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Review Article

The BFS complex – a unique facility to justify the neutronic parameters of the new generation fast reactor cores

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Abstract

The BFS complex comprising two fast critical facilities – BFS-1 and BFS-2 – is a unique experimental base for research into fast reactor physics, reactor safety, core optimization, justification of the closed fuel cycle parameters. The critical facilities have the same pitch of the core lattice, they are loaded with the same materials for core simulations but they differ in size. Over 60 years of the BFS operation, IPPE specialists have gained considerable experience in operating the facilities and carrying out experiments. More than 150 critical assemblies have been studied in BFS.

Keywords

BFS complex, reactors, fast critical facilities, neutronic parameters, full-scale simulation, refining neutron data, verifying computer codes, benchmark

Brief description of the BFS complex

Paragraph 2.1.4 of the "Nuclear Safety Regulations" (Blair Briggs J et al. 2012) stipulates test and reactor modelling for justification of both new reactor projects under development and operating reactor upgrade projects. There is a similar stipulation in the regulations of other countries. The USA, France, Japan and certain other countries had critical facilities for modelling fast reactors. However, all of them are now decommissioned, and, for the time being, the BFS complex of critical facilities (the BFS CFC) belonging to IPPE is the only experimental facility in the world to offer the possibility of full-scale simulation of different fast reactor mock-ups (and not of fast reactor mock-ups only) (Dulin VA 1979; Yoo J et al. 2012; Kazansky YuA et al. 1977).

The BFS CFC allows full-scale simulation of reactor cores with different types of nuclear fuel (metal, mixed oxide, nitride, with added minor actinides (MA)), with different types of coolant (sodium, lead, lead-bismuth, water, etc.), with different control rod materials.

New techniques for measuring the neutronic parameters (for the benefit of high-power new generation reactors as well) are developed and implemented at the critical facility complex (Kazansky YuA et al. 2012; Matveenko IP et al. 2012).

Besides, the BFS CFC can be used to conduct experiments for refining neutron data and verifying computer codes.

Numerous test instrumentation systems of the BFS CFC help to measure a lot of parameters that cannot be measured in the real reactor. These parameters include, for example, the dense (void) reactivity effect of the coolant.

The key technical parameters of the BFS critical facilities are given in Table 1.

Table 1. Key technical parameters of the BFS critical facilities

Parameter	Value
Power (max), kW	
BFS-1	0.2
BFS-2	1.0
Simulated coolant	Na, Pb, Pb-Bi, etc.
Reflector	U, UO_2, Pb, Pb-Bi, steel, etc.
Fast neutron flux density, (max.), n/(cm^2·s)	
BFS-1	up to 10^8
BFS-2	up to 10^9
Core cooling	Natural convection or forced air cooling

The BFS-1 critical facility (CF) is designed to study neutronic characteristics of future fast reactors and to assemble benchmarks. It is also possible to simulate light water reactor cores by using water or, instead of water, polyethylene disks. Nuclear fuel storages and technological processings of nuclear fuel cycle can be modelled as well. The BFS-1 facility achieved its first criticality on February 2, 1962.

The vessel of BFS-1 is a vertical steel tank which is 2 m in diameter and 2.7 m in height. The size of the tank allows simulating full-scale mock-ups of up to 1000MW future research and power fast reactors with various core and blanket layouts. The reactor vessel provides for a thermal column and a metal column. The former is used for thermal calibration and the latter – for simulation of the neutron shielding and other reactor zones away from the core centre. At the base of the vessel there is a spacer grid, which is a steel plate 100 mm thick. It has openings which are 35 mm in diameter and arranged in a triangular lattice with a pitch of 51 mm. The vessel is filled up with steel or aluminum tubes (~ 1500 in number) 50×1 mm in diameter, whose shanks enter the openings of the spacer grid. The tubes are filled with disks of fuel, fertile, structural materials and coolant. The number of the disks, their proportion and order are the same as in the cores, fertile blankets and reflectors of simulated reactors.

Some of the tubes placed in the central part of the vessel have pulley drives and act as safety, shim or control rods in the core. These tubes help simulate CR mock-ups. Their composition (a set of disks containing reactor materials in the core and blankets) is similar to the composition of the surrounding tubes in the core.

