BOR-60 Reactor Operational Experience and Experimental Capabilities

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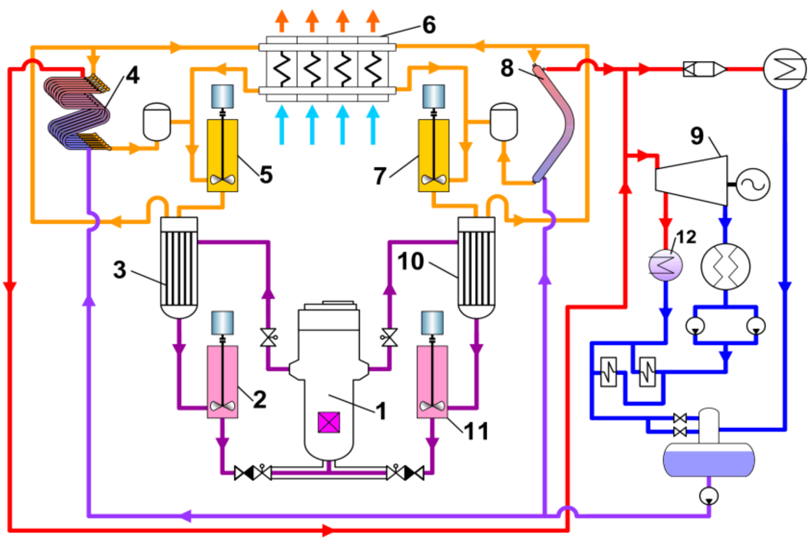
Abstract. Test fast reactor BOR-60 is one of the leading Russia’s and world’s facilities to test fuel, absorbing and structural materials intended for fast, advanced water-cooled, gas-cooled and fusion reactors and to justify the lifetime extension of the VVER and BN-type reactors. BOR-60 reactor was commissioned in December 1969 and by 2017 it has been operated at power for ~275000 hours. This parameter makes BOR-60 the world’s leader and the reactor keeps demonstrating its potential in extending the lifetime of sodium-cooled fast reactors. For more than 47 years the reactor has been safely and effectively operated. Nowadays it is practically the only fast test reactor that has unique capabilities to carry out comprehensive research in different areas supported with well-equipped material testing laboratories and fuel fabrication and reprocessing facilities. Experimental results generated at the reactor allowed the justification of fuel pin and absorber materials for reactors BN-350, BN-600, BN-800 and others. Recently, a package of works has been done that included the inspection of equipment, calculation-experimental investigation of parameters and operational conditions of non-replaceable components of the reactor, examinations of structural materials after a long-term operation under irradiation and others. The generated results allowed the extension of the BOR-60 operation till 2020. Now the work on the further reactor modernization and lifetime extension (after the year 2020) is going on.

Key words: BOR-60 reactor, equipment, operation, damage dose, fuel element, heat rate, moderator.

1. Introduction

Fast test sodium-cooled reactor BOR-60 was constructed to try out the technologies and test fuels and materials for industrial fast sodium-cooled reactors [1]. Figure 1 shows the reactor layout. BOR-60 has two circuits for cooling with sodium. The third circuit is water-steam and equipped with a turbine and heating unit.



1 – reactor; 2, 5, 7, 11 – primary and secondary circuit pumps;
3, 10 – intermediate heat-exchangers; 4, 8 – steam generators;
6 – air heat-exchanger; 9 – turbine; 12 – heating unit
FIG. 1. BOR-60 layout.

The BOR-60 design was developed to provide for a 20-year operation. However, the experience in the reactor equipment operation as well as experimental results on the structural materials behavior generated both at BOR-60 and other fast sodium-cooled reactors showed it possible to extend the facility lifetime. So, since 1988, the reactor lifetime has been gradually extended, first for ten years (to 30 years), then for another ten years (to 40 years) and then for five years (to 45 years). In 2014, the BOR-60 reactor was licensed to operate till 2020 [2]. The BOR-60 lifetime extension was done in a full compliance with the RF regulations [3] setting the key criteria and requirements to the safety when extending the lifetime of nuclear facilities. The above-said regulations meet the IAEA-TECDOC-792 recommendations on the management of research reactor ageing.

