USE OF RUSSIAN TECHNOLOGY OF SHIP REACTORS WITH LEAD-BISMUTH COOLANT IN NUCLEAR POWER

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Abstract

The experience of using lead-bismuth coolant in Russian nuclear submarine reactors has been presented. The fundamental statements of the concept of using the reactors cooled by lead-bismuth alloy in nuclear power have been substantiated. The key results of developments for using lead-bismuth coolant in nuclear power have been presented.

1. INTRODUCTION

In the beginning of the 1950s nearly at the same time the USA and USSR launched the development of the nuclear power installations (NPI) for nuclear submarines (NS). In both countries the work was carried out for two types of NPIs: with pressurized water reactors and reactors cooled by liquid metal coolant (LMC).

In the USA sodium was chosen as LMC as it possessed the better thermo-hydraulic characteristics. The ground-based test facility-prototype of the NPI and the experimental nuclear submarine "Sea Wolf" were constructed. Yet the operation experience has pointed that the choice of coolant which was chemically active with respect to oxygen and water has not justified itself. After several sodium\water interaction, the RI was decommissioned together with the compartment and replaced by the pressurized water RI.

In the USSR the lead-bismuth eutectic alloy was chosen as LMC [1].

In the USA the scientific and research development works were conducted on using lead-bismuth coolant (LBC), but the alternative to solving the problem of corrosion resistance of structure materials and maintenance of coolant quality (the coolant technology) did not give any positive results and those works were stopped.

In our country the comprehensive problem on coolant technology, structure materials corrosion and mass transfer has been solved as a result of systematic work of several organizations (IPPE, TSNII KM "Prometey", EDO "Gidropress", OKBM, NITI and some others) for about 15 years. (Some failures happened in early days of mastering the new technology). The solution to this problem has ensured long and reliable operation of NPIs using LMC at the NSs.

Proceeding from the experience gained from design and operation of NPIs using LBC, there have been developed both the concept of exploiting the reactors cooled by LBC for nuclear power plants (NPP) and a number of proposals for using LBC in nuclear power (NP).

2. THE ANALYSIS OF EXPERIENCE OF OPERATING THE REACTOR INSTALLATIONS USING LEAD-BISMUTH COOLANT AND THE ACCIDENTS HAPPENED

When the last NS with the RI cooled by LBC was removed from the Navy, the specific phase of ship nuclear power development had been completed. The innovative nuclear power technology which could not be comparable with any other one in the world was demonstrated to our industry.

Due to ongoing developments of the RIs using LBC for civilian NPPs the experience gained needs for the thorough analysis and control in these designs in order to make the best use of the LBC advantages (high boiling point, chemical inertness, etc.) and to minimize the effect of its disadvantages (melting point is ~ 125°C).

The operation experience also accompanied by the number of accidents which are inevitable for any new technology mastering (the history of technique has demonstrated it) and revealed the difficulties of servicing these RIs at their base places and refueling has received inconclusive assessment by the experts, who have been familiarized with it more or less.

Some authors, who have been familiarized with this experience only by "hearsay", allege in their memoirs that for the NSs adoption of Projects 705 & 705K (class "Alpha" according to the NATO terminology) of RIs using LBC was a tragic error.

Below there is the presentation of the key results of operating the RIs using LBC, the results of analysis of the deadly serious accidents happened, causes of the difficulties in servicing the RIs at the base places and fuel reloading and at the other phases of RIs life time, measures which have been realized in the designs of the RIs using LBC and which eliminates the accidents causes and operation difficulties. This presentation enables us to make an unbiased assessment of the experience gained [2].

2.1. The Analysis of the Accidents Happened and Difficulties of RIs Base Serving

In the course of design and operation of RIs using LBC there were accidents at three NSs, that was the cause of impossibility of further NSs operation. These were the RI accident at the left side of the NS of Project 645 in 1968 [3] when the core was melted partially, the OK-550 RI accident at the NS of Project 705 (task order 900) in 1971 when the additional pipelines of the primary circuit lost their tightness and the BM-40/A RI accident at the NS of Project 705K (task order 105) in 1982 when the global corrosion damage of steam generator (SG) pipe system of the water-steam circuit happened and there was about 150 l spill of radioactive coolant into the compartment [2].

One of the difficulties of RI servicing at the NS base places and refueling is the necessity of continuous steam ingress into the steam heating system (SHS) of the primary circuit in order to provide the liquid form of coolant and join up periodically the RI with the base installation to perform the maintenance works on coolant technology.

As far as the RIs of the NPPs are concerned, the problem on coolant liquid form maintenance is not so urgent because of existing the outside power sources and stationary arrangement of RIs.

Below there is given the description of the accidents mentioned in the course of their progress, the analysis of their origin causes and technical measures on their elimination.

2.1.1. The RI Accident at the NS of Project 645.

It was the only accident the cause of which is concerned with LBC using. As a result of low studying physical and chemical coolant operation processes, the substantiated specifications on impurity compositions in coolant, instrumentation for coolant quality control and equipment to provide the maintenance of required coolant qualities in the course of operation have been lacked. This level of LBC knowledge can be compared with that of water coolant when it was allowed to convey water from the water pipeline into the steam boiler.

As a result of that operation, uncontrollable accumulation of significant masses of lead oxides in the primary circuit happened, they could have formed when the pipelines of the primary circuit gas system, which were necessary for its repair, were depressurized, and thus the air penetrated into the primary circuit. Besides that, the primary circuit was contaminated by products of oil pyrolysis, which was the working medium for seals of rotational shafts of pumps that provided the gas leak proofness of the primary circuit. Masses of oil were spilled into the primary circuit because the oil seals had not been reliable enough.

When the rate of SG leakage increased suddenly (it had started some time before the accident), the oxides accumulated and other impurities filled the core, that was the cause of the violent decline of heat removal. Negative temperature reactivity effect was the cause of transfer the automatic power control rod up to the upper switch terminal and spontaneous power reducing to 7% of nominal one. This was the first symptom of the accident.

But the operational documentation did not include any necessary instructions for the operator how to act when that kind of situation arose. Instead of resetting the emergency protection (EP) at the left side reactor, he followed the commander's directions (it occurred in the course of navy training) and tried to maintain the given power level by continuous removal of compensative rods (CR) out of the core. All reactivity reserve for 12 CR was released in about 30 minutes, though it was intended to provide for the power reserve generation about 4000 efficient hours. When CR were removed, the fuel in the local core area, where heat removal was deteriorated, melted and left the core together with the coolant flow. Signals of radiation hazard in the compartment that called for shutting down the RI and removing the crew into the compartments being distantly removed from the RI were not taken into account.

After this accident the works on the coolant technology problem have been launched. For many years these works have been carried out at the number of organizations under the scientific supervision of SSC RF IPPE. As a result, the problem has been solved successfully and the many-year experience of the following RIs operation has corroborated it.

Later the NS of Project 645 was withdrawn from the Navy and after special preservation of the RI and reactor compartment sunk in the Kara Sea.

It is necessary to point out the main technical measures for eliminating the causes of such accidents:

- in order to eliminate the accumulation of oxides the maintenance of some excess inert gas pressure in the gas system of the primary circuit has been provided when repair works of equipment and fuel reload are to be performed. In order to eliminate the possibility of air penetration into the primary circuit and the radioactivity yield into the environment the most possible tightness of it has been provided. For this purpose special repairing and refueling equipment have been developed;
- the sensors of thermodynamic oxygen activity which enable to control the content of oxygen dissolved in LBC and detect alloy oxidizing at the very early phases have been designed and introduced;
- rejecting the use of the oil seals of the pumps shafts and adoption of water seals or gas-tight electric drivers of the primary circuit pumps. This eliminates the oil penetration into the primary circuit and contamination of LBC by the products of oil pyrolysis ;
- using the ejection system of high-temperature hydrogen regeneration that has been built into the RI and ensures chemical recovery of lead oxides by hydrogen (the explosion proof compound of helium and hydrogen is used) and enables, if necessary, to purify even hard contaminated circuit from lead-oxides;
- using the continuously operated system of coolant purification from irreducible impurities on the glass fabric filters;
- using the automatic system of coolant quality control which is equipped by sensors of continuous control of coolant quality and protective gas, ensures the preservation of oxide films on the

surfaces of the primary circuit structure materials contacting with coolant and eliminates their corrosion deterioration as well as ensures the early diagnostics of sabnormal states.

2.1.2. The RI Failure at the NS of Project 705 (Task Order 900)

Since the beginning of the RI testings in 1970 and its further operation in 1971 and in 1972, the RI operation has been accompanied by the higher content of moisture in the air of the tight compartment (TC) where the RI was mounted. The tests have demonstrated that the causes of moisture were bad air-tightness of the seal of one SG cover because of the flaw in the nickel gasket, that was changed then, and steam leakage through the steam heating system welds which have been made unsoundly, and there was no possibility to eliminate this leakage because of compact assembly.

As a result of cold surface sweating inside TC, forming water drops were the cause of wetting the heat insulator and "dry" protection materials which involved chlorides.The drops of water saturated with chlorides touched the primary circuit hot ancillary pipelines made from austenite steel and gave rise to their corrosion cracking on the outside surface that has been fully verified by the results of the RI inspection performed. Through corrosion damages of the primary circuit ancillary pipelines at two of three heatexchanger loops and impossibility of their repair because of compact assembly caused the decision of removing this NS out of the Navy and carrying out the RI inspection.

Thus, this accident is not concerned with LBC use. Similar accidents followed by the RI failure happened with pressurized water reactors when sea water touched the rustproof pipelines of the primary circuit.

Technical measures eliminating the causes of such accidents are the following:

In the designed perspective RI it has been used the pool type integral system of the primary circuit arrangement, which eliminates fully any primary circuit pipelines out of the monoblock unit vessel including comparatively thin-wall ancillary pipelines of small diameter, there are no valves. Therefore, ramified system of steam heating is eliminated. Fabricating the RI monoblock unit under the plant conditions ensures high quality and delivery of the reactor block available for operation. The integral arrangement almost completely eliminates the possibility of coolant leakage. Besides that, the use of safety vessel is provided. The looser RI arrangement at NPP ensures the better conditions for performing the assembly works and controlling their quality.

2.1.3. The RI Accident at the NS of Project 705K (Task Order 105)

Global corrosion damage of the SG evaporation sections pipes made from perlite steel occurred as a result of not meeting the requirements for the water-chemical regime (WCR) for feeding water of the SG. It was the result of the fact that under real operation conditions the way of reducing the oxygen content in feeding water by electron-ion-exchanging filter with copper-containing charge, which was provided by the Project, caused to copper escape into the secondary circuit that was the cause of severe electric-chemical corrosion of the piping system of the SG evaporation sections.

