Concept of European Molten Salt Fast Reactor (MSFR)
Dr. Elsa Merle, CNRS, France

Berta Oates: Good morning. Welcome, everyone, to the next Gen IV International Forum webinar. Before we actually get started this morning, let's just do a little bit of housekeeping. There are three, actually there are four pods that you should see. One is of course the Presentation pod. There's also a Question and Answer pod with a chat box for you to enter your questions. We will take those questions at the end as time allows. The second pod has the file PDF version of today's presentation slides. And in the third pod is a link to an online survey that we invite you to take to provide feedback. We appreciate your feedback and take those comments quite seriously to make improvements to the GIF webinar program and appreciate that information.

It sounds like in the background we do still have one computer speaker broadcasting. I hear a little bit of echo in my headset, so if we can please check our computers one more time and make sure that we're muted.

With that, we will start today's presentation.

Doing today's introduction is Dr. Paviet. Patricia is the Director of the Office of Materials and Chemical Technologies within the DOE Office of Nuclear Energy. She is also the Chair of the GIF Education and Training Task Force.

Patricia Paviet: Thank you, Berta. It's a pleasure to be here today to reintroduce Prof. Elsa Merle. She is a Professor in France at the PHELMA engineering school of Grenoble Institute of Technology, Director of the Master's Program in Reactor Physics and Nuclear Engineering. She is also working at the Laboratory for Subatomic Physics and Cosmology of Grenoble in the research staff, in the MSFR research team.

Since 2000, she has been actively involved in the CNRS program dedicated to the conceptual design of innovative Gen IV reactors. As such, she is contributing to various studies and validations of the concept of molten salt reactors, and more specifically, since 2008 on the definition and optimization of the concept of MSFR. She is in charge of the work package 1, “Integral safety approach and system integration” of the Euratom project SAMOFAR of Horizon2020, and she represents the CNRS at the steering committee on Molten Salt Reactors of the GIF.

So I'm very happy to have you here, Elsa. Thank you so much again for volunteering to give this webinar, and you have the floor.

Elsa Merle: Thank you, Patricia. And thank you for the introduction. So hello, everyone. I am very happy today to have the opportunity to present you some historical elements and the current R&D activities on the specific MSR concept, the ??? [00:03:20] in Europe since some years, the molten salt fast reactor.

So before arriving at the concept of MSFR, I would like to start at the very beginning and why this name of molten salt reactor? So at the very beginning, in the first studies on nuclear reactors, of course people had a specific look at liquid fuel reactors. So why? Because of the advantages of having a liquid fuel. You have homogeneity of the fuel, no loading plan required.

The heat may be produced directly in the heat transfer if you use the fuel as the coolant. And then there is no delay in the heat transfer between the production of the extraction and the very fast thermal feedback.

Then, you have the possibility to reconfigure passively the geometry of the fuel. With one configuration optimizing the electricity production in the core and one configuration allowing long-term storage and cooling of the fuel. Between the two of these configurations, you can only have a gravitational draining of the fuel, which is very fast and very easy.

Then, you have the possibility to reprocess the fuel without stopping the reactor during reactor operation. Then you can have better management of the fission products and have a better characteristic in terms of neutronics and physicochemisty. And then, there is no need for a reactivity reserve when you start the reactor because the fertile and fissile matter may be adjusted during reactor operation.

When you consider all these advantages, then the next question is, which liquid fuel because liquid fuel seems very interesting, but which constraints do we have under liquid fuel?

So to choose your liquid fuel you have to have a not-too-high melting temperature but a high boiling temperature, low vapor pressure, liquid transparent to neutrons because as a fuel you have to have both thermal and hydraulic properties because this fuel is also the coolant. And to have all the advantages mentioned before, it has to be stable under irradiation of course to allow good solubility of fissile and fertile matter in it because we want to use it as fuel. And to avoid production of radio-isotopes hardly manageable, and finally, we have to have a solution to process and control the fuel salt.

And then, when you will list all these constraints, finally you find that the best candidate to have a liquid fuel reactor is to use fluoride or chloride salts, and that's why it's called the molten salt reactor.

And then to finalize the choice of the fuel salt that we did for the MSFR, the last question is fluoride or chloride salt? Then ??? on neutronic and chemical studies, and by combining them we finally chose one of the options. But first, when I look at the neutronic issue, you can consider, as shown here, the neutronic spectrum in green for fluoride sale and in purple for chloride salt, and when you combine these, a neutron spectrum with the neutron availability, on here we consider the thorium fuel cycle. Then, you can observe that the breeding in the reactor, the fuel breeding, is better for the fluoride salt compared to the chloride salt. When you combine all these neutronic considerations, you have a better breeding ratio for the fluoride salt.

So why choose the thorium fuel cycle? Because it's the basis of our reactor right now because there is only one fissile matter available on Earth, it's uranium-235, and then we have to do nuclear energy production starting from this uranium-235, and these can be used to produce either uranium-233 in the thorium-uranium fuel cycle or plutonium-239 in the uranium-plutonium fuel cycle.

The uranium-plutonium fuel cycle is more classical because it's already used in light water reactors, part of them, a part of the energy for PWR is based on this, but the thorium fuel cycle is not yet used, and it's the only alternative to the plutonium fuel cycle, so we chose to study this thorium fuel cycle in a liquid fuel reactor. And then on the other side, thorium is emitting high energy gamma rays, and so it's easier to use thorium in a liquid fuel reactor than in a solid fuel reactor to avoid the fuel fabrication.

So the thorium and molten salt reactor presents a very interesting characteristic when you combine them. So it's for first ??? for the advantage for the fluoride salt; it's better when we want to use a thorium fuel cycle.

And then, another neutronic consideration, it's irradiation damages. I show again the neutron spectrum, and you can see that the spectrum is much harder in the chloride salt compared to the fluoride salt. And then, since the neutron spectrum is less fast with fluoride salt, it results in reduced irradiation damage, both DPA and helium production by a factor of 5 to 7, which is really better for the material, for the structural materials.

Then, another consideration is chemical issues. When you look at the production of radioisotopes in the case of fluoride salt or in the chloride salt, you can see the production of chlorine 36 in the case of the chloride salt, which is a very radioactive element and very hard to manage and to confine, and for us it seems to be really a key point to choose instead the fluoride salts.

So when you combine all of this, we have chosen the fluoride salt because of chemical considerations to avoid production of chlorine 36 and to reduce the irradiation damage, which is already important, and we have also chosen the thorium fuel cycle right now to have a higher breeding ratio combined with the fluoride salt on spectrum and to have a smaller production of minor actinides.

So these are the bases of what we are studying right now on the MSR concept.

