REVIEW OF FAST REACTOR OPERATING EXPERIENCE GAINED IN 1998 IN RUSSIA. GENERAL TRENDS OF FUTURE FAST REACTOR DEVELOPMENT

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ABSTRACT

Review of the general state of nuclear power in Russia as for 1998 is given in brief in the paper.

Results of operation of BR-10, BOR-60 and BN-600 fast reactors are presented as well as of scientific and technological escort of the BN-350 reactor.

The paper outlines the current status and prospects of South-Urals and Beloyarskaya power unit projects with the BN-800 reactors.

The main planned development trends on fast reactors are described concerning both new projects and R&D works.

1. RESULTS OF NPP OPERATION IN 1998

In 1998, 29 power units were in operation in Russia as integral parts of 9 NPPs having 21.242 GW total power, namely:

- 13 power units with VVER integral type reactors (including 6 VVER-440 and 7 VVER-1000 reactors);
- 15 power units with uranium-graphite channel type reactors (including 11 RBMK and 4 EGP-6 reactors);
- 1 power unit with the BN-600 fast neutron reactor.

Energy production by the NPP was $103.493 \cdot 10^6$ MW·h, that is 95.6% of the previous year value.

In 1998, average load factor value for all NPPs was 55.62% as compared to 58.2% value obtained in 1997.

During 1998, 102 malfunctions took place on the NPPs, only 4 of these being safety related. Three of these malfunctions are classified by the first level and one – by the second level of the INES scale.

Fig. 1 shows the change of the frequency of NPP abnormal operation cases including those influencing safety during previous 6 years. Classification of safety related incidents according to the INES scale is presented in Fig. 2.

Fig. 3 is the diagrams showing distribution of incidents occurred on different reactors between the quarters of 1998. Fig. 3a gives the total number of abnormal operation events for each reactor type, while in Fig. 3b specific values per one reactor of each type are presented.

It should be noted that last year some increase of the total number of abnormal operation cases took place as compared to that of 1997. However the number of these cases in the NPP, classified by the INES scale, have been kept almost the same during several years.

2. FAST REACTOR OPERATING EXPERIENCE

During 1998, three fast reactors were in operation in Russia (BR-10, BOR-60 and BN-600). In Kazakhstan, the BN-350 fast reactor is in operation under scientific and technological supervision of the Russian experts.

2.1. NPP with the BN-600 reactor

One preventive repair procedure was carried out on the BN-600 reactor in 1998. Neither unscheduled reactor power decrease nor loop shutdowns nor reactor scrams took place.

Diagram illustrating the BN-600 reactor power unit operation in 1998 is presented in Fig. 4.

Generation values of electricity and heat were respectively 2519 and 81 million kW·h.

Table 1 gives the main technical and economical indices of the BN-600 reactor in 1998, compared to the average statistical values taken for the whole operating period.

Table 1.

No.	Parameter	Unit	1998	Average value
1.	Load factor: Heat supply is neglected Heat supply is taken into account	%	48 49.5	74
2.	Unscheduled decrease of load factor	%	0	1.5
3.	Number of cases of power decrease	-	0	3
4.	Number of reactor scrams	-	0	0
5.	Collective irradiation dose	man-Sv	1.78	0.88
6.	Radioactive inert gas release	kCi	0.44	0.44
7.	Amount of low active solid wastes	m^3	60	36
8.	Maximum fuel burn-up	% h.a.	11	

Unlike the previous years of the reactor operation, when one annual 54 days preventive repair and 21 days refueling procedures were planned, in 1998 one combined scheduled preventive repair procedure was carried out during 6 months (182 days). This decision was caused by the necessity of elimination of seizure in the small rotating plug of the reactor during refueling, which was increasing since 1995.

In order to determine the cause of the plug seizure, in 1997 a hole was drilled in the plug body during the reactor planned shutdown, and inspection of its bearing unit was made. Sodium was detected in the bearing unit.

An attempt was made to remove the sodium by heating the bearing, but it turned lame, and then decision was made on removal of sodium and replacement of (presumably) damaged bearing unit in the course of preventive repair procedure in 1998.

