Working Document
for the Post-Accident Review Meeting

USSR STATE COMMITTEE ON THE UTILIZATION OF ATOMIC ENERGY

THE ACCIDENT AT THE CHERNOBYL' NUCLEAR POWER PLANT AND ITS CONSEQUENCES

Information compiled for the IAEA Experts' Meeting,

25-29 August 1986, Vienna

PART II. ANNEXES 2,7

D R A F T

August 1986

ANNEX 2

2. DESIGN OF THE REACTOR PLANT

The reactor plant is designed to produce dry saturated steam at a pressure of 70 kgf/cm^2 ($\simeq$ 7 MPa). It consists of the reactor proper with its monitoring, control and protection systems, and the piping and equipment of the multiple forced circulation loop (primary coolant circuit).

2.1. Reactor

The RMBK power reactor is a heterogeneous thermal neutron channel-type (pressure tube) reactor, in which graphite is used as the moderator, while the coolant is light water and a steam-water mixture circulating through vertical channels passing through the core.

The reactor core (1) takes the shape of a vertical cylinder with an equivalent diameter of 11.8 m and height of 7 m (see Fig. 2.1). It is surrounded by lateral and end graphite reflectors 1 and 0.5 m thick, respectively. The core is composed of fuel channels with the fuel assemblies inside them, a graphite moderator, channels with neutron absorber rods (control rods) and the sensors of the monitoring system. Some of the channels in the core are made of a zirconium alloy. The graphite stack consists of blocks assembled into columns with axial cylindrical openings into which the fuel channels are inserted. The fuel channels are located in 1661[*] cells in a square lattice with a 250 mm pitch. The channels of the control and protection system (CPS) number 211 and are arranged in the same way as the fuel channels in the central openings of the graphite stack columns (the arrangement of the channels is shown in Fig. 2.1a).

The grapite stack is located in a leaktight cavity (reactor space) formed by the cylindrical cowling (2) and the plates of the upper (4) and lower (3) metal structures. To prevent oxidation of the graphite and to improve heat transfer from the graphite to the fuel channels the reactor space is filled with a helium-nitrogen mixture with a volumetric composition of 85-90% He and 15-10% N_2. To prevent the possibility of helium leaking from the reactor space the inside cavities of the metal structures and the space around the cowling are filled with nitrogen at a pressure 50-100 mm H_2O (~ 0.5-1.0 kPa) greater than the pressure in the reactor space.

---

* The reactors of the first construction stages of the Leningrad, Kursk and Chernobyl' nuclear power stations contain 1693 fuel assemblies and 179 CPS channels.

The fuel channels are housed in tube ducts welded to the metal structures (5). The upper and lower metal structures and the water-filled annular tank (6) around the cowling serve as biological shielding for the rooms surrounding the reactor. The coolant (water) is fed in from below to each fuel channel through separate pipes. As it rises and flushes the fuel elements, the water heats up and partially evaporates; the steam-water mixture is led off from the top of the channels likewise through separate piping.

Nuclear fuel is reloaded without a reduction in reactor power by means of the refuelling machine.

Under steady-state operating conditions the intensity of the refuelling when the reactor is operating at nominal power is 1-2 assemblies per day.

The reactor is equipped with a control and protection system (CPS) and with monitoring systems which transmit information on the state of the core and the operation of various components, as well as sending the necessary signals to the CPS and the emergency signalling system.

Main characteristics of the reactor

Coolant flow through the reactor, t/h	37.6×10^3
Steam pressure in the separator, kgf/cm^2	70
Pressure in the group pressure headers, kgf/cm^2	82.7
Mean steam content at the reactor outlet, %	14.5
Coolant temperature, ^oC	
Inlet temperature	270
Outlet temperature	284
Maximum channel power, with allowance for 10% power distortion, kW	3000
Coolant flow rate in maximum power channel, t/h	28
Maximum steam content at channel outlet, %	20.1
Minimum critical power margin	1.25
Core height, mm	7000

Core diameter, mm	11 800
Fuel lattice pitch mm	250 x 250
Number of fuel channels	1661

2.1.1. Design of the fuel assembly and fuel element

The fuel assembly of the RMBK 1000 reactor consists of the following main parts (Fig. 2.2):

- Two fuel sub-assemblies (1);
- Supporting rod (2);
- Guiding tail and nose pieces (3 and 4);
- Nuts (5).

The fuel assembly is 1015 mm long.

Each sub-assembly consists of 18 fuel elements, a casing and 18 pressure rings.

