STATUS OF FAST REACTOR ACTIVITIES IN THE RUSSIAN FEDERATION

M.F. TROYANOV, A.A. RINEJSKIJ

INSTITUTE OF PHYSICS AND POWER ENGINEERING

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1.1. THE CURRENT STATUS OF POWER IN REPUBLICS OF THE FORMER USSR

Still before the disintegration of the USSR and CIS formation, the Power Program of the country was being developed. All republics presented information and their proposals on the state and development of the fuel and power complex.

The most significant, from our point of view, information prepared by republican and former Union structures is presented in Table 1. As is seen from the table, the per-head fuel consumption is maximum in the Russian Federation, 9.3 t/year*), then follow: Estonia, Lithuania and the Ukraine (7-8 t/year). The minimum per-head consumption of fuel and energy resources is in the Central Asian republics Tajikistan and Kirghizstan (2.3 t/year and 2.4 t/year, respectively). Only four republics are provided with energy resources of their own: Turkmenia, the Russian Federation, Kazakhstan and Azerbaijan. Moldova, Armenia, Latvia are least provided with fuel resources.

After passing over to selling of fuel resources at world prices some republics will have to spend annually substantial funds (the Ukraine - 9.0, Byelaruss - 3.4, Georgia - 1.2 milliards of US dollars).

But it is not only a matter of figures. The power industry is an integral system with its established technological and economic ties. For its functioning coordinated actions of all republics are essential. On the territory of the former USSR the Integrated Electric Power System has been created. It is hardly probable to ensure reliable power supply of the republics without the participation of the European Economic Community.

About 65 % of the total electric power generation fall

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*) For comparison, corresponding values for some other countries are as follows: USA - 10.8, FRG - 6.2, France - 5.0, Japan - 4.4, China - 0.94 t/man year.

Table 1.

POWER RESOURCES STATUS OF THE FORMER USSR REPUBLICS*)

	Fuel consumption ton/man year	Indigenous power resources, %	Fuel and power import cost, billion $/year
AZERBAIJAN	4.5	104	-
ARMENIA	4.4	4.3	0.9
BYELARUSS	5.4	11.5	3.4
GEORGIA	5.1	19.8	1.2
LATVIJA	5.5	8.1	0.8
LITHUANIA	7.2	24	0.7
KAZAKHSTAN	6.8	127	-
KIRGHIZSTAN	2.4	52	0.3
MOLDOVA	3.6	1.5	0.9
RUSSIA	9.3	146	-
TADJIKISTAN	2.3	40	0.5
TURKMENIA	4.6	650	-
UZBEKISTAN	3.1	84	1.0
UKRAIN	7.2	48	9.0
ESTONIA	8.1	75	0.4

*) 1990 Status, "Energia", № 7, 1991, p. 44-47

to the share of the Russian Federation.Of 515 mln. t of oil and gas condensate produced last year, 456 mln. t fall to the share of the Russian Federation.

The structure of the primary fuel and power resources reveals that both at present and in perspective the oil-gas complex will keep its leading role in satisfying the national economy requirements of the Russian Federation. According to forecasts, a proportion of oil will decrease (from 38.8 % in 1990 to 25-27 % in 2010), a proportion of gas will increase (from 38 % in 1990 to 46-49 % in 2010). The share of coal in the fuel-power balance in the nearest decade will be approximately stable (up to 14 %) and after 2000 it will start to increase and will grow to 15-16 %.

The development of electric power in Russia involves not only meeting its own growing energy needs but also electric power supply to other States (to Kazakhstan, Central Asia, the Ukraine).

As a whole, the fuel and power balance of the Russian Federation will remain to be an active one, as the Republic does not only meet its own needs but also is a large-scale supplier of fuel for export. E.g., in 1990, in the Russian Federation the difference between the produced and used in the Republic fuel was on the whole 637 mln. t.

1.2. THE STATUS OF NUCLEAR POWER

It is known that not all of the 45 power units in operation comply with the new safety requirements. This is also true not only for the CIS and East European countries. The first-generation nuclear power units were designed in accordance with less stringent standards as compared to the present-day ones.

By the present time there are realized measures on increasing the safety level of practically all operating nuclear power plants.This calls for the appropriate equipment and control systems supply for replacement and repair, for training and maintaining at a high level of operating personnel qualification.

The complex of all these activities with account of the current economic situation in republics will be difficult to realize in a short time.

Further development of nuclear power will be promoted, besides a drastic rise of prices for fossil fuels, by research and development activities on safety improvement. In new reactor concepts a basic approach is involved that is based on already developed technologies so that for the introduction of these reactors no prototype plants should be built but their commercial construction could be started at once. Within the frame of such an approach the projects of NPPs with medium-side reactors of 600-800 MWe and with large-size reactors of 1000-1600 MWe, both thermal and fast, are being developed. Of course, the pressurized water reactor remains the main reactor type as yet.

In order to achieve these aims as soon as possible it would be advisable for us to cooperate with industrial nuclear states.

In view of a drastic rise of prices for coal, oil, gas, the NPPs become economically competitive in most republics of the former USSR. At present the price of electricity generated at NPPs is much lower compared to that produced by fossil fuel plants.

2. FAST REACTOR OPERATIONAL EXPERIENCE

2.1. SOME GENERAL RESULTS

The experience gained in research and operating activities shows that the problem "sodium-water" does exist for the reliability of SG operating, but the accidents related with water penetration in sodium or in a box where steam generators are located would not cause the irradiation consequences. For this purpose in the project an intermediate circuit and special SG protection system were designed.

Certainly, sodium leaks and possible fires are still dangerous, but in the design these events are reduced to minimum. Basic accident of primary circuit loss of coolant is to bare the reactor core. In order to avoid this possibility guard vessel has been provided and siphon effect has been excluded. It's not expensive, and double wall vessel together with a low sodium pressure gave a decisive advantage in the safety of fast reactors in comparison with PWR type reactors. Another danger of sodium fire is caused by formation of radioactive aerosols. To reduce a probability of such an accident we used a pool-type configuration of the primary circuit, reduced the number of pipe-lines outside the reactor vessel, and provided the passive means in the design to eliminate the sodium fire. A danger of fires caused by secondary sodium leaks is eliminated using the same method. Thermal effects of sodium burning are considerably less than effects of organic fuel burning (gasoline - by 4.6 times; wood - by 1.5 times).

