JPRS/L 3082

24 July 1969

NUCLEAR REACTORS IN THE SOVIET UNION

BY

DEVANA LAVRENCIC

- USSR -

JPRS/L 3082

24 July 1969

NUCLEAR REACTORS IN THE SOVIET UNION

- USSR -

[Article by Devana Lavrencic*; Rome, Notiziario, Italian, May 1969, pp 51-65]

Soviet experts believe the basis for the choice of a breeder clearly favors the adoption, in power programs, of high speed sodium reactors, initially functioning as converters and later passing to the phase of self-fertilization at a doubling time of seven to eight years. The construction of a nuclear plant takes 4 to 5 years; 3 to 5 years to obtain operating experience. Thus, the assessment of results that can be obtained with a given plant can be carried out in a time period of about 10 years. Therefore, if an industrial high-speed reactor is to be developed, and if the necessary industry is to be developed by the end of the 70's (a period of time during which it is expected that competitive high-speed nuclear reactors will be built in other countries) energetic action is now necessary.

There are two main reasons for the development of high-speed reactors: the first refers to low-cost reserves of uranium minerals, and the second refers to construction of nuclear plants for the simultaneous production of electricity and heat, as well as of very low-cost fuel.

Reserves of U minerals at 10 $\div$ 22 \$/kg known about so far are completely inadequate for a construction program of large electronuclear plants equipped only with thermal reactors. During its life -- calculated at about 30 years -- it is believed that a 1,000 MWe reactor, of the light water type, supplied with slightly enriched uranium requires $\simeq$5,000-6,000 t of natural uranium (170 divided by 200 t/year). For an installed nuclear power of 100,000 MWe, therefore, 500,000 to 600,000 t of natural uranium in a 30-year period would be required.

* Daise [Economic Research Service]

[I - USSR - K GUO]

In the USSR, the doubling of installed power takes place every 7 years; however, a doubling every 10 years is expected beginning in 1980. On the basis of these hypotheses, Soviet experts estimate that in the year 2000, installed power in the country will be $\sim 2 \times 10^6$ MWe. Installed nuclear power at the end of the century is estimated at 30 $\div$ 50 percent of total installed power, corresponding to several hundreds of thousands of MWe.

The construction of nuclear plants equipped with tested thermal reactors is justified in those regions where they can be economically competitive in the coming years. It is known that in the USSR, such reactors are those equipped with VVER reactors (with water under pressure) and U-graphite-H_2O, and the regions where they will be built are situated in the European part of the USSR and in the extreme north. Since the far eastern regions of the USSR have enormous reserves of low-price coal as well as water reserves, they will not need nuclear power for a considerable period of time.

The study of possibilities of the industrial use of high-speed reactors began in the USSR in 1949. The importance of experiments and plans showed it was possible to build high-speed reactors, and among them, the most promising were those cooled by liquid sodium. The multiplicity and complexity of problems involved in the use of high-speed reactors for the production of electrical energy required the construction of a series of experimental high-speed reactors.

The high-speed reactor development program went through the following phases:

1. BR-1, BR-2 - 1955-1956 -- Low power, experimental high-speed reactors.

2. BR-5 - 1958 -- High-speed experimental reactors for use in physics, thermodynamics and engineering research.

3. BN-50 - 1958-1960 -- Prototype high-speed reactor - project BFS-1 (1961), critical overall.

4. BN-350 - 1969 -- High-speed reactor of 1,000 MWt, planned on the basis of operating conditions of the BR-5.

5. BOR-60 - 1968 -- 60 MWt high-speed prototype reactor planned on far more advanced bases than the BR-5.

6. Na circuit on the MIR reactor -- 1968 BFS-2 (1969) critical overall.

7. BN-600 -- First 5 years of the 70's, economically competitive, electricity-producing, planned on the basis of the characteristics of the BOR-60.

BR-1, BR-2

The high-speed research reactors BR-1 and BR-2 were developed for the study of physical characteristics, to develop theory and calculation methods for high-speed reactors systems and to study technological problems connected with the use of liquid metals as coolants.

By 1955 it was already demonstrated with the 50 W BR-1 that the conversion factor in the ^{238}U-Pu cycle in reactors with a hard neutronic spectrum can be equal to 2. On the basis of this, it was concluded that for efficient fertilization of the nuclear fuel, it is possible to successfully use high-speed reactors (Figures 1, 2 appended).

In 1956, the 100 kW BR-2 mercury-cooled reactor became critical for the study of dynamics, control and conversion factors. The BR-2 was dismantled and the BR-5 was built to replace it.

In 1957, it was decided that it was more convenient to use ceramic fuel and liquid sodium as a coolant for higher-power reactors.

In 1957 at Obminsk, a critical intermediate/high-speed BR-3 complex became operational. It was fueled with natural U and Pu (Figure 3 appended).

BR-5

The sodium-cooled 5,000 kWt experimental high-speed reactor was built in 1957-1958 for the purpose of studying engineering problems linked with the planning of high power electronuclear plants equipped with high-speed reactors.

The reactor's basic parameters are shown in Table 1.

Chief studies in connection with the reactor which became critical in 1958 are: the various types of fuel elements in conditions close to those of industrial reactors; the multicircuit system with liquid metal coolant; and conventional apparatus in operation conditions with liquid metal.

TABLE 1

Characteristics of the BR-5

Fuel	PuO_2
Charge	50 kg
Diameter of fuel rod	4 mm
Core dimensions	280 x 280 mm
Density of maximum power	500 kW/liter
Maximum neutron flow	8×10^{14} n/cm^2 s
Maximum thermal flow	1.2×10^{6}kcal/m^2/hr
Average Na temperature at core outlet	500°C
Na capacity through core	240 m^3/hr
Na speed	2.5 m/s
Coolant circuits	2

Operation of the Br-5 passed through several phases. Initially the apparatus functioned at a low power level without coolant. In January 1969 the Na coolant circuit was put into operation and studies of the physical characteristics of reactor were carried out at a power level not exceeding 1 MW (December 1958-June 1959).

The second phase began when the nominal power level was reached (July 21, 1959); later the reactor functioned at various power levels with an Na output temperature of 400-500°C (July 1959-October 1960). During the third phase the fuel reached an irradiation rate of 2 percent at the end of 1960. Meanwhile, in 1960 the Na emergence temperature was raised to 500°C to raise the temperature level for technological research on the BR-5 to the temperature range for industrial plants.

On January 1, 1961, the reactor had produced 21×10^6 kWh reaching an irradiation rate of 2.8 percent. On September 1, 1961, the thermal energy produced was 39×10^6 kWh and the maximum irradiation rate reached was 5 percent. In September 1961 the reactor was extinguished to examine the condition of the fuel elements because of considerable contamination of the coolant and the primary circuit (November 1960-September 1961). Following examination of the fuel elements and purging of fission products from the primary circuit, in March 1962 the PuO_2 core of the BR-5 was substituted by a core with PuO_2, and UO_2 elements. Eighty percent of the old PuO_2 fuel elements which had reached an irradiation rate of 5.8 percent were replaced in the center of the reactor for further use together with UO_2 elements to study the release of solid and gaseous fission products of the elements exposed to the coolant and the gas in the primary circuit in function of the reactor's temperature and power.

In 1962 and 1963 the reactor functioned at 20 percent and 40 percent of nominal power, according to needs of particular experiments, with Na output temperature of 300 to 430° C. During 1964 power was raised to 3,000 kW and later to 3,500 kW (70 percent of nominal). The maximum irradiation rate reached in PuO_2 elements until March 1964 was 5.8 percent (March 1962-November 1964).

The BR-5 was extinguished in November 1964 because of an increase in the release of fission products from the PuO_2 elements and in order to charge the apparatus with carbide fuel elements. The maximum irradiation rate of PuO_2 elements of the original charge was 6.5 percent the rate for UO_2 elements enriched with 90 percent of ^{235}U was 1.4 percent. The irradiation rate of U monocarbide placed in the reactor in 1964 reached 0.8 percent. In January 1965 the entire core was discharged and placed in storage.

Following recharge of the core, maintenance work was done, mainly to replace the drainage valves in the primary circuits and the cables in the various control systems because their properties of electrical insulation were unsatisfactory. Other experimental equipment was installed. In May 1965, the reactor became critical with uranium monocarbide fuel and the characteristics of the new core were studied; the reactor was raised to a power of 5,000 kW on June 25, 1965. In the change to U carbide fuel, the primary coolant was not changed; the primary circuit was decontaminated through cold traps before other maintenance chores were completed. The cold traps used in the primary circuit have the following characteristics:

volume	200 liters
capacity	1 m^3/hr
Na temperature	120°C
coolant fluid	toluol

As the toluol boils its vapors are condensed by water coils. The trap capacity is assured by the prevalence of primary pumps.

The characteristics of carbide fuel elements are included in Table I and the physical characteristics of the BR-5 with monocarbide fuel in Table II.

The reactor has been functioning at full power since June 1965. Until October 1, 1965 it functioned for 38,500 hours at various power levels, of which 12,300 hours were at 5,000 kW. During that time, total thermal energy produced was 104 x 10^6 kWh. The irradiation rate reached as of October 1, 1967 was 2.5 percent. In March 1968, the mean irradiation level was 30,000 MWd/t. It is hoped to achieve 100,000 MWd/t but more time is needed. In order to do this, the possibility of increasing the power of the BR-5 to 10 MW is being studied.

