Russia BN 600 and BN 800
Mr. Ilya Pakhomov, IPPE, Russia

Berta Oates
Welcome everyone to the next Gen 4 International Forum webinar presentations. We thank you for joining us this morning. Today's presentation on ‘BN-600 and BN-800 Operating Experience’ will be presented by Mr. Ilia Pakhomov. Doing today's introduction is Patricia Paviet. Dr. Patricia Paviet is the technical group manager of the Radiological Materials Group of the Nuclear Sciences Division at the Pacific Northwest National Laboratory. She is also the Chair of the Gen 4 International Forum Education and Training Task Force. Patricia?

Patricia Paviet
Good morning, everybody. It's a pleasure to have Mr. Ilia Pakhomov with us. He is the Head of Laboratory in the State Scientific Center of the Russian Federation, Institute for Physics and Power Engineering, named after A.I. Leypunsky. Since 2006, he has been charged with developing advanced sodium fast reactors as an engineer, junior researcher, and head of laboratory. In 2014, he became a member of the working group on scientific and technical support of the BN-1200 project in IPPE. Currently, he is head of laboratory, management of experiments and engineering safety of fast sodium reactors. He is responsible for research of operability elements of the core, safety issues of sodium fire, and safety during interloop leaks in the sodium-water steam generators. He is also involved in the formation of a Research and Development plan for the Fast Sodium Reactors.

It's a pleasure to have you, Mr. Pakhomov. Thank you so much for volunteering to give this webinar. I give you the floor. Thank you.

Ilia Pakhomov
Thank you, Patricia. Hello, everybody. I will make a presentation about the operating experience of BN-600 and BN-800 power units. In this presentation, I’ll make an introduction. I’ll tell you about the operating experience of the BN-600 reactor power unit. I will tell you the main characteristics of the power unit of the core. I’ll tell you the main stages of this power unit operation, and the main results of the BN-600 power unit operation.

The next one will be BN-800. I’ll tell you about the construction and operation of this reactor. I’ll tell you the main characteristics of that power unit and the principal stages of construction, commissioning, and operation. And the last one will be the prospects of the sodium fast reactor development in Russia.

Introduction. This report reviews the Russian experience gained in the field of sodium fast reactors where Russia takes the lead in the world using the example of power units with the BN-600 and BN-800 reactors. Sodium fast reactors development began in the 1950 of the 20th century, including the USSR. But in most countries, the development of this reactor technology was subsequently suspended. In the USSR and Russia, the SFR development has been ongoing for more than 60 years.

The results of these developments are as follows. Experimental fast reactor BR-5/BR-10 located in IPPE in Obninsk. Experimental fast reactor BOR-60 operating at the RIAR in Dimitrovgrad. Prototype reactor unit with the BN-350 reactor that was in operation in Shevchenko. Industrial power unit number 3 with BN-600 reactor operating, and unit 4 with BN-800 reactor put into operation at the Beloyarsk Nuclear Power Plant in Zarechny. Project of the multifunctional fast research reactor, MBIR, being constructed in Dimitrovgrad. And the project of a commercial power unit with the BN-1200 reactor.

This table presents the data on the operating lifetime of various sodium fast reactors that existed earlier and are currently operating in the world. The lifetimes shown in the table are calculated from the time of the birth, startup, to the time of the final shutdown of the reactor. The table presents data on the operating life of various SFRs existing before and currently operating in the world. Here you can see such research reactors like EBR, BR-5/10, Fermi, Rapsodie, BOR-60, and JOYO. And you can see also the powers of sodium fast reactors like BN-type reactors, Phenix, PFR, Superphenix, and MONJU. For all countries, it's more than 550 years' experience.

This data provide a visual representation of the accumulated experience of SFR in various countries. You can see that the biggest experience of SFR is in Russia, USA, France, and Japan.

In this slide, you can see the timeline of the sodium fast reactor development in the USSR and Russia. It starts from BR-5 and doesn't end by the BN-1200.

Now we will talk about the operating experience of the BN-600 power unit.

In the course of BN-600 reactor design development and engineering concept selection, all the experience accumulated at BR-5/BR-10, BOR-60, and BN-350 was taken into account. The fundamental difference of the BN-600 from the previous SFR designs is a pool-type arrangement of the primary circuit. Correctness of the decisions made in the BN-600 design was later confirmed by its successful operation for almost 38 years. Power startup of the BN-600 reactor took place on April 8, 1980, the design power level was reached in December 1981. Since 1982, power unit 3 of the Beloyarsk Nuclear Power Plant with the BN-600 reactor has been in commercial operation as a commercial power unit.

