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Concept of the Self-Fuel-Providing LMFBR

Georgy Toshinsky

Institute of Physics and Power Engineering, Russia

ABSTRACT

The concept of nuclear power development on the basis of fast reactors using only fertile material (depleted uranium) as fuel make-up has been studied. It enables for 100 years to develop nuclear power without fuel reprocessing. The expediency of using lead-bismuth alloy as liquid metal coolant has been demonstrated. The key results of investigations performed have been presented.

INTRODUCTION

The development of power and nuclear power (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.

This time period would be characterized by the fact that available natural and enriched uranium resources would be exhausted. It would result into the necessity of searching, mining and developing new uranium deposits at hard-reached areas that would be the cause of increasing the fuel component of the cost of electric power produced by NPPs with thermal neutrons reactors.

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

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 thermal power plants (TPP) using organic fuel. Under the market economy conditions when construction should be performed on the base of repayable credits 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 radiational 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, quotas on releasing the greenhouse gases, significant increasing the cost of electric energy produced by TPP, 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 aud tbat one when NP with FBR and closed NFC can be assuredly competed with that using organic fuel. Then NP would be less economically effective in comparison with that using orgauic fuel ("gas pause").

The circumstances considered makes 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 [1]. The main concept goal is to increase sharply the efficiency of utilization of the natural uranium energy potential without radio-chemical fuel reprocessing and fuel cycle closing for "gas pause" elimination.

FAST REACTOR OPERATION IN THE OPEN NFC (PHYSICAL CONCEPT)

Fast reactors (FRs) are commonly dealt with in the closed NFC when they operate under the breeder regime. And plutonium is mainly accumulated in zones of reprocessing it from depleted uranium. The breeding ratio is less than that one in the core. The core make-up under partial refueling 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 reprocessing" fuel cycle that was first theoretically considered by S.Feinberg and E.Kunegin ("Kurchatov Institute", Russia) in 1958 [2] and then by K.Fuchs and G.Hessel (Germany) in 1961 [3].

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. That reactor we call Self Fuel Providing Reactor (SFPR).

In the reactor under consideration the start-up load of core consisting of fuel elements 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 released energetic plutonium). In the course of reactor operation the FSA of start-up load under partial refueling 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 refueling 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 refueling.

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 refueling must be low.

The chosen scheme of partial FSA refueling and reshuffling should also provide flattening the energy release 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 release. That is why zones with higb 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 uranium (WPU). In this case the efficiency of using energy potential of natural uranium increases several times in comparison with LWR (see below).

FBR SAFETY IMPROVEMENT DUE TO USING LEAD-BISMUTH COOLANT

To achieve high indices of reactor installation (RI) safety, the choice of the primary circuit coolant is of great significance. It determines also the main technical solutions of RI design and its equipment and NPP technical and economical characteristics to a great extent.

Among all the liquid-metal coolants it is sodium that has gained the widest acceptance. This coolant has been chosen for FBR because, due to its good thermal and physical properties, sodium offers possibilities for an intensive heat removal and thus for a short Pu doubling time. This was an indispensable requirement in the early stages of FBR design and development in the sixties-seventies. For this very reason, when considering various liquid metal coolants (LMC) for FBR in 50s, academician A.I.Leypunsky gave preference to sodium though initially he had considered lead-bismuth coolant (LBC) for this purpose [4].

Yet there are some complicated problems related to realization of the reactor concept concerning a slightly enriched or depleted uranium makeup in reactors with sodium coolants. They are associated with a considerable positive sodium void reactivity effect which is characteristic of reactors with low neutron leakage. Besides, the neutron slowing-down due to sodium nuclei decreases the CBR by about 0.06 which plays a leading part in achieving a high efficiency of natural uranium utilization. For instance, as the calculations have demonstrated in the case of sodium coolant, it is difficult to provide the reactor criticality when there is a depleted uranium makeup.

At present and in the foreseeable future there is no need for so short Pu doubling time as can be only provided by FBR with sodium coolant. So the opportunity was afforded to turn back to studies of LBC for FBR.

In our country a long-term experience has been accumulated with respect to the eutectic LBC using in the reactors of the nuclear submarines (NS) [5] which have been developed by IPPE as a supervisor of development. In the process of developing this coolant for use in RIs of submarines several principal problems have been encountered. One of them related to the coolant quality ensurance in the course of operation, another – to the radiation safety associated with formation of alfa-active radionuclide polonium-210 and also some others have been settled.

