STATUS OF SFR DEVELOPMENT IN RUSSIA

Y.M. Ashurko

State Scientific Center of the Russian Federation –
Institute for Physics and Power Engineering (SSC RF-IPPE), Obninsk

Submitted to the International Workshop on Prevention and Mitigation of Severe Accidents in Sodium-cooled Fast Reactors, Tsuruga, Japan, June 11-13, 2012

GENERAL INFORMATION ON NPPs IN RUSSIA

- *In Russia there are in operation 33 nuclear power units located on 10 NPP sites.*
  - On 20 October 2011, first criticality of the 4th power unit of the Kalinin NPP was reached.
  - On 12 December 2011, it was put into trial operation at 50% of rated power.
  - Total electric power capacity of the all Russian NPPs (without the 4th Kalinin NPP power unit) is equal to 24.242 GWe.
- *Now more than 15 power units are under construction.*
- *Total electricity production by the NPPs in Russia in 2011 was 172.68153 billion kW·h (it was increased by 1.0% in comparison with 2010).*
- *In 2011, load factor of the Russian NPPs was reached 81.24% (81.34% in 2010).*
- *There were 44 events of abnormal operating conditions of NPPs in 2011, which are classified in accordance with INES scale as follows:*
  - 14 events were classified as below INES scale, i.e. “out of scale” cases;
  - 29 events were classified by "0" level;
  - 1 event was classified by "1" level.

LIST OF NPP POWER UNITS OPERATING IN RUSSIA (1/2)

No	Power unit	Reactor type	Year of commissioning	Installed electric power, MW_e
Operating power units				
1	Novovoronezh-3	VVER-440	1971	417
2	Novovoronezh-4	VVER-440	1972	417
3	Kola-1	VVER-440	1973	440
4	Leningrad-1	RBMK-1000	1973	1000
5	Kola-2	VVER-440	1974	440
6	Bilibino-1	EGP-6	1974	12
7	Bilibino-2	EGP-6	1974	12
8	Bilibino-3	EGP-6	1975	12
9	Leningrad-2	RBMK-1000	1975	1000
10	Bilibino-4	EGP-6	1976	12
11	Kursk-1	RBMK-1000	1976	1000
12	Kursk-2	RBMK-1000	1979	1000
13	Leningrad-3	RBMK-1000	1979	1000
14	Beloyarsk-3	BN-600	1980	600
15	Novovoronezh-5	VVER-1000	1980	1000
16	Leningrad-4	RBMK-1000	1981	1000
17	Kola-3	VVER-440	1981	440

LIST OF NPP POWER UNITS OPERATING IN RUSSIA (2/2)

No	Power unit	Reactor type	Year of commissioning	Installed electric power, MW_e
Operating power units				
18	Smolensk-1	RBMK-1000	1982	1000
19	Kursk-3	RBMK-1000	1983	1000
20	Kola-4	VVER-440	1984	440
21	Kalinin-1	VVER-1000	1984	1000
22	Kursk-4	RBMK-1000	1985	1000
23	Smolensk-2	RBMK-1000	1985	1000
24	Balakovo-1	VVER-1000	1985	1000
25	Kalinin-2	VVER-1000	1986	1000
26	Balakovo-2	VVER-1000	1987	1000
27	Balakovo-3	VVER-1000	1988	1000
28	Smolensk-3	RBMK-1000	1990	1000
29	Balakovo-4	VVER-1000	1993	1000
30	Rostov-1	VVER-1000	2001	1000
31	Kalinin-3	VVER-1000	2004	1000
32	Rostov-2	VVER-1000	2009	1000
33	Kalinin-4	VVER-1000	2011	1000
	Total			25242

FAST REACTORS

- *In Russia there are 2 facilities with fast reactors in operation:*
  - Test reactor BOR-60 at the SSC RF RIAR (Dimitrovgrad);
  - Commercial power unit No. 3 of the Beloyarsk NPP (BelNPP) with sodium cooled fast reactor BN-600 (Zarechny).
- *Research reactor BR-10 at the SSC RF – IPPE (Obninsk) is on the stage of preparation for its decommissioning.*
- *No. 4 power unit of the Beloyarsk NPP with sodium cooled fast reactor BN-800 is under construction.*

FTP "NUCLEAR POWER TECHNOLOGIES OF A NEW GENERATION" (1/3)

- *In accordance with approved by the Government of the Russian Federation the Federal Target Program (FTP) "Nuclear power technologies of a new generation for period of 2010-2015 and with outlook to 2020" activities is provided in area of:*
  - Sodium cooled fast reactors (development of the BN-1200 design);
  - Fast reactors with heavy liquid metal coolant (development of the BREST-OD-300 with lead coolant and SVBR-100 with lead-bismuth coolant) and
  - Correspondent to them fuel cycles.
- *The FTP is aimed at the development and construction of a new technological platform for nuclear power based on transition to the closed nuclear fuel cycle (CNFC) with fast reactors of the 4^{th} generation.*
- *In order to implement the transition to the new technological platform the FTP envisages the activities in the following directions:*
  - Development of advanced reactor technologies of the 4^{th} generation;
  - Construction of new test facilities and installations, upgrading and development of an experimental and bench-scale base in support and justification of the reactor technologies under development;
  - Development of the technologies for production of advanced types of fuel for reactors of the next generation;
  - Creation of materials and technologies of the CNFC for nuclear power systems with fast and thermal reactors of the new generation;
  - Development of the integrated code systems of a new generation for analyses and substantiation of safety of advanced NPPs and CNFC.