The basic specifications of the BN-600 power unit are shown in this table. I will pay attention to the main of them. First of all, the thermal power of 1470 thermal megawatts. We have three circuits: primary, secondary and tertiary circuits, and the three loops for each circuit. As I said, is a pool-type reactor and the sodium temperature in the outlet from the core is 550-degree Celsius. The core diameter is approximately 2 meters and the height is 1 meter.

All our pumps in the primary and secondary circuit are centrifugal type. In steam generator-type, we have once-through section and modular with three modular in each section, and we have 8 sections, so we have 24 modules. We have a standard turbo generator with a power of 210 megawatts. We have a decay heat removal system. It's located bypass with the air heat exchanger on the loop of the secondary circuit. Our rotating system is made by 2 rotating plugs and we have a vertical refueling mechanism. We have 3 spent fuel storage. One of them is universal storage. Second is the intermediate sodium storage. The last one is the water pool. We wash our sub-assemblies from the sodium by the steam, gas and water.

The program of gradual enhancement of the uranium oxide burnup design level was carried out at the BN-600 reactor. The first modification of the BN-600 core, 01M, involved changing the fuel element and core configuration to decrease linear heat rating in fuel elements, as well as optimizing the scheme of core refueling.

Further increase of fuel burnup was achieved by changing the structural material of the fuel element cladding and fuel assembly ducts. It's the core, 01M1 modification. The successful operation of the BN-600 with this core and the complex research work made it possible to increase the design value of fuel burnup up to 11.1% of heavy atoms and to change over to the longer fuel element lifetime with 4-fold reactor refueling. It's the 01M2 modification.

Here you can see the main design characteristics of all BN-600 core modifications that were changed during its modification. You can see how we changed the reactor core high, how we changed the core structure materials, how we increased our 'maximum fuel burnup.' You can see the 'fuel operating life' in official days, and 'fuel inventory in core.' As you can see, we have two different few cycles, summer fuel cycle, and winter fuel cycle.

This figure shows the load factor variation of the Beloyarsk Nuclear Power Plant power unit with BN-600 reactor during the period of commercial operation. The average load factor for the entire period of operation of the power unit till the end of 2017 equaled to 74.25%. For the period of commercial operation, from 1982 to 2017, it was equal to 74.55%. Some load factor decrease during the period from 2005 till 2010, that can be seen in the figure, was caused by the necessity to carry out the work on BN-600 lifetime extension through the replacement of the equipment whose lifetime was over, specifically steam generator modules and modernization of safety systems.

The results of a detailed analysis of the reasons for the load factor decrease for the period of operation from 1982 till 2004 inclusive. The value of scheduled load factor decrease is caused by duration of power unit shutdown periods necessary for carrying out scheduled preventive and repairing work and reactor refueling.

Now, the duration of the annual reactor shutdown period for scheduled preventive repair is mainly determined by the rated time of complete overhaul of the turbine generator and time required for reactor refueling. In the recent years of operation, the average scheduled preventive repair duration has been about 71 days.

Here we can see the distribution of reasons of load factor decrease for BN-600. As you can see, only approximately 3% is the equipment failure and personnel errors.

This figure shows the time distribution of equipment failure and personnel errors that took place in BN-600 power units from 1982 till 2017. It should be maintained that after the personnel had mastered the sodium fast reactor technology at the early stage of the BN-600 power units operation and adjusted the main sodium equipment, the failures occurred mostly in electric supply system and technological equipment of the 3rd circuit and were not connected with the sodium systems. The unscheduled value of load factor loss is equal to 1.1% per year.

Operation period and lifetime performance of sodium fast reactor equipment without overhaul. We can see the components. It’s non-replaceable equipment like reactor vessel and primary circuit piping. And sodium pumps, intermediate heat exchanger, and steam generator. You can see how long the lifetime was of this equipment without overhaul. The presented data testify to good compatibility of sodium coolant with structural materials used and its low corrosion activity in the mastered range of SFR parameters. The replacement of the following main equipment of the BN-600 power unit was implemented during its operation. It’s the 4 sets of the primary sodium main circulating pumps, 1 set of the secondary sodium main circulating pumps, 1 set of the mechanisms of the control and protection System, 3 sets of guide tubes of this protection system, full set of steam generator modules and plus 1 set of steam generator evaporator modules. And 1 intermediate heat exchanger. The unique repair work was performed on the small rotating plug during their life.