Table 1 presents the key BOR-60 characteristics.

TABLE I: KEY BOR-60 CHARACTERISTICS

Characteristics	Value
Nominal thermal capacity, MW	Up to 60
Nominal electrical power, MW	Up to 12
Neutron flux density, $n \cdot cm^{-2} \cdot s^{-1}$	Up to $3.6 \cdot 10^{15}$
Primary and secondary circuit coolant	sodium
Sodium flowrate through the reactor, m^3/h	Up to 1100
Sodium temperature: - inlet, ºC - outlet, ºC	 310÷340 Up to 530
Na velocity in the core, m/s	Up to 8
Na flowrate in two secondary loops, m^3/h	~ 1400
Coolant in the third circuit	Water/steam
Steam pressure in the third circuit, MPa	10
Overheated steam T, ºC	Up to 480
Annual operation at power, days	240
Micro-run duration, days	Up to 90
Period between micro-runs, days	(9 - 45)

The following activities have been done to upgrade the equipment and enhance safety:
- industrial anti-seismic protection system has been put into a pilot operation;
- storage batteries and rectifiers of the emergency power supply system were replaced;
- two backup diesels were installed into the emergency power supply system;
- portal monitors “ARKA” have been installed to detect radioactive substances and nuclear materials at the BOR-60 entry control point;
- TV system has been mounted to control processing procedures;
- 3rd circuit processing procedure control system of has been upgraded;
- backup control board has been upgraded;
- fire alarm system has been put into operation;
- three transformers in the power supply system have been replaced;
- outlet fitting of the steam generator buffer tank has been replaced;
- loud-speaker communication system has been mounted;
- control rod equipment has been replaced.

2. Analysis of the conditions and safety justification for the reactor components affected by irradiation

Figure 2 presents the BOR-60 reactor vertical cross-cut. To evaluate the residual lifetime and justify the operational safety, the following additional calculation-experimental activities were done:

– distribution of neutron flux density and damage dose accumulation rate inside the core and beyond was studied to verify the MCU-based calculation models [4]. The study resulted in the reduction of the calculation uncertainty and specification of neutron fluence (F) and damage dose (D) values in different reactor components as of the end of 2016. The study results were used to make predictions till 2020;
– coolant temperatures were measured at different elevations above the core at different power levels and temperatures of the internals were specified with the account of the heat rate effects to further use these data in the strength calculations;
– stress-strain state of the components was calculated based on the maximal damage values of the materials affected with operational factors, including neutron fluence.

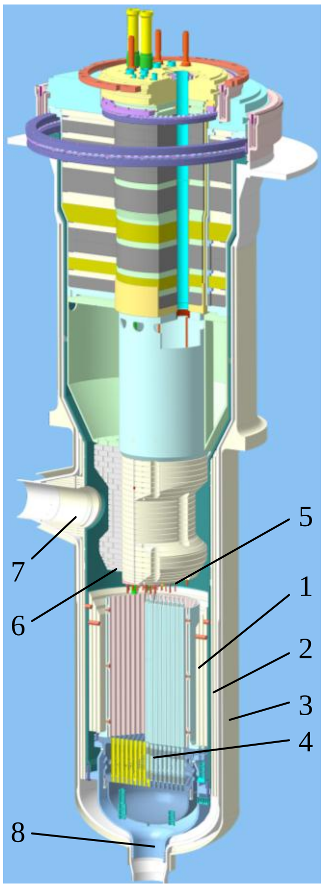


FIG 2. BOR-60 vertical cross-cut.

Table 2 presents the critical BOR-60 components and operational temperature (T, ºC), fast neutron (E>0.1MeV) fluence ($F_{0.1}$) and damage dose (D) for each of them as of December 2016 and till 2020 (predicted) [5].

