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Variants of Nuclear Power Plants of Small and Medium Power with Heavy Liquid-Metal Coolants

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Abstract

New design solutions have been proposed for a BRS-GPG type reactor circuit, which are different from transport and stationary low and medium-powered reactor installations cooled with heavy liquid-metal coolants, and which correspond to the evolutionary development of such installations. While developing these solutions, the available experience in creating and operating Soviet pilot and commercial power plants cooled with lead-bismuth coolants was used, including investigations, primarily experimental ones, carried out by team of authors in justification of a capacity range (50 - 250 MW) of low and medium-powered reactor plants with horizontal steam generators (BRS-GPG) proposed and elaborated at the NNSTU.

Keywords

Heavy Liquid Metal Coolant (HLMC), Nuclear Power Plant, Lead, Lead-Bismuth, Low and Medium Power Reactor, Steam Generator Solution, Main Circulation Pump Solution, BRS-GPG, Multifunctional Reactor

1. Introduction

The development of small and medium-sized nuclear power plants is in great demand at the present time. The proposed design of the first circuit of a small and medium-sized nuclear power plant with a lead coolant, developed by a team of authors, is promising for providing electricity to remote research or industrial facilities. As presented in [1] [2], projects like this are attracting more and more interest.

Designing heavy liquid-metal coolant (HLMC) circuits begins with the development of circuit principle diagrams defining the composition of equipments and tie lines (pipelines, channels) between them. These diagrams comprise normal operation elements (basic and secondary) as well as, where necessary, protective and localizing systems being part of safety systems. The latter are designed to localize after-effects when any failure of the normal operation elements occurs.

Principle diagrams are used during developing design and operating documentation, during training operating personnel, during operating circuits, in case of repairs and in some other cases.

The structure and technical solutions of circuit principle diagrams are primarily determined by the functional purpose of a facility comprising a circuit (power installation, research test bench, etc.).

Diagrams of power circuits cooled with HLMC are usually required to be viable, *i.e.* fail-safe in case of any malfunction of one or more elements. This requirement is implemented by backing up vital elements of diagrams. An example of such backup includes introduction into a diagram of several circulation pumps or heat-exchangers (steam generators) connected through cut-off devices in a collector circuit or several autonomous groups "steam generator-pump" shut off by valve gates in a loop circuit. Increasing the number of equipment units, including those having equal reliability performance, detracts from circuit reliability in general.

Principle diagrams of HLMC circuits, based on the functional purposes of the relevant installations with these circuits, may be divided into power-producing and non-power-producing. Power-producing circuits include:

- Heat-removing circuits of epithermal and fast neutron heavy nucleus fission reactors;
- Reactor (target) circuits of currently developed accelerator-driven;
- Circuits to remove heat from the blanket and diverter of a nuclear-fusion reactor, in which the possibility of using HLMC is conceptually feasible.

Non-power-producing HLMC circuits include:

- Circuits to test and deliver equipment (main circulation pumps, steam generator sections, etc.), shut-off devices and other components of power circuit circulation routing;
- HLMC circuits of test (research) benches.

Considering reactor plant circuits it has to be chosen the correct coolant.

In terms of safety requirements, HLMC reactor plants have an obvious advantage over installations cooled with sodium and water coolants, have a 20-time lesser specific energy per unit of volume than in VVER type reactors and 10 times less than in reactor plants cooled with sodium coolants. In HLMC reactor plants, there is no potential "compression energy", chemical energy of interaction with zirconium as with water and no potential energy of escaping hydrogen with air as with water and sodium, etc.

In terms of cost effectiveness, HLMC power-generating units have an obvious advantage over reactor plants cooled with water and sodium, are commercially viable as compared with installations powered by cheap hydrocarbon fuels. The high cost effectiveness of HLMC units is achieved owing to a high efficiency of the power-generating unit (up to 45%), a high utilized temperature of HLMC (up to 500˚C - 550˚C) and, consequently, produced steam (400˚C - 520˚C) at its pressure 4.0 - 24.0 MPa that is low being close to the atmospheric pressure in the reactor circuit (0.1 - 0.5 MPa) and, consequently, small thickness of walls, reactor plant double-circuit arrangement, through using advanced types of fuels (nitride) and other advanced solutions, through using typical, standard steam-turbine plants in power-generating units [3] [4].

2. Reactor Plant with HLMC

The networked reactor circuit cooled with HLMC and having a loop-type arrangement in its turn required availability of a branched and powerful system for heating up the reactor circuit elements and maintaining them in a "hot" condition. All that, in general, reduced the reliability of the power plant and increased its dimensional characteristics. The specific weight and mass indicators of reactor plants having a loop-type arrangement may be not only equal to, but also less than similar characteristics of multi-block and single-block configurations due to a very high density of HLMC.

The SVBR-100 reactor plant has been developed as a unified reactor plant with a power up to 100 MW (electrical) for multifunction application within the structure of modular nuclear power plants considerably reducing investment risks or as independent energy sources. Parameters of generated steam, based on the power unit intended use, may vary within the range of pressures from 4.0 to 9.0 MPa. The temperature of generated steam may vary within the range from saturated steam to overheated steam having T = 400˚C. (These design parameters may change in future).

In low-powered plants, it is allowed to implement technical solutions resulting in some decrease in the working fluid loop parameters and the thermal Rankine cycle efficiency as compared with the technical and economical benefits conditioned by such decrease.

The plant uses a fast neutron reactor cooled with a lead-bismuth coolant if there is pressure in the reactor circuit, with gas pressure above the HLMC level close to the atmospheric pressure. A possibility is provided in the reactor as to the use of different types of fuel (UO_2, MOX-fuel, mixed oxide fuel with minor actinides—TRUOX-fuel, nitride fuel) without redesigning the reactor and with meeting the safety requirements. Unlike plants cooled with sodium, the SVBR reactor plant uses a two-loop circuit of heat removal and a steam generator with repeated natural circulation along the secondary loop (working fluid loop). In heat-rejection loops of the reactor monoblock unit, there is natural circulation (NC) of the coolant, which is enough to cool the reactor without any dangerous overheating of the core. The reactor monoblock unit having a guard vessel is located in the tank of the passive heat-rejection system. The tank filled up with water also performs the function of a neutron shield. Provision is made for a non-recurrent cartridge-to-cartridge fuel discharge at the end of the core service life and charging of new fuel in the form of an all-in-one cartridge (core).

The reactor plant arrangement is integral, whereby the reactor core, the primary circuit equipment and steam generators are located in a single inner casing of the reactor monoblock unit, with complete elimination of shut-off devices and pipelines of the lead-bismuth coolant. The main components of the reactor monoblock unit and the reactor plant are made in the form of individual modules, whereby there is a possibility of their replacement and repair.

