# European Sodium Fast Reactor: An Introduction
# Dr. Konstantin Mikityuk, PSI, Switzerland

**Berta Oates**
Good morning. Welcome everyone to the next Gen IV International Forum Webinar Presentation. Today's presentation is on the 'European Sodium Fast Reactor: An Introduction' by Dr. Konstantine Mikityuk. Before we get started I would like to go through. The first is that the audio is through your computer speaker. Please be sure that they are unmuted, select the 'mic and speakers' radio button on the right hand side, audio and pane display. If you do experience any technical difficulties today, please contact the GoToWebinars Help Desk on the number shown there at 888-259-3826.

Because all of the attendees are muted, the questions will be taken today through the questions pane on your screen. Type in your question. I apologize for that typographical error there. Please share today's presentation with others or watch it again. It is being recorded. The audio-video recording of the webinar and the slide decks will be made available at the Gen IV International website at www.gen-4.org. The slide deck is also available from your viewscreen as a handout. Please take the survey. The link is listed there. It will also be there in the email including today's presentation.

With that, I would like to get started by introducing Patricia Paviet to introduce today's presenter. Dr. Patricia Paviet is the Technical Group Manager of the Radiological Materials Group in the Nuclear Sciences Division at the Pacific Northwest National Laboratory. She is also the co-Chair of the Gen IV International Forum Education and Training Taskforce. Patricia?

**Patricia Paviet**
Thank you so much Berta for the introduction. Good morning everyone. It's really a great pleasure today to have our presenter, Dr. Konstantine Mikityuk, because he is the co-chair of the Gen-IV International Forum Education and Training Taskforce. And for 3 years, Konstantine and I and all the other members of this taskforce are working together to give you these webinars.

Since graduating from Moscow Engineering Physics Institute in 1992, Dr. Mikityuk has been involved in research of safety-related aspects of various nuclear reactors with fast neutron spectrum: first, at the Russian Research Center Kurchatov Institute and then at the Paul Scherrer Institute.

His current interests are safety analysis of sodium-cooled fast reactor, in particular neutronics and thermal-hydraulic aspects of sodium boiling.

Konstantine is a Group Leader at PSI. He is maître d'enseignement et de recherché at Ecole Polytechnique Federale de Lausanne. He is also a lecturer at the Eidgenössische Technische Hochschule Zürich. He is also the coordinator of the Horizon-2020 European Sodium Fast Reactor-Safety Measures Assessments and Research Tool, the ESFR-SMART project. Without any further delay Konstantine, I give you the floor. Thank you again so much for volunteering to give this GIF webinar. Thank you Konstantine.

**Konstantine Mikityuk**
Thank you very much Berta. Thank you very much Patricia for your kind introduction. First of all, I would like to thank also Generation IV International Forum for giving me this opportunity to make this webinar. It's a great pleasure and honor for me to deliver you this talk. And I in particular also would like to thank Dr. John Kelly [ph] for his support and help in preparing this webinar.

So, welcome everybody. You know the topic of the webinar. I would like to start with an outlook of this plan. I organized it in terms of goals. So what I would like to achieve today during this 1 hour. So first of all, I would like to present you a brief history of the conceptual development of a large-power (3600 megawatt thermal) European Sodium Fast Reactor. Then for the rest of my talk I would like to give you some detail, some introduction into the Horizon-2020 ESFR-SMART project mentioned by Patricia already.

So, first, I would like to discuss with you the start of the content of activities of this project focused on Generation IV ESFR Safety Enhancement. And then, I would like to provide you an overview of New Safety Measures proposed for improvement of the three safety functions: reactivity control, heat removal, and radioactivity containment. This is the first part of our project. We propose new modifications, new ideas for the design. And then, the fourth goal is to introduce you experimental programs which are currently ongoing in Europe in support of the Sodium Fast Reactor Research and Development which makes the second part of our project. I will conclude with summary of the activities which we plan to perform during the next phase of the project.

So, let us start. There is some delay with the reaction of the slides. We start with the first goal. Before presenting you the history of European Sodium Fast Reactor, I would like to say a few words about concept of sodium fast reactor in general. Here you can see a conceptual diagram of this system. As in any nuclear reactor, thermal power in this system is generated in the core, which you can see here, due to fission of heavy metal nuclei by neutrons of high energy which is also called fast neutrons. For this reason, the reactor is called fast reactor. The fission chain reaction is controlled by inserting in the core the rods absorbing neutrons. You can see them here. The core is located in the pool of liquid sodium at about 500 degree Centigrade. This sodium circulates through the core where temperature is growing up and heat exchanger where temperature is reducing and the heat is removed to the secondary loop, to the secondary coolant. Circulation occurs due to primary pumps. I already said that there is a secondary loop. This is secondary sodium loop, which circulates with the use of secondary pump through the heat exchanger located in the primary pool of sodium and through the steam generator where steam is generated. This steam drives turbines which generates electric power. So, the goal of this system is to generate power indeed.

The focus of our project, our concern, we mainly focus on safety aspects, so how to make this reactor safer? We systematize, we categorize the safety performance in terms of basic safety functions. When we develop design, we try to guarantee to demonstrate that these safety functions are fulfilled in every reactor state. These are three functions. The first is control of reactivity, so the chain reaction should always be under our control. The second is removal of heat from the fuel. So, we would like to demonstrate to guarantee that the heat is removed at acceptable temperatures, so there is no overheating. The third is confinement of radioactive materials. Again, all radioactivity should be contained inside the primary system, inside the primary pool.