TABLE II: FAST NEUTRON FLUENCE AND DAMAGE DOSE IN THE REACTOR COMPONENTS

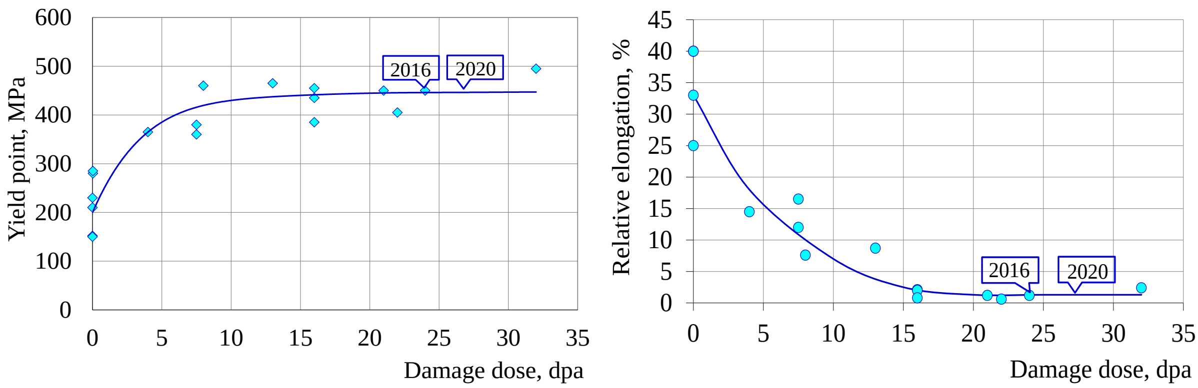
№*	Components	T, ºC	December 2016		December 2020 (predicted)	
			$F_{0.1}$, cm^{-2}	D, dpa	$F_{0.1}$, cm^{-2}	D, dpa
1	Basket	200-515	5.3E+22	17	5.9E+22	19
2	Vessel	200-400	2.7E+22	8.6	3.0E+22	9.5
3	Jacket	200-350	2.1E+22	6.7	2.3E+22	7.5
4	Collector	200-360	4.5E+22	16	4.9E+22	17
5	SRP plate pin	200-515	6.6E+22	24	7.2E+22	27
6	LRP plate pin	200-515	4.6E+22	17	5.1E+22	19
7	Outlet sleeve	200-515	2.7E+21	0.9	2.9E+21	1.0
8	Inlet sleeve	200-360	9.8E+19	0.05	1.1E+20	0.06
* – See Fig. 1						

The structural materials used for the BOR-60 components, including the vessel, internals, primary and secondary equipment and pipelines, are austenitic stainless steels 1X18H9 and 0X18H10T (analogs of 347 and 321). As it follows from the data given in Table 2, the highest damage dose is observed for lower plates of the biological shielding of small and large rotating plugs (SRP and LRP). The SRP and LRP biological shielding is located above the core and consists of 16 steel plates fastened to each other and mounted to the rotating plugs with pins. The pins are operated under axial tensile stress, intensive reactor emission, temperature gradients and cyclic temperature changes resulted from reactor start ups and shut downs. In addition, the pins are affected with lower plates swelling under neutron irradiation. So, when justifying the strength characteristics of the reactor units, a special attention was paid to the integrity of the SRP and LRP biological shielding.

Since the replacement of the rotating plugs and their biological shielding is rather a time- and labor-consuming activity, these components refer to non-replaceable ones. To get experimental data on the condition of the plates and pins of the biological shielding, there was remotely cut out a fragment of an automatic control rod guide tube ((∅48×2mm, steel 0X18H10T) located close to the lower plates of the SRP biological shielding. The guide tube had been operated in the reactor since its start up in 1969 till the mid of 2008; the operational time made up ~ 217000hs at (500÷520)°C (the reactor was operated at power) and ~120000hs at (200÷270)°C (during shut downs), the fast neutron fluence made up 5.7×10^{22} cm^{-2}, damage dose – 21.5dpa.