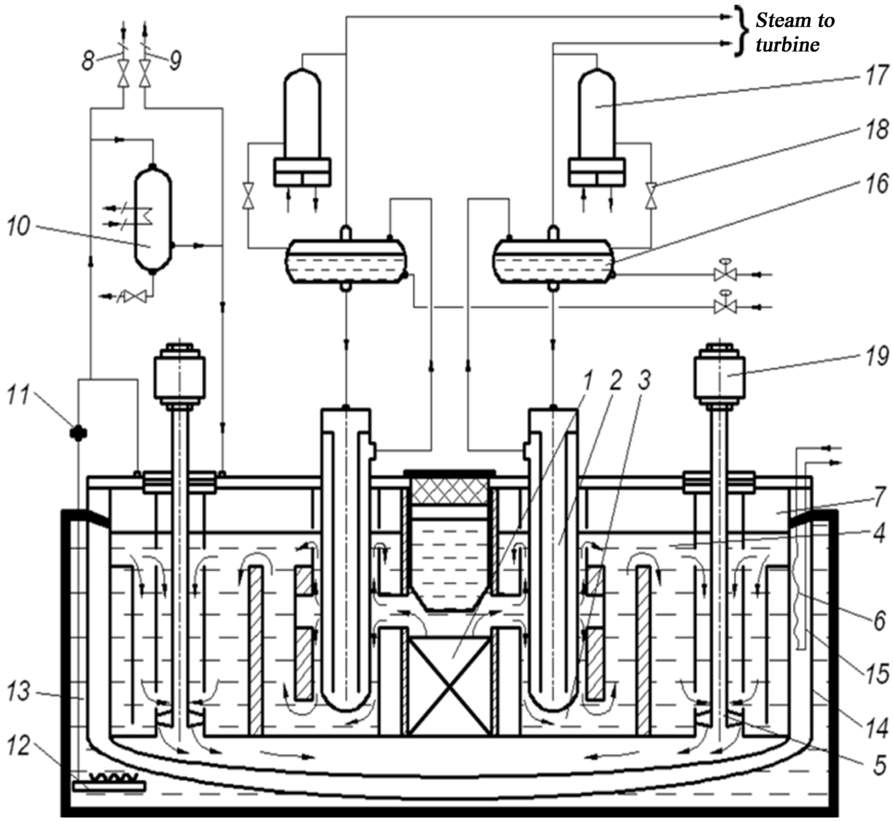
The circuit of heat removal from the reactor using a lead-bismuth eutectic alloy (Circuit I) is designed to remove heat from the reactor core and transfer it to water-steam (working fluid) in the steam generator. The primary circuit comprises capacities filled up with a liquid-metal coolant and protective gas systems. This system is designed to:

- compensate for temperature variations of the coolant volume;
- protect eutectics against any contact with the atmospheric oxygen and the coolant oxidation;
- prevent radioactive gaseous (vaporous) products from getting into the reactor plant room;
- create an anti-cavitation static head at the suction of the circuit circulation pumps (when required);
- localize emergency situations referring to water and water steam inflow to the liquid-metal coolant in case of any interloop leakiness (leak) of the steam generators;
- carry out processing treatments of the circuit.

Application of a lead-bismuth coolant makes it possible, as compared with a sodium-based coolant, to further safety of the plant since eutectics, unlike sodium, do not burn in the air at the reactor circuit parameters and do not interact with water forming a highly explosive hydrogen and releasing heat, which offers the possibility of using a two-loop circuit, rather than a three-loop circuit.

In comparison with a water coolant, the lead-bismuth eutectics contribute to using not slow (thermal) but fast fission neutrons in the reactor core, which in turn creates the possibility of obtaining neutron-fissionable isotopes from non-fissionable ones, like the possibility of using available raw material stocks for the atomic energy industry ten times and more as well as using fast neutrons for transmutation of long-lived isotopes and for other purposes. In addition, using liquid-metal coolants allows for almost twofold increase in the thermal efficiency of power units due to a possibility of obtaining high parameters of steam generated by the reactor plant at pressures in the reactor circuit close to the atmospheric pressure. Steam parameters are $T > 500^\circ C$, $P > 16.0$ MPa (**Figure 1**).

In contrast with reactor plants cooled with gaseous coolants, under otherwise equal conditions, reactor plants cooled with liquid metals have qualitatively



1—reactor core; 2—steam generator; 3—steam generator sub-header; 4—steam generator top header; 5—main circulation pump; 6—free coolant level; 7—unit gas plenum; 8—gas supply pipeline; 9—gas discharge pipeline; 10—isolation condenser; 11—burst diaphragm; 12—sparger; 13—water region of passive heat-rejection system tank; 14—guard vessel; 15—steam heating equipment; 16—separator; 17—cooling condenser; 18—shut-off control valve; 19—pump motor.

Figure 1. Principle Diagram of SVBR-100 Reactor Plant Circuit (Pneumohydraulic).

smaller mass and dimensions parameters and greater safety due to better thermal and physical characteristics of liquid metals as compared with gases.

The basic characteristics of the SVBR-100 reactor plant core are shown in **Table 1** [5].

The SVBR-100 reactor plant, owing to the physical peculiarities of fast neutron reactors, the relatively low power, the chemical inertness and high boiling temperature of lead-bismuth eutectics as well as the integral configuration of the reactor circuit catchment-based equipment, exhibits the properties of inherent safety. All safety systems of the reactor plant operate inactively. Most of them are used as normal operating systems. The modular design of a reactor plant unit having relatively low mass-dimensional characteristics of the reactor and other equipment makes it possible to provide for complete manufacture at the factory and deliver to the site of a nuclear power plant using any types of transport.

In the course of Russian developments of reactors cooled with lead coolants, the type and design of a reactor (from a graphite-moderated reactor to a fast neutron reactor), the pattern of coolant circulation in the circuit and other technical

Table 1. The basic characteristics of the SVBR-100 reactor plant core.

No.	Parameters	Value
1	Installed thermal power, MW	280*
2	Power plant electrical output, MW	100
3	Lead-bismuth coolant temperature, ˚C • at the outlet from the reactor core • at the inlet to the reactor core	 490 340
4	Reactor core dimensions D × H, m	1.645 × 0.9
5	Reactor core mean volumetric power density, kW/dm^3	140*
6	Mean linear load on nuclear fuel elements, kW/m	~24.3*
7	Fuel: • type • uranium-235 charging, % • mean uranium-235 enrichment, %	 UO_2 ~1470* ~16.1*
8	Reactor core service life, thous. eff. hours	~53
9	Reloading interval, years	~8 (~10)
10	Volume of lead-bismuth coolant in circuit, м^3	18
11	Reactor shell overall dimensions, m	↓ 4.53 × 6.92
12	Primary circuit coolant, volume m^3	Pb-Bi, 32
13	Number of loops	4

*these characteristics refer to the SVBR-100 reactor plant within the structure of a modular NPP having two units of 100 MW electrical power each. If the SVBR reactor plant is used within the structure of any other NPP, the specified characteristics may vary.

