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THE PROBLEMS OF THERMOHYDRAULICS OF PROSPECTIVE FAST REACTOR COMCEPTS

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Abstract

In this report the main requirements to fast reactors in system of future multicomponent Nuclear Power with closed U-Pu fuel cycle are regarded. The peculiarities of different liquid-metal (sodium and lead-alloyed) coolants as well as the thermohydraulics problems of prospective FR concepts are discussed.

Presently the rate of development of the Nuclear Power is significantly lower comparatively with that one during 60 – 80th years of the century. Assured resources of cheap uranium, saturated market of the light-water reactors, fierce competition from the side of Gas-Oil Power in electricity production, "cold" attitude of public as the result of nuclear accidents are practically stipulated an absence of interest to construction of new nuclear power plants, despite of permanent advancement of the operated plant's safety, considerable improvement of analytical and numerical modelling and hardware support, appearance of great deal of the new concept projects of NPPs, being prospective from a view-point of safety and economy.

The fast sodium-cooled reactors LMFRs are in the most squeeze situation now. High specific construction costs of LMFR NPP. sodium chemical aggressive reactivity, expensiveness of the U-Pu closed fuel cycle, while plenty of cheap uranium is available, are the main reasons of closing the LMFR programs in USA, Great Britain, German and now in France. However, an interest to fast reactors (FR) does not fall down, because on the base of these very reactors it is possible to develop the future Nuclear Power, which is able to provide enhanced fuel breeding, usage of maximum fissile actinides. minimization of radioactive wastes and decrease of the scale of disturbance of current gas, thermal and radioactive planet balance in the next century.

The LMFR experience shows that they can be reliable and safety operated, provided high technical culture. However, the LMFBR ideology was formed in period of solving the task of minimization of the fuel doubling time. In the result the Fast Breeder Reactors (LMFBR) with tight - lattice ducted Fuel Assemblies have been created. This causes a number of LMFBR's lacks and disadvantages, which lower the operation reliability and reactor safety potential. Some of them are:

- High volumetric share of structure materials in core (~ 20%);
- Significant shift of radial power profile in course of fuel burning, because of MOX-fuel Core Breeding Ratio is below 1;
- Considerable excess of the core reactivity for burn-up (~ 5$);
- High value of the FA hydraulic resistance, and hence, low capability of the core to develop natural coolant circulation under accident condition;
- Insertion of the positive reactivity in the result of blockage of FA flow area and consequent boiling of sodium coolant;
- Significant temperature axial and azimuthal gradients in structural elements under condition of large fast-neutron fluence . resulting in considerable deformation of FE, fuel rods bending, local declination of the flow area and consequent overheating of the fuel rod claddings;
- Destructive impact of vibration and corrosion-erosion process onto core structural

elements under condition of high-speed coolant flow (7-10 m/s) and high temperatures of fuel rod claddings (up to 650 – 700 °C).

Without doubts, future FRs should base on the experience of the current LMFBR reactors. But, they will have to meet the requirements of the future Nuclear Power, in front of which there will be a number of its own problems as well as the troubles, derived from present Nuclear Power. So, conceptual projects of prospective FR should now oversee a few steps from the current situation on the "chess board" of Nuclear Power. These conceptual projects should answer the principle questions of work with equilibrium quantity of radionuclides, providing the safety, positive neutron balance of the Nuclear Power for enhanced fuel breeding, minimization of rate of radioactive waste accumulation and non-proliferation. The answering this questions requires carrying out experimental-theoretical works in justification of new kinds of fuel compositions, coolants and structure materials, development new approaches to physical and mathematical modelling.

In structure of future large-scale Nuclear Power should be multicomponent [1]. This NP should solve the task of minimization of all actinides in fuel cycle and minimization of their final disposal. In such NP the Power Fast Reactors (PFR) will carry out a function of enhanced breeding of fuel, for the Power Thermal Reactors (PTR) and enlarging of NP scale. Another important role of FR-closing the fuel cycle for all dangerous actinides.