The BFS-1 critical facility is used to:

- conduct research for justification of design parameters and safety of fast reactors (and other types of reactors) with various types of coolant;
- develop and introduce new techniques for specifying neutronic characteristics of the designed advanced reactors;
- perform benchmark experiments for validation of neutron data and codes for calculating neutronic characteristics of advanced reactors and their fuel cycle parameters. The mock-ups of fast IBR-2, BOR-60, BN-350 reactors designed in the USSR were studied in this facility.

The BFS-2 critical facility is designed to study large-size fast reactors. The BFS-2 CF achieved its first criticality on September 30, 1969.

The BFS-2 CF is structurally identical to BFS-1, but it is of a larger size, which makes it possible to assemble high-power reactor mock-ups (thermal power up to 3000 MW). The vessel is 5 m in diameter and 3,3 m in height, the number of tubes in the vessel is about 10000. The tubes have the same diameter as those at the BFS-1 facility and they are filled with the same (as at BFS-1) disks containing reactor materials. Next to the reactor vessel, there are some volumes which make it possible to simulate reactor shielding and to perform a number of additional measurements.

The BFS-2 critical facility is equipped with a coordinate manipulator used for shuffling the tubes in the vessel, rearranging samples and detectors within the critical assembly in the automatic control mode and for operation in the oscillation mode.

BFS-2 was used for research studies on the mock-ups of BN-600, BN-800, BN-1600 reactor cores and fertile blankets (with uranium oxide and MOX fuel) and for research into fast reactor mock-ups with inserts of alternative fuels.

The stock of disks used for simulations at BFS-1 and BFS-2 is the same and includes:

- fissile materials (uranium metal and its oxide of enrichment to 36% and 90% U-235, plutonium metal of two isotopic compositions);
- fertile materials (uranium metal and its oxide – depleted in the U-235 isotope, depleted uranium nitride, natural uranium oxide, thorium metal);
- minor actinides (samples of their various isotopes, disks containing Np-237 oxide);
- structural materials (a wide range of steels and different metals);
- coolants (sodium metal, lead, lead-bismuth, polyethylene for water simulation);
- absorbers (boron carbide – natural and enriched in the B-10 isotope, neutron absorbing metals and their compounds);
- moderators and inert materials (graphite, polyethylene, etc.).

The BFS critical facilities are equipped with experimental and auxiliary devices including:

- automatic remote controlled manipulators (which help to implement the oscillation mode);

- devices for synchronous insertion of CR mock-ups into the core;
- devices for heating samples to 600 °C;
- devices for forced core cooling;
- mechanical devices for fixing various detectors and for shuffling them during experiments.

To make measurements at the BFS complex, there is a wide range of techniques and devices, of which the main are:

- a multichannel wide-range digital reactivity meter; corresponding techniques and their practical evaluation were cross-checked in international experiments (a modification of the reactivity meter was applied in BN-600);
- a set of fission chambers of different geometries, with a wide range of fissile isotopes, allowing to perform measurements of fission reaction rate distributions and spectral indices in the reactor core, reflector and shield;
- a variety of samples for oscillatory measurements (in the cold and heated (to 600 °C) states);
- a set of foils and samples for neutron activation analysis;
- a variety of sources of neutron, gamma and other kinds of radioactive radiation and their detectors;
- systems for detecting various kinds of radiation.

Possible types of critical assemblies in the BFS critical facilities:

- fast reactor mock-ups;
- mock-ups of reactors with a different neutron spectrum;
- benchmarks for testing neutron data;
- benchmarks for justifying nuclear safety;
- benchmarks for developing techniques for measuring neutronic parameters.

Research fields in the BFS CFC and their main results before the start of retrofitting and upgrading

From the latter half of the 1980s on, experiments carried out in the BFS CFC have involved diverse research studies not only in support of Russian reactors (funded by Russian customers), but also as part of international cooperation including ISTC projects and bilateral international contracts.

As for sodium-cooled fast reactors (e.g. BN-800 project, etc.), special attention was paid to the justification of parameters for cores with a sodium plenum (a series of BN-800 reactor mock-ups and benchmarks to find an acceptable value of SVRE), to the simulation of accidents and to the measurement of CR worths:

- BFS-44 (1981) – the first mock-up of BN-800, BFS-39 (1985) – the BN-1600 mock-up;
- BFS-46, BFS-46, BFS-50, BFS-52 – BN-800, BN-1600 heterogeneous cores;
- BFS-54, BFS-56 – benchmarks for tackling the sodium plenum issue;
- BFS-66 (2002) – an insertion of nitride fuel in the centre.

