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COMPARATIVE ANALYSIS OF COOLANTS FOR FBR OF FUTURE NUCLEAR POWER

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INTRODUCTION

Selection of fast reactor (FR) coolant of the future Nuclear Power is a complex task that has no a single solution.

On the one hand, it is caused by the fact that to a great extent the coolant applied determines the neutron spectrum in the reactor. Such neutronic and physical reactor parameters as breeding ratio (BR), void reactivity effect (VRE), which largely determining safety, and some others are dependent on the neutron spectrum.

On the other hand, the coolant applied determines the engineering design of the reactor and reactor installation (RI) on the whole, technical and economical parameters of the NPP and NPP operational safety.

Due to existing alternative energy resources, NP is developing in conditions of keen competition to power sources that use fossil fuel. For that reason, the economical parameters of the NPPs with FRs should always provide their competitive ability in case the government is not capable to subsidize the constructing of such NPPs from the budget.

In recent decades the requirements to NPP safety parameters have been heightened. These requirements are expected to grow in future as the summary power and operating time of the NPPs are increased. It is necessary to keep the risk value (statistical expectation) of the severe accident at the current level that can be considered as socially acceptable.

Contrary to this tendency, in recent decades the requirements to the ability of FRs to breed plutonium have been considerably reduced.

When the FR coolant is selected, the degree of its mastering and expected term of the beginning of FR industrial introduction should be taken into account. It is caused by the fact that mastering the new coolant requires large R&D expenses, designing and long testing of the demonstrational reactor. (It is considered that new coolant is not only a new substance or chemical element but also familiar coolant which parameters are used in the new range, e.g. water steam under overcritical parameters, etc.).

The experience of NP development indicates that this period can last 25-30 years. For that reason, it worth developing a new coolant if the forecasted costs of its mastering (which is usually reduced) could be justified by advantages expected of its application (which are usually overestimated).

REQUIREMENTS TO FUTURE LARGE-SCALE NUCLEAR POWER

General Requirements

It is evident that substantiated selection of coolant cannot be done without accounting the requirements to the reactor. These requirements are changing in compliance with changing the missions that arise at different stages of NP development.

Currently NP is in crisis. This crisis is displaying in the following events. Some countries have made a decision to close down the NP, the world's NP increase paces have reduced sharply. There are certain cases, when already constructed new NPP units have not been commissioned.

It is caused by the fact that endeavors to meet the increased requirements to NPP safety with traditional types of reactors results in rising NPP capital and operational costs.

Other reasons are related to the nuclear fuel cycle (NFC) because of increasing the risk of unauthorized plutonium proliferation, caused by necessity of plutonium extraction from spent nuclear fuel (SNF) and lack of solution on handling with the long-lived radioactive waste (RAW).

All listed above issues also concern the traditional sodium-cooled FBR with short plutonium doubling time.

Along with this, NP development without FRs cannot be imagined because only FRs are capable to provide functioning the future large-scale NP for thousand of years without any constraints on fuel resources.

Challenge of new RI generation (IV generation in USA terms [1]) can be formulated as "Design one or more nuclear power systems that can be licensed, constructed, and operated in a manner that will offer a competitively priced supply of electricity while satisfactorily addressing nuclear safety, waste proliferation, and public perception concerns of the countries in which it is deployed."

For these reasons, in recent years some countries launched works on developing the requirements to new generation FRs and designing the concepts of RIs that can meet these requirements.

Investigations into these issues have been carried out in Russia [2, 3], Japan [4] and in the USA [1]. They have resulted in formulating the following requirements.

Economics:

- In the places of NPP siting economical parameters of the NPP with FRs should be competitive with those of the NPP with LWRs and alternative energy sources.
- Acceptable investment risk
- Limited project and construction lead time

Safety:

- The safety level should be not lower than that of new generation LWRs. (Russian concept presumes that this level should be much higher and should be ensured by deterministic elimination of causes of severe accidents).
- Tolerance of human error
- ALARA (as low as reasonably achievable) radiation exposure
- Low likelihood of core damage

- Demonstration of severe core damage impossibility

Fissile materials nonproliferation:

- Minimal attractiveness to potential proliferators
- Intrinsic and extrinsic proliferation resistance

Lack of environmental impact:

- Solutions for all waste streams
- Public acceptance of waste solutions
- Minimal waste

Others.

The particular content of these criteria and their priorities are discussed by the world's experts. Different value parameters are cited and all is summarized as follows:

Economics is regarded as competitiveness with alternative energy sources in conditions of a particular site and this criterion is one of the most important.

Safety is regarded as attractiveness for energy enterprises from the standpoint of ensuring their economical safety in case of any accidental situations at the NPP and as social acceptance for the population who live nearby the site. I.e. there should be reliable technical assurance that no accidental situations that might happen at the NPP can damage the population's health and properties.

Generation IV reactors must demonstrate that severe core damage will not occur for any plausible initiating event. This demonstration must be accomplished through integrated reactor testing.

No credible scenario should exist for release of radioactivity requiring offsite response to ensure public safety.

Generation IV reactors must be highly tolerant to human error. Generation IV reactor designs must afford ALARA radiation exposure over the total system lifetime, including all stages of fuel cycle from nature uranium mining to RAW disposal.