For minimization of equilibrium quantity of Pu in NP closed fuel cycle this actinide should be effectively applied in PTR. These reactors should also widen a sphere of nuclear energy utilization.

For utilization (fission) of minor actinides (Cm, Am, Np) and burning dangerous long-lived fission products (^{129}I, ^{99}Tc, ^{90}Sr, ^{137}Cs, ^{135}Cs), some share of burner-reactors must be essential. In the concept of future NP [1] function of reactor-burner will carry out high-flux molten-salt reactor (MSR).

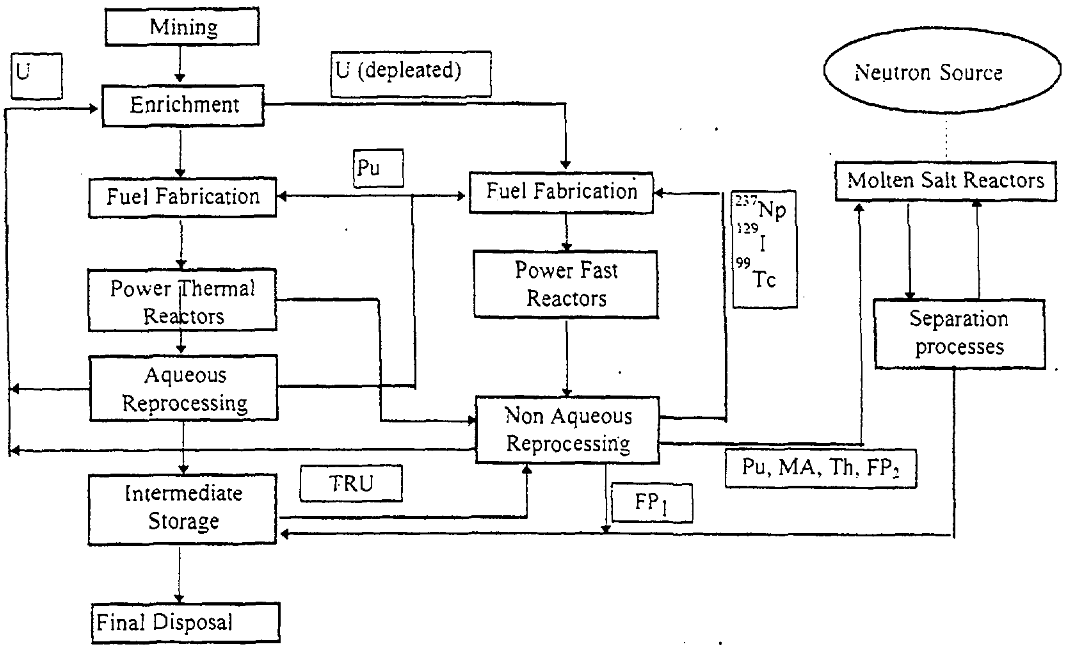

FIG. 1. Multi-component nuclear energy system with closed fuel cycle for all actinides and dangerous long lived fission products. FP_1, FP_2 — short lived and long lived FP respectively. Designations: FP — Fission Products; MA — minor actinides (Cm, Am, Np): TRU — transuranic actinides.

Anyway, whatever future NP to be, fast reactors will be one of more important its component. In carrying out above-mentioned functions in structure of Nuclear Power, fast reactors should meet a number of requirements, imposed by NP:

(1) Specific mass power rate (power rate per unit of mass) of fuel should be as great as possible. This requirement is derived from intention to minimize the equilibrium quantity of Pu and other actinides through minimization of lifetime of them in NP closed fuel cycle. In turn, this requires to decrease irradiation time of fuel in reactor while increasing its burn-up.

(2) Core design should guarantee, that excess of core reactivity for burnup is less than 1$. Such quality can principally enhance a safety potential of reactor.