Recent years – BN-800 mock-ups with MOX fuel and a hybrid core.

Starting from the BFS-61 assembly (a benchmark model of a lead-cooled fast reactor with mixed nitride fuel, 1991), a series of assemblies were studied in support of the BREST reactor project:

- BFS-77 (1999);
- BFS-64, BFS-95 (2002).

To justify the project of the SVBR reactor with lead-bismuth coolant, the following critical assemblies were mounted:

- BFS-85, BFS-87 (2000) – benchmarks, BFS-107 (2012) – SVBR mock-ups.

As for research on other types of reactors, the following are worth mentioning:

- the high flux research reactor BRV-150 (benchmark model) – BFS-51 (1987);
- a benchmark model of a fast reactor cooled with water and steam – BFS-105 (2008);
- evaluation experiments for GCR – BFS-91.

Regarding the problem of weapon-grade plutonium utilization (in accordance with the RF-USA agreement), the critical facilities were used to:

- justify a hybrid core with MOX fuel for the BN-600 reactor on a series of critical assemblies BFS-62, BFS-66;
- justify the use of MOX fuel for VVER reactors on a series of the BFS-93 assemblies;
- justify the safety of plutonium geological disposal on BFS-79 (1999), BFS-81 (1999) (under the contract with the USA).

To study the neutronic parameters of the cores with fuel in an inert matrix (without fertile material), the following critical assemblies were mounted:

- BFS-58 for a fast BN-800 reactor;
- BFS-91-1 for a fast reactor with MOX fuel;
- BFS-91-2 for a reactor with a softer neutron spectrum.

The new ideas of using novel materials in the cores led to benchmark experiments:

- with plutonium metal cores – BFS-55-1 and BFS-55-2 (Zr volume fraction of up to 40%);

- with steel components in the blankets – BFS-66 (9 options of axial reflectors – Ni, Cr, Fe, Zr, U-met, UO_2).

Lack of experimental data on the nuclear fuel cycle safety (contracts with the USA and France) called for criticality studies on wet MOX fuel – BFS-97, BFS-99 (8 options).

In 2010 the Federal Target Programme called "Next-Generation Nuclear Energy Technologies for the period from 2010 to 2015 further extended to 2020" (NGNET FTP) was launched. As part of this programme, a wide range of research studies on advanced fast-neutron reactors was planned and conducted, from the projects of research reactors and small power reactors (MBIR, SVBR-100) to commercial high-power reactor projects (BN-1200 and BREST-1200) (NP-082-07 2008). All of these new projects were supposed to include research into neutronic parameters of the reactor cores, provided by experiments at the BFS critical facilities. Let us enlarge on some of them.

MBIR

The basic design of the MBIR research reactor provides for the start-up core configuration with MOX fuel, a steel reflector and an in-core storage with boron shielding, a central loop channel, two peripheral loop channels, three experimental channels, twelve material test assemblies and a CPS system including control rods from highly enriched and natural boron carbides. An important feature of the MBIR reactor core that other fast reactor developments have never offered before is that for every three fuel assemblies in the core there is one empty assembly (an experimental channel that can be loaded with fuel or structural materials when necessary). Another important feature is the unspecified contents of the experimental cells. These can be loaded with structural and fuel materials, coolants, absorbers, etc. Hence, there arose a need for verification of calculation programs for such cores. The BFS structure and the available materials allow the simulation of MBIR's core material composition, the radial blanket and the shield layouts. Experimental studies on MBIR's core mock-up were carried out at the BFS-1 facility in two stages. During the first stage, measurements were made to study the reactor state at the start of the fuel lifetime when shim rods were partially inserted (the BFS-111-1 critical assembly). During the second stage, studies were conducted on the BFS-111-2 critical assembly that simulated the state of the MBIR reactor at the end of the fuel lifetime when shim rods were withdrawn from the core. The core geometries of the two critical assemblies were kept the same.

