Experimental R&D in Russia to Justify Sodium Fast Reactors
Dr. Iuliia Kuzina, IPPE, Russia

Bertha
Welcome everyone to the Next GEN IV International Forum webinar presentation. This morning's presentation on 'Experimental R&D in Russia to Justify Sodium Fast Reactors' will be presented by Dr. Iuliia Kuzina. Doing today's introduction is Dr. Patricia Paviet. Dr. Paviet is the Group Leader of the Radiological Materials Group at Pacific Northwest National Laboratory. She is also the Chair of the GEN IV International Forum Education and Training Working Group. Patricia.

Patricia Paviet
Thank you, Bertha. Good morning everyone. Before I introduce Dr. Iuliia Kuzina, I just wanted to remind you that this month is our fifth anniversary of us doing a [ph] webinar on GEN IV system. We would like to thank you for following these webinars and showing your interest in these advanced reactor systems. If you can, please talk to your colleagues about these GIF webinars, and let's continue this important research in the nuclear energy arena.

Dr. Iuliia Kuzina is with us today. She is Deputy Director General, Director of Nuclear Power Department in the State Scientific Center of the Russian Federation, the Leypunsky Institute for Physics and Power Engineering, Joint-Stock Company. Since 2000 she has been involved in studies of heat transfer in liquid metal coolants and in 2003 she earned her Ph.D. degree. Since 2016 she has been holding the position as Director of the Department in charge of sodium fast reactor, lead-cooled fast reactor, and light-water reactor design justification. She is a leader of computational and experimental work aimed at justification of safety and operability of fast reactor designs with liquid metal coolants. In addition, she has been teaching at the university as a lecturer in thermohydraulic calculations since 2004. She is nominated as an expert from Rosatom in GIF lead-cooled fast reactor provisional System Steering Committee.

Without any further delay, I thank you again, Iuliia, for volunteering to give this webinar and you have the floor. Thank you Iuliia.

Iuliia Kuzina
Thank you, Patricia. Good day dear colleagues or good evening, good morning, I don't know. I see that a lot of my friends are here. Hello everyone. I thank the organizers for giving me the opportunity to be a speaker at this webinar. I'm a little nervous because it's my first experience of participating in such an event. I hope that it will be interesting for you

will hear something new about the research that has been conducted and is being conducted in Russia for sodium reactors.

I'm going to talk about the ongoing experimental R&D work in Russia for sodium fast reactors mainly illustrated by an example of the BN-1200 project. This reactor currently under development meets the criteria of the fourth generation reactors, although I will mention other projects as well. When [Unclear] reactor is designed and developed, it's required to perform the experimental justification of new technical solution and earlier developed systems operating with new parameters and under new conditions. As an example, in the slide, you can see the BN-1200 reactor with least of equipment most part of which requires conducting the experimental R&D work. For instance, the in-vessel internal structures were justified in BN-600 and the BN-800 designs, whereas the newly designed, designed meaning circulating pump fuel and all basic safety system, emergency heat exchangers, and other components are used for the first time. The R&D program is envisaged for the new systems, and it's clear that it includes both, calculation, and experimental parts. Today, I'm going to dwell only on the experimental part.

Dimensionally, the reactor can be divided into two big parts, the core with its elements and in-vessel equipment of the primary circuit. In this and the next slide, you can see the main components of the BN-type reactor system such as intermediate heat exchange, main vessel's pressure chamber, core pressure pipeline, primary main circulating pump, control, and protection system, actuators, and so on. Not only the reactor as a whole, but also the new reactor equipment and systems require justification, I mean it's true for many components, from the reactor to the heat conversion system and in our case it's the steam generator. The BN-1200 reactor has a three-circuit configuration with four heat removal loops.