Summarizing, we can affirm that the technical solutions reduce the fire and irradiation conditions caused by sodium leak.

Fast reactor operating personnel noted the reactor stability, ability of self-control using effective feedbacks, the stability of energy release distribution in time. Due to a high breeding ratio, during 0.5 year of operation there is no need to have exceeding reactivity. That is why even with erroneous manipulation by control rods, the reactor conducts as a single whole without formation some local critical masses. It is an important fast reactor advantage to compare to thermal reactors.

Experience gained is positive and exceeds expected result. Sodium technology has happened to be simple: coolant quality control, operating and maintenance, cleaning of devices from sodium residues, regeneration of cold traps became ordinary operations.

During fast reactor operation we could not avoid some sodium coolant problems:

- Oil penetration into sodium of the BOR-60 secondary circuit.

- Condensation of sodium vapour on cold surfaces of the BN-350 gas communication and rotating plugs and of the cold part of the roof of the BN-600 reactor vessel (that one likely caused the fall of accumulated deposits into sodium and resulted in slight primary circuit parameters changes).

- Contamination of sodium by cesium-137 due to fuel elements failure that made us implement a special purification of sodium from cesium.

- Tritium accumulation in cold traps is to be considered during their regeneration.

- Several incidents of sodium leak trough the seal failures and inflammation which were quickly eliminated.

We can appreciate the advantages of sodium coolant:

- Good sodium corrosion compatibility with structural materials up to the temperature of 700°C.

- High thermal conductivity.

- Low vapour pressure and high boiling temperature.

One may consider that the difficulties to achieve the sodium circuit reliability are behind now. The most serious of them were as follows:

- A failure of intercircuit tightness in steam generator units of BN-350 (17 occurrences in the period of 17 years) BN-600 (11 occurrences, 11 years). The majority of incidents happened during the first operating year and were connected with deviations in welding technology and testing of welded zones. The SG's thermal cycling during start-up process is very important.

- The design secondary-circuit overpressurization protection system in these incidents was effective enough. The purification of the circuit from contaminants (alkali, hydrides) in case of intercircuit loss of tightness is not complicated. However, the repair of failed SG module may not be possible to guarantee; in our practice there were examples when a damaged module had to be replaced.

- Cavitation damages of the BN-350 primary sodium pump impeller in the region of the overflow orifices and a damage of the BN-600 primary pump shaft and its coupling as a result of excitation of resonance rotating vibrations in the system "pump shaft - motor shaft".

- Hydraulic shocks during check valves closing at the outlet of the BN-350 primary pumps that made necessary to change their removable parts.

There were no other serious damages in the sodium circuit valves, intermediate heat exchangers, cold traps, refuelling systems.

2.2. THE BN-600 NUCLEAR POWER PLANT OPERATIONAL EXPERIENCE

2.2.1. General Remarks

On February 2, 1991 a leak in a reheater module on one of steam generators was detected. By estimates, 4 kg of water penetrated into sodium. The corresponding steam generator was shut off and the reactor was operating at 70 % power for one week. By means of valves and by freezing of the sodium piping parts the failed section was localized on water and sodium sides and the reactor upto its shutdown for operating at power level of 600 MWe with three steam generators in one of which two sections were out of work.

The principal operating characteristics of the BN-600 reactor nuclear power plant for the last years and from the start of operation are presented in the Table 2.

Operating histogram of the BN-600 reactor for 1991 is shown in Fig.1. During this year there have been two shut downs of the reactor. The first was caused by the need of planned repair and maintenance work and core refuelling in the first half of the year, and the second-mainly by the core refuelling in the second half of the year.

2.2.2. Radiological Conditions at the BN-600 Reactor

The perspectives of nuclear power development will undoubtedly depend on the NPP radiation safety level under various operating conditions. As known, the determing factors are radioactivety releases to the environment and personnel radiological doses during repair, inspection, transportation and maintenance activities.

Rarioactive inert gas and aerosol release into the stack of the BN-600 reactor during the whole period of operation is as follows:

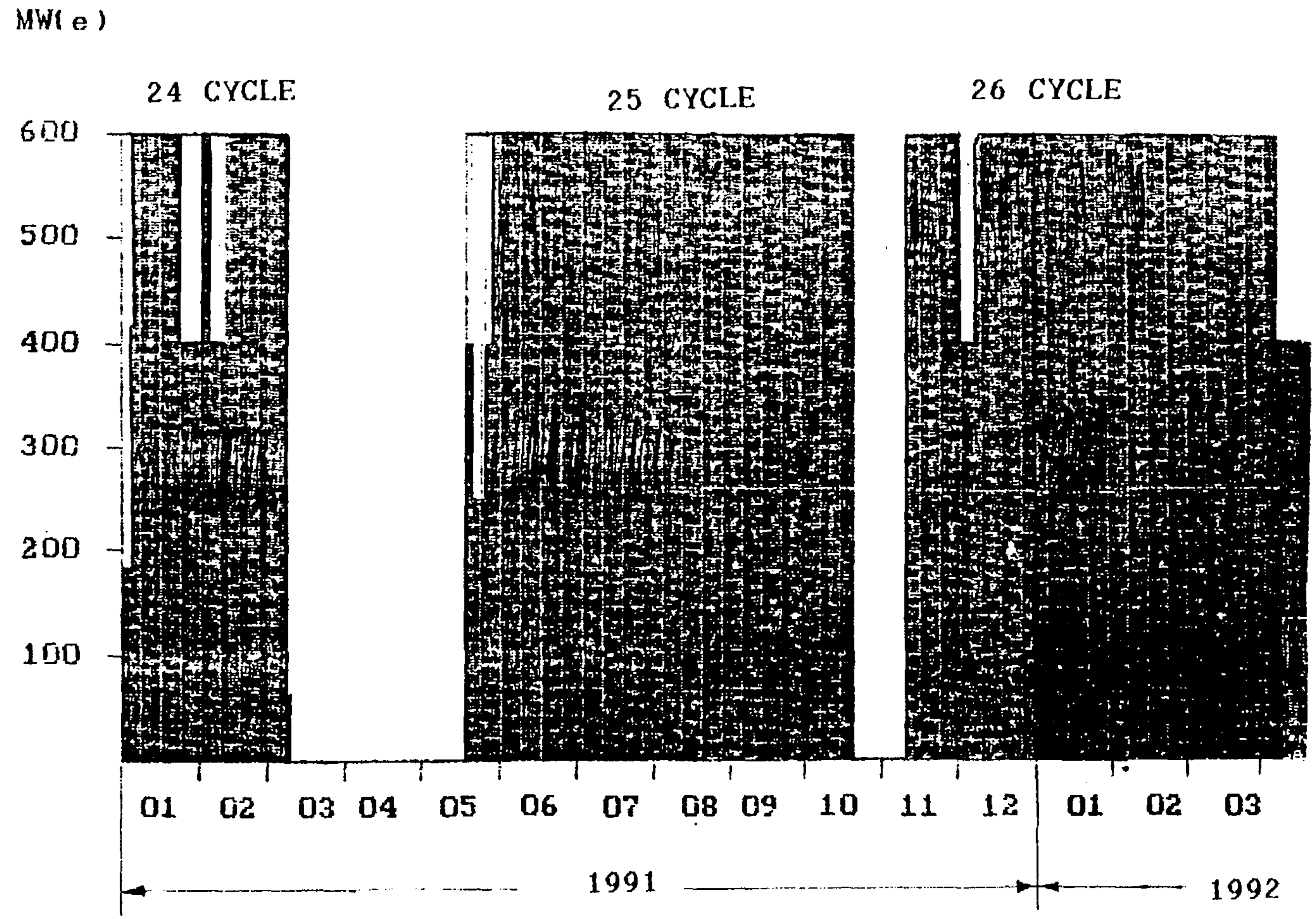


FIG.1 BN-600 OPERATING HISTOGRAM

Table. 2.

Main BN-600 reactor characteristics for the last 5 years and from the start of operation

No.	Characteristic	Units	1987	1988	1989	1990	1991	From the Start of Operation up to Jan.1,1992
1.	Electric Power output	10^6kWh	3895	4037	3988	3464	3670	40992
2.	Load factor	%	74.11	76.6	75.89	65.91	69.83	66.17
3.	The number of unit shut-downs		1	1	3	3	2	67
4.	The number of loop outages		1	2	1	7	4	55

Year	1980	1981	1982	1983	1984	1985	1986	1987	1988	1989	1990	1991
$\frac{\text{Curie}}{\text{day}}$	1.2	11.5	5.1	24.7	21.9	7.1	14.4	7.9	2.5	1.43	0.95	0.77

The main components of release are the hazardous noble radioactive gases. There has been practically no release of iodine-131, of long- and short-lived nuclides. A considerable decrease of inert gas activity release after 1987 is connected with the reduction of activity in the reactor gas blanket as a result of changing over to the advanced core. Radioactivity release is a few curies per day, at a permitted level of 500 curies.

A study of the radiological characteristics - on-site gammabackground, atmospheric fall-outs, flora and fauna radioactivity, - has revealed no radiation effect of the BN-600 reactor upon the environment.

2.2.3. Operating Experience with the Core

As was noted in our previous papers, in the BN-600 reactor the reference core design had been used till 1986. Its characteristics were as follows:

- relatively high linear power value on fuel pins (upto 540 W/cm);
- movement of the core fuel subassemblies from the periphery to the centre with their rotation through 180°.

5-year operation experience has shown that the design concept of the core does not allow to achieve high fuel burn-ups. Therefore, in 1986 work on the BN-600 reactor core modification was begun aimed at creating more favourable conditions for fuel pin operation. Core design with an increased height (from 75 cm to 1.0 m, at the expense of a corresponding decrease of the axial blanket and gas plenum height) has proved to be more preferable. There were used three instead of two zones of different fuel enrichment. All these have allowed to reduce essentially the linear power rating of fuel elements, has eliminated operations on movement and rotation of fuel subassemblies.

At present (the end of March - the beginning of April) the 26th run of the BN-600 reactor core is coming to an end. The core is operating without untight fuel elements. This run is characterized by that in this period the change-over to the core with maximum fuel burn-up of 10 % h.a. was begun, the maximum dose being 75 dpa. The core has been charged with fuel subassemblies the fuel pins of which have an increased height of the fuel column (1030 mm instead of 1000 mm) due to a decreased height of the lower axial blanket. Maximum values of linear power rating and cladding temperatures remained close to the former ones (475 w/cm and 696°C, respectively). The fuel subassembly duct material is EP-450 Type ferritic-martensitic steel (13Cr-2Ni-Nb-P-B), the fuel pin cladding material is ЧС-68 cw austenitic steel 16Cr-15Ni-2Mo-2Mn-Ti-P-B).

The performance of fuel subassemblies with such a combination of materials has been validated by positive experience of irradiation of experimental fuel subassemblies. In all, in the BN-600 reactor core there have been irradiated 250 fuel subassemblies with ducts of EP-450 steel and fuel pin clads of EP-847 cw steel (16Cr-15Ni-3Mo-Nb), EP-172 cw steel (16Cr-15Ni-3Mo-Nb-B),ЧС-68 cw steels. Of them, 11 fuel subassemblies (1397 fuel pins) have reached burn-up > 10 % h.a., 5 fuel subassemblies (635 fuel pins) > 11 % h.a.. Maximum burn-up was 11.7 % h.a., maximum dose 90 dpa. Now work on preparation of the required documentation and specifications to change over the BN-600 reactor core to a burn-up of 12 % h.a. is under way. The conceptual design has been developed, irradiation of experimental fuel subassemblies is carried out.

Similar work is carried at the BN-350 reactor. In this reactor core 50 fuel subassemblies with ducts of EP-450 steel were irradiated. Maximum fuel burn-up is 11 % h.a., the maximum dose - 92 dpa.

In the 49th run there were discharged 6 experimental fuel subassemblies with a burn-up of 13 % h.a.. Cold-worked austenitic steel was used for fuel pin clads, ferritic-martensitic steel - for ducts.

2.3. BN-350 REACTOR

The BN-350 reactor has been operated since 1973 at the rated power level (electricity and fresh water production). Reactor parameters are shown in the Table 3.