So just a small word on the historical studies, and more precisely, historical studies in Europe because there was of course a study in the United States with the aircraft reactor experiments and the molten salt reactor experiments, experimental reactors, molten salt reactors, in Oakridge in '15 and '16, and these reactors were based on a graphite matrix while the fuel salt was circulating.

And then, we started again, some people started again studies in France for example, and in Europe, and for example, the TIER I project of Mr. Bowman. Also, a reevaluation of the MSBR, molten salt breeder reactor, which was the project of Oakridge for a breeder industrial reactor and molten salt reactor.

And we started again calculations on the MSBR using the tools available right now, computer tools and capacities available now, and for example using a probabilistic neutronic code at the ??? MCNP, combined to an in-house code for material evaluation.

And then the MSBR has been recalculated here. And we highlight some, of course, great advantages of the MSBR project, a very promising project, with many studies and details. It was very, very useful to start on MSR right now, but there are some problems to be solved, for example, the null to positive thermal feedback coefficient, a positive void coefficient, very heavy fuel processing due to the thermal spectrum. There are some problems related to the graphite moderator. Then the ??? was not to start from the MSBR and to optimize key point-by-point the MSBR but to start again from the advantages of a liquid fuel reactor.

So I list again the advantages of the liquid fuel reactor and to combine them to the neutronic optimization of MSR that was based on the Generation IV criteria, because since then, the Generation IV International Forum has defined criteria for the next generation of reactors, and then we can use this criteria to design a new (ration) of MSR just by considering the advantages of liquid fuel and the neutronic Gen IV criteria.

It is safety by looking at the negative feedback coefficients to have a stable reactor, sustainability by trying to reduce the irradiation damage in the core, deployments by combining good breeding of the fuel and reduced initial fissile inventory.

And then, to do so, we started again from this basis, and there were sensitivity studies, systematic studies – they were from CNRS at that time – by considering different moderation ratios in MSR by varying the fraction of graphite compared to the fraction of fuel.

So here, what is shown here on the left is the different reactor configuration with the same fuel cell volume, the fuel cell volume in the yellow is constant, but we varied the graphite amount in the core, a core with a large amount of graphite to a core without graphite inside.

And when you modify the graphite amount in the core, you modify the moderation ratio, it needs to modify the neutron spectrum, from the blue one corresponding to the configuration at the bottom, a very thermal configuration, to the purple one, which is the version with no graphite moderator in the core, which is a fast spectrum ratio of the MSR.

And then the (energy) for the different versions is to study the advantages and drawbacks of each configuration.

And then, the conclusion of these studies were that for a thermal spectrum configuration of MSR, we have a positive feedback coefficient, which is not acceptable for us, an iso-breeder reactor, which is correct, a quite long graphite lifespan, but really important, the amount of graphite in the core. So the graphite management will be very heavy, and the very positive point is a very low uranium-233 initial inventory, fissile inventory.

In the second one, the epithermal spectrum configuration, you have a quite negative feedback coefficient, which is better, still an iso-breeder configuration, but a very short graphite lifespan, and still a lot of graphite in the core, so the graphite management is still very complicated. And the initial fissile inventory is correct.

And finally, for the fast spectrum configuration, it means you have finally no more graphite in the core, only a fuel salt. We obtained a very negative feedback coefficient, which is very good for the stability of reactor, a very good breeding ratio, no problem of graphite lifespan since there is no more graphite in the reactor, only a larger initial fissile inventory.

And finally, which was this configuration, after some optimization that I will show you, will lead to the molten salt fast reactor.

So the last optimization that I will show you is to resolve this problem of initial inventory. So there exists to reduce the initial and fissile inventory by modifying either the reactor design and/or the produced power. So what we chose to do is to play on the specific power of the reactor and to fix the produced power, here.

So we fixed the produced power to 3 GWth to be equivalent to the current reactor under construction, that is the EPR, to have a high reactor producing a large amount of energy, and just to have a challenge, because if we have a problem we can reduce this produced power and then find another solution, but if we find an optimized configuration with high power, then we will find optimization, we will find a configuration with reduced power as well. So we fix the produced power and then we play on the fuel salt volume and the core geometry.

And one very important thing is that if the liquid fuel, there is no solid matter inside the core, we can reach a specific power much higher than with solid fuel.

The constraint we have here are the capacities of the heat exchangers to extract heat because the fuel is a coolant. If we reduce the amount of fuel, we reduce the amount of coolant, and we have to extract heat with this coolant. We have to reduce the neutronic irradiation damage to the structural materials. And the larger the core is, the smaller the specific power is and the smaller the irradiation they make is ???.

And finally, we want to have external neutronic characteristics of the reactor in terms of burning efficiency and deployment capacity.

And when we combine all these factors, we can observe that we finally find a very satisfying, promising configuration with 18m3 of fuel salt with a specific power of 330 W/cm3, and this corresponds to an initial fissile inventory of only 3.5 tons per GWe. We already reduced fissile inventory when compared to all of the (future) reactors.

And I have to mention one thing. We only have one fissile load required per reactor. There is not one core used for energy production and one core under reprocessing. The processing actually is done during the reactor operations.

We only need one fissile load per reactor (total), and this is very small, 3.5 tons only, is required for each reactor in operation.

So, to summarize, when we combine the advantages of the liquid fuel to the Gen IV criteria in terms of neutronic optimization, we are trying to redefine an innovative MSR concept called by the GIF Policy Group the molten salt fast reactor, MSFR, in 2008. And we have here a part of the GIF Annual Report in 2008, and the MSFR is selected as a long-term alternative for future fast-neutron reactors with an excellent feedback coefficient. The ???, we have been asked to assess the specific technology challenges and the safety approach, and I will come back to this in this presentation.

So we reached all the targets in terms of neutronic optimization.

Then, the work now relies on other parts of ??? neutronics because what I have presented up to now is only neutronic considerations, neutronic issues, but a reactor, a nuclear reactor, needs the other expertise to be optimized, so then we add the European Project, Evaluation and Viability Of Liquid Fuel Fast Reactor, EVOL, of FP7, which was a Euratom/Rosatom cooperation, which I listed here the European partners involved at the very beginning of the project, with two observers, Politecnico di Milano or the Paul Scherrer Institute, working now on the MSFR, and this project was coupled to the MARS project, Minor Actinides Recycling in Molten Salt, of Rosatom. And together, we did a very nice job to propose the design of the MSFR given the best system configuration issued from physical on this type, chemical, and material studies. And you can find three work packages of EVOL listed here, Design and Safety, Chemistry and Reprocessing, and Structural Materials.

And finally, some examples. I will show some examples of the outputs of the project.