(3) Power Coefficient of Reactivity (PCR) and Temperature Coefficient of Reactivity should be negative with a good margin to provide reliable feedback compensation of power and temperature rise in normal operation and accidents. Requirement of negative Void Effect of Reactivity (VER) should not prevail over other important reactor characteristics as breeding gain and minimum excess of reactivity for burnup. Instead, the general requirement to NPP with FR, limiting the risk of prompt and delayed fatalities in the result of accident on such NPP by the value 0.1% of the risk, which is characteristic for given place.

(4) Breeding ratio of FRs should amount 1.3-1.5 to supply thermal and burner reactors by Pu (Th). So, it causes a necessity to apply both axial and radial blankets in reactor. Another trouble, neutron fluence in reactor vessel and structures, require to make tight and thick blankets to mitigate problems with NPP decommissioning.

(5) Coolant of reactor contour should not accumulate a large quantity of dangerous long-life radioactive products.

(6) Core design should assure a 'good' thermohydraulics, providing reliable heat removal from core fuel elements in normal and accidental regimes, high margin to boiling, minimum non-uniformity of coolant heating and minimum temperature gradients in structures of Fuel Assemblies in core and blankets.

(7) Core design together with coolant chemical properties and technology should not allow appear dangerous sediments of impurities (corrosion and erosion products, debris etc.), which could negatively affect to capability of heat transfer in reactor core, heat exchangers etc., and lead to local concentration of high-radioactive products upon streamlined surfaces.

Let's do now a brief comparative analysis of compositions on the basis of lead and sodium as coolants for the fast reactors. The application of sodium engineering for NPP with fast reactors has solid experience, and in this sense sodium as the coolant for a today's day is rather competitive. However, such negative its features as fire-risk, not so large margin up to boiling (see. Tab. I[1]), yield of long-life radioactive ^{26}Al in it, rather high radio-activating of primary circuit structures result in necessity of searching for new compositions for use them as coolants in reactors, and first of all in fast reactors in order to increase their safety.

In Russia there is an experience of designing, creation, operation and maintenance of the ship nuclear installations with liquid-metal coolant. In the time, when a problem was decided on what material to use as the coolant, it was selected an eutectic alloy 44.5%Pb-55.5%Bi, mainly, on the reasons of its fire safety, while the melting temperature point of the given alloy not strongly exceeds that one for sodium (see Tab.I). The operated installation had low power and was enough compact. So, its mass-dimensional performance was rather acceptable.

[1] Data presented in Tab. I are taken from [2-5]

Quite another matter is a power reactor. Here, the high density of lead-alloyed coolants at, practically, identical volumetric thermal capacity results in essential increase of weight of reactor installation.

Let's compare coolants on the basis of lead and sodium in specific weight of the coolant in the primary (reactor) circuit per unit of the reactor power:

$$M_{coolant} = 1000\frac{\rho\tau}{c_V \Delta T} = 1000\frac{\tau}{c_P \Delta T}[\text{tons / MWt}], \quad (1)$$

Where

τ – time period of circulation of the coolant in the primary circuit; $c_V = \rho\, C_p$ – volumetric thermal capacity of the coolant; C_p – mass isobaric thermal capacity of the coolant, (see Tab.I); ρ – coolant density; ΔT – mean-mixed heating of coolant in reactor.

So, for the reactor with thermal power 1000 MWt and reactor temperature heating 100 K with the time of circulation in the primary circuit 30s, the weight of the sodium coolant will make 240 tons, while the weight of the lead coolant at the same parameters – 2040 tons. As one can see from (1), so essential difference in weight is caused by a difference only of one thermodynamic parameter – mass thermal capacity, which is a 7.5 – 8.5 times lower for lead-alloyed coolants, than for sodium (see Tab.I).