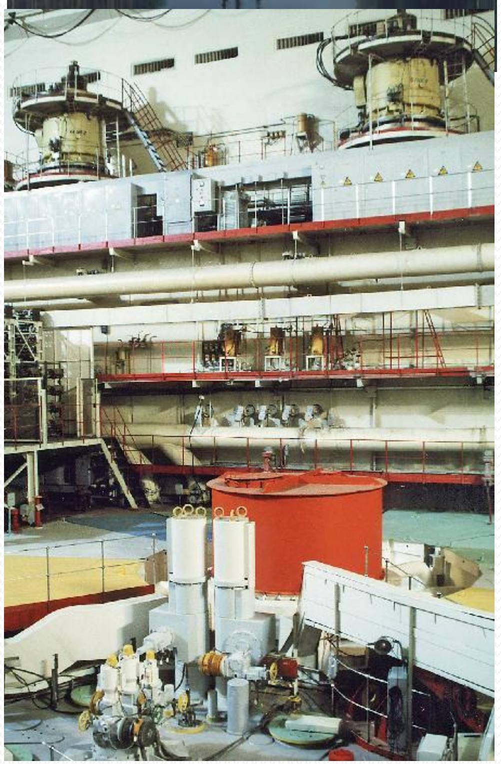
FTP “NUCLEAR POWER TECHNOLOGIES OF A NEW GENERATION” (2/3)

- *The FTP implementation should be realized in two stages.*
- *The first stage (2010-14) envisages performance of the following main works:*
  - Development of basic designs of the above-listed fast reactors of the 4th generation;
  - Completion of designing and commissioning of uranium-plutonium oxide fuel production plant for fast reactors of the new generation;
  - Development of a detailed design for construction of a multipurpose research fast reactor (MBIR) with sodium coolant;
  - Development of new radiation-resistant structural materials for the reactors of the new generation.

FTP "NUCLEAR POWER TECHNOLOGIES OF A NEW GENERATION" (3/3)

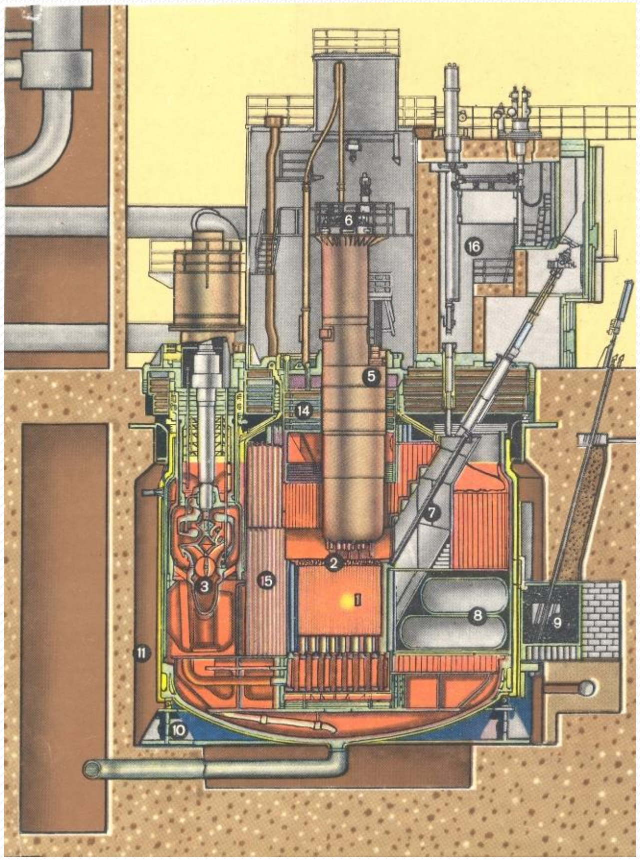
- *The most important works should be implemented at the second stage (2015-20) are as follows:*
  - Construction of demonstration and prototype facilities with the BREST and the SVBR reactors;
  - Set-up of a pilot plant for production of compact fuel for nuclear reactors of the new generation;
  - Construction of a demonstration semi-commercial pyrochemical complex for fuelling nuclear reactors of the fourth generation;
  - Construction, refurbishment, technical upgrading and commissioning the required research base purposed for the justification of the new technological platform of nuclear power, including the MBIR reactor construction and commissioning the technically reequipped complex of big test facilities (BFS).