The loops have been chosen symmetrical, thus making it possible to simplify the process of designing and manufacturing in contrast to the BN-800 reactor. Each loop is connected to an intermediate heat exchanger. There is one main circulating pump and two modules of steam generator. The emergency heat removal system is also represented by four identical valves [ph] with one autonomous and one heat exchanger for one loop. In contrast to BN-800, the pipeline length and the reactor building dimensions were significantly reduced. This factor required new expansion bellows to be developed for the main pipeline. The requirements to the reactors of the fourth generation [Unclear] that's forcing the designers to improve safety on the one hand, but at the same time to reduce the cost of the entire reactor as a whole. For the BN-1200 project, the experimental R&D work has been performed and is planned to be performed to justify such equipment as, for example, the emergency heat removal system, steam generator, main circulating pump, expansion bellows, and others.

Let's start with the research at justification of the core characteristics as the most high energy part of the reactor.

The experimental R&D work on the core justification starts with the core simulation at the critical facilities. At the Institute for Physics and Power Engineering, there are two facilities like these BFS-1 and BFS-2, they primarily differ in their sizes. Before the experiments, first the computer simulation of the BFS loading is to be performed and then the core is assembled and the test studies are performed with the aim of determining the core neutronic characteristics. This data forms the basis for the basic design of the core and justification of its safety.

The research program at the BFS critical facilities can be conventionally divided into two big classes, benchmark and mock-up experiments. Benchmarks are critical assemblies of a maximum possible simple design intended for simulation of spectral characteristics of the cores and media and study. Assembly geometry simplification is for the purpose of minimizing model uncertainty and identifying a constant [Unclear] of computer codes and the experimental results are obtained. Such assemblies are primarily used for verification of constants of different reactor materials. Another class is mock-up experiments or full-scale simulation, on the contrary implies the most plausible description of the geometry of the object under study and its fuel composition. The results of such experiments are for verification of design codes and design nuclear safety analysis at regulatory authorities. A special mention should be made of the work in preparation for the first criticality of the reactor. It involves development of measurement procedures for the first criticality including sensitivity testing of the detectors as part of the startup instrumentation, fuel loading pattern in the course of first criticality gaining, measurements to control the rod worth [ph] and spatial power density distribution to specify the reactor power and so on. An example neutronic parameters obtained in the experiments is presented on the next slide.

Cooperation with France can be considered as a good example of using BFS core research and physics to justify nuclear safety. One of the principles of designing safety of generation IV reactors is safety as an intrinsic property of the facility or in other words the presence of negative reactivity feedback, which if the situation gets out of control will prevent depravation [ph] of an uncontrolled chain reaction and allow a nuclear accident to be avoided. It was the study of the BN reactor for physics and in particular the void reactivity effect was dedicated in the frame of the joint French [Unclear] program of research launched at the BFS critical facilities in 2012. The research program involved experiments at four critical assemblies, which differed in both, their core size and its composition. Sodium [Unclear] heterogeneity of the core were considered as the design solutions resulting in negative sodium void reactivity effect. Each of these solutions was considered both, individually and in combination. The first two critical assemblies were models of the [Unclear] medium typical of the core of the Russian [ph] BN-1200 reactor and the other two critical assemblies were similar to the French ASTRID reactor. The sodium void reactivity effect of the core was studied in detail to increase the reliability, the measurements were made by two independent methods.

Cooperation with the Republic of Korea can be taken as an example of using the BFS with the aim of justifying specifically the design neutronic characteristics and design computer codes at the regulatory authority. Since 1997, a number of experimental programs have been carried out at the BFS critical facilities to simulate the Korean fast reactor. The reactor cores of various configurations were studied in a wide range of capacities. These studies dealt with both, full-scale models and benchmarks. The focus was on the issues related to nuclear safety. Consideration was given to the control and protection system, both, sodium void reactivity effects due to the core expansion in the course of heating and many, many other issues. The extensive experimental material obtained was used to justify the license of the design project of the PGSFR-150 reactor facility. They are prototype of the fourth generation reactor.

It should be noted that the BFS is an excellent example of international cooperation, which gives both sides many advantages. For the BFS itself, these are experimental possibilities. In the context of the test objectives, the BFS instrument and material resources are managed and experimental capabilities are increased. New test procedures are mastered and available ones are improved at the state-of-the-art level. Calculation analysis of the experiment and estimation is performed with the use of the most advanced computer codes and it makes it possible to both, improve the national neutron data library and contribute data to the libraries of international partners. The stakeholders generate a research program and as a result exclusively receive new experimental data. Finally, due to international collaboration, dozens of experiments performed at the BFS were subject to a thorough expert review and included into the handbooks.