First, for example, the design of the MSFR before this project was very simple, and as you can see here, thanks to (thermalizer's) calculations, to have uniform heating of the fuel salt in the core or no hot point in the core, we finally proposed an optimized toroidal shape of the fuel salt, of the core, sorry, and we proposed also an optimized initial fuel salt composition. I will detail this later on.

We performed a neutronic benchmark to compare the tools from the nuclear databases used for the MSFR, and we concluded, because the (partners) of the tools we have are very reliable and we obtain equivalent results, but the main uncertainties are coming from nuclear databases and nuclear data and on the thorium cycle.

Then we started some first developments of the safety assessment method for the MSR, and recommendations for the choice of the core structural materials.

So to illustrate some part of this, for example, considering the fuel processing, as you know, 4th generation reactors have to be breeder reactors. It means that the fuel processing is mandatory to recover the produced fissile matter. And when you have liquid fuel, the processing can be done on the same site and during reactor operation, which is very useful.

Then, there are two parts of the processing. The display here is a simplified version of the MSFR with two ??? processing that have been chosen. First, the motivation for this processing is also the extraction of the fission product to control the physicochemical properties of the salt and to avoid erosion, corrosion, et cetera, and to keep good neutronic properties to the reactor, which is the case for all reactors.

And you have two kinds of reprocessing: the injection of bubbles in the core and the processing by batch of fuel salts on a daily basis.

So the physical separation is an injection of gas to extract the non-soluble fission products. It means the gas and the metallic fission products, and then the chemical separation is done on some liters of some, some 10 liters of fuel per day, and it's on site but outside of the reactor vessel. It's located outside of the reactor vessel.

Then the next study is to see the link between the reactor operation on the processing. Because it's done during reactor operations, people are asking the impacts of the processing on the safety. So the first study that has been done was the impact of the processing on the neutronic properties of the reactor. So on the left, as I display, the breeding ratio evolution as a function of the reprocessing rate. So the time to process the whole fuel salt volume.

As you can see, up to some years, up to the processing of some years of the whole core, the 18m3, you have no impact at all of the chemistry on the neutronics. Because I listed those, I present the absorption of the neutrons by the nuclei present in the fuel salt, and you can see that absorption, the capture on the fission products are negligible, completely negligible. And you can see exactly the same thing on the bubbling processing. The breeding ratio is completely independent from the bubbling time, the time to extract the non-soluble fission products.

And this is due to the fact that we have in the fast neutron spectrum very low capture cross-sections, and then the chemistry has no impacts on the neutronic operation of the reactor. So it's very interesting because there are no safety issues due to the chemistry or failure of the chemical processing unit to the neutronics and then to the safety of the reactor operations, and then it's possible to do (power) studies of the chemistry on the neutronics. The studies are not directly linked. And this is very interesting.

Then another question that we asked in EVOL was how to start MSFR. So in a fuel cycle that is not already used in the world, there is no uranium-233 on Earth, as well there is no plutonium-238, -239 on Earth, so how can we have the initial fissile, the first fissile load to start MSFR?

So we can start directly using uranium-233 produced in another reaction, Gen III for example or other Gen IV reactor, or we can start with enriched uranium, but it requires an enrichment higher than 20%, so it's not okay regarding pressurization issues. Or we can start with plutonium currently produced in the reactor under operation, but we have to take into account the solubility limits of these elements in the fuel salt. And finally, the best idea is to do a mix of these solutions to have thorium as fertile matter combined with uranium-233 and transuranic elements for example, or enriched uranium with transuranic elements, et cetera.

So we did a lot of studies by combining these and studying the deployment capacities of the breeding ratio of (inversion).

And finally, the frame of the EVOL project, we did a selection of the optimized fuel salt composition regarding not only neutronic consideration but also chemical consideration and material issues.

And finally, we proposed an optimized initial composition of the fuel salt displayed here containing enriched uranium at the level of 13% with transuranic elements produced by a current reactor, TWR, light water reactor for example. And thanks to these, we can start with matter existing on Earth, available. It's possible to start MSFR right now. And the reactor operation will be perfect.

I illustrate here the evolution of the fuel from this version of the reactor. Below is the solubility unit for plutonium, and as you can see, in this reactor we burn these transuranic elements, the plutonium, the zirconium, and the americium amounts are reduced. This reactor burns these elements while producing uranium-233 from thorium, and finally, it's also a breeding ration of the MSFR that we can start with what we have on hand right now.

We also performed some more complicated studies, deployment studies, as you can see here, with many versions of MSFR combined because in the first time, we have produced fissile matter, then we have to stabilize the amount of fissile matter, and we go up to the end-of-game to see how we can stop such a fleet of reactors, and it's completely possible. You just change the salt in MSFR and you put salt with ???, and then you burn what was the inventory of the produced MSR reactor.

And we show that we have very good deployment capacity. A transition to the fuel cycle may be achieved, and we can also close the current fuel cycle by reducing the stockpiles of produced, currently-produced transuranic elements.

So this is what's going on up to the EVOL project.

And finally, at that time, the MSFR looks like that. So as we say here, the energy is produced in the middle of the core, so the fuel salt is staying red, just to see it. It enters the core here. Fission occurs here, heating the salt that is extracted and sent to the heat exchangers so that the heat is extracted. The salt is cooled and then ??? liquid. And one cycle like that is very fast, some 3 to 4 seconds.

It's a liquid circulating fuel reactor. The fuel is used as a coolant. And a mean fuel temperature around 700° with a fast neutron spectrum on a thorium fuel cycle.

I display here what we call the fuel salt circuit. (Naturally it came around) to a primary circuit, but it's where the fuel salt is located during power production.

Of course there are other circuits in the system: an intermediate circuit, and the core, of course, the fuel cell circuit, in an intermediate circuit also here, a thermal conversion circuit. And of course the draining of storage tanks, if we have to do something in the core without the fuel, and processing units located on site.

And right after these, by the definition of the system, right now the R&D activities need multidisciplinary expertise in reactor physics, chemistry, safety, design, materials, and we have collaborations in a national frame, for example, in France with CNRS in the IN2P3 Department, with universities, with AREVA, with IRSN, the CEO in France, EDF, and CDA, in the European frame, in the frame of the EVOL project, finished, and the current SAMOFAR project, I will speak on this, and on a worldwide frame, the frame of GIF, and recently with IAEA following a technical meeting organized last year and I hope with collaboration to continue.

Just to summarize the technical characteristics of the fuel salt circuit of the MSFR, I remind you of the produced power. With the temperature right in the core around 100°. And maybe something interesting, the negative feedback coefficient around -8 pcm/°C, ??? due to the density, ??? due to the Doppler effect. The core, which is very compact, as you can see, around 2m by 2m, it's a very compact reactor, which is a very good point, and 18m3 with height in the core, highest outside of the core to extract heat in the heat exchanger, et cetera, with a quite fast fuel salt circuit in the fuel circuit.