Sodium removal and bearing replacement were carried out after the plug had been lifted by 650 mm into the leak tight container. Inspection of the replaced bearing unit showed that it was blocked up with sodium, and the cause of sodium penetration into the bearing is to be studied. Preliminary analysis has shown that the amount of sodium (15 kg) corresponds to the rate of sodium vapor transfer in the annular gap between the large and small rotating plugs.

Then the plug was elevated by 2100 mm height for inspection and measurements, and some scratches on its lateral side were detected and eliminated. After these operations, the rated value of the force required for the plug rotation was restored.

In 1998, irradiation of experimental subassemblies with uranium-plutonium fuel was in progress without any problems. Neither were encountered problems in the subassemblies intended for cobalt-60 production, irradiation of which was completed.

Studies are planned for the future on the possible extension of the power unit operating life from 30 to 40 years, as well as on determination of the possibility of weapons grade plutonium utilization in the BN-600 reactor.

2.2. Experimental reactor BOR-60

In 1998, power of BOR-60 reactor facility was within 45-55 MW range. The main parameters of the reactor in 1998 as well as the data characterizing the whole period of the reactor operation are presented in Table 2.

Table 2.

1	2	3	4	5	6	7	8
Parameter	Unit	1 qtr	2 qtr	3 qtr	4 qtr	1998	Since start-up to 31.12.98
Time of reactor operation on the power exceeding minimum detectable level	h	1958	1635	1866	1219	6678	160409
Reactor usage factor	–	0.906	0.748	0.845	0.552	0.762	
Maximum reactor power	MW	55	49	45	50	55	
Energy production: heat electricity	MW·h	 85101 8865.6	 67208 10024.8	 66935 10965.6	 47844 6300	 267088 36156	 6626307 1078168.4
Steam generator operating time: OPG-1 OPG-2	h	 1958 1958	 1616 1616	 1834 1834	 1216 1216	 6624 6624	 92967 45935
Heat supply to the consumers	Gcal	40192	6144	3655	19220	69211	390935

In 1998, tests of the fuel, absorber and structural materials were in progress under conditions of high coolant parameters and neutron flux.

The reactor core consisted of standard fuel subassemblies with the vibro-packed oxide fuel (UO_2, $UPuO_2$).

Maximum fuel burn-up in the standard subassemblies and in the experimental fuel subassemblies of standard design is respectively 15.5% h.a. and 30.3% h.a. In some of the experimental fuel elements (namely three fuel elements located in special demountable device) ~ 32.3% h.a. maximum fuel burn-up was achieved.

2.3. Test reactor BR-10

In 1998, the BR-10 reactor was in operation on variable power during 4634 hours, including 3243 hours operation within 6.0 – 7.6 MW power range.

The following works were carried out on the reactor:

- target irradiation to produce isotopes required for pharmaceutics (^{32}P, ^{33}P, ^{35}S and ^{89}Sr) as well as ^{99}Mo;
- irradiation of uranium and magnesium fuel composition specimens and four specimens of various fuel element cladding structural materials;
- irradiation of lavsan film by fission products in thermal neutron beam for the purpose of production of track membrane having sterilization filtration properties (0.2 μm pore diameter, 10^9 pores/cm^2 pores density);
- radiation treatment of the oncological patients.

Design burn-up value for the nitride fuel is 8% h.a. By now, permissible 9% h.a. burn-up value has been justified for the BR-10 reactor nitride fuel.

Design burn-up value was exceeded in the fourth quarter of 1998 in many fuel subassemblies, maximum burn-up value being equal to 8.62 % h.a.

Permanent monitoring of the fuel element cladding integrity made on the basis of measurement of the delay neutron predecessors in the primary coolant and gaseous fission product activity in the cover gas of the primary pumps has made it possible to timely detect 7 fuel element failures, including one case of the fuel-coolant contact.

Fast neutron fluence (En $\geq$ 0.1 MeV) upon the reactor vessel is as high as $5.95 \cdot 10^{22}$ n/cm^2, design value being equal to $7.0 \cdot 10^{22}$ n/cm^2.

In 1998, works were under way on preparation of the report on the BR-10 test reactor operation safety taking into account the IAEA recommendations.