Fissile materials nonproliferation should preserve the current regime, where the misuse of fissile materials is the least attractive route to potential nuclear weapon proliferators.

Reducing the quantity of radioactive waste. Politically and publicly acceptable technologies must exist for all nuclear waste streams and an implemented solution must exist for wastes from previous and existing plants.

Other requirements are the following:

- opportunity to export these reactors to any (first of all, developing) countries;
- capability of reactor's fuel-self-providing in the closed fuel cycle (Russian concept presumes that BR slightly exceeding 1.0 is quite sufficient. In Japanese requirements BR ~ 1.2);
- capability to operate on different kinds of fuel, including various compounds of plutonium and MA. That is the consequence of the task on reducing the quantity of RAW.

To a certain extent coolant's nuclear, physical and chemical properties influence on all listed requirements.

Response for the challenge

Technical measures to meet these requirements can be illustrated by the ANL approach for development of the concept of reactor STAR-LM [5]. These are:

- **Simplification** – An important element of regaining economic competitiveness is to simplify the NPP and its operations. The goal is to reduce the scale and complexity or to eliminate systems to the greatest extent possible to reduce capital costs and to minimize maintenance and staffing needs.
- **Factory Fabrication** – Significant cost savings can be introduced by minimizing the extent of on-site fabrication and construction. Accordingly, the reactor system is to be of modular construction to the greatest extent possible and configured in a pool (monoblock) rather than loop arrangement. Vessels need to be small enough to be transported by various means to the site including overland.
- **Existing Technology of Applied Materials** – The requirement to complete R&D in a 3-5 year period means that materials must be selected based on proven experience in a reactor environment, including the fuel, coolant, and structural materials. The same is true regarding the compatibility of materials.
- **Existing SG Technology** – In STAR-LM the authors limit their consideration by conventional steam parameters (7 MPa) and design approaches that produce steam superheat to improve efficiency while retaining design simplicity are sought. The SG is to be a standard means of decay heat removal during shutdown. The SG is to be modular and designed for ease of replacement.
- **Proliferation Resistance** – The reactor system is to be designed to be exportable to developing countries, and hence proliferation-resistance is required. For example, particular features relevant for the STAR-LM concept include "lifetime" reactor fueling and no on-site fuel access. In this approach there is no need for refueling equipment and spent fuel storage.
- **Ultra-long-life Core** – Consistent with the cost competitive objective, the goal is an ultra-long life core design (15 year with 100 % loading factor). It is required to achieve this within the existing technology as regards fuel burnup and material exposure ranges, nominally 150,000 MWd/T peak and 200 dpa (cladding), respectively. Reactor operation with untight fuel elements must be also considered for a long life core.
- **Passive core shutdown** – should be provided while an off-normal transient condition.
- **Coolant technology**. It is required to provide a system for the purpose of maintaining coolant purity and corrosion protection. As regards purity, it is required to prevent the precipitation and accumulation of impurities that could degrade the long-term performance of the heat transport system or endanger normal coolant flow through the core.
- **Inherent Safety**. Inherent safety is a broadly encompassing term, which implies that safety-in-the-design is implemented based on the inherent behavior of materials and passive safe response of design features to the greatest extent possible. The reactor and coolant systems are to operate at nominally atmospheric pressure; it is required to prevent interruption of the heat transport function in the event of a leak. It is required to provide passive means of removing decay heat, and effective reactor cool-down, in the event that the SG heat sink is unavailable.
- **Containment**. A containment boundary is to be provided. (It is considered unrealistic to design a reactor system without containment no matter what level of inherent safety is achieved). The containment is to be part of the seismic-resistant

zone. The design basis for the containment is to be determined based on the spectrum of internal and external events identified.

VIABLE VARIANTS OF COOLANT FOR ADVANCED FRS

FR viable coolants with completely developed package of technologies are light and heavy liquid metals: sodium, mercury, and lead-bismuth alloy.

Sodium

Sodium is the most widely used as FR coolant due to its extremely high heat conducting properties. This made it possible to obtain the high core power density that in case of BR > 1 provided short plutonium doubling time. It was this property of sodium that determined its selection as FR coolant at the initial stage of NP development. Large experience of operating, equipment repairing, eliminating the consequences of the accidents happened was gained for the sodium coolant.

Mercury

Due to its chemical inertness and liquid state at ambient temperature, mercury is used as coolant for experimental FRs "Clementina" (the USA), BR-2 (USSR). Mercury coolant technology was developed in the 40s when the experimental heat electric plants with binary cycle were constructed. Mercury steam was used as working body in the high-temperature part of the thermodynamical Rankine cycle.

However, due to its high chemical toxicity, high section of fast neutrons capture, comparatively low boiling point (~340 °C) and high cost, mercury was not extended as coolant for the power FRs.

Lead-Bismuth eutectic

Lead-bismuth coolant (LBC) of eutectic composition has been mastered in conditions of many years of operating Russian nuclear submarines' (NS) reactors. The main benefit of LBC is its chemical inertness and high boiling point, which give the inherent safety property to the RI.