SVBR-100

The development of the basic design of the SVBR-100 reactor with lead-bismuth coolant, a uranium core, an internal flow-through lead-bismuth radial reflector, an outer steel reflector, boron carbide shielding, a cluster of CPS rods from highly enriched boron carbide required an additional detailed justification of physical parameters and computer code uncertainties. Studies were carried out in two directions. First, a simplified benchmark model of the SVBR-100 reactor (the BFS-107-1 critical assembly) was mounted and studied at the BFS-1 critical facility to justify its physical parameters which are necessary for refining the nuclear data component in calculations. Then, full-scale mock-ups of the reactor core at the start and at the end of the fuel lifetime were mounted at BFS-2 (the BFS-80-1 and BFS-80-2 critical assemblies). Data required to verify the calculations of the reactor core design parameters were obtained from these experiments.

BREST OD-300

Before the completion of retrofitting and upgrading works at the BFS CFC, the stock of the disks with depleted uranium mononitride did not exceed 500 kg, which is not enough for full-scale simulations of the BREST reactor. That was why preliminary experiments were first conducted on benchmark models whose compositions and spectral characteristics were close to those of the simulated reactor. These benchmarks were used for experimental research required for refinement of neutron data and verification of computer codes developed within the scope of the NGNET FTP for reactor facilities with liquid-metal (lead) coolant and nitride fuel. There were two stages of research. At the first stage, the BFS-113-1A critical assembly was studied. Its core consisted of two parts: a central zone with oxide uranium-plutonium fuel and a driver with oxide uranium fuel. At the second stage, the BFS-113-1B critical assembly was studied. The core of this critical assembly consisted of three parts: a central zone with nitride uranium-plutonium fuel, a second zone with oxide uranium-plutonium fuel and a driver with oxide uranium fuel. The BFS-113-1C critical assembly was mounted to study the effect of the change in Pu isotopic composition in the central subzone, which consisted in the replacement of low-radiation Pu disks in the fuel tubes of the central subzone with high-radiation Pu disks, the rest of the disks in the fuel tubes of this subzone remaining unchanged.

Neutronic parameters studied on different critical assemblies at the BFS critical facilities throughout their operation are presented in Table A1 of the Appendix.

Experience of international cooperation

Starting from the 1990-s, the BFS-1 and BFS-2 critical facilities have been intensively used for international cooperation (under bilateral contracts):

- France, CEA. The BFS-67, BFS-69, BFS-71 critical assemblies (up to 14% Np in the fuel, different Pu fractions) (Rimpault G et al. 2012);

- PRC, simulations of the CEFR reactor – BFS-65 (1992), BFS-83 (2000), BFS-119 (2019);
- USA – weapon-grade plutonium utilization, creation of the materials protection, control and accounting systems at BFS;
- Japan – benchmarks with sodium plenum, hybrid core of the BN-600 reactor;
- Republic of Korea – mock-ups of the KALIMER reactor – BFS-73 (1997), BFS-75 (1999); PGSFR – BFS-84 (2015); SFR –BFS-76 (2010–2011) (Zaetta A 2012), BFS-109 (2012);
- India – a series of the BFS-24 critical assemblies, the result analysis.

A number of joint experiments were performed, and evaluation of the experiments performed earlier were evaluated:

- with France and Japan, – measurements of the effective delayed neutron fraction on various critical assemblies;
- with France and other participants in the project – measurements of the fission reaction rate distributions by using fission chambers at the MASURCA facility;
- with France and the USA – reactivity measurements under conditions of the strong impact of spatial effects;
- contracts with ISTC (over 15 assemblies);
- the International Criticality Safety Benchmark Evaluation Project (12 assemblies);
- the International Reactor Physics Experiment Evaluation Project (11 assemblies);
- international experiments on comparison of techniques for reactor performance measurements (foreign participants – Japan, France; critical assemblies apart from BFS – MASURCA, FCA).

From 1994 on, much work has been done to improve safety of handling nuclear materials and operating the BFS on the whole as part of US/Russia Materials Protection, Control and Accounting (MPC&A) Programme. A technology for putting individual marking on inventory items containing nuclear materials was developed and introduced. The technology allows continuous control of the handled inventory items by means of computer technologies. For this purpose, a quick detector of nuclear materials in the inventory item is used. It is combined with a sensor-identifier of the inventory item, which sends information to the computer database in the online mode. This significantly mitigates human error in the MPC&A system operation. The introduced technology makes it possible not only to identify inventory items automatically (barcode scanners of the identifying numbers, with the data transferred to the computer database), but also to perform simultaneous verification measurements of a given set of nuclear material characteristics in the inventory item.