BASIC OPERATIONAL PARAMETERS (1/2)



- *The industrial power unit No.3 of the Beloyarsk NPP with the BN-600 reactor, being the largest operating sodium-cooled fast reactor in the world, is under operation more than 32 years.*
- *As of 31.12.2011, the BN-600 power unit:*
  - had been in critical condition more than 220 000 hours;
  - had produced about 120.73 billion kW·h of electricity;
  - operated with average value of the load factor for the period of its commercial operation since 1983 equal to 74.57%.
- *Design lifetime of the BN-600 power unit equal to 30 years had been expired by April 8, 2010.*
- *A large program on the BN-600 power unit lifetime extension had been done in the following main directions:*
  - Work on substantiation of lifetime of non-replaceable reactor facility elements,
  - Complex of works on additional investigation and substantiation of residual operation life of the power unit’s systems and elements,
  - Replacement of the equipment.
  - Measures on enhancement of the power unit’s safety,
  - Development of the Report on Profound Safety Estimation of the power unit and a set of the proving documentation for reception of the license for additional period of operation.
- *On April 7, 2010, the Beloyarsk NPP had got the Rostechnadzor’s license for lifetime extension of the BN-600 power unit up to March 31, 2020.*
- *In 2011:*
  - 4249.84 million kW·h of electricity had been produced,
  - 287 Tcal of heat had been delivered for heat consumers from power unit’s collectors,
  - NPP load factor was equal to 80.87%.

BASIC OPERATIONAL PARAMETERS (2/2)

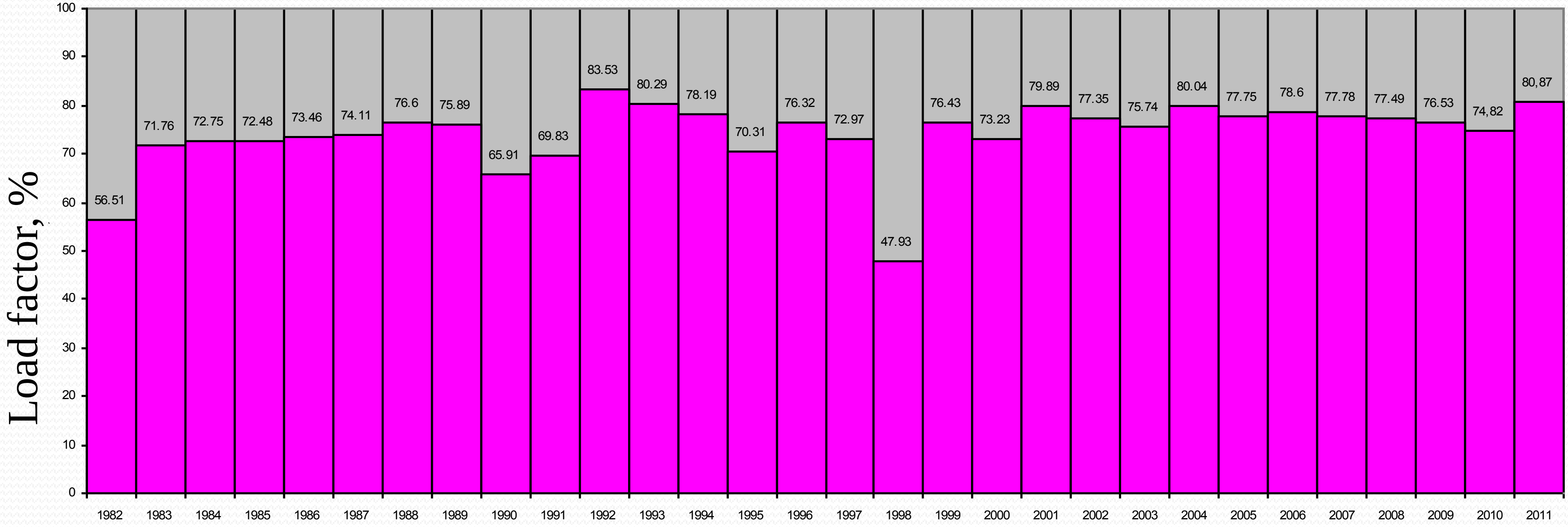


- *Within 2011, there were no unscheduled shutdowns of the power unit.*
- *There were 2 scheduled power unit shutdowns for refueling and preventive maintenance works (PMW) in 2011:*
  - Spring PMW – from 13.04 till 05.06.2011 (53 days);
  - Autumn PMW – from 22.10 till 06.11.2011 (14.4 days).
- *Two power decreases occurred during 2011:*
  - on 05-07.02.2011, switch-off of turbogenerator No.6 due to decrease of electrical resistance of insulator in the rotor's excitation chain of the generator No.6 ,
  - on 09.08.2011, decrease of the units's power up to 60% for elimination of leak in tube bundle of condensator of turbogenerator No.5.
  - Both events are classified in accordance with INES scale as "out of scale" cases.