The experience in sodium leaks outside and inter-circuit leaks in steam generator was gained at the early stage of the BN-600 operation. All 27 sodium leaks that occurred at the early stages of the BN-600 reactor operation were mostly small leaks. In 21 leaks, the amount of sodium leaked didn't exceed 10 liters. It was from 100 milliliters to 10 liters. In 6 other leaks, the amount of sodium leaked was 30, 50, 300, 600, 650 and 1000 liters. This table contains the main characteristics of these sodium leaks. As you can see, we have only one leak, a big leak in the primary sodium system.

There were 14 cases of sodium fires. All the 27 sodium leaks were detected in due time by detection systems or operators. Powders were used for confining and extinguishing non-radioactive sodium fires. The design algorithm of sodium fire consequences confinement was implemented only in one case of radioactive sodium leak from the primary circuit, and it proved its value: the radioactivity release of 10.7-curie caused by this incident was well below the permissible limit. The accumulated sodium leak experience proved the effectiveness of the protection systems aimed at leak consequences confinement.

Total number of leaks can be distributed with respect to the reactor facility components in the following way. So, 5 is the sodium reception system, 5 cut-off valves of the steam generator modules, 12 is auxiliary secondary systems, and 5 auxiliary primary systems. It should be noted that the last sodium leak at the BN-600 reactor took place 24 years ago, in May 1994.

The sectional/modular steam generators used in the BN-600 reactor have demonstrated high performance characteristics for the whole period of power unit operation. Overall, there were 12 leaks in the steam generator. All these leaks took place in the early stages of the unit operation. Half of these 12 leaks of steam and water into sodium happened in the first year of operation because of manifestation of hidden manufacturing defects.

Inter-circuit leaks took place mainly in the modules of superheaters, six events; and reheaters, five events, while only one leak occurred in the evaporator. All the listed steam generator leaks were suppressed by regular means and didn't result in emergencies. During the last 33 years of operation, there was only one minor steam generator leak that happened in 1991. The steam generators have operated without any inter-circuit leaks for 27 years despite numerous replacements of steam generator modules that were performed during this period as part of the power unit life extension procedures.

This figure shows the BN-600 steam generator section that consists of 3 modules. It's evaporator in the center, superheater is on the left, and reheater is on the right. The steam generator of each coolant loop consists of 8 steam generator sections connected in parallel.

This table presents the main characteristics of inter-circuit leaks in BN-600 power unit steam generator modules. You can see the type of module and leak rate. As you can see, there is only one leak in the evaporator. Evaluating all the deviations from normal operating mode that took place during the BN-600 operation, including those connected with sodium leaks, it should be emphasized that none of them resulted in any radiation impact on the population and environment. By the off-site impact criteria, all of them are below the International Nuclear Event Scale, and therefore are insignificant.

The main result of the BN-600 power unit operation. You can see. During the operation of the BN-600 power unit, the following achievements were reached. Long-term endurance tests of large-size equipment operating in sodium. Mastering the sodium technology on an industrial scale. Development and optimization of operating modes. Mastering the technology of replacement and repairing of sodium equipment including the primary components like pumps, steam generators, and so on. Reaching the acceptable level of fuel burnup.

During the entire period of its operation – as of the end of 2017, it's more than 265,000 hours in critical state. BN-600 produced more than 147 billion kilowatts of electrical energy, making a notable contribution into the Urals power supply as one of the most cost-effective and eco-friendly power units.

Amount of gaseous radioactive products emission to the atmosphere, as a rule, does not exceed 1% of the acceptable level. Amount of solid and liquid radioactive waste is also minimal, and it doesn't exceed 50 cubic meters per year.

Personnel radiation exposure is lower than the average level existing in the nuclear industry. One of the most important results obtained during the BN-600 operation is the fact that the design parameters for sodium large-scale equipment operation period and lifetime have been achieved and even exceeded. During the period of industrial operation, the BN-600 reactor demonstrated high safety and reliability characteristics and thus solved its task which was to industrially justify the reliability and safety of the SFR technology and specifically to the technology of sodium coolant.

The 30-year design lifetime of BN-600 was to be over in 2010. In 2005 till 2010, the activities were performed to extend the BN-600 lifetime for 10 years, up to March 31, 2020. It should be noted that the obtained results showed there is a possibility to exceed the BN-600 lifetime to 45 years. Currently, the activities are underway to extend the BN-600 lifetime beyond the year 2020. Preliminary results prove the principal technical possibility of extending the BN-600 reactor lifetime to 60 years. It depends only on the replacement equipment.

Now I'll tell you about BN-800 reactor power unit construction and its operation. One of the main objectives that have to be solved in the course of BN-800 operation is to demonstrate a closed nuclear fuel cycle. An arrangement of the closed nuclear fuel cycle with an SFR will indicate mastering the entire set of SFR technologies. Over the long term, this will address the issues of expanding the nuclear fuel base and disposal of spent nuclear fuel, including minor actinides.