For the safety enhancement purpose, the so-called "enhanced safety island" was arranged around a group of buildings including the BFS complex, equipped with engineering controls and physical protection means.

Over the past decade, international cooperation has had a new lease of life:

KAERI, the Republic of Korea

Several experimental programmes were implemented on the BFS critical assemblies to simulate the Korean fast breeder reactor SFR with metal fuel (uranium-zirconium or uranium-plutonium-zirconium alloy) and sodium coolant, its power ranging from 100 to 300 MW (e). They were full-scale core mock-ups of SFR-300 (BFS-76), SFR-100 (BFS-109), PGSFR (BFS-84) reactors. Since the reactivity effects due to the core thermal expansion are important for reactors with metal fuel, the standard experimental programme was supplemented with the measurements of the reactivity effects due to the core axial and radial expansion. Spectral indices of minor actinides (neptunium, americium and curium) were measured as well. The obtained large experimental material was used by the Korean side for licensing the project of the PGSFR reactor facility.

CEA, France

Series of experiments were carried out under the Implementing Agreement on core physics in the field of sodium-cooled fast reactors.

Critical assemblies were mounted for simulating MOX-fuelled cores whose spectral characteristics were close to those in ASTRID and BN-1200 reactors, aiming at optimized cores as regards the sodium void reactivity effect (SVRE). The purpose of the research was to obtain experimental data that would help extend the range of the validated application of computer codes used in designing future fast reactors. They are integral, high-power, plutonium-fuelled SFRs designed to achieve a high conversion ratio with radial steel reflectors and a low, possibly even negative, SVRE. Of particular interest are the radially-stretched, pancake-shape cores (H/D < 0.35), with a low sodium-to-fuel volume fraction, a sodium plenum region just above the fissile column, and an axially-heterogeneous fuel sub-assembly structure with alternating fissile and fertile sections. In fact, all the critical assemblies were benchmarks, with simple one- or two-zone cores. Two of them (BFS-82-2 and BFS-82-3 assembled at BFS-2) were benchmarks of the core whose central part simulated MOX fuel and was surrounded by a driver zone with uranium oxide. They were used for developing the experiment procedure and assessment methods. The third critical assembly, which was mounted at BFS-1, was a one-zone benchmark of the ASTRID reactor, with MOX fuel. It had several modifications (BFS-115-1, BFS-115-2 and BFS-115-3), with different plenum sizes and with insertion of an axial depleted UO_2 plate of variable axial position and thickness. The most extensive experimental programme was performed on this series of critical assemblies. It included measurements of radial and axial local SVREs, axial fission reaction rate distributions of the main fissile and fertile isotopes, boron capture reaction rate distributions, axial fission neutron importance distributions, spectral indices, boron absorber rod worth at the core centre. The fourth assembly (BFS-117-1) was the last in a series. It was a one-zone benchmark of the ASTRID core without the fertile plate.

CIAE, the People's Republic of China

A high priority for China is commissioning of CEFR with all-MOX core, which is to be followed by construction of the MOX-fuelled CFR-600 reactor seen as a commercial demonstration fast reactor. The CEFR core mock-up was assembled at BFS-1. Two full-scale core mock-ups were studied: for the start of the fuel lifetime (BFS-119-1) and for the end (BFS-119-2).

Retrofitting and upgrading of the BFS CFC

The BFS-1 and BFS-2 critical facilities successfully operated for half a century deserved to be retrofitted and upgraded. The retrofitting and upgrading works were carried out in 2012–2016 as part of the NGNET FTP and included:

- renovation of the BFS complex building, including the nuclear materials storage, and upgrading of the engineering systems of the critical facilities;
- supply of new materials (including nuclear and structural materials, absorber and coolant materials);
- upgrading and replacement of the scientific equipment.

As for the engineering systems of the complex, of special mention are complete replacement of the CPS equipment, upgrading of the materials protection, control and accounting systems, replacement of the complex radiation monitoring and self-sustaining reaction emergency alarm systems, replacement of the ventilation and radioactive drain systems.