TABLE II

U Monocarbide Fuel Elements in the BR-5

Enrichment, % ^{235}U	90
Element section	hexagonal
2 apothems, mm	26.1
Thickness of hexagonal casing, mm	0.3
Element height, mm	833
Number of rods/element	7
Outside diameter of rod casing, mm	8.55
Casing thickness, mm	0.4
Diameter of fuel in the rods, mm	7.65
Height of fuel in the rod, mm	380

TABLE III

Physical Characteristics of the BR-5 with Monocarbon Fuel

Nominal power, kW	5,000
Core power density, kW/liter:	
average	210
maximum	285
Coefficients of disuniformity in heat production	k_r 1.14
	k_z 1.19
	k_v 1.36
Maximum neutron flow n/cm^2 s	4.8×10^{14}
Efficiency of control elements,	
% compensation cylinder	4.1
blanket compensator	4.8
automatic regulator	0.1
Efficiency of fuel elements, %	
U monocarbide	
core: center	0.84
periphery	0.36
Natural U at periphery of core	0.11
Effect of Na drainage from core, %	-4
Core temperature coefficient, 1/°C	-3×10^{-5}
Asymptotic coefficient of power, 1/Mw	-8×10^{-4}

The BR-5 has demonstrated the possibility of reliable prolonged operation with parameters close to those of the BN-350: mean power density 360 kW/liter (500 kW/liter maximum); Na discharge temperature from the core 500°C. The heat exchangers of the BR-5 primary (Na-NaK) function with Na for 60,000 hours, without losing impermeability at a thermal flow of 1×10^5 kcal/m^2h. The exchangers of the secondary (Nak-air) with U (ø 25 x 1.5) in steel 1Cr 18Ni 9Ti functioned for 16,500 hours at a mean thermal flow of 0.25×10^5 kcal/m^2h.

During this period, the safety rods were activated 227 times; about 35 percent of the times, this happened while power was being increased.

The 10-year operational span of the BR-5 and the positive experience achieved made it possible to abandon the development and planning for the 50 MWe BN-50 and to move on to planning and construction of the BN-350 with a thermal power of 1,000 MW.

BFS-1, BFS-2

The principal instruments for experiment 1 studies of the physics of high-speed reactors are the critical complexes BFS-1, which became operational in June 1961 and the BFS-2 which will begin functioning in 1969.

The vertically charged BFS-1 complex is placed in a steel vessel 2 meters high and 2 meters in diameter with a single hexagonal grid. The complex consists of 1,500 vertical tubes (ø 50 mm) arranged in a hexagonal lattice (pitch 51 mm) in aluminum or in steel 2,080 mm long and with the thickness of the wall equal to 1 mm. The tubes are charged with pellets of various materials to simulate the desired composition. The pellets are cylindrical, measuring 46.7 mm in diameter and 10 mm in height except for the fuel: in fact, the enriched metallic uranium jacketed with stainless steel and the Pu oxide encased in stainless steel are in the form of pellets measuring ø 47.6 by 5.6 mm.

The control system consists of 14 mobile tubes, generally of the same composition as the tubes that surround them. The control tubes are placed in fixed positions in the core. The core is not entirely surrounded by the reflector but only in one direction in order to achieve a simulation of the reflector for the entire thickness. Instead, in the opposite direction, there is a thermal column of graphite used to calibrate the detectors.

The electronic apparatus consists of three channels for tests, of three impulse channels and one channel for measurement of excursion. The impulse channels are usually used as channels for measurement of neutron density, coefficient of multiplication, etc. These devices permit control of all levels of operation of the complex from 10 W to 100 W. The maximum operating capacity of the BFS-1 is 100 W, but the usual operating level is 10 W (Figures 5, 6 appended).

It is expected that the BFS-2 will become operational in 1969. It is similar to the BFS-1 in structure, grid, tubes diameter, but it is larger in size. It is five meters in diameter and its working height is three meters. The BFS-2 will be used as a model for larger high-speed power reactors as well as for general work in the determination of impact sections.

Work of this kind will be transferred from the BFS-1 to the BFS-2 in order to permit utilization of the BFS-1 in connection with the microtron for the measurement of neutron spectrums with the time of flight method.

The microtron is an electron accelerator very similar to the microton-injector of the IBR installed at Dubna (Figure 7 appended) and differs from it only because of the horizontal placement of the surfaces of the orbits and because of a different charging system. The characteristics of the system are as follows:

Electron energy	30 MeV
Impulse duration	1-3 us
Impulse current	100 mA
Repetition frequency	5 div. by 500 imp/s
Neutrons emitted from the target	10^{10}n/imp

The resolution power of the spectrometer at a distance of 800 meters is 2-3 ns/m which corresponds to:

$$\frac{\Delta E}{E} = \begin{cases} 0.5\% \text{ for neutrons of } \sim 1 \text{ keV} \\ 15\% \text{ for neutrons of } \sim 1 \text{ MeV} \end{cases}$$

There are two intermediate experimental stations at a distance of 50 meters and 230 meters from the reactor.

The plan of the BFS-1 microtron is shown in Figure 8 appended.

BN-350

The BN-350 plant equipped with high-speed reactor is in the stage of advanced construction at Shevchenko on the Mangyshlak Penninsula, on the northeast shore of the Caspian Sea. It was planned for the production of electrical energy (150 MWe) and for the production of 120,000 t of desalted water per day. The plant is expected to begin functioning in 1969. For the initial period of work the experimental and verified parameters of the BR-5 (Figures 9, 10, 11 appended) were selected.

The main characteristics of the plant are shown in Table III.

The study of the functioning over many years of the BR-5 reactor -- which operates on parameters fairly close to those of the industrial reactors presently being planned -- and analysis of research and planning as well as construction tests permitted passing to the construction of a nuclear plant with high-speed reactors cooled by sodium with a thermal power of 1,000 MW (BN-350).

The BN-350 is the first large high-speed power reactor in the USSR calculated according to parameters already achieved and verified experimentally which guarantee adequate safety of the reactor.

The input temperature of the sodium coolant (500°C) permits the achievement of sufficiently high values for steam parameters and as a consequence, for output through the thermal section of the plant. The electric power of the plant will be 350 MW with use of turbine with standard condensation (K-100-45).

However, it is planned to use counter-pressure P 50 turbines for the BN-350. This guarantees the solution of two problems: developing an industrial plant of 150 MW and obtaining 1,200 t/h of industrial steam to be fed to the desalting plants for the production of 120,000 m^3 of distilled water per day.

The BN-350 plan permits the use, without special variations, of fuel of different composition. For the first cores the fuel will be a mixture of uranium dioxide and plutonium and in the first phase, enriched uranium dioxide.

The well-known advantages of ceramic materials are their high fusion temperatures, the isotropic structure, the absence of phase transitions and the capacity to guarantee a high level of irradiation without significant variations in dimensions. Furthermore, the oxide fuel is very compatible both with the sodium coolant and the casing material of steel.

Extensive experiments were made with the BR-5 regarding the stability of the fuel rods of the oxide type with stainless steel casings. The dimensions of the space between the oxide and the casing and the free volume were calculated beginning with an irradiation rate of 2 percent. Presently the irradiation rate reached is over 6 percent. The examination of the rods, which reached an irradiation rate of 5 percent, and then 6 percent, has revealed a satisfactory state of the casings for most of the rods.

Experimental research on variations produced by samples of uranium oxide and a mixture of uranium oxide and plutonium, when exposed to irradiation, confirmed that the irradiation rate easily reached 10 percent and over. Such irradiation rates can be obtained for thermal flows which reach a level of 2 X 10^6 kcal/m^2h.

TABLE IV

Characteristics of the BN-350

Thermal power, MW		1,000	
Electric power, MW		350	
Distilled water produced, 10^3 t/day		up to 150	
Fuel		PuO_2+U^{238} or enriched UO_2	
Collant		Na	
Core:			
volume, m^3		1.87	
diameter, m		1.5	
height, m		1.06	
composition	fuel	Na	steel
	44.7%	31.1%	24.2%
Blanket composition:	fertile material	Na	steel
axial	43.8%	41.9%	14.3%
radial	60.8%	21.6%	17.6%
Power density, kW/liter		500	
Maximum thermal flow, kcal/m^2h		1.86 x 10^6	
Maximum temperature of shell, °C		675	
Maximum irradiation level, MWd/t (%)		50,000-60,000 (5)	
Core life (charge factor 0.85)days		300	
Pu^{239} charge, kg		850	
Conversion factor:			
total		1.5	
of the core		0.7	
Maximum Na speed, m/s		8	
Na temperature, °C			
at input		300	
at output		500	
Steam parameters:			
temperature, °C		430	
pressure, atm		50	

The mean density decrease of the dioxide within the shell compared to that of the rods in the BR-5, of up to 8 g/cm^3 (maintaining the density of the pellets at 10-10.5 g/cm^3 made it possible to reach an irradiation rate of the fuel in the BN-350 reactor above 5 percent.

In order to improve the coefficient of conversion and other characteristics, a levelling of thermal density was achieved in the reactor through the use of fuel with two different degrees of enrichment.

In the core with a fuel containing a single enrichment, the disuniformity coefficient of power density is equal to 2. The principal disuniformity is found in a radial direction (1.67).

The reactor and the pressure manifold are placed in a vessel filled with sodium. The storage facility for spent fuel and the first stratum of the shield are placed in contact with the radial blanket.

In the center of the core are placed seven compensators, two regulating rods and three safety rods. The reactor vessel is closed above with two rotating plugs with mechanisms for recharging and for activating the safety and control systems.

In the radial direction, after the container, there is a thermal insulation stratum and the lateral biological shield. In a special lining of graphite within the biological shield are placed the ionization chambers of the safety and control systems. The activating mechanisms for the safety and control systems are placed above the rotating plugs on a special support. The gas chambers of the mechanisms are linked with the gas chambers of the reactor, filled argon. The sodium level of the reactor is selected so that fuel elements extracted from the core or from the blanket remain below the sodium level during transfer to the internal storage place.