As it was described in our previous report, in January 1989 water into sodium leaks happened in the evaporator modules of two Chechoslovak made steam generators. Since that time four loops has been in operation, reactor thermal capacity being of 70 %.

There is additional data on steam generator loss of tightness: sodium leakage from the evaporator vessel took place as a result of the wall burnthrough caused by sodium-water reaction after one tube failure (Fig.3). The mass of sodium leaked is appreciated to be 1 t. Sodium poured into the guard vessel, surrounding SG modules, where it was extinguished by means of the nitrogen supply; the increased content of nitrogen was maintained until the metal temperature was lowered. The vessel wall burnthrough occurred because of the SG module design: there is little distance between the wall and tubes, and the spray of water from the failed tube was directed straight to the wall.

Now one of the two steam generators is completely repaired, and the other is under the repair. Operating histograms of the BN-350 reactor are shown in Fig.2..

2.4. BOR-60 REACTOR

In 1991 experimental studies have been carried out on the BOR-60 reactor in order to detect sodium boiling and gas cavities appearance in the experimental core fuel subassemblies.

The BOR-60 facility is the first experimental nuclear power plant with the fast reactor of thermal capacity up to 60 MW and electric capacity up to 15 MW. The BOR-60 reactor has been played a great role in validation of principal engineering solutions for the BN-350 and BN-600 reactors. On

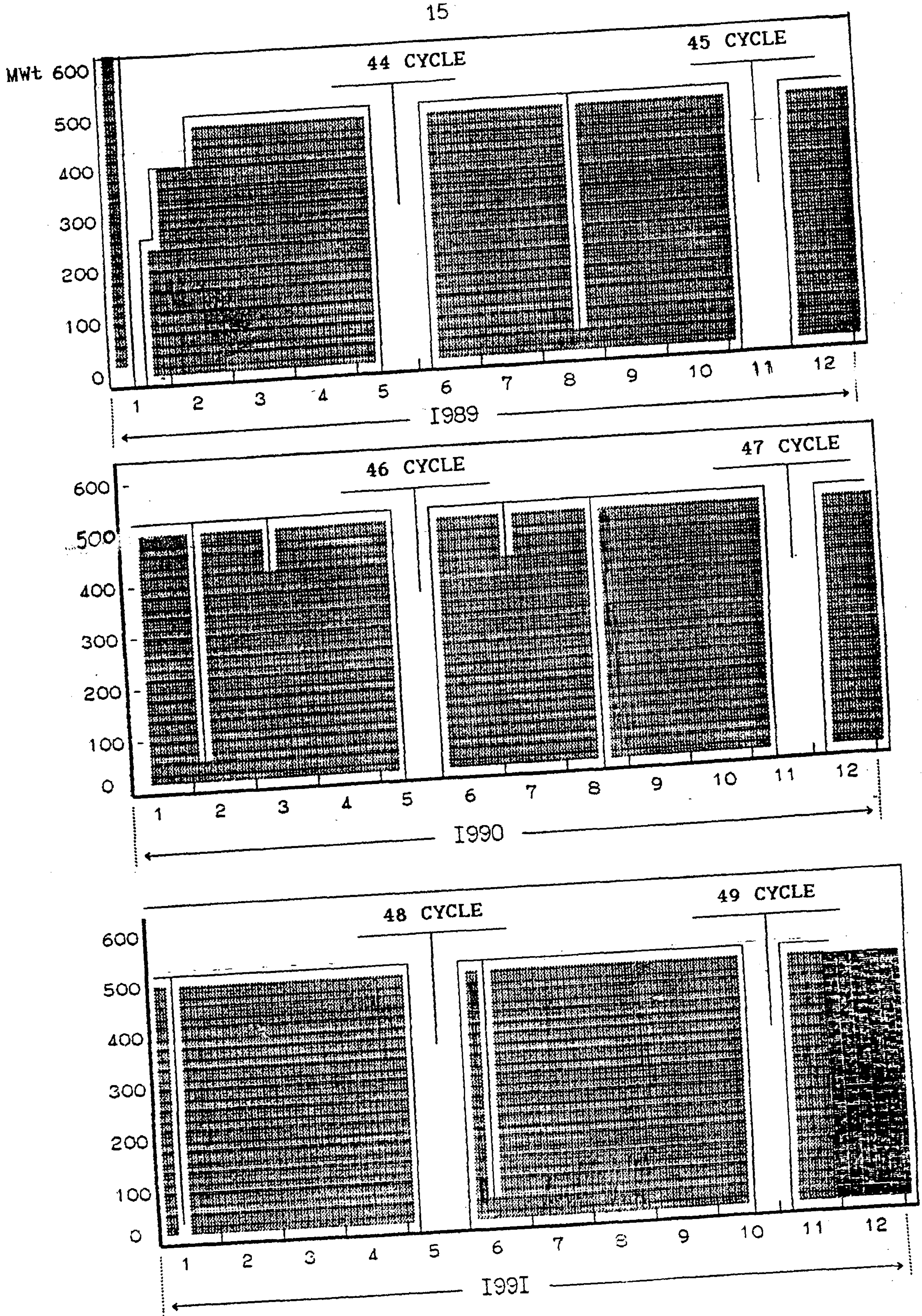
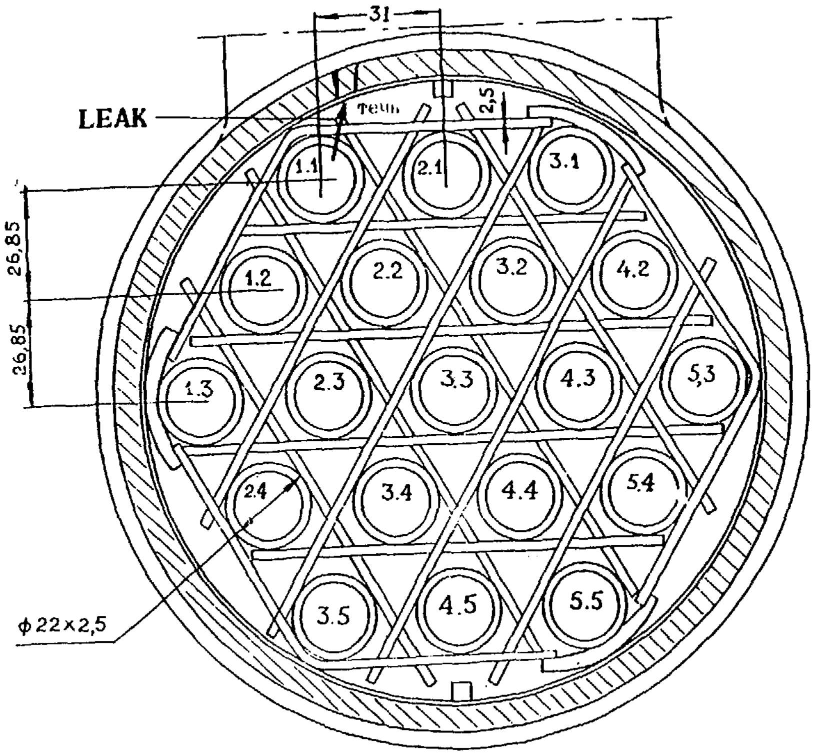