Below, we will use the following symbols to remind you about the three safety functions. It's 1, 2, 3 in different colors. Here I have a footnote which is very important to remember. Even if we control reactivity, even if we shut down the reactor, it means we stop the chain reaction, there is always heat generation because the fuel contains radioactive isotopes which decay, and this decay generates heat. We call it residual heat or decay heat. In particular, when we talk about safety function 2, we will have in mind often the decay heat removal, so even in the case when chain reaction has stopped.

A few important points related to advantages and challenges very briefly because you can refer also to Webinar 4 given by Dr. Hill from Argonne National Laboratory about sodium-cooled fast reactor where it's much more detailed. I would like to mention a few advantages of using sodium in particular, but in general sodium fast reactor. First of all, we should remember that in fast reactor due to fast neutron spectrum we have potential for breeding, for generating new fissile fuel. The second, we should remember about excellent thermal conductivity of sodium. This is the best thermal conductivity among available metals. For this reason we have very efficient cooling which helps us to fulfill safety function too.

The third advantage is large margin to boiling. This is also very important because we don't need pressurization. So, in our design, the primary pool remains at atmospheric pressure of gas above free sodium level. This is a big advantage for safety but in particular to safety function 3 which is confinement of radioactive materials. We don't pressurize, we don't have inside additional potential energy which can be dangerous for preservation of the boundaries of the reactor pool.

Four advantages I mentioned here. For challenges, for disadvantages, for problems, I listed here two items which are most important in my opinion. The first one is sodium is chemically active in contact with water or air. We should have in our design the measures which prevents contact of primary sodium, which potentially can be radioactive with water or with air. For this reason, we introduced intermediate circuit which I mentioned before in order not to have in the reactor pool steam generator. Because steam generator in the reactor pool would be potential danger in case of steam generator tube rupture, potential danger of contact of water, and primary sodium. This also contributes to safety function 3.

The second challenge relates to neutronics, the sodium from viewpoint of neutrons it has significant neutrons scattered in cross section. It means that the presence of sodium makes a spectrum softer, shifted towards thermal energies. And correspondingly, when we remove for some reason sodium, the spectrum becomes harder. We observe spectrum hardening when sodium is removed. Spectrum hardening because of cross section nature, when fast reactor becomes faster it means positive reactivity field. It means the power will grow up. For this reason, special safety measure is needed in order to fulfill safety function 1 which is controlled reactivity. We will discuss it. There are several reactors under operation, several sodium fast reactors under operation in the world, three observed in Russia: BOR-60, BN-600, and BN-800; and one in China, Chinese Experimental Fast Reactor, CEFR.

This slide shows a brief history of European Sodium Fast Reactor development. In terms of the plot it's thermal power of reactor versus time of startup. I put on this plot both existing reactors. So, I put here reactors existing in Western Europe and the three programs. UK program is DFR and PFR. You can see approximately the time of startup. I don't show the shutdown but all these reactors are already shut down. The second program is a German one with KNK-II and SNR-300. The third program, the biggest is French with Rapsodie followed up by Phenix, followed up by Superphenix. Currently under development and currently under reevaluation is the ASTRID program. This is a program on creation of new sodium fast reactor at the place of formal Phenix reactor. So, I put it in the future like 2030 but as you know, it's difficult to predict more or less exactly at this time.

What I would like to emphasize here with the top line is that in parallel to real designing and building an operational field reactor, in Europe there was a line of R&D projects of large power of 3600 megawatt thermal power. This line started with Superphenix 2 developed during the preparation of Superphenix during startup in the timeframe from 1980 to 1990 roughly. I showed it in green because it was French program. But in 1990, this program was followed up by now real European project which was called European Fast Reactor. The timespan was about 10 years, started in 1990 and finished in 2000. And then there was a gap. And then a new frame, in Framework Program 7 we started a 4-year project called Collaborative Project European Sodium Fast Reactor CP ESFR. So, from 2008 to 2012. And again we have a 5-year break and in 2017 we started a follow-up project called ESFR-SMART, which is a subject of this webinar.

The main goals of all these projects were partly similar to Generation IV system's goals. We think about improving safety, improving economics, and improving management of nuclear materials. So, I've put here another symbol for economics because I would like to emphasize it in a couple of places what we are doing in ESFR-SMART in order to have in mind improvement of economics. We don't have this as a goal, as a work package evaluation of economics. But we have in mind that simplification is good not only for safety which we believe the case, but also in economics. I have shown here the diagrams of three systems, the timeframes, and to show that this was aligned, this was development and every time we used the achievements of previous projects.

The design of this European Sodium Fast Reactor has been developed step by step, and of course taking into account SFR operation experience and multiple experiments. The most general parameters are listed here, like thermal power of 3600 megawatt, electric power of 1500 megawatt, mass of sodium in the primary pool at about 2500 tons, primary sodium temperature changing from 395 to 545 degrees centigrade, which is exactly the value at which Superphenix fast reactor was operated. In all these designs, we had six heat exchangers, three primary pumps, and 36 steam generators.

So, this was a very brief introduction into history and concept of European Sodium Fast Reactor. Now we continue with discussion of status of the current R&D activities and status and content of activities in the frame of ESFR-SMART project. I would like to make a general introduction. This project is focused on safety enhancement of ESFR.

First of all, we selected a phoenix bird as a symbol, as a logo of our project in order to emphasize the link with the previous project like previous reactors, sodium fast reactors like Phenix and Superphenix. But also, we promote the symbol of phoenix as a logo for fast reactor in general because this fairytale bird after living hundreds of years, it burns itself, and then recreate, reborn from its own ash. This reminds us about recreation of new fissile fuel from fertile fuel in fast reactor.