Core and Blankets

The core is 1,500 mm in diameter and 0.060 mm high. It is composed of 211 hexagonal fuel elements; 12 positions in the core are occupied by the organs of the control and safety system. Each element contains 169 fuel rods placed at 7 mm intervals in a triangular grid. Fuel consisting of groups of rods was selected for the BN-350; the external diameter of the rod is 6.1 mm. The rod casings are stainless steel, 0.4 mm thick. The PuO_2 and UO_2 fuel has a density of 10.5 g/cm^3. The mean volumetric density of the fuel rod is 8 g/cm^3. The core is surrounded by a blanket 600 mm thick which includes both the axial as well as the radial blanket. In the hexagonal box of the core element at the extremity of the rods are placed 37 rods (external diameter 12 mm) which contain weak UO_2. The density of the uranium dioxide pellets is 10.5 g/cm^3, the mean density of the rod sections is 9.5 g/cm^3. The rod casing in the axial blanket is 0.4 thick and is made of stainless steel. The portion of the UO_2 in the axial blanket is equivalent to 43.8 percent. The lateral reflector is composed of 440 hexagonal elements whose external configuration is similar to that of the core elements.

Each element contains 37 weak uranium rods (external diameter 14.2 mm) placed in a triangular grid (at a pitch of 14.8 mm). The casing of the radial blanket rod, which is 0.5 mm thick, is in stainless steel. The portion of UO_2 is 60 percent. The height of the radial blanket rod is 2,400 mm. The fuel elements of the core and of the radial blanket are at a distance between the opposite sides of the hexagon of 96 mm and the thickness of the hexagonal box is 2 mm.

The elements are placed in the reactor at a pitch of 98 mm. To the hexagonal box of the elements, at both ends, are welded the upper tong with hook for extraction of the element and the lower tong for fixing the element in the pressure vessel. The total length of the element is 3,600 mm. The upper ends are free and can be moved in the axial and radial directions within the limits of the tolerances of the materials. The lower tongs are made in the form of tubes with tapered ends; on the lateral surface of the lower tong there are openings for sodium input. By this method of introducing sodium, the possibility was eliminated that the element could rise under the action of the fluid's dynamic pressure.

The radial blanket is subdivided in regard to the system of heat extraction in the internal part containing 120 elements and in the external part containing 320 elements. Two rows of elements of the radial blanket (internal blanket) with high power density are included in the cooling system along with the core elements (to the high pressure vessel). These elements can be exchanged with core elements and vice versa; the core elements can be exchanged with elements of the first two rows of the blanket.

In this way. it is possible to correct the critical dimensions of the reactor.

The successive rows of elements of the radial blanket (320 elements) are connected to the low pressure vessel since the power density in that vessel is notably lower than that of the elements in the first rows. The radial blanket is surrounded by 41 channels for use as internal storage in which are placed, in the period between recharging, the fuel elements extracted from the core by decay. The last external row of the radial blanket can be used as a supplementary facility for internal storage of the elements.

The radial blanket and the storage facility are surrounded by a neutron shield in steel, 200 mm thick, designed to decrease the neutron flow which reaches the reactor vessel. When the reactor operates at nominal power, the temperature of the sodium at input is 300°C. The mean temperature of sodium output from the reactor is 500°C.

The maximum power density is 850 kW/liter. The temperatures and stresses of the fuel rod have been determined both for nominal conditions and under conditions which take into account possible deviations of various parameters from values corresponding to the nominal.

Reactor Vessel

The reactor vessel is a container of variable diameter made of stainless steel Cr18Ni9. The maximum diameter of the vessel is 6000 mm; the minimum 2,200 mm; wall thickness is 30 mm; total height 13,000 mm. The upper part of the vessel serves as a support for the revolving plugs and for the recharging mechanisms. The container is a load-bearing structure in which are placed: the central part of the reactor with the collector, the thermal shield of sodium which fills the vessel, safety plugs and recharging mechanisms supported by the vessel itself. In the central part of the vessel there is a flange fixed to the welded annular metal construction.

The lower part of the vessel forms a pressure barrel to which the sodium flows through six nozzles 500 mm in diameter. The flange of the pressure chamber serves as a support for the collector on which the core and the blanket are placed. In the upper part of the vessel there are six nozzles 600 mm in diameter through which the sodium flows from the reactor to the heat exchangers. The inside of the vessel is lined with sheets of "thermal shield" of stainless steel whose overall thickness is 60 mm. This is designed to decrease thermal stress within the reactor vessel -aused by changes in temperature. The use of the thermal shield made it possible to decrease the degree of thermal stress below the elastic limits of the vessel's material at operating temperatures.

In the annular space formed between the walls of the vessel and the thermal shield stratum runs the sodium which skims the container and the output nozzles and is mixed with the total flow. This flow of sodium in the space has made it possible to lower the operating temperature of the vessel from 500°C to 420°C. The flow of sodium through the annular space is equal to 2 percent of the total amount of sodium through the reactor. In order to prevent the emptying of the reactor in case of a leak in the vessel a protective sheathing 10 mm thick was provided (Figure 12 appended).

The Reactor's Safety and Control System

The control, regulation and safety system of the reactor is part of the centralized control system of the entire plant.

The system for safety and control is composed of the following mechanisms:

a) Systems for measurement and control of power and the reactor's period, including the subcritical period;

b) Automatic regulation systems;

c) Reactivity variation compensating systems;

d) Safety systems.

The characteristics of the control mechanisms are:

For the automatic regulation system (Figures 13, 14 appended)

Number of mechanisms	2
Number of rods for each mechanism	1
Distance from core center	98 mm
Rod efficiency	.20%
Rod stroke	750 mm
Maximum speed of movement	150 mm/s

For the reactivity variation compensation system (Figures 15, 16 appended).

Number of independent mechanisms	7
Number of elements per mechanism	1
Overall efficiency	2.1%
Weight of U/element	18 kg
Element stroke	1,060 mm
Speed of movement	10 mm/s
Movement pitch	20 mm

For the safety system (Figures 17, 18 appended).

Number of independent mechanisms	3
Overall efficiency	3.5%
Rod stroke	1,260 mm
Activation time	0.7 s
Lift speed	5 mm/s

The operating elements of the power controlled mechanism are the absorption rods of the compensating elements. Each power driven mechanism unit consists of a support with the rod, a ring device, coupled motor for the absorption rod or compensating element. The support with the rod is a unit fixed on the flange of the central reactor tube. The ring device is attached to the corresponding support with studs. The coupled motors of the power mechanism are placed on a special plate. The motors are coupled to the ring device with sealed motors. The rotation of the sealed electric motor is changed by means of the ring device in the rod traverse on whose lower end is a terminal with hooks by which the absorption rod is fixed. The hooks are controlled by means of levers of the upper part of the mechanism. There is also control of coupling and release of the rod and of the absorber by means of a shaft inside the rod.

The automatic regulation mechanism's absorber contains seven rods 9.6 mm in diameter, each of which contains B carbide with an 80 percent enrichment of B^{10}. In the upper part of the absorber there is a gas chamber which captures the helium formed by the boron.

Reactivity variation is compensated through movement along the core of six elements with active materials (burn-up compensators) and of one element with boron absorber (temperature compensator). The active part of the former is formed by elements structurally similar to elements of the core; the latter consists of elements containing boron carbide.

Reactor Shield

The shield has a series of special characteristics due to the high intensity of the flow of fast neutrons leaving the reactor. At the limit of the conversion zone the neutron flow is $5 \times 10^{13} n/cm^2s$ (taking into account the accumulation of plutonium in the blanket and the placement of fuel elements in the intermediate tank). In a radial direction, the shield comprises the primary shield within the reactor vessel, the secondary shield which decreases the flow of radiation toward the beton and the beton shield. The primary shield consists of steel blocks 200 mm thick placed immediately after the intermediate tank; of a stratum of sodium 500 mm thick; and a stratum of steel 60 mm thick. The production of heat in the tank wall is 0.1 W/cm^3 while the total neutron flow during a 20-year operating period is estimated at $5 \times 10^{21} n/cm^2$. The secondary shield is made of a stratum of steel (150 mm) and a stratum of iron oxide (1,000) and lowers the flow of radiation toward the beton to $5 \times 10^9 n/cm^2s$.

The normal beton thickness is 2,000 mm. The upper shield includes a stratum of sodium, a steel plate immersed in sodium to simplify the transfer of heat, and of alternating strata of iron and graphite.

BOR-60

The BOR-60 reactor was built at Melekess in the Institute for Scientific Research on Nuclear Reactors where it became critical for the first time on December 30, 1968. A photograph of the plant and a plan of the layout are contained in Figures 19 and 20.

It is a small nuclear plant with a complete turbogenerator system (the turbine power is 10 to 12 MW), a coolant tower, substation, batteries, etc. The characteristics are given in Table IV. [Tr. note: reference to Table IV may be erroneous; see Table V for BOR-60 characteristics].

The BOR-60 was built for the purpose of carrying out experiments with various types of fuel elements under conditions of high power density (900 to 1000 kW/liter), at high irradiation levels (up to 100,000 MWd/t) at an Na output temperature up to 600°C. As the figures show, these parameters permit development of nuclear plants with high-speed reactors with economic indices completely comparable to those of coal power plants. The achievement of such parameters will make it possible to build a high-power nuclear reactor as a prototype for a first series of nuclear plants which could assure production of electrical energy at low cost for heat production, as well as production of new nuclear fuel in larger quantities than those consumed.

The BOR-60 also will serve to assure reliable construction of the heat exchanger, steam generators and accessories which will work at high temperatures and for power densities greater than those obtained under conditions prevailing at the Shevchenko plant.

Enriched UO_2 and PuO_2 or U or Pu carbide may be used in the BOR-60 core. The fuel elements, at least in the first charge, will be hermetically sealed. The cylindrical lower end of the fuel element is welded below the core support plate while the upper part is free. The spacing system of the rods consists of helically wound wires soldered at the ends.

The height of the elements is 1,495 mm of which 400 mm is the active zone. Above this is a fertile zone of 150 mm and below it, of 100 mm. Beyond the fertile zones, at the two ends of the elements there are collection tanks for gaseous fission products.

A three-circuit cooling system was planned for the BOR-60: Na-Na-H_2O-steam. The two primary circuits are independent and each is estimated to produce 50 percent of the maximum power. The dimensions of the apparatus are such that the heat produced by the reactor can be removed at 3 percent of the power through natural circulation in the first, second and third circuits.

