Molten Salt Actinide Recycler and Transforming System with and Without Th-U support: MOSART
Dr. Victor Ignatiev, Kurchatov Institute, Russia

Berta Oates
… on 'Molten Salt Actinide Recycler and Transforming System With and Without Thorium-Uranium Support,' otherwise known as 'MOSART,' will be presented now. Doing today's introduction is Dr. Patricia Paviet. She is the Director of the Office of Materials and Chemical Technology within DOE, Office of Nuclear Energy. And she also leads Gen IV International Forum Education and Training Task Force. Patricia?

Patricia Paviet
Thank you so much, Berta. Good morning, everyone. It's my pleasure today to introduce Dr. Victor Ignatiev. He works at the National Research Center, Kurchatov Institute, in Moscow, Russia, where he is the head of the Molten Salt Reactor Laboratory since 2012. He is also a professor since 2009. He graduated from the Nuclear Power Systems Moscow Physical Engineering Institute in 1976. He received his Ph.D. in 1986 from the Kurchatov Institute of Atomic Energy in Moscow as well, with his Ph.D. research focusing on Molten Salt Reactors.

Since 2014, he is the co-chair of the Gen IV MSR pSSC. In 1985, he received the Kurchatov Award on the Fundamental Studies of Molten Salt Reactors. In 2016, he received the Kurchatov Award on Engineering studies of Molten Salt Reactors. The main area of his research activities focuses on Molten Salt Reactor: The thorium-uranium fuel cycle, and transuranic burners; the combined materials compatibility and salt chemistry control in selected molten salt environments at parameters simulating designs operation; the physical and chemical properties for fuel and coolant salt compositions; and finally, the flow sheet optimization, including reactor physics, thermal-hydraulics, and safety-related issues.

It's a great honor to have Dr. Ignatiev with us. Without any delay, I am going to give the floor to Victor. I thank him again strongly for volunteering to give this webinar. Thank you so much, Victor, and you have the floor.

Dr. Victor Ignatiev
Thank you, Patricia, and good day everybody, and let's start our presentation concerning the MOSART approach. Here, you have the contents of our presentation. In my presentation, I will call the issues that deal with the MOSART neutronics, thermal hydraulics, and fuel cycle properties. Safety aspects. Key physical and chemical properties of fuel salt. I would say some words concerning materials compatibility and salt chemistry control. Let's start our introduction.

As you might be aware, from the beginning of nuclear power, two approaches were under consideration. The first one was focused on ‘mechanical engineering device’ which presumes that the fuel solid has to be used in a maximum condensed form that excludes reprocessing and has the advantage of technical simplicity while reactor operating.

Chemical engineering device, which represents another approach, has not only possibilities of general benefits such as unlimited burn-up, easy and relatively low cost of purifying and reconstituting the fuel because it’s fluid, but also there are some more specific potential gains. As you can see on the figure, in MSR devices, solid fuel elements are replaced by liquids, and the fuel is dissolved in the molten salt fluorides. It circulates in the closed circuit. The primary circuit is connected to the processing unit and via boundary with the intermediate circuit.

Our study concerns MOSART system, considering different scenarios for these reactors without and with thorium-uranium support, fuel it with transuranic elements from used nuclear fuel. MOSART design options with homogeneous core and fuel salt with high enough solubility for a higher plutonium and minor actinide trifluorides were examined. The webinar has the main objective of presenting the fuel cycle flexibility of MOSART system while accounting technical constraints and experimental data received in this study. A brief description is given of the experimental results on the key physical and chemical properties of fuel salt and combined materials compatibility to satisfy MOSART system requirements.

On this figure you can see on the left figure and also on the right. You can see the typical flow sheet of the fuel cycle which is under development in Russia. You can understand that in Russia we support the closed fuel cycle. And the recent Rosatom developments concerning the MOSART concept address the advanced large power unit with the main design objective being to close nuclear fuel cycle for all actinides including neptunium, plutonium, americium, and curium.

For example, MOSART started with transuranic fluorides from used fuel, has a flexible fuel cycle and can operate in different modes. The first of which is Transmuter. We now consider [Unclear] and chemical combined as a possible site for construction of the MOSART reactor plant. The unique technical and technological capabilities of this site provide the opportunity to place an experimental MOSART unit in close proximity to the reprocessing facilities of the VVER used fuel, linking it to an experimental demo center under construction now near Krasnoyarsk.