In December 1998, operating life time of the BR-10 reactor was extended until 31.12.2000.

Basic design of the BR-10 reactor decommissioning is now under development.

2.4. Cooperation with the Republic of Kazakhstan on the BN-350 reactor NPP

The BN-350 reactor has been out of operation since March 1998.

Agreements are still in force on participation of the Russian institutions in the assessment of conditions of the BN-350 reactor components and systems and justification of its safety, as well as preparation for its decommissioning.

2.5. Current status of the BN-800 reactor NPP construction

According to the "Program of Nuclear Power Development in Russia in 1998-2005 Period and Prognosis to 2010", two power units with the BN-800 reactors are supposed to be built on South-Urals and Beloyarskaya NPP sites.

The construction of these two power units has been suspended because of the lack of financing.

Currently, works on the improvement and finalization of the BN-800 reactor design are carried out. The modifications introduced to the design are aimed at the considerable reduction of the power unit construction cost.

3. THE MAIN GOALS OF DESIGN STUDIES IN FAST REACTOR AREA

Two main directions can be identified in fast reactor development in Russia:

- works on the improvement of fast reactor designs under development and those of the reactors in operation;
- advanced reactor design studies.

The former direction covers works on the BN-800 reactor design modification and development of hybrid core design for the BN-600 reactor.

The latter implies organization of the development of the advanced competitive large size fast reactor with sodium coolant and organization of works on justification of the alternative coolants application in the power fast reactors. Lead, and lead-bismuth are considered as the alternative coolants.

In particular, 75 MW lead-bismuth cooled fast reactor design with two heat removal circuits is under development. The concept design stage is near completion.

The main efforts in the area of the alternative coolants for fast reactors are now supposed to be applied to the development of BREST-300 lead cooled fast reactor design.

3.1. Works on improvement of the BN-800 reactor design

R&D works program on the correction and appropriate justification of basic design of some systems and components of the BN-800 reactor has been developed and approved by now in order to increase its technical and economical parameters.

These works are planned for the period of 1998-2005. In order to avoid duplication and to reduce cost of the NPP these works will be performed to meet the requirements for both Beloyarskaya and South Urals sites.

In particular, the following systems and components should be modified within the framework of R&D works:

- reactor building;
- BN-800 reactor control system;
- emergency power supply system;
- decay heat removal system with air heat exchangers (DHRS AHX);
- passive safety system (PSS);
- steam generator;
- components and systems of the 3rd circuit, etc.

3.2. Justification of hybrid core design of the BN-600 reactor

The development of hybrid core design using MOX fuel for the BN-600 reactor is in progress now. It is supposed that this design realization will assure efficient incineration of the weapons grade plutonium in the BN-600 reactor.

This work is carried out under financial, scientific and technological support from the USA, France and Germany. Financial support of the design is planned by Japan.

Basic design of the hybrid core has been completed by the SSC RF IPPE and OKBM. As it has been mentioned above, several MOX fuel subassemblies are now irradiated in the BN-600 reactor in order to have experimental justification of the hybrid core design.

Currently, justification of the BN-600 reactor hybrid core is under way for the abnormal operation conditions, design basis accidents and more severe beyond design accidents.

3.3. Development of advanced fast reactor designs

As it has been indicated above, two main directions can be identified in the development of the advanced fast reactor designs, namely:

1). improvement of technical, economical and safety characteristics of traditional sodium cooled reactors in order to meet rather high requirements and increase their competitiveness;

2). development of improved safety fast reactor designs with heavy liquid metal (in particular, lead) coolant.

Sodium cooled fast reactors are the most promoted designs. They have been realized in operating plants and have developed technology and extensive operating experience. Sodium cooled reactors have demonstrated high level of safety and reliability, and according to the development made these have high potential in terms of self-protection and improved safety.

Currently, the development of a large size fast reactor concept with the improved safety is being initiated in Russia. It is supposed that this will be three circuit facility of about 1600 MWe capacity with passive systems of the reactor shutdown and air heat exchanger based decay heat removal.