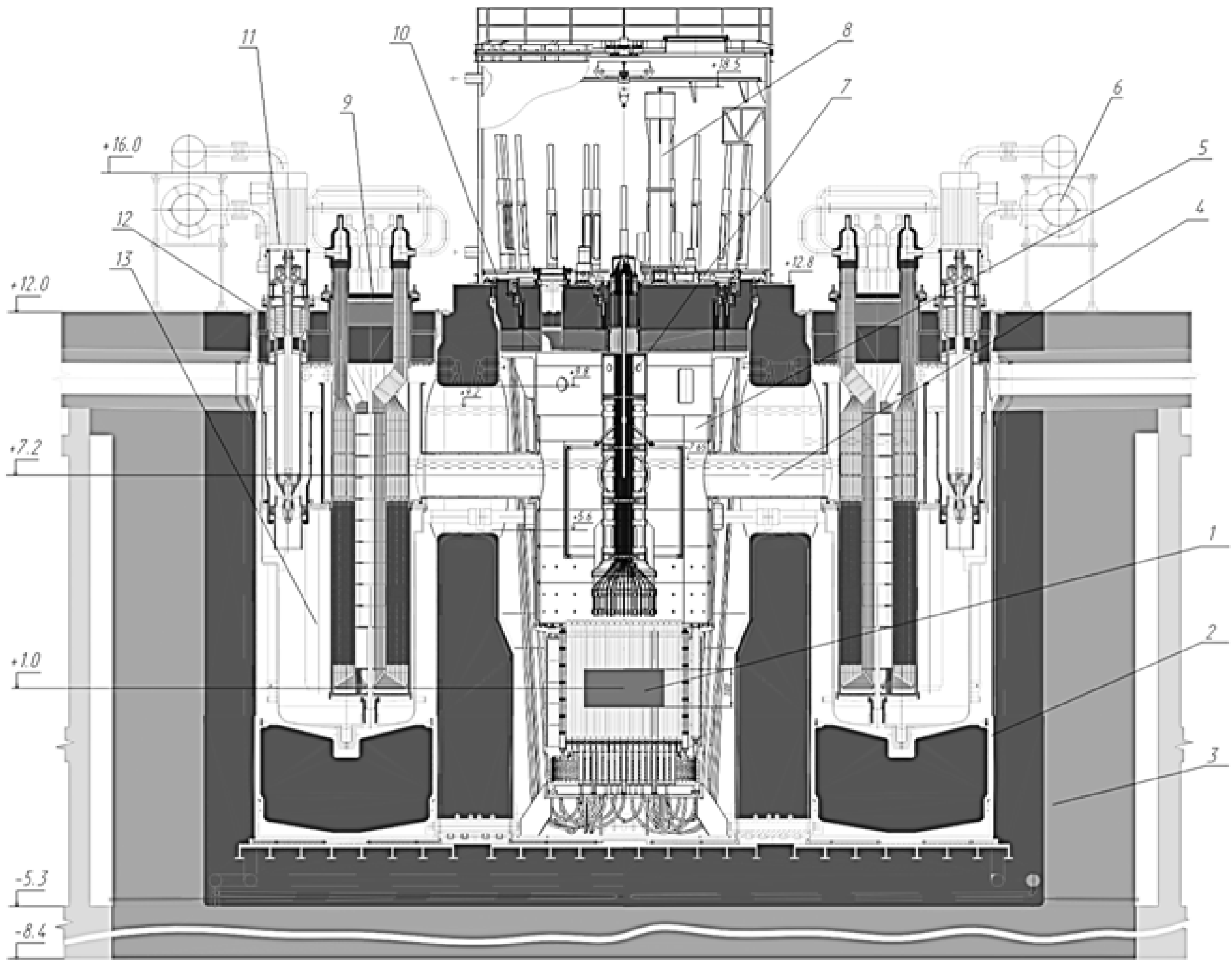
solutions have evolved. From the previous options of this plant, its final options are distinguished by a unique, previously unused pattern of coolant circulation across the reactor circuit. Against a head of four main circulation pumps, the coolant heated up to 420˚C rises in the tank to the top of free level in the circuit. Then the lead is supplied to an annular chamber and from it to an annular downtake clearance, from which—to the reactor pressure chamber. Passing the core, the coolant heated up to 540˚C rises to the free level above the reactor outlet chamber. From the latter, the lead is fed through flumes to inlet chambers of steam generators, which have a free level. Giving up heat to the steam-water circuit working fluid, the liquid metal goes down passing through the steam generator tubing system and is further fed to suction chambers of the main circulation pumps. The specific feature of this pattern includes the presence of free levels at the coolant inlet and outlet from the steam generator. Such pattern prevents the working fluid (water, steam) from getting into the reactor core in case of the emergency "interloop steam generator leakiness". Potential drawbacks of such pattern may include fluctuations of interconnected free levels in the circulation circuit. The disadvantages of latest alternative design options of the BREST-OD-300 reactor plant unit include its structural complexity and poor manufacturability (**Figure 2**).

A reactor plant of the BREST type has no pipeline and shut-off device, which improves reliability of the plant.

In the BREST-OD-300 successor options, the main circulation circuit has a tank-based, catchment-based configuration, a reactor and four steam generator-pump loops made in separate casings (vaults) of the reinforced concrete framework and having common free coolant levels.

3. Main Design Solutions of a Modern New Circuit of BRS-GPG Reactor Power Installation Cooled with HLMC

Scientific group in Nizhniy Novgorod State Technical University have set a task:



1. Reactor core; 2. Casing unit; 3. Reactor pit; 4. Header pipeline; 5. Reactor core barrel; 6. Cooldown system; 7. Measuring column; 8. In-vessel transfer machine; 9. Steam generator; 10. Top covering; 11. Main circulation pump; 12. Steam generator-MCP unit; 13. Filter.

Figure 2. BREST-OD-300 Reactor View in Longitudinal Section. The conceptual designs of the more powerful BREST-OD-1200 reactor plant having electrical power of 1200 MW are made on the basis of the same principles as the BREST-OD-300 reactor plant.

creation and experimental verification of a new reactor power plant with fast neutrons, cooling with HLMC, called BRS-GPG, with means Fast Lead Reactor With Horizontal Steam Generator.