The fuel element (2.2a) consists of the cladding (6), fuel column (7), holding spring (8), plug (9) and end piece (10).

The material of the cladding and end pieces is a zirconium alloy with 1% niobium (alloy 110). The spring is made of Ts2M zirconium alloy.
The outer diameter of the cladding is 13.6 mm and the minimal thickness 0.825 mm.

As the fuel use is made of sintered uranium dioxide pellets. The pellets are 11.5 mm in diameter and 15 mm high; to reduce the heat expansion of the fuel column the pellets are concave at the end. The mean mass of fuel in a fuel element is 3600 g, the minimum density of the pellets is 10.4 g/cm^3, and the diametric gap between the fuel and the cladding is 0.18-0.38 mm.

The fuel elements are made leaktight by resistance butt welding of the nose piece on to one end of the cladding tube and of plug on to the other.

The initial medium under the cladding is helium at a pressure of ~ 1 kg/cm^2 (0.1 MPa). The fuel column in the element is held in place by the spring with a constrictive force of about 15 kg.

The casing consists of a central tube 15 mm in diameter with a wall thickness of 1.25 mm, an annular grid (11) and 10 spacer grids (12). The central tube and end grid are made of a zirconium alloy with 2.5% niobium (alloy-125), while the spacer grids are made of stainless steel.

By means of two flairings the central tube is joined to the end grid in such a way that there is no possibility of an axial air gap at the join, and twisting of the grid with respect to the tube is also prevented. To keep the sub-assemblies in position and prevent them twisting with respect to one another, the casing tubes are fitted with special grooves. The spacer grids are fixed to the central tube at intervals of 360 mm. Each grid is secured by insertion of the projecting end of the central sleeve into two grooves on the tube in such a way that it can move along the tube if there is a small azimuthal air gap.

The spacer grid is assembled from individual shaped cells (12 cells in the peripheral row and 6 in the inside row), the central sleeve and an encircling rim. The parts of the grid are joined together by resistance spot welding. The openings for the fuel elements in the grid are 13.3 mm in diameter. On the rim of the grid there are projections making it easier to load the assembly into the channel. The diameter across the rim projections is ~ 78.8 mm.

The cells are made of tubing with a wall thickness of 0.35 mm; the central sleeve is made of tubing with a wall thickness of 0.5 mm, and the rim from tubing with a 0.3 mm in wall thickness.

The fuel elements are secured to the end grid by means of pressure rings made of stainless steel. The securing system cannot be taken apart, since the pressure rings deform when the fuel elements are secured.

The design of both fuel sub-assemblies is identical.

When the fuel-assembly is being put together, the nose piece, the two sub-assemblies, and the tail piece, which is fixed with a nut, are mounted on the central rod. The nut is prevented from unscrewing by means of a pin.

Two types of fuel-assembly are inserted in the reactor: a working assembly and an assembly for use as a monitor for the power density (over the core radius) which is different from the working assembly in terms of the design of the tie rod. The latter is hollow and consists of a tube with a 12 mm outside diameter and wall thickness of 2.75 mm, and a plug, both made of zirconium alloy (alloy-125), a steel-zirconium transition piece and an extension tube made of stainless steel.

2.1.2. Fuel channel (Fig. 2.3)

The fuel channel is intended to house the fuel assemblies with the nuclear fuel and to control the flow of coolant. The casing of the channel is a welded structure consisting of a middle and end part. The middle (2) is made of zirconium alloy (Zr + 2.5% Nb) and composed of a tube 88 mm in outside diameter with a wall thickness of 4 mm, an upper (1) and lower (5) end piece made of corrosion-resistant tubing (steel 08 Cr18Ni10Ti). The middle part is joined to the ends by means of special steel-zirconium transition pieces (3, 4).

The transition joints - corrosion-resistant steel-zirconium alloy - are manufactured by means of vacuum diffusion welding (Fig. 2.3(a)).

The transition joints are designed to produce programmed configurations and stresses in the area of the joint that guarantee strength and reliability under operating conditions. The inside part of the transition is made of zirconium alloy, while the outside part around it is made of corrosion-resistant steel. During the diffusion welding a thin layer of mutually diffusing products forms on the contact surface of the parts being joined together. The quality of the diffusion welding is checked by ultrasonic flaw detection and metallographic devices. As part of the fuel channel the transition pieces are also tested for helium leaktightness and hydraulic pressure.

The channel tubes are joined to the zirconium parts of the transition pieces by electron-beam welding. To improve the corrosion properties of the welded joints they undergo additional strengthening and heat treatment.