Large funds were invested into mastering the technology of LBC usage (not only maintaining the required quality of LBC is implied), all principal issues of its industrial usage were solved; the experience of long operation has been gained (~80 reactor-years). Reactors were designed under IPPE scientific supervision.

On the basis of this experience the project of multipurpose reactor module SVBR-75/100 has been designing [6].

Coolants with Incompletely Developed Package of Technologies

FR viable coolants with incompletely developed package of technologies are lead, water-steam with supercritical parameters, molten salts, and gas coolants.

Lead Coolant

Lead coolant was proposed by RDIPE (Russia) as an alternative to LBC. The following considerations were taken into account:

- much lower level of LC polonium radioactivity as compared with LBC (10^4 times less);
- much lower cost of lead as compared with that of bismuth (~10 times less);
- large scales of world's lead production and its resource base that do not limit development of future large scale NP.

All listed statements are true. However, they are not sufficient reasons for rejecting the LBC as a perspective coolant due to the following:

- Polonium problem for LBC has been studied; solutions to the issue of minimizing personnel's radiation risk have been obtained and developed. Polonium is not hazardous for the population near the NPP. This corroborates the fact that the issue of ensuring the radiation safety was solved successfully. American and Japanese scientists who independently conducted researches into this issue also concluded that polonium activity in LBC was not a barrier to using it in NP [7].
- As estimations have revealed, higher cost of bismuth does not noticeably influence on the value of specific capital costs of NPP construction.
- The situation on bismuth production looks much likely that on uranium production in 1940 when nobody needed it in large quantities.

The very important disadvantage of LC is high boiling point (~330°C against ~125°C of LBC). Necessary experimental base should be constructed; carrying out a great body of long R&D should be done. For example, the issues of maintaining corrosion resistance of steel for fabricating the fuel elements' claddings and maintaining the required quality of LC during operation have not been solved. Besides, there is no representative experimental circulation facility with LC in the primary circuit and steam of overcritical parameters in the secondary circuit. This facility could be used for simulating the running of different emergency regimes and for carrying out verification of codes for calculating the dynamical processes. It has been estimated that the constructing cost of demonstrational reactor BREST-OD-300 is ~2 billion US dollars. Constructing cost of SVBR-75/100 demonstrational reactor equals to ~100 million US dollars.

Gas Coolants

There is no proper experience of applying gas coolant in FRs. It is expected that solving the issue of FR safety in case of the accident with coolant's loss requires a great deal of endeavors due to much higher power density of the FR core and makes this reactor ineffective economically. Gas coolant's high pressure in the primary circuit of this reactor is the cause of lack of inherent safety properties.

Water Steam of Overcritical Parameters

In different countries water-steam of overcritical parameters was more than once considered as a viable FR coolant. High moderating ability of hydrogen nuclei are compensated by low density of coolant under overcritical parameters and low volumetric coolant fraction in the core (tight lattice design of fuel elements) that provides a sufficiently hard neutron spectrum in the reactor.

However, this idea was not developed further than its initial studies. That was caused by difficulty in providing safety in loss of coolant events, difficulties in ensuring reliability under coolant's superhigh pressure, etc.

Molten Salts

Molten salt coolants should not be considered due to the following reasons. Due to higher melting point (over 400°C), the issues of maintaining the operation reliability of molten salt coolants are more difficult than those of lead coolant. Besides, molten salt coolants include rather large quantity of light nuclei of chemical elements (Li, Be, F).

COOLANTS SELECTED FOR FURTHER ANALYSIS

Carried out brief survey of coolants variety has revealed that number of coolants for the further analysis may be reduced. Among all kinds of viable coolants for FRs, LMCs most completely meet the requirements to the advanced reactors and has the considerable database. Among the LMCs sodium coolant and LBC should be emphasized because they have a complete package of technologies for their handling. Sodium coolant, LBC and LC are accepted for further consideration.

COMPARISON OF LMCs PROPERTIES.
ITS INFLUENCE ON THE NPP DESIGN AND ECONOMICAL PARAMETERS

Proposed earlier approaches to the issue of coolant comparison [8] were based on comparison of physical and chemical properties of various coolants. The properties of coolants were considered for the most advanced and developed RI. However, it is evident that coolant's selection to a great extent determines particularities of design of different equipment's elements and the whole NPP. For example, LBC properties enable to design two-circuit FRs, but high chemical activity of sodium requires three-circuit design.

Influence of coolant's selection on the NPP design features can be accounted. All the way from NPP designing to decommissioning, including the experience of NPP constructing and operating, revealing the design errors and their correction should be passed. It is evident that this approach cannot be fully realized on practice. Only detailed designing of NPP alternative concepts makes it possible to estimate the influence of coolant properties on the NPP parameters. It can be performed only for an NPP, which is developed for specific objectives and under specific economical of a certain country.

The situation established in Russia makes it possible to realize this approach in practice. Russia has gained a great experience of designing, operating and decommissioning the RIs with sodium coolant and LBC. There are NPPs with sodium coolant (BN-350, BN-600). Recently the technical project of the NPP with sodium reactor BN-800 [9], conceptual projects of LBC cooled NPP SVBR-75/100-1200 [10, 11], the concept of LC cooled reactor BREST-1200 and its demonstrational prototype BREST-OD-300 [12] were designed. NPP design and its basic parameters were determined for all these reactors. Therefore, an opportunity is given to carry out comparison on coolant parameters that cannot be estimated without complete design of the RI and NPP.