In the table on the next slide, you can see the main parameters of both burner and breeder system which represent a single fluid or a two-fluid

system without or with a blanket. As you can see, it's a large power unit, about 2400-megawatt/thermal. It's a high-temperature system with the outlet temperature of the core more than 700 Celsius. The main attractive features of the MOSART system include the use of a simple configuration of homogeneous core, no solid moderator or construction materials under high flux. Proliferation resistance, multiple recycling of actinides because separation coefficients between transuranic elements and lanthanide groups are extremely high for fluorides. But within the group, they are low. The proven container materials, high nickel alloys, and the system components, pump, heat exchanger, drain tank, operating in the fuel circuit at the temperatures below 1000 Kelvin. Also, inherent safety of the core due to first large negative temperature reactivity coefficient of about minus PCM per Kelvin. And a long period for the soluble fission products removal of about 1 or 2 years.

In this figure, you can see the primary circuit of MOSART which includes the homogeneous core. Fuel salt pump on the hot leg of the primary circuit. And heat exchanger. The maximum temperature of the fuel salt in the primary circuit made of special nitrate-based alloy is mainly limited by tellurium intergranular, dependent on the salt redox potential. The minimum temperature of the fuel salt is determining not only its melting point but also the solubility for actinide trifluoride in the solvent for this temperature because they make up for – the fuel salt is the transuranic elements and we do not use it in these burner system, uranium or thorium support to get criticality.

As you can see, the main construction material is the high nickel alloys, but also the core may include graphite reflector or nickel reflector. And the distribution plate can be made from the chromium carbide ceramics. The addition of these materials to the system can affect on the corrosion rate in the system.

On this figure, you can see the results of the thermal-hydraulic evaluation of the MOSART core. The purpose of thermal-hydraulic analysis was to provide fuel salt flows according to the power distribution in order to decrease the maximum fuel salt temperature in core. To avoid reagents [ph] of stagnant or reverse flows, to decrease the maximum reflector temperature. As you can see, the reference MOSART core configuration satisfies the two most important thermo-hydraulic configurations. Arrangements of reverse or stagnant flows are avoided. And second, the maximum temperature of solid reflectors is low enough to allow its use for a suitable time.

In this figure, you can see in the table the different solvent systems we used in our study. The different feed material characterized by a ratio of minor actinides to transuranic element in the system from 0.1 which is typical for spent fuel of LWR, UOX fuel, up to minor actinides to

transuranic elements ratio of 0.45. Here you can also see the spectrum in the system at the beginning, on the left. And at equilibrium. You can see that it is intermediate between the thermal and fast spectrum.

On the left bottom figure, you see critical concentration of transuranic elements in molar percent versus the time of its result of depletion calculation for the system with the different solvents. Trinary: Lithium, sodium, beryllium fluoride system. And Binary: lithium, beryllium fluoride solvent system.

For the ratio of minor actinides to TRUs equals 0.1. You can see that, when we avoid sodium fluoride from the solvent system, the critical concentration, we need to provide drastically decreased.

Also, if we go to another picture where we can see the lithium-beryllium solvent system. You can see how an increase of minor actinides to TRU ratio from 0.1 to 0.45 increase the critical concentration required. But still, we are within solubility and we have some margin to provide reliable operation in the system. We now will discuss the MOSART operation for the system without [Unclear] materials support.

But we also can add to the system thorium tetrafluoride. On the right figure, you can see that the addition of 2% of thorium tetrafluoride to the system and the step by step increase within 10 years, up to 6 molar percent, provide us operation in the self-sustainable mode. In this case, we do not feed the system by transuranic fluorides.

In the table you can see a contribution to coefficient of different isotopes. You can see that in such a system, about 30% [Unclear] equilibrium we manage with the curium-245.

And next, in this figure we should discuss the limitations we have for the system. For example, for the MOSART core with the typical dimensions, we consider operating in transmuter mode with specific power of 70-megawatt per cubic meter. The possible operational time of the top reflector made of nickel with a decreased concentration of boron is estimated as 10 to 15 years. But for the core with a specific power of more than 120-140 megawatt/cubic meter, specific power will decrease the lifetime of the reflector down to 5 years.

Because we have problems with the helium embrittlement for the nickel-based alloys at temperatures of 700 Celsius. But if we will use graphite reflector, we are also limited by its expansion and irradiation by fast and thermal neutrons. In this case, we will also meet the additional problems with increased corrosion due to the addition of graphite to the primary circuit.

Now, we will discuss the issues dealing with MSR engineering safety features. As I mentioned already, nuclear fuel is fluid and you can see it circulate through the reactor coolant system, transfer heat to the heat exchange and becoming critical only in the core. Possible initiators of reactor coolant system breach accident can be pipe failure missiles and pressure or temperature transience in the reactor coolant system. Failure of the boundary between the 1st and 2nd salts in the heat exchanger. And the problem of developing the reactor coolant system which will be reliable, maintainable, inspectable over the plant's lifetime will probably be a key factor in demonstrating ultimate safety and licenseability.