As a result of through pipes damage, steam from the secondary circuit began to penetrate into the primary circuit, where after separation from coolant it condensed in the emergency condenser (EC) specially provided in gas system in case of leakage in the SG. As internal volume of the EC had been filled by the condensate step by step, according to the signal the operator many times drained the EC by removing the condensate accumulated into the suitable reservoir, and thus he eliminated the essential pressure increase in the primary circuit gas system.

However, the EC drainage was stopped because of not clear reasons. Heatexchanging surface of the EC was completely flooded by water and condensation of steam penetrating stopped. The pressure increase began at the primary circuit gas system. The strength of the gas system and primary circuit could bear the full working pressure of the secondary circuit. That is why in that case there could not be any tightness loss of the primary circuit.

Nevertheless, the tightness loss occurred, and it was caused by the following. The gas pocket in the leakage reinjection pump (LRP) located in the pump tank below the LBC level had adjusting manometer with ultimate pressure of 4 kg per cm^2 . According to the instruction, if the RI was in operation, this manometer had to be shut off by the valve. The instruction requirement was violated and the valve occurred to be open. Due to this fact, when steam pressure in the gas pocket of the LRP tank reached ~ 6 kg/cm^2 and the LBC level in the internal pocket of the LRP was increased with corresponding increase of gas pressure, the sensitive manometer element destroyed, gas escaped from the pump pocket, and under the steam pressure that was the cause of filling up the gas pocket of the LRP by lead-bismuth alloy and its further leakage through the damaged manometer into the inhabited section of the reactor compartment. (The scheme is presented in Fig.1).

Radioactive air contamination by polonium-210 aerosols reached 10 MPC. Due to the following proper actions, the crew irradiation and radioactive contamination were within the permissible limits. The analyses of crew bio-samples, which had been performed by medical service, demonstrated that none of the crew had the content of polonium-210 more than 10 percent of the maximal permissible value.

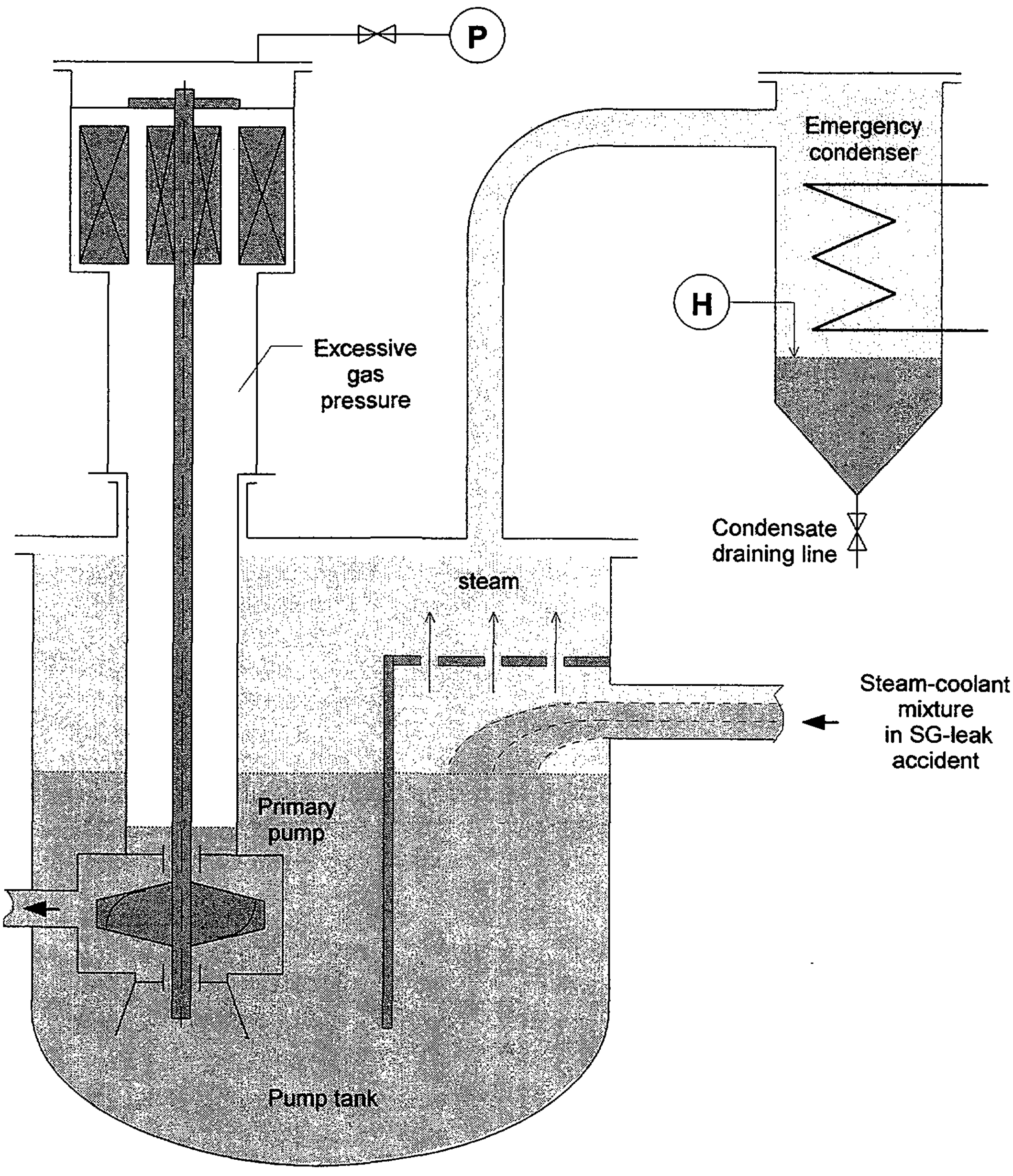


Fig. 1.

RI examining showed that it could be reconditioned. However, another decision was accepted. It was decided to change the whole reactor compartment of this NS by the new one fabricated earlier. The motive for this decision was the following. In the course of this RI fabrication at the Machine-building plant in Podolsk there was faulty change of the SHS pipes fabricated from high-nickel corrosion-resistant steel by the pipes made from common stainless steel of the same size.

This error was found out after the RI unit was fabricated and it was impossible to change the pipes. Because the service life of stainless steel pipes was restricted by corrosion conditions, there was accepted the decision to limit the service life time of reactor unit up to 25000 hours and fabricate the reserve RI unit in order to use it for changing off-spec one in the course of the NS overhaul period. In 1982 the service life of the SHS stainless steel pipes had to be expired, and that was the motive for changing the NS reactor compartment.

The analysis performed has demonstrated that the cause of this accident and the accident at the NS of task order 900 is not concerned with the use of LBC.

The following technical measures ensure elimination of such accidents at the new generation RIs:

- eliminating copper-containing materials out of water-steam circuit;
- using more corrosion-resistant steel as material for SG pipes under water -steam conditions instead of low-alloyed steel of perlite class;
- providing passive drainage of the EC when it is filling up by the condensate up to the given level;
- uniting the gas volume inside the pump electric motor with the total one above the free coolant level;
- providing the great extent of control-fitness and repair-fitness of the RI.

2.2. The Problems of Mastering the RIs Cooled by LBC

Among the key problems which have been solved in the course of design and operation of this type installations we must emphasize **the LBC technology problem** - i.e., development of systems and devices ensuring measurement and maintenance of the LBC quality required during its long-time operation both under normal conditions of leak-proof circuit and in the case of partial tightness loss of tightness of the circuit in the courses of repair works and reactor reload. Functioning those systems and devices is necessary for eliminating structure materials corrosion and slagging the circuit by lead oxides [4]. It should be pointed out that in the early days of mastering LBC, when the necessity of developing and implementing the measures on the coolant technology had not been realized, there were cases of reducing the coolant cross sections up to the full blockage of coolant flow rate because of depositing lead oxides and other impurities and all the resulted consequences (see 2.1.1.).

Corrosion resistance of structure materials have been ensured by using special steel alloyage, applying protective films to them in advance and maintaining necessary concentration of corrosion inhibitor - dissolved oxygen - in LBC [5]. The importance of these measures has been corroborated by the fact that when the necessary coolant quality was maintained, for several thousands of hours under the 650°C temperature there were no corrosion of fuel element steel cladding. However, when the dissolved oxygen concentration was inadequate (under the specially provided conditions), it took about 20 hours for through corrosion damage of the 5 mm thickness pipe under the same temperature.

Melting point of LBC is about 125°C. LBC maintaining in liquid state under all RI operation regimes is ensured by using the SG with multiple circulation over the secondary circuit, besides that, the inlet temperature of the water supplied to the SG is higher than LBC melting point. For initial heating-up and maintaining the primary circuit under hot condition at a low level of power release in the core the system of steam or electrical heating may be used.

The substantiation of the possibility of multiple coolant "freezing-defreezing" in the RI was an important practical problem. Low shrinkage of LBC during solidifying and rather high plasticity with low strength in solid state facilitate the elimination of RI damage when alloy is transiting from liquid to solid form and its further cooling down up to the ambient temperature. A special order of the temperature-time heating regime has been developed for safe RI "defreezing". This problem is dealt with in paper [6].

The specific feature of LBC is the formation of α-active polonium-210 radionuclide with a half-life of ~ 138 days when bismuth is irradiated with neutrons.

The major reason for its radiation danger is the formation of radioactive polonium aerosols when hot LBC contacts with air. It could happen under conditions of emergency tightness loss of the primary circuit and coolant spillage. In this case, as the RI operation experience at the NS has displayed, the yield of Po aerosols and air radioactivity (according to the thermodynamics laws) reduce quickly with temperature decreasing and spilled alloy solidifying. Fast solidification of spilled LBC restricts the area of radioactive contamination and simplifies its removal in the form of solid radioactive wastes.

Low polonium concentration in the coolant (at the level of 10^{-8} of at.%) and formation of thermodynamically proof chemical polonium-lead compound, which additionally reduces the polonium pressure by 1000 times, individual and collective protection personnel's facilities developed, methods for equipment decontamination and recording activity on the surfaces, methods of performing repair works have lead to that fact that for a long-term period of operating the reactor using LBC including the repair works of the primary circuit equipment and removal of coolant leakage (also in the case of LBC penetrating into the secondary circuit as a result of NS crew errors), there have been no cases of personnel extrairradiation by this radionuclide above the permissible limits. This positive practical result agrees with conclusions of foreign experts who have investigated the polonium hazard problem if LBC is used for nuclear reactor cooling [7], [8]. Paper [9] is focused on the analysis of this important problem

The following characteristics have been obtained in the course of NPI tests and operation: power and parameters of installation, the campaign lifetime, the reactivity margin, reactivity coefficients, poisoning effects, temperature distributions, dynamic parameters, coolant radioactivity, dose rates of neutron and γ-radiation behind the shield. They were in sufficiently good agreement with calculation results.