Each primary circuit contains a submerged centrifugal pump, the intermediate exchanger, control and shutoff valve. In every secondary after the heat exchanger and the pump there is a steam generator and a 30 MWt exchanger which can be coupled by any secondary circuit in place of the corresponding generator.

The steam generators of the various BOR circuits differ structurally: natural circulation generator (Figure 21) and single changeover (Figure 22).

TABLE V

Characteristics of the BOR-60

Maximum thermal power, MW	60
Maximum electric power, MW	12
Core dimensions: (cm)	
height	40
diameter	41
Maximum/mean power density, kW/liter	1,000/900
Maximum thermal flow, kcal/m^2h	2.65×10^6
Na temperature, °C	
input	360-450
output mean/maximum	580/650
Na capacity, t/h	750-1,000
Na velocity, m/s	10
Core Fuel Element	
shape	hexagonal
fuel	UO_2enr. 90% in U^{235}
number	70-80
U^{235} charge, kg	150-160
number of element rods	37
rod diameter, mm	6.1
working length, mm	400
rod casing	stainless steel
thickness, mm hermetic elements	0.4
ventilated elements	0.25
element casing	stainless steel
thickness, mm	1
Fuel Element of Radial Blanket	
shape	hexagonal
number of rods/element	7
rod diameter, mm	14.5
triangular grid pitch, mm	15.2
fertile material	weak UO_2 pellets
thickness rod casing, mm	0.35
length of fertile material, mm	900
mean radial blanket thickness, mm	150
number of core elements + radial blanket	259
Pressure Vessel	
shape	cylindrical
diameter at core level, mm	1,100
maximum diameter, mm	1,500
height, mm	6,500
wall thickness, mm	16-20
Na input duct, ø mm	1, ø 300
output duct	2, ø 300

TABLE V (cont'd)

Safety and Control System	
efficiency of the 3 safety rods, %	2.3
efficiency of temp. comp. rod, %	1.0
efficiency of irradiation level compensating rod, %	1.7
coefficient of reactivity temp. 1/°C	-4×10^{-5}
coefficient of reactivity power, 1/MW	-1.5×10^{-4}
Parameters of steam generator output	
temperature, °C	540
pressure, atm.	100

BN-600

Construction of the BN-600, 600 MWe, was begun at Beloyarsk. This reactor will represent the next phase of the BN-350 in the development of large reactors for production of electrical energy which are economically competitive. The fundamental data of the project are roughly the same as the BOR-60. In fact, the irradiation rate will be raised by 5 to 10 percent above that of the BN-350; the output temperature of sodium from the reactor will be raised from 500°C to 545°C; thermal yield will be raised from 35 percent to 42 percent; the period between charges will be increased from 2 months for the BN-350 to 4-5 months. It is expected that the plant will be completed within 4 to 5 years (Table V) [sic].

The economic indices for high-speed reactors were referred to the third Geneva Conference (1964) for the two electricity-producing reactors:

	Capital Costs	Cost of Energy
BN-350	240 rubles/kW	0.63 kopeks/kWh
BN-600	150 rubles/kW	0.32 kopeks/kWh

While the BN-350 is being developed at Shevchenko and the BN-600 at Beloyarsk, the USSR is studying plans for high-speed reactors on an industrial scale of 1,000 MWe and above with highly advanced steam characteristics (130-240 atm., 540°C). It should be possible to use standard turbogenerators in plants with this type of reactor and the net thermal yield should not be less than 40 percent.

On the basis of examination of preliminary data it is possible to expect that it will be possible to achieve capital costs comparable with those of nuclear plants equipped with thermal reactors, and lower than the cost component of kWh produced through the fuel cycle.

TABLE VI

BN-600, Project Characteristics

Thermal power, MW	1,430
Electric power, MW	600
Charge, (3.3 kg./MWe) kg Pu	1,980
Coolant	Na
Two-region core	active region (driver) fertile region (blanket)
Two-zone active region	
internal zone	low-enr. UO_2
fuel	(UO_2PuO_2)
external zone	high enr. UO_2
Number of core elements	397
Core dimensions, [as given]	2.048
diameter, ∅	0.7
height, h	
Core Fuel Element	
rods/element	127
element section	triangular
rod diameter, mm	6.9
rod casing thickness, mm	0.4
maximum permissible casing temp. °C	720
Fertile Region Fuel Element	
rods/element	37
material	weak U
triple finned casing	stainless steel
external diameter finned tube, mm	15.25
casing thickness, mm	0.4
Temperature of Na °C	
at reactor input	375-410
output	545-580
Na capacity in primary, t/h	24,000
Time between partial charges, days	147
Irradiation rate, %	10
Primary System	
number of circuits	3
number of pumps	3
number of Na-Na intermediate heat exchangers	6 (3 x 2)
Secondary System	
number of circuits	3
temp. in Na-Na exchangers at full power	
input	340°C
output	550°C
steam generators (Na-H_2O	3

TABLE VI (cont'd)

water temp. °C	
input	240
output	505-540
steam production, t/h	1,836
steam parameters	
pressure, kg/cm^2	140
temperature, °C	505 div. by 540
turbogenerators	3, parallel
nominal power, MWe	200 (each)

Future Electronuclear Plants

Further progress in the development of high-speed reactors will be determined not only by research but will depend to a considerable degree on the construction of prototypes of increasingly greater power and on accumulated operational experience. At the same time the thermal reactors whose technical and economic possibilities are still far from being exhausted also will be studied and developed.

The variant of the U-graphite reactor, which shows great promise, is the 1,000 MWe reactor with saturated steam at a pressure of 70 atm. The yield, which is lower than that of reactors of this variant with superheating, can be compensated for with better exploitation of nuclear fuel, using Zr alloys low in neutron absorption in the core instead of stainless steel. Thus the cost of electrical energy from nuclear plants equipped with these reactors could be lower than that for contemporary coal power plants.

Work is being done on U-graphite reactors with supercritical steam parameters: pressure, 240 atm; temperature, 540°C. The production of energy and the plant yield could be increased with steam having these characteristics. Another advantage which is not less important is the use of standard turbines of 500 MW for supercritical parameters such as those now being installed in the new thermal plants.

The 2,000-4,000 MW electronuclear plants with two to four reactors of this type can be economically competitive with thermal plants even in fuel-producing areas. In order to achieve such high steam characteristics and high levels of irradiation of nuclear fuel it is necessary to set up vast experimental and research projects as well as to resolve the series of complex metallurgical, thermodynamic and nuclear physics problems.

The parameters for reactors of thermal power, of U-graphite and water under pressure are selected in order to obtain the maximum quantity of energy from every fuel charge, that is, the maximum "combustion" of both the initial U isotopes and those formed during fission.

The thermal reactor, in contradistinction to the high-speed reactor, does not permit a rapid accumulation of new nuclear fuel for the construction of new reactors. Nevertheless, the electronuclear plants with thermal reactors could be preferable where they work on a low charge factor which has an essential influence on the accumulation time of plutonium in the high-speed reactors.

Projects for high-speed 1,000 MW reactors are concentrating on study of a shift to uranium and plutonium carbide fuels believed obtainable through self-fertilization around 1.75.

These reactors could be expected to produce steam having the following characteristics: pressure, 240, atm.; temperature, 580°C; the fuel must be able to reach an irradiation level on the order of 100,000-150,000 MWd/t.

BIBLIOGRAPHY

(1) Memoranda presented by the Soviet delegation to the III International Conference on Peaceful Uses of Nuclear Energy, Geneva, 1964.

(2) Nuclear Energy in the USSR, CNEN, Rome, 1966.

(3) Memorandum presented by the Soviet delegation to the conferences and symposia on high-speed reactors at Detroit, 1965; London, 1966; Dubna, 1967.

(4) Atomnaya Energiya, 23, n. 5, November 1967

(5) Sovetskaya Atomnaya nauka i tekhnika, Atomizdat, Moscow, 1967.

(6) Jaderna Energie, CSSR, 14, 41-56, February 1968.

(7) Memoranda presented by the Soviet delegation to a symposium on the present state and the future prospects of development of electronuclear plants with VVER-type reactors, Moscow, April 1968.

(8) Nuclear Industry, 15,31, September 1968

(9) Atomo e Industria, 1 February 1969.

(10) CNEN Bulletin, 5, 661-2, October 1959; 6, 555-63, June 1960; 915-8, October 1960; 7, 678-8 July 1961; 8, 69-71, April 1962; 9, 95-101, June 1963; 10, 59-60, December 1964; 11, 49-51, April 1965; 58 July 1965.

FIGURE APPENDIX

Figure 1 - BR-1 reactor; outside view, upper section. Photo not reproduced.

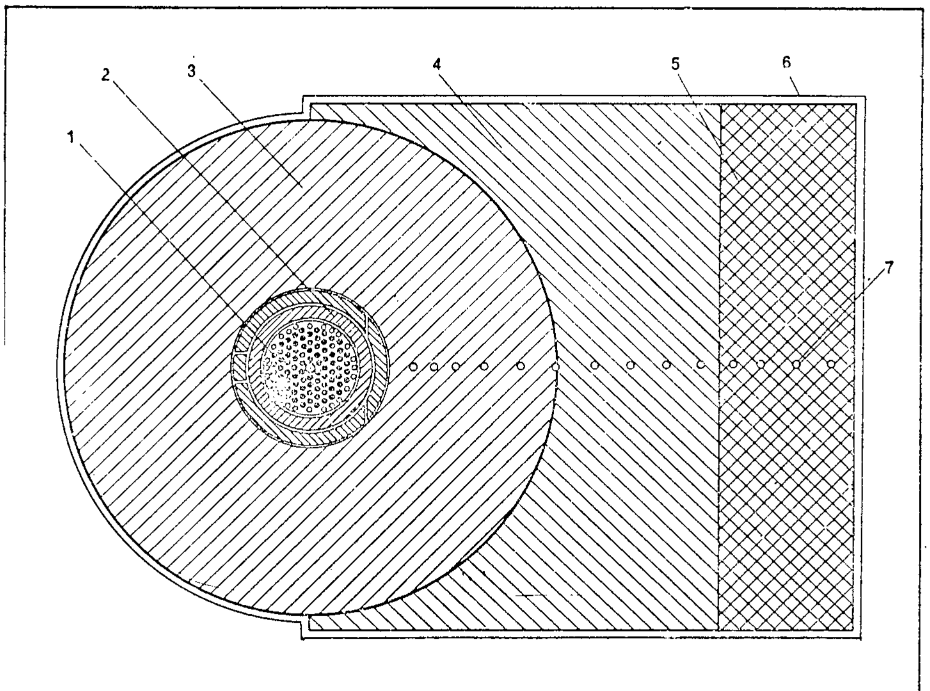


Figure 2
Cross section.
1) core with Pu rods in center and U rods at periphery;
2) compensation cylinder; 3) U blanket; 4-5) U blocks;
6) Cd casing to prevent thermal neutrons reflected from the walls from seeping into blanket; 7) measurement holes.

