MOLTEN SALT REACTOR FOR THE MANAGEMENT OF MINOR ACTINIDES FROM VVER 1000/1200 USED FUEL

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Abstract

This paper mainly considers the MOlten Salt Actinide Recycler & Transmuter (MOSART) system without U-Th support fueled with different compositions of transuranic elements from VVER 1000/1200 used nuclear fuel (UNF). Last developments concerned single fluid MOSART design addresses advanced large power unit with main design objectives to close nuclear fuel cycle for all actinides, including Np, Pu, Am and Cm. The optimum spectrum for Li,Be/F MOSART is fast spectrum of homogeneous core without graphite moderator. The effective flux of such system is near 1x1015 n cm-2 s-1. Single fluid 2.4 GWt MOSART unit can utilize more than 250 kg of minor actinides per year from VVER 1000/1200 UNF. The main attractive features of MOSART system deals with the use of (1) simple configuration of the homogeneous core (no solid moderator or construction materials under high flux irradiation); (2) proliferation resistant multiple recycling of actinides (separation coefficients between transuranic (TRU) and lanthanide groups are very high, but within the TRU group are very low); (3) the proven container materials (high nickel alloys) and system components (pump, heat exchanger etc.) operating in the fuel circuit at temperatures below 1023K, (4) core inherent safety due to large negative temperature reactivity coefficient (-3.7 pcm/K), (5) long periods for soluble fission products removal (1-3 yrs). The fuel salt clean up flowsheet for the Li,Be/F MOSART system is based on reductive extraction in to liquid bismuth. The paper has the main objective of presenting the transmutation advantages and fuel cycle flexibility of the large power Li,Be/F MOSART system while accounting technical constrains and experimental data received in this study. The main design choices and characteristics for MOSART concept are explained, including fuel maintenance and engineering safety features. The need for the experimental small power test MOSART unit to demonstrate the control of the reactor and fuel salt management with different minor actinides loadings for start up, transition to equilibrium, drain-out, shut down etc. with its volatile and fission products, is discussed

1. INTRODUCTION

After 2020 facilities of Experimental Demonstration Centre (EDC) being built at the site of the Mining and Chemical Combine (MCC) will begin reprocessing of UNF from VVER-1000 reactors on the basis of innovative technology, providing a recovered nuclear material (refined products) for recycling in thermal and fast solid fuel reactors [1]. After adjustment of all technological processes EDC will become the reference basis for a large-scale RT-2 plant, which should provide an environmentally and economically acceptable reprocessing of VVER-1000/1200 UNF. In accordance with the EDC flowsheet, the highly active raffinate, containing long-lived actinides (243Am, 245Cm, 247Cm, 248Cm), is sent for conditioning. The obtained vitrified high level waste (HLW) belong to the 1st class of radwaste. Use of dedicated reactor unit as a burner of TRUs, remaining after recycle the main part of U and Pu to solid fuel thermal and fast reactors, may reduce the volume and radiotoxicity of HLW.

It is obvious that for operation with TRU loadings, which cannot be managed by traditional solid-fuel reactors, suitable reactor systems must allow: 1) widely vary the nuclide composition of fuel loading without changing the core design, 2) maintain the inherent safety features of the reactor when changing the fuel composition, 3) abandon the manufacture of fuel pellets, 4) ensure the minimum possible actinides losses to waste in multiple recycling. Finally, technologically such a reactor must be prepared for implementation, i.e. should rely on proven technology.

Solid fuel reactors with a fast neutron spectrum are theoretically able to burn TRU successfully. However, the introduction of minor actinides into traditional fast neutron reactors will complicate the design of these reactors, will require the development of new fuels, will complicate and lead to an increase in the cost of its fabrication. The scientific and engineering issues of fuel pellet manufacturing a with significant minor actinides (MA) vector, as well as safety issues for a fast reactor with a such fuel, are not solved nowhere in the world. In addition, for solid fuel reactors with a limited burnup, the loss of TRU to waste stream in multiple recycling will be comparable to the amount of MA burned during the same time.

In molten salt reactors (MSR) the nuclear fuel is in fluid form (molten fluoride salt) and is circulating throughout the primary coolant system. From the outset MSRs were developed as graphite-moderated designs. The first experimental studies and design developments of MSR were performed in the 60-70s of the last century in the US ORNL [2-5]. The 8 MWt MSRE reactor was built and successfully operated from year 1964 to 1969. The success of MSRE stimulated the development of a thorium-uranium 1 GWe MSBR design with thermal-neutron-spectrum. In the Russian Federation, the MSR studies began at the NRC “Kurchatov Institute” in the second half of the 1970s [6].