TABLE I. COMPARISON OF THERMAL-PHYSICAL PROPERTIES OF PB- AND NA-BASED COOLANTS

Properties(at atmospheric pressure)	**Coolant**			
	Pb	**44.5%Pb 55.5%Bi**	**97.7%Pb 2.3%Mg**	**Na**
Melting temperature, °C	327.4	123.4	248.7	98
Boiling temperature, °C	1745	1670	1610	881
Fusion heat, kJ/kg	23	25	26	112
Vaporization heat, kJ/kg	860	820	920	3890
Density at 500°C, kg/m^3	10470	10050	9560	833
Volumetric expansion coefficient, 1/K	$1.12*10^{-4}$	$1.34*10^{-4}$	$1.22*10^{-4}$	$2.93*10^{-4}$
Heat conduction at 500°C, W/m K	15.4	14.4	22	66
Specific heat at 500°C, kJ/kg K	0.147	0.146	0.168	1.254
Kinematic viscosity at 500°C, m^2/s	$17.6*10^{-8}$	$12.8*10^{-8}$	$15.3*10^{-8}$	$29.0*10^{-8}$
Prandtl number, Pr	0.0178	0.0131	0.011	0.0046
Saturated vapor pressure, bar	$5.1*10^{-6}$	$8.7*10^{-5}$	$2*10^{-5}$	0.009
Surface tension coefficient, N/m	0.44	0.42	0.35	0.155

The decrease of specific weight of the coolant in the primary circuit $M_{coolant}$ is possible, for instance, due to increase of volumetric fuel rating in reactor core and decrease of the time of circulation t. Both of them require reduction of flow cross-sections in the primary circuit. However, deterioration of the circuit hydraulic characteristics will simultaneously take place. So, the limitation on weight of the reactor installation with the lead-alloyed coolants could be also a reason of limitation of its power.

Let's consider now those characteristics of coolants, which influent hydraulics of reactor installation. From the thermal balance in reactor core it is easy to define the coolant velocity in it:

$$W_{coolant}^{core} = \frac{q_V H^{core}}{\varepsilon_{coolant}\rho\, c_P \Delta T}, \quad (2)$$

Where q_V – specific volumetric fuel rating in reactor core; $\varepsilon_{coolant}$ – volumetric fraction of the coolant in the core.

One can define a hydrodynamic pressure of the coolant P_{din} as a function of the coolant velocity:

$$P_{din} = \frac{\rho\left(W^{core}_{coolant}\right)^2}{2} = \frac{\left(q_V H^{core}\right)^2}{2\rho\left(c_p \Delta T \varepsilon_{coolant}\right)^2} \qquad (3)$$

The value of hydrodynamic pressure is a parameter, determining the pressure loss due to friction and drag, and, besides, the dynamic vibrational loads on the fuel assembly in the core.

The value of volumetric fuel rating q_V, the height of the core H^{core} and volume fraction of the coolant $\varepsilon_{coolant}$ are determined mainly by desirable neutron-physical characteristics of reactor (reactor breeding ratio, core breeding ratio, void reactivity effect etc.), which, in turn, are caused by a functionality of the given reactor type in the Nuclear Power system. The upper limit of the core coolant heating ΔT is caused by required lifetime of structural materials under conditions of high-temperature cyclic loads, stresses from temperature non-uniformity, high-temperature corrosion; the lower limit of the core coolant heating ΔT is caused by some guaranteed temperature margin to the freezing point of the coolant, and also by parameters of steam-power heat cycle (e.g. in LMFBR the value of ΔT amounts 190-210°C, in the case of power reactor using the 44.5%Pb-55.5%Bi coolant that value should be decreased to 130-170°C; as for such lead-alloyed coolant as Pb and 97.3%Pb-2.7%Mg with high temperature point of freezing, the value of ΔT should not exceed 130°C).

As one can see from above adduced expression for P_{din} , value of hydrodynamic pressure is inversely proportional to density and squared thermal capacity. At the same geometric and power parameters of the core the value of the hydrodynamic pressure of the lead-alloyed flow is a 5-6 times higher than that one of the sodium flow.

In fast sodium breeder reactors with a rather tight fuel lattice (pitch-to-diameter ratio is 1.15 - 1.2) the value of dynamic pressure reaches 20 - 40 kPa. The arising vibrational loads, affecting the fuel rods in the assemblies are rather great. Under such conditions the fuel assemblies keep their workability due to dense packing of bundle of the rods, spaced by wrapped wires; the rod bundle, in turn, is kept by a thick duct.