Figure 3 - Photo not reproduced BR-3 Reactor

Figure 4 - Photo not reproduced BR-5 Reactor

Figure 5 - Photo not reproduced BFS-1

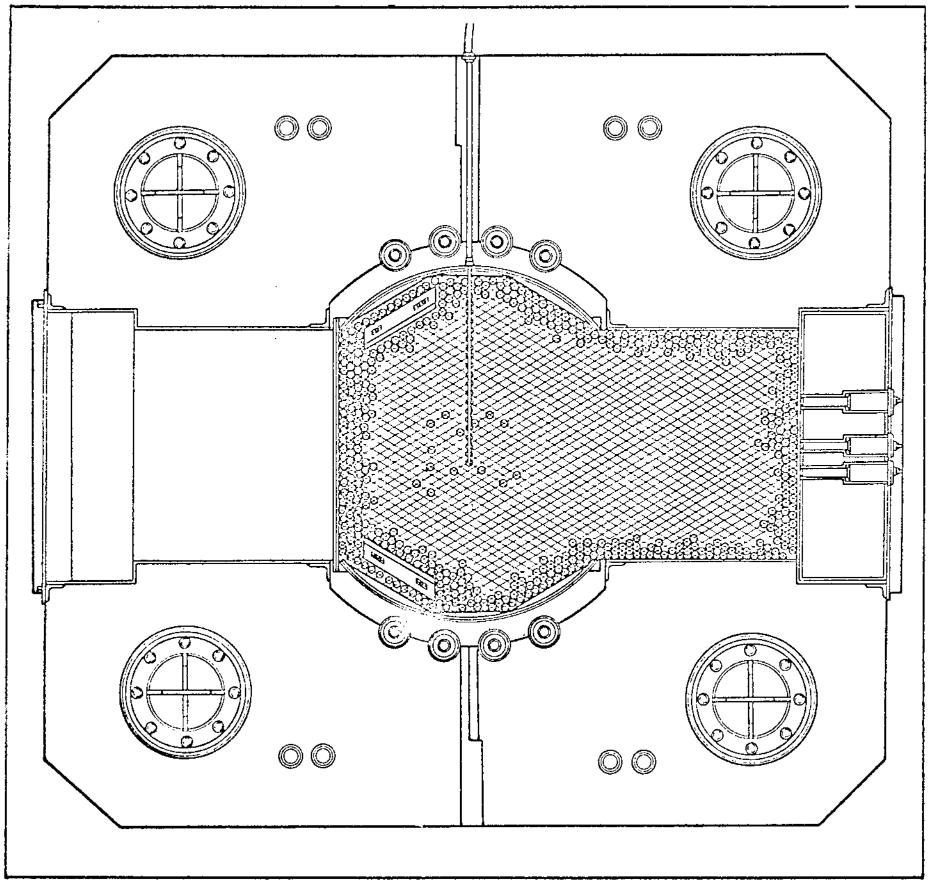


Figure 6 - Section of BFS

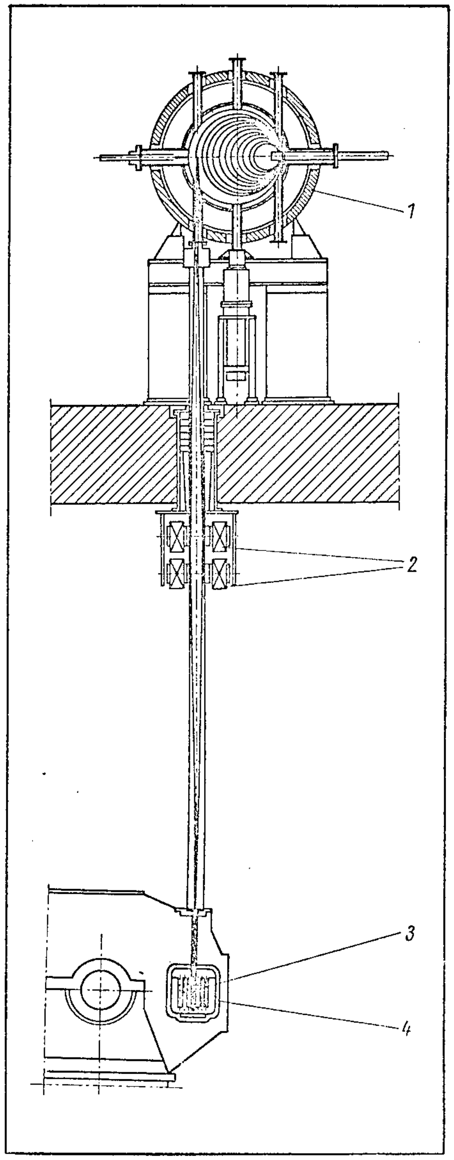


Figure 7 - Installation of microtron in IBR reactor

1) microtron chamber
2) focusing magnets
3) target
4) IBR

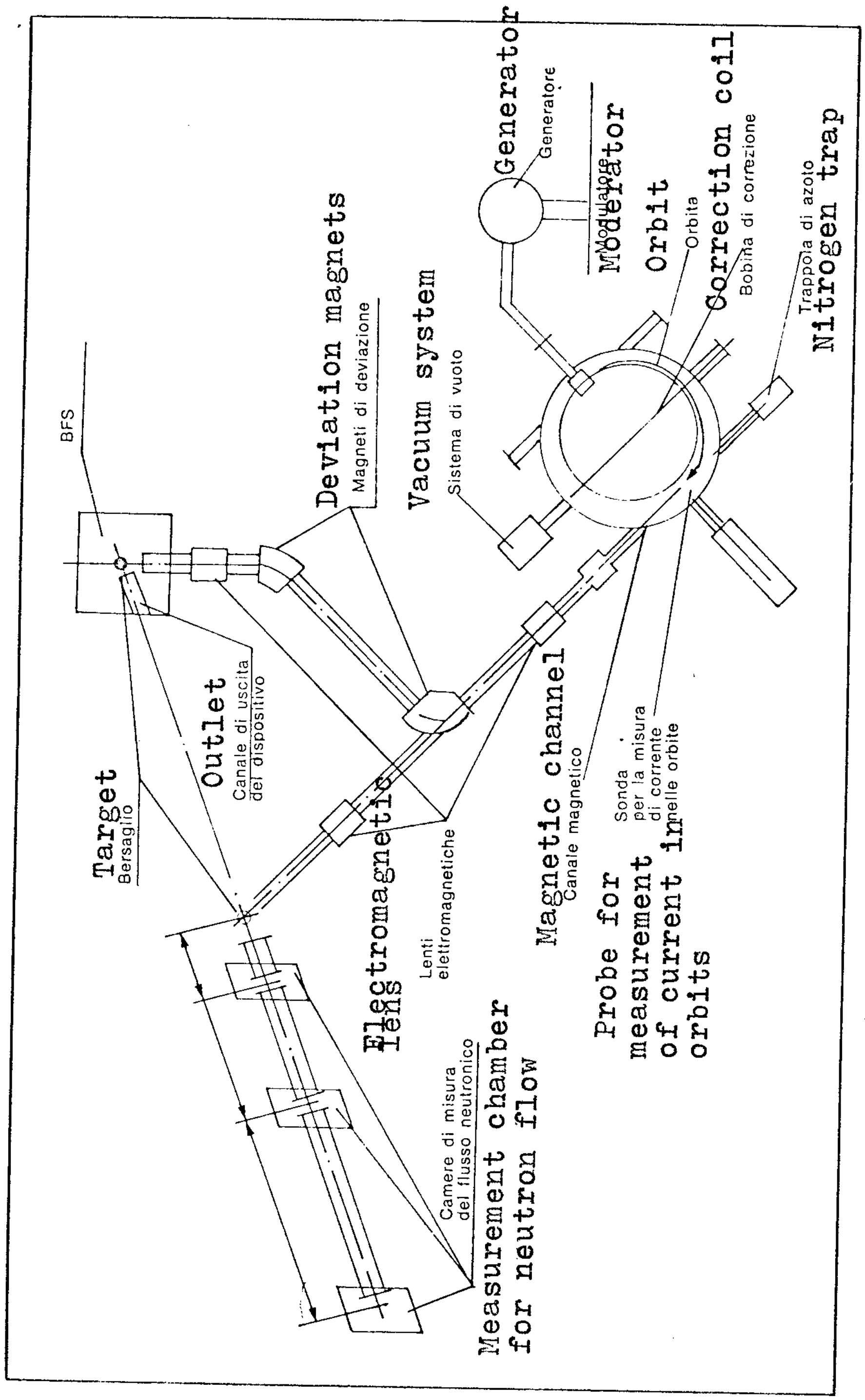


Figure 8 - Plan of the BFS microtron complex

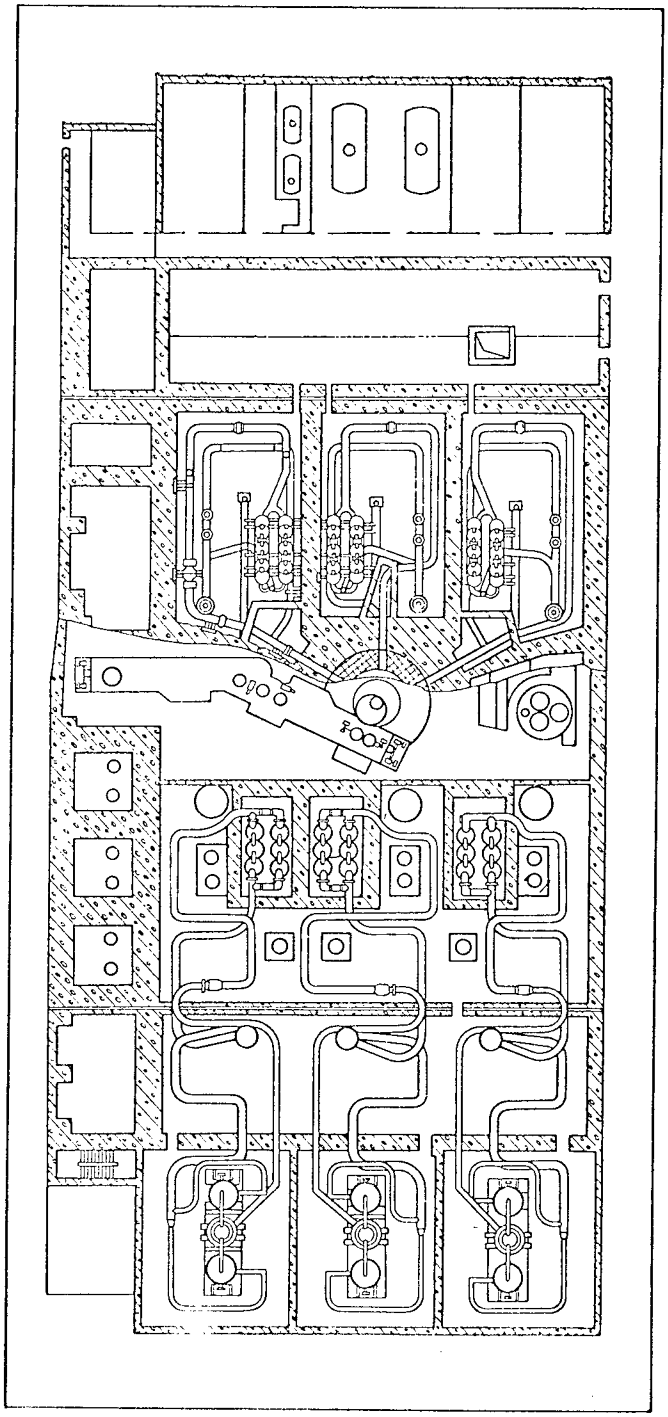


Figure 9 - Section of BN-350 reactor building