This figure shows a general scheme of a nuclear steam-supply system with BN-800. Currently, a hybrid core is used in the BN-800 reactor. It includes both MOX-fuel and uranium dioxide-fuel. Here you can see, the first one is the reactor. It's the reactor vessel, the main reactor vessel, and the guard reactor vessel.

Intermediate heat exchanger. Here you can see the steam generator. Our steam generators has two modules, evaporator and superheater. Here, number 2 is the core. The 13 is the air heat exchanger for our decay heat removal system.

This figure illustrates the longitudinal cross-section of the BN-800 reactor. You can see here the main vessel, the guard vessel. You see the core. This one is intermediate heat exchanger. The core diagrid in the bottom of the core. And the 10 is refueling mechanism.

The next three slides will be the basic technical parameters of the BN-800. We can see here the reactor thermal power is 2100. Unit net efficiency is 40%. You see the parameters of the coolant temperature are very close to BN-600, the outlet is 547. The primary sodium, they are only 1000 tons.

And we have the same scheme in BN-800 like in BN-600. We have 3 circulation pumps and 3 loops. Here you can see the diameter of the core is 2.5 meters and the core height is less than 1 meter. It's not as high as BN-600, but the diameter is much more than in BN-600. We used, as I said, the different types of fuel. We used MOX fuel and dioxide-uranium fuel.

The reactor main vessel is quite big. It is almost 13 meters in diameter and 14 meters in height. It's stainless steel. In the secondary loop, we have steam generators. The temperature of sodium in steam generator, inlet, 505 degrees, and outlet 308 degrees. We have two decay heat removal air cooling sections. We have decay heat removal system in each loop. In BN-600 we have 8 steam generator sections; in BN-800 we have 10 steam generator sections in each loop. We call all these 10 sections like one steam generator. We don't call that the steam generator is only one metal.

Here we can see the principal stages of BN-800 construction, commissioning and operation. The BN-800 reactor design is to a significant extent a logical development of the BN-600 reactor and contains its main design, scientific and engineering solutions. Nevertheless, the BN-800 design has a number of conceptually new things that differentiate it from the BN-600 reactor. The principal differences are the following. A passive emergency shutdown system with hydraulically suspended rods; a special sodium cavity over the core to reduce sodium void reactivity effect; a core catcher in the bottom of the reactor vessel to collect and retain core debris; a decay heat removal system is connected to the secondary circuit, bypassing the steam generator in each loop. A as I said, one turbine generator for all the three heat-removal loops. And in steam generator section, a reheater module is eliminated so each steam generator section comprises of an evaporator module and a primary superheater module.

Here you can see the principal stages of commissioning BN-800. In June 2014, BN-800 had the first criticality. In July 2015, completion of the first criticality stage. This year, at the end of 2015, the onset of the power start-up stage; first connection to the turbine generator to the grid. In February of 2016, we completed the power start-up stages and onset of the pilot operation stages. In September 2016, we completed the pilot operation stages and the preparation for power unit commissioning. In October 2016, BN-800 power unit commissioning. During the whole period of its operation, the BN-800 reactor was critical for 14,000 hours, having generated over 9.4 billion kilowatts/hour of electrical power. The average capacity factor during the whole period of its operation till the end of 2017 was equal to 59.83%, with this value equal to 71.82% in 2017.

In these figures, you can see the principal stages of BN-800 construction and commissioning from 2008 till the present day. You can see the view of reactor pit construction. Here, the mounting of the reactor vessel bottoms. Beginning of fuel assembly loading. You can see the main control room of the BN-800. And just a view of the power unit with BN-800 in the day and at night

What are the prospects for further sodium fast reactor development in Russia? In compliance with further objectives in the development and improvement of sodium fast reactor technologies, the following stages can be highlighted. Demonstration of BN-type reactor closed fuel cycle in BN-800. Sodium fast reactor technology commercialization stages; designing and construction of BN-1200. And large-scale sodium fast reactor technology development stages, construction of a small series of commercial power units with BN-1200 reactors.

Summarizing. The overview of the experience in the operation of power units with BN-600 and BN-800 reactors and particularly the results of successful and stable operation of the third power unit at the Beloyask Nuclear Power Plant presented in these slides makes it possible to draw a conclusion about the industrial development of sodium fast reactor technology and in particular sodium technology. The experience gained in the course of BN-600 operation formed the basis for designing high-power sodium fast reactor BN-1200.

Thank you for your attention. If you have any questions, I'll be glad to answer them. You also can ask me by my email.

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