Today MSR R&D has mainly focused on fast-spectrum MSR options combining the generic advantages of fast neutron reactors (extended resource utilisation, waste minimisation) with those related to molten salt fluorides as both fluid fuel and coolant (low pressure; high boiling temperature; good compatibility with high Ni-alloys, SiC ceramics and graphite; no exothermic reactions with water, air and, optical transparency) [7]. The main attractive features of advanced MSR designs under consideration are as follows:

- Minimum number of parasitic absorbers and as a consequence less number of fissile materials in the core;
- Non limited fuel burn-up with minimal losses of actinides to waste in multiple recycling;
- Flexibility of the fuel cycle - the ability to utilize fuels of various nuclide composition without reactor shutdown and special modifications of the core;
- On-site fuel processing - no temporary storage is required to hold UNF, transportation of UNF and fuel loading for the next transmutation cycle;
- High thermal efficiency, due to the high temperature of the fuel salt (650-750oC).
- Operation in load follow mode.

MSR developments in the Russian Federation on the 2.4 GWt MOSART design address the concept of large power unit with a fast neutron spectrum in the core without graphite moderator. The main characteristics of the MOSART design are given in Table 1. Since year 2000, the NRC “Kurchatov Institute” carried out complex studies on the MOSART project, which included for configurations selected: neutronic, thermal hydraulic and safety analysis; experiments concerned the main physical and chemical properties of fuel / coolant salts; compatibility of structural materials and fuel / coolant salts with its chemistry control [8-10].

The MOSART feasibility at present is beyond doubt. In this paper focus is placed on MOSART system without Th support with main design objective to close nuclear fuel cycle for all actinides, including Np, Pu, Am and Cm.

TABLE 1. MAIN CHARACTERISTICS OF THE MOSART DESIGN

Fuel circuit	MOSART
Fuel salt, mole%	$LiF-BeF_2+TRUF_3$
Temperature, ^oC	650-750
Core radius/height, m	1.7/3.6
Core specific power, W/cm^3	74
Container material	HN80MTY alloy
Removal time for soluble FPs, yrs	1-3

The effective flux of such system is near 1x1015 n cm-2 s-1. The possibility of creating a high neutron flux and the lack of structural materials in the liquid homogeneous core, leads to optimization of the neutron balance, as well as the possibility to change the fuel salt composition without core modification and reactor shutdown, creates favorable conditions for the TRU utilization. The MA burning rate is directly proportional to the core specific power. When choosing this parameter, it is advisable to be within technical limits.

Summary times and possible methods for fission product removal and actinides recycling in MOSART system are presented in the Table 2. Even in the homogeneous core, where removal times for soluble fission products are long enough, taking away of neutron poisons is, of course, the primary purpose of fuel processing. All actinides are immediately returned to fuel circuit.

The main advantages of MOSART are the ability to vary widely the MA content in fuel salt without losing the inherent safety and the absence of stages related to the fuel fabrication and re- fabrication in multiple actinides recycling. The molten salt fluoride mixtures, due to the high separation coefficients between actinides and lanthanides, make it possible to organize an effective removal of soluble fission products, based on the reductive extraction, to substantially reduce the time of the external fuel cycle for actinides and its losses in waste stream in multiple recycling in comparison with solid fuel reactors.

TABLE 2. SUMMARY TIMES FOR FISSION PRODUCT REMOVAL AND ACTINIDES RECYCLING IN MOSART UNIT

Element	Time
Kr, Xe	50 s
Zn, Ga, Ge, As, Se, Nb, Mo, Ru, Rh, Pd, Ag, Tc, Cd, In, Sn, Sb, Te	2-4 hrs
Zr	1-3 yrs
Ni, Fe, Cr	1-3 yrs
Pu, Am, Cm, Np, U	1-3 yrs
Y, La, Ce, Pr, Nd, Pm, Gd, Tb, Dy, Ho, Er	1-3 yrs
Sm, Eu	1-3 yrs
Sr, Ba, Rb, Cs	5-10 yrs