In PWR fuel assemblies with looser fuel lattice (pitch-to-diameter ratio is 1.35-1.44) the value of the dynamic pressure amounts 10-12 kPa. The vibrational loads at such hydrodynamic pressure are not so high as in LMFBRs. PWR ductless fuel assemblies, spaced by thin-walled grids, keep rather well their workability under such vibrational loads.

The velocity of lead-alloyed coolants in the power reactors should be limited by the value of 1–1,5 m/s, and hydrodynamic pressure by 10-15 kPa. The reason of such limitation is an intensification of processes of core structure materials dissolving and erosive wear with increase of the coolant hydrodynamic pressure. Let's mark here, that the less is difference in densities of structural materials and coolant and the higher is the coolant temperature, the more intensive is the rate of structural material dissolving.

Limitation on velocity at use of lead-alloyed coolants results in necessity of decrease of volumetric fuel rating and increase of coolant fraction in the core, that, on one hand, is useful from a point of view of reduction of the core reactivity excess for burnup compensation and decrease of the void reactivity coefficient and effect, but, on the other hand, results, as it was said above, in increase of the reactor and whole primary circuit weight.

Fig. 1 demonstrates an influence of the fuel lattice pitch-to-diameter ratio, specific mass fuel rating in the core and the fuel rod diameter to the value of hydraulic resistance of the primary circuit in the fast liquid-metal reactors. On the Figure the points are marked, which respect to some operated reactors with the sodium coolant. Also a typical example of hydraulic resistance of the primary circuit with the 44.5%Pb-55.5%Bi coolant and loose fuel lattice is adduced.

Let's compare now the fast reactors cooled by lead-alloyed and sodium coolants on ability of the primary circuit to develop the natural circulation in it in nominal operation and emergencies. Under a level of the natural circulation (n.c.) of the coolant let's imply a percentage of the coolant flowrate relatively to its nominal value due to developing of the natural circulation in the primary circuit at the nominal level of heating in reactor.

Ability of the primary circuit to develop n.c. determines, in main, a level of temperatures of the core structural and fuel components in emergency processes. In LMFBR reactors, for example, rather dangerous is the situation with de-energizing of electric drivers of the Main Circulation Pumps (MCP) at malfunctioning of emergency protection. At a low level of n.c. in such situation the temperature feed-backs have no time to stabilize reactor in acceptable asymptotic state. Heating of the sodium coolant, eventually, results in its boiling and, at presence of positive void reactivity coefficient, lead to a neutron runaway of reactor.