Let’s pass [ph] from the core characteristics to the safety analysis. Experiment to justify the core safety is a crucial task, its result effect are following. They make it possible to verify the calculation codes, which simulate accidents including severe ones, and also to remove excessive conservatism in the safety evaluation. [Unclear] high temperature processes are started in such experiments, and it should be noted that today an experimental program has been developed in Russia which covers a rather broad range of core safety issues. We should implement it within a few years.

Now I will focus on the experimental studies that have been carried out by the present time. To carry out project design work, R&D is required to study high temperature processes that occur in the cold or fast reactors under emergency conditions. For such tasks the IPPE developed the AR1 [ph] facility, which is used to conduct experiment from hydraulic studies of transient and emergency operating modes of fast neutron reactors with liquid metal coolants. At this test facility, a series of experiments was carried out which simulated boiling in fast reactor core with a design, which is specific to the BN-1200 reactor. State-of-the-art modeling of dynamic liquid metal coolant boiling is of prime importance for an integrated analysis of neutronic and thermohydraulic characteristics of a fast reactor core for safety justification in accident conditions.

Experimental data obtained at IPPE have shown that the liquid metal boiling in fast reactor fuel assemblies has a complex structure and is characterized by both, stable and pulsation regimes with significant fluctuations of the parameters, which are capable of giving rise to a critical heat flux. Stable nucleate boiling in the simulated fuel assemblies is absorbed only in a restricted region of heat fluxes, and its transition to unstable slug boiling is determined by various factors, for example, roughness. For sodium boiling in the fuel assembly model with sodium [Unclear] located above the reactor core which is designed for compensation of the positive sodium void reactivity effect in case of boiling onset, the feasibility of long-term sodium cooling of fuel rod simulators in the fuel assemblies has been demonstrated, all these conditions during 10 minutes.

The data on liquid metal boiling heat transfer in the bundles were generalized and cartogram of the flow regimes for liquid metal two-phase flow in the bundles was constructed.

New safety elements in the fourth generation reactor require additional experimental justification. In BN-1200 reactor, such a completely new system is a passive safety system based on the temperature principle. Unlike BN-800, in addition to hydraulically activated safety rods, BN-1200 also has temperature actuated rods. The design decided to give preference to the fusible element. At the stage of concept design, options with shape memory effect, curie equivalent, and some of expansion were conceded.

The passive safety system has an actuating component and this component of various designs is currently being tested. For testing the passive safety system devices of various designs with various fusible elements, a high temperature section was created at IPPE at the 6B test facility, which provides the parameters of temperature variation rate close to those in a reactor. Endurance tests are carried out prior to the test dedicated to actuation processes. The temperature increase rate under emergency conditions is high, up to 25 degrees per second, and high temperature and its increase rates impose serious requirements on the characteristics and quality of the facility and its equipment.

This slide shows the design of the test chamber as well as the principle of a passive safety system element [Unclear] limit. As the sodium temperature in the chamber rises to 720 degrees at a predetermined rate of about 1 meter per second. The actuation time of the sensitive element after reaching the set temperature is measured and the actuation temperature is recorded. At the bottom of the slide, you can see photos of one of the elements before and after the test.

Next slide is about test facility Pluton at the IPPE – Pluton is designed for investigation of accidents through the use of corium simulators including uranium-containing ones, material movement in the course of fuel melting in the fuel assembly models of various geometries, fuel heat-to-mechanical energy conversion factors, and so on. Measured parameters are coolant power, temperature, flow rate, pressure, and so on. There are lots of experiments carried out at the test facility are used not only for understanding comprehension or the physics of processes, but also for verification of all simulating severe accidents. You can see, there are a list of studies we can perform using this test facility.