Figure 10 - Photo not reproduced Reactor BN-350.
Progress of construction in 1967.

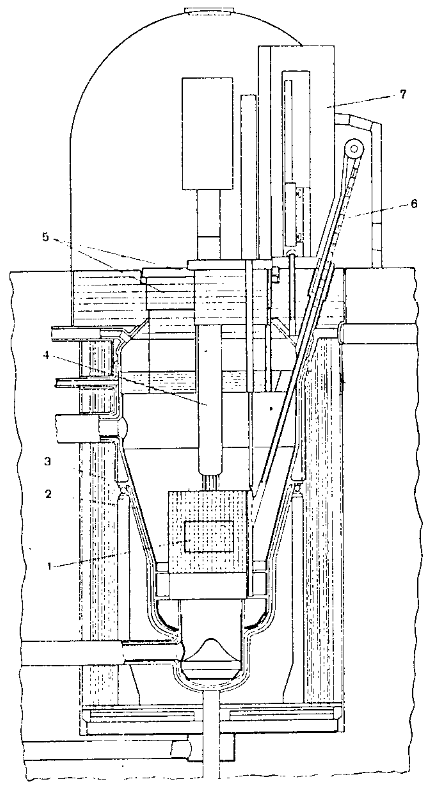


Figure 11 - Section of the plant

1) Core; 2) blanket; 3) reactor vessel; 4) central column for control organs; 5) rotating plugs for charging of fuel elements; 6) mechanism to discharge spent fuel elements; 7) storage chamber.

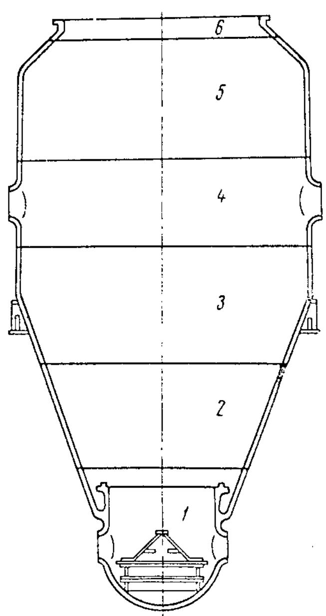


Figure 12 - Section of the vessel

1) Pressure chamber; 2) cone;
3) support flange (two halves);
4) nozzle area, upper (two halves);
5) upper cone (two halves);
6) upper flange.