Among the positive properties of the RI using LBC, which have been discovered in the course of operation, one can point out the simplicity of control, high manoeuvrability and short time of reaching the power regime out of subcritical reactor state, the possibility of RI operation if there is small leakage in the SG pipe system, high repair-fitness of the SG by plugging the depressurized pipes, the possibility of RI stable operation at any low power levels, the possibility of quick changing the circulation regime of coolant with essential change of its flow rate, almost complete generation of designed power by cores under normal and acceptable conditions of leak-proofness of fuel rod claddings.

While performing repair works and reactor reloading, it is not necessary to carry out primary circuit decontamination which is concerned with collecting, storing, transporting and reprocessing masses of liquid radioactive wastes (LRW).

The cores and absorbing rods of the control and protection system (CPS), which have ensured total design power service, have demonstrated high operation serviceability.

During the last 10 years of NPIs operation there have been no problems with either structure materials corrosion in the primary circuit or the deviations from specifications on the circuit purity.

Experience of development and operation of NPIs using LBC at the NSs and ground-based facilities-prototypes enables to make a number of practically important conclusions on the arrangement and equipment of the primary circuit for reactors cooled by LBC for NPPs [10] .

The best engineering and economical parameters should be expected if the arrangement of the primary circuit equipment is integral one.

The most convenient design scheme of the SG is that one in which liquid metal circulation is performed in the interpipe space and that of water or steam is carried out in pipes. That design ensures the possibility of repairing the SG by plugging the separate pipe, which has lost its tightness, without dismantling the SG or breaking the primary circuit.

Shut-down regimes, the regimes of start-up and cooling down are realized in the easiest way if the SG operates by using multiple circulation in the water -steam circuit.

Both mechanical upright pumps with a turbine or electric driver and electromagnetic pumps may be used for coolant circulation.

During last period eight NSs with RIs using liquid metal coolant (LMC) have been constructed [11]. The first experimental NS of Project 645 had two reactors. Other seven NSs of Project 705 (according to the NATO terminology - "Alpha") had one reactor. Due to its high-speed qualities this NS was put down into the Guinness Book of Records. Besides that, two full-scale ground-based reactor test facilities were constructed and put into operation (the prototypes of NPIs at NSs at IPPE in Obninsk and at NITI in Sosnovy Bor). The total RI operating time added up to about 80 reactor-years.

3. THE CONCEPT OF USING REACTORS COOLED BY LBC INNUCLEAR POWER

3.1. The Fundamental Statements of Safety Concept

Priority for realizing the safety requirements in comparison with the other has become obvious after happening the Chernobyl accident.

An intensive development of NPP new concepts for the future has been launched all over the world, their safety would be more based on RI inherent self-protection allowing to eliminate deterministically the possibility of the severest accidents which result in catastrophic consequences not only in the event of failing the safety technical systems and personnel's errors but also in the event of ill-intentioned actions (terrorism, diversions, military attacks) and natural disasters.

Nowadays there are two approaches to ensuring the RI safety.

The first approach is traditional and is based on increasing the amount and efficiency of various protecting and localizing systems which reduce the possibility of severe accidents and hazardous factor of their results. Practical realization of this approach, theoretically based on probabalistic safety analysis (PSA), results in more complicated and expensive installation, deterioration of its other characteristics and, nevertheless, in principle, does not exclude the possibility of severe accident with catastrophic consequences because there have not been eliminated the internal causes of arising the accident.

The second approach is based on the concept of the RI with inherent safety ensuring its self-protection in which the causes of arising the severe accidents with catastrophic consequences have been deterministically eliminated by nature laws. This approach does not require the construction a lot of protecting and localizing systems, which, in some cases, themselves may become the causes of accidents.

There is no need in complex substantiations of safety either by using many calculative and experimental works in the frameworks of abstract scenarios for severe accidents in progress, by constracting expensive large-scale test facilities. Most consistently this approach has been developed in the USA by Prof. A. Weinberg [12], and in Russia by Prof. V.V. Orlov[13].

Neither the first approach nor the second one is realized in its pure form. Each of them possesses the elements of the other. Nevertheless, proceeding from the concept "SAFETY CULTURE" [14], when developing the advanced RI, it seems necessary to use the approach most based on the inherent safety.

It is due to the fact, that the large-scale advanced RI must meet the stricter safety requirements because the probability of severe accident specified in active standard documentation gives the socially acceptable expected value of frequency of its realization only for today's level of the NPP reactors service time, which is less than 10000 reactor-years.

If in the future this factor increases ten times, it will take us to reduce this probability ten times for 1 reactor-year in order to keep today's expected value of frequency of severe accident realization [15].

For traditional types of NPPs, which safety is usually substantiated by PSA methods, it takes us to increase the amount of protective barriers (double containment), localizing systems (corium trap), etc. The result of it will be inevitable going up the NPPs cost and losing their competitive ability in comparison with electric plants that use organic fuel. We are watching the symptoms of that process even now.

Inadequacy in substantiating NPPs safety for future NP by using PSA methods is the result of the fact that the PSA considers failures of technical devices and operational personnel's errors to be occasional events, which probability may be estimated only with high uncertainty. Low probability of the severe accident is neither the proof of its impossibility nor the evidence of the fact that it might happen not earlier than thousands or tens of thousands of years later. Besides that, if there were people's ill-intentioned actions, that should be taken into account, these events would not be occasional but programmed in advance. In that case the PSA conclusions lose their validity completely.

3.2. Substantiation of Choosing the Fast Reactor Cooled by LBC

Liquid metal cooled fast reactors are classified as RIs which safety is ensured principally due to their inherent self-protection. It is associated with a number of their internal features.

Absence of poisoning effects in the fast reactor (FR), low value of negative temperature reactivity coefficient, compensation of fuel burn-up and slagging processes by plutonium generation as well as partial reloadings enable to ensure the operative reactivity margin to be less than delayed neutron share and to diminish or eliminate the probability of runaway by prompt neutrons in the reactor under operation conditions.

LMC used for FR cooling motivates considerably constructional and thermal scheme of the RI, technical and economical characteristics of the NPP.

Among the LMCs used in NP sodium is the most commonly used.

Choosing this LMC for fast reactors was caused by its possibility of intensive heat removal due to its good thermal and physical properties. It enabled to provide short plutonium doubling time that was obligatory requirement at the early phases of designing the fast breeder reactors in the 60s and 70s years, and was caused by the fact of unproved forecast of very high NP development rate and, therefore, need for fuel self-providing. That was the reason why in the 50s Academician A.I.Leypunsky, who considered various LMC for cooling the FRs, preferred sodium, though LBC was initially considered for these purposes [16].

At present and in the foreseeable future there is no need for such short plutonium doubling time which can maintain the FR cooled by sodium. The necessity for designing fast breeder reactors has been postponed to many decades [17]. It enables us to return to the opportunity of using LBC for cooling the FR, one should take into account many-year experience we have gained by using this coolant in the NS reactors designed under the IPPE scientific supervision.

It should be highlighted that the experience of using sodium coolant has been gained under the conditions of industrial operation of power reactors at NPPs and could be used at once, whereas the experience of using LBC has been gained under the conditions of RI operation at the NS, which were different from those at NPP in scale factor, and requires applicable adaptation to new conditions. However, these circumstances should not be the reason for not using LBC in NP if it has weighty backgrounds.

These backgrounds are the following:

- Enhancing reactor safety due to elimination of coolant boiling (coolant boiling point is ~ 1670°C, boiling point of sodium is ~ 870°C) in the most heat stressed fuel subassemblies (FSA) even in the case of the severest accidents. It makes the realization of coolant positive void reactivity effect practically impossible. Besides that, the use of chemically inert LBC eliminates occurrence of explosions and fires if there is coolant contacting with air and water which is possible in emergency situations;
- Improving technical and economical parameters due to using two-circuit scheme of heat removal, eliminating some safety systems, systems of accident localization, simplifying the technology of managing the SNF;
- Solving the number of principal problems on RIs reliability and safety: ensuring the proper coolant quality and its maintenance in the course of operation, ensuring radiation safety associated with forming alfha-active polonium-210 radionuclide, etc. These tasks have been developed for the NS's RIs.

The expediency of using eutectic lead-bismuth alloy as primary circuit coolant in nuclear reactors is due to its physical and chemical and thermodynamical characteristics which enable to meet the safety requirements the most completely.

Extremely high boiling point about 1670°C allows low pressure in the primary circuit with the 400...500°C outlet reactor coolant temperature that ensures high steam parameters. Thus the RI structure is simplified and its reliability increases.

Natural properties of LBC such as high boiling point practically eliminate the possibility of the primary circuit overpressurization and reactor thermal explosion if coolant is accidentally overheated, because there is no pressure increasing in this case.

Impossibility of coolant boiling enhances the reliability of heat removal from the core and safety because there is no phenomenon of heat removal crisis.

Low pressure in the primary circuit enables to reduce the thickness of reactor vessel walls and to use for its fabricating less strong austenite steel being resistant to radiation embrittlement under operation conditions and eliminate the possibility of vessel brittle damage. This enhances safety and eliminates the restrictions on temperature change rate on conditions of thermocycling strength and radiation life time of reactor vessel.

Loss of coolant with its circulation interruption through the core if there is failure of the main reactor vessel tightness (postulated accident) with suggested integral design of the RI can be eliminated by introducing the safe-guard shell, small free volume between the main reactor vessel and the safe-guard shell and impossibility of coolant boiling-up off in the case of loss of tightness by primary circuit gas system.

In the case of failing all systems of emergency cooling down (postulated accident,) elimination of core melting down under heat decay effect and keeping the vessel of small and medium-sized power reactors intact is ensured completely by passive way with large margin to boiling due to accumulation of heat in internal reactor structures and in coolant with short-time increase of its temperature. In this case heat is removed through the reactor vessel (which temperature increases correspondingly) to the tank of water storage around the reactor vessel and after its boiling away (in the case of full switching off and interruption of cooling down systems work) into the ambient air by its natural circulation.

In the case of emergency overheating and simultaneous postulated failure of emergency protection systems (EPS) down to the level which does not cause the core damage the reactor power decrease is ensured by reactivity negative feed backs.

In the core and within the RI there are no materials which yield hydrogen as a result of thermal and radiation exposures and chemical reactions with coolant. The coolant itself reacts with water and air very slightly, their contact is possible if the circuit has lost its tightness. Thus, the possibility of arising chemical explosions and fires caused by internal reasons is eliminated completely.

Slight chemical activity of LBC enables to realize two-circuit scheme of heat removal, so the RIs become simpler and cheaper. The emergency processes associated with tightness loss by primary circuit pipes and intercircuit SG leaks occur without hydrogen yield and any fire and explosion hazards.