The abbreviation ESFR-SMART as already mentioned is European Sodium Fast Reactor Safety Measures Assessment and Research Tools. In the title already we have two directions, two main parts, and two main general goals which I formulated here. First of all, we would like to select and assess innovative safety measures for European Sodium Fast Reactor concept. Second, we would like to develop new research tools related to SFR safety, more general, not only to European Sodium Fast Reactor but to all sodium fast reactors which are interested in these research tools like ASTRID program or Chinese program or Japanese program or Indian program, or Russian program. Under new research tools, we mean calculational codes, we mean new experimental data, and we mean sodium facilities.

The budget of project is €5 million of Euratom contribution plus about €5 million of consortium’s own contribution. Timeframe is from September 2017 to September 2021. This is a four-year project.

We managed to create very strong consortium. We have 19 partners from 9 countries. And on this chart I tried to show the logos of all partners but also I tried to make groups which shows one of several expertise. For example, we have some organizations which possess experience in past SFR safety related tests like IRSN here or KIT. We have organizations with experience in SFR license like IRSN and GRS which is technical and support organization supporting safety authorities. We have universities which have experience in SFR safety-related education. We have industrial partners like framatome, EDF wood, with experienced in past SFR operation. We have few organizations which are running SFR safety related tests right now like KIT, PSI, or University of Lorraine.

We have experience in sodium facilities and instrumentation, or in fuel measurements. A majority of our partners are at the same time members of IRDECO [ph] platform which is Industry Research and Development European Cooperation. We cooperate also and we exchange information in frame of R&D activities related to ASTRID.

Any project – and I would like to emphasize this that our project is first of all people. In one project mailing list I have almost 150 persons. This is a very strong team. We have all the leading experts in SFR area in Europe in this team. This makes us really strong. Here I listed key persons and task leaders. You can see many names I am sure you know and some of these people already presented you webinars. I would like to emphasize and mention one person, which is Dr. Joel Guidez from CEA who was in charge of Phenix reactor, who was participating in Superphenix operation and was very active in EFR project.

Dr. Guidez is serving like a bridge between this previous experience, previous project and the future. We appreciate very much his coordination and his advice. You can also download and listen to his webinar about experience from Phenix and Superphenix. This is webinar 15.

The structure of the project, we organized work in three subprojects. In total we have 12 work packages and 47 tasks. I showed here these subprojects in different colors. First two subprojects corresponds to the main goals I presented to you. Subproject 1 shown in blue is devoted to analytical assessment of new safety measures for ESFR. We use as input the results of the previous Framework Program 7 projects of Generation IV International Forum activities and of ASTRID R&D cooperation. We plan to provide this as a result, a new ESFR concept which is simpler, safer, and cheaper.

The second subproject devoted to research and development to support ESFR safety enhancement, more generally we consider the legacy data in which we collect as inputs to our work. We plan to provide the result, validated codes and new data and new facilities. Subproject 2, 3 is management and interaction. It takes care of dissemination, education, training, and management of the project.

I would like to give you a little bit more information about the structure. So, in particular, subproject 1, it includes five work packages. The first work package is devoted to new safety measures. So in this work package during the first year, so this is now almost finished. So, during the first year we made lot of efforts in order to produce proposals for new safety measures which will be evaluated at the next phases. We have first of all task here on definition of safety requirements taking into account already our ideas and taking into account Generation IV International Forum methodologies. And then, we have two tasks which focus on specification of new core safety measures and of new system safety measures. We also have tasks which continue working during all 4 years which will monitor consistency of R&D studies in the whole subproject.

We proposed the new safety measures for ESFR which I will present you today. The rest, the four remaining work packages will focus on assessment of how these new safety measures will impact operation of reactor in different states? Here we used concept of defense-in-depth. We follow the levels of defense-in-depth in organization of our work. You probably know defense-in-depth is the arrangement of defensive lines so that they can defend each other. This concept was used, for example, in medieval castles design. It is used currently in information protection concepts and also in development of nuclear power plant designs.

First of all, we have work package 1.2, which works in normal operation conditions. We evaluate our new design for normal operation. We look at initial core performance and burn-up. We calculate safety and performance parameters at the end of cycle. We look at fuel performance and we produce the recommendation how to calculate good conductance for the fuel rods. We work on coupled some hydraulic and neutronic simulations, everything normal operation. Here we produce safety coefficient cross-sections which is used in all other work packages.

Work package 3 is focusing on the next defense-in depth level and we call it assessment of measures to prevent sodium boiling. Basically we model, we simulate here, we think about protected loss of flow transient, protected loss of offsite power. We work with decay heat level and we assess transition from forced convection to natural convection. We assess primary pump's performance, Decay Heat Removal system performance, performance of passive quotient on systems, so everything before sodium boiling.

In the next work package we look at the next defense-in depth level and we evaluate measures to prevent severe accidents, how they perform in particular for the new core design which I will present to you in a moment. We expect a new phenomenology, new behavior in unprotected transients. In transients when they still have chain reaction. We have, for example, the stop of primary pumps. This is a very severe situation. But we believe that our new safety measures, our new core design allows us to avoid severe consequences like core meltdown. So, we would like to study the behavior of special conditions of sodium boiling due to use of this innovative core design. We think about assessing the pressure waves and how they propagate through the core and how these pressure waves can impact the reactivity due to change of the geometry of the core which is the last task focusing on dynamic reactivity effect of pressure waves.

Okay, the last defense-in depth level, when all other levels failed for some reason, variable probability event. We assess measures to mitigate severe accidents. Here we organize work in terms of the phases of severe accidents. We look at transition phase, at expansion phase, and we will analyze introduction of absorber into the core to mitigate the consequences. This is the structure to give you overview of what we did, what we are doing currently and what we plan to do in subproject 1.