Molten salt reactor design including MOSART must be designed so the decay heated fuel salt reach the drain tank under any credible accidental conditions. Note that in our system we have common boundaries between the fuel circuit and the processing unit and as well as the fuel circuit and intermediate circuit.

Within our study, we evaluated the MOSART transients. As you can see, for different cases included ultimate loss of flow, loss of fuel heated sink, over-cooling, and insertion of reactivity up to 500 PCM. The conclusion was that MOSART is expected not to be seriously challenged by the major unprotected transients. The system was shown to buffer reactivity insertions up to 500 PCM. In these cases, we can see that the temperatures are expected to rise only about 300 Celsius above nominal under these severe transient conditions. The mechanical and structural integrity of the system is not expected to be impaired.

Concerning the severe accident with the rapture of the main fuel salt pipe and fuel discharged on the reactor bottom, we started consequences for these accidents. Of course, we used the model based on the mass transfer and the experimental data we got from the MSRE operation. As a criteria characterizing an isotope yield from the fuel salt was accepted. And the ratio of isotope activity changed into the gas-phase to its full activity built up in the reactor by the moment of accident. Remember that after an accident, all noble gases and metals available should move to the gas phase. However, as noted before, during the normal operation, these nuclides almost completely leave the fuel salt and are stored outside the primary circuit, in special containers.

For Molten Salt Reactor, to conclude, the total release of radioactivity as you can see from the table on this figure, would be significantly lower by 1 or 2 orders of magnitude compared to PWR. So, for several particular nuclides such as iodine 131 and iodine 133, the difference, as you can see, will be smaller.

The probability of an accident with a relatively low impact for MSR is higher than for LWR. In our opinion this is due to the possibility of

leakage of radioactive liquid fuel in case of accidents in the pump, piping, and valves. But the consequences of severe accidents, in particular, leading to the release of radioactive products into the environment for MSR, in our opinion, significantly less than for LWR. First of all, because the fuel is already liquid and we survive these accidents at the low temperature, for example, at 700 Celsius, less than 1000 Celsius, compared to more than 2000 at LWR, and less content of radioactivity in the fuel.

Now, let's move to the consideration of the materials portability issues. As you can see from these figures, during operation the combined environment's effect on the materials like radiation in the case of MOSART. It's fast neutrons, high temperature, more than 700 Celsius. And the corrosion deals with redox potential in the system, heat up of the fuel salt in the core, and the difference between the highest and the lower temperature in the circuit. Also, less, it depends on the fuel salt fluoride.

Also, we should take into account that fuel addition, graphite addition, ceramic addition in the system, also impurities like iron, like nickel, like chromium, and fission products like tellurium will affect on the corrosion of metallic materials in our system.

As you might be aware, in Kurchatov Institute we started experiments, about 70 different alloys, mainly nickel-based alloys. Among alloying elements, there were tungsten, niobium, uranium, molybdenum, aluminum, and copper. We started our studies with original materials developed in the US, like Hastelloy N, Hastelloy N modified. In the table are summarized the element content in these alloys. As you can see, for our advanced alloy in Russia, we alloy material by aluminum and niobium.

In the last two columns are summarized data for recently developed nickel-molybdenum, and E-721, nickel-tungsten alloys. Also, now we have a Chinese replica of Hastelloy N called GH3535. In our laboratory we studied all the alloys. Experimental results in the polythermal loops with the redox potential measurement demonstrated that operations with FLiBe-based salts also fueled by uranium or plutonium fluorides are feasible using carefully purified molten salts and loop internals. The corrosion rate for domestic Russian alloys in these fuel salts was less, 5 micrometers per year. We have no traces of intergranular corrosion of alloys in the system with the proper control of the redox potential without tellurium. We also studied, not in the natural convection loops but in another facility, the effect of addition of tellurium on intergranular corrosion for lithium, beryllium, thorium, uranium salts, and all for lithium, beryllium, uranium salt. Here in the table, you can see the summary of results. The concentration of the fuel was changed from up to 2 molar percent and the uranium 4 to uranium 3 ratio in the system was changed from 1 to 500. In the last case, we simulated conditions in the processing

unit with a very high redox potential. In line 5, we simulated operational condition of the MSRE reactor with a uranium 4 to uranium 3 ratio of about 100.