A very important point here, I need to mention that some MSFR design characteristics have a strong impact on reactor operation. The fuel is the coolant. There is no control rod foreseen in the core to drive it. And the reactor will be driven by the heat extraction, and off this, we craft a definitional assessment of normal operation procedures and of the safety approach dedicated to the MSFR, to a liquid circulating fuel reactor.

And what we are working on right now is the frame, not as a frame, but mainly in the frame, a collaborative frame with this SAMOFAR Project. It is the Assessment of a Molten Salt Fast Reactor, which is a project of Horizon2020. We are in the middle of the project. I listed here the partners on the project led by Delft and Prof. Kloosterman, with CNRS, in ??? JRC-ITU on the ???, our colleague from Italy, POLIMI and POLITO, in France, TSO, IRSN, AREVA, CEA, EDF, PSI, of course Paul Scherrer Institut in Switzerland, and an institute in Mexico working on structural materials, on materials.

And the idea of SAMOFAR is to deliver experimental and calculation proof of the following of some key safety features on the reactor. And these rely on five technical work packages: the first one, which I will detail later on, integral safety approach and system integration; the second one, dedicated to measurements of chemical and physical properties of fuel salt relating to safety; then to proof of concept of key safety features of the MSFR with two experiments on the natural degradation on the thermal properties of the salt; then one package dedicated to numerical calculation of transients; and finally, one package dedicated to the safety evaluation of the chemical processes and plant. This last point is very important because usually we are working on the safety of reactors, but in the MSFR, since the chemistry and the reactor are on site, on the same site, we are also working on the safety in the fuel processing, and this Work Package 5, driven by Dr. Delpech, is dedicated to the study of the fuel processing in relation with the safety issue.

So now I will concentrate on safety and just remind you of some very fundamental elements on nuclear safety because a nuclear reactor is an industrial facility, an industrial plant. In addition to the constraint of any industrial facility, we have some additional constraints or, let's say, specificities, due to the site as well of a nuclear reactor. So we have a huge energy reserve concentrated in the fuel which is the advantage of nuclear energy. We have an accumulation of, production of radioactive elements in the fuel, which are dangerous and that produce heat. And then, we have a large release of energy even after the reactor is shut down.

Then these, ??? is decline in terms of (safety rules), they need some nuclear safety to control the reactor and to control everything, and then we have three safety functions. We have a (safe) for every reactor, including the molten salt reaction. We have to evacuate heat even after the generation stops. It means the residual heat management. We have to control the chain reaction at any time, in any place on the reactor, so in the core or in the processing unit or the reactors, and we have to confine the radioactive elements at any time under any circumstances. That's why there are three confinement barriers in light water reactors for example.

And now, we have to foresee three safety functions with a molten salt reactor, which is quite different regarding its design. So the design aspects of the MSFR impacting its safety analysis.

First, the salt is liquid, so we have no core combusting is not a severe accident. Of course, the molten salt is also the coolant. We have relative uniform fuel irradiation but a significant part of the fissile inventory circulates out of the core, in the reactor, as I said, but out of the core.

The fuel processing and loading is done during reactor operation. We really can control the fuel composition at any time, which is very positive. Now there are no control walls in the core. The reactivity will be controlled by the heat transfer rates in the heat exchangers, the fuel salt feedback coefficient, which are very good, the continuous fissile adjustment, and by geometry of the fuel salt mass.

I'm sure there is no requirement for controlling the neutron flux shape, on the contrary, of a future reactor.

So it's possible to drain quickly and passively the fuel salt by draining for example, but also, we can do it actively, and the cold shutdown may be obtained. The best cold shutdown is to let the fuel salt in the core, but if we have to do something, maintenance in the core, it's possible to drain the molten salt from the fuel circuit to change its fuel geometry very easily to have a correct shutdown margin and cooling, it's very easy, and the fuel salt, it can be done passively by reactor operation in two dedicated systems, a normal operation storage system or an emergency draining system, which are under study.

And then to fulfill the three safety functions, we are taking into account these design characteristics, to redesign, to take the safety into account during the design definition of the reactor, the core and the draining system at least. We have to define the normal operation procedures. And then we have to evaluate the safety of the system once designed, accident initiators and the accident scenarios. And finally, regarding the safety approach, we have to define what is a severe accident first? Which kind and how many barriers do we have to have? What about the reactivity control in such a reactor, et cetera? These are the key points of the current R&D activities.

So if I consider the work package tasks that we are doing in the Work Package 1 of SAMOFAR, in charge of "Integral safety approach and system integration," the first step was the definition of good design taking into account the safety aspects.

So when you consider the LOLF accident, it's a loss of liquid fuel accident, there are no tools available right now for a quantitative analysis, but qualitatively, in the first time, as I remind you here, the fuel circuit was quite complicated, with pipes, multiple connections, and there was also potential leakage, so we can't put collectors. But also, what we do is the proposition of a new integrated MSFR design to suppress pipes and leaks.

So right now the core, the fuel circuit does not look like this, but it's much more, it's a vessel here, containing the fuel salt with the storage tanks around it, and this vessel contains the fuel salt, and then we put in this vessel some cooling device, a cooling system or sectors, composed of (bolts) and of a heat exchanger, and these are immersed in the vessel containing the fuel salt. There is no pipe taking the fuel in the ground and putting it in the heat exchanger, so no pipe, no more problem with the pipe.

In the same spirit, we already find the emergency draining system, which is now, so I display here the fuel circuit, and below it is the emergency draining system. It's a vessel right now containing a fuel salt after the draining of course, and in which we put cooling rods to cool the fuel salt in a ??? passively.

The emergency draining has to be triggered and achieved by redundant or reliable devices. It's a technological problem. We have to maintain the fuel salt in a passively safe situation for long periods, months or years, with materials resilient to high temperature using large volumes of fuel salt, avoiding criticality, of course it's the geometry and the composition, a problem of geometry and composition of the materials, and allowing the passive decay heat extraction in any circumstances.

The advantage of this system is that it gives a large grace period, which is not the case in the core because the fuel salt volume is reduced, but the draining can be very fast, and then the grace periods are very large in the draining system, so we can have very efficient cooling, of course fuel solidification maybe if necessary, and possible heating because the idea to recover the liquid fuel is not a ??? system. The idea is that we can recover the fuel from this emergency draining system to restart the reactor without any problem. And it's the basis of the definition of the emergency draining system that we are working on in the framework of SAMOFAR.