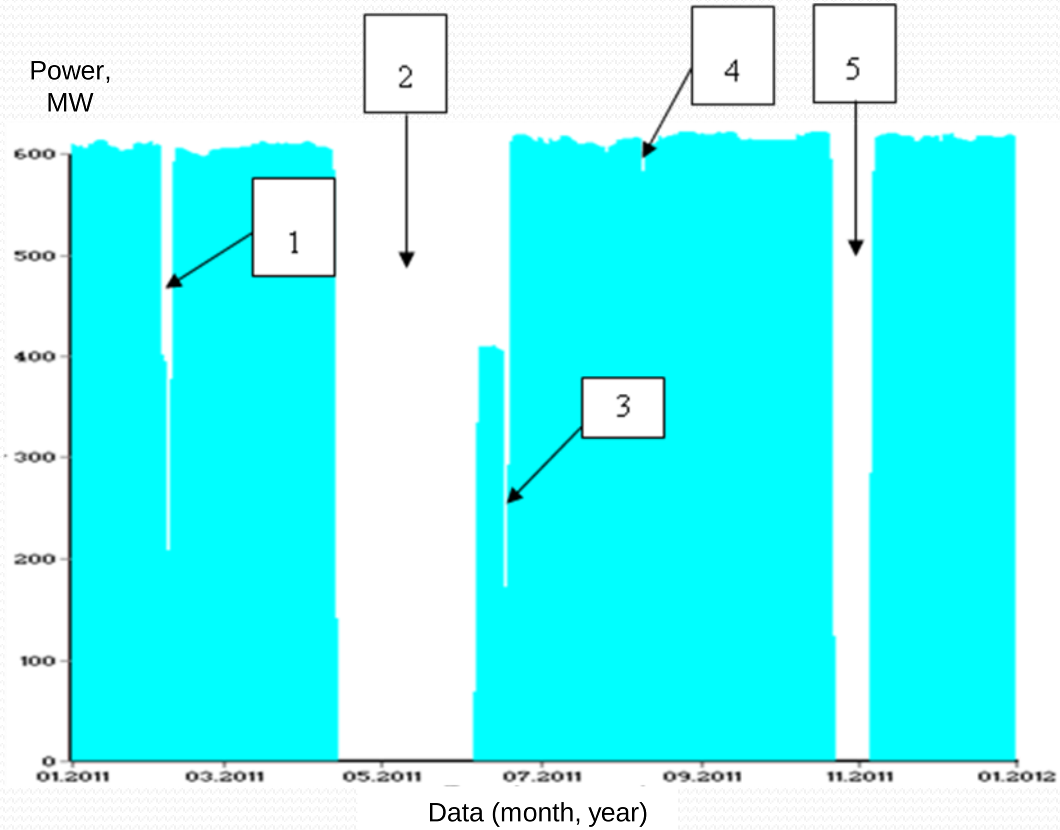
Basic parameters of the BN-600 power unit

Parameter	Value
Thermal power, MW	1470
Electric power, MW	600
Number of heat removal loops	3
Configuration of the primary circuit	Pool
Steam generator design	Once-through, sectional-modular
Maximum neutron flux density, $n \cdot cm^{-2} \cdot s^{-1}$	$6.5 \cdot 10^{15}$
Fuel	UO_2
Max. fuel burn-up, % h. a.	11.1
Inlet/outlet core coolant temperature, ºC	377/550
Inlet/outlet SG coolant temperature, ºC	518/328
Inlet/outlet SG water/steam temperature, ºC	241/507
Live steam pressure, MPa	13.2
Design lifetime, year	30

CHANGE OF LOAD FACTOR DURING BN-600 POWER UNIT COMMERCIAL OPERATION



LOAD MAP OF THE BN-600 POWER UNIT IN 2011



1 – switching-off turbogenerator No.6 due to decrease of electrical resistance of insulator in the rotor's excitation chain of the generator No.6;

2 – scheduled shutdown of the power unit for medium repair (from 13.04 till 05.06.2011);

3 – putting into operation of loop No.5;

4 – decrease of the units's power up to 60% for elimination of leak in tube bundle of condensator of turbogenerator No.5;

5 – scheduled shutdown of the power unit for routine repair (from 22.10 till 06.11.2011).

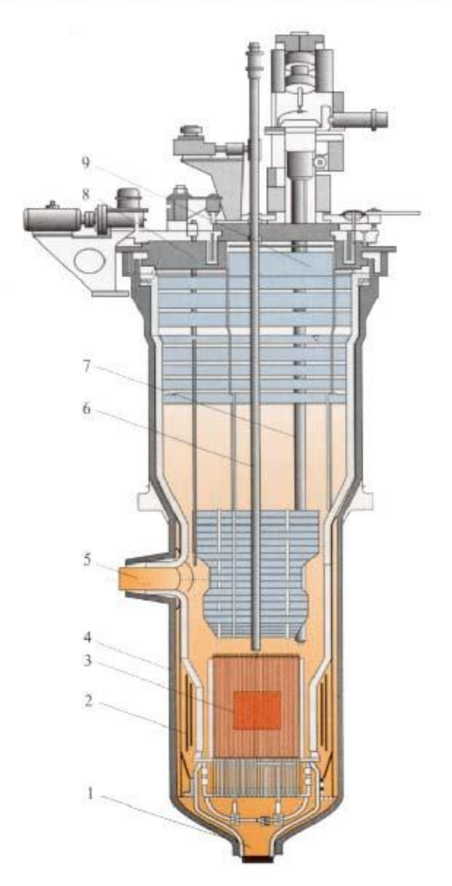
BASIC REPAIR AND MAINTENANCE WORKS CARRIED OUT IN 2011

- *Reactor refueling;*
- *Operating control of structural material of the spent fuel storage;*
- *Overhaul of generator-6 with replacement of the rotor's band;*
- *Overhaul of 3 feedwater pumps, medium repair of 1 feedwater pump, routine repair of 2 feedwater pumps;*
- *Replacement of 1 emergency feedwater pump with electric motor,*
- *Replacement of drainage pipes on the tertiary side in SG-4,*
- *Replacement of valves in the tertiary circuit in loop 4.*