The next slide shows you our plans for subproject 2, which is also divided in five work packages. Here we perform work. We support research and development activities for SFR safety enhancement in general. So first of all, in work package 1 we collected legacy data, so the experiments and data from the past. We work on calibration and validation of our codes which are used for SFR safety. So, in particular, we used the Superphenix startup static and transient data for validating core static neutronics and operational transience codes. In the second task, we work with data on sodium boiling and transitional convection flows. In the third task, on the molten fuel interaction and molten fuel behavior. And this validation work is full and work started in FP7 JASMIN project. Finally, we also used data to validate codes calculating source term.

In the second work package, we performed new experiments for safety. The first task is focused on support of tests at KASOLA loop in Karlsruhe Institute of Technology. We would like to conduct experiment on transitional conditions, transition from forced convection to natural convection. In the second task, we are doing some model experiments to study sodium boiling phenomenon. Also, we plan small real sodium boiling tests. In the third task, we look at corium jet impingement study. And finally we make experiments on the corium behavior and core catcher. I will give you examples in the second part of my webinar.

Work package 3 is focusing on the support of European sodium facilities. This is important for us to identify the places where we have sodium loops in Europe, and to provide them both financial support but also support in the form of student mobility. As the last task in this list, we have the grant program to support mobility of students to let them perform part of their work at sodium loops in Europe. We also support here the development of design guidelines for sodium loops of procedures and standards for testing selected elements of sodium technology. We review the worldwide technologies here.

In subproject 2.4, which is called Instrumentation for Safety, we decided to focus on just one device, one measurement technique which is the Eddy Current flowmeter. And we continue activities here started in FP7-ICT Plus project. This flowmeter will be qualified by means of model experiments, will be checked or validated or tested in mockups under sodium conditions. Finally, we plan some tests in cooperation with KASOLA team to test this measurement technique, this flowmeter at high temperature in real sodium conditions.

The main idea is to create a very compact device which allows us to measure flow rate of sodium at the outlet of every subassembly. This will be a very great step forward in terms of monitoring the behavior of SFR from viewpoint of safety and first of all for detection of blockage of subassemblies.

Last but not least, and subproject 2 is work package 5, which is devoted to new measurements of fuel properties. Again, we continue the previous work started in ESNI [ph] Plus. The idea is to organize the transportation of burnt and fresh fuel samples irradiated in a real fast reactor to laboratory and to obtain new point for some more physical properties, first of all, thermal conductivity but not only.

So, we work on preparation of experimental programs. We already performed transportation of both fresh fuel samples and burnt fuel samples. We start with fuel characterization and measurements of fuel properties. Our goal is to issue the new catalog of fast reactor MOX fuel properties.

This was a brief presentation of content and status of R&D activities in our project ESFR-SMART. We look now closer to the content to achievements of SP1. So, I will provide you briefly an overview of new safety measures proposed for improvement of three safety functions. We already discussed reactivity control, heat removal, and radioactivity containment.

This is the overall view of new European Sodium Fast Reactor. You can see here the primary sodium pool which is roughly 20-meter high, 20-meter in diameter with a core inside with three pumps and six heat exchangers. Every heat exchanger you can see in the yellow here is connected to one secondary loop. You can see here in the red and blue the hot leg and cold leg, the pump, the sodium receiver. This box contains six steam generators. I will show you on the next slide. You also can see these tall chimneys which we designed to promote natural circulation of atmospheric air to remove the decay heat.

So, this is the same design but with some more open components, like we opened here the reactor pit to show you the core. To show you above construction with control rod drive lines, heat exchangers in yellow, the pumps in blue. You can see the corium catcher device in green. You can see insulation in dark green, the gap between insulation and main vessel in blue. You can also see six steam generators. So we removed here the box, the containment. You can notice that we have these windows in these boxes. The idea here is that we can open these windows and organize natural circulation through these boxes in order to remove decay heat from the surface of six steam generators, modular steam generators. So, I opened here the chimney to show you the sodium air heat exchanger. And also, there is a window which is opening to organize the circulation. This is an additional part of Decay Heat Removal.

Here I show as a zoom, a design of core region. I would like to list here briefly the main innovations which we propose. First of all, this is low-void effect core which I already mentioned. I will give you details on the next slide. The idea of new designs is to improve the safety function 1, control of reactivity. For the same goal, we designed the new control rods designed for control rod driveline, which is passively activated due to curie temperature. We designed in our spent fuel storage for 50% of core loading to improve the performance of safety function 3. So, we keep the spent fuel assembly. We don't remove it from the reactor but we put it in intermediate, in inner storage, and keep it here for another 3 years to cool down and to reduce decay heat power level. We also created in the core many holes to make it similar to Swiss cheese. We called these holes corium discharge channels. This makes our core in a way more transparent for the very low probability case of core meltdown. We created these holes to facilitate the discharge, the movement of this liquid Corium down towards core catcher where we would like to collect the molten part of the core in order to organize efficient cooling, passive cooling.

We also consider in our design hydraulic diodes at the pump outlet somewhere here. The goal is to reduce the backflow in the pump when it stops for some reason, just one pump. And if you don't have this device, the other two pumps will start to pump primary sodium into the failed pump. This is undesirable situation because it will mean bypass of the core and loss of flow through the core. To avoid or to reduce this effect we consider special device which gives normal path for sodium in forward flow and which in a way prevents or creates hydraulic resistance for backflow.

A little bit more details about new core design. You probably heard about this low void effect design. Here I showed the axial map of the core. It's vertical cut, and in different colors you can see different materials like orange is fissile fuel. We decided for simplicity to have initial fuel with the same plutonium content. It's around 18 mass percent. You can notice that in the inner region, the height of fissile fuel is smaller and in the outer region it's bigger. In this way we try to flatten the radial power profile, in order to obtain the more flat, more uniform outlet coolant temperature distribution.