You can see that in this table results we got for tellurium corrosion in the lithium, beryllium, uranium salt where we compared in the first column the addition of Hastelloy N, and high-nickel alloys, HN80MTY alloyed by 1 mass percent of aluminum at increased temperatures of up to 800 Celsius, with the uranium 4 to uranium 3 ratio from 30 to 90. And as you can see, if we carefully control the redox potential in the system at uranium 4 to uranium 3 ratio equal 30, there are no traces of cracks, no tellurium corrosion. But if we increase the redox potential in the system up to 60, we already have traces of the intergranular corrosion for the Hastelloy N. But still, alloy alloyed by aluminum works very well. Even we increase both, temperature and uranium 4 to uranium 3 ratio up to 90. In this case, we will have alloy alloyed by aluminum 10 times more resistant to tellurium corrosion.

Note that the reaction on the bottom of the figure which is responsible for the transfer of free tellurium to the structural materials and prevents such corrosion rate.

And now some words concerning the properties of the fuel and coolant salts we consider for the MOSART. Of course, the choice of the fuel in coolant salt depends on the concept of what it should be, burner, breeder, or another type of system where we use the molten salt only as a coolant. But in most cases it's lithium-beryllium fluoride salt like FLiBe. The problem is that in concentrations of individual salts in the system, for example, for the burner, in order to increase the solubility of actinides, trifluorides, we need to decrease the concentration of beryllium difluoride in the system for example. Of course, in this case we will get a system with a bit higher melting temperature and so we pay for increased solubility by the increase of melting temperature. But if we want to decrease the melting temperature, of course, we can increase the concentration of beryllium difluoride up to 50 molar percent, but in this case we will have a significant increase of viscosity in the system. We should take into account all these peculiarities, but now we are focused on the MOSART which operates mainly on trifluorides.

First of all, we managed and measured the actinides and lanthanides trifluoride solubility in the system. We measured it by two methods. First is the thermal saturation and second is gamma scanning of plutonium 9 in the system. You can see on the figures, on the right part of these figures, the solubility increased with the temperature. We measured it both, for plutonium and for americium. We got the data for individual solubilities for actinide and lanthanides. As I told before, mainly for the FLiBe-based system but we also measured it for FLiNaK without and with the addition

of actinides, trifluorides, and tetrafluorides. I should mention that when we measured not only individual solubility of trifluorides but also joint solubility. Take a look on the right table. Yes, you can see the liquid temperature for the ternary lithium, thorium, uranium salts. The thorium trifluoride liquid significantly displace plutonium trifluoride in the system and we should take it into account when we prepare our design.

I should tell you that we have all the data required to prove MOSART concept concerning its transport properties and their behavior versus the temperature is quite clear for us now.

Concerning the radiation, stability of the salts under irradiation. Of course we have a lot of radiation tests made in Oak Ridge Labrador. But also in Kurchatov we did experiments on radiation stabilities of tenfold [ph] FLiBe-based compositions without fuel addition, with the addition of uranium, and with the addition of fertile materials. Also, we replaced lithium by sodium fluoride in order to decrease the fluoride evolution.

The main conclusion is that the radiolysis of molten fuel salt at reasonable power densities is not a problem up to very high temperatures, more than 1000 Celsius. It seems unlikely though, it's possible that MSR fuels will evolve fluoride on cooling, at temperatures below 100 Celsius. If so, we should prepare some arrangements for their storage at elevated temperatures until the fraction of the decay heat energy is dissipated.

Now, some words concerning the fuel processing. In single-fluid MOSART which is based on FLiBe, we have quite a simple processing scheme. The required fuel maintenance operation in our case should continuous removal by dispersion and stripping [ph] section of reactor of fission product, krypton and xenon.

Second, addition of uranium if necessary, and thorium, and the transuranic elements to replace a loss by a burnup. Third, ensure the production of uranium (III) to keep redox potential of the fuel at the desired level. Also, recycling of all actinides in the system should be provided. Removal of soluble fission products, principally the array of elements should be included in the system. If necessary, we should manage isolating of protactinium-233 from the region of high neutron flux, but it's not the case for the MOSART, so we do not manage it.

We should also provide the removal of oxide contaminants from the fuel. In addition, they may include both, additional thorium tetrafluoride, replace that loss by transmutation or stored with the fuel removed from the operating circuit. Finally, the removal of the portion of insoluble noble and semi-noble metals will be appreciated. In this figure, on the right, you can see the flow sheet for the core of the MOSART. And in this case, first we remove zirconium. After that, we remove actinides by reductive

extraction. And only after that, lanthanides, and after that we reintroduced actinides to the fuel salt, make the valency adjustment, and return the fuel salt to the core.