Within the framework of the second direction, works on justification of BREST-300 lead cooled fast reactor are set in Russia. The reactor construction is planned in the vicinity of Beloyarskaya NPP site.

The key problem to be solved for the reactor design justification is successful mastering of lead coolant technology.

The international conference on “Heavy Liquid Metal Coolants in Nuclear Technologies” was held last October in Obninsk, where the comprehensive information on BREST-300 reactor design (Fig. 5) was presented.

The main characteristics of the reactor are presented in Table 3.

Table 3. Design characteristics of BREST-300 reactor with lead coolant

No.	Characteristic	Unit	Value
1	Reactor thermal power	MWth	700
2	Reactor electric power	MWe	300
3	Coolant		Lead
4	Core inlet coolant temperature	°C	420
5	Core outlet coolant temperature	°C	540
6	Maximum lead velocity	m/s	1.8
7	Maximum temperature of the fuel element clad	°C	650
8	Steam temperature at the SG outlet	°C	520
9	Steam pressure at the SG outlet	MPa	24.5
10	Core life	year	5
11	Refueling interval	year	1.0
12	Core breeding ratio		~ 1
13	Core diameter	mm	2300
14	Core height	mm	1100
15	Fuel element diameter	mm	9.1/9.6/10.4

Fig. 1. Change of frequency of NPP abnormal operation cases in Russia during 1992 - 1998 period

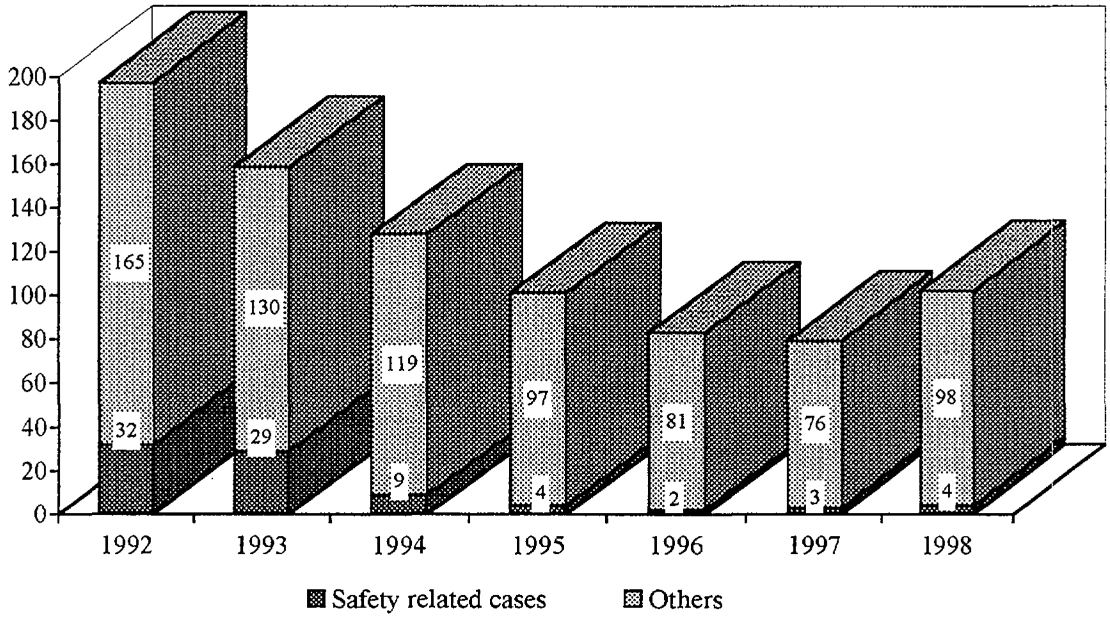


Fig. 2. Change of frequency of safety related abnormal operation cases in NPP during 1992-1998 period