The final aim on the first stage of development was:

1) selection and justification of the coolant;

2) creating the construction, that eliminate the steam generator leakiness accident or minimize its effects;

3) decrease the mass-dimensional index of whole facility;

4) provision of heat removal in standing regime;

5) integrating the one coolant technology system;

6) additional system of safety in case of coolant flow out from reactor shell.

The absolute advantage of reactor plants cooled with liquid-metal coolants (Pb, Pb-Bi, Na, K, etc.), as compared with reactor plants using water coolants (pressurized-water reactors, shell-type water-boiling reactors), includes, along with more complete utilization of raw materials resources, a higher thermal efficiency of the power generating facility due to higher steam parameters generated by the reactor plants. Furthermore, the reactor loop pressure in reactor plants using liquid-metal coolants is close to the atmospheric pressure, unlike reactor plants using water coolants, in which the reactor loop pressure equals 6 - 16 MPa (60 - 160 kgf/cm^2) and higher. Reactor plants cooled with heavy liquid-metal coolants, as contrasted with reactor plants cooled with primary sodium, have a double-loop design, rather than three-loop design. The basic characteristics of the BREST reactor plant core are shown in **Table 2** [5].

Upon occurrence of one of the most potentially hazardous emergency situations in reactor plants cooled with heavy liquid-metal coolants—"interloop steam generator leakiness", water and steam are supplied from the working fluid loop under a pressure of 6 - 16 MPa and higher to the reactor loop under a pressure in the steam generator from approx. 0.01 MPa (0.1 kgf/cm^2) to 0.6 - 0.7 MPa (6 - 7 kgf/cm^2) determined by the HLMC pressure head in the steam generator and the pressure produced by the main circulation pump (MCP). The design of reactor plants should exclude both water entrance into the core, which in the case of a fast neutron reactor may result in its uncontrolled runaway, and overpressure and destruction of the reactor loop that is not rated for the water and steam pressure in the working fluid loop.

Unlike the reactor loops of nuclear-powered submarines cooled with heavy liquid-metal coolants, the fast neutron reactor loops of HLMC power-generating units, due to economic considerations, may not be designed for a pressure equaling or exceeding the working fluid pressure in steam generators. This in turn poses the requirement to use technical solutions preventing any significant pressure rise in the reactor loop in case of the emergency under review. It is obvious that implementation of such solutions is clearly determined by the accepted reactor plant design, the maximum flow of water and steam supplied to the HLMC loop upon occurrence of such emergency and, accordingly, the leakiness area in

Table 2. [5] The basic specifications of BREST-OD-300 and BREST-OD-1200 reactors.

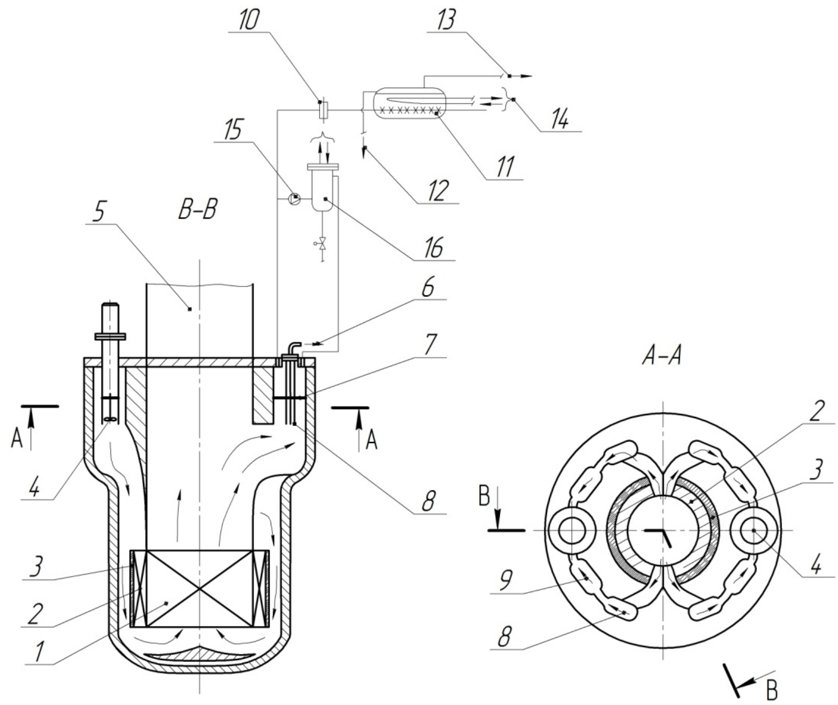
Characteristic	Brest-300	Brest-1200
Thermal power, MW	700	2800
Electrical power, MW	300	1200
Number of fuel assemblies in the reactor core, pcs.	185	332
Reactor core diameter, mm	2300	4755
Reactor core height, mm	1100	1100
Fuel element diameter, mm	9.1; 9.6; 10.4	9.1; 9.6; 10.4
Fuel element pitch, mm	13.6	13.6
Reactor core fuel	UN + PuN	UN + PuN
Fuel element column, (U + Pu)N, t	16	63.9
Fuel element column, Pu/(Pu^{239} + Pu^{241}), t	2.1/1.5	8.56/6.06
Fuel-element lifetime, years	5	5-6
Reloading interval, years	1	1
KVA	~1	~1
Power effect, % DK/K	0.16	0.15
Entire effect, % DK/K	0.35	0.31
Delayed neutron fraction, b_{eff} %	0.36	0.34
Lead in/out temperature, ˚C	420/540	420/540
Maximum temperature of fuel element cladding, ˚C	650	650
Maximum lead flow rate, m/s	1.8	1.7
Steam temperature at the steam generator outlet, ˚C	340/520	340/520
Steam generator output pressure, MPa	24.5	24.5
Lead flow, t/s	40	158.4
Steam generator capacity, t/s	0.43	1.72
Net efficiency of power-generating unit, %	43	43
Expected useful life, years	30	60

Note. This design data may be adjusted at later stages of work.

the steam generator tubing system. The designer may include in the project an allowance for potential rupture (destruction) with discharge of the working fluid from two cross sections of one steam generator tube. The design may include an allowance for any destruction with a lesser discharge area or an allowance for potential destruction of several tubes or a section of the steam generator (for example, due to cyclic thermofatigue stresses). The maximum safety will be applied to that reactor plant whose design may contain an emergency with the maximum reasonable flow of water and steam supplied to the reactor loop, in which there is no overpressure of the reactor loop, and the entrance of water and steam into the core is restricted to a value reliably excluding any reactor uncontrolled runaway.