In green you can see the regions of breeding. This is called fertile blanket. In this region in particular, the new fissile, new plutonium is generated. There is also a steel blanket. Below there is fission gas plane, a dedicated place where we collect the gaseous fission products generated during the operation. There is an inner spent fuel storage you can see here on the periphery. You can see the holes which is corium discharge tubes. They are open now. We analyze the design of the thermal hydraulics but the idea I already mentioned, to facilitate the corium discharge.

The last but not least design feature is sodium plenum shown here in gray above the fuel. This has in a way empty space inside every subassembly field only with sodium. This layer of liquid sodium in normal operation, this layer above fuel reflects neutrons down. It serves as a liquid reflector. So, neutron born in fuel goes up, reflects, and goes down. It again can cause chain reaction. It's not lost. But in situation when we have sodium boiling, the void is generated at the top of the fuel where sodium is hottest and propagates in the sodium plenum. And in this way leakage sodium disappears from this effective reflector layer. And we don't have liquid reflector anymore and neutrons born in fuel fly up and absorbed in the [Unclear] which is strong absorber.

So, in this way, we reduce the reflecting back of the neutrons and we introduce negative reactivity. So, by boiling passively we remove reflector and we efficiently insert absorbent as a core in a way. This lets us obtain close to zero global reactivity effect and it allows us to obtain negative reactivity effect during realistic scenario of sodium boiling.

We can look at also the radial map from above. We see perfectly symmetric core with six batches. There are 6-year fuel cycles. Every year one-sixth of the core is moved from the operating region to the inner storage. This is fuel assemblies of one color. The empty space is filled with fresh fuel. This is called mixed scheme because there is no reshuffling. We don't change the position of the assembly inside the core. Every assembly stays on its place for 6 years and then moves to inner storage where it is cooled for 3 years and then it's removed to external storage outside the reactor.

We can host 50% of the core in internal storage, which helps us to improve safety function 3. I already mentioned that Curie-point lock at passively activated safety rods and corium discharge channels which contributes to all three safety functions.

So, we can look a bit closer to passive control rods. Here I give you illustration. I don't have the European design, so I found a similar design developed by JAEA, a Japanese organization. They call it SASS concept, standing for Self Activated Shutdown System. It's based on two parts installed on the control rod driveline. Here is control rod driveline. And then it continues here and here somewhere we have the absorber rod. The two parts of this device are kept together due to balance between magnetic force and the gravity. Magnetic force is created by the coil current shown here. The feature is this red layer which is temperature sensitive alloy written here.

The idea is that at the increase of the outlet sodium temperature which flows here, when temperature is increasing then temperature of this red alloy is increasing. At some moment, at some point called Curie temperature, Curie point, the magnetic properties of this alloy is disappearing. And then the balance between magnetic force and gravity is changing in favor of gravity. The device is falling down and the absorber is inserted in the core passively.

Therefore, activation of absorber insertion or we call it scram are provided both in response to the scram signal.  When we switch off the current in the coil, then the magnetic force is disappearing or the off-normal core conditions when we have rise of the temperature.  And at Curie point the temperature-sensitive alloy lost its magnetic properties and it has the same effect of introduction of the absorber in the core.

Again, about the path for corium, just to illustrate that in case of very low probability core meltdown event the corium discharge channel helps to avoid re-criticality because re-criticality can be created when we have big, dense volume of molten core, for example, to promote transfer of the corium to the core catcher.  So in this way we have this melt in the place which is designed to host it and designed to withstand this very aggressive high temperature material.  So, it contributes to the control containment of radioactivity.  And finally, to efficiently remove decay heat because again it is designed so that sodium can circulate by natural convection and remove the heat generated in corium to Decay Heat Removal systems.

So, a few words about Decay Heat Removal system.  We designed three new systems to improve the second function which is removal of heat, in particular decay heat.  The first Decay Heat Removal System, DHRS-1 is connected to the IHX.  It's connected to secondary sodium.  It's not the separated loop with its own sodium circuit but it is used in secondary sodium.  It allows us to simplify the design because we don't need additional purification system, for example.

Another advantage is that we designed it in such a way that even when we completely lose the secondary system, we still have DHRS-1 available. So natural circulation of secondary sodium and the exchange with atmospheric air promoted by the tall chimney which I already mentioned.

At the same time, if we have secondary system and feed water in steam generators available, first of all we rely on normal task of heat removal in normal operation in order to remove decay heat also.  If we lose the feed water, then we still have possibility to organize atmospheric air circulation through the steam generator containment, I already mentioned, by opening the windows and allowing air to circulate.  So we base this idea on the experiment conducted during end-of-life test at the Phenix and it was a very successful experiment.

Now we are studying how it operates because we have much higher power here.  We also consider the concept of passive thermal pumps which is shown here.  It's a kind of electromagnetic pump but it uses permanent magnet and it uses electric current which is generated by the difference of temperatures.  So, we have kind of more or less thermal couple which generates electric current due to difference between sodium temperature and environment temperature. We don't expect very strong head but we believe that this small pump can contribute to natural circulation and can help us to establish good natural circulation. For this reason, we consider, we analyze these devices both in secondary loop and in Decay Heat Removal System loop.