One of the studies was dedicated to distribution of materials along the height of fuel assembly models and plugging of their flow sections in the course of simulating an uncontrolled loss-of-coolant accident in BN reactors. The goal of the research was to identify the mechanisms of fuel cladding failure [ph] in the 19 rod test bundle and then numerical evaluation of the intensity of these mechanisms in the process of core degradation. Power density in the experiment was provided by the mild reaction. The markers such as corium were introduced into the composition of the mild mixture of materials. In the initial state, they are localized in a strictly defined position along the height of the fuel rod simulators. It made it possible to assess the nature of the mild movement inside the test bundle on the basis of the available data on the final distribution. Then in accordance with performed experiments three main mechanisms of cladding degradation have been identified. Thermal stresses in the cladding material; second is melting of cladding materials, dynamic effects posed by rapid conversion of the thermal energy of the corium simulator melt into mechanical block in the course of melt sodium thermal interaction. The slide shows photos of the 19 rod assembly after a series of experiments. Flow area plugging, global degradation, and assembly [Unclear] through and other consequences of simulator accident are demonstrated here.

Other experiments related to severe accidents, experimental study of corium behavior at the boundary between corium and reactor structures. Phase stratification of the melt into metallic and ceramic phases is shown experimentally. The influence of thermal interaction of corium with nitrogen, which simulates the vapor phase of sodium is considered. Melt ejections from the interaction area due to the gas phase expansion were experimentally estimated. Material wear and deformations of specimens are recorded for the categories subject to melt shock impact only. The results obtained make it possible to improve the high-temperature part of the Russian COREMELT code, it's core melting part.

Now some words on the R&D work on justification of new fuel. Actually, a special webinar would be dedicated to a new type of fuel, nitride, and in this presentation I will indicate the main principles of its justification. In BN-1200 reactor, it's supposed to combine the flaw in new technical solutions for the fuel element. Fuel element cladding, austenitic steel at the initial stage, and improved steels of ferritic-martensitic class at the following stages. Two fuel types, MOX fuel and new fuel type, mixed nitride uranium-plutonium. Irradiation tests are carried out at fast neutron reactors, BOR-60 reactor and BN-600 reactor.

Within the framework of the Priority project, a comprehensive program for computational and experimental justification of fuel elements with mixed uranium-plutonium nitride fuel for the BN-1200 and BREST reactors for the period until 2020 was developed. This program includes pre-reactor experimental studies of the properties of nitride fuel and cladding material; testing of experimental fuel elements with mixed nitride uranium-plutonium fuel in BOR-60 and BN-600 reactors; post-reactor studies of irradiated fuel elements; development of methods, codes, and criteria for substantiating the operability of fuel elements; and the improvement of fuel elements with mixed nitride uranium-plutonium fuel, development and optimization of technologies for their manufacture.

The third section is a certain group of studies [Unclear] justification of the primary in-vessel equipment.

Let me start with the main circulation pump of the primary circuit. A radically new rotor turning gear lubrication system is used in the design. It has nothing comparable in BN-type reactors. Lube oil supply system is downsized because an oil bath lubrication system was applied and many elements of the system were removed, it reduced the mass of this unit. A full-scale mockup of the upper bearing unit has been created and is being tested.

The design has used the following approach. The equipment that has reference samples and requires no justification at high temperatures undergoes water tests. The components that require liquid metal and high temperatures are tested at sodium test facilities. For example, shaft seal mockups undergo water tests.

Let's consider a unit such as an emergency heat exchanger and its check valve. A heat exchanger with a check valve has a completely new design. The valve opens once a year during loading, it's approximately once every about 330 days, and it is exposed to high sodium temperatures up to 550 degrees. There is a probability of the check valve welding with the heat exchanger. The tests of the check valve model carried out continuously for 265 days at a sodium temperature of 550 degrees and at pressing force and confirmed durability of this element in conditions close to real. There were no signs of self-[Unclear] of a check valve blocking element with the model seat [ph].

Another very interesting development is a small-sized plugging meter of impurities in sodium. The system of coolant purification and quality control is located in the reactor vessel. The design of a small-sized plugging meter to measure impurities in sodium and a system for its non-dismantling periodic verification have been developed and are being tested at the sodium test facility. A small size of the equipment allows it to accommodate a plugging meter inside the vessel of BN-type fast reactors. The reduction in size has been achieved by combining the functions of a pump and flow meter in one device. This design has obtained several patents.