The samples underwent metallography that showed changes in the steel structure caused by thermal ageing of the material under a long-term operation. The structure is characterized with equiaxial grains and small amount of duplicates; it remained fine-grained with grain size corresponding to index 9-10 according to the State Standard (GOST 5639-82). Carbonizing-caused darkening was observed in the surface layers located close to sodium. The metallography showed the carbonizing depth to be 150-160μm on the inner surface of the guide tube and 200-250 μm on its outer surface. The maximal swelling value measured by a hydrostatic weighing made up (1.4±0.8)%. It should be noted that selling data correlate well with the results obtained in the EBR-II reactor for steel 304SS: at a fast neutron fluence of ~ 5.7×10^{22} cm^{-2}, swelling does not exceed 1.5% [6].

To justify the BOR-60 lifetime extension, there were summarized all the data on the mechanical properties of steels X18H9 and X18H10T that were either purposely irradiated or operated as part of replaceable reactor components (FA ducts, blanket assemblies, cases, etc.). When irradiating steel X18H10T under 520-550°C, hardening of the material is observed up to 10-12dpa and then the yield point achieves its saturation at 32dpa and does not change any longer. The hardening is accompanied with a decrease of plasticity characteristics. As the damage dose rises up to 32dpa, a certain level of the plasticity characteristics saturation is observed ~ (1-2)% (Fig. 3).



a) b)

FIG 3. Dependencies of the yield point (a) and relative elongation (b) for steel X18H10T irradiated in BOR-60 at 520-550 °C.

The conditions of all critical reactor units under operation were analyzed and the behavior of the material was predicted for a 300000-hour operation period; the strength calculation technique was revised and the static and long-term strength was calculated with the account of mechanical, thermal and radiation effects and cycling load for all critical reactor components. The calculation results showed the reliability and strength criteria for the most stressed components to be met with the account of all operational factors till, at least, 2020.

3. Analysis of performance and safety justification of sodium equipment and pipelines

According to the regulations and norms of the BOR-60 equipment and pipelines operation, the state of materials is controlled by specific methods agreed by both regulatory authorities and design agency. The visual inspection of the equipment and pipelines is done in the accessible places; the pipeline material thickness is measured by ultrasonic method and some pieces of equipment (circulation pumps, etc.) undergo the vibration measurements.

When preparing for the reactor lifetime extension for up to 40 years and beyond, the state and mechanical characteristics of the primary and secondary pipeline material (steels 1X18H9 and X18H10T) were examined by destructive methods. Fragments were cut out either from replaced pipelines or from the operational ones provided it was feasible. The key examination results of the materials operated for 35-38 years are as follows: no significant degradation of the material is revealed, structural changes are insignificant, the amount of non-metallic inclusions (carbides, carbonitrides) does not exceed the limits, corrosion damage is no higher than 10µm, plasticity meets the requirements.

Let us take as an example the examination results obtained for the secondary main pipeline (∅325×12mm, steel 1X18H9) operated for 38 years (maximal temperature ~ 480°C):

- the chemical composition corresponds to the Certificate for Materials;
- index of non-metallic inclusions is no higher than 3;
- the metal structure with grain-size index (6-7) has both larger grains (grain-size index 1-2) and smaller ones (grain-size index 8-9);
- at a test temperature of 20°C a significant increase in the ultimate strength up to ~700MPa is observed as well as in the relative elongation up to 70%;
- at a test temperature of 600°C, the relative elongation makes up 33%, the allowable limit being 22%.