The performed calculations show that the 2.4 GWt MOSART, starting with TRUs from UNF of VVER-1000 with the ratio of MA to (Pu + MA) equal 0.1 or 0.15 (see Table 3), without core modification and changing temperature in the fuel circuit, can use any TRUs make up with the MA to (Pu + MA) ratio up to 0.33. For scenarios 1 and 2 from Table 2 initial core loading of TRUs required will be 3.5 t and 4.5 t, respectively. Core loading of TRUs at equilibrium might be varied from 3.6 t till to 18.0 t depending from the MA to (Pu + MA) ratio in the core make up. 2.4 GWt MOSART with a fuel salt selected can utilize up to 250 kg of MA and about 500 kg of Pu per year (in the case of make up with 33% of MA) [3]. Note, that after each next recycling fuel is getting less attractive for fissile material diversion. During 50 yrs of operation 2.4 GWt MOSART can utilize more than 12 t of MA. Last TRU loading will be transferred to the next MOSART unit to be constructed at the MCC site.

TABLE 3: ISOTOPIC MASS PROPORTION IN INITIAL MOSART LOADING, IN MASS. %

Isotope	Scenario 1	Scenario 2
Np^{237}	6,42	6,51
Np	6,42	6,51
Pu^{238}	3,18	2,77
Pu^{239}	43,93	48,36
Pu^{240}	21,27	19,97
Pu^{241}	13,52	8,30
Pu^{242}	7,88	6,25
Pu	89,78	85,65
Am^{241}	0,55	5,56
Am^{243}	2,33	1,69
Am	2,88	7,25
Cm	0,92	0,59
Total	100	100

For core reflectors located inside the reactor vessel, it is proposed to use nickel. The radiation resistance of high nickel alloys determines the upper limit for core specific power in the MOSART design. For high-nickel alloy at temperatures above 500°C, the embrittlement associated with the formation of helium along the grain boundaries, is the most important process, caused by both fast and thermal neutrons. To obtain an acceptable service life for reflectors (> 5 years), the core specific power should not exceed 130-150 W/cm3 In addition, with this limitation for the core specific power, there is no problem of heat removal from the fuel circuit. In this case the service life for the reactor vessel, made of the HN80MTY alloy, will be about 50 yrs.

In the MOSART design, a well-established molten LiF-BeF2 salt mixture, is chosen as a solvent for TRU trifluorides. The solubility of TRU trifluorides (PuF3 and AmF3) in molten LiF-BeF2 (in mole %) salt mixture with decreased beryllium difluoride fraction of 0.27 and the minimum temperature in the fuel circuit of 650 °C is about 3,1 and 3,7 mole % respectively [9]. 2.4 GWt MOSART the primary circuit will contain about 50 m3 of fuel salt, of which only half is in the core.

The creation of a 2.4 GWt MOSART unit is proposed to be preceded by the construction of small power test unit to demonstrate operation of the reactor plant loaded by different TRU compositions with fuel salt processing unit.

Thus, the MOSART concept, using the advantages associated with the liquid fuel structure, opens the prospect of a significant improvement in nuclear power technology with regard to the closure of the fuel cycle for all actinides. It is proposed to use the technical and technological capabilities of the MCC site to place MOSART in the immediate vicinity of UNF reprocessing facilities, linking it to the EDC infrastructure. It is assumed that the fuel cycle of this complex will be organized as follows: the bulk of the removed uranium and plutonium return to thermal and fast solid fuel reactors, and the remaining TRU are transferred for utilization in the MOSART system. The co-location of MOSART and UNF reprocessing plants, will provide the complex and the surrounding by electricity, facilitates the problems of nuclear materials transport and radwaste management.

2. FUEL MAINTENANCE

In general, to achieve fuel maintenance, (1) the fuel must be delivered to and into the reactor in a proper state of purity and homogeneity, (2) the fuel must be sufficiently protected from extraneous impurities, and (3)

sound procedures must exist for addition and recycling of the actinides required and (4) provision of the required redox potential in the system.

For MOSART that propose chemical reprocessing to remove fission products (see Table 2), the required fuel maintenance operations also include (1) continuous removal (by the sparging and stripping section of the reactor) of fission-product krypton and xenon, (2) addition of Pu, Np, Am and Cm to replace those lost by burnup, (3) in situ production of NpF3 to keep the redox potential of the fuel at the desired level, (4) recycling of all actinides, (5) removal of soluble fission products (principally rare earths); in addition, they may include (6) removal of inadvertent oxide contaminants from the fuel; (7) removal of a portion of the insoluble noble and semi noble fission products. Each of these is discussed briefly below.