BASIC OPERATIONAL PARAMETERS

- *The BOR-60 reactor is used for:*
  - Material tests;
  - Isotopes production ;
  - Tests of the various equipments of fast reactors;
  - Heat and electricity production.
- *BOR-60 reactor facility is in operation more than 42 years.*
- *Its operation had been permitted by 31 December 2009.*
- *In December 2009, Rostechnadzor has given the license to the RIAR for further operation of the BOR-60 reactor facility up to 31.12.2014.*

BASIC PARAMETERS OF THE BOR-60 REACTOR



BOR-60 reactor longitudinal section

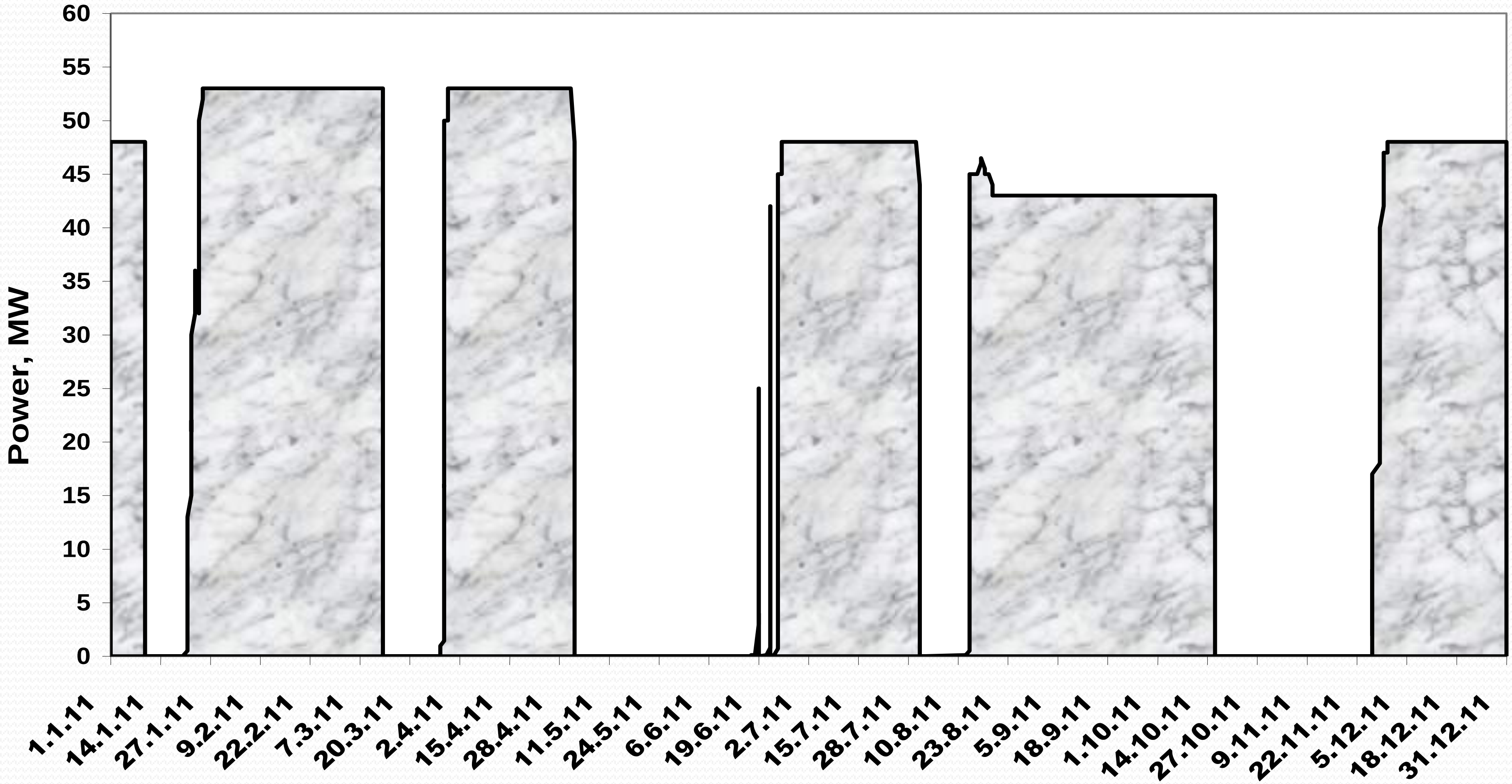
Parameter	Value
Thermal reactor power, MW	60
Maximum neutron flux density, $n \cdot cm^{-2} \cdot s^{-1}$	$3.7 \cdot 10^{15}$
Maximum core power density, kW/l	1100
Average core neutron energy, MeV	0.45
Fuel	UO_2, UO_2-PuO_2
Fuel burn-up rate, %/year	up to 6
Neutron fluence per year, $n \cdot cm^{-2}$	$5 \cdot 10^{22}$
Damage dose rate, dpa/year	up to 25
Inlet coolant temperature, ºC	310-330
Outlet coolant temperature, ºC	up to 530
Core run duration, days	up to 120
Reactor availability	up to ~0.73
Number of cells: for FSAs for absorber rods instrumented cells	265 156 7 3