FIG. 2. BN-350 OPERATING HISTOGRAM



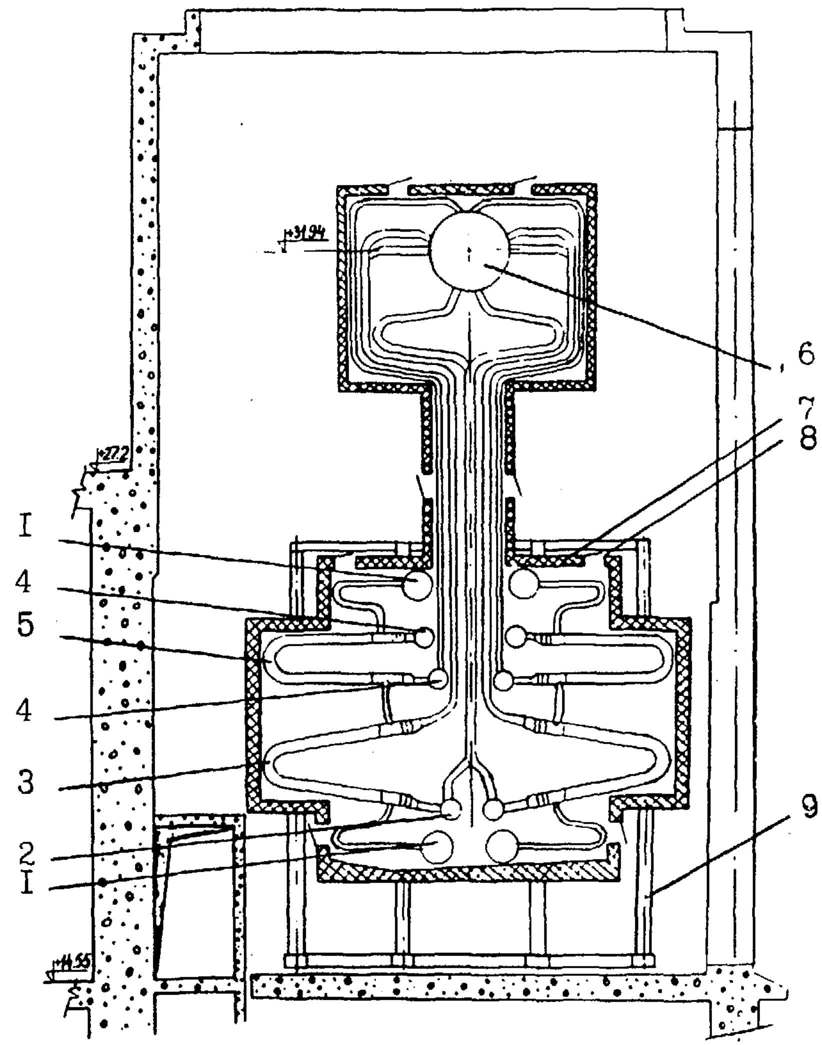


FIG.3 BN-350 STEAM GENERATOR

1. SODIUM HEADER; 2. WATER HEADER;
3. EVAPORATOR MODULE; 4. STEAM HEADER;
5. SUPERHEATER MODULE; 6. STEAM DRUM;
7. INSULATED AIR-TIGHT HOUSING;
8. AIR CIRCULATION VALVES;
9. SUPPORT STRUCTURE.

Table 3.

BN-350 reactor parameters in 1991

Parameters	Units	Value
Thermal capacity	MW	520
Electric capacity	MW	80
Desalinated water production	t/day	80000
Primary sodium temperature:		
- reactor inlet	^{o}C	288
- reactor outlet	^{o}C	437
Secondary sodium temperature:		
- steam generator inlet	^{o}C	420
- steam generator outlet	^{o}C	260
Superheated steam:		
- pressure	MPa	4.5
- flowrate	t/hour	750
- temperature	^{o}C	405

this reactor fuel subassemblies and steam generator modules of large power reactors designed in the USSR were tested. In recent years on the reactor vast experimental studies on new fuel technologies, structural materials have been conducted.

Since its commissioning (28.12.69) up to the beginning of 1991 the BOR-60 reactor had operated on power for 115832.5 hrs. The plant availability factor during 21 years of operation was 60 %. The total power output was as follows:

- thermal 225013 MWh
- electric 31777 MWh.

2.5. THE BR-10 REACTOR OPERATING EXPERIENCE

In 1991 the BR-10 reactor was operated with the 90 %-enrichment uranium mononitride core at a rated power level of 8000 kW. In the reactor core and nickel reflector channels experimental samples and devices were irradiated with fast neutrons. In the core, regular irradiation of sulphur in order to obtain radioactive phosphor-32 was carried out.

In the three vertical reflector channels the irradiation devices with a high flux of moderated neutrons (up to 10^{14} n/cm^2s) were tested. Water was used as moderator. It is assumed to generate molybdenum-99, mercury-197, aurum-198, iridium-192, iodine-131, cadmium-109, tungsten-188, etc. in these channels.