To justify the extension of the BOR-60 lifetime till 2020, additional calculations were done for the sodium circuit equipment and pipelines using up-to-date software and regulatory requirements on the statistical and cyclic strength, long-term statistical and cyclic strength, resistance to the brittle/ductile damage and maximal seismic effect with the account of the actual and predicted operational parameters. The calculation results are as follows: the statistical and long-term strength as well as resistance to the brittle/ductile damage are provided for all reactor components; the predicted accumulated fatigue damage for the most stressed element is no higher than 0.8, i.e. significantly less than 1.

The results of the operational history analysis, survey of the technical state and strength calculations done for the sodium equipment and pipelines show them to be in the satisfactory condition and suitable for the further operation till, at least, 2020.

In 2015, the 3rd section of the steam generator OPG-1 was cut out and material tests were done for steel (15312+ Nb) of the heat exchange tubes.

The examination results were as follows:

1. A significant accumulation of sediment was observed on the heat transferring surfaces of the evaporator and economizer that caused worsening of the heat exchange characteristics of the steam generator. The contamination of the evaporator tubes made up 145÷186

g/m^2. The most contaminated tubes were those of the economizer, the contamination value achieved 300 g/m^2.

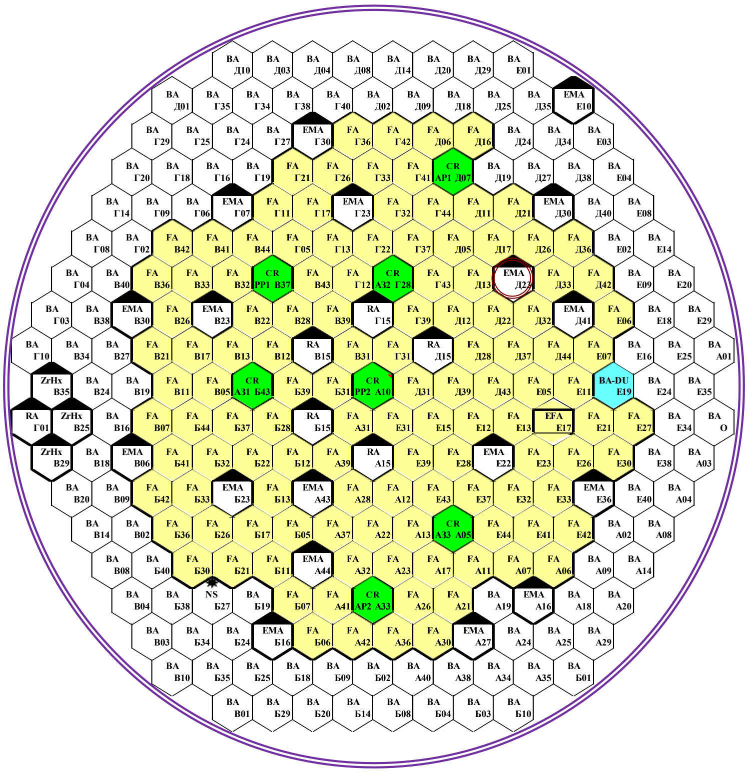
2. The examination of heat exchanging tubes materials regarding the effect of water and steam showed:

- the total corrosion rate in the steam superheater estimated by the oxide film thickness made less than 0.001mm/year; it was 0.003mm/year in the evaporator and 0.005÷0.006mm/year in the economizer;
- the corrosion pit depth in the steam superheater did not exceed 0.05mm;
- the maximal corrosion pit depth in the evaporator made up 0.20÷0.30mm;
- the maximal corrosion pit depth in the economizer achieved 0.15÷0.18mm;

3. No corrosion damage of the tubes from the sodium side was revealed.
4. Typical changes in the base metal structure related to thermal ageing were noted.
5. The results of mechanical tests of ring-shape and transverse samples showed the strength of the steam superheater material to increase by 30÷40MPa at a test temperature of 20°C and the plasticity to decrease as compared to the initial values. At that, the mechanical properties remained at rather a high level.