INDICES OF BOR-60 REACTOR OPERATION IN 2011

Index	Value
Time of reactor operation on power exceeding minimum controlled l evel, h	5711
Reactor load factor	0.65
Max reactor power, MW	53
Energy output: heat, MW·h electricity, MW·h	 264165 38326
Time of SG operation: SG-1, h SG-2, h	 5633 5633
Heat delivery to consumers, Gcal	57063

- *In 2011, there were 7 reactor shutdowns:*
  - 2 unscheduled shutdowns and
  - 5 scheduled ones for implementation of preventive maintenance work, partial refueling, loading and unloading of experimental devices and assemblies with radioisotopes.
- *One unscheduled shutdown occurred due to switching-off turbogenerator, the other was caused by switching-off secondary circulation pump due to failure of its electric motor.*

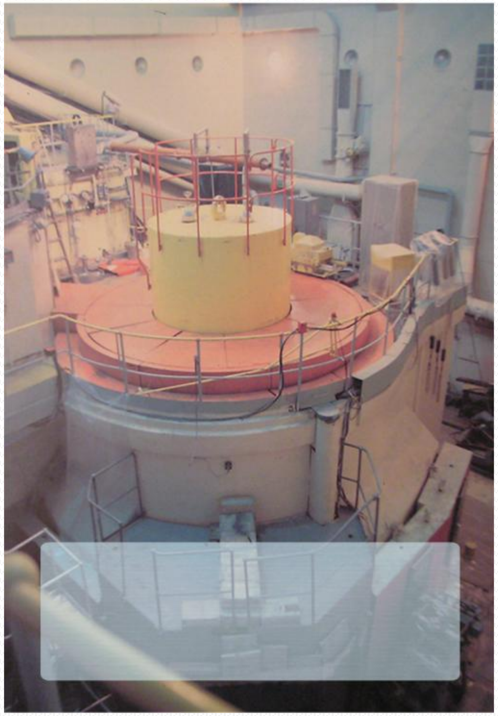
DIAGRAM OF THE BOR-60 REACTOR OPERATION IN 2011



EXPERIMENTAL WORKS CARRIED OUT IN 2011

- *Irradiation of assemblies with structural materials within temperature region from 320℃ to 450℃;*
- *In-pile studies of capsules with hafnium hydride samples under temperature from 500℃ to 600℃;*
- *In-pile tests of the SVBR-100 fuel pin models and fuel pin claddings from EP-823 steel;*
- *Irradiation of the assembly for substantiation of serviceability of fuel pins with vibropacked and pellet MOX-fuel;*
- *Production of radioisotopes of strontium-89 and gadolinium-153 .*

CURRENT STATUS



- *After operation during about 44 years the BR-10 experimental reactor was finally shutdown on December 6, 2002 and now it is on the preparatory stage of its decommissioning.*
- *Current status of the BR-10 reactor is as follows:*
  - All FSAs have been unloaded from the core and replaced by the dummy subassemblies;
  - All FSAs are placed into the interim storage;
  - Sodium has been drained from the primary and secondary circuits to the storage tanks and frozen;
  - The inner surfaces of the primary circuit have been cleaned from sodium and decontaminated;
  - Both loops of the secondary circuit have been cleaned from sodium.

WORKS DONE IN 2011

- *Preparation works on installation of the following facilities were implemented:*
  - *GETTER facility for demercurization of sodium-potassium coolant;*
  - *MAGMA facility for conditioning radwaste of sodium and sodium-potassium coolant by solid phase oxidation method;*
  - *LUIZA facility for treatment of coolant by method of gas-phase purification.*
- *Fulfillment of regulation works on maintenance of serviceability and safety of reactor systems being in operation;*
- *Continuation of preparation of the documents required for getting the license for the BR-10 decommissioning and their submission to Rostechnadzor.*

GENERAL INFORMATION

- *Construction of the 4th power unit with the BN-800 reactor is carried on Beloyarsk NPP site.*
- *Date of construction completion and power unit commissioning is scheduled in 2014.*
- *The BN-800 is planned to be used for:*
  - Closing fuel cycle and
  - Recycling stocks of weapon-grade plutonium.

Main BN-800 parameters

Parameter	Value
Thermal power, MW	2100
Electric power, MW	880
Number of heat removal loops	3
Configuration of the primary circuit	Pool
Steam generator design	Once-through, sectional-modular
Maximum neutron flux density, $n \cdot cm^{-2} \cdot s^{-1}$	$8.8 \cdot 10^{15}$
Fuel	UO_2-PuO_2
Max. fuel burn-up, % h. a.	9.9
Inlet/outlet core coolant temperature, ºC	354/547
Inlet/outlet SG coolant temperature, ºC	505/309
Inlet/outlet SG water/steam temperature, ºC	210/490
Live steam pressure, MPa	13.7
Design lifetime, year	40

WORKS PERFORMED IN 2011

- *Construction works were carried out on the following places of the site:*
  - Reactor compartment (erection of walls at level +36.3 - + 45.0 m);
  - Steam generator compartment (installation of steam generator modules);
  - Turbine hall (erection of the turbogenerator basement);
  - Special building;
  - Administrative buildings;
  - Diesel generator facility;
  - Outdoor switchgears;
  - Power unit’s pump station.
- *Manufacturing the main equipment continues.*

VIEWS OF BN-800 NPP SITE (1/4)