All parameters are summarized in comparative Tables. The certain coolants' parameters, which are the results of corresponding RI and NPP design particularities, are compared by the parameters of particular projects of the NPP:

- Sodium coolant – BN-600 and BN-800 (IPPE)
- LBC – SVBR-75/100-1200 (IPPE)
- LC – BREST-1200 and
  – BREST-OD-300 (RDIPE)

Maintenance of Coolant's Liquid State.
Opportunity of Coolant's "Freezing-Unfreezing"

Under solidifying temperature sodium possesses the property of high plasticity. This makes it possible to avoid lots of troubles related to sodium cooling. Nevertheless, the technology of sodium "freezing-unfreezing" has not been developed and used. High chemical activity of sodium would not make it possible to benefit from using this technology.

LBC is chemically inert and has the lowest coefficient of volumetric expansion among considered coolants (see Table 1). Package of these properties makes it possible not only to exclude RI damage caused by accidental RI "freezing" but also to use the regimes of coolant's "freezing" for heightening safety of RI SVBR-75/100 under long conservation and its transportation in assembly together with the core. This technology makes it possible to eliminate all on-site refueling operations, simplifies RI operation in developing countries and minimizes the amount of radioactive waste at the NPP site.

Approaches used for substantiating the regimes of LBC “freezing-unfreezing” may be used for LC. However, it will be much more difficult to realize those regimes for LC due to its higher coefficient of volumetric expansion and melting point (see Table 1).

Basic Physical Properties of LMCs — Table 1

Properties	Na	Pb	Bi	45%Pb-55%Bi
Density in solid state at melting temperature, kg/m^3	950	11 050	9 800	10 700
Density in liquid state at melting temperature, kg/m^3	930	10 680	10 110	10 560
Specific volume changing caused by melting, %	+2.7	+3.6	– 3.2	+ 1.5
Melting temperature, ^{0}Ñ	98	327	271	125
Melting heat, kJ/kg MJ/m^3	113 107	23 244	53 500	21 223
Boiling temperature (at atmospheric pressure), ^{0}Ñ	887	1745	1552	1670
Interaction with air in liquid state	Ignites spontaneously at T>200^{0}Ñ	very slow oxidation, protective film is forming under operation temperature conditions		
Interaction with water and water steam	very reactive, with H_2 generation	no interaction		

Performance Characteristics of LMCs — Table 2

Properties	Na (BN-600)	Pb-Bi (SVBR-75/100)	Pb (BREST-ÎÄ-300)
Operation temperature, ^{0}Ñ — at the core inlet —at the core outlet	 300…380 500…550	 250…330 430…480	 > 420 < 520
Average operation temperature, °Ñ	450	380	470
Density at operation temperature, kg/m^3	845	10 210	10 500
Thermal capacity: mass, kJ/kg K volumetric, MJ/m^3 K	1.27 1.07	0.146 1.49	0.147 1.54
Kinematics viscosity, m^2/s	3.07×10^{-7}	1.62×10^{-7}	1.85×10^{-7}

Thermal conductivity, W/(m·K)	69	13.5	15.4
Value of heat transfer coefficient on the surface of fuel elements, kW/m^2·K	60…100	25…35	25…35
Specific density changing, kg/(m^3·K)	0.24	1.375	1.2
Coefficient of volumetric expansion, 1/K	2.71×10^{-4}	1.20×10^{-4}	1.14×10^{-4}
Permissible LMC velocity under conditions of core elements vibration-survival ($\rho \cdot w^2$=const), m/s·	10	2.9	2.8

Corrosion Resistance of Structural Materials

Table 3

♦ Properties	*Na* *BN-800*	Pb-Bi SVBR-75/100	*Pb* *BREST-OD-300*
Permissible cladding temperature in the "hot spot ", ^{0}Ñ long-term short term	 720 ~850	 600 650 (up to 5000 h)	 650* no data available
Operation temperature, ^{0}Ñ — at the core inlet — at the core outlet	 300…380 500…550	 250…330 430…480	 > 420 < 520
Coolant heating in the core, ^{0}Ñ — regular mode — permissible under natural convection	 160…200 up to 300	 130…180 up to 300	 < 100 < 180
Corrosion mechanism	steel components solubility in LMC	steel components solubility in LMC caused by diffusion through the protective oxide coating	intensive cladding oxidation or steel components solubility at T > 600^{0}C (depends on oxygen concentration in LMC)

* - The value, has been declared in the BREST reactor designs, is not substantiated experimentally