Preparation of initial fuel. Initial purification procedures for the MSR present no formidable problems. Nuclear poisons (e.g., boron, cadmium, or lanthanides) are not common contaminants of the constituent raw materials. All the pertinent compounds contain at least small amounts of water, and all are readily hydrolyzed to oxides and oxyfluorides at elevated temperatures. The compounds LiF and BeF2 generally contain a small quantity of sulfur as sulfate ion.

Purification procedures used to prepare materials in many laboratories and engineering experiments have treated the mixed materials at high temperature (usually at 600°C) with gaseous H2-HF mixtures and then with pure H2 in equipment of nickel or copper. The HF-H2 treatment serves e.g. to (1) reduce the Np5+ and Np6+ to Np4+, (2) reduce sulfate to sulfide and remove it as H2S, (3) remove Cl- as HCl, and (4) convert the oxides and oxyfluorides to fluorides. Final treatment with H2 serves to reduce FeF3 and FeF2 to insoluble iron and to remove NiF2 that may have been produced during hydrofluorination. Such a purification procedure can provide a sufficiently pure and completely homogeneous fuel material for initial operation of the reactor.

Addition of actinides. It will apparently be necessary, assuming the fuel volume changes from these additions or other causes do not require removal of any fuel to storage. These will be inherently more complex (and radioactively dirty), and stating which of the options would be preferred is not presently possible. If making a few additions of plutonium and minor actinides to the reactor fuel during its lifetime is necessary, then adding it e.g. as a liquid containing 7LiF-PuF3-NpF4-AmF3-CmF3 mixture should be possible. Alternatively, a procedures presumably could be developed for addition of solid minor actinides trifluorides.

Maintaining the desired UF3/UF4 or NpF3/NpF4 ratio. Operation of the MSRE demonstrated that in situ production of UF3 could be accomplished readily and conveniently be permitting the circulating fuel to react in the pump bowl with a rod of metallic beryllium suspended in a cage of HN80MTY. This technique could be adapted for use in other MSR designs; beryllium reduction would be desirable if the fissionable neptunium additions are to be made as Li3NpF7

Removal of fission-product krypton and xenon. Stripping of krypton and xenon makes possible their continuous removal from the reactor circuit by the purely physical means of stripping with helium. Such a stripping circuit would remove an appreciable (but not a major) fraction of the tritium and a small (perhaps very small) fraction of the noble and semi noble fission products as gas-borne particulates. In addition, the stripper would remove BF3 if leaks of secondary coolant into the fuel were to occur. None of these removals (except possibly the last) appreciably affects the chemical behavior of the fuel system.

Partial removal of noble and semi noble metals. The behavior of these insoluble fission-product species, as indicated previously, is not understood in detail. If they precipitate as adherent deposits on the MOSART heat exchanger, they would cause no particularly difficult problems. However, should they form only loosely adherent deposits that break away and circulate with the fuel, they would be responsible for appreciable parasitic neutron captures. To the extent that they circulate as particulate material in the fuel, insoluble fission-product species could probably be usefully removed by a small bypass flow through a relatively simple Ni based-wool filter system. Presumably, such a system would need to have a reasonably low pressure drop and probably would need to consist of sections in parallel so that units whose capacity was exhausted could be reasonably replaced.

Fuel chemical processing. In MSRs, from which xenon and krypton are effectively removed, the most important fission products poisons are among lanthanides which are soluble in the fuel. Also, the trifluoride species of actinides and the rare earth's are known to form solid solutions so, that in effect, all the LnF3 and AnF3 act essentially as a single element. In combination of all trifluorides, actinides solubility in the melt is decreased by lanthanides accumulation. Since actinides must be removed from the fuel solvent before rare earth's fission products the MSR must contain a system that provides for removal of all actinides from the fuel

salt and their reintroduction to the fresh or purified solvent. This fuel processing system can be based principally on three types of operations: removal of actinides, rare earths, and other fission products from the salt by extraction into molten bismuth. The chemical basis on which the processing system is founded is well established (the coefficients of the distribution of actinides and lanthanides in the Li,Be/F-liquid bismuth system with respect to plutonium at T = 600oC are respectively 6 for curium, 3.000 for neodymium and 25.000 for lanthanum); however, only small engineering experiments have been carried out to date, and a considerable engineering effort remains.