The task of modelling the new generation fast reactors (both full-scale mock-ups and various benchmarks) required fabrication of a large number of hermetically sealed disks containing:

- plutonium metal (in addition to the available reserves);
- uranium nitride;
- enriched boron carbide;
- sodium metal.

The new disks allow simulations of almost all new generation cores, with any type of fuel and coolant and various CPS rod systems.

Future prospects for the BFS CFC in 2022–2025

Following the launch of the Integrated DETS Programme (programme for the development of equipment, technologies and scientific studies in the field of nuclear power uses in the Russian Federation for the period up to 2024), a wide range of research studies will be conducted at the BFS CFC on full-scale core mock-ups of the BREST OD-300 reactor with dense mixed nitride uranium-plutonium fuel (MNU-Pu) and BN-1200 with both MNU-Pu and MOX fuel. At that, new, radially-stretched, pancake-shape core configurations with an axial layer of fertile material are expected to be studied.

As part of the above-mentioned programme, neutronic parameters of advanced pressurized water reactors with supercritical coolant parameters will be studied and justified.

Research to justify the nuclear safety of sodium-cooled fast reactors in beyond-the-design basis severe accidents may be one of the important future tasks for the BFS complex, too. A number of experimental programmes under international contracts are going to be carried out as well.

Conclusion

The BFS CFC consisting of two critical facilities for simulating fast reactor cores varying in size is one of a kind complex, for the simple reason that all the foreign counterparts are decommissioned. The completed retrofitting and upgrading works proved a milestone in the operation of the critical facilities, offering a possibility to justify experimentally the neutronic parameters of the new generation fast reactors in the first place. Available nuclear materials allow for simulating full-scale core mock-ups with various types of fuel (metal, mixed oxide, mixed nitride, with added MA), with various types of coolant (sodium, lead, lead-bismuth, water, etc.), with different control rod materials. The critical facilities have all the necessary auxiliaries and instrumentation for conducting experiments. Measurement techniques have been verified mane times in joint international experiments, in the presence of experts from France, the USA, Japan.

Technical capabilities of the facilities make it possible to simulate not only fast reactor cores but also cores and benchmarks with other spectral characteristics.

The unique capabilities of the BFS complex have been effectively used for many years in the international cooperation through bilateral contracts (France, China, the USA, South Korea, Japan, India) or multilateral agreements (ISTC, NEA OECD).

It is not possible to avoid mentioning most valuable experience of teaching students and pesonnel (from abroad as well) reactor experiment techniques and measurement techniques that are specific to the operation of the critical facilities (reaching critical conditions, passportization of critical assemblies, materials control and accounting).

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Appendix 1

Table 1A. Studies performed at BFS

Number of the critical assembly	Index of the critical assembly modification	Year	Simulation object	Experimental research											
				Criticality	Fission Rate Distribution	Void Reactivity Effect	Control Rod Worth	Spectral Indices	Central Reactiv. Coefficients	Minor Actinides	Delayed Neutrons Fraction	Doppler Reactivity Coefficients	Spectrum	Neutron Lifetime	Miscellaneous
BFS-1															
1		1961	BFS-1	+											
3		1962		+	+			+	+				+	+	
8	A	1963		+										+	
9	A	1963	Assembly with Be	+	+			+	+						
10	A,B,C	1963		+	+										
11		1963		+	+		+	+	+					+	
12		1963		+	+			+	+			+			
14	-1,2–7	1963–1964	BOR-60	+	+		+	+	+					+	
15	-1,2,3A	1964	OK-500 (BN-250)	+	+		+	+	+				+	+	
16	-1	1965	BN-350	+	+		+	+	+	+		+	+	+	
17	-1,1o	1966–1967	BN-350	+	+		+					+	+		
16	-1,2,3	1967	BN-350	+	+			+				+	+		
18	-1,2,3	1968	IBR-2	+	+									+	
20	-1	1968	IBR-2	+	+			+	+					+	
21		1968	BOR	+	+	+	+	+	+	+					8
22		1968	BN-350	+	+	+	+	+	+				+	+	
23		1970	Insert of Pu (hr)	+	+				+			+			
25		1970	Insert of Pu (lr)	+											
26	-1,2,3	1971	Electron cyclotron	+	+			+	+				+		3, 6
27	-1,2	1972		+				+	+				+		
33	-1,2,3		SCHERZO-UO_2 -740	+				+							
35	-1,2,3	1974	SCHERZO –U-556	+				+							
45	A-1, B-1	1980–1982	BN-600 LEZ	+		+		+	+						
47	-1	1985	BN-600 LEZ	+			+								
49	-1,2,3,4		MOX with moderator	+											
51	-1	1987	BRV-150	+	+		+	+							
53	-1,1H, 2, 3	1986	Hybrid, metal-oxide	+	+	+	+	+							1, 7
55	-1,1A,2	1987–1989	U-Pu met + *Zr* + *Th reflector*	+	+	+	+	+	+		+	+			1, 2, 5