Work is nearing completion on creation of a plant for polymer film irradiation for the production of nuclear membranes with a pore density up to 10^9 pores/cm^2. For this purpose in the graphite thermal column of the reactor an uranium target is placed that will irradiate the moving film with fission fragments. After chemical processing the film will be used for filter fabrication. Experimental irradiation and etching of the film activity will be lower than 10^{-7}Ci/kg.

On the B-3 neutron beam the fast neutron irradiation of oncologic patients was carried out. From the beginning of the work 160 patients have been treated with a good effect.

In the reactor channels, colourless jewelry topazes were irradiated. After irradiation the topazes became intensively coloured.

There were continued measurements of concentration of tritium in primary and secondary sodium, in cover gas, in the interspace of the reactor vessel and primary piping, as well as in the technological rooms atmosphere.

3. POWER UNITS WITH THE BN-800 REACTORS

On the design stage the safety stresses became general. The project was reconsidered twice to satisfy the new safety demands. The decay heat removal system, based on passive principles, improved reactor control system, antiseismic devices and solutions, considerably reduced sodium piping system drastically changed the quality of the project concerning safety. The last significant change is dealing with a new core concept which assumes practically zero sodium void effect of reactivity. The construction of reactors started simultaneously both on Beloyarsk and South Urals sites in 1986 and simultaneously was stopped in 1990 because of the negative public attitude to the atomic energy due to Chernobyl accident, and because of the economical crisis in the country as well. Recently, the consent of the South Urals local authorities and government was obtained to continue the plant construction.

It was taken into account:

- The Chelyabinsk region where the power plants are going to be built suffers the lack of electrical power.
- In the neighborhood, the "MAYAK" plant dealing with the fissile fuel of VVER-440 reactors is situated, and here the plant for the mixed uranium-plutonium oxide fuel fabrication is being built.
- On the plant "MAYAK" four reactors for military purpose plutonium production were shut-down, and the problem of unemployment of well-qualified personnel must be solved.

Additionally. some ecological reasons are in favour of the power plant building. The complex commissioning that is expected till the end of the century permits us to implement for the first time the full fuel cycle.

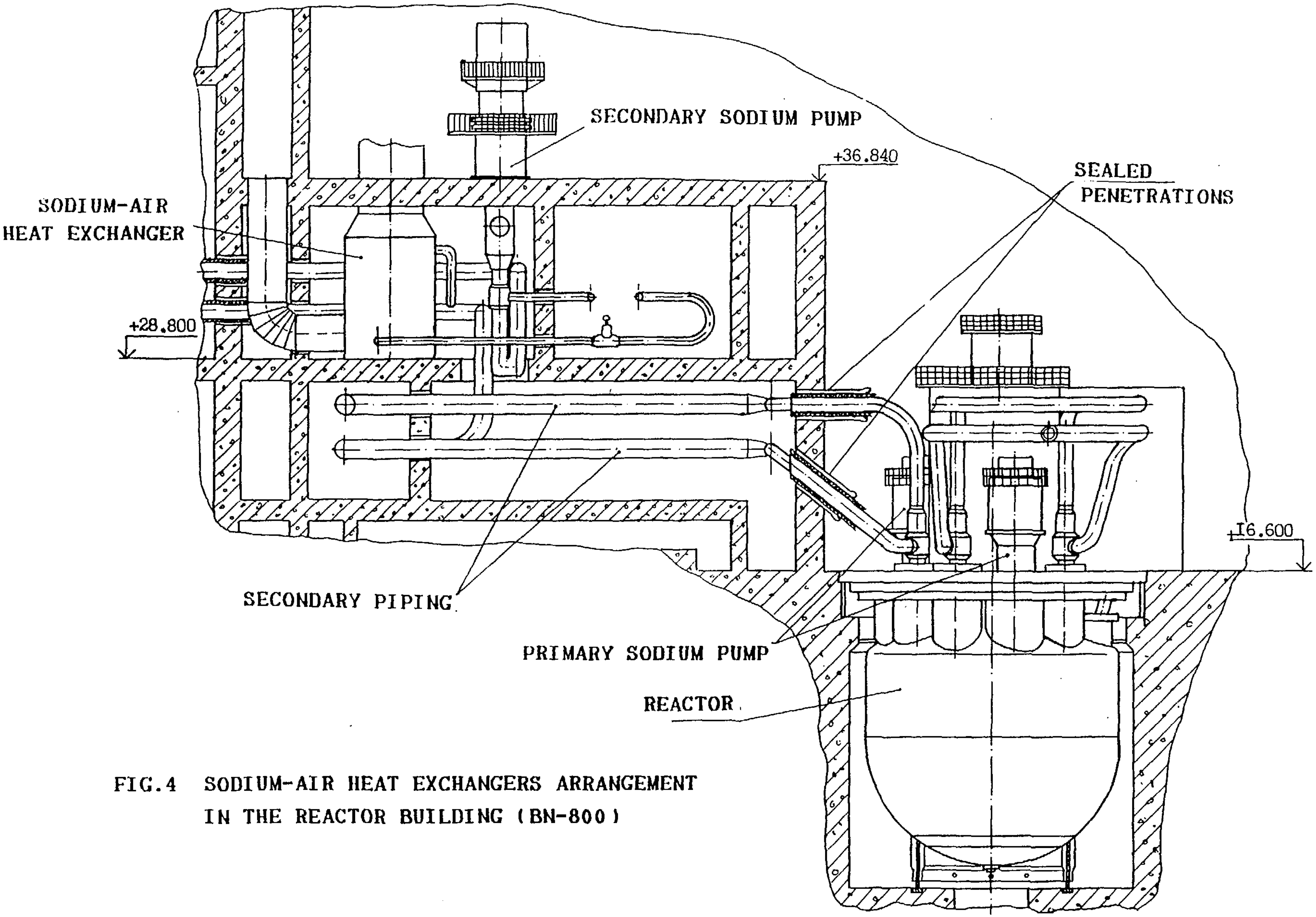


FIG.4 SODIUM-AIR HEAT EXCHANGERS ARRANGEMENT
IN THE REACTOR BUILDING (BN-800)

On the basis of the analysis of the design and research and development work performed we have come to a deep conclusion of its perspective, high operation and economic characteristics. The main thing here is the development and testing of all design components at the operating BN-600 reactor. The BN-800, as is known, is an advanced version of this reactor. During the past more than ten years period of highly reliable BN-600 operation all basic technological solutions used in the BN-800 have been confirmed.