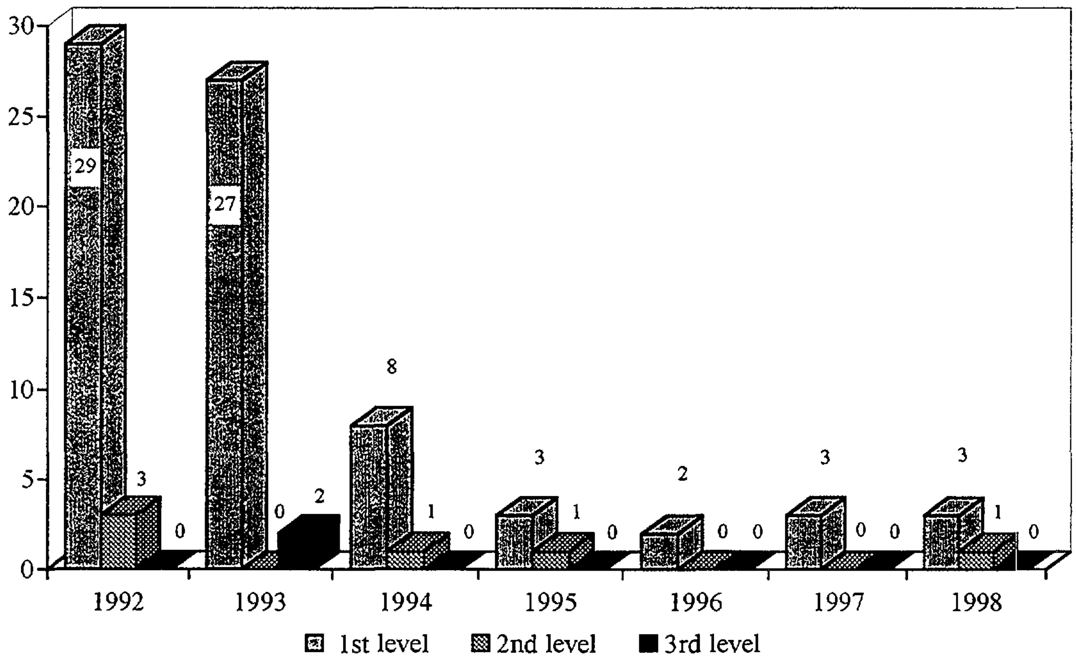
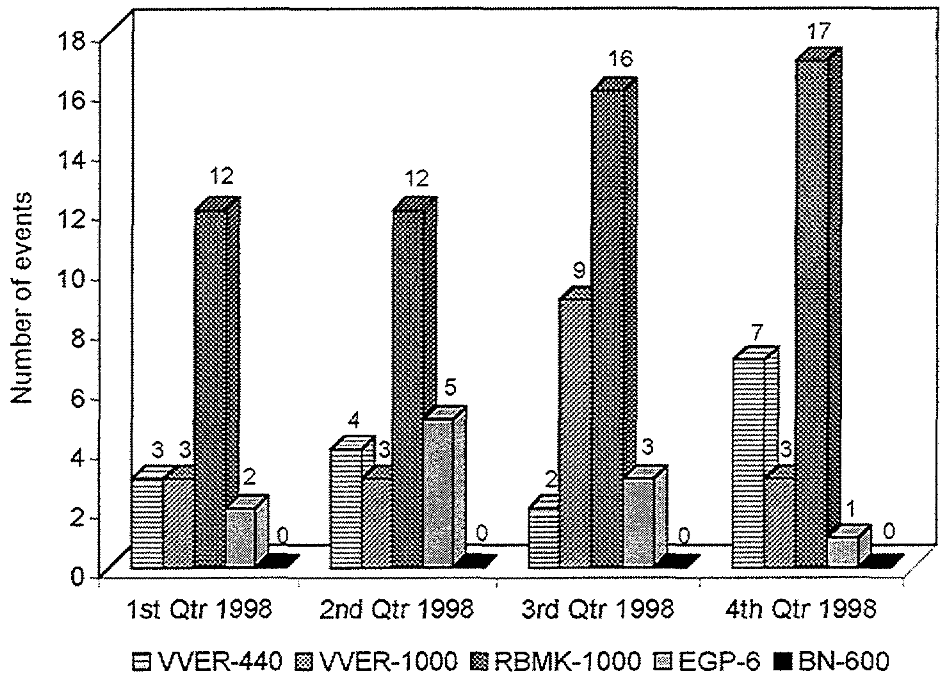
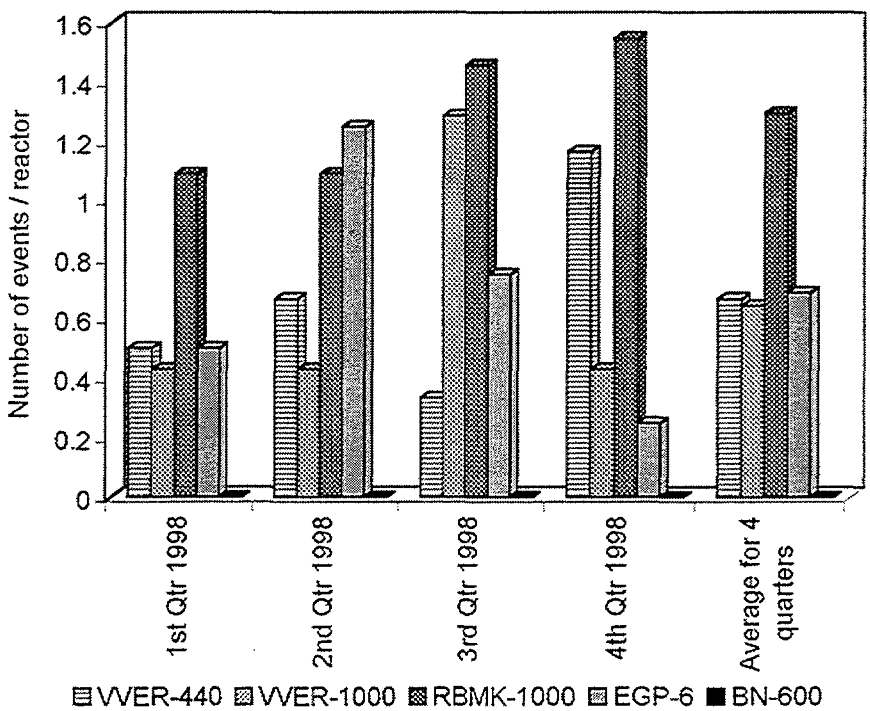