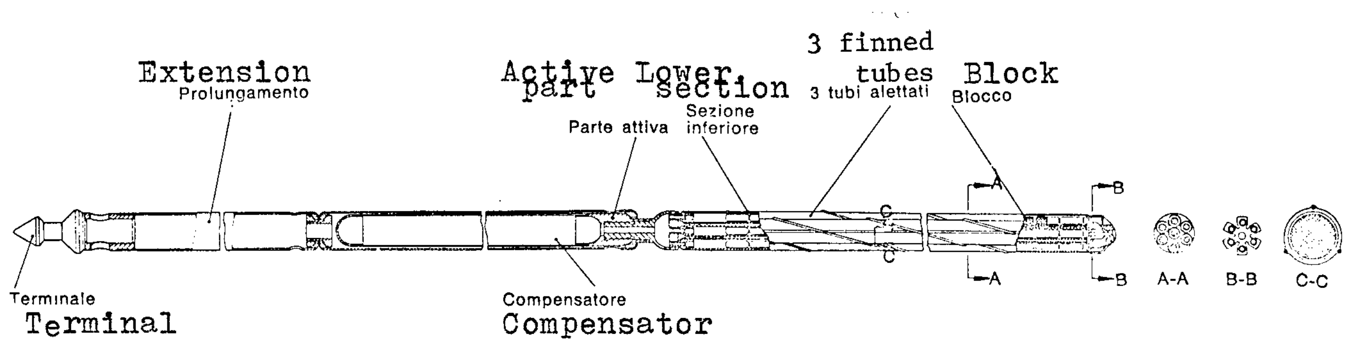


Figure 13 - Automatic regulating rod

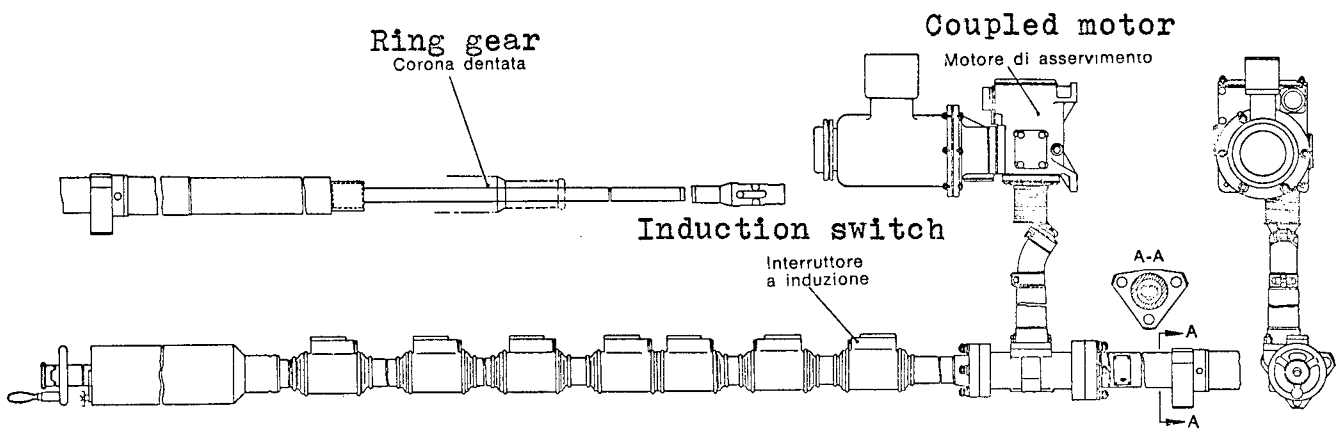


Figure 14 - Activating mechanism of automatic regulating rod

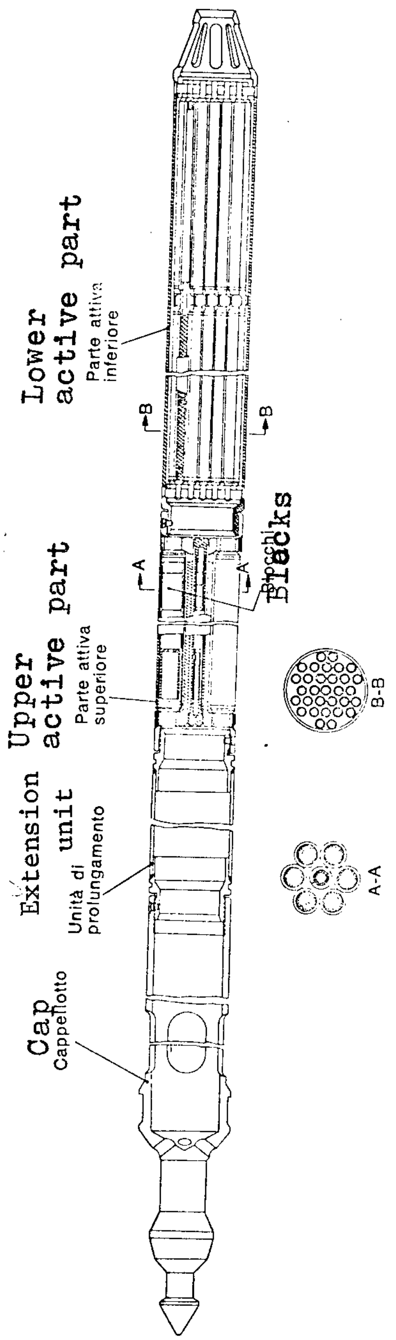


Figure 15 - Temperature compensating rod

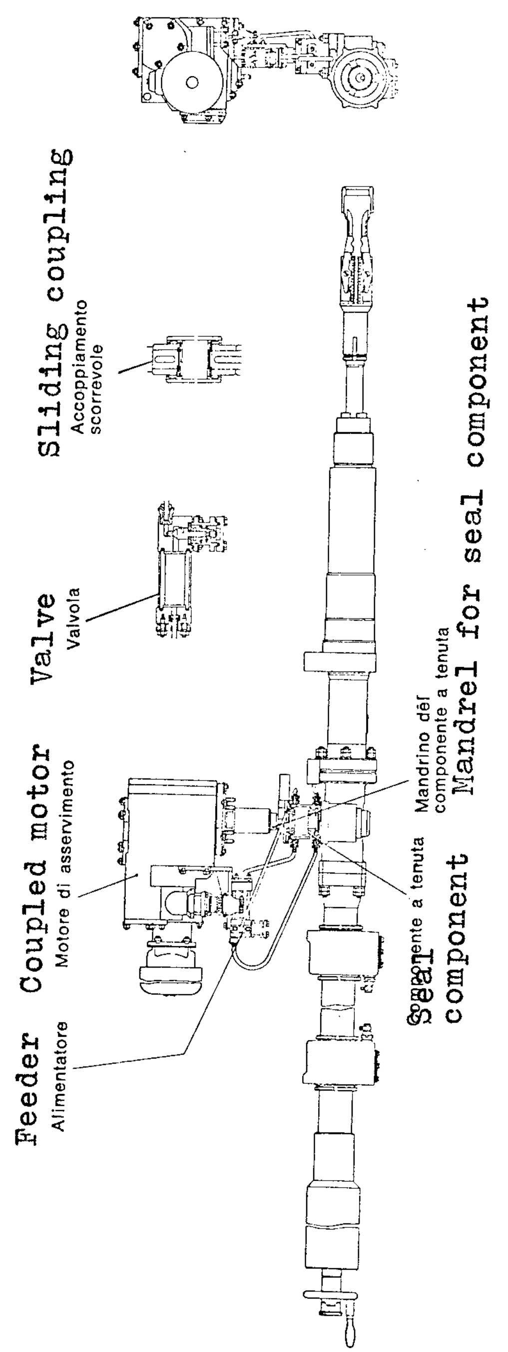


Figure 16 - Auxiliary mechanism for compensating rod

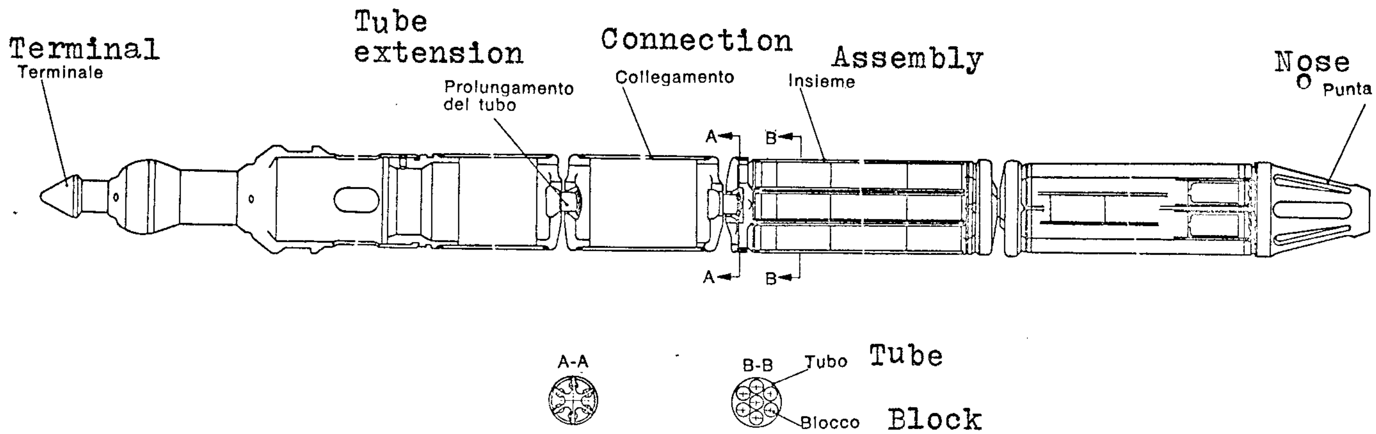


Figure 17 - Safety rod

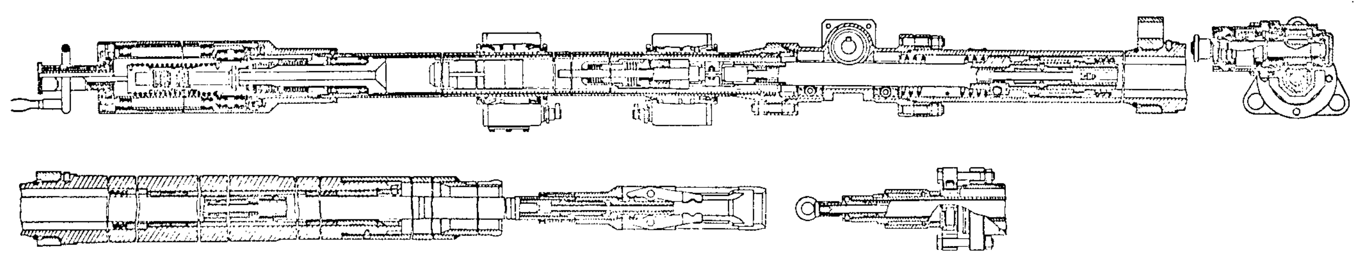


Figure 18 - Gear and pinion system for safety mechanism

Figure 19 - Photo not reproduced Bor-60 Reactor
Building shown in construction phase.

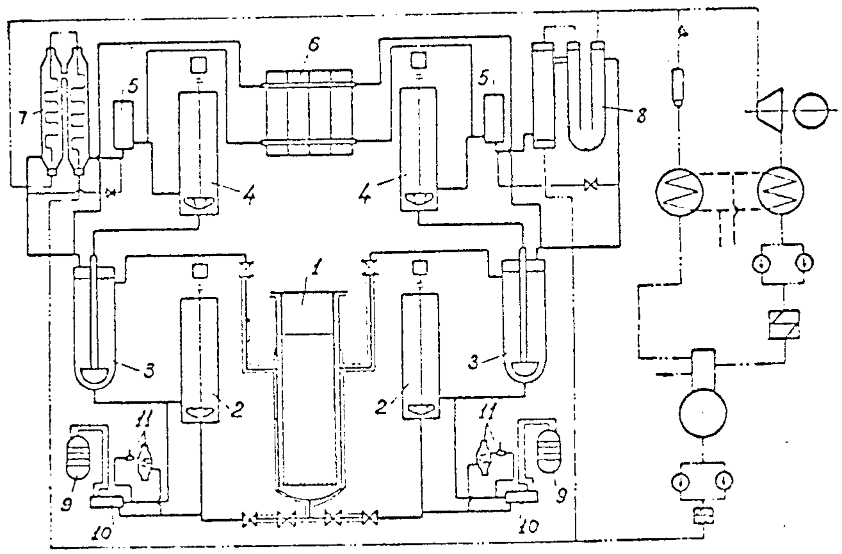


Figure 20 - BOR-60 reactor blueprint

1) Reactor; 2) primary circulation pump; 3) heat exchanger; 4) secondary circulation pump; 5) mixer; 6) air exchanger; 7) single changeover steam generator; 8) natural circulation generator; 9) cold trap; 10 recuperator; 11) oxide detectors.