Bird's-eye view of the reactor compartment of the main building

VIEWS OF BN-800 NPP SITE (2/4)

Bird’s-eye view of the steam generator compartment of the main building

VIEWS OF BN-800 NPP SITE (3/4)

Bird's-eye view of the turbine hall

VIEWS OF BN-800 NPP SITE (4/4)

Bird's-eye view of the special building

PROSPECTIVE ACTIVITIES ON SFR IN RUSSIA (1/2)

- *Near- and mid-term plans on SFR in Russia are determined by the FTP "Nuclear power technologies of a new generation for period of 2010-2015 and with outlook to 2020".*
- *The following works on SFR are scheduled within the FTP framework:*
  - Development of a design of the large size sodium fast reactor BN-1200 and implementation of the R&D for its substantiation;
  - Designing and construction of the research fast reactor MBIR with sodium coolant;
  - Design development and construction of the pilot plant for manufacturing MOX-fuel for the BN-800;
  - Upgrading of an experimental base for justification of the SFR, including BFS.

PROSPECTIVE ACTIVITIES ON SFR IN RUSSIA (2/2)

- *The following works on SFR are performed outside of the FTP framework :*
  - Upgrading and completion of certain parts of the BN-800 detailed design;
  - Conceptual study of a high-temperature sodium fast reactor BN VT for its application in high-temperature industrial technologies;
  - Development on the base of the BR-10 of technological processes proposed to be used for SFR decommissioning.

BN-1200 (1/2)

- *Now the Russian organizations according to the FTP conduct a development of a design of the sodium cooled fast reactor of the 4th generation BN-1200.*
- *The most important new conceptual technical and design decisions accepted for the BN-1200 design are as follows:*
  - Pool type arrangement of the primary circuit with location of all sodium systems including cold traps and chemical-engineering control systems within the reactor vessel that makes it possible to eliminate in fact a danger of radioactive sodium release outside of the reactor vessel and its fire;
  - Simplification of a refueling system by exception of intermediate storage drums of fresh and spent FSAs and organization of a capacious in-reactor vessel storage (IVS) of spent FSAs (SFSAs), providing a direct unloading of SFSAs (after their exposure in the IVS) from the IVS into washing cells and further into an exposure pool;
  - Transition from sectional-modular SG scheme to integral one based on application of straight-tube large-capacity modules;
  - Maximum enhancement of inherent safety features of the reactor facility and application of safety systems based on passive principles of functioning: passive systems of emergency protection, passive decay heat removal system through independent loops connected to a reactor vessel.

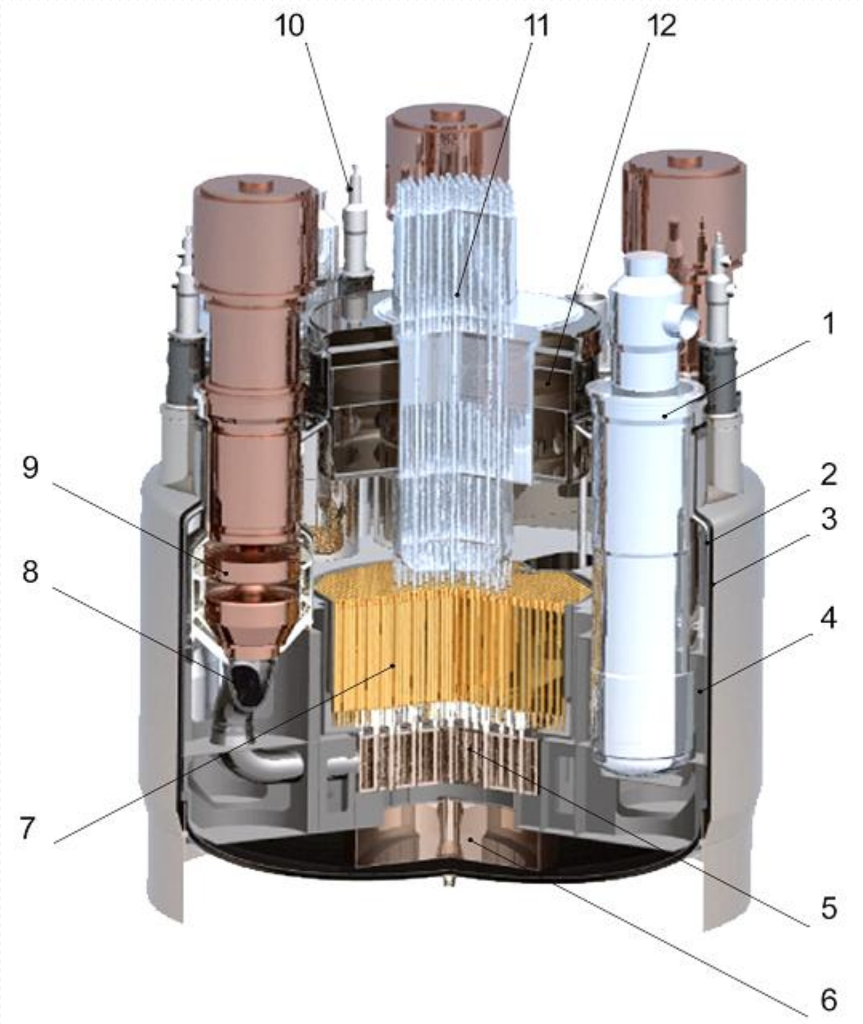
BN-1200 (2/2)