The steel parts of the transition pieces are welded to the top and bottom parts of the fuel channel by argon welding. A metallic coating of aluminium is applied to the outer surfaces of the steel parts in the channel to protect them against corrosion.

To improve heat flow from the graphite block to the channel, slotted graphite rings 20 mm high are fitted onto the middle of it and positioned very closely together along the channel so that every other ring is directly in contact, by means of its lateral surface, either with the pipe (7) or with the inside surface of the block (6), as well as being in contact at their ends.

The minimum gaps between the channel and ring - 1.3 mm - and ring and block - 1.5 mm - are designed to prevent wedging of the channel in the stack through radiation-induced thermal shrinkage when the reactor is in operation.

The channel body is housed in the reactor in tube ducts (3, 4) welded to the top and bottom metal structures (Fig. 2.4). It is attached immovably to the upper duct by means of a thrust collar and filament seam made by argon arc welding (1). The lower part of the body is welded to the metal structure duct, being joined to it through the bellows compensating unit (2); this makes it possible to compensate for any difference in thermal expansion of the channels and metal structures, as well as ensuring reliable leaktightness of the reactor space. The channel body is designed to operate safely for 30 years, but whenever necessary a defective channel body can be taken out of the reactor and replaced by a new one with the reactor shutdown.

The fuel assembly with its fuel elements (5) is mounted inside the channel on a suspension (6), which keeps it in the core and enables the refuelling machine to replace a spent fuel assembly without stopping the reactor.

The suspension is fitted with a closing plug (7), which is mounted in the housing of the upper duct. This plug hermetically seals the inside of the duct by means of a ball-type shutter fitted with a sealing washer. The unsealing operations during refuelling are carried out by the refuelling machine using remote control.

2.1.3. Control channels (Fig. 2.4)

These channels are intended to contain the control system rods, vertical power density monitors and ionization chambers. The middle of the channel (3) is made of a zirconium alloy (Zr + 2.5% Nb) and constitutes a tube 88 mm in diameter and with a wall thickness of 3 mm. The upper (1) and lower (4) end parts are made of corrosion-resistant piping (steel 08 Cr18Ni10Ti). The middle part is joined to the end tubes by means of steel-zirconium transition pieces similar to those used for the fuel channels. The channels are secured immovably to the upper tube duct by means of a thrust collar and a filament seam, and to the lower duct via the bellows compensating unit. The CPS channels in the upper part have heads (5) designed for the attachment of actuators and for supplying cooling water to the channel. Graphite sleeves (6) are placed over the channel and provide the requisite temperature conditions for the graphite column. At the bottom of the channel is a throttle device (2), which ensures that the channel is completely filled with water.

Placing of the control channels in the graphite columns independently of the fuel channels guarantees their preservation and, consequently, the efficiency of the control elements contained in them in the event of possible accidents due to rupture of the fuel channels.

2.1.4. Metal structures of the reactor (Fig. 2.1)

The lateral biological shielding tank (6) takes the form of a cylindrical reservoir with an annular section 19 m in outside diameter and 16.6 m in inside diameter: it is made of low-alloy steel sheeting of the pearlite class (10 CrSiNiCu) 30 mm thick. Inside the tank is divided into 16 vertical leaktight compartments filled with water, the heat from which is removed by the cooling system. The top metal structure (4) is a cylinder 17 m in diameter and 3 m high. The upper and lower plates of the cylinder are made of steel (10 CrNilMo) 40 mm thick welded to the lateral shell by means of leaktight welds, and welded to each other by means of vertical strengthening fins. The holes in the top and bottom plates are for the welded-in tube ducts (5) holding the fuel and control channels. The space between the tubes is filled with serpentinite (a mineral containing bound water of cystallization). The metal structures are mounted on 16 roller-type supports attached to the projection of the annular part of the lateral biological shielding and bear the weight of the loaded channels, the floor of the central hall and the piping of the upper steam-water and water communication lines.

The bottom metal structure (3), which is 14.5 m in diameter and 2 m high, is similar in design to the top structure. It is loaded by the graphite stack mounted on top of it together with the supporting units and lower water communications. The number and arrangement of the lower fuel and control channel ducts welded to the top and bottom of the lower metal structure are the same as in the upper structure. The cavity inside it is filled with serpentinite. The supporting metal structure on which the lower metal structure is mounted is composed of plates with reinforcing fins 5.3 m high which intersect at the centre of the reactor and are perpendicular to each other (7).