The unit capacity of 800 MWe is very suitable for the present-day power systems of many countries.

4. WORK ON LARGE-SIZE REACTOR DESIGN

4.1. SOME CONSIDERATIONS ON POWER LEVEL

Last year due to funds limitations work on the BN-1600 reactor project was carried out with relatively small forces.

Primary emphasis was focused on estimating the effect of a power level on the following important characteristics of the reactor and of the power unit as a whole:

- Economical characteristics
- Safety
- Reliability
- Feasibility.

As a result of research and development work performed the following preliminary results were obtained.

Capital cost. It has been found out that the most effective way to reduce the components structure metal demand and, respectively, the fast reactor costs, is to increase their unit power with minimization at the same time of the number of equipment operating in parallel loops (pumps, heat exchangers, steam generators, etc.), i.e., due to their maximum enlargement in size.

Comparative characteristics of structural metal demand (tons/MWe) for the 1600 MWe three-loop reactor are presented in the Table 4.:

Table 4.

Reactor	Reactor unit	Heat transport system, including SG	NSSS
VVER-1000	0.91	2.20	3.11
BN-1600	1.55	1.20	2.75
EFR-1500	1.50	1.15	2.65
PWR-1300	0.59	1.55	2.04

The BN-1600 reactor by its metal demand compares favourably with the present-day thermal reactors. So, there are some reasons to state that by means of an increase of fast reactor power up to 1600 MWe, the development of a number of optimum design solutions and optimization of parameters it will be possible to improve considerably the economic characteristics of the LMFBRs and to solve the problem of their competitiveness relative to other sources of energy.

Nuclear and engineering safety. Calculations have shown that with increasing of core size and reactor power above 1000 MWe, at an optimum core characteristics and fuel type, it is possible at the same time to fulfil the following conditions: SVER < 0 and BR_{core} > 1 characterizing reactor nuclear safety.

The engineering safety problems - reliable tightness of the radioactive circuit, decay heat removal systems operating on the basis of passive principles, -are successfully solved in a large size reactor.There is a possibility to accommodate in a large-sized vessel all auxiliary systems with radioactive coolant (cold traps, intermediate sodium storage for spent fuel subassemblies, etc.), special heat exchangers for decay heat removal connected to the sodium-air heat exchangers. As a result, there are practically precluded radioactive coolant leakages from ramified auxiliary systems of conventional design, in specially arranged parallel decay heat removal channels heat is removed due to natural circulation of coolant.

Operational reliability. Calculations were carried out using statistical data on reactors in operation in our country: BN-600, BN-350, BOR-60. The following circuit design was considered: 3 primary pumps, 6 intermediate heat exchangers, 6 secondary pumps, 6 steam generators and two turbines. It has been found out that at improvement of some equipment units with due regard for the available operating experience the BN-1600 NPP load factor at a level of 70-80 % can be assured.

Feasibility. Analysis and design experience have shown that all in-vessel components of the 3-loop reactor (pumps, heat exchangers, the central column, etc.) can be carried by rail in the assembled form and the vessel can be assembled on site using the developed techniques.

4.2. MAIN LAYOUT DECISIONS

Pool concept with arrangement of all the equipment in the reactor vessel was adopted for the primary system.

The plant's secondary system consists of three loops, each of them icluding two IHXs, SG, secondary pump, installed in cold leg and secondary pipelines, the thermal expansion compensation of which is provided by the use of multi-layer bellows.

BN-1600 reactor is placed in a pit, the concrete walls of which are covered by leaktight metal lining. The pit lining above the reactor head goes over into protective dome under which control rod drives, primary pump drives and in-reactor refuelling mechanisms set are located.

The plant equipment including secondary pumps is placed inside 45 m diameter concrete containment. Air heat-exchangers of DHR system and SGs are located in separate boxes outside the containment (see Fig. 5).

BN-1600 reactor design represents a tank with toroidal-spherical bottom and flat head (Fig. 6). Reactor