We at our institution are quite confident with the processes of reductive extraction into the liquid bismuth. Of course, it's only static laboratory experiments but our basis is that the coefficients of distribution of actinides and lanthanides in the FLiBe liquid bismuth system with respect to plutonium are for curium 6 and for lanthanides equal 3000, and for niobium [ph] and lanthanides, it's 25,000. So, they are very high between groups as I told before.

And it will be quite a short and simple scheme if we will operate with the fuel without thorium support. But if we add the thorium in the system, the separation coefficients between lanthanides and thorium are very low, and they are close to 1. And in this case we need to complicate the processing flow sheet by the use of a chloride system and move these elements to chloride system where we have much better separation coefficients to manage it.

We are very close to the conclusion in our system. What I want to say in my last figure, that MSR concepts offer alternative options for the new fuel breeding and long-lived waste incineration with the added value of liquid fuel. As we discussed, intrinsic safety features, fuel cycle flexibility, simplified fuel processing, in-service inspection, no fuel transportation and re-fabrication required. Significant progress has been made last year on the resolution or cancellation of critical viability issues. Material compatibility, salt physical and chemical properties, reprocessing feasibility, intrinsic safety.

Pre-conceptual studies of the whole reactor and reprocessing unit must be performed to establish the MSR viability. I should stress experimental infrastructure, analytical and integral salt loops with real fuel salts, of course, including forth convection loops are required to proceed further in the mastering of MSR technologies including tritium control and the main components like long-shaft pump, heat exchanger.

Thank you for your attention. I finished.

Berta Oates
Thank you, Victor. If you have questions for today's presenter, please go ahead and type those into the Q&A chat pod now. I see that there are a couple in there. Before we address those questions, let's take a look at the upcoming webinar presentations. In July, we anticipate a presentation by Dr. Varaine on Astrid: Lessons Learned. In August, a presentation on BREST-300 Lead-Cooled Fast Reactor from Dr. Rachkov from Russia. In September, a presentation on Advanced Lead Fast

Reactor European Demonstrator," otherwise known as 'ALFRED' project, from Dr. Alessandro Alemberti from Italy.

Victor, on the Q&A pod, there are a couple of questions. If you scroll over, there are two tabs. One is the 'presenter view' and one is the 'participant view.' If you look at the 'presenter view,' you can see where I have advised about the audio test, but there is the fourth one. There is a question from Boris Hombourger. 'Why put thorium in the fuel salt in the case of a two-fluid configuration? Doppler effect? Improved breeding?' Do you see the questions?

Dr. Victor Ignatiev
I cannot see the questions, sorry.

Berta Oates
Use the mouse. You see the Q&A pod, the chat pod for Q&A. Along the header of that, if you scroll over, there are two tabs. When you scroll over, you can click between presenter or participant view. You click 'Presenter' and then you can see the questions as they line up.

Dr. Victor Ignatiev
Okay. I have a question from Boris Hombourger. 'Why put thorium in the fuel salt in the case of a two-fluid configuration? Doppler effect? Improved breeding?

We considered the two-fluid scheme for two reasons. First of all, to simplify the processing flow sheet. We can simplify the processing flow sheet in the case of a two-fluid system. And also for this system we will have better neutronic characteristic. Is it okay?

Next question, 'Which computer codes are used for calculation?' Okay. For neutronic calculations, we used mainly MCNP plus origin code. Now, we also make calculations with the Serpent 2 version and it's quite good in agreement. But also, we used for calculation Russian domestic Monte Carlo code called MCU. Is it okay?

Okay. Thank you.

Berta Oates
There is a follow-up question where Damir has asked 'And for thorium?'

Dr. Victor Ignatiev
Yes, Damir, we also use this code for both for uranium/plutonium systems, and for the MOSART system with the thorium/uranium support. 'For thermal-hydraulics?' Yes. As you might be aware that Serpent 2 is suitable for open form. We also use the domestic Russian code Flow Vision [ph] for thermal-hydraulics. Okay? Okay, thank you.

Berta Oates
Any other questions?

John Kelly
Yeah. Are there any other questions? This is John Kelly.

Berta Oates
Hi, John.

Dr. Victor Ignatiev
Yes, John. Yeah.

John Kelly
Yeah. Victor, I really appreciate your webinar. It was really excellent. If there are any other questions, please put them in now because we have a little bit of time. I guess not. So, again, Victor, thank you so very much.

Dr. Victor Ignatiev
Thank you, John.

Patricia Paviet
Yeah, thank you, Victor.

Dr. Victor Ignatiev
Thank you. Shall we finish?

Patricia Paviet
Yes. I think we are done, Victor. Thank you so much. Thanks, everybody.

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