Technology of LBC Quality Maintenance

Table 4

Properties	*Na* *BN-800*	Pb-Bi SVBR-75/100F	*Pb* *BREST-OD-300*
Method of maintaining the coolant quality	Cold trap for keeping the given impurity concentration in LMC	Hydrogen regeneration for circuit purifying, dosed oxygen feeding for corrosion prevention	Filters, maintaining the oxygen concentration in the narrow operation range due to hydrogen regeneration and dosed oxygen feeding
Operation mode of equipment for maintaining the LMC quality	Permanently	Periodically	Permanently
Arrangement of equipment for LMC quality maintenance	Outside the reactor vessel, pipelines with primary circuit coolant are necessary	Inside the rector vessel	Inside the rector vessel
Coolant technology effectiveness	Proved by long-term operation	Proved by long-term operation	Unavailable, under development

Liquid State Maintenance

Table 5

Properties	*Na* *BN-800*	Pb-Bi SVBR-75/100F	*Pb* *BREST-OD-300*
Regimes requires external heating to keep the LMC in liquid state when decay heat is low	1) core refueling 2) repairing (replacement) of the reactor equipment 3) scheduled shut down		
Temperature level necessary for maintaining the liquid state in compliance with a heating method, ^{0}Ñ	200 (gas or electric heating)	170 (steam heating) 200 (electric heating)	420 (electric heating)
Systems for maintaining LMC in liquid state	1) gas heating 2) electric heating	1) steam heating that provides high uniformity when heating and accuracy when maintaining LMC in liquid state; 2) electric heating	electric heating

Opportunity of Coolant's "Freezing-Unfreezing"

Table 6

Parameter	*Na* *BN-800*	Pb-Bi SVBR-75/100F	*Pb* *BREST-OD-300*
Motives for carrying out coolant "freezing"	No motives	1) changing over the reactor into the safe state for temporary of long conservation and transportation as an assembly; 2) localization of severe accident consequences	1) emergency "freezing" caused by heating system failure; 2) changing over the reactor into the safe state for temporary of long conservation; 3) localization of accident consequences
Technical feasibility of safe coolant's "unfreezing"	—	multiple "freezing-unfreezing" of the reactor and related equipment has been mastered	1) lack of experience; 2) volumetric expansion of Pb caused by phase changing exceeds that of Pb-Bi more than twice; 3) predetermined uniformity of heating-up should be provided under the higher temperature of phase changing

Neutron and Physical Characteristics of Cores cooled by Different LMC

Influence of LMC properties on neutronic and physical characteristics of cores is cited in Table 8. The most important are safety characteristics such as void reactivity effect, and reactivity margin to compensate the fuel burn up, capabilities for plutonium breeding, capabilities for minor actinides transmutation.

Currently the factor of plutonium doubling time is not so significant.

Void Reactivity Effect

Void reactivity effect (VRE) determines a reactor hazard caused by coolant boiling or accidents related to coolant's loss.

It is well known that for sodium cooled FRs the basic issues of providing safety (except those caused chemical activity of sodium) are related to positive VRE, which is caused by relatively large sodium cross-sections of fast neutrons absorption and moderating. It should be highlighted that the value of positive VRE directly depends on the sodium quantity, i.e. on the core dimensions. In this case, an intention to achieve zero value of VRE contradicts to achieving CBR ~ 1. An intention to improve one of safety characteristics inevitably causes deterioration of the other and that is fully determined by sodium coolant properties.

Better neutron and physical properties of lead and bismuth make it possible to achieve negative VRE for LBC and LC cooled FRs easily.

Capability of Minor Actinides Transmutation

Capability of minor actinides transmutation relates to both "own" MA generated during the FR lifetime and MA of LWRs' spent fuel. The main obstacle in this way is nuclear safety providing that is caused by diminished delayed neutrons fraction in the reactor with MA and positive VRE increasing.

From this standpoint, LBC cooled RIs have optimal characteristics. LBC allows designing the core with CBR ~ 1 and with negative VRE for any fuel type. It makes possible to incinerate effectively not only "own" MA but also MA of LWRs' spent fuel [11] under acceptable safety characteristics.

For LC CBR ~ 1 is achieved only for nitride mixed U-Pu fuel of equilibrium concentration. It is impossible to add MA of LWRs' SNF in the BREST type reactors for transmuting.

It is expected that sodium, which properties do not allow to achieve CBR ~ 1 and negative VRE simultaneously, would not make it possible to design the safe core for MA transmutation.

Plutonium Doubling Time

No doubt that thermo-physical properties of sodium make it possible to achieve the shortest plutonium doubling time (see Table 7).

LBC makes it possible to design the core with CBR ~ 1.05 for MOX fuel, maximal lifetime duration and high safety characteristics. Another feature of LBC cooled FRs is a capability of using any kind of fuel without worsening safety characteristics (first of all, by providing negative VRE). E.g., when nitride fuel and axial blanket inside the fuel elements are used in the SVBR-75/100, it is possible to achieve BR~ 1.14 and reduce plutonium doubling time up to ~70 years. For that reason, rather high rate of developing NP are possible with FR using only "own" fuel (plutonium doubling time corresponds to time of installed nuclear capacity doubling).

LC capabilities are much more limited. BREST type reactors are designed for only fuel self-providing without uranium blankets. To increase power capacities of these type reactors requires supply with external plutonium of the specified isotopic composition.