And the second point, remember, after design is defining the right operation procedure, so we are also working on this. I remind you that the MSFR characteristic requires the definition of normal operation procedures dedicated to the MSFR, which is a liquid circulating fuel reactor. So I take here the example of one normal operation mode is the load following because the MSFR, due to its liquid fuel, is designed to be very flexible in operation and to allow easy and long load following. The idea is to accomplish this load following without using any control rods, only by varying the power extracted from the core. And of course you have to keep the structure materials at a constant temperature so that we have several levers.

For example, we can play on the fuel salt circulation speed, on the intermediate fluid circulation speed, or on the temperature of the intermediate fluid in the heat exchangers by, for example, bypass or other technological devices.

And the idea is now to evaluate (??? is not following) exactly impacts the core. For that, we need precise transient calculations, at least a coupling of neutronic and thermal-hydraulics, and I will present you a tool developed for the MSFR, also used now for other reactors but developed for the MSFR and adapted for such a reactor.

And we are working now on the plant simulator, on system code developments, to more ??? study and design these, all the reactor operations, all the operations procedures of the MSFR.

Regarding transient calculations, as you know, the main ??? explaining in nuclear power is neutronics that heat the fuel and thermal-hydraulics to cool the fuel. And neutronics, due to the power production, ??? temperature, buoyancy effect, precursor production, and on the other side, thermal-hydraulics will give a temperature field and precursor of the decay position.

Why these second elements? Because we have a liquid fuel that is circulating with the precursor of delayed neutrons. It means that the delayed neutrons' precursors are not stable in the core, and they depend on the fuel circulation and the velocity field, et cetera. And the precursor can be produced in the core and delayed out of the core with no importance in the fission (chain). So it's very important to have an idea to have a code taking into account the motion of this precursor.

So it will give the delayed neutron source, which are mandatory to drive the reactor, the Doppler feedback effect, and the density feedback effect. And the idea is to couple, to have a code, which is both precise and quick in the calculations. Actually, it's very complicated, so what has been used here by Axel Laureau, it's the CFD model for (thermal-)hydraulics and the calculation is done in the code OpenFOAM, in the thermal-hydraulic code OpenFOAM.

And on the other side of neutronics, the idea is to have something precise and quick, and so the idea was to use SERPENT, for example, the neutronic Monte Carlo calculation code, the probabilistic neutronic calculation code. It was the SERPENT code. And to do calculation of the reactor and then to store the reactor behavior in fission matrices, so where neutrons produce fission. And then to have an idea of the time matrix. It means the time for 1 neutron to produce the fission in another part of the reactor. When you have the fission matrices and the time matrices, as displayed here, you use these kinetic equations to follow the evolution of the prompt neutron population and the delayed neutron population. And then this equation on the fission matrices are calculated once very precisely before the beginning of the transient, and then they are used very quickly, instantaneously. They are implemented in the OpenFOAM code and are used very quickly, deterministically during the transient calculation coupled with thermal-hydraulics.

And we can then do some calculations. For example, here it's the load following of 33% ???, which is quite a large load following, and we extract it, for example in red. Due to the cooling, the power, the criticality ??? in the core, you have on the left the margin to prompt criticality, you are very far from prompt criticality. And then the cooling, in tracing the reactivity, and then for the ???, what we put in the code is the dotted line is the extracted heat, and what is produced by the reactor, the produced heat, is the plain line, and you can see that the produced power exactly follows the extracted one.

Maybe we can have a look now at this effect in 3D in the video. So as you can see, at the bottom, you have the power on the left and the temperature on the right, and I think, you see the production of heat at the center of the core on the left, so you have cooling of the fuel, sorry, on the right, leading to a production of power ??? on the left. So these rely on 3D precise calculations. And then, what we show here is that the reactor is completely stable and therefore exactly on a very (flat slope) following ???. Thank you, Berta.

And so to conclude on these graphs, the ??? nuclear heat is deposited directly in the coolant. The produced power exactly follows the extracted power. The reactor is very flexible and very well adapted to load following. From neutronic and thermal-hydraulic issues, the load following is driven by the extracted power and no control rods are needed and this confirms the excellent load following capacities of the core.

So now we arrive to the safety part of the analysis.

And to do a safety evaluation we choose to mix the ISAM methodology of this combined with a systematic risk analysis.

So to start first with the systematic risk analysis. Thanks to that we have listed the preliminary main accident types for the MSFR, which are listed here. For the fuel circuit accidents, for example, loss of heat sink, loss of fuel flow, over-cooling, et cetera, reactivity anomalies accident. We have listed the accidents on the draining system and for design extension conditions. If I talk to you on reactivity anomalies accident, then we are calculating failure trees, et cetera. And finally up to the deterministic part of the calculation that what we will see now in the video.

So here I will present you a reactivity session of 100 ??? in one ???, which is the maximum that can be inserted in the reactor, and if we start again, you have the reactivity on the top, the power button, and then the temperature on the bottom right. And as you can see here, okay, the reactivity is inserted, and you have on the upper part on the right of the window, you have the power on the left and the temperature on the right, and you will see again, you insert reactivity, then we have an increase of the power on the left, right now, leading to a temperature increase on the right, and then once more ??? ??? are coming from the heat exchanger. And then everything is stable again, and the reactor is stable without anything, doing anything. Thank you, Berta, for this video.

As for the normal operations case, in this case we also see that we have very good behavior of the reactor to compensate here a very fast reactivity insertion. There are very important spatial effects in such a core. You have to have a good simulation tool, and there is, I have to mention now, we don't reach prompt critical regime up to (500) pcm inserted in 1s, which is completely impossible.

But I want to mention something. Parametric studies have been performed on overcooling up to prompt critical regime, and what's the effect on such a reactor is that even if we reach prompt criticality, there is no cliff-edge effect.

No sudden violent behavior observed on the MSFR even when the critical regime is reached. And then it's very important because the reactivity control is not that important. If we reach prompt criticality in the reactor, the behavior is still very, very good. And this is very important for the safety evaluation.

So the safety evaluation is going on right now, and combined with the ISAM methodology, we have written the safety MSFR white paper. We have exchanges with the research on safety working group since last year, and we are still doing this together and I am hope it will continue.

Right now, in this frame we are designing, for example, the number and the kind of barriers that are necessary to confine the fissile matter. We don't know how much of a barrier we need, how much, what they are, but it's under working in the PhD theses of Delphine Gérardin and Anna-Chiara Uggenti in the frame of the SAMOFAR project.

And finally, we want to develop a safety approach dedicated to a fast spectrum MSR with a circulating fuel, combining both deterministic and probabilistic approaches based on the current safety principles but adaptive to the MSFR characteristics to define severe accidents, the barriers, the practical elimination, et cetera.