The elimination of water or steam penetration into the core (if the SG leakage is large) and consequent overpressurizing the reactor vessel designed for maximal possible pressure, if that accident occurs, are ensured by coolant circulation scheme. In this scheme steam bubbles and water drops are thrown out on the free coolant level by upgoing coolant flow. Thereby steam effective separation occurs in the gas space of the primary circuit above the coolant level, whence steam goes to the system of emergency condensers operating passively, and through the ruptural membranes it goes to bubbler if these condensers fail postulatedly.

The operation experience of the RI using LBC at the NS has illustrated the possibility of RI safe operation during some time under conditions of small SG leakage, which does not cause any significant deviations of the designed engineering parameters. This fact allows to realize necessary repair works not urgently but at the convenient time.

The coolant does not boil if the primary circuit loses its tightness, and has the property to retain iodine, as a rule, its radionuclides represent the major factor of radiation danger just after the accident as well as the other fission products (inert gases are exception) and actinides. This reduces sharply a scale of radiation consequences of that accident in comparison with pressurized water reactors.

The containment above reactor serves as the additional safety system barrier. Its main purpose is the protection against external effects. Low storage of potential energy in the primary circuit restricts the RI destruction scale to only external impacts.

Extremely high safety potential peculiar to this type of the RI is characterized by the fact that even when such initial events as containment destruction and primary circuit loss of tightness coincide, neither reactor runaway, nor explosion and fire occur, and the radioactivity yield is lower than that one which requires the population evacuation.

Taking into account that energy stored in the coolant (heating, chemical and compression potential energy) is minimal in comparison with other coolants used and previously mentioned physical special characteristics of fast reactors and RI integral design, one could look forward to designing the RI of ultimately achievable self-protection.

It makes possible and expedient the use of that type reactors not only for electric power generation at the NPP but also for simultaneous heat generation at the NHPP with their locating near large cities or for sea water desalination. On the base of such reactors NP would become not only socially acceptable for population but also socially attractive if it gained economical competitive ability with heat electric generation plants using organic fuel (we have all backgrounds for it).

The high level of self-protection of reactors considered makes expedient their use for nuclear transmuting NP long-lived radioactive waste.

3.3. The Problems of Increasing the Overall Bismuth Production and Using the Lead Coolant

The factor limiting LBC use in large-scale future NP depends on deficiency of today's bismuth manufacturing which has been determined by its consumption level.

Today's bismuth situation looks like uranium one during the period from 1906 to 1940 when only 4000 t of uranium were mined over the 34 years. But in 1980 the world's uranium mining (except the ex-USSR) reached 40000 t per year [18]. The explored uranium resources increased greatly too.

One should point out that bismuth content in the Earth's crust is one-fifth as high as that of lead [19]. However, deposits of bismuth with high content of about 5-25% are very rare and locate in Bolivia, Tasmania, Peru and Spain. That is why 90% of world bismuth has been manufactured from the wastes of lead-refining, copper-smelting and tinning plants.

The experience of the USA and Japan has revealed that equipping copper-smelting, tungsten and lead plants with dust-catching and dust-utilizing systems can essentially enhance bismuth extraction and has great significance for the environment.

If the problem of developing the future large-scale NP with that type of reactors is solved positively, it is necessary to carry out bismuth geology works that have not been performed in necessary scale until today because of lacking demands.

In Russia that work can be launched at MINATOM enterprises being the members of AO "Atomredmetzoloto". According to VNIPI Promtechnologia information, the production of bismuth of about 2500-3000 tons per year, together with gold and other metals, may be organized in the south-east of Chita Region where the resources of gold-bismuth ore have already been explored. That volume of bismuth production will ensure the year input of 2,5-3,0 GWe for the NPP using RI SVBR-75 and 1 GWe for the NPP using SVBR-600 (if aggregate power of the RI increases, specific expenditure of bismuth increases too).

The cost of bismuth is ten times as much as that of lead and is a very small part of capital costs for NPP construction. It should be taken into account, that coolant is not spent and could be used again in other RIs.

RDIPE proposes to consider lead coolant as an alternative to lead-bismuth alloy because the scales of lead production do not limit the rate of large-scale NP development.

However, use of lead coolant is associated with some engineering problems. Due to the higher lead melting point (it is 327°C in comparison with 125°C for eutectic coolant), lead coolant temperature must be increased significantly. It complicates the solution to the problem on coolant technology, structure materials corrosion and mass transfer. Being applied to the lead-bismuth coolant, the problem has taken about 15 years for its solving. It will demand changes from existing steam parameters to overcritical ones which have not been used at the NPP. Besides that, it results in more complicated RI operation because of great possibility of forming solid "sows" in the primary circuit under transitive regimes, accidents, repairs, refueling. Nowadays the works on mastering the lead coolant are at their initial phase.

Taking into account all mentioned above, use of lead coolant would be justified only if the rates of power capacities increase for the NPPs with the RIs considered were high enough, and expenditures for increasing the annual bismuth production and its cost were put up as much that the increase of specific capital costs of NPP construction would not be economically reasonable.

It should be highlighted that there is no sharp boundary between lead-bismuth eutectic alloy and pure lead. As bismuth has been in deficiency, one can consider non-eutectic alloy with bismuth content decreased up to 10% (versus 56% in eutectic alloy). Being compared with lead coolant, its melting temperature is decreased by 77°C (to 250°C) and that facilitates RI operation and reduces maximal temperatures of fuel elements claddings up to the values tested for eutectic coolant under the conditions of long-term operation tests.

Mastering the technology of alloy with 10% of bismuth content further ensures, if necessary, gradually to introduce lead coolant use.

4.PROPOSALS ON USING LEAD-BISMUTH COOLANT

4.1. The Near-Term Prospect up to the year 2030 (On the Basis of Multipurposed Reactor Module SVBR-75/100)

Currently the NP development meets significant difficulties in many countries. First of all it relates to complicating and cost rising of NPPs due to essential enhancing the safety requirements.

Increasing the unit power of the nuclear power units with light water reactors aimed at specific capital costs reducing enhances the total cost of the construction and the constructing term. This is unfavorable for receiving a credit.

The possibility and expediency of developing the NP based on unified small power reactor modules SVBR-75/100 with fast neutron reactors cooled by lead-bismuth eutectic coolant (LBC) is substantiated for the nearest decades in the paper.

The properties of self-protecting the RIs using LBC allow to create ecologically pure nuclear power sources of enhanced safety.

For the small power RIs using LBC long-term passive heat removal through the reactor vessel into the environment is ensured without core damage in postulated accident with complete failure of heat removal and emergency protection systems.

Characteristic features of small nuclear power plants (SNPP) are

– transportable or modular-transportable design if SNPP is assembled by using completed units with possibility of their transporting by any kind of transport;

– improved (in comparison with RIs of other types) weight and overall dimensions parameters of SNPP and its separate units;

– long work without refueling (about ten years) that enables not to reload fuel at the site of SNPP operation if there is no suitable infrastructure (this possibility is realized in the RI with the power lower than 100 MWe), and reduces the risk of unauthorized plutonium proliferation;

– the possibility of SNPP reactor unit transportation in order to reload fuel and, after serving its term, to the central base of technical service under the conditions of nuclear and radiation safety and additional physical protection (coolant and fuel assemblies "freezing" in the reactor);

– economical competitive ability with other types of SNPP.

The project of that SNPP "Angstrem") gained the first place among its power group in the SNPP project competition held by Nuclear Society [20].

The closeness of scale factors of the RI at SNPP and the NPI at the NS ensures continuity of main engineering solutions and enables creation of SNPP in very short terms if there is a certain customer. SNPPs of the type considered satisfy the principle which is the most acceptable for the customer: "I build, own, operate, decommission up to the "green" lawn".

The great interest to SNPP of 50-150 MWe power was revealed at the IAEA working group meeting on using ship reactor technologies for civilian needs (July 1998, Obninsk, Russia) such as producing electric energy, desalinated water and heat generation [21

It is followed from the USA Report, in which demand for such reactors is valued to be 50-70 after 2015. The USA suggests to organize the international cooperation on realizing the design that satisfies these requirements the most completely.

The analysis of unique experience of operating the NPIs using LBC, that has been performed by Russian experts, has testified that in developing small and medium-sized power reactors for civilian NP needs the causes of the accidents happened and the operation difficulties have been eliminated.

The designs considered have shown that NPPs with reactors cooled by LBC satisfy the modern safety requirements, non-proliferation of Pu and can be economically competitive with traditional NPPs. For those NPPs entering the market of nuclear technologies (they could be attractive for developing countries and for economically effective replacement of removing NPP's powers capacities) there is need for construction and operation of that reactor in Russia. Now that perspective is been studied.

On the basis of the concept presented and the experience gained during the last years under the scientific supervision of SSC RF IPPE named after Academician A.I.Leypunsky a number of proposals for civilian small and medium-sized NPPs have been developed at EDO "Gidropress: renovating of NPP units which operation term has been exhausted; regional nuclear heat power plants (NHPP) of 100-300 MWe power which need near cities location; large power modular NPPs (~ 1000 MWe) like US concept PRISM or Japanese concept 4S; nuclear power complexes for sea water desalinating in developing countries which meet non-proliferation requirements, reactors for Pu utilization and minor actinides transmutation.

Their brief description is presented below.

4.1.1.Multipurposed Reactor Module SVBR-75/100

RI SVBR-75/100 is designed for generating steam which parameters enable to use it as working medium in thermodynamical cycle of turbogenerator installations. It is possible to vary the steam parameters according to the needs. Today the base variant of RI SVBR-75 [22] has been developed for generating the saturated steam under the pressure of 3,24 MPa, i.e. the pressure which is produced by the SG of the Novovoronezh NPP (NVNPP) 2-nd unit, and when turbogenerator with intermediate steam superheating is used, this enables to generate electric power of about 75 MWe when working under condensation regime.

The design of RI SVBR-75 have two-circuit scheme of LBC heat removal for the primary circuit and steam-water for the secondary circuit. The integral design of the pool type is used for the RI primary circuit (see Fig. 2). It enables to mount the primary circuit equipment inside the one vessel. RI SVBR-75 includes the removable part with the core (the reactor itself), 12 SG modules with compulsory circulation over the primary circuit and natural circulation over the secondary circuit, 2 main circulation pumps (MCP) for LBC circulation over the primary circuit, devices for controlling the LBC quality, the in-vessel radiation protection system and buffer reservoir which are the parts of the main circulation circuit (MCC).