4. Irradiation capabilities

Figure 4 presents the BOR-60 core arrangement. The reactor core is hexagonal having 265 cells. There are 156 cells for fuel assemblies, 7 cells for control rods and the rest are for blanket assemblies. The experimental rigs can be installed in any cells, except for those taken by control rods. At various times, the BOR-60 core contained different number of fuel assemblies (from 75 to 130 pcs.) depending on the properties and characteristics of fuel, its burnup depth, core and blanket arrangement and number and type of experimental rigs, etc. [1, 7].



FA – fuel assembly,

CR – control rod,

EMA – experimental material test assembly,

ZrHx – assembly with moderator (zirconium hydride),

BA – blanket assemblies,

NS – neutron source,

RA – assembly for radionuclides accumulation,

D23 – instrumented cell

FIG. 4. BOR-60 core arrangement.

The reactor design provides for changing the core arrangement in the wide range; a large number of experimental rigs can be inserted into the reactor cells. The neutron flux density (Fn) can be three times different from cell to cell, the maximal value being 3.6×10^{15} cm^{-2}s^{-1}

(thermal reactor capacity 60MW, all assemblies are fuel ones). It allows loading the reactor with different fuel compositions and achieving practically any burnup. Up to 20 experimental rigs with structural materials can be inserted into the core, while the number of irradiation rigs in the blanket is unlimited. There is a thermometric channel allowing for an instrumented irradiation and output of data on the irradiation conditions via 30-50 communication lines. Figure 5 presents the key reactor neutronic characteristics at a power of 55MW [7].

An experimental channel to carry out instrumented in-reactor tests is located 196cm from the core above the 5th row cell D23 (Figure 4). The lower part of the experimental rig has the same shape s the standard fuel assembly (fixture and hexagonal tube 44mm across flats). Table 3 presents the key characteristics of cell D23.

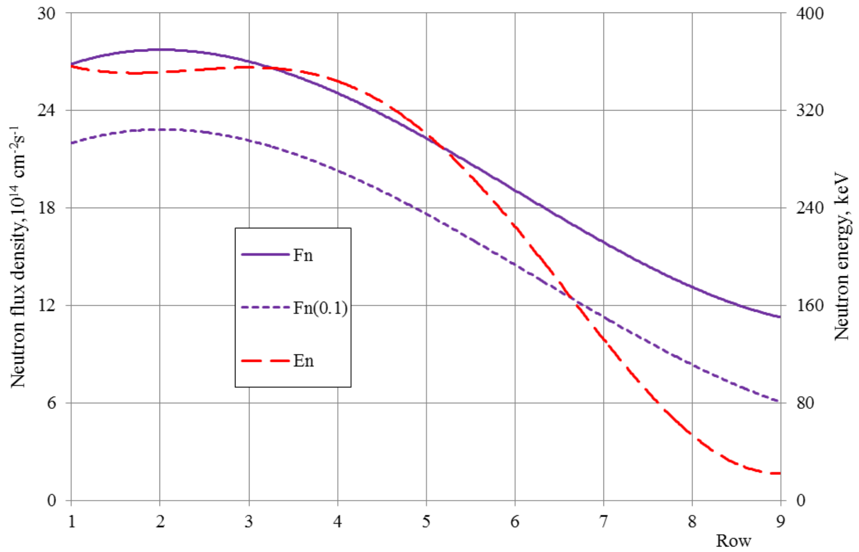


FIG. 5. Radial distribution of the key BOR-60 neutronic characteristics (average neutron energy (E), integral (Fn) and with E>0.1MeV neutron flux density (Fn(0.1))

TABLE III: IRRADIATION CONDITIONS IN CELL D23

Parameter	Value
Neutron flux density, $cm^{-2}\cdot s^{-1}$	$2\cdot 10^{15}$
Specific heat rate in structural materials (with atomic number Z = 26÷30), W/g	4
Absorbed gamma-emission dose rate, Gy/s	$4.5\cdot 10^{3}$
Factor of irradiation density non-uniformity distribution along the core (450 mm): for neutrons for gamma-emission	 1.13 1.25
Na flow rate, m^3/h: from high pressure chamber from low pressure chamber	 Up to 8 Up to 2