Doubling Time of Plutonium Table 7

Parameter	BN-600	SVBR-75/100	BREST-OD-300
Breeding ratio	1.3 (ÌÎÕ)*	1.05 (ÌÎÕ) 1.14 (UN-PuN)**	1.03 (UN-PuN)
Load on kilogram of plutonium loaded, kW/kg	1100	250	140
Plutonium doubling time in the reactor, years ***	~7 (ÌÎÕ)	~200 (ÌÎÕ) ~70 (UN-PuN)	~ 600 (UN-PuN)
Possibility to start FR operation using uranium fuel	Present	Present	Absent

* - due to the side and axial blankets
** - with the axial blanket
*** - without accounting the fuel cycle (~3 years).

Table 8

Neutron and Physical Characteristics of Cores cooled by Different LMC

Parameter	Na BN-600	Pb-Bi SVBR-75/100	Pb BREST-OD-300
Prompt neutrons lifetime, s	$3{\cdot}10^{-7}$	$2{\cdot}10^{-7}$	$1{\cdot}10^{-5}$
Coolant fraction in the core, %	~ 35	~ 30	55...65
Core power density, MW /m^3	~ 490	~ 135	~ 155
Fuel compositions compatible with LMC	Oxides, nitrides, metallic fuel	Oxides, nitrides	Oxides, nitrides
Local void reactivity effect	Positive $> \beta_{eff}$	Positive $< \beta_{eff}$	Positive $> \beta_{eff}$
Core void reactivity effect	+ 0 (BN-800)	Negative	Negative
Possibility of prompt neutron runaway in beyond-design accidents of the LOCA type	Present	Absent	Absent
Possibility of incinerating own MA	Limited	Effective	Effective
Possibility of incinerating LWRs' MA	Limited	Effective	Absent

Thermal-Hydraulic Performance of Cores cooled by Different LMC

Comparison of influence of LMC thermal-physical properties on reactor parameters (see Table 9) was carried out under the following conditions:

1. The cores cooled by three different LMC have identical power, height, inner structure (diameter and pitch of fuel elements).

2. 180 °C as a mastered value of coolant heating in the core was adopted for LBC and sodium coolant. For LC, which has higher melting temperature and the same limit of maximal temperature as that for LBC, mean heating does not exceed 120 °C.

3. Mean velocities of coolants were chosen proceeding from the assumption of identical hydro-dynamical LMC influence on circuit's elements (vibration, hydro-dynamical pressure, erosion). This influence is determined by flux kinetic energy proportional to ρW^2. Values of velocities cited in Table 9 correspond to ρW^2 =50 kPa.

The basic parameters of the cores cited as ratios to those values for sodium coolant were determined proceeding from the following expressions:

Table 9

Thermal-Hydraulic Performance of Cores for Different LMC Used
(Power and Core Structure are Identical)

Parameter	Na	Pb-Bi	Pb
Reactor power, relative	1	1	1
Velocity, m/s	7.21	2.21	2.17
Heating, ΔT, ^{0}Ñ	180	180	120
LMC flow-rate, relative	1	0.73	1.05
Free cross-section of the core, relative	1	2.5	3.7
Core diameter, relative	1	1.6	1.9
Core hydraulic resistance, relative	1	1.09	1.15
Pumping loss	1	0.79	1.21
Level of natural circulation (at h_{NC}=3 m), % of rated power Where possible heating in the core, °C	1 300	0.9 300	0.3 150
Mean heat flux	1	0.40	0.27
Heat transfer coefficient, relative	1	0.28	0.30
Temperature drop on the boundary layer (between the fuel elements cladding and the coolant flow core)	1	1.42	~1.0

Heat Removal from the Core
due to Natural Circulation Heat Transport Capability

The level of NC is much determined by the reactor design. Selection of coolant also influences it through limitation of coolant's heating and the number of RI circuits (see Table 10).

Capability for Passive Heat Removal from the Core

Table 11 summarizes capabilities of RIs cooled by considered LMCs for passive residual heat removal in the absence of the regular heat removal through SGs (or intermediate circuit's heat exchangers).

The analysis of the data presented has revealed that the FR cooled by LBC satisfies the most stringent safety requirements in case of happening the accidents of the LOHS type and LOHS-WS type (without scram).

Passive cooling systems of reactors of BN and BREST types cannot operate permanently due to the risk of coolant's "freezing" under RI normal operation. Hence, ACTIVE valves for switching off the system at normal operation should be designed.

LBC properties make it possible to design this system not only without using means of active circulation but operating constantly, i.e. really the PASSIVE one.

Issues of Providing Radiation Safety of FRs with Different LMCs

Under normal operation of all FRs cooled by LMC, their radiation impact on the environment is approximately identical (see Table 12).