The scheme of coolant moving within the MCC is as follows: through the windows of reactor outlet chamber the coolant heated in the core flows to the inlet of the SG twelve modules which have parallel connection. It flows from top to bottom in the intertube gap of the SG modules and is cooled there. Then the coolant penetrates into the intermediate chamber, from which it moves in the channels of in-vessel radiation protection system, cooling it, to the monoblock upper part and there it forms the free level of "cold" coolant (peripheral buffer chamber), further from the monoblock upper part the coolant flow moves to the MCP suction inlet.

The adopted circulation scheme with free levels of LBC existing in the monoblock upper part and SG module channels, which contact the gas medium, ensures the reliable separation of steam-water mixture out of coolant flow when the accidental tightness loss of SG tube system occurs, and existing of gas medium ensures the possibility of coolant's temperature changes.

Monoblock is placed in the tank and is mounted there (see Fig. 3). The tank is filled by water and is designed for cooling the RI in case of beyond design accidents. The gap between the major vessel and safe-guarding one is chosen to ensure the circulation circuit disrupture in case of accidents related to the tightness loss by the monoblock major vessel.

The secondary system is designed to operate the steam generator producing saturated steam with multiple natural circulation through the evaporator-separator, as well as to provide the scheduled and emergency RI cooling by using steam generator.

The design provides three systems of heat removal to the heat sink both in scheduled and emergency reactor core cooling. First system includes normal operation RI and turbine generator

Реакторный модуль СВБР-75.
Reactor plant module SVBR-75.

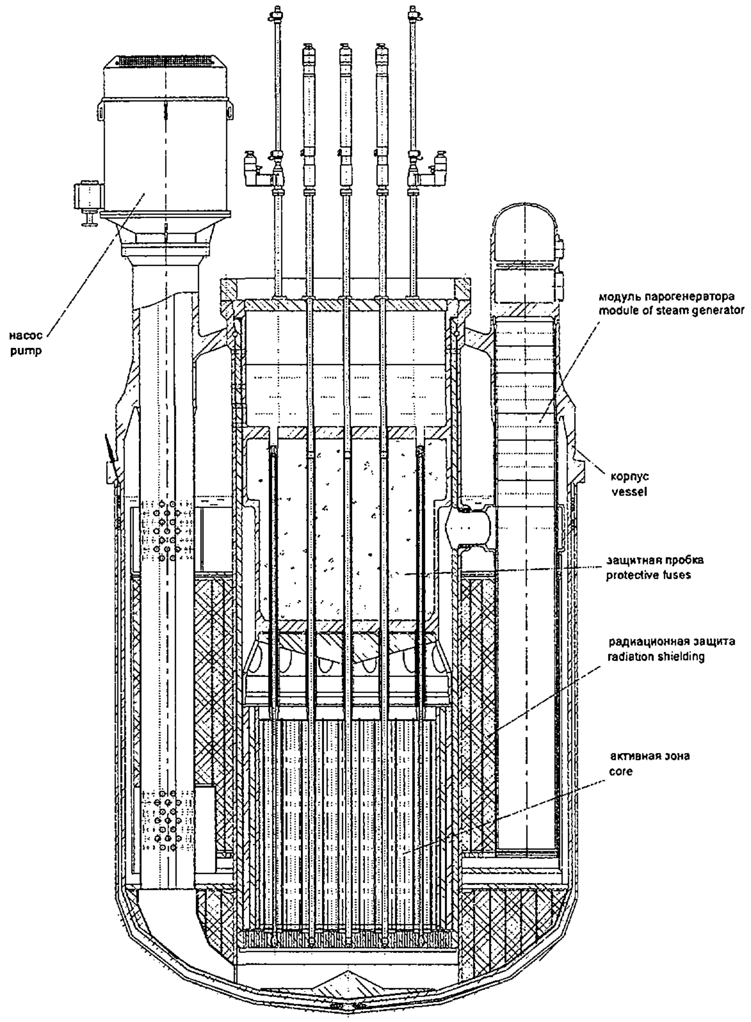


Fig. 2

equipment and systems. The system is cooled by the primary heat removal via steam generator heat exchange surfaces, steam being dumped to the turbine generator systems (TGS).

The second heat removal system is an independent cooling system (ICS), which includes, besides a part of primary and secondary circuit equipment, a loop separator-cooling condenser with natural circulation. Via this loop the heat is removed to the intermediate circuit water. This system ensures independent (from the turbine generator systems) reactor cooling and independent reactor plant operation at a constant power level up to 6 % N_{nom} at the nominal steam pressure. In case of total RI de-energizig the system ensures cooling of the reactor over several days. Connection/disconnection of ICS is realized with no operator action and without using external power supply systems.

Third core heat removal system is a passive heat removal system (PHRS). The heat is removed from the monoblock to the water storage tank located around the monoblock vessel. This system ensures the reactor core cooling in case of postulated maximal accident with all secondary equipment failed, reactor protection system failure and total de-energizing of the NPP.

The principal technical parameters of RI SVBR-75 are presented in Table 1.

RI SVBR-75 operates for eight years without core refueling. During this period there is no need in carrying out fuel works. At the initial stage the use of mastered oxide uranium fuel in the uranium

Реакторная установка СВБР-75.
Reactor plant SVBR-75.

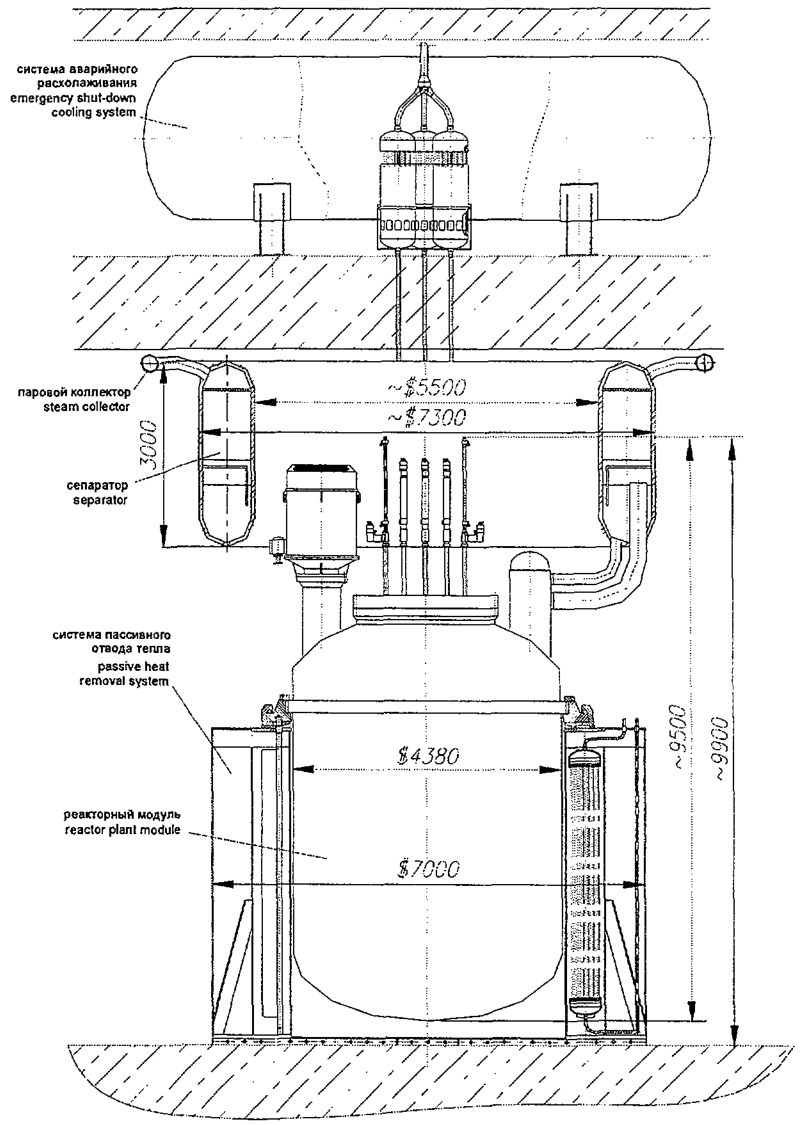


Fig. 3

closed fuel cycle is provided similar to that in reactor BN-600. Further the use of dense uranium and plutonium nitride fuel is possible. In this case the core breeding coefficient is more than 1 and in the plutonium closed fuel cycle the reactor would operate by using only depleted waste pile uranium.

The refueling is performed after lifetime ending. Refueling means the complex of works on restoring the reactor full power resources which includes core replacing works, as well as decommissioning and commissioning works associated with them. Cassette by cassette fuel unloading out of the reactor vessel is provided and loading of the fresh core as a part of new removable part is provided as well. Core refueling is realized by using special refueling equipment. Unloaded FSAs are placed in special penals with liquid lead which solidifies further.

RI SVBR-75/100 is designed on the design base of RI SVBR-75 and distinguishes from it only by SG operating in one through regime and generating the superheated ste°am of 400°C temperature and 9 MPa pressure. Thus the electric power is ensured to be of about 100 MWe.

TABLE 1. PRINCIPAL TECHNICAL PARAMETERS OF RI SVBR-75

Parameter	Value
Number of reactors	1
Rated heat power, MW	268
Electric power, MWe	75
Steam production rate t/h	About 487
Steam parameters:	
-Pressure, MPa	3,24
-Temperature	238
Feed water temperature, °C	192
Primary coolant flow rate, kg/s	11180
Primary coolant temperature, °C	
-core outlet	439
-core inlet	275
Core dimensions, DxH, m	1,65x0,9
Average value of core power stress, kW/dm^3	135
Average value of fuel element stress, kW/m	~ 22
Fuel:	
-type	UO_2
-U-235 mass loading, kg	1476
-Average U-235 enrichment, %	15,6
Steam generator (SG) numbers	2
Evaporator numbers in SG	6
Evaporator dimensions DxH, m	~ 0,6x4
Numbers of PCMPs	2
PCMP electric driver power rate, kW	400
PCMP head, MPa	~ 0,5
Primary circuit coolant volume, m^3	18
Monounit vessel dimensions, DxH, m	4,53x6,92
Designed earthquake	Of magnitude 9 (MSK)
Designed construction terms (months)	36

4.1.2. Renovation of NPP's Units with Exhausted Lifetime

The number of NPP's units which lifetime has been exhausted is growing in the NP of many countries. It needs for huge expenditures on withdrawing the units out of operation and constructing the new ones for replacing the removing power capacities. At the same time there is an opportunity for untraditional solving this task by using NPP's units renovating. Renovating means replacing the Risfor units with exhausted lifetime by new Ris, using the NPP's existing buildings and structures with full replacing of removed power capacity. However, this way of replacing removed power capacities rigidly restricts the type of the RI used for renovation:

- the possibility of mounting the RI in existing rooms after dismantling the equipment of the "old" RI (the reactor which dismantling is followed by great radiation doses is an exception);
- satisfying the regulation requirements for safety including "old" units without containment.