A few important innovations for reactor pit and for Decay Heat Removal System 3. So, we spent some efforts in order to redesign the structures of the primary system, in particular reactor roof and the reactor pit. The goal was to eliminate reactor dome. This dome above primary system was used. In Superphenix it was additional cost. It made more difficult to operate the reactor when everything is closed under the dome. We decided to eliminate the reactor dome and to improve the confinement of reactor roof. We needed a big solid piece of metal. We minimized the penetrations. We even considered temporary welding of perimeter of all components. We consider a liquid freezing shield for rotating plugs, just to mention a few most important proposed measures. The second simplification, which also will contribute to economics emphasized here, the first is elimination of reactive dome. And the second, which is quite revolutionary, is we proposed to eliminate also the safety vessel.

In the past, in traditional design, every sodium fast reactor has two vessels, main vessel, then gas gap, and then safety vessel and then reactor pit which is concrete. Our proposal is to minimize the gap between reactor vessel and pit and eliminate safety vessel. Instead of safety vessel, we propose to use metallic liner on the insulation. First, we introduce insulation shown here in green, which allows to minimize the gap between the free gap for inspection in particular.

We need to minimize this gap because in case of main vessel failure, we would like the reduction of the liquid sodium level when sodium goes outside and fills this gap. The height of this level will be still high enough to cover windows of intermediate heat exchanger. Even if in case of main vessel break and the leak of primary sodium, we still don't interrupt the primary circuit circulation and decay heat removal correspondingly.

For cooling the reactor pit concrete, we designed two cooling system, one in oil and one in water. They are forced under forced convection and we consider that they are also suitable for decay heat removal. In particular, we now have this system very close to the reactor vessel and we rely on radiation heat exchange. This system which uses concrete cooling pipes, concrete cooling loops for decay heat removal we called DHRS-3.

This was a brief overview of the subproject 1 activities devoted to new safety measures. I tried to present you the most important proposals. These proposals are under assessment now. The next goal, the next part I would like to talk a little bit about SP2, Subproject 2 and introduce experimental programs which are currently ongoing in Europe in support of the sodium fast reactor R&D but also talk a little bit about legacy. What we have from the past as an important database which we are using in the project.

So, I organized all experiments we have in the project on this table. I used a color code of this pale green for legacy data. This orange or pink for new experiments, and I structured every column of the table correspond to defense in depth levels. We have few experiments for normal operation, for sodium boiling, for severe accident management, and for severe accident mitigation. I put aside the MOX fuel measurement experimental program because it contributes to all defense in depth levels.

I would like to give you a few details. I already mentioned Superphenix. So, we used openly available legacy data obtained during the start-up tests at Superphenix operated in France for validation of computer codes for coupled neutronic and thermal-hydraulic calculations. We wait for new data to be obtained at KASOLA, new sodium loop which is currently under commission at Karlsruhe Institute of Technology, Germany. This data will be used for validation Computational Fluid Dynamics, CFD codes.

I also mentioned about the new Eddy Current Flowmeter under development at Helmholtz-Zentrum Dresden Rossendorf in Germany to measure sodium flow rate at the fuel subassembly outlet. For sodium boiling domain, in cooperation with Karlsruhe we collected legacy data obtained at KNS-37 sodium boiling loop at Forschungszentrum Karlsruhe, Germany to validate the sodium boiling models. I also mentioned the small test we planned at Karlsruhe Institute of Technology to make very compact sodium boiling facility with pulse laser heating to gain experience with two-phase sodium flow experiments, measurements, instrumentation, and to provide safety and to provide data for validation of sodium boiling codes. At the Paul Scherrer Institute we also work on new water steam facility in order to study chugging boiling conditions. So, we start with water and steam because it’s cheap. It allows us to prepare sodium boiling tests. It allows us to test and calibrate our thermal-hydraulic codes.

For the domain of severe accident measurements, we used legacy data from twin reactors CABRI and SCARABEE for validation of severe accident codes. So in CABRI reactor we used tests on molten fuel ejection for single fuel rod, which was really melting down, and the interaction of molten fuel with sodium channel. For SCARABEE, it was even larger scale experiment with a bundle of fuel rods with the measurements on mild propagation into the bundle. So, both reactors were operated at Cadarache Center France CEA [Unclear].

These two experiments are located in the Karlsruhe Institute of Technology in Germany. The left one called LIVE is to study interaction of molten corium stimulant with core catcher. The right one is designed to study interaction of molten corium stimulant with concrete. Finally, for this domain we have small facilities in University of Lorraine which are using similarity because between ice water interaction and molten corium with core catcher interaction, in particular the importance of formation of pool effect. These experiments are ongoing now.

For the domain of severe accident mitigation, we obtained, we collected legacy data from different facilities like FAUST and NALA from Forschungszentrum Karlsruhe to have data on behavior of aerosols in sodium vapor atmosphere to check our codes and also on FANAL experiment conducted in the past at CEA Cadarache France, some data on sodium pool fires.

Already mentioned several times, MOX fuel measurement we expect new data on fresh and burned mixed uranium-plutonium oxide fuel thermal-physical properties will be obtained for the use in computer simulations, not only even probably for fast reactors. It's very important new knowledge useful for the whole nuclear industry.

So I mentioned also the network of sodium facility in Western Europe. We have four sites where there are currently sodium loops available for experimentation. This is CEA, South of France. We can mention CHEOPS and PAPIRUS platforms. The two sites in Germany, one in Karlsruhe which is KASOLA and SOLTEC and KARIFA facilities one place at HZDR which is Dresden facility, and one site is in Latvia in University of Riga which is AMPERE and TESLA mainly related to electromagnetic pumps studies. I also mentioned that we support attachments of students working in our project to these facilities by dedicated mobility grants.

So, it was a short presentation of Subproject 2 and the experimental programs which we are going to use to support our R&D activities. Finally, I would like to summarize and say that the current achievements of the ESFR-SMART project in the context of development of European Sodium Fast Reactor are as follows.