- *Besides, the BN-1200 design envisages:*
  - Traditional three-circuit design of the power unit;
  - Number of loops in the primary circuit - 4 (each loop contains one IHX and one MCP-1);
  - Number of loops in the secondary circuit - 4 (each loop contains one IHX, one MCP-2 and one SG);
  - Number of turbine units per power unit - 1;
  - Steam reheating;
  - Operation of the NPP at the stable (mainly, rated) power level with load factor equal to at least 0.9;
  - Flexible nuclear fuel cycle allowing transition from fuel made on the basis of plutonium extracted from SNF of thermal reactors to fuel made on the basis of own plutonium and providing opportunity of using different types of fuel (oxide, nitride).
- *A FOAK power unit with the BN-1200 reactor is scheduled to be constructed by 2020. The BelNPP site is considered as probable place of its location.*

MAIN BN-1200 PARAMETERS

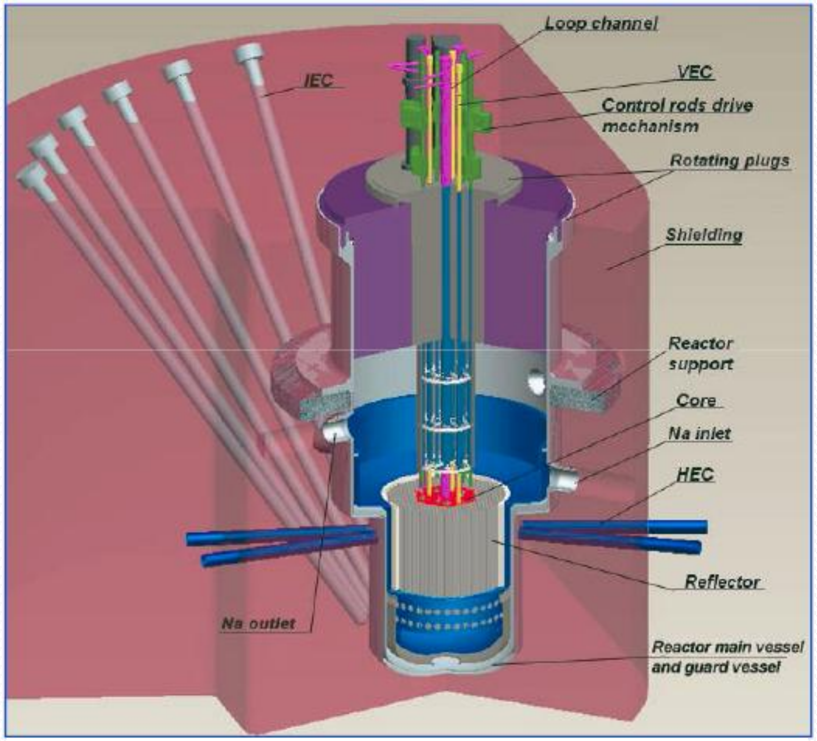
Parameter	Value
Rated thermal power, MW	2900
Electric power, MW	1220
NPP efficiency, %: gross net	 42.0 39.0
Number of heat removal loops	4
Design lifetime, year	60
Primary circuit coolant temperature (IHX outlet/inlet), °C	410/550
Secondary circuit coolant temperature (SG outlet/inlet), °C	355/527
Tertiary circuit parameters: live steam pressure (SG outlet), MPa live steam temperature (SG outlet), °C feedwater temperature, °C steam reheating temperature, °C	 14.0 510 240 250

VIEW OF THE BN-1200 REACTOR VESSEL



1 – IHX;
2, 3 – main and guard vessels respectively;
4 – supporting structure;
5 – inlet plenum;
6 – core debris tray;
7 – core;
8 – pressure pipeline;
9 – MCP-1;
10 – refueling mechanism;
11 – CRDM;
12 – rotating plugs.

MBIR



Lay-out of the MBIR reactor vessel and its experimental channels

- *Now development of the design of the research fast reactor MBIR with sodium coolant is performed.*
- *Putting into operation of the MBIR is scheduled in 2019.*

BASIC CHARACTERISTICS OF THE MBIR REACTOR

Parameter	Value
Thermal power, MW	~150
Electric power, MW	~40
Maximum neutron flux density, $n \cdot cm^{-2} \cdot s^{-1}$	$\sim 6.0 \cdot 10^{15}$
Driven fuel	Vi-pack-MOX, (PuN+UN)
Test fuel	Innovative fuels, MA fuels and targets
Core height, mm	600
Maximum core power density, kW/l	1100
Maximum neutron fluence per year, $n \cdot cm^{-2}$	$\sim 1 \cdot 10^{23}$ (up to 45 dpa)
Design lifetime, year	50
Number of autonomous test loops with different coolants	up to 4
Total number of experimental subassemblies and target devices for radioisotope production	up to 12 (core) up to 5 (radial shielding)
Number of experimental channels	up to 3 (core)
Number of experimental horizontal channels	up to 6 (outside reactor vessel)
Number of experimental vertical channels	up to 8 (outside reactor vessel)

Thank you
for your attention !