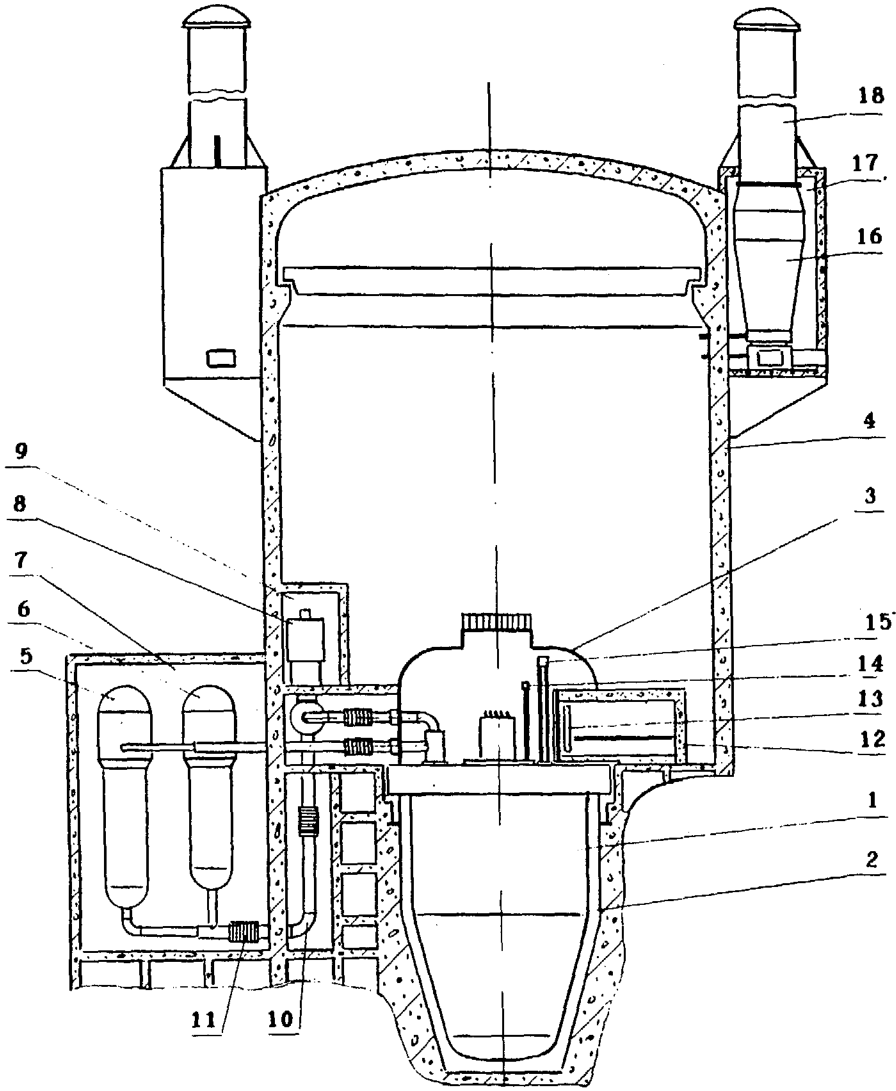


FIG.5 BN-1600 NUCLEAR ISLAND BUILDING LAYOUT

1-reactor; 2-reactor plant; 3-protection dome; 4-containment; 5-steam generating module; 6-intermediate steam superheater; 7-SG box; 8-secondary pump; 9-pumps box; 10-secondary pipelines; 11-sylphon compensator; 12-transfer box; 13-refuelling machine; 14-refuelling mechanism; 15-vertical elevator; 16-ERHRS air HX; 17-air HX box; 18-chimney.

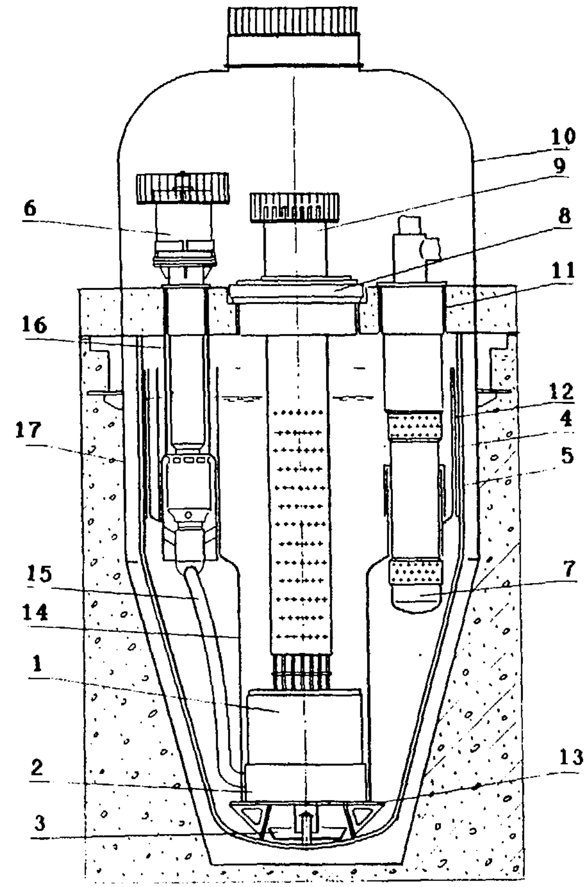


FIG.6 PRIMARY SYSTEM, CROSS SECTION

1-core; 2-discharge chamber; 3-tray; 4-vessel; 5-guard vessel; 6-primary coolant pump; 7-IHX; 8-rotating plug; 9-CPS column; 10-protection dome; 11-reactor head; 12-vessel thermal stabilization shell; 13-core support structure; 14-"hot box"; 15-discharge pipeline; 16-pump barrel; 17-reactor pit lining.

head has a box-like structure. The lower head plate is protected by thermal insulation, cooling of the head is provided by air supply through special channels.

Core diagrid is mounted on a special support structure at the reactor vessel bottom. A core catcher designed for localization and cooling down of core fragments at beyond-design accident is set up under the diagrid.

Primary coolant pumps and IHXs are arranged in penetrations of reactor head that supports them. Compensation of thermal expansions between primary system equipment and reactor vessel structures is executed by compensation of pumps discharge pipelines and gas gates, making IHXs openings in the "hot box" leak tight, excluding the leakages between hot and cold reactor cavities.

Central column with reactor control rod drive mechanisms is mounted on rotating plug besides refuelling mechanisms. The drives are of two types based on active and passive principle of operation. The first type of drives is an active one actuating by the signals coming from control system. The second type of drives with several varieties is a passive one.

Above there was presented one of the reactor designs under consideration - with the increased vessel height, practically without in-vessel neutron shielding. At present the developments of other versions aimed at choosing the optimum one are carried out.

5. FAST REACTOR R&D, DESIGN AND MANUFACTURING ORGANIZATIONS

Fast reactor R&D, design and manufacturing activities are carried out by research institutes, design and construction organizations of Russian Ministry of Nuclear Power. The main participants of the project are as follows:

- Institute of Physics and Power Engineering (Obninsk) - scientific supervisor of the design;
- Experimental Machine Building Design Bureau (Nizhny Novgorod) - general design organization of the BN-350, BN-600

and BN-800 reactors;

- Experimental Design Bureau "Hydropress" (Podolsk) - general design organization of the BOR-60 reactor and BN-350, BN-600 and BN-800 steam generators;

- Research Institute of Non-Organic Materials (Moscow) - general design and technology development organization of fast reactor core elements;

- Sankt-Petersbourg Research and Design Institute of Complex Power Technology - general design organization of the NPPs with BOR-60, BN-350 and BN-800 (South Urals site) reactors;

- Sankt-Petersbourg Research and Design Institute ("Atomenergoproject") - general design organization of the NPPs with BN-600 and BN-800 (Byeloyarsk site) reactors;

- Research Institute of Nuclear Reactors (Dimitrovgrad) - irradiation tests of fuel, absorber and structure materials, fuel cycle, sodium technology;

- "Polymetall" works (Moscow) - design and technology developing enterprise of absorber rods;

- "Atommash" Machine Building Factory (Volgodonsk) - the BN-800 reactor and equipment manufactures;

- Machine Building Factory* (Podolsk) - manufacturing enterprise of the BN-350, BN-600 and BN-800 reactors and fast transfer equipment (SG, IHX).

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* Not relating to Russian Ministry of Nuclear Power.