Natural Circulation (over All Circuits) — Table 10

Parameter	Na (BN-600)	Pb -Bi (SVBR-75)	Pb (BREST-300)
Coolant heating in the core: regular permissible for a short time	 200 300	 160 300	 120 180
LMC density change, kg/m3: under regular heating under permissible heating	 48 71	 220 412	 145 216
Natural circulation pressure head, kPa regular permissible	hNC=3m 1.4 2.1	hNC=3m 6.5 12.1	hNC=3m 4.0 6.4
Power level with natural primary circulation, % to rated power: regular mode permissible mode	4 10	11 > 25	3* 5*

— due to BREST-OD-300 features

Passive Heat Removal — Table 11

Parameter	Na BN-600	Pb-Bi SVBR-75/100	Pb BREST-OD-300
Permissible media for passive heat removal from the primary circuit	Na	air water	Air
Capability for passive heat removal through the reactor vessel	Absent due to self-developing the coolant's leak	can be realized, heat removal to the water pool under self-localization of the vessel's leak	limited, due to low effectiveness of air cooling (only air can be used for cooling)
Capability for passive heat removal through the secondary circuit system	can be realized, through the air coolers (Na-air) integrated into the secondary circuit	can be realized, through the emergency condensers connected to the separators by using the reverse valves	absent, due to impossibility to organize natural circulation in the secondary circuit (water/steam) and ensure coolant "nonfreezing"
PASSIVE heat removal system operating constantly	absent, valves in the air line are necessary for elimination of coolant "free-zing" in normal operation	present, constantly operating system of heat removal through the reactor vessel	absent, valves in the air pipeline are necessary for elimination of coolant "freezing" under normal operation

Impact of Different LMCs on NPP Radiation Safety Table 12

Parameter	Na BN-600	Pb-Bi SVBR-75/100	Pb BREST-OD-300
Coolant radioactivity	high	high	low
Features	sodium–24	polonium-210	—
Primary circuit pipelines beyond the reactor vessel	present, for the cold traps	—	—
Radiation safety in normal operation	high	high	high

The Analysis of Accidents' Development for FRs Cooled by Different Coolants

The data on the analysis of accidents' development and resistance of various LMC cooled FRs to the external effects are summarized in Table 13-Table 19.

The following possible accidents are considered:

- NPP blackout;
- SG leak;
- LMC cooling with its "freezing".

RI's resistance to earthquakes and personnel's malevolent actions and terrorists' acts is considered as well.

For the accidents considered to be within the design basis, which include total long blackout of the NPP, LBC parameters make it possible to maintain the reactor's operation ability that cannot be maintained for sodium and LC cooled reactors.

NPP Blackout

Table 13

Parameter	Na BN-600	Pb-Bi SVBR-75/100	Pb BREST-OD-300
Coolant temperature at the core outlet at the first minutes after NPP blackout with core shutdown (temperature runaway), ^{0}Ñ	700…880 Na boiling is possible	to 550	800…1000 due to low temperature reverse feedbacks
Maximal temperature of the fuel elements cladding at the first minutes after the accident, ^{0}Ñ	700…880	< 600	800…1000
Maximal temperature in the core under long term blackout, ^{0}Ñ / time moment, hours	700…800 / 20	1) 600 / 13, heat sink through the reactor vessel 2) < 400, temperature is permanently decreasing after initial runaway with heat sink through the autonomous condenser-separator system	700…800 / 20
Reactor status after long-term blackout	safety operation margins are exceeded	complete serviceability is retained	safety operation margins are exceeded

Steam Generator Leak

Table 14

Condition	Na BN-600	Pb-Bi SVBR-75/100	Pb BREST-OD-300
Steam generator micro-leak (up to 1 kg/h)	stoppage of the emergency loop of the intermediate circuit	long operation is possible	reactor should be shutdown for repair due to possible deteriorating the coolant quality
Single SG tube rupture	active interaction of Na with water, neighbor SG tube rupture occurs, leak increase is possible, SG section replacement is needed, immediate shutdown of the emergency loop is required	within design basis accident, immediate reactor shutdown is not needed, neighbor SG tube rupture does not occur, SG can be repaired by plugging the accidental tube	within design basis accident, immediate reactor shutdown is not needed, neighbor SG tube rupture does not occur, SG can be repaired by plugging the accidental tube
Multiple rupture of SG tubes	considerable release of hydrogen out of the secondary circuit, overheating of core elements does not occur	design measures for large leak localization are provided, radioactivity release does not exceed permissible limits, overheating of core elements does not occur	design measures for large leak localization are provided, radioactivity release does not exceed permissible limits, coolant “freezing” in the emergency section of the SG is possible, core elements overheating is caused by SG flow rate blockage

RI Coolant "Freezing" Table 15

Parameter	Na BN-600	Pb-Bi SVBR-75/100	Pb BREST-OD-300
Feasibility of coolant "freezing"	Is impossible for coolant in the reactor vessel, is possible in external pipelines	Is only possible after long (about a year) time delay	Is possible in case of within design-basis accidents (SG leak, heating system failure caused by refueling) and in case of long shutdown for equipment repair
Consequences of coolant "freezing"	Absent	Absent	The case of reactor total incapability to operate is possible
Feasibility of coolant "unfreezing "	Multiple "freezing-unfreezing" of pipelines is technically feasible	Multiple "freezing-unfreezing" of the reactor is technically feasible	There are problems in technical feasibility of coolant "unfreezing"

RI Seismic Resistance Table 16

Parameter	Na (BN-600)	Pb-Bi SVBR-75/100	Pb BREST-OD-300
FR feature	Coolant density is less than that of water, mass and dimension parameters are close to those of LWRs	Coolant density is ~12 times higher than that of water or sodium	
Seismic resistance providing	The problem has been studied well, corresponding technical solutions have been found and are realized	The problem is solved due to small dimensions and high stiffness of the reactor vessel (reactor mass: ~500 t, that includes LBC mass: < 200 t)	Solution to the problem is not found and complicated due to large mass of coolant, large dimensions and low stiffness of the reactor vessel structure