Taking into consideration potential occurrence of the emergency "interloop steam generator leakiness" in a reactor plant using a heavy liquid-metal coolant, as it has been shown by the experimental and calculation-theoretical research, preference will be given to the structural design of a reactor plant with horizontal steam generators [6] [7] [8] whose version is shown in **Figure 3** Reactor core 1, steam superheater 8, evaporator 9 and main circulation pump 4 are located beneath free coolant level 7.

The steam generator tubing system is designed horizontal with a view to reducing the HLMC bed height over the tubes of the steam generator tubing system and forming, in case of any significant emergency destruction of the tubing system elements, a solid steam (steam-water) channel between the destruction point and the gas plenum over free coolant level 7. The tubing system of steam superheater 8 and evaporator 9 is designed in the shape of flat coiled horizontal tubes whose ends are sealed in tube plates of chambers of steam-superheating and evaporating sections having detachable covers for mounting, when it is necessary, throttles and plugging emergency tubes when they are depressurized.



1—reactor core; 2—neutron reflector; 3—biological protection; 4—main circulation pump; 5—reactor control and protection system rods; 6—superheated steam line to turbine; 7—free HLMC level; 8—steam generator superheater; 9—steam generator evaporator; 10—burst diaphragm; 11—relief tank; 12—condensate to special water treatment system; 13—gas to gas treatment system; 14—cooling water; 15—gas circulator; 16—safety steam condenser.

Figure 3. Structural design of a HLMC Reactor Plant Unit with a horizontal steam generator.

To stir (prevent stratification of) the coolant, combs (or other devices) are mounted in the tubing system, whose teeth are located between the tubes of the horizontal steam generator. The gas plenum over free coolant level 7 in the steam generator sections is higher or equal to the amount of the coolant contained in them and is connected through burst diaphragm 10 with relief tank 11 whose gas plenum is open to the atmosphere via the gas treatment system. The water volume of the relief tank is cooled with process water and is connected through overflow line 12 with drain tanks or other capacities receiving and treating the condensate supplied to them upon occurrence of the emergency "interloop steam generator leakiness".

The gas plenum of each section of the steam generator is connected through pipelines with emergency steam condensers and gas circulator 15.

In order to reduce the circulation routing length of the HLMC reactor loop and decrease the amount of the coolant contained in it as well as to lessen the mass-dimensional characteristics of the reactor loop, above reactor core 1, there is an annular channel whose inner diameter is larger than the outer diameter of reactor core 1. The annular channel (its sections) contains steam-superheating 8 and evaporating 9 sections of the horizontal steam generator.

The entrance region of steam superheater 8 is connected with the coolant capacity above reactor core 1, and the output region of evaporator 9 is connected with the inlet chamber of main circulation pump 4 (axial). The pump outlet cavity is connected via a downtake channel with core 1 of the reactor.

Such arrangement of the reactor loop contributes to reducing its hydraulic resistance, increasing the percentage of natural circulation in the loop and improving the reactor plant safety in the event of any emergency destruction of the steam generator tubing system elements.

A nuclear power plant is operated under the potentially dangerous emergency condition "interloop steam generator leakiness" in the following manner. Depending on the interloop leakiness size and, accordingly, the flow rate of water and steam supplied to the HLMC loop, it is possible to differentiate two conditions of the plant when such emergency occurs. In case of a minor flow rate where steam and water inflow in the form of bubbles or jets, an instance of such emergency is proven by an increase in the condensate level in condenser 16, a certain rise of pressure in the gas system and a growth of the free coolant level in the reactor loop, primarily in the leakiness area of the horizontal steam generator tubing system. In this mode, the gas circulator is put into operation, and the steam is condensated in emergency condenser 16. From there, the condensate is removed with opening a drainage valve by signals from the top level alarm to the drain tank. By a signal from the bottom level alarm, the drainage valve closes. If such emergency occurs, the reactor plant may operate with subsequent search for and plugging of a faulty tube (tubes) or the point of their sealing into the tube plate. In case of the reactor plant continuous operation in this emergency mode, there occurs the coolant oxidizing in the loop due to the following. Steam and water are contaminated with oxygen that oxidizes the HLMC. Water molecules inflowing to the coolant volume partially dissociate forming oxygen and hydrogen both due to ordinary chemical reactions and under the action of ionizing radiation. The formed oxygen almost instantly enters into a reaction with the liquid-metal coolant oxidizing it in the melt or forming (at saturation) a solid disperse oxide phase. The hydrogen formed out of water molecules due to gravitation and other effects is dumped to the gas system only slightly deoxidizing the coolant in the loop and restoring its oxides residing within slags on its free surface.

In case of a high flow rate of steam and water through a faulty leakiness, *i.e.* a rupture of one or more or all tubes of the steam generator section, the operating signs of such emergency include a heavy increase in pressure in the gas system due to inflow of a considerable amount of steam and water into the protective gas system capacity and a growth of the free HLMC level in the faulty steam generator section.

Due to insignificant embedding (~1000 mm and less) of the horizontal tubes beneath the coolant level, the steam generator section has a sparging crisis of light phase through the HLMC layer without any dynamic shock effect typical of other-type reactor plant arrangements. Between the point of discharge and inflow of the working fluid into the HLMC capacity and the free HLMC surface developed in the horizontal steam generator, there forms a steam (steam-water) channel. From the walls of this channel, the finely-divided HLMC phase may be trapped by steam to the steam space over the free coolant level. The combs and other devices creating the HLMC flow between the flat coils of the steam generator tubing system limit the sparging area and speed up the sparging crisis process in the HLMC volume in the steam generator section. The coolant level in the faulty section, as also in other steam generator sections, increases insignificantly, by a value of the steam channel volume.

At the moment of any interloop leakiness appearance with reasonably maximum damage to the tubing system elements and up to any burst of diaphragm 10, no overpressure of the reactor loop and no hydraulic impact will occur at such arrangement, unlike other arrangements. The pressure in the reactor loop rises up to the bursting pressure of diaphragm 10 in addition to a minor static pressure ~0.2 MPa (up to approx. 1 kgf/cm^2 (atm)) over the faulty horizontal tubes. Following the diaphragm bursting, the loop pressure increases by a value of hydraulic resistance of the steam-gas mixture flow from the horizontal steam generator gas plenum up to the pressure in the relief tank that is open to the atmosphere. After determination and shutdown of the horizontal steam generator according to water and steam, plugging of the faulty steam generator tubes is carried out.

At such emergency, the increased pressure in the faulty steam generator section blocks any supply of the coolant to it and contributes to termination of the HLMC circulation through the faulty section and partial drainage of the coolant in the faulty section.