Making use of this coolant allows FBR safety to be improved further due to its chemically inert behaviour and higher boiling temperature (about 1700 °C) that actually rules out the coolant boiling in the most stressed FSA even in the case of the most severe accidents.

Taking into account that heating energy, chemical and potential compression energies stored in this coolant are minimum in comparison with those of other coolants used it might be result in a RI class with the safety as high as reasonably achievable.

So there is a chance to realize a two-circuit design-scheme of steam generation and to improve technical and economical indices.

The problems limiting LBC use in large-scale future NP depend on deficiency of bismuth resources and its a relatively high cost. It might happen in this case that a bismuth cost (as a part of the overall capital costs) for constructing NPP will be too large, especially when there is a demand to use the ore with a low bismuth content. It may become reasonable to change over to a purely lead coolant proposed for cooling FBR by RDIPE for which there are no raw material base restrictions.

However, due to lead higher melting point, the temperature interval of lead coolant application should be considerably increased. It makes the problem of the coolant technology more complicated as well as the corrosion resistance of structural materials and mass transfer. Being applied to lead-bismuth coolant the problem took 15 years for resolving. Besides, it results in a more complicated operation of RI.

Accordingly, during the preliminary stage of concept development a lead-bismuth alloy is considered as a coolant of non-eutectic composition with bismuth content decreased up to 10 % (versus 56 % in the eutectic alloy). Hence there are no problems with bismuth resources and deterioration of cost indices. At the same time (as compared to a lead coolant) its melting temperature is decreased per 77 °C (up to 250 °C) that facilitates RI operation and allows to diminish considerably the pressure in the secondary circuit (not more than 14.0 MPa, in comparison with 240 MPa for a lead coolant) and reduce maximum temperatures of fuel element claddings up to the level which has been checked up under conditions of long-term operation tests.

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 GW (e) - year).

For considered type of FR EUU is increasing 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, is decreasing with increasing the number of fuel campaigns burnt up in the reactor.

If the use of WPU is ensured as make-up fuel, thus EUU would be maximum achieving under the large number of campaigns the value which is equal to fuel burn up depth (in % on heavy metal). It corresponds to EUU increasiug 10-20 times in comparison with LWR and is explained by the fact that for fabricating make-up fuel there is no need in 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 1). But thus EUU is decreasing, being still several times as high as this one for the VVER-1000 reactor. We can see it from Table 2 where the ratio of EUU for the SVBR-600 reactor with equivalent electric power of 625 MW 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 1

The Depth of Fuel Burning up, Fast Neutrous Damaging Dose, Microcampaign and Campaign Duration as the Functions of Make-up Fuel Enrichment

X_5	g	D	$T_{мк}$	$T_к$
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.a. ,

D - fast neutrons damaging dose, dpa,

T_{mc} - the microcampaign duration, eff. days,

T_c - the fuel campaign duration, eff. years.

Taking into account these results it is expedient to use the flexible regime of reactor operation. In the first campaign the uranium enrichment in the make-up fuel is chosen to be such that the fuel elements operation conditions are to he within the framework 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.

Table 2

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 Natural 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%.

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, radiational and ecological accidents possible for this period if corresponding enterprises are functioning. By assessment 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.

MAINTENANCE OF THE SPENT NUCLEAR FUEL (SNF)

When there is long NP functioning in the open NFC the workload for storages with burnt up FSA increases. These storages must be of suitable volumes and provide long safe and controlled storage of the spent FSA. High 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» storages, it is not exposed to any attacks resulting in damage of shielding 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 shielding resistant to external attacks and decreases the storage cost.

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 : ($D \times H \cong 4.0 \times 1.4$ 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 the exploitment of LBC, thus the higher value of CBR has been provided, which has had a key part in reaching the EUU most 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% h.a. (that is justified at experimental assemblies of EBR-2 reactor), fast neutron damage 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 FSA is about 30 years (that is three times over that obtained by operation of the RI 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 [6].

On the base of the results obtained the engineering design of RI SVBR-600 of the 625 MW 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 SVBR-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 follows.

Capital costs for constructing the NPP along with the SSNF (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 first circuit pipelines and valves, significantly less quantity of auxiliary systems and systems ensuring safety, significantly less (4 times) volume of the SSNF under the same power generation, almost total absence of liquid radioactive wastes (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 cousidered.

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