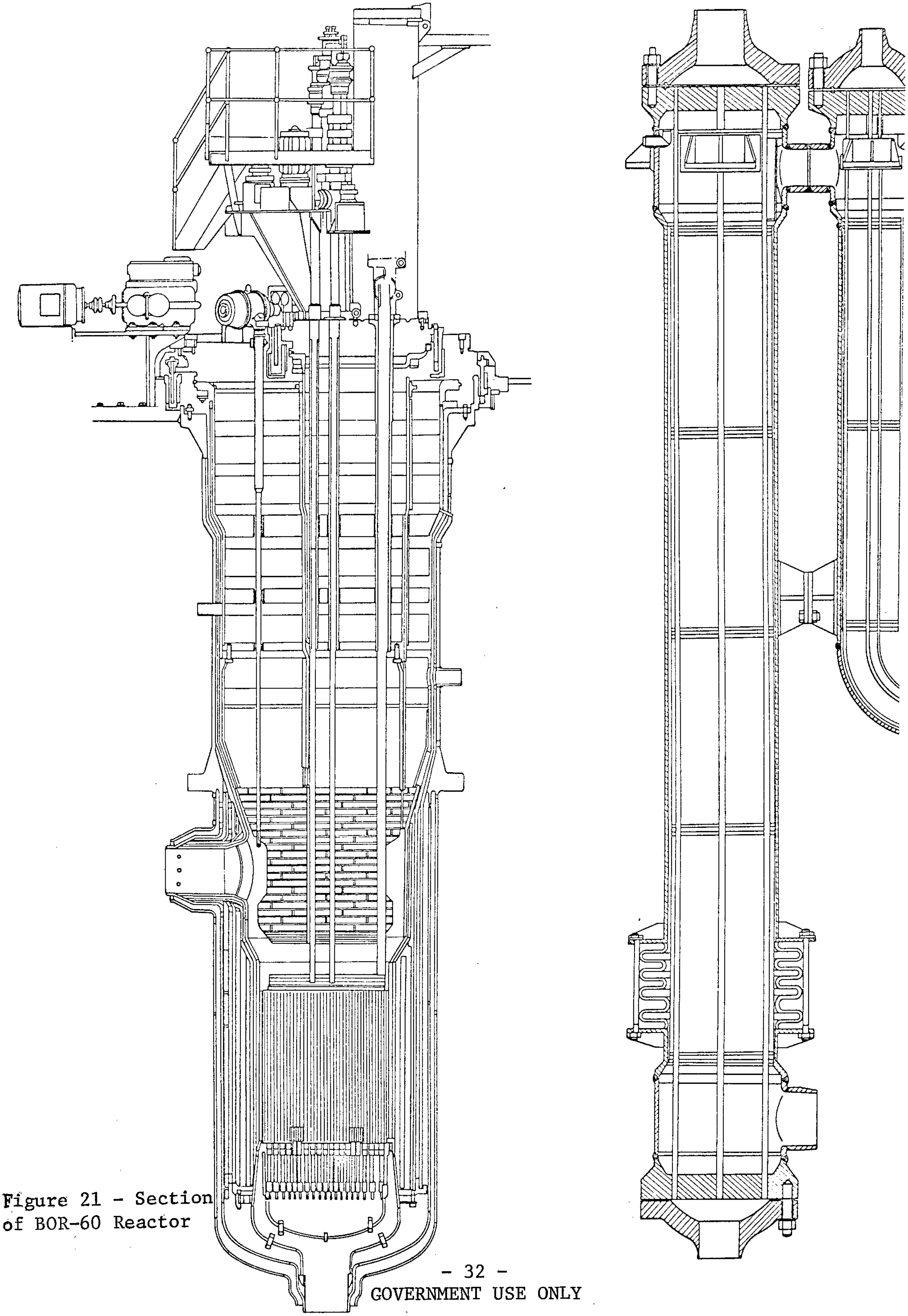


Figure 21 - Section of BOR-60 Reactor

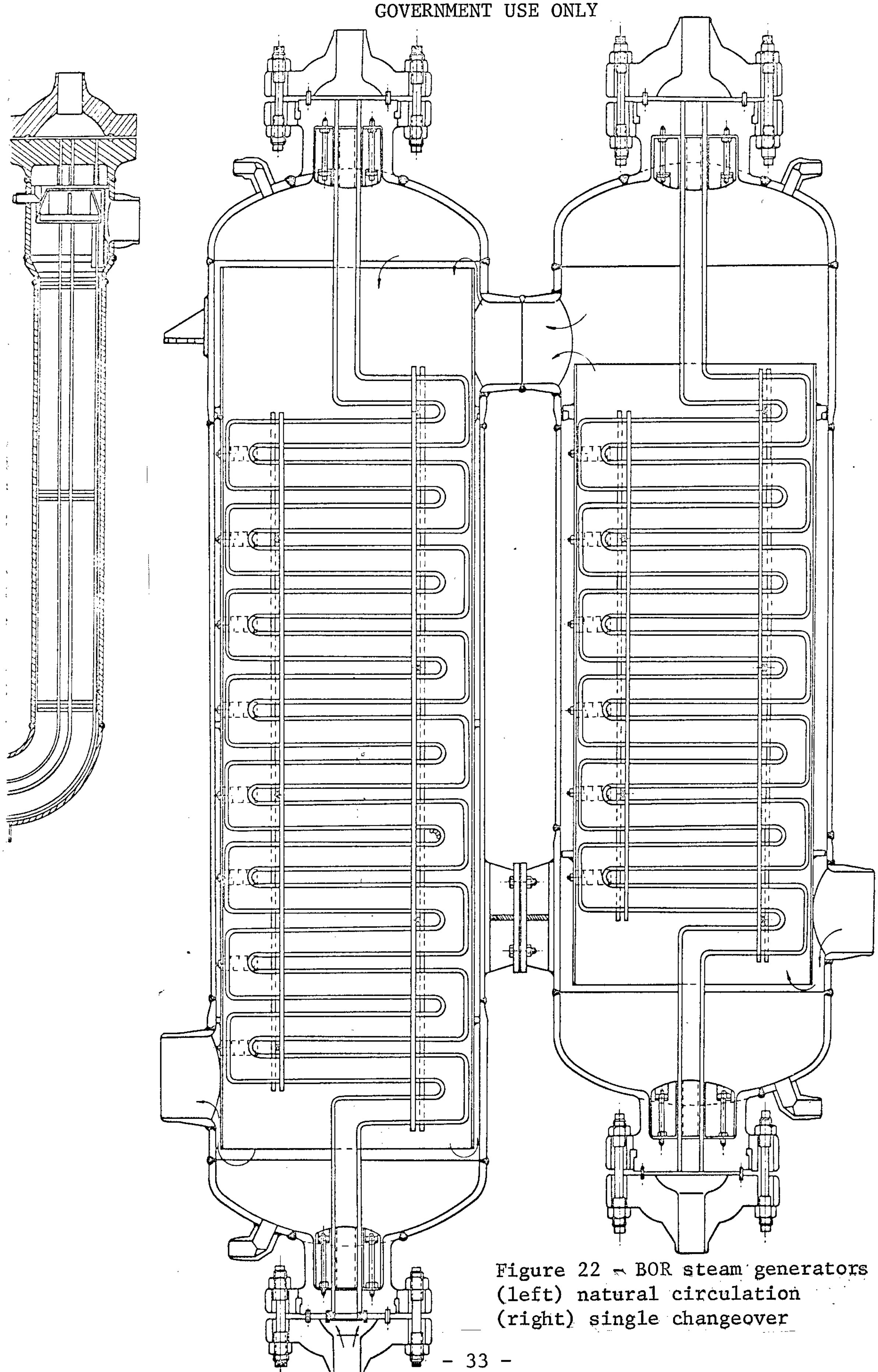
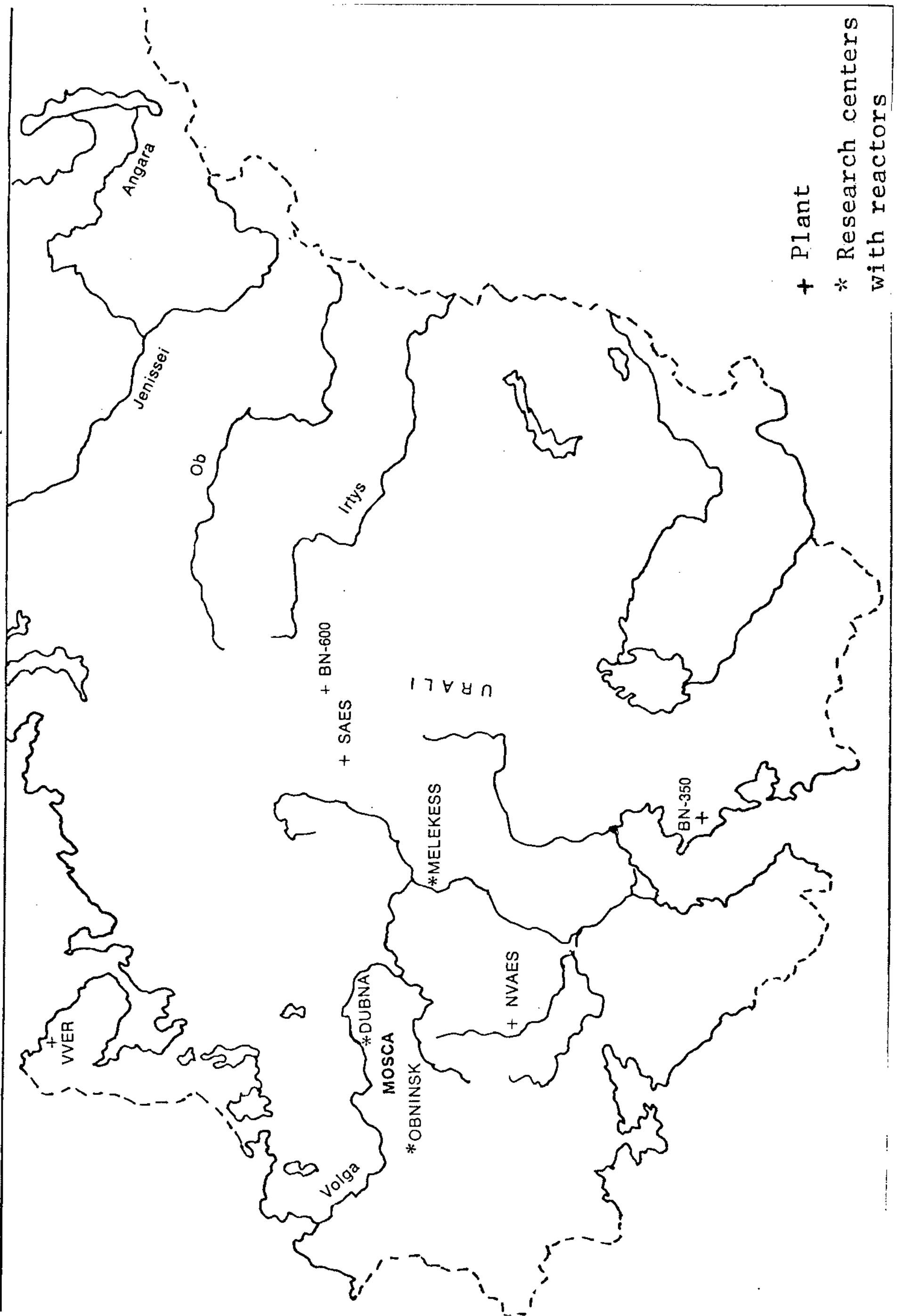


Figure 22 - BOR steam generators
(left) natural circulation
(right) single changeover

Location of major research centers and of electronuclear plants in the USSR



- END -

6034
CSO: 19622-D