The arrangement of the reactor plant with horizontal steam generators makes it possible to:

- prevent water from getting into the fast neutron reactor core and eliminate the reactor uncontrolled runaway at the potentially hazardous emergency situation "great interloop leakiness" of the steam generator at almost any flow rate of the working fluid in the reactor loop;
- exclude the loop overpressure and hydraulic surges at almost any flow rate of the working fluid in the reactor loop when the emergency "interloop steam generator leakiness" occurs;
- increase dynamic pressure of natural circulation and, consequently, flow rate of natural circulation due to concentration of the heat transfer section (in the steam generator) in the upper loop area, minimization of the reactor loop length and exclusion from it of any additional upcomer and downcomer regions increasing the loop hydraulic resistance and creating dynamic pressure of natural circulation;
- provide for sufficiently simple detection and damping of the steam generator emergency tubes;
- ensure compactness and minimize mass-dimensional characteristics of the reactor plant [5].

Coolant Selection and Justification

The technologies of creating and operating power-generating reactor plants cooled with lead-bismuth coolants have been successfully tried and tested in Russia. This coolant is compatible with water as a working medium in the Rankine cycle. Its melting temperature of 125^{0}C corresponds to the saturated steam pressure of 0.23 MPa, which allows for reliable heat removal with water in the equipment using this coolant at its operationally acceptable pressure of more than 0.3 MPa without coolant freezing. This, in turn, makes it possible to ensure the reactor shutdown cooling and, when required, heating of the reactor circuit elements with water and steam in standby and transient modes with excluding any possibility of liquid metal freezing in the reactor circuit. This property of Pb-Bi significantly improves its consumer appeal as compared with lead. The disadvantage of a lead-bismuth coolant includes a high degree of activity during the reactor operation according to Po-210 isotope, which is 20,000 times higher than the activity in a lead coolant circuit, as well as the cost of bismuth is by an order of magnitude greater than that of lead.

The lead melting temperature of 326˚C corresponds to the saturated vapor pressure of about 12.2 MPa. This virtually eliminates the possibility of heat removal with water from the equipment cooled with a lead coolant during the reactor shutdown cooling and in standby modes because any pressure reduction in the water cavity below this value will cause the coolant freezing and the channel permeability suspension in the cavity filled with lead. Keeping of a pressure larger than 12.3 MPa in the steam generator cavities or other heat-exchangers in transient, standby and maintenance modes is technically challenging and virtually impossible, which makes this coolant ill-compatible with water. The extensive experience in operating benches with lead-bismuth coolants during heat tracing of pipelines and equipment with HLMC shows no perceptible difference in their maintenance.

Lead as a coolant cooling a heavy nucleus fission reactor circuit has the following major advantages: a high boiling temperature, chemical inertness upon contact with water and air, and a minor radioactivity. Small values of slowing-down and neutron-absorption cross sections in lead allow increasing its volume concentration in the core up to 60%; decreasing the value of the coolant heating in the core; reducing the velocity value and, consequently, power expenditures for pumping the coolant, and ensuring a high level of natural circulation.

According to other characteristics, lead and lead-bismuth as reactor plant coolants are almost identical. Based on the cost effectiveness and safety criteria, the application of a lead coolant under modern technologies is probably more reasonable than the application of a lead-bismuth coolant in low and medium-powered installations (in BRS-GPG reactor plants).

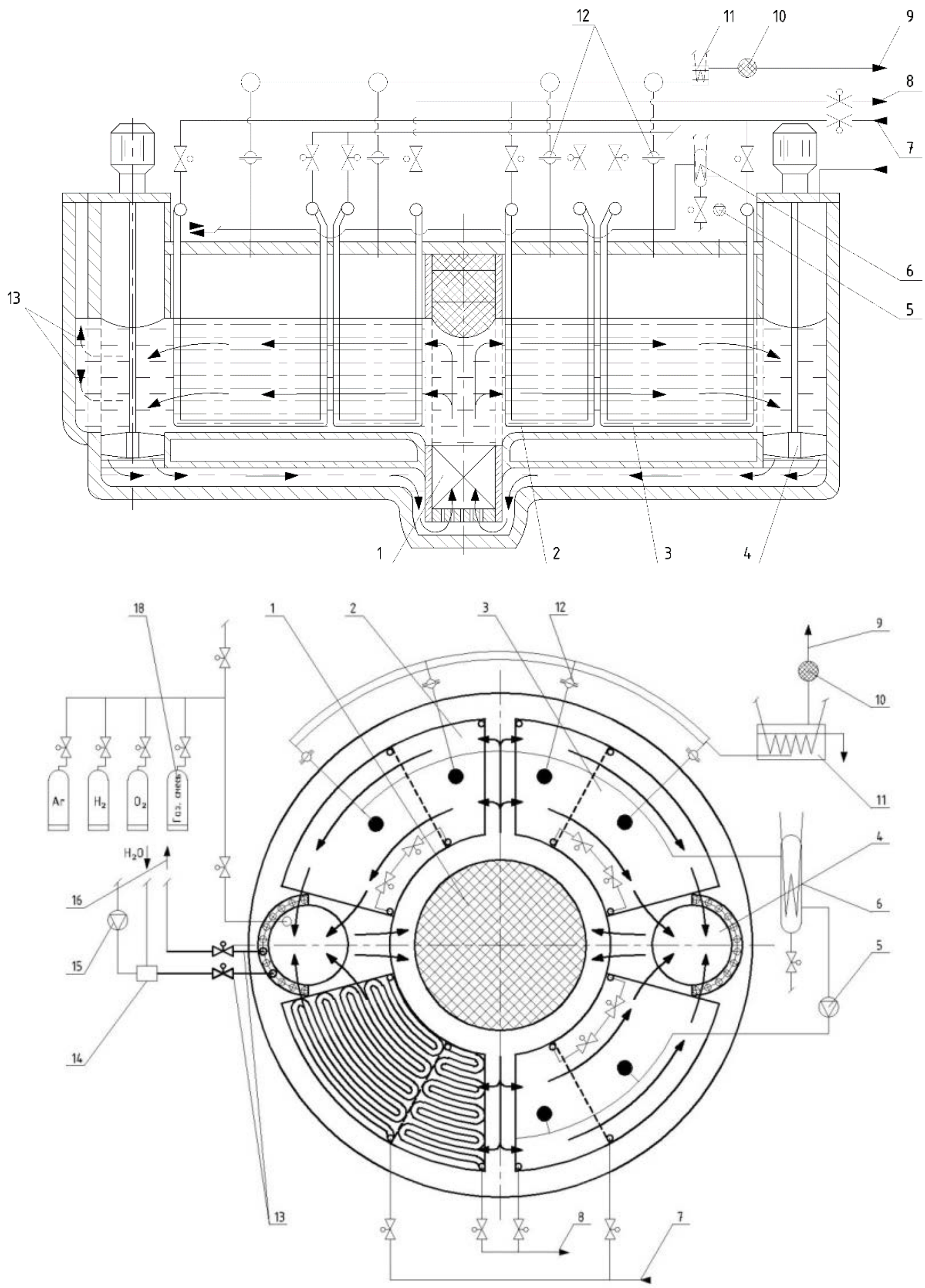
Coolant Circulation Pattern in Reactor Circuit and Reactor Unit Arrangement