Okay, some technical problems. The further few slides are dedicated to the study of the coolant flow in the course of decay heat removal. Two test facilities were used to study coolant flows inside the reactor vessel. The working fluid of both test facilities is water. Test facility TISEY simulates only one sector of the BN-1200 reactor and test facility V-200 simulates the entire BN-1200 configuration.

I would like to give you more details on the results obtained at the V-200 test facility. At the IPPE, thermohydraulic studies were performed with the water model of the primary loop at the scale of about 1 to 10 for a pool-type fast liquid metal cooled reactor. The model houses all the main components of the in-tank equipment as it's shown on this slide.

Coolant velocity fields in the upper chamber of the reactor were measured and average coolant temperature distribution along the height of the upper chamber was determined. Experimental studies of the coolant flow and its temperature fields demonstrated the areas of steady state coolant temperature stratification. At the interphases of the stratified circulation regions, high temperature gradients and significant fluctuations were detected. It can result in an impact on structural materials, thermal fatigue, and reduction of the service life of reactor equipment which is confirmed by experiments. It's a must to model these local [Unclear] characteristics of stratified flows with computer codes.

The comparison was made of the charts showing thermohydraulic processes in the reactor vessel under the nominal conditions and the steady-state natural convection regime. The steady-state natural circulation conditions are characterized by significantly lower temperature gradients in the vertical direction above the radial blankets as compared to the nominal conditions. It was shown that the emergency heat removal system with autonomous heat exchanges has a certain margin in terms of its heat removal capacity. In other words, heat removal under the conditions of emergency cooling is possible without any exceeding of nominal temperature.

The next section is dedicated to justification of the secondary circuit equipment.

Within the framework of BN-1200 reactor power unit optimization, the decision was made to develop a leak-proof pump, the second main circulating pump. The pumps like these haven't been designed in Russia yet and more so haven't been yet tested. Currently, we are getting ready for testing the second main circulating pump model at our track 1 [ph] test facility at IPPE. The specific feature of reactor design consists in the fact that the leak-proof pump of the secondary circuit is installed directly on the intermediate heat exchanger, offering new design with a synchronous motor. In the near future, we are going to do the following. Testing of a bearing with a full-scale axle-bearing model; testing of the individual components of the pump, sealed or [Unclear] motor, a thin-wall shield; testing of a full-scale axial/radial-bearing model of scale 1 to 3 of the secondary pump. All the tests are conducted with the parameters similar to the operating ones both in sodium and water.

The following tests concern expansion bellows. The BN-1200 design envisages expansion bellows installed in the main pipelines and steam generator modules. The tests cover a few steps. Heating, cooling, melting, and solidification; flexibility of bellows in different planes; examination after all the test cycles completed. Expansion bellows confirmed their operability at the preset parameters, up to 35 cycles at a temperature higher than 500 degrees.

Steam generator, the scope of R&D work on BN-1200 steam generators includes a great number of tests such as research into the processes in heat tubes with single-tube and multi-tube models; development of the assembling process for a straight-tube steam generator of a long length; development of water-to-sodium leak detection systems; and testing of rupture disks.

Now, in the design BN-1200 reactor, we use a large-modular steam generator. This new design should be tested and justified experimented. The experiments were performed with a single-tube model at the SPRUT test facility in IPPE. The aim was, as I mentioned, to justify the design parameters of a new design of a large-modular reactor steam generator where within one shell the processes of steam evaporation and superheating take place. The experiments were conducted at the modes of 12% and 75% of the nominal sodium flow rate with the real parameters in the sodium loop and high-pressure water loop. The experimental critical heat flux data obtained demonstrate a satisfactory agreement with the data of skeleton tables on calculated critical heat flux in tubes. A significant effect of water pressure on both, the critical steam quality and the value of heat flux density was observed. With pressure growth, the value of heat flux density goes up and the value of critical quality goes down. Now, this year we are getting ready to perform similar experiments at the SPRUT test facility for seven tubes model and we are going to conduct such experiments this year.