Resistance to Malevolent Actions and Terrorists' Acts

Table 17

Parameter	Na (BN-600)	Pb-Bi (SVBR-75/100)	Pb (BREST-OD-300)
Reactor elements which are the most sensitive to malevolent actions	Secondary circuit pipelines, passive heat removal system, CR drivers	*CR drivers*	Passive heat removal system, CR drivers
Possibility of forced melting (destruction) of the core	Present	Absent	Present
Radiation consequences	If the fire with the primary circuit sodium occurs, considerable radioactive contamination of the territory is possible	Population evacuation is not necessary, lack of radioactive contamination beyond the reactor compartment	Radioactive contamination beyond the reactor building can occur in case of the accident with leak of cooling heat exchangers immersed into the primary circuit and damage of air coolers

Coolant's Cost Contribution to the NPP Cost

TABLE 18

Parameter	Na	*Pb-Bi*	Pb
Major capital cost items related to LMC	▪ coolant ▪ system of initial heating the reactor ▪ coolant's filling system ▪ heating system ▪ coolant purification system ▪ cover gas system ▪ system of washing the reactor elements from coolant ▪ sodium fire extinguishing system	▪ coolant ▪ coolant's filling system ▪ heating system ▪ system of maintaining the coolant and cover gas quality	▪ coolant ▪ system of initial heating the reactor ▪ coolant's filling system ▪ heating system ▪ system of maintaining the coolant and cover gas quality
Coolant cost, % of the NPP cost	~ 1 (BN-800)	< 1 (SVBR-75)	~ 1.5 (BREST-300)
Capital costs of the maintenance systems, % of the NPP cost	~ 6	~ 1.5	~ 3..6 (taking into account the coolant filling expanses)

Coolant's Influence on the NPP Economical Parameters

Table 19

Parameter	Na BN-600	Pb-Bi SVBR-75/100	Pb BREST-OD-300
FR features	Three-circuit scheme of the reactor Large body of commissioning works on the NPP site (on-site assemblage of the reactor vessel) special system for fire extinguishing special system for localizing the SG leak (when sodium interacts with air) system of washing the spent cassettes from coolant	Completely factory-fabricated reactors are delivered (serial production, unification) Simplification of safety systems due to developed properties of reactor's inherent safety and LBC chemical inertness Opportunity to show at the demonstrational RI resistance to accidental situations and tolerance to personnel's mistakes	Large body of R&D for developing and mastering the coolant technology Very large body of commissioning and start-adjusting works on the NPP site (on-site assemblage of the reactor vessel, filling it with coolant) Complicated infrastructure (for maintaining the operation ability of the turbine with supercritical steam parameters)
Impact of FR features on the economical parameters	Long term of construction Deterioration of economical parameters as compared with LWRs	Reduction of the construction timeframes Reduction of the investment risk Improvement of economical parameters as compared with LWRs	Long term of construction Great uncertainty of economical parameters Great investment risk

CONCLUSIONS

- The requirements to FRs are reconsidered. Gradual transition from the FR as builder up of plutonium to the FR as economically effective energy source with fuel self-providing ability (BR~1), which safety level is higher than that of the LWR, is taking place.
- Among all types of coolants viable for FRs, LMCs cover the most complete range of requirements to advanced reactors and have a complete database.
- Sodium and LBC are selected among the LMCs because there is complete package of technologies for their handling.
- HLMCs, being at a disadvantage of heat transfer rate in relation to sodium, make it possible to give the inherent safety properties to the reactor and, as a result, to simplify essentially the reactor design and its safety systems. This results in capital and operation costs reduction.
- Neutronic characteristics of HLMC cooled reactors make it possible to transmute the own MA safely, and LBC cooled reactors are able to transmute LWRs' MA, providing CBR>1 and high safety characteristics.
- Basing on the comparison carried out, it can be concluded, that both LBC and sodium are perspective coolants for future FRs.
- LC properties (higher melting point) significantly complicate the FR design and operation and cause certain problems, which solutions have not been found yet.
- Single-valued selection of coolant for advanced FRs requires deeper investigations into the issue of NPPs with FRs cooled by sodium and LBC. Comparison of their technical and economical characteristics for each particular country is necessary as well.

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LIST OF ACRONYMS

BR – breeding ratio
CBR – core breeding ratio
FR – fast reactor
HLMC - heavy liquid-metal coolant
LBC - lead-bismuth coolant
LC – lead coolant
LMC – liquid metal coolant
LWR – light water reactor
MA – minor actinides
NC – natural circulation
NFC - nuclear fuel cycle
RDIPE - Research and Design Institute for Power Engineering, Russia
NP - nuclear power
NPP - nuclear power plant
NS - nuclear submarine
R&D – research and development
RAW – radioactive waste
RI - reactor installation
SG - steam generator
SNF – spent nuclear fuel
VRE – void reactivity effect