After chemical processing in fuel salt approximately 10% of the initial amount of lanthanides remains (mainly cerium). The purified salt is then transferred to the actinide recycling. The lanthanide precipitate with salt residues is sent for vacuum distillation of the salt components. The lanthanide salts remaining after the distillation are sent to the EDC for conditioning and subsequent near-surface disposal.

3. ENGINEERED SAFETY FEATURES

As mentioned above the main feature of the MSR which sets it apart from the solid fuel reactors is that the nuclear fuel is in fluid form (molten fluoride salt) and is circulating throughout the primary coolant system, becoming critical only in the core. Thus, for any MSR to have equivalent overall containment, greater requirements must be placed on the containment barriers from the fuel salt outward.

The possible problems and engineered safety features associated with this type of reactor will be quite different from those of the present day solid fuel designs. In the MSR, the primary system coolant serves the dual role of being the medium in which heat is generated within the reactor core and the medium which transfers heat from the core to the primary heat exchangers. Thus the entire primary system will be subjected to both high temperatures (>700°C at core outlet) and high levels of radiation by a fluid containing most of the daughter products of the fission process. On the other hand, the fuel-coolant barrier in a solid-fuel reactor, interposed between the heat source and the cooling fluid, is the barrier most vulnerable to damage in a nuclear excursion so that its protection and the consequences of its failure tend to impose more restrictive nuclear safety requirements on a solid-fuel reactor. Because of the low fuel salt vapor pressure, however, the primary system design pressure will be low, as in an liquid metal cooled reactor (LMR) designs. The entire primary coolant system as analogous, in terms of level of confinement, to the cladding in a solid fuel reactor. Although much larger, it will not be subjected to the rapid thermal transients with melting associated with accident scenarios for VVER-1000/1200 and liquid metal cooled designs. Two additional levels of confinement will be provided in the MSR in accordance with present practice. The problem of developing a primary coolant system which will be reliable, maintainable (under remote conditions), inspectable, and structurally sound over the plant's lifetime will probably be the key factor in demonstrating ultimate safety and licencebility.

It is the breach of the primary coolant system boundary, resulting in a large spill of radioactive salt into the primary containment, which will provide the design basis accident. The analogues level of occurrence in a solid fuel reactor would be from major cladding failure (min) to core meltdown (max). Possible initiators of this accident include pipe failure missiles, and pressure or temperature transients in the primary coolant system, failure of the boundary between the primary and secondary salt in the intermediate heat exchanger could be especially damaging. In the event of a salt spill, a possibly redundant system of drains would be activated to channel the salt to the cooled drain tank. The primary system containment, defined as the set of vertically sealed, concrete-shielded equipment cell, would probably not be threatened by such a spill, but cleanup operations would be difficult.

A unique safety future of the MSR is that, under accident shutdown conditions, the fuel material would be led to the emergency core cooling system (represented by drain tank cooling), rather than vice versa. The reactor and containment must be designed so that the decay heated fuel salt reaches the drain tank under any credible accident conditions. In any case, the decay heat is associated with a very large mass of fuel salt, so that melt through does not appear to be a problem.

The safety philosophy for accidents involving the reactor core is very different for fluid-fueled reactors and for solid-fueled ones because the heat source is (mainly) in the liquid and not in a solid, which requires continuous cooling to avoid melting. An LMR, for example, has a tremendous amount of stored energy in the fuel pins which must be removed under any accident conditions. Dry out, which leads to almost immediate

meltdown in an LMR, would not be nearly as severe in the MSR because the heat source would be removed along with the heat sink capability.

For 2.4 GWt MOSART severe accident with the rupture of the primary circuit and fuel discharged on the reactor box bottom was estimated (see Table 4). The model based on mass transfer theory describing main radionuclides distribution between the fuel salt, metallic surfaces of the primary circuit, graphite and the gas purging system was applied for calculation releases to the containment atmosphere. A great deal of practical information on the disposition of these different fission products groups was provided by operation of the MSRE.