Fig. 3. Distribution of NPP abnormal operation events between the reactor types in 1998

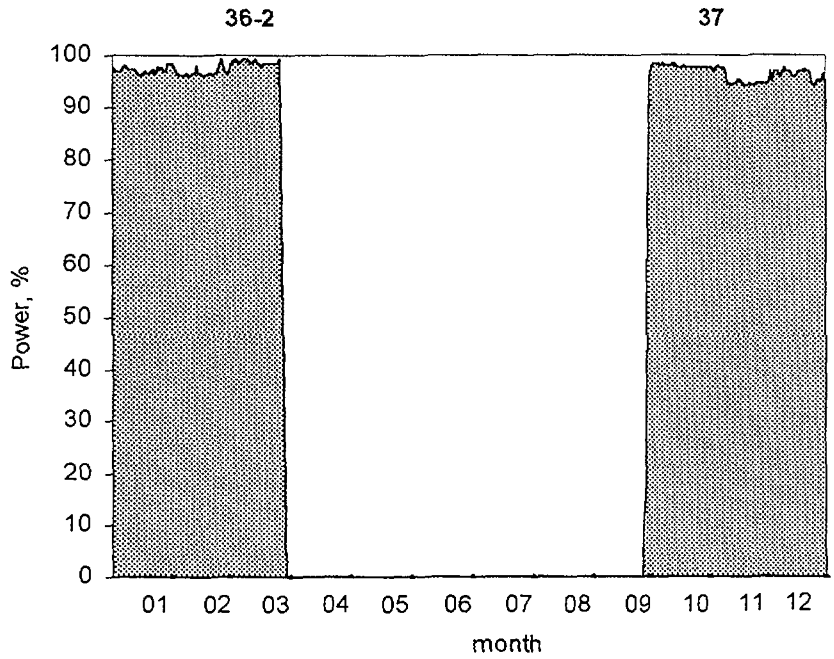


a)



b)