And we want to build a reactor risk analysis model very flexible to identify the postulated initiated events, the hazards, the high risk scenarios, to estimate the and risk due to the residual heat and the radioactive inventory, to evaluate the barriers, and of course to have something flexible enough to build the design is changing, it's optimizing, and we want to ??? evolve with the design.

Finally, one small word on PR&PP. We are working, some work on proliferation issues right now, since some time, with a preliminary analysis of the MSFR proliferation resistance initiated by the GIF's PR&PP working group, Proliferation Resistance and Physical Protection Working Group. And our idea, the first idea, was the application of one part of the threat "fissile diversion," but we look at this whole MSFR system, the reactor and the processing units together. The idea was to see what are the risks and can we do something on the design to reduce such risk?

The conclusion, the preliminary conclusion right now, in the case of the fissile diversion is that it's not possible to take directly fuel salt because the radiation level is too high. So this may be possible, to take the ??? in the reprocessing unit and then to wait for the decay of ??? in uranium-233, and then we have to study the possibility of course with some levels, the 2.6 MeV radiation that can be detected very easily, the detection by mass balance, et cetera, so we are working on these different points.

And next, from processing resistance, we have to identify and to analyze other proliferation resistance threats related to the whole MSFR system, and we need interactions with the PR&PP Working Group in the GIF and with IAEA or with any people interested in this topic or any experts or industry persons interested in that. But we'll still keep interactions with the official working group on this topic.

And now I arrive at the last part of my presentation because where I speak of MSFR and people are serious that you may need the calculation on this kind of reactor, yes? And what is your dream? What is your demonstrator? And when and how can you build it? So I will infer that it's not one demonstrator. Of course, I hope there will be one, but there are many demonstration steps before. You have to consider the sizing for example, from the small size 1 liter to test basic chemical data, monitoring, et cetera, a medium size reactor and 100 liters to test hydrodynamics, noble fission product extraction, heat exchanges, and finally a full size experiment, 1m3, to validate the technology, the hydrodynamics, et cetera.

Then, you consider different levels of radio protection. You can use first an inactive simulant salt, no radioactivity, to do hydrodynamics, first measurements, model validation of the thermal-hydraulics, et cetera.

You can then have a low activity level, thorium or depleted uranium. Then it's still a standard laboratory, and you can do pyrochemistry, corrosion, and chemical and monitoring studies.

And then you have to go to the high activity level, which is a nuclear facility, for the fuel salt processing. There are operations, pyrochemistry, actinide recycling, et cetera. And then (issue taken).

The medium size with an inactive simulant salt, you have what we have here going on, this FFFER loop, the forced fluoride flow experiment, with ??? reprocessing the gases and particle extraction with 1/10th of the whole reactor with a simulant salt. And the idea is to test some technological aspects, ultrasonic measurement, the valve, by the valve cold plug to automatically drain the fuel, and level measurement, et cetera.

And the next step will be the SWATH facility in the SAMOFAR project to measure the thermal-hydraulic properties of the fuel salt.

And then each ??? full size experiment with a high activity level, it's what we call a ??? demonstrator. The power demonstrator on the MSFR, we do some preliminary studies on this, but it's the next step to be studied. Let's say for example 100 MWth reactor, with a volume reduced or power reduced by a factor of 30 or volume reduced by a factor of 10, we have 2m3 in the reactor, and a fuel temperature rise in the core is 30 degrees. And so show that to demonstrate the characteristic may be representative of the MSFR in terms of heat exchanges, of velocity of the fuel, of intermediate, et cetera.

And then, we can go maybe one step further because when we look at this power demonstrator, we can show that the amount of uranium to start it is around 650 kg. It's an under-breeder reactor, and there is no impact of the chemical processing on the reactor operations.

And then we add a fertile blanket for example in such a small reactor, and we obtain a power demonstration from a power demonstrator, maybe a small modular reactor MSFR or a small MSFR, which is a breeder, a small reactor, reduced reactor, with radial constraint on the reactor operations, and only it's a small modular breeder reactor, which is very, very small.

And then, just to finish, such a small modular reactor, S-MSFR, just at the beginning of the study, it's a very compact reactor that may be operated 30 years without chemical processing with the same fuel, only a salt control, maybe bubbling, which is very stable, and which, maybe, not in this configuration but we have to study it, it may be a breeder also.

So I list here some PhD theses in France on the MSR and they are all available at this horrible address or by typing "MSFR LPSC" in Google search. There are papers produced on CNRS and Reactor Physics and Safety.

And also I list here other MSR publications, some in the frame of the GIF MSR Steering Committee.

And finally, I thank you for your attention.

Oates: Thank you, Dr. Merle. If you have questions regarding any of the information in today's presentation, please feel free to type it into the Q&A chat pod and we'll take as many of those as we have time for.

While you're typing in questions, the upcoming presentations in the Gen IV International Forum series of presentations, in June, Lead Fast Reactor by Dr. Craig Smith. In July, we'll have a presentation on the Thorium Fuel Cycle by Dr. Franco Michel-Sendis. And in August, a presentation on the Metallic Fuel Cycle for SFRs by Dr. Steven Hayes.

At this time, I don't see a lot of questions coming in. Here we go. There's a little bit of time delay between the time the questions are entered and the time that they post. So Elsa, if you look on the Q&A pod, there are two tabs. When you scroll over, there's a presenter view and a participant view, and if you'll click on the presenter view you can see the questions coming in.

Merle: Okay, right now, I only see the questions from the ???. [*laughs*] But... I don't see any questions.

Oates: So roll over to the presenter view and click that. There's a question from Casper Sun. "Why fuel salt division...?" [*noise*]

Merle: I'll try to see that. Ah. Sorry. "Why fuel salt division...?" Sorry, I will try to find the fuel salt division part. Fuel salt division.

Okay, so why is it complicated to (steal) directly fuel salt? Because when you have the fuel salt you have many actinides in them. It's identical to take directly a fuel assembly, a used fuel assembly. It's complicated and impossible because of the content in terms of radioactive elements of the fuel salt. The radioactivity level will be too important and it will kill the people that try to take it directly. When you look at the radioactivity level, it's very, very, very important.

So the idea is to wait because the first step of the processing in the MSFR is to take all the heavy nuclides, all the transuranic elements on the fissile matter, and to redirect them directly immediately in the core. So the first step when you will extract fuel salt from the processing unit is to separate the heavy nuclides and to send the transuranic element on the uranium and plutonium, et cetera, and send these back to the core.

Then the salt after that is less radioactive, and the idea is, can we (steal) something somewhere? But the fuel salt as a whole is already very, very, very radioactive and it's impossible to take it directly.