4.2 Reactor operation schedule

Heat produced by BOR-60 is transferred either to the heating system, converted into electricity or discharged into the atmosphere by the air heat exchanger. Due to this and depending on the season, the reactor power varies within the range of 50÷55 MW. The duration of the continuous operation (no shutdowns) depends on the reactivity margin, experimental and test programs and lasts no longer than 90 days, as a rule. In addition, methodical experiments are carried out in the reactor to justify the irradiation parameters for experimental rigs and materials. These experiments are performed either at the beginning or at the end of the reactor run in cell D23.

Figure 6 presents an actual power change of the reactor during the year 2016. Two long-term outages (~45 days) are used to refuel the reactor and perform maintenance and repair activities, two short-term outages (less than 20 days) are used to load/unload non-fuel assemblies, change position of some assemblies (if needed) and organize short experiments. The data acquisition and measurement system records all the parameters during the operation.

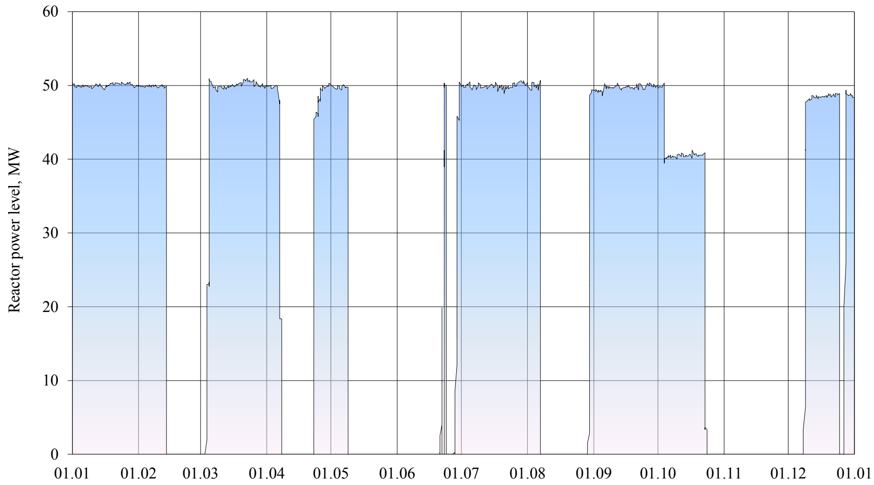


FIG. 6. BOR-60 reactor power change during the year (2016)

4.3 Calculation and methodical support

A thorough study of the neutron-physical, thermo-hydraulic and dynamic characteristics of the reactor allowed developing software to support on-line the reactor operation and perform investigations.

Based on a wide experience in the investigation of reactor characteristics as well as on the verified software, techniques were elaborated to control the irradiation modes and parameters of the materials in the non-instrumented cells.

Table 4 gives the uncertainties of the key irradiation parameters for standard fuel assemblies and experimental rigs.

TABLE IV: UNCERTAINTIES OF THE KEY IRRADIATION PARAMETERS

Parameter	Uncertainty, %
Thermal capacity	5
Na flowrate	4
T of Na and samples	5
Power of the core FA	6
Specific heat rate	6÷9
Neutron flux density	3
Neutron fluence	9
Damage dose	6
Decay heat	8÷12

4.4 Experimental methods and rigs

To irradiate a wide range of materials and items under different modes and parameters, specific irradiation rigs are used, including capsules, dismountable material test rigs, autonomous instrumented channels, instrumented assemblies, etc. To study creep, corrosion resistance and swelling of cladding and duct materials, irradiation rigs were developed to test samples at

320-1200°C [8, 9]. They have simple design and can be inserted practically in all core and blanket cells. The key task facing us when developing the rigs is to provide the required samples' temperatures. For this purpose, the rig design has either insulating gaps or intensive cooling or additional heating due to the radiation heat rate or fuel fission. These rigs also allow for the pre-set temperature non-uniformity along the height and azimuth.