Number of the critical assembly	Index of the critical assembly modification	Year	Simulation object	Experimental research											
				Criticality	Fission Rate Distribution	Void Reactivity Effect	Control Rod Worth	Spectral Indices	Central Reactiv. Coefficients	Minor Actinides	Delayed Neutrons Fraction	Doppler Reactivity Coefficients	Spectrum	Neutron Lifetime	Miscellaneous
57		1989	VVER – tightly-packed – U	+	+		+	+	+			+	+		1, 3
59		1990	VVER – tightly-packed – Pu	+	+			+	+		+				1
61	-0,1,2	1990	Pb – benchmark	+	+			+	+	+	+	+	+		
63	-1,2,3	1992	Complex heterogeneity	+	+	+	+				+				3
65	-1,2,3	1992–1993	CEFR	+	+	+	+	+	+		+				
67	-1,2,3, 3B	1994–1995	SUPERPHENIX	+		+	+	+	+	+					
69	-1,2	1995	CAPRA	+	+	+	+	+	+	+		+			
71	-1	1996–1997	57% Pu	+	+	+	+	+	+	+					
73	-1	1997	KALIMER	+	+		+	+	+	+	+	+			
75	-1	1999	KALIMER	+	+	+	+	+	+	+		+			
77	-1	1999	BREST-300	+		+*		+	+	+					
79	-1,2–5	1999	Waste disposal	+	+				+						
81	-1,2–5	1999	Waste disposal	+	+			+	+						
83	-1	2000	CEFR	+											
83	-2	2000		+											
83	-3	2000		+	+	+	+								
85	-1,2	2000	Pb-Bi benchmark U	+	+			+							
87	-1	2000	Pb-Bi benchmark Pu	+	+			+			+				
87	-2	2000		+	+			+							
89	-1,2, 2A,3	2000	SSS with HLMC	+	+	+*	+		+						1
91	-1,2,3	2001	ROX-fuel, GCR	+				+	+	+		+			
93	-1,2–6	2002	MOX in VVER	+	+			+	+						
95	-1,2	2002–2003	BREST-300	+	+	+*	+	+	+	+	+				
97	-1,2–4	2004	MOX fabrication	+	+			+		+	+				
99	-1,2	2004	MOX fabrication	+	+			+		+	+				
101	M-1,2 2A,3	2005	MOX fabrication	+	+			+							9
103	-1,2,3	2005–2006	MOX-12,5%, conversion ratio in the core~ 0,8–1,1	+		+	+	+	+	+					
105	-1,2,3, 3A	2008	Pu-LR, modified spectrum	+			+	+	+	+					15
107	-1	2011	SVBR	+	+		+	+	+		+				14
109	-2A	2012	SFR	+	+	+	+	+	+	+	+	+			13
111	-1,2	2013	MBIR	+	+		+	+							12
113	-1,1A, 1B,1C	2013–2014	BREST-OD-300. UO2+PuUN+Pu	+	+	+*	+	+							9, 11
115	-1,2,3	2014	ASTRID	+	+	+	+	+							10
117	-1	2020	ASTRID	+	+	+	+						+		10
119	-1,2	2019	CEFR	+	+	+*	+	+			+		+		9
BFS-2															
24	-1,2–16	1971	BN-600	+	+		+								4
29	1,2–5	1973–1975		+	+		+	+	+				+		1, 20, 21
31	-1,2,3	1974	Core with Pu-UO2 fuel and K-infin.=1	+	+			+							17
34	-0,1	1975	Sodium-cooled fast reactor with UO2 fuel	+	+	+	+	+	+					+	1, 7, 24
39	-1	1977-1979		+	+	+	+	+	+						4, 17, 20, 22, 23
44		1979	BN-800 with Pu-UO2 fuel	+	+	+	+	+	+					+	1, 17, 18–19
46		1981	BN-1600 with heterogeneous Pu core	+	+			+							
48	-1,2–4		Non-full-scale model of BN-800	+	+		+								
50	-1,2–5		BN-800, UO_2, PuO_2, axial heterogeneity	+	+	+	+	+	+						1, 5
52	-1		BN-800, UO_2, radial heterogeneity	+	+		+	+	+				+		3 ,6 , 7
52	-2,V, V/1-2, B, B/2, B/2(M)	1988–1990		+	+	+	+	+	+					+	8
54	-1,2–4	1990–1992	BN-800 with Pu (HR) +Na plenum	+	+	+	+	+	+			+			1, 3, 8
56	-1,1A, 1B	1992	BN-800, mixed vibro-packed fuel, Na plenum	+	+	+		+	+	+					1, 16