You can say that maybe a state can organize something, but they are controlled by IAEA and it's not possible to take apart directly the fuel salt. You have to separate it and then try to (steal) the fissile matter or protecting it. I hope that answers the question.

So, sorry, I'll try. It's complicated because the window is quite small. "What is the estimated...?" So from Mr. Piette, "What is the estimated time to market for MSFR?"

If a state is interested in that, it can be efficient, and if we stay where we are right now, let's say, it can take some time. It can take a while. I think if you see it, what happens there were several world wars. Nuclear energy was developed very, very, very quickly, and then after that, when energy was required we developed reactors very, very quickly after that because there was a need from society. And if we put more money and more people on that, it may be fast.

And you can see that there is a demonstrator built in China, equivalent to the MSRE, graphite moderated MSR, and they are working very well and very fast. It's important for them, and I think they will succeed in let's say ten years or something like that to build a demonstrator or fertile thermal version of course the first time, but it depends on the people, the money you put in that, I think. And in some countries you also have, with no nuclear energy right now, you have to develop a nuclear safety authority and maybe there will be an adaptation of the nuclear safety to this new kind reactor. Maybe 20 years to have a final version, but it really depends on the political decision, let's say.

I wonder the color difference… Sorry, I'm reading… Oh. I don't know how I can read a long question. From Mr. Erbay, I cannot read the end of the question. I don't know. I can't see… Oh, maximize. Yes, it's okay. I don't understand the question. Sorry, maybe you can edit again. "I wonder the color difference at the numerical solution of the transient temperature change in the reactor." Sorry, I don't understand for right now.

Yes, then the question from Mr. Reim is, "on slide 4," if I may elaborate something on the graphite impact on the reactor. And so I come to slide 4.

So on slide 4, for me there is no… So maybe the question is in the neutron spectrum because, yes, here, we have a fast neutron spectrum.

Oates: Slide 6.

Merle: Oh, okay. Yes. So I don't really detail the graphite moderator. The problem is that we don't know right now how to manage graphite, nuclear, graphite in contact with nuclear radioactivity. We have some in the gas graphite reactor in some countries, and right now the management of the graphite is quite complicated because this graphite is contaminated with carbon-14 and chlorine-36 and it's very radioactive matter, and it's very complicated to manage it as waste.

And then regarding fire risk, graphite may have fire problems. In some cases, people met this problem in the graphite gas reactor also. Then having something that can be on fire in a reactor, in a real nuclear reactor, for me it's not satisfying, let's say. Regarding safety and regarding a possible source of dispersion, of radioactive matter dispersion, something that can take fire is for me not satisfying.

And then regarding the point on reprocessing, when you have a graphite moderator in the core, then you have a thermal spectrum, then you are very sensitive to fission products, and the MSDR in the reprocessing with some cubic meters per day, it's very, very heavy and very, very difficult to handle.

And in case of a problem in the reprocessing unit, it will have been mandatory to stop the reactor because the neutronic capacities of the reactor, they directly depend on the processing, which is an online reprocessing, and that safety relies on chemistry, on chemical operation. It's also not satisfying for us.

So Sven.

Oates: Do you also see the question on the radioactive waste? "Is there any radioactive waste inventory for this kind of reactor comparing to...?"

Merle: Mr. or Ms. Kicevic. Is there any radioactive waste inventory?

Yes, of course there is a waste inventory, but in fact when you extract the fission products, you will store them and it's waste. But you have two kinds of waste, waste produced during the reactor operations; in our case, it's only the fission products extracted during reactor operation out of the technological waste. So there is fission product produced on extraction, and then when you stop the reactor, you have the fissile inventory of the reactor, which is waste also.

In our case, what we intend is mainly to put this fuel, because it's completely controlled, in another reactor and to reuse it. But when you stop nuclear energy, when we stop a nuclear energy project one day, we will find a wonderful and better and massive energy production way, which does not exist for me right now, then we'll stop maybe nuclear energy, and then all the inventories of the reactor on the operations at that time will be waste. It's the way we are ??? as end of game scenarios to see can, we burn this in reactors dedicated to that. And there are also studies of equivalent (risk) for all the kinds of reactors.

The advantage here is we (already quit reactors) from putting large radioactive elements in them, like americium or things like that, it's easier in a liquid salt because you don't have to fabricate the pin, the assembly, et cetera. It's liquid and the fabrication is easier, so it's wider. So we think that it's easier for reducing the waste inventory at the end.

But of course, the fuel salt is our objective, our primary objective. And all the parts of the reactor in contact with the fuel salt will be our objective.

Oates: I'm going to post the question that you didn't understand. It was, "Thank you for the good presentation. I wonder the color difference at the numerical solution of the transient temperature change in the reactor," on the right.

Merle: Yes, from the colleague from Turkey. Yes.

Oates: Yes, and I'm not…

Merle: Do you understand the…? On the right of the…? Do you understand the question?

Oates: I didn't, but I posted it so that the person asking could perhaps type in clarification.

Merle: Let me ask. Can you please, if you are here until now, can you detail your question, please? So maybe I come to the next question.

Oates: "Are fission gases produced during operation, and if so, how are they treated?"

Merle: Sorry?

Oates: "Are fission gases produced during operation, and if so, how are they treated?" From Sven Bader.

Merle: I don't… Sorry. Ah.

Oates: On your right-hand side there should be a scroll bar that lets you keep going down.

Merle: Yes, yes, yes. So from Casper Sun, to read your question, I think. And Sven Bader. "Are fission gases…" Oh, sorry… Okay. So yes, fission gases are produced during reactor operation. It's why we have the bubbling. In fact, there was no bubbling for us in the MSRE. Bubbling occurs continuously due to the gas indicated in the pump and through the fission product, so if we don't have a bubbling system, it will be automatically in the reactor because of the production of fission gas.

So yes, fission gases are produced. They are extracted by the bubbles, and then we have a gas processing unit. For some of the gases we can wait for the decay in another element, for example, xenon in cesium. It can decay out of the core. It's no more gas after that. And so the remaining gases, including (tritium), the idea is to store them in a dedicated part of the plant. So, yes. The treatment is under definition. On evaluation, it's what we call the bubbling reprocessing unit.

I hope I answered the question.

Oates: And then it looks like the clarification on the question regarding the color, "in the solution there may be smooth color changes." And maybe preconditions can be evaluated. One notice.

Merle: In the solution. In fact, I don't really... Maybe it's regarding the... I don't know. Sorry. I propose that we can have interaction after all by email on this question because I don't understand. The solution is a numerical solution or is the fuel salt, but of course the boundary recognition has been surveyed. Yes, on the calculation ??? right now, ??? is available and we are doing some calculations from SAMOFAR WP4 when doing the new calculation of new transuranic states, and I hope this is the question.