So, first of all, we selected and specified the number of design modifications aimed at design simplification and safety enhancement. We developed a quite detailed description of this design. We have design drawings, we had card drawings. We have a lot of interfaces between card drawings and the input models of the files. It's a big achievement. Then, we already evaluated the new ESFR core neutronics performance in normal operation. We started a number of benchmarks and new experiments. Burnt fuel samples were prepared for measurements of thermal properties. This is very fresh news that fresh fuel samples are also available for measurements of thermal properties.

In the next phase of the project we plan to evaluate the new ESFR core and system performance in normal and accidental conditions. We continue benchmarks and new experiments. We focus on measuring fresh and burnt fuel samples thermal properties.

So, with this, I would like to thank you for your kind attention for attending this seminar. Please visit our project at this website. You can find there are a lot of information like project video, list of publications with possibility to download most of them, list of students and their projects, announcement of events, and so on. Thank you very much.

**Berta Oates**
Thank you Dr. Mikityuk. If you have questions for the presenter of today, please do go ahead and type them into the question pod. While those questions are coming in, we'll take a quick look at the upcoming webinars. In May, a presentation on the Formulation of Alternative Cement Matrix for Solidification and Stabilization of Nuclear Waste. In June, a presentation on the Interaction JOG/Sodium in Case of a Clad Breach in a Sodium Fast Reactor, and in July a presentation on Security Study of Sodium-Gas Heat Exchangers in the Framework of Sodium-Cooled Fast Reactors.

There are some questions coming in. I am going to see if I can share these with you Konstantine.

**Konstantine Mikityuk**
Yes please. Okay, I see the questions. Just let me read them. I don't know should I start from the top or the bottom?

**Berta Oates**
You can start with the top and work your way down, but it's certainly up to you how you would like to field them.

**Konstantine Mikityuk**
Okay I have a quite lengthy question. Why it was decided to develop such a powerful reactor concept knowing that the capital cost to build it will be high and the construction period will be rather lengthy, therefore, it might be quite difficult to sell the concept. Currently, the SMRs are getting popularity due to the relatively low price and quick deployment. What is the strategy to commercialize it or it is rather just being developed for research purposes?

Okay, you probably answered it already in your question. So first, I would say that we have some legacy. What I showed you as a history, this is important for us. We started this line, this work of large power reactor started with Superphenix 2 and European Fast Reactor, and at the time when the conditions were completely different in Europe. It was significant work, significant legacy for large power reactor of this design. So, one of the rationale, one of the reasons why we would like to continue this line is knowledge preservation.

I emphasized several times that we are sure that the work, the results which we produce including validated codes including field new experiments, new data, new ideas for safety measures, all these also developed for large power reactor they can be used by SMRs, probably with some rethinking, with some modifications. But we believe they will be useful for SMRs. Of course, in the current project we don't have work packages. We don't have work to evaluate the cost and to think about commercialization licensing.

No, we create the platform. This is the second probable reason we create the platform to keep the research ongoing on sodium fast reactor. We created the platform to start and budget to launch new Ph.D. studies and to organize cooperation between them. We create this framework. This is the goal. We don't think commercialization. We think how to preserve the knowledge and how to continue research and how to let young people who would like to work on this topic, how to give them the platform, the opportunity. We don't compete with SMRs. We are happy to cooperate. I am very much personally in favor of creating the project of small sodium fast reactor in Europe. I hope I answered the question.

The next one, what is the main advantage or logic behind the idea of having six steam generators? In this case you have six boxes. Each containing six steam generators which makes 36 steam generators total. Besides the increased cost, the higher number of steam generators increases the complexity and the probability of unplanned outages caused by various uncertainties. As well, it should greatly increase the length of downtime during the scheduled outages and replacements which leads to the lower capacity factor.

Great, thank you very much. You did an analysis which we can include in one of the deliverables. I like it very much. Okay, again similar answer, the legacy. We can ask why it was like this before? Before the compact, the printed circuits, heat exchangers didn't exist, the technology did not exist. The economic situation was different. I mentioned few advantages. For modular steam generator we have big surface area which we can try to use for decay heat removal. They are modular so they probably could be isolated and inspected and even probably replaced.

But I fully agree with your analysis of disadvantages. And in fact for the third year of our project or probably for the last year, we think about making in parallel the analysis of complete replacement of modular steam generator with very compact printed circuit heat exchanger. It will allow us to review significantly the building size also, which is important for economics. Thank you.

Next question, having in mind the behavior of sodium in contact with the water or air, how normal is the fire or the accident is suppressed, controlled during the small break or large break LOCA?

Okay first of all, we don't use this abbreviation. Loss of Coolant Accident as LOCA, well, historically it's used for pressurized systems where they are very energetic because of the high pressure. In sodium fast reactor there is no excess pressure, so there is no LOCA in this sense. In particular, in the primary system all pipes are inside the primary pool. In a way, the LOCA practically eliminated event. Or we can have very low probability, loss of integrity of reactor vessel. But that's probably loss of – more relevant to speak about loss of inventory than loss of coolant. We don't lose cooling ability.

Okay, I am not an expert in sodium fires. I know that there are safety measures like interaction of argon in the building in order to push out the oxygen or use of special extinguisher measures which excludes hydrogen, so no water and so on, special powders. But I am not the best person to give you the details about sodium fire management.

Let me go to the next question. You are welcome. I would like to ask about breeding of fissile materials, specifically high quality plutonium due to the inclusion of blanket which you have mentioned on Page 8 under advantages. So, you probably mean nonproliferation issue. Yes, it's important. It's one of the goal generation for system as far as generation for system. For this reason we don't have in our design a radial blanket. Because radial blanket means that this is a blanket which is movable and it doesn't have any other materials. It means that potentially misuse can occur with radial blanket, for example irradiation for a limited time of depleted or natural uranium and then extraction of assembly and the separation of high quality plutonium.