That way of replacing the removing power capacities of the 2-nd, 3-rd and 4-th units of NVNPP based on the unique Russian lead-bismuth technology has been suggested by SSC RF IPPE together with EDO "Gidropress" and GNIPKII Atomenergoproekt and is been developed according to the task of concern "Rosenergoatom".

SVBR-75 nominal power is chosen to be 75 MWe due to limited dimensions of NVNPP renovation units' SG compartments which do not enable to install the large power module, necessity for ensuring the equality of generated steam and feeding water consumption for SVBR-75 and RI

VVER-440 SG, possibility of reactor module complete plant fabrication and its transportation by the railway, as well as closeness of the scale factor to NS's RIs that enables to use some developed technical solutions and reduce R&D. For replacing the power capacities of the 2-nd unit four SVBR-75 modules are installed in SG compartments, and six modules are installed for each of 3-rd and 4-th units.

Arrangement of modules in SG compartments of the MCP of NVNPP's 2-nd unit is presented in Fig. 4.

The great economical efficiency of renovating the "old" units is expected: specific capital renovation cost is 560 dollars per kWe, that is half as many as that for constructing the new NPP block[23]

4.1.3. Regional NHPP Based on Module R-75/100

In the world many regions and first of all medium-sized and large cities face with serious difficulties in energy supplying especially heat supplying in winter.

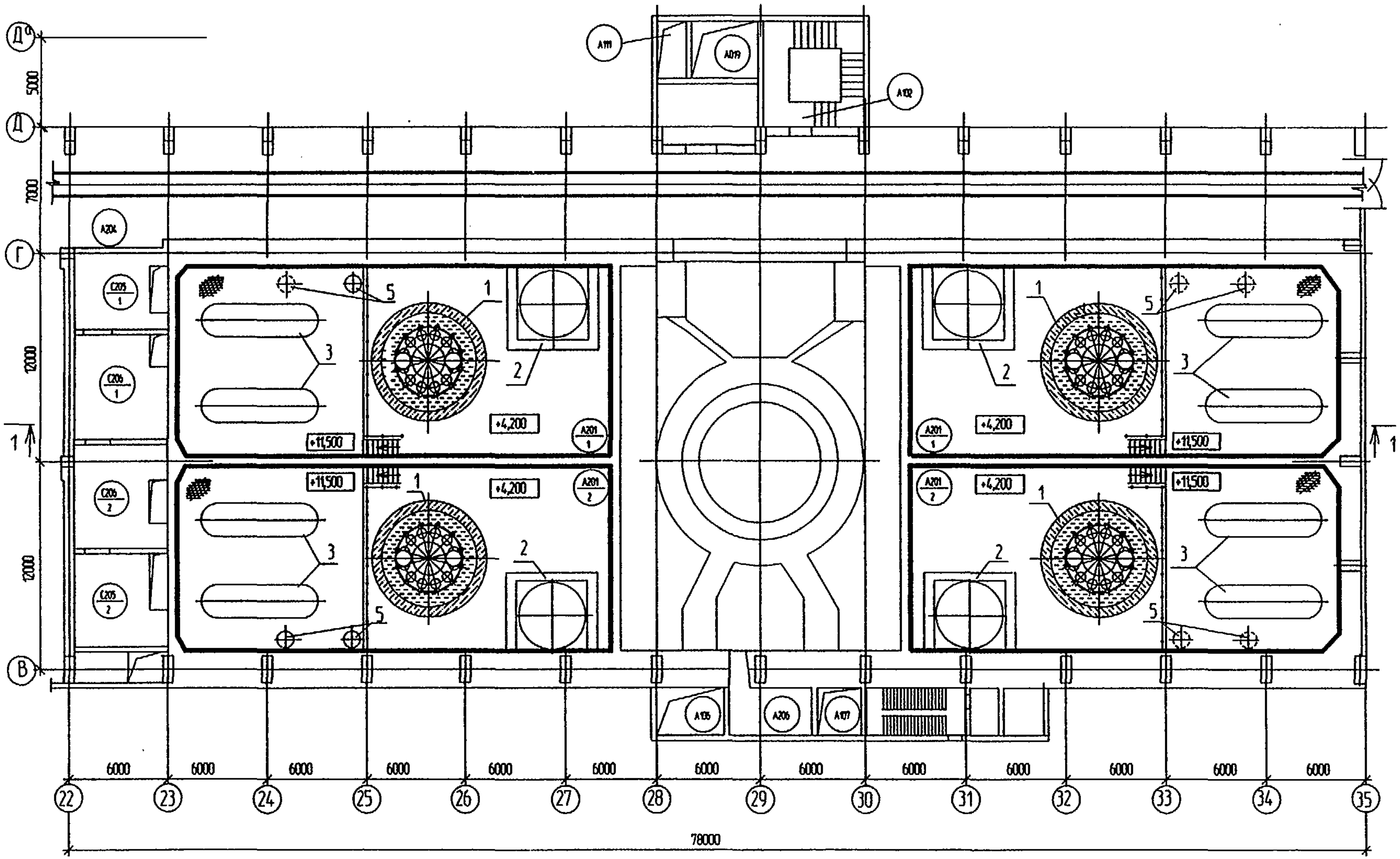
One way for solving this problem is use of nuclear heat power plants (NHPP). However, use of traditional type RIs (which use water under the high pressure for reactor cooling) for these purposes needs designing the number of additional safety systems if compared with those accepted for NPPs situated at the distance of 25 km or more from the cities. This results in going up the NHPP cost and at the same time does not eliminate the principal possibility of scarcely probable nuclear accident with severe consequences because the high pressure in the reactor, which is the internal cause of its arising, is not eliminated.

The high level of SVBR-75/100 reactor module inherent safety makes it possible and expedient to use it simultaneously for producing electric energy at the NPP and for heat generating at the NHPP which needs near city's location. So we eliminate the possibility of arising severe accidents accompanied by explosions, fires with unpermitable radioactivity exhausts which require population evacuation beyond the NHPP site not only if there are personnel's errors and equipment failures but if they coincide, if there are terrorist groups' actions. We can construct NHPPs of 100-300 MWe by using these standard modules.

For the major vessel of regional NHPP unit for RI SVBR-75/100 the principal design solutions distinguish from those for traditional type reactors. Small dimensions of monoblock SVBR-75/100 and developed properties of inherent safety of FRs with LBC require the RI protection from only those external effects: aircraft falling, shock waves, maximal computated earthquake (MCE). There is no need for designing the tight shell withstanding significant internal pressure. Small dimensions of protected reactor compartment and simple scheme of RI enable to reduce the terms of NHPP unit construction and significantly reduce the construction cost.

The simplicity of automated control system (ACS) conditioned by using passive systems for cooling down the RI facilitates the construction cost reducing.

As it has been assessed by experts, the economical competitive ability with HPP using fossil fuel will be ensured due to the following facts: almost lack of expenditures on nuclear fuel transportation; long lifetime of the fast reactor core, which ensures without refueling RI operation during about 8 years; low cost of spent nuclear fuel storing; almost lack of liquid radioactive waste and expenditures for its recycling; complete plant fabricating of the RI and possibility of its transportation by car, railway or sea to the NHPP constructing site that reduces the constructing terms and approaches them to those of traditional HPPs and reduces the investment cycle; sharp reduction of expenditures for designing safety ensuring systems due to RIs' high inherent safety; high commercial production due to great demand for these NHPPs; possibility of export delivery of these RIs. The cost of these NHPPs is one-fifth - one-tenth as many as that of large NPPs.



Plane on the point 4.200

ПЛАН НА ОТМЕТКЕ 4,200

Figure 4

4.1.4. Large Power Modular NPP

On the basis of commercially produced modules SVBR-75/100 it is expedient to develop the design of modular NPP of large power (1 GW and more at the same unit). The prospect for that principle of designing NPP is shown in conceptual design developed in the USA (PRISM) [24] and in Japan (4S) [25]. However, use of this principle for LBC cooled reactors is the most effective.

The economical gain is achieved due to: constructing volumes reduction because of eliminating the number of safety systems and localizing systems, reducing the specific material expenditure (including bismuth demands) as compared to traditional reactors of large power, reducing the fabricating cost due to high commercial production, reducing the NPP constructing terms when reactor modules are delivered to the constructing site in high plant readiness. It enables to improve the conditions of credit receiving and repayment and to increase the competitive ability of NPP. The preliminary assessments have revealed that the capital cost of constructing such NPP is expected to be not more than that for constructing the NPP's VVER-1000 unit.

4.1.5. Dual-purpose nuclear desalinating power complex for developing countries

Many developing countries in Africa an Asia suffer from deficiency of fresh water and electric energy. The majority of these countries do not have sufficient own resources of fossil fuel, which can meet their demands. In some countries fuel transporting is difficult, there are no powerful electric power transmission lines. The marketing researches conducted recently by IAEA [26] have revealed that in many cases nuclear power sources of 100 MWe small power can be used economically effective for these purposes.

However, the developing countries' particularities concerned with not sufficiently high level of education, technical culture, social and economical development, as well as possibilities of arising local military conflicts, put forward special requirements to the nuclear power technology which are stricter than those for developed countries.

First of all, these requirements are RI inherent safety against severe accidents that is based on RI inherent properties ensuring safety not only in cases of personnel's errors and multiple failures of technical systems coincidences, but in cases of sabotage terrorist actions, etc. Besides, they must meet strict non-proliferation requirements [21], including that refueling in the country-user must be eliminated and due to this fact the lifetime duration must be 10 years or more. The opportunity for reactor unit transportation to the country-manufacturer in the state of nuclear and radiation safety for refueling and then transporting it to the country-user again must be ensured. Thefts of fuel must be technically eliminated as well. Besides, the competitive ability to the alternative resources of receiving fresh water and electric energy must be ensured.

RI SVBR-75/100 meets these requirements the most completely. It has extremely high safety potential, lifetime duration needed, ensures the regime of non-proliferation due to the following:
- use of uranium with enrichment less than 20%,
- lack of refueling in the country-user,
- opportunity of transporting the reactor module after ending its lifetime to the country-manufacturer in the state of nuclear and radiation safety with LBC "frozen" in the reactor.

4.1.6. Plutonium Utilization and Long-lived Minor Actinides Transmutation

In different countries the policy on managing plutonium, which quantity increases steadily, is different. It is determined by the fact that on the one hand Pu is very valuable fuel for the future NP using FRs. On the other hand it can be used for political and military purposes. Besides, Pu is high radiotoxic material and is considered to be NP dangerous radioactivity waste. Long-lived minor actinides belong to it too.