It is suggested that a BRS-GPG type reactor plant has a new, unconventional in terms of HLMC, arrangement of the reactor circuit and coolant circulation pattern minimizing its length and excluding any additional upcomer and downcomer regions. The coolant circulates in the following way. Passing through the reactor core, the lead enters the superheater section, then the evaporation section of the horizontal steam generator with the coolant free level and further the submersible axial main circulation pump, from the head of which it descends to the reactor core inlet.

With such arrangement of the reactor circuit, the maximum possible natural coolant circulation is achieved, which significantly improves the reactor plant safety. In such case, a gain in performance of the power-generating unit is possible both due to the maximum value of its specific power and due to the minimum mass-dimensional characteristics of the reactor circuit (**Figure 4**).

Experiments with introducing relatively large supplies of water and steam (kilogram and more), with embedding discharge orifices up to 4.0 m under the HLMC free level, with a temperature up to 600˚C, at a pressure fall at such discharge orifice up to 8.0 MPa show that the steam and steam-water mixture independently organize a "light-phase" vertical channel from the discharge point to the HLMC free level, regardless of HLMC availability and speed of circulation [3] [5].

This result of investigating (simulating) one of the most hazardous emergencies in HLMC installations—"steam generator large break" enables to propose a technical solution that will competently change running of such accident providing the power-generating unit safety when it occurs. The application of a steam generator (horizontal) whose piping system is positioned at the minimum depth (up to 1.0 m) under the HLMC free level provides for a spontaneous arrangement of a "light phase" channel from the point where the working medium



1—reactor core; 2—steam superheater; 3—evaporator; 4—main circulation pump; 5—gas circulator; 6—isolation condenser; 7—feed water supply; 8—steam to turbine; 9—gas blow-off line; 10—filter; 11—condenser; 12—burst diaphragm; 13—to standby and emergency heat removal system; 14—mixer; 15—compressor; 16—air-steam mixture outlet

Figure 4. Principle diagram of BRS-GPG reactor circuit side view (top) and top view (bottom).

enters HLMC to the gaseous (steam-gas) plenum above the coolant free level in the steam generator emergency section. Then the steam, water and gas enter a condenser through a burst diaphragm, from where the gas passes to the atmosphere through the gas treatment system, and the condensate to the "dirty water" tank. The application of such technical solution renders a BRS-GPG type installation innovative in terms of safety as compared to other HLMC installations.

Steam Generator Design Solutions

All steam generators that were used within the structure of nuclear-powered submarine installations had coolants longitudinally flowing around heat-exchanging pipes, U-shaped type. SVBR-type steam-generator units use Field's tubes; BREST-type steam-generator units use coiled tubes with a small coil inclination angle.

In steam generators of BRS-GPG reactor plants, the steam generator piping system design has a maximum depth for the HLMC free level, such as in the shape of a plate coil system. Pipe ends of the steam generator sections are embedded into tubeboards of water and steam chambers of the reactor unit top plate. The NNSTU has carried out activities for experimental determination of heat-exchange characteristics of a horizontal piping system flowed around by a high-temperature lead coolant. Piping systems of the steam-superheating and evaporating sections are located in the annular channel above the reactor core. The gas plenum above the HLMC free level is equal or exceeds the coolant volume in the section. The volume of each section opens through a burst diaphragm into a steam condenser, such as a relief tank whose gas plenum is open to the atmosphere through the gas treatment system.

The gas plenum of the steam generator sections opens into isolation condensers and a gas circulator, which conceptually makes it possible to operate the power-generating unit at the power level in the emergency mode "steam generator small break" [9].

The structural scheme of the reactor plant using HLMC with a steam generator with a tubing system in the form of horizontal tubes or tubes with a slight incline (straight, flat coils, etc.) with the minimum deepening for a coolant free level (**Figure 4**) is the most safe in case of any steam generator interloop leakiness of the steam generator up to the breakdown of all the tubes of the superheater and flash sections [10]. In case of such an emergency, a continuous steam (steam-water) channel is generated in the coolant volume from the leak spot with a small (up to 1.0 - 1.5 m) deepening under a HLMC free level of up to the volume above a coolant free level. This channel removes steam and water with a minimum hydraulic resistance from the reactor circuit through the burst diaphragm installed in the gas plenum of each section of the steam generator. The HLMC level in a breakdown steam generator grows, which slightly increases a free level in the unit isolating gaseous cavities of the breakdown steam generator. Reactor cooling is carried out through fault-free steam generators or through other channels. Standard technologies are used for breakdown tubes search and plugging. In doing so, over-pressurization of the HLMC circuit is excluded as well as supply of water and steam into the core of the fast neutron reactor at any size of leakiness in one of several sections of the steam generator.

In case of a small size of the interloop leakiness, steam and water supplying to the HLMC space of the horizontal steam generator are separated on the developed free coolant surface and removed from the reactor circuit by standard methods.

A potential drawback of such a design of the steam generator is temperature stratification (gradient) of the HLMC flow horizontally moving along the height. To exclude this effect, installation of vertical grids in the tube system is provided.

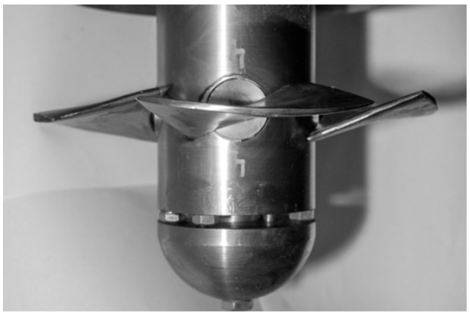
Small and medium powered reactor plants using HLMC with horizontal steam generators have substantial benefits as compared with other types of reactor plants using HLMC and currently they are under development [3].