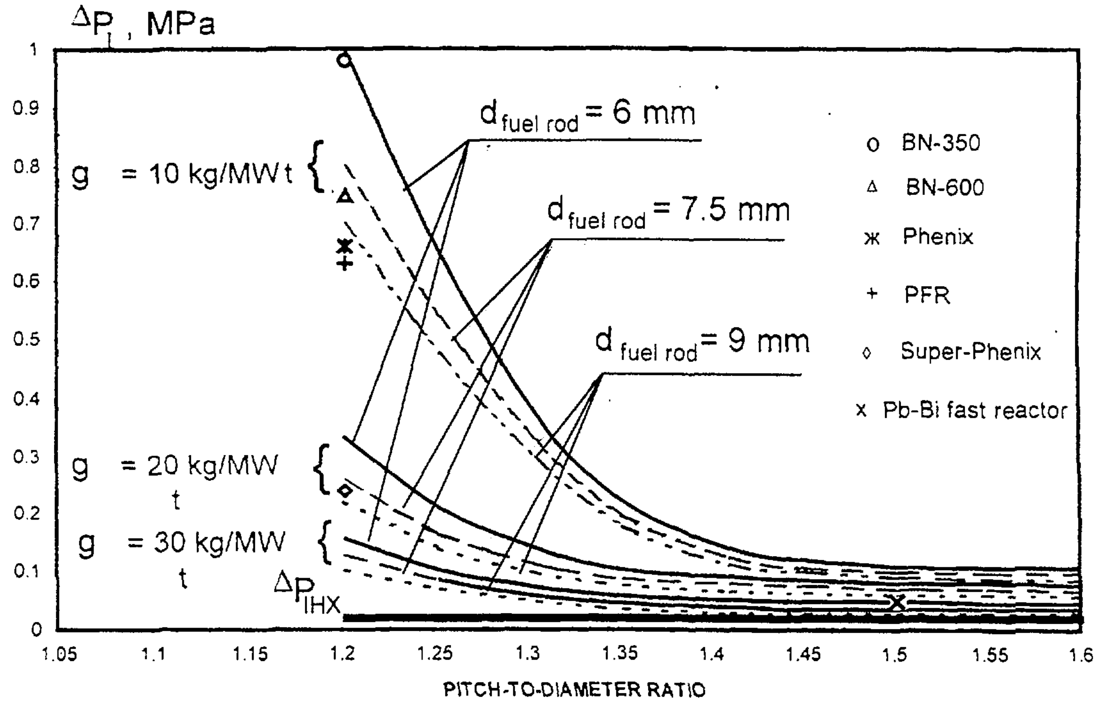

Fig. 1. Hydraulic resistance of the primary circuit of the fast reactors of integral layout with a free coolant surface in the reactor vessel versus a fuel lattice pitch-to-diameter ratio, specific mass fuel rating and fuel rod diameter at a nominal coolant heating in the reactor $\Delta T=160$ °C; core height $H_{core}=1$m. Notation: ΔP_I - hydraulic resistance of the primary circuit; g - specific mass fuel rating in the core (kg-of-fuel/ MWt); $d_{fuel\ rod}$ – diameter of fuel rod; ΔP_{IHX} – hydraulic resistance of Intermediate Heat Exchanger on a side of the primary circuit

As an example let's consider a situation with a primary circuit blow-down more detail. Provided the asymptotic condition of the primary circuit is got, the pressure head of natural circulation

$$\Delta P_{nc} = \Delta T_0 \, t \, \rho \, \beta \, g \, H_{pull}$$

counterbalances the hydraulic resistance of the primary circuit with the running-out pumps

$$\Delta P_I \approx \left(\Delta P_0^{core} + \Delta P_0^{INX} + \Delta P_0^{pump}\right)\omega^2 \qquad (4)$$

Here: ω – level of the natural circulation; ΔT_0 – nominal heating in reactor; $t=\Delta T/\Delta T_0$ - relative heating; ρ, β – density and volumetric extension coefficient of the coolant; g - gravity; H_{pull} – primary circuit «pulling» height (equals approximately a distance from the core middle up to middle of IHX); ΔP_0^{core}, $\Delta P\ P_0^{IHX}$, – hydraulic resistance of the core and IHX respectively in the nominal regime; ΔP_0^{pump} – hydraulic resistance of the stopped pump at the 100 % flow rate in the primary circuit.

By equating the expressions for ΔP_{nc} and ΔP_I, one can get a percentage of the flowrate through the reactor in a mode of natural circulation:

$$\omega(t) = K\sqrt{t}\,, \tag{5}$$

where

$$K = 100\%\sqrt{\frac{g\rho\beta\Delta T_0 H_{pull}}{\Delta P_0^{core} + \Delta P_0^{IHX} + \Delta P_0^{pump}}}\,. \tag{6}$$

Percentage of the reactor power (from nominal value), can be approximately defined as follows:

$$q(t) = Kt^{2/3} \tag{7}$$

From adduced expressions one can see, that both the percentages of the reactor flowrate and power are determined by the relative heating t and some parameter K, which reflects hydraulic quality of the primary circuit, its ability to develop the natural circulation. The value of K equals to percentage of the reactor flowrate and the power at the nominal heating (t=l):

$K=\omega(1)=q(1)$.