For the first case the issue of Pu managing (both extracted one and that is contained in spent nuclear fuel) results in long reliably controlled storing.

The second case results in the task of Pu transmuting into the form that reduces the risk of its unauthorized proliferation or its complete burning up and minor actinides nuclear transmutation.

For solving this task the works on mastering the new nuclear technology using accelerator-driven systems are carried out. The main stimulus for developing this technology is Pu and minor actinides blanket subcriticality that eliminates the prompt neutrons runaway nuclear accident.

Along with it, this task can be solved on the basis of already mastered technology. For example, during eight years one reactor module SVBR-75/100 can transmute about 1000 kg of Pu (weapon or reactor one) into the form protected against unauthorized proliferation ("spent fuel standard") at reducing its quality as a weapon material compared to the weapon Pu. In terms of 1 GWe - year 1,25 tons of Pu will be utilized in those reactors. If minor actinides (first of all amerithium) is introduced into fuel, their transmutation into short-lived radioactive wastes will take place.

The safety level needed will be ensured due to developed propertied of RI SVBR-75/100 inherent safety which have been mentioned above.

5. THE DISTANT FUTURE TASK (AFTER THE YEAR 2030) LONG-TERM NP DEVELOPMENT IN CONDITIONS OF LIMITED NATURE URANIUM RESOURCES

The development of power and NP especially is very sluggish. That is why many countries, where there is developing NP, consider conditions under which NP can exist and develop in the middle of the next century. As far as Russia conditions are concerned, this time period would be characterized by the fact that available nature and enriched uranium resources which enable to develop NP and ensure export supplies without large costs for uranium mining and enriching would be exhausted. It would result into the necessity of searching, mining and developing new uranium deposits at hard-reached areas or into the orientation towards scale uranium import that would be the cause of increasing the fuel component of the cost of electric power generated by NPPs with thermal neutrons reactors.

The problem of NP fuel supplying could be solved by FR operating in closed nuclear fuel cycle (NFC) and enabling to involve effectively uranium-238 into power generation.

However, nuclear fuel-power complex development on this base inevitably yields to increasing specific capital costs and duration of investment cycle in comparison with modern and promising HPP using organic fuel. Under the market economy conditions when construction should be performed on the base of credits repayment it can deteriorate the NP competitive ability if consumers have free access to the wholesale electric energy and power market. Now this situation is forming in the USA, where constructed NPPs with light water reactors (LWR) is not putting into operation because of their low profits, even under conditions of the absence of the costs for fuel recycling and fuel elements refabricating. The situation is not going to improve even if weapon plutonium is involved into NFC with not complete fuel cycle closing (the absence of costs for reprocessing, minimal radiation characteristics during fabricating).

At the next century in the course of NP development based on the evolutionary improvement of traditional reactors the situation would be deteriorated because in order to keep up the value of severe accident risk at the today's existing socially acceptable level, the probability of that accident should be reduced inversely proportional to the number of reactor-years generated by the NPP units. It would inevitably yield to increasing the capital costs for increasing the NPP safety.

NP situation may be improved in more distant future when the resources of cheap natural gas have to be exhausted (the share of natural gas is the dominant one in electric energy production), quotas on releasing the greenhouse gases, significant increasing the cost of electric energy produced by HPP, controlling the environment have to be introduced.

There might be the long time interval (50-100 years) between the moment when NP loses its competitive ability after exhausting cheap uranium resources and that one when NP with closed NFC can be assuredly competed with that using organic fuel. Then NP would be less economically effective in comparison with power industry using organic fuel ("gas pause").

The circumstances considered make it actual to search for the concept of NP development during the "gas pause" period, which enables to overcome arising difficulties. One of possible concepts has been designed by SSC RF IPPE [27]. The main concept goal is to increase sharply the efficiency of natural uranium energy potential utilization without radio-chemical fuel reprocessing and fuel cycle closing for "gas pause" elimination.

5.1. Fast Reactor Operation in the Open NFC (Physical Concept)

FRs are commonly dealt with in the closed NFC in which they operate under the breeder regime. And plutonium is mainly accumulated in zones of its reprocessing from depleted uranium. The breeding ratio is less than that one in the core. The core make-up under partial refuelings is performed by fuel with fissile material content as that was in start-up load fuel. For providing core reactor make-up by built-up plutonium, closing the NFC with fuel reprocessing and fuel elements refabricating by exploitation of released plutonium is needed.

However, FR operation may be performed in non-traditional regime, its realization would enable to close the "gas pause". It is the FR with so called "without chemistry processing" fuel cycle that was first theoretically considered by S.M. Feynberg and E.P. Kunegin ("Kurchatov Institute", Russia) in 1958 [28] and then by K.Fucks and H.Hessel (Germany) in 1961 [29].

The fast reactor operation under that regime is not in conformity with the existing view points on the FR role in NP, as in this case built-up plutonium is neither extracted nor reused, but mainly utilized directly inside the reactor. This is the reason for reconsideration of traditional FRs.

In the reactor under consideration the start-up load of core consisting of fuel subassemblies (FSA) with comparatively high enriched uranium fuel (10...12%) is realized only once. (Certainly, there may be used U-Pu fuel containing weapon plutonium if the problem of its utilization is actual or yielded energetic plutonium). In the course of reactor operation the FSA of start-up load under partial refuelings are being replaced gradually by previously loaded make-up FSA where plutonium has been already built up, and their place is occupied by fresh make-up FSA with depleted or slightly-enriched uranium. There are no separate breeding blankets in that reactor.

The principal condition of ensuring reactor operation under that regime is core breeding ratio (CBR) being more than one. It enables the criticality to be maintained due to plutonium built-up. Thus, on the one hand, it is necessary the average plutonium concentration in the core to be more than critical one. On the other hand, this concentration should not exceed the equilibrium value, which is determined by the equality of the rate of forming plutonium out of uranium-238 and that of its burning-up. The reactor parameters and the regime of partial refuelings should be selected in such a way that the reactivity loss after partial refueling should be compensated by the reactivity increase during the operation period between refuelings. This reactivity increase is compensated by introducing the absorbing rods into the core. After partial refueling the reactor must be critical with the absorbing rods extracted.

Carrying out these conditions must meet the number of reactor requirements. The core must be of large dimensions. The volume share of fuel and its uranium density should be high. The share of FSA reloaded per one cycle of partial refuelings must be low (high refueling multiplicity).

The chosen scheme of partial FSA refuelings and reshufflings should also provide smoothing the energy yield distribution in the core. The problem is complicated because simultaneously in the core there are both FSA with large plutonium content that is close to equilibrium one and fresh make-up FSA without plutonium with very low energy yield. That is why zones with high and low plutonium content must be alternated.

After unloading the last FSA of start-up load out of the core the reactor goes into the self-maintaining operation regime. Thus criticality is maintained mainly by own plutonium and reactor uses only low-enriched uranium (LEU) or that from the waste pile (UWP). In this case the efficiency of using energy potential of natural uranium increases several times in comparison with LWR (see below).

5.2. Efficiency of Natural Uranium Energy Potential Utilization

Efficiency of natural uranium energy potential utilization (EUU) is determined as the ratio of the power generated by the reactor over the define time period (fission products mass) to the mass of natural uranium (NU) used during this period to provide the reactor operation.

For LWR this value is about 0.5% that corresponds to the consumption of NU of about 200 tons per one ton of fission products (or per 1 GWe - year).

For considered type of FR EUU increases as there is the number increase of fuel campaigns burnt up in the reactor. It is concerned with the fact that contribution of start-up load, which fabrication requires much NU, into the total reactor energy-generation decreases with increasing the number of fuel campaigns burnt up in the reactor.

If the use of UWP is ensured as make-up fuel, thus EUU would be maximal, achieving under the large number of campaigns the value which is equal to fuel burn up depth (% of h.a.). It corresponds to EUU increasing 10-20 times in comparison with LWR and is explained by the fact that for fabricating make-up fuel there is no need for NU. To ensure such EUU increase the fuel elements must meet the most stringent requirements for campaign duration, depth of fuel burning-up and the value of fast neutrons damaging dose.

If there is the increase of uranium enrichment in the make-up fuel, there would be the decrease of requirements for fuel elements operation conditions (see Table 2). But thus EUU decreases, being still several times as high as this one for the VVER-1000 reactor.

TABLE 2.THE DEPTH OF FUEL BURNING UP, FAST NEUTRONS DAMAGING DOSE, MICROCAMPAIGN AND CAMPAIGN DURATION AS THE FUNCTIONS OF MAKE-UP FUEL ENRICHMENT

X_5	g	D	Òìê	Tê
0	20,4	434	575	41
1	18,3	387	520	37
2	16,2	342	463	33
3	14,2	296	407	29
4	12,0	250	351	25
5	9,8	200	281	20

X_5 - make-up fuel enrichment, %
g - the depth of fuel burning up, % h.à.
D - fast neutrons damaging dose, dpa,
$Ò_{mc}$ - the microcampaign duration, eff. days
T_c - the fuel campaign duration, eff. years

We can see it from Table 3 where the ratio of EUU for the SVBR-600 reactor with equivalent electric power of 625 MWe which has been considered as an example of realizing the FR operating in the open NFC, to EUU for VVER-1000 reactor has been presented. This ratio demonstrates the increase of the functioning time for NP using SVBR-600 reactors in comparison with that using

VVER-1000 ones in the open NFC under the same NPP's total power maintained and NU resources. If the enrichment of make-up fuel is taken to be 4,4%, as it concerns the VVER-1000 reactor, then EUU for the SVBR-600 reactor would be three times of that for the VVER-1000 reactor even in the fourth campaign. As a result, the consumption of natural uranium would decrease three times, i.e. the possible term of existing the open NFC would increase three times.

TABLE 3. THE COMPARATIVE EFFICIENCY OF NATURAL URANIUM ENERGY POTENTIAL UTILIZATION. (THE INCREASE OF OPERATION TIME FOR NP USING SVBR-600 REACTORS IN THE OPEN NFC IN COMPARISON WITH VVER-1000 UNDER THE SAME POWER AND NATURE URANIUM RESOURCES)

X_5 / n	0.2	1	2	3	4	5
1	1.05	1.00	0.96	0.90	0.87	0.84
2	3.15	2.79	2.45	2.19	1.98	1.81
3	5.24	4.33	3.57	3.04	2.65	2.35
4	7.34	5.66	4.44	3.64	3.09	2.69
5	9.44	6.85	5.14	4.11	3.42	2.93
10	19.90	11.15	7.22	5.33	4.23	3.51

X_5 - make-up fuel enrichment, %
n - a number of campaigns
EUU for VVER-1000 is considered to be 0,5%

Taking into account these results, it is expedient to use the flexible regime of reactor operation. In the first campaign the value of uranium enrichment in the make-up fuel is chosen in such a way that the fuel elements operation conditions are to be within the frameworks of the verified values. But the EUU will not be the best.