Number of the critical assembly	Index of the critical assembly modification	Year	Simulation object	Experimental research											
				Criticality	Fission Rate Distribution	Void Reactivity Effect	Control Rod Worth	Spectral Indices	Central Reactiv. Coefficients	Minor Actinides	Delayed Neutrons Fraction	Doppler Reactivity Coefficients	Spectrum	Neutron Lifetime	Miscellaneous
58	-1, 1I, 2–4	1993–1997	BN-800, Na plenum, Pu fuel (without U), mixed vibro-packed fuel	+	+	+	+	+	+	+				+	16
60	-1,2	1997–1998	Burners of Pu and MA	+	+	+	+	+	+	+					15
62	-1,2–6	1999–2002	Hybrid core of BN-600	+	+	+	+	+	+			+			
64	-1	2002	BREST-OD-300	+	+	+*	+	+		+			+		7, 12
66	-1,B,2, 2A,3, 3A	2003–2006	BN-600, MOX fuel	+	+	+	+	+	+			+			
68	-1,2–4	2006	BN-600, partial simulations	+	+			+					+		
70	-1,2	2007	BN-600	+			+								14
72	-1,2,3, 3A,4	2008	BN-800	+	+	+	+	+		+					
74	-1,2	2009	BN-800	+	+										6, 8, 13
76	-1,1A	2010–2011	SFR	+	+	+	+	+	+	+	+				11
78	-1	2011	Hybrid core of BN-800	+	+	+	+	+							
80	-1,2	2012	SVBR-100	+	+		+	+					+		
82	-1,2,3	2012–2013	Benchmarks of BN and ASTRID	+	+	+	+	+							9
84	-1	2015	PGSFR	+	+	+	+	+			+				10, 11

Notes: Indices are fission reaction rate ratios that characterize the neutron spectrum (F^{238}/F^{235}, C^{238}/F^{235}, F^{239}/F^{235}, C^{Au}/F^{235}, $A^{Li\text{-}6}/F^{235}$, etc.).

Symbols in the rightmost column of the table:

* – void reactivity effect of heavy metal coolant; **1** – hydrogen reactivity; **2** – Al(n,a) reaction distribution; **3** – neutron importance; **4** – prompt neutrons damping decrement; **5** – prompt neutron lifetime; **6** – subcritical state; **7** – boron acid reactivity; **8** – shielding experiments; **9** – replacement of FR^{Pu} with FR^{U}; **10** – fission neutron importance distribution; **11** – Pu isotopic composition; **12** – radial distribution of rod worth; **13** – reactivity effect of FR thermal elongation; **14** – effect due to replacement of Pb-Bi reflector rods with steel reflector rods; **15** – methodological work on the reactivity meter; **16** – worth of BN-800 real fuel assemblies with mixed vibropacked fuel; **17** – cell reactivity; 18 – AlN(14,15) reactivity; **19** – measurements of subcriticalities; **20** – studies on the effects of small heterogeneity; **21** – reactivity of uranium of different enrichment; **22** – studies on the effect of the core cylinderization; **23** – worth of BN-600 subassemblies; **24** – worth of trap rods.