To make more clear: how the coolant thermodynamic properties influent ability of the primary circuit to develop n.c., let's transform the expression (6), factoring the term ΔP_0^{core} out off the brackets in denominator:

$$K = 100\%\sqrt{\frac{g\rho\beta\Delta T_0 H_{pull}}{\Delta P_0^{core}(1 + x^{IHX} + x^{pump})}}\,, \tag{8}$$

where $x^{IHX} = \Delta P_0^{IHX} / \Delta P_0^{core}$ and $x^{pump} = \Delta P_0^{pump} / \Delta P_0^{core}$.

Let's express now ΔP_0^{core} through hydrodynamic pressure P_{din} (formula 3) and core drag coefficient ζ_{core}. Now we can transform formula (8) as follows:

$$K = 100\%\sqrt{\frac{g\rho\beta\Delta T_0 H_{pull}}{\Delta P_0^{core}(1 + x^{IHX} + x^{pump})}} = 100\%\sqrt{\frac{g\rho\beta\Delta T_0 H_{pull}}{\zeta_{core}\Delta P_{din}(1 + x^{IHX} + x^{pump})}} =$$

$$= 100\%(\rho\, c_P\sqrt{\beta})\frac{\varepsilon_{coolant}(\Delta T_0)^{2/3}}{q_V H_{core}}\sqrt{\frac{2gH_{pull}}{\zeta_{core}(1 + x^{IHX} + x^{pump})}} \tag{9}$$

From (9) one can see, that the percentage of natural circulation K is determined by design features of the core (q_V, $\varepsilon_{coolant}$, H_{core}, ζ^{core}), hydraulic characteristics of the primary circuit (H_{pull}, x^{IHX}, x^{pump}). The influence of thermodynamic properties of the coolant is exhibited by a complex $(\rho\, c_P\sqrt{\beta})$. And taking into account that the core hydraulic resistance coefficient ζ^{core} is proportional to $Re^{-0.2}$ (Re – Reynolds number), and Re is inversely proportional to kinematic viscosity of the coolant, one can finally obtain, that the complex consisted of the coolant properties, influencing the percentage of the core flowrate in n.c. regime looks like the following: ($\beta^{0.5}\rho\, c_P\, \nu^{-0.1}$).

It is interesting to mark, that the value of this complex is about the same for all the coolants adduced in Tab.I, independently either a coolant is on the basis of lead or sodium.

So, the hypothetical replacement one liquid-metal coolant for another in the given reactor, practically, will not change anything from a point of view of ability of the primary circuit to develop the natural circulation.

Actually, the main parameter, influencing this ability, is the core fuel volumetric rating q_V. This is the parameter, which forms hydraulic performance not only reactor, but also whole primary circuit, since it influents determinantly a volumetric coolant fraction in the core, hence the values of the flow cross-sections in the core as well as the value of the core flowrate, coolant velocity and hydrodynamic pressure. The latter, as it was mentioned above, defines the pressure loss from friction and drag forces in reactor and entirely primary circuit. Hydraulic resistance of the reactor and required reactor flowrate, in turn, determine a performance of MCP, and, therefore, its design. The design of IHX in integral layout of the primary circuit with a free level of the coolant in the vessel is determined by the value of reactor flowrate and vessel sizes (diameter, height), since the hydraulic resistance of IHX in integral layout with free coolant level is limited by the value 0.2 – 0.25 bar, that is stipulated by the requirement of cavitation-free operation of the Main Circulation Pumps and inhibition for gas bubbles, captured by the coolant flow from the free surface, to get the core.

Another one important moment of the core thermohydraulics concerns a scale of non-uniformity of temperatures of Fuel Assembly structures. As rule, temperature distortion across FA, first of all, depend on a scale of temperature heating:

$$T_{max}(z) - T_{mean}(z) = \left(\left(1 + \frac{k_{power}\sqrt{\dfrac{\rho}{\rho_{max}}} - 1}{k_{mixing}} \right) - 1 \right) \Delta T_{mean}(z).$$

Therefore, in decreasing temperature drop in reactor core one can decline temperature differences and hence, reduce both stresses and bending of Fuel Assembly.