In the next campaigns, as the experience is being gained, the uranium enrichment in the make-up fuel will be decreasing, and the increase of EUU will be the result of it.

To achieve the maximal possible EUU there should be provided the possibility on exhausting the RI service term to use its fuel load, in which plutonium has been built up in quantities ensuring the reactor criticality, as the start-up load for the new RI.

The significant increase of EUU in comparison with LWR ensures the possibility of long-lived existence of the open NFC under the limited NU resources. It enables to postpone the necessity of introducing large-scale reprocessing the spent nuclear fuel (SNF) over 50 to 100 years, to save much money and eliminate the risk of nuclear, radiation and ecological accidents possible for this period if corresponding enterprises are functioning. By assessments to realize this possibility under the limited cheap uranium resources phasing out the LWR and introducing the FR of the SVBR-600 type into NP might be starting in 2030.

Nevertheless, as in the long-term prospect the SNF reprocessing and plutonium recycling are necessary, the search and development of the most economically available, safe and ecologically pure processes of SNF reprocessing must be carried out within the required scales.

5.3 Managing the Spent Nuclear Fuel

When there is long NP functioning in the open NFC the workload for repositories with burnt up FSA increases. These repositories must be of suitable volumes and provide long safe and controlled storage of the spent FSA. Enhanced safety of burnt up FSA storage can be reliably ensured because there are some barriers (fuel matrix, fuel element cladding) at the reactor for radioactive products release from FSA into the environment, and the additional ones can be designed (e.g., the capsule with burnt up FSA is filled by liquid lead and then "frozen"). If the spent fuel is stored in the "dry" repositories, it is not exposed to any attacks resulting in damage of protective barriers. Comparatively low volume of the spent FSA per 1 GW(e)-year, in comparison with VVER-1000, caused by 3 or 4 times deeper fuel burning up, facilitates the designing reliable physical protection resistant to external attacks and reduces the repository cost.

In addition to it, the reactor operation in the open NFC almost eliminates the possibility of unauthorized plutonium proliferation and utilization of built-up plutonium for the war purposes because it exists in the spent fuel together with high-radioactive fission products ("spent fuel standard").

It will be promoting to the rise of political stability in the world. It should be also highlighted that the risk of unauthorized plutonium prolifiration out of the SNF repositories after accumulating the threshold plutonium quantity making the fabrication of several hundreds of nuclear charges available does not increase proportionally to the total quantity of plutonium in the repository.

Turning over the spent nuclear fuel repositories (SNFR) to the International Guarantees facilitates to reducing the risk of plutonium thefts.

5.4. The Key Results of the Investigations Performed

Results of the investigations carried out have corroborated the feasibility of FR operation with make-up under partial refuelings by slightly enriched or depleted uranium. In this case the EUU highest value is achieved if the core dimensions are: (D x H≅ 4.0 m x 1.4 m), fuel volumetric fraction is not lower than 60%, there is utilization of fuel with high uranium density (~11 g/cm^3) and formation of possibly harder neutron spectrum in the core (there is lack of light nuclei).

The formation of harder neutron spectrum has been influenced by LBC using, thus the higher value of CBR has been provided, which has had a key part in reaching the EUU biggest value. The calculations have demonstrated that for sodium coolant, which can moderate neutrons more effectively, it is hard to provide the reactor criticality if it has been made up by depleted uranium.

In the case of using metal alloyed (10% of Zr) uranium fuel with 75% effective density of theoretical one, the reactor can utilize waste uranium as the make-up. Thus the highest EUU is ensured (about 20%). In this case the burn-up depth achieves about 20% of h.a. (that is justified at experimental assemblies of EBR-2 reactor), fast neutron damaging dose on the fuel element cladding material accounts for approximately 430 dpa (it is twice the value that has been achieved by tests for ferritic and martensitic steels), the total operation period of FEA is about 30 years (that is three times over that gained for RI operation at the NS).

As it has been mentioned above, increasing the make-up fuel enrichment reduces these demands considerably. It should be also pointed out that the concepts of the core complete equipment, scheme and partial refuelings schedule, as they have been accepted in calculations at the phase of preliminary study, are not optimal. They need further multicriterion optimization by using calculational algorithms in which reactor dimensions are reflected equivalently. It enables to reduce the requirements for fuel elements operation. One such method has been presented in paper [30].

On the base of the results obtained the engineering design of SVBR-600 RI of the 625 MWe electric power has been carried out at EDO "Gidropress.", and the design of NPP using that RIs has been carried out at VNIPIET. Their aim was to choose the principal engineering and designing approaches and the preliminary assessment of their engineering and economical characteristics.

The results of calculation of engineering and economical characteristics of the two-block NPP with RI VVER-600 being compared with those of the two-block NPP with RI VVER-640 which has the highest safety characteristics for that type of RI, have demonstrated the following.

Capital costs for constructing the NPP along with the SNFR (the costs of initial fuel and coolant load calculated per 1 kW of electric power maintained have been taken into account) have been almost equal and the cost of electric energy has been 21% less than that of RI VVER-640.

The equality of specific capital costs, in spite of higher costs of FR start-up fuel load and coolant which are typical for RI SVBR-600, has been accounted by simplier scheme of this RI and reduction of constructing volumes, which is caused by the total absence of the primary circuit pipelines and valves, significantly less quantity of ancillary systems and systems ensuring safety, significantly less (4 times) volume of the SNFR under the same power generation, almost total absence of LRW (by the experience of operating the RI using lead-bismuth coolant at the NS) and, as a result of this, significant reduction of special chemical water purification systems which compensates the factors pointed out above. The less electric energy cost for SVBR-600 is accounted for the significantly more depth of fuel burning-up in the FR considered.

As a result of the investigations carried out, there have been gained the conclusions of the economical preference of the NPP using SVBR-600 RI in comparison with that using VVER-640 RI. Of course, these conclusions are preliminary because the level of studying the RI SVBR-600, that corresponds to the phase of engineering recommendations, defines the more error of engineering and economical calculations than that for VVER-640. However, these conclusions point to the expediency of further investigating the concept suggested along with the development of the NPP concept design. And the design of the NPP using VVER-1000 RI (V-392), that has the best engineering and economical characteristics, should be accepted as the comparison basis.

6. CONCLUSION

The analysis of experience of operating the RIs at the NSs and at the ground-based facilities-prototypes has demonstrated that it was only one accident that was caused by using LBC. Further work has enabled to solve the problem of LBC technology and ensure the reliable exploitation of the RIs primary circuits. The causes of other accidents and emergency situations were not concerned with the use of LBC, and similar accidents could have happened at any type of the RIs.

The developed measures on radiation safety have enabled to eliminate the personnel's irradiation by polonium-210 above the permissible limits not only under the normal operation conditions, but also if there are refueling, repair works, emergency coolant spillage.

The problem of multiple coolant “freezing-defreesing” in the RI has been solved. It enables to use this regime for providing nuclear and radiation safety during the RI transportation (for low power RIs) and long SNF storage.

The experience gained has enabled to design the number of small power RIs providing the most complete realization of the self-protection principle for the severest accidents and their deterministic elimination because of using the fast reactor, LBC and the primary circuit integral design (SVBR-75/100). On the basis of these RIs renovation of NPP’s units served their lifetime can be carried out, regional NPHPs for energy shortage regions can be built, large power NPPs of modular type can be constructed, meeting IAEA requirements power complexes for electric energy producing and sea water desalinating can be built in developing and other countries, reactors for utilizing Pu and transmuting the long-lived minor actinides can be designed.

In order to maintain the NP competitive ability on exhausting the cheap uranium resources and necessity of using the FR with closed NFC and plutonium recycling (plutonium significantly deteriorates the ecological characteristics) there has been developed the concept of the FR cooled by LBC and using the depleted uranium as the fuel make-up. It enables to postpone the large-scale SNF reprocessing with plutonium recycling over 50 to 100 years.

The existing scales of metal bismuth production which have been restricted by bismuth usage will limit the rate of introducing the reactors cooled by LBC in NP. That is why after the period of operating the demonstrative reactors using LBC and corroborating their practical reliability, economical efficiency and safety and accepting the decision of their wide introduction into NP, the actions on increasing the bismuth production must be taken. Further, when the bismuth cost begins to deteriorate noticeably the NPP's economical characteristics (as the exploiting mines are being depleted), it will be expedient to start using LBC with lower bismuth content (e.g., as low as 10%), and then pure lead coolant, which is more difficult to exploit and has not been mastered yet.

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LIST OF ABBREVIATIONS

NPP - nuclear power plant;
NPHP - nuclear power heating plant;
RI - reactor installation;
NPI - nuclear power installation;
NS - nuclear submarine;
LMC - liquid metal coolant;
IPPE - Institute of Physics and Power Engineering;

LBC - lead-bismuth coolant;
NP - nuclear power;
SG - steam generator;
SHS - steam heating system;
EP - emergency protection;
CR - compensative rods;
TC - tight compartment;
WCR - water-chemical regime;
EC - emergency condenser;
LRP - leakage reinjection pump;
FR - fast reactor;
MPC - maximal permissible concentration;
FSA - fuel subassemblies;
PSA - probabalistic safety analysis;
SNF - spent nuclear fuel -
EPS - emergency protection systems;
SNPP - small nuclear power plant;
HPP - heating power plant;
MCP - main circulation pump;
SRD - scientific and research developments;
LRW - liquid radioactive waste;
RAW - radioactive waste;
NFC - nuclear fuel cycle;
LWR - light water reactor;
FEA - fuel elements assemblies;
LEU - low-enriched uranium;
CBR - core breeding ratio;
UWP - uranium waste pile;
EUU - efficiency of natural uranium energy potential utilization;
NU - natural uranium;
SNFR - spent nuclear fuel repository;
NVNPP - Novovoronezh nuclear power plant;
RDIPE - Research and Design Institute of Power Engineering;
NITI - Technological Research and Development Institute (Sosnovy Bor);
TSNII KM Prometey - Central Research Institute of Structural Materials;
OKBM - Experimental Design Bureau of Mechanical Engineering;
EDO Gidropress - Gidropress Experimental and Design Bureau (Podolsk);
SSC RF IPPE - State Scientific Center of Russian Federation Institute
of Physics and Power Engineering;
St.P MBM "Malakhit - Malakhit Marine Engineering Design Bureau (St. Petersburg);
GNIPKII Atomenergoproekt - All Russia Research and Design Institute;
VNIPIET - All Russia Research and Design Institute for Integrated
Power Engineering Technology;
VNIPI Promtechnologya - All Russia Scientific Research and Design Institute.