TABLE 4. ACTIVITY RELEASES INTO THE MOSART CONTAINMENT IN CASE OF FUEL CIRCUIT FAILURE

Isotope	A_s / A_o	A_g / A_s	A_g / A_o
Te129	0,25	1	0,25
Te132	0,005	1	0,005
Ru103	0,01	1	0,01
Ru106	0,001	1	0,001
Nb95	0,034	1	0,034
Zr95	0,99	0,0011	0,0011
Sr89	0,99	0,00046	0,00046
Sr90	0,98	0,00046	0,00046
Ba140	0,97	0,006	0,006
La140	0,98	0,026	0,025
Ce141	0,99	0,0024	0,023
Ce144	0,96	0,0024	0,023
I131	0,62	0,43	0,27
I133	0,94	0,43	0,43
Cs137	0,7	0,016	0,011

As a criteria characterizing an isotope yield from the fuel salt is accepted the ratio of this isotope activity changed into a gas phase of a containment (Ag) to its full activity built up in a reactor by the moment of the accident (A0). For a molten salt fuel there are three broad classes of fission products: whose fluorides are stable in the salt at its redox potential (soluble fission products), the noble gases and the noble metals. The major soluble fission products are rare earths (including Y), Zr, Ba and Sr, Rb and Cs, I and Br. The noble gases have very low solubility in molten salt and take first opportunity to migrate to any gas phase in contact with fuel. The competitive sinks for noble gases are the pore spaces in the graphite and the circulating bubbles of the cover gas. Another major group of fission products, consisting largely of Nb, Mo, Tc, Ru and Te, does not form fluorides that stable at the redox potential of the fuel salt and is therefore called noble. They are not wet by the salt and also tend to migrate to the salt surfaces. For noble metals it was estimated that about 50% on the metal surfaces and 50% would go into the off-gas system with the bubbles.

After accident considered all noble gases and metals available should move to the gas phase (Ag /As = 1, where Ag /As - the ratio of isotope activity in the gas phase of the containment after an accident to its activity concentrated in the fuel salt by the moment of the accident). However, already as it noted before during the normal operation these nuclides are almost completely leave the fuel salt. As can see from the Table 3 only from 0,1 to 3,4 % of them is remained in the fuel. Therefore, the release for the noble metals is not so big. Only for Te129 it amounts to 25%. This isotope has a sufficiently small half - life period (T1/2=69 min) to leave the fuel salt completely, for example in comparison with Te132 (T1/2=78 h).

Alkaline and alkaline earth metals and rare earth form in the fuel salt stable and well soluble fluorides which have enough high temperature of melting (above 1200oC). For this group in normal operation As / Ao ratio is about 0,95. As a result for isotopes of these group the relative activity yield Ag / Ao comes up 0,1 to 2,5 % . Note, that such isotopes as Sr89 and Cs137 have gas precursors Kr89 and Xe137 with a low half life. But because of low concentration of noble gases in the fuel salt an escape of Sr89 and Cs137 into the gas phase due to gas precursors is insignificant and relative activity yield for these isotopes is not different for others.

4. CONCLUSION

The advantages of MOSART as a TRU burner from UNF reprocessing are primarily due to the lack of the need to manufacture a fuel pellet and the possibility of widely varying the content of long-lived actinides in fuel salt without core modification.

MOSART will allow to burn all the MA produced (250 kg / year) at EDC and approximately 500 kg / year of recycled reactor grade plutonium, while producing 1 GWe of electricity. The construction of a large power MOSART is proposed to be preceded by the operation of 10 MWt test MOSART unit to demonstrate the reactor control and fuel salt management with different TRU loadings for start up, transition to equilibrium, drain-out, shut down etc. with its volatile and fission products. There are opportunities to further improve the efficiency of MA burning in MOSART, which will be justified by the results of the experimental setup.

In order to avoid nuclear materials diversion MOSART reactor plant is integrated (1) at the front end with UNF aqueous reprocessing plant and (2) at the back end with the high temperature fuel salt clean up facility all located at the MCC site. All fresh fuel fluorides containing significant quantities of fissile materials (Pu+MA) for initial loading and make up, will be manufactured onsite by hydrofluorination process. In molten salt pyroprocessing facility the higher actinides would always accompany the plutonium, this operation would never produce a “clean” material would be attractive for diversion. Last TRU loading will be transferred to the next MOSART reactor plant to be constructed at the MCC site.

The industrial MCC site has all the necessary engineering communications, automobile and railway access roads, areas for the expansion of storage facilities, heat and power supply systems, water supply, electric networks. MOSART plants can use the existing radiochemical infrastructure of the MCC, which should somewhat reduce investment costs through the use of existing mine workings, the absence of long arms to transport UNF and processed products, and the availability of qualified personnel.

Introduction of MOSART into the Russian nuclear power system as an integral element will allow solving the problem of utilization of long-lived actinides from UNF reprocessing. The development of the proposed technology on an industrial scale will certainly require solving of a number of technical tasks, however, there are no deadlock problems on this path.

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