Let's compare now the coolants, adduced in Tab.I on their properties, which influent heat-transfer capability of the coolant. The main thermodynamic characteristic, influencing heat removal efficiency of the coolant is its thermal conductivity. Obviously, the sodium coolant has not a competitor on this parameter. Its thermal conductivity is a ~ 4.4 times higher, than that one of lead or Pb-Bi alloy.

Besides, it is necessary to mark, that lead-alloyed coolants are more troubled, than the sodium one, by formation of additional thermal resistance at the fuel rod surfaces consequently of a separation of impurities from the coolant flow and their adhesion at the hot surfaces. Experiments [6-7] have shown, that the heavier impurities move, mainly, in to the flow core, whereas the light impurities and gases are separated at the walls of the channels. For example, lead oxides, transported by the flow of Pb - Bi alloy are accumulated on the walls of the channels, whereas iron particles are suspended in the flow core. Measures, preventing similar appearances can be a systematic monitoring and control of a liquid metal chemistry, mainly, on the contents of oxygen in it. It is important also to apply initially pure metals in processing the coolants, careful cleanup of the protective inert gases from oxygen and moisture.

In some cases the heavy metals Pb and alloy Pb-Bi can be protected by nitrogen. More preferable. however, to use reductant gas mixtures (argon - hydrogenous, nitrogen - hydrogenous and so on) with periodic change of a gas ullage under the free coolant level, where water vapor and other volatile oxidant products can be accumulated.

Rather perspective coolant among lead-based liquid metals is an eutectic alloy of lead and magnesium 97.3%Pb-2.7%Mg. Its thermal conductivity is a 1.5 time higher, than Pb and Pb-Bi one, the thermal capacity is higher by 15 %, that it is important for improving the mass-dimensional characteristics of the reactor (see formula 1) and decrease of hydrodynamic pressure (see formula 3).

In the work [5] it was carried out the experimental study of thermodynamic properties of this alloy, regularities of its interaction with structural materials and ceramics, gaseous solubility and oxides behavior in the liquid metal. The work [5] and consecutive studies of alloy 97.3%Pb-2.7%Mg have shown, that the process engineering of this material in many respects is similar to a process engineering of more investigated lead-based coolants - Pb and Pb-Bi.

It is necessary also to mention a problem, related with a volatility of saturated vapors above a free coolant surface in reactor. In the result of condensation of these vapors on the more cold surfaces of the structures, located in upper plenum of the reactor, impurities can be concentrated due to adhesion at these surfaces from condensing vapors, and these concentrates can be high-radioactive ones.

In Tab.I the data on saturated pressure for the liquid-metal coolants at the temperature 500°C are adduced. One can see, that sodium is a leader there (p_s = 0.009 bar). Volatility of saturated vapors of Pb-Bi alloy results in carry-over of high-radioactive and high-toxic Polonium, being a yield of neutron capture by Bismuth in reactor core, and deposition of Po-containing concentrates on «cold» surfaces. Besides, the value of saturated pressure of Pb-Bi alloy ($p_s = 8.7 \cdot 10^{-5}$bar) is the highest among the lead-based coolants because of presence of a great quantity of Bismuth in this alloy. The liquid lead ($p_s = 5.1 \cdot 10^{-6}$ bar) and alloy 97,3%Pb-2.7%Mg ($p_s = 2 \cdot 10^{-5}$ bar) are in the better position.

CONCLUSION

In multicomponent Nuclear Power system the Power Fast Reactors (PFR) will carry out a function of enhanced breeding of fuel, for the Power Thermal Reactors (PTR) and enlarging of NP scale. Another important role of FR – closing the fuel cycle for all dangerous actinides, minimization their equilibrium quantity. This will require from future Power Fast Reactors to increase mass power rate of fuel, provide high level of breeding (BR ~ 1.3-1.5), minimize a loss of neutrons and fissile actinides together with providing high level of operation reliability and safety.

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