Main Circulation Pump Design Solutions

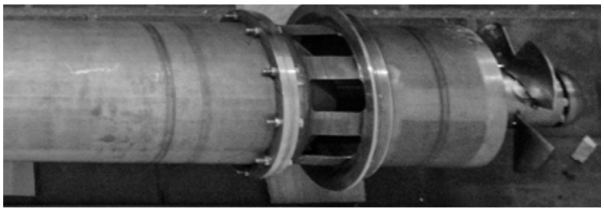
Based on the HLMC head and supply ratio in a BRS-GPG type installation, the design of an axial deep-water pump with the "from top to bottom" coolant flow direction has been chosen for development, which is conditional on the accepted reactor circuit arrangement. The NNSTU carries out investigations aimed at a reasoned designing of the BRS-GPG main circulation pump, stepwise including:

- experimental determination in a natural environment of basic characteristics of the pump impeller model cascades caused by circulation of speed when a lead coolant flows around the blades at the rate up to 200 t/hour and with a temperature of 440˚C - 500˚C defining an optimum alternative design of the full-sized impeller [4];
- experimental determination in a natural environment of an optimum blade profile of the impeller model in an optimum cascade defined at the first stage;
- experimental determination in a natural environment of characteristics and optimum geometry of the pump input and output sections, including the outlet straightener.

The NNSTU has developed a model design of an axial electric pump with rotating blades and uses it during experimental studies (**Figure 5**). The use of the BRS-GPG installation main circulation pump design with rotating blades enables to:



(a)



(b)

Figure 5. Pictures of (a) Shaft-Mounted Impeller and (b) Main View of Pump with Rotating Blades.

- cut off the coolant backflow during the pump emergency stop replacing the non-return valve in the circulation routing, and
- minimize the pump hydraulic resistance under HLMC natural circulation in the reactor circuit [3] [5].

Reactor Shutdown Cooling and Ensuring Reactor Plant Standby Modes

To ensure reliable and safe heat removal during shutdown cooling and in standby modes of a BRS-GPG type installation, it is suggested for regulated heat removal from HLMC to use an air-water mixture at a pressure close to the atmospheric pressure with finely-divided water droplets . The amount of heat being removed is efficiently regulated by changing the water weight content in a two-component flow on signals from HLMC temperature sensors at the heat-exchanger outlet. The heat developments generated in the reactor core may be removed through air supply from a standalone compressor and water supplied independently of the steam-turbine plant. The performance characteristics of such system are investigated and practiced to perfection on the NNSTU benches.

Heavy Liquid-Metal Coolant Technology

It is suggested to install several sensors in the BRS-GPG reactor circuit to monitor oxygen thermodynamic activity in HLMC. Upon corresponding signals from these sensors, it is suggested to monitor the status of the circuit and the serviceability of the protective oxide coatings. After receiving a signal stating a reduction in oxygen content in HLMC, it is suggested to carry out input of gaseous oxygen into the HLMC volume through capturing an oxygen-containing gaseous phase in HLMC jets (sprays) at the pump inlet. After receiving a signal stating an increase in oxygen content in HLMC, the same method is used to input a hydrogen-containing gaseous mixture into the HLMC volume at the main circulation pump inlet. The finely-divided oxidizing or fuel-rich gaseous phase within the two-component flow downstream of the pump is supplied to the reactor core and further to the steam generator sections. The gaseous mixtures are separated from HLMC at free levels in the circuit (in steam generators, pumps). Such method and the mass-exchanger have been investigated and practiced to perfection on HLMC benches at the NNSTU and are used as a standard device in an installation for testing and developing models of the BREST-OD-300 plant pump flow part with a rated consumption of the lead coolant up to 2000 t/hour [3].

4. Conclusions

Using HLMC in fast rectors is maximum effective and economical in comparison with gas, water or sodium reactors. Developing of fast reactors with HLMC is one the priority goals in atomic industry. Creating of this rector can solve many important problems, like increased safety, creation of closed fuel cycle, etc.

New unconventional technical solutions of the capacity range (50 - 250 MW) of low and medium-powered reactor plants cooled with lead and lead-bismuth coolants have been proposed and are experimentally developed in NNSTU. These solutions will improve the safety of a power-generating unit in case of the hazardous emergency "interloop leakiness of steam generator" at any reasonable amount of water and steam ingress to the reactor circuit. The new solutions make it possible to enhance the cost effectiveness of a power-generating unit through reducing the reactor circuit mass-dimensional characteristics.

Conflicts of Interest

The authors declare no conflicts of interest regarding the publication of this paper.

References

[1] Freire, L. and Andrade, D. (2021) Novel Technological Developments with Impacts on Perspectives for Mobile Nuclear Power Plants. *World Journal of Nuclear Science and Technology*, **11**, 141-158. https://doi.org/10.4236/wjnst.2021.114011

[2] Popa-Simil, L. (2021) Nuclear Power Renaissance Based on Engineered Micro-Nano-Nuclear Materials. *Energy and Power Engineering*, **13**, 65-74. https://doi.org/10.4236/epe.2021.134B007

[3] Beznosov, A.V., Dragunov, Yu.G. and Rachkov, V.I.M. (2006) Heavy Liquid-Metal Coolants in the Nuclear Power Industry. AT Publishing House, Moscow, 370.

[4] Beznosov, A.V., Bokova, T.A. and Bokov, P.A. (2017) Steam Generators: Design, Types and Applications. Nova Science Publishers, Inc., 1-134.

[5] Beznosov, A.V., Bokova, T.A., Bokov, P.A. and Meluzov, A.G. (2020) Installations and Stands for Investigating and Testing of Constructions in Lead and Lead-Bismuth Coolants. N. Novgorod Litera, 103.

[6] Beznosov, A.V., Bokova, T.A. and Bokov, P.A. (2015) Technologies and Major Equipment of Circuits Cooled with Pb, Pb-Bi. LAP LAMBERT Academic Publishing.

[7] Beznosov, A.V., Bokova, T.A. and Molodtsov, A.A. (2006) Experimental Research of Processes Accompanying Interloop Leakiness of Steam Generators Cooled with Lead and Lead-Bismuth Coolants and Optimization of Their Design. *Izvestiya VUZov. Nuclear Power Engineering*, Russian Federation, Obninsk, No. 4, 3-11.

[8] Beznosov, A.V., Molodtsov, A.A., Bokova, T.A., *et al.* (2008) Russian Invention Patent No. 2320035 RF. Nuclear Power Plant, Bull. No. 8.

[9] Beznosov, A.V., Molodtsov, A.A., Bokova, T.A., *et al.* (2007) Russian Invention Patent No. 2313143 RF. Nuclear Power Plant, Bull. No. 35.

[10] Beznosov, A.V. and Bokova, T.A. (2012) Equipment of Power Circuits Using Heavy Liquid-Metal Coolants in the Nuclear Power Industry: Training Aids. Nizhny Novgorod State Technical University, Nizhny Novgorod, 536.