So I have another question. What is...? Oh, okay, yes. So from Mr. Piette, "What is the material of the vessel?" So the material salt vessel is a nickel-based alloy. Right now the (aim) to show the ??? nickel-based alloy. I did not mention it, but it's basically an NSBR project relying on a new development from Russia and in France and also in China right now, materials are under study, and yes, it will be a nickel-based alloy.

So, "In the design with half of the fuel..." Sorry. So from Mr. Pázsit, "In the design with half of the fuel in the core and half of the fuel outside...," yes. The idea is, how far are we away from criticality out of the core since half of the fuel source is out of the core? And, "How well is the fuel in the heat exchanger discoupled neutronically in such a compact system?"

So we have neutronic protection between the core on the heat exchanger. The idea is that there are (brown) or (fertile) materials to avoid neutrons from the core from going to heat exchangers. And it has been calculated, so right now the source of neutrons from the fuel salt is higher in the heat exchangers as the neutrons come in from the core. So, yes, we are studying this.

And then, the risk of criticality out of the core has been studied, but the configuration is not optimized. The fuel is located in small parts or not so concentrated as in the core. So of course we have to (study) in case of leakage the risk of criticality out of the core, but if we have leakages, small leakages, then the fuel salt will freeze because it will be freezing at normal temperature in a room. And if it's a large leakage, it will have a configuration very, we have to study of course what is the possibility, but the idea is to have a configuration which is far away from critical. But in the heat exchanger, there is no problem with that because of the material from everything. So yes, we are studying, we are calculating this.

And also, we are calculating the risk of criticality in the processing unit, which is very important as well. It’s a topic of WP5 of SAMOFAR. I hope that answered the question.

Oates: And there's a follow-up on the fission gas question, "...over the expected lifetime of the plant approximately what quantity of fission gas is decay stored?"

Merle: I think I answered to the fission gas. Ah, no. Okay. The fission gas question... Oh, okay. The idea is not to store all the fission gas; it is not to keep all the fission gas on the site. So the idea is to put them in dedicated containment and then to store them, but not on site during the whole life of the plant. But of course they will decay. I don't have the exact number, but yes, there is a lot of decay heat in this part, and what I did not mention is that when we study the safety of the system, we have to study, it's a very good point that the fission products are extracted out of the core during reactor operation because you have, you reduce the (??? source) in the core, but of course your have to study the safety and the confinement in the processing unit. Yes, and I fully agree.

We can study that in parallel, but we have to study. It's why we study the safety not only for the reactor but also for the processing unit. And it's under study.

Oates: "Is the low heat capacity of molten salt a challenge? And if yes, how it is addressed regarding the design?"

Merle: So the low heat capacity... Low heat capacity. I don't really understand because the molten salt... Oh, you mention the capacity, yes, to exchange heat. So, yes. Molten salt is not... Yes, it's very useful to transport heat, and to cool heat you have to, yes, you have to design heat exchangers to have very good, to maximize exchanges with an intermediate coolant. And then, yes, it does impact the design of the heat exchangers, which is a key point of the MSR, of the fuel circuit.

We have to have small channels of small plates of pure salt heat exchangers to extract heat. Yes, right. But it's under design and we are working on that with people in Milano and people in ??? and some people, yes. But, yes, I'll confirm on hearing at CNRS here. But yes, the heat extraction is, the characteristic of salt has to be taken into account.

And there... I don't know if there are further questions, Berthe? I think this... This one? I find really wonderful to (avoid) this question in the interests of... Maybe there are other questions, but up to now, thank you very much for the questions and please do not hesitate to contact me or SAMOFAR because the idea is really to open the community and to have exchanges with experts in materials, in ???, et cetera, in safety, et cetera. So we are very interested.

Oates: Thank you. Thank you very much, Elsa, for your presentation. I know it takes a bit of time and energy to put these together and the information is very well received, as you can tell from the accolades that you got even during the questions. We appreciate it very much.

Merle: Thank you very, very much, Berthe, because your help was very, very important.

Oates: There was a follow-up question. "Do you have an idea of the burn-up MSFR could reach with an initial load of TRU+Thorium without any reprocessing, neglecting other problems?"

Merles: So the burn-up is a question of classic for a molten salt reactor because you don't load and unload the fuel, and the fuel is all the time in the reactor during the whole time of the plant, so you inject some small amount of fuel, you don't extract the spent fuel. Really the operation is continuous. And defining the burn-up for a solid fuel reactor is very complicated.

So as for the safety aspect, we have maybe to define what... A direct equivalent to a burn-up is not easy to calculate for MSFR because you never have spent fuel. So if you use the fuel in another reactor and another reactor and another reactor, I think you can use it completely, but I don't know if that's clear.

Oates: Thank you. Thank you very much.

Merle: Ah, there is... Are there outage schedules? Oh. Yes. Yes, so what I did not mention is, yes, we can have a plant failure or a hole in the heat exchanger during reactor operations of the MSFR, and then we have to change elements in the fuel circuit and the IED. It's why we have cooling sectors. The question is, "Are there outages schedule in the normal operation of MSFR?"

So if you have problems on elements to replace in the core of the reactor, then we will have... It's the way we have a cooling sector. The idea is to replace a whole cooling sector if there is a problem with one. If we have a pump failure, then we take the cooling sector corresponding to the failure and we replace it. We have 16 cooling sectors, there are 16 cooling sectors in the reactor. If we have a failure in one of the sectors, it's possible to operate the reactor, to stop it slowly, and then to change something by replacing a sector, a complete sector.

But the idea is to operate the reactor as long as possible without stopping it of course, but it's possible to change at least quickly. They need to change the parts quickly.

Oates: Thank you.

Merle: Thank you.

Oates: One last item to note. Today's presentation was recorded. The recording will be presented on the GIF webinar website. Just give us a few days to get that link posted so you can go back and review the presentation at your leisure. And after this presentation, we are going to try something new to the Gen IV webinars, and that is that we are planning to queue up some certificates of attendance for the people who participate in the live presentations, so watch your in-boxes for those to distribute.

With that, again, thank you very much, Elsa, for your presentation. Amanda, thank you as always for running behind the scenes there keeping things flowing. And I thank everyone else for their attendance and participation and great questions.

Merles: Also, thank you very much Berthe and Patricia and (John) for the organization and ??? for the organization, and thank you to everybody for finding here.

(John): And just to remind everyone to take this survey. We really appreciate the feedback we get from them. You can see the link there.

Oates: Okay. Have a great day.

Merle: Bye-bye, thank you.

Oates: Bye-bye.