In our case, we have first very limited amount of fertile material. Second, it is always coupled in the same assembly as fissile material. It has a radioactivity protection from for example minor actinides, from fissile region which creates a very strong barrier or complexity for misuse of fissile material generated in the blanket. I think it's quite common approach to avoid radial blankets and still keep limited axial blanket.

Okay, next question. One of the goals of the Generation-IV reactor is to make sure that the advanced reactors are proliferation resistant. Yes, my question is in order to achieve the Gen-IV goals fully can't we just remove the blanket in order to minimize the breeding in the core from ESFR to make it proliferation resistant, for example, currently MIR 600 and TG-SFR?

Yes, again I am not the best specific expert in proliferation resistance. I think I already answered this question. Personally, I think it's possible, yes. We don't have – it's not a must, I would say like this, to have fertile blanket in the core. It improves the breeding. It improves probably even some safety parameters in some cases. But if this is the most important problem and issue for the country and for the organization or for the designer, yes it's possible. But in my opinion it would require a careful analysis of pro and contract. Clear analysis, what is this scenario which you would like to protect against when you have a very simple track of obtaining these materials. In my opinion, this analysis was already done and the conclusion is what I already said about radial and axial blankets.

So, how do fuel change possible during the operation? How do fuel change possible during the operation? Not sure I catch, change refuel. Okay, I go to next one. You can write it to me to my email. So, welcome Antonio. I have a question which is not related to the ESFR-SMART project. I was wondering about the current status of BN-800 Russian reactor. Do you know how is its operation? Could it be possible to apply its relative experience to the ESFR-SMART project? Thank you so much. Welcome.

Okay, there was a webinar devoted to BN-600 and BN-800, webinar number 24 given by Mr. Pakhomov from IPPE. And this was the right place to ask this question. I am not very much aware of details of operation. I know that BN-800 is very successful project and we are all proud of this project. This is great. This new reactor is really great for the community, for future of fast reactors on one hand. On the other hand, we look for any opportunities to cooperate with Russian colleagues, with Indian colleagues, with Chinese colleagues.

The two platforms which we have very efficient platforms are probably three: R-IEA technical working group on fast reactors is Generation-IV International Forum, one is OECD and the platforms. So, we cooperate everywhere. There is a dialog between ESFR-SMART and IBP and Indian colleagues, Chinese colleagues. This is great that we have this opportunity.

So then, which computer codes have been used for some hydraulics validation? We have many codes historically used for some hydraulic calculations of fast reactors. Correspondingly, every code should be validated for the domain of application, either normal operation, sodium boiling, or severe accidents. It means interaction with fuel and so on. If we talk about some hydraulic codes, okay, the dedicated codes are, for example, SAS4A or SASSYS which is US-Japanese and German development. This is a SIMMER [ph] code which is also consortium. France also I should mention.

We use CFD codes both commercial and open source like open forum. It's under development. They are very established for single phase but just we start using them also for boiling. We are using light-water reactor codes like TRACE, RELAP, ATHLET, or CATHARE which we modify by extending properties database or by extending models. They perform very well in fact because we can use all developments, huge validation database from light-water reactors. We see which parts we can use for fast reactors, in particular for sodium fast reactors. If you look at the modern system codes, I listed almost all of them already have coolants, equation of state for the advanced coolants like sodium, lead, lead-bismuth or helium.

Do you have an economical target in your design activities such as construction cost and/or power generation cost? Is it possible to satisfy your economical requirement with the current design strategy?

Thank you for the question. I already mentioned that no, we don't have in our project dedicated tasks for economic evaluation. If you don't have evaluation, it's difficult to have the target. I would probably recommend you to listen to webinar number 14 by Dr. Rothwell from NEA which is about estimation of cost of duration for systems. Then you can have an idea about the tools, about the level of uncertainties, something like this. But as I answered to the very first question, this is not our primary goal. We don't think how to build this reactor in the near future and how much it will cost. We have different goals.

During the presentation, you mentioned that spent fuel moved to outer place of core, make it far from center, exchange with fresh fuel. Is it possible to make this exchange during the operation?

Yes, so there are two approaches. One approach is what I described that the fuel assembly, the spent fuel assembly, 6-year old fuel assembly is taken by the machine from the core and put in a free space of internal fuel storage at the periphery of the core. When this fuel is 9-year old, so in 3 years this fuel is taken by the arm, by the mechanism and taken away off the reactor pool using gas heating. Then, it is located in the storage after treatment probably in a water storage.

This is quite expensive from viewpoint of time. It requires a long time. To accelerate it and to improve economics in a way, we can use another strategy. We can take the assembly directly, spent fuel assembly from the core, and move it to external storage. But for this case we should organize decay heat removal from this assembly by circulating sodium.

So, it should be a dedicated machine which is huge, expensive. If you look at some last images of ASTRID, so this machine was introduced in ASTRID but it was used in Superphenix also. You can have the idea how it works.

**Berta Oates**
Thank you everyone for such a great round of questions. Thank you Dr. Mikityuk for taking such time to answer so thoroughly those questions. I don't see more questions. There are some accolades coming in.

**Konstantine Mikityuk**
Okay, I would like to thank everybody for attendance, for very interesting questions. As I said, we will use some of them in our reports. Thank you very much. Thank you to Berta, Patricia, and to Generation-IV International Forum. It was a great opportunity. Thank you.

**Berta Oates**
Thank you. Thank you very much. Have a great day.

**END**