Fig. 4. BN-600 Operating Histogram in 1998 year



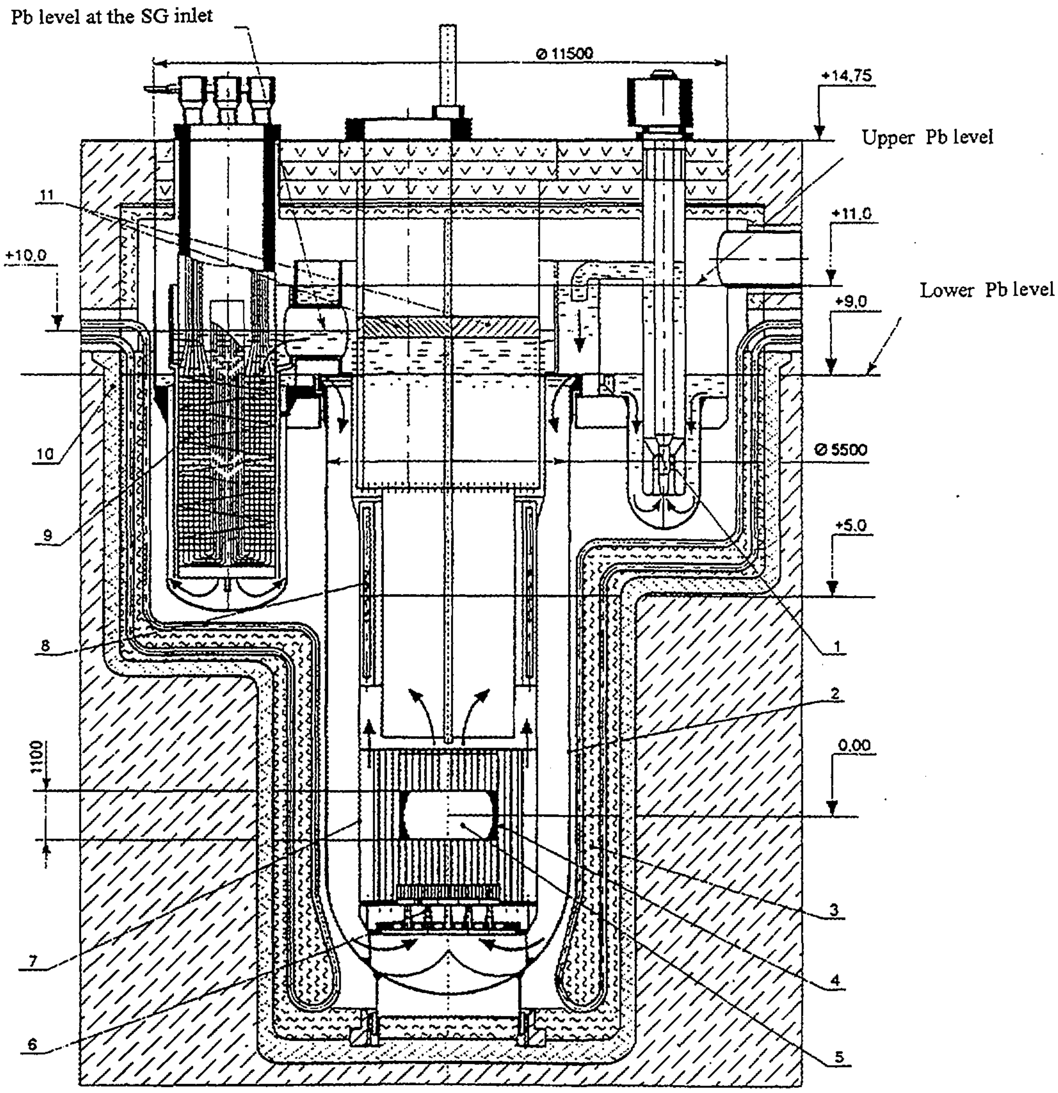


Fig. 5. Elevation of BREST–300 reactor

1 - pump; 2 - vessel; 3 – thermal insulation; 4 – control and safety rods; 5 - core; 6 – support sticks; 7 – separating casing; 8 – SA storage; 9 – steam generator; 10 – concrete cavity; 11 – rotating plugs