The long-term strength and irradiation-induced creep of cladding materials are examined using pressurized tubular samples. The reactor allows testing different types of promising fuel and structural materials under high heat rate (100 kW/m), temperatures (1000°C) and fluence (1.8×10^{23} cm^{-2}, E>0.1 MeV).

The lowest irradiation temperature that can be provided in the BOR-60 is 310°C that is 40-80°C lower as compared to other fast reactors. It widens significantly the testing capabilities of the reactor regarding the examination of mechanical properties of zirconium alloys and materials of VVER internals. If it is necessary to unload samples for interim measurements and then to continue their irradiation, a dismountable rig design is used. Different capsule designs are used to test samples in gas, sodium, potassium, lithium, lead, etc.

Irradiation rigs of the same design as the standard FAs are used to study abnormal phenomena and .justify the fast reactor safety. Severe accidents are studied using autonomous loops located inside the reactor, the gas lines and gage signals being output. The loops are two-circuit and cooled with sodium. Depending on the design, the sodium circulation can be either natural or forced by either a centrifugal or electromagnetic or MHD pump. The loop with the natural sodium circulation was used in the experiment with a cladding temperature increase up to 780°C. The loop with the forced sodium calculation was used for a demonstration experiment on coolant flow blockage and cladding melting. The gained experience allowed the development of similar channels with liquid metal coolant to test the fuel pins for promising fast reactors. Information about the irradiation rigs types is given in Table 5.

TABLE V: BOR-60 IRRADIATION RIGS

IR types	T, °C	Heater	Heater power, kW	Coolant flowrate, m^3/h	T maintained by	Reactor
Unsealed, refreshed:						
- no heater	310-370	-		2,5-3,5	-	
- metallic heater	350-500	tungsten	20-50	0,2-1,0	-	BN
- fuel heater	400-650	NF	50-200	1,0-4,0		
- based on standard FA	Up to 650	NF	Up to 400	3,0-4,0	-	
Unsealed non-refreshed	500-700	irradiation	-	-		
Sealed						BN, BREST, SVBR, VVER, RBMK, HTGR, Fusion reactor
heat sealed gas gap medium:						
- non-refreshed liquid metal	400-1000	Self heat	-	1,0-4,0	-	
- inert gas	400-650	The same		1,0-4,0	-	
T maintained by:			-			
- regulated thermal resistance	400-500	The same	-	-	gas gap height and thickness	
- cone capsule	Up to 600	The same	-	-		
- evaporating siphon	650-900	The same	-	-	Tube with boiling Na	
Autonomous loop channels	300-1000	FA	Up to 100	Up to 3,0	Coolant flowrate change	BN, BREST, SVBR, Fusion reactor

5. Further operation

BOR-60 has been under operation for more than 47 years, the designed lifetime being 20. A decision to extend its lifetime was taken on the basis of the survey of the state of equipment and materials, strength calculations done for the sodium equipment and pipelines critical for the safety. A decision [2] has been taken to extend the BOR-60 lifetime till 2020 and beyond.

Long-term R&D programs have been developed and financed in the frame of both Federal Target Program "Nuclear Power Engineering of New Generation" and international contracts. The BOR-60 core will be occupied with the experiments for the next several years and the research activities will be carried out till its decommissioning.

By the end of the BOR-60 lifetime, a new multi-purpose fast test reactor MBIR will be constructed and commissioned under the Federal Target Program "Nuclear Power Engineering of New Generation". The MBIR will replace the BOR-60 after its final shut down [10]. It will satisfy the demand in experiments related to fast reactors. The MBIR will have a wider range of experimental capabilities. Being located at the RIAR's site, it will not take much time to transfer all the experiments from the BOR-60 into the MBIR [11]. Nowadays such experiments are being carried out.

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