We paid a lot of attention to the creation of a small water-to-sodium leak detection system. It was designed on the basis of solid electrolyte sensors that monitor hydrogen and oxygen contents in sodium. On the slide, you can see that the…

The next slide gives you a brief information about the steam generator over pressure passive safety system, which is designed on the basis of pressure safety devices in sodium and gas. The slide presents the structural design of the first-of-a-kind sample of the rupture disk device, and we are going to test this rupture disk device next year.

The following tests concern sodium leak detection with the use of fast-removable thermal insulation as a protective jacket in the secondary circuit. Six experiments were conducted, three experiments with a horizontal pipe and three experiments with a vertical pipe. Different sodium leak rates through a flaw with a diameter of 2 millimeters were used.

You can see steady-state conditions for experiments. The signs of sodium leak through the flaw appeared 3-5 seconds after and it was smoke. The detector was actuated immediately. A minute after the contact sensors were actuated, output to our measures and external electrical heat. Approximately 3 minutes after, the [Unclear] block was actuated. The sodium leak through the floor lasted about 4 minutes. The amount of leaked sodium was about 2.5 liters. The use of contact sensors for sodium leak detection, wire and meshed detectors, demonstrated its effectiveness in terms of sodium leak detection at early stages with the capability of sodium leak localization.

Next slide is about test research into sodium fires. Test procedures are sodium, the sodium pipeline, and test section are heated to the preset temperature. Then, the required pressure is achieved and the valve is opened and sodium goes to the test section under thermal insulation with a specified flow rate. Different types of thermal insulation required experimental justification. The conclusions based on the experimental results were no sodium spray fire at the preset leak rates, confirmation of the concept 'leak before break,' performance testing of the sodium leak and fire detectors.

The last section of my review is some issues of sodium coolant technology.

Some [Unclear] about testing of a cold trap model at the sodium test facility. We performed it at the test facility Protva-1 in IPPE. A computer code based on the open phone [ph] package was developed to calculate sodium oxide [Unclear] in the cold trap volume. Then, the code was verified with the use of experimental data obtained and calculations of sodium oxide mass transfer were performed for a standard code test. On this slide, you can see that cold trap model schematic, the experimental conditions and experimental results. Distribution of deposits along the trap model length was determined.

Let's move to the development of a combined system of primary sodium purification from oxygen with the use of getters, sorbents, and filters. The three options of a combined [Unclear] primary sodium purification system was suggested and tested. The first, a getter trap and filter to retain zirconium oxides in the outage conditions with heat map [ph] in the conditions of operation the nominal parameters at the maximum temperature of the primary [Unclear]. The second, a getter trap and filter to retain zirconium oxides and a cold trap. A cold trap to clean sodium in the outage conditions and a getter trap to clean sodium at the nominal parameters. The third was high temperature and low temperature suction [ph] sodium purification.

Finally, the last slide, which demonstrates research into mass transfer of steel corrosion products in sodium up to 780 degrees. The distribution of the flux density of corrosion products on the walls along the channel was obtained. It is consistent with calculated data for chromium under conditions with low content of oxygen and hydrogen in sodium. This fact confirms the possibility of using the previously obtained physical and chemical constants to calculate chromium mass transfer in high temperature sodium circuits with an increased hydrogen content in sodium up to 6 PPM. The issue was of the obtained constants makes it possible to estimate corrosion products transfer in the sodium circuits with both, allow content of oxygen and hydrogen in sodium and their increased content. Such estimates are required to predict possible accident.

As a summary, I would like to say that a lot of experiments have already been carried out and now we are planning a number of experimental studies to ensure that BN-type reactors are safe and reliable.

Thank you very much for your attention. On the slide you can see my email. If you have any questions, please feel free to ask me right now or by mail. Thank you again. Thank you. So long report.

Bertha

The upcoming webinars. In October, we have a presentation planned on Metal Fuel for Prototype Generation-IV SFR: Design, Fabrication and Qualification. November, we have a presentation on Geometry Design and Transient Simulation of a Heat Pipe Micro Reactor. In December, Development of an austenitic/martensitic gradient steel by additive manufacturing planned.

Bear with me, I will have to…

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