The cylindrical shroud (2) is a welded shell with an outside diameter of 14.52 m and height of 9.75 m made of steel sheeting (10 CrNilMo) 16 mm thick. To compensate for longitudinal heat expansion the shroud is fitted with a lens-type compensator. The shroud, together with the top and bottom metal structures, forms the closed reactor space.

The metal structure of the top covering (8) has an opening for the insertion of the fuel and other special channels. It is covered over by a removable floor (9) consisting of individual slabs. The floor acts as biological shielding for the central hall and, furthermore, serves as heat insulation for it. The floor consists of upper and lower slabs and blocks resting on the fuel and reflector channel ducts. The slabs and blocks are metal structures filled with iron-barium-serpentinite cement stone.

Air is extracted from the central hall through gaps in the floor and then passes to the ventilation shafts. The air cools the floor and prevents the possibility of radioactivity releases entering the hall from the room containing the steam-water communications.

2.1.5. Graphite stack (Fig. 2.1)

The graphite stack (1) is assembled on the lower metal structure inside the reactors space. It takes the form of a vertical cylinder made up of 2488 columns of graphite blocks with a density of 1.65 g/cm^3. The blocks are shaped like parallelepipeds with a 250 x 250 mm section and height of 600 mm. The mass of the stack is 1700 t. There are openings 114 mm in diameter along the axis of the blocks, forming ducts in the columns to hold the fuel channels and CPS channels. Each graphite column is mounted on a steel base plate (10), which in turn rests on a cup welded to the top plate of the lower metal structure. The graphite stack is made secure against movement in a radial direction by means of rods positioned in the peripheral columns of the lateral reflector. At the bottom the rod is welded to the supporting cup, while at the top it is joined immovably to the tube duct welded to the bottom plate of the upper metal structure. The hollow rod, made of corrosion resistant steel (08 Cr18Ni10Ti) piping, holds the channel for cooling the reflector blocks. The heat released in the stack is removed basically to the fuel channels and partially to the CPS channels. The presence of firm-contact rings on the channels and the helium-nitrogen mixture with which the channel-ring and ring-block gaps are filled keep the stack at a temperature not exceeding 700°C.

In the case of the graphite blocks the highest temperature zones are to be found on the block edges, while the lowest temperatures are found on the inner surface of the vertical openings into which the fuel and other channels are placed. The highest temperature is found in by the blocks located in the middle of the centre part of the core.

The greatest temperature differential - between the edge and inner surface of the opening - is to be found in the block with the fuel channel and amounts to ~ 150°C.

2.1.6. Biological shielding

The biological shielding of the fourth unit reactor of the Chernobyl' nuclear power station has been designed in accordance with the requirements in force in the USSR - "Radiation Safety Standards NRB-76" and "Health Regulations for Designing and Operating Nuclear Power Plants SP-AEhS-79".

The dose rate for external exposure in the central hall and serviced buildings adjoining the reactor vault do not exceed 2.8 x 10^{-2}mSv/h (2.8 mrem/h). During refuelling, at the time when the spent fuel assembly is removed and passed through the floor of the central hall, the gamma dose rate close to the refuelling machine briefly rises to 0.72 mSv/h. In the room containing the water communication lines below the reactor the shielding ensures that the neutron flux density drops to values at which there will be no appreciable activation of the piping and structures. It is only permitted to enter that room when the reactor has been shut down.

Shielding against radiation from the coolant in the piping and equipment of the main circuit makes it possible to carry out repair and adjustment operations while the reactor is in operation; for example, channel-by-channel adjustment of the coolant flow by means of multipurpose valves fitted in the group headers, repairs to the electric motors of the main circulation pumps, and so on. Radiation heat release is reduced to values at which the temperature of the supporting metal structures (top, bottom and tank) and the reactor shroud is not more than 300°C, which makes it possible to use low-alloy steel.

The fast neutron fluence with an energy of more than 0.1 MeV reaching the reactor shroud and sheeting of the metal structures close to the core has not exceeded 10^{20}n/cm^2 in 30 years of operation.

The shielding designed takes the following form (Fig. 2.1).

Mounted on each graphite column, between the end reflectors 500 mm thick and the upper and lower metal structures, are steel blocks (10) (the lower ones are 200 mm thick and the upper ones 250 mm) designed to reduce the fast neutron fluence onto the metal structures supporting the load, as well as to reduce the energy released in them.

The space between the tubes in the top and bottom metal structures is filled with serpentinite (3, 4), which makes it possible to reduce the length of the fuel channels and the overall dimensions of the building.

Above the steam-water communication lines is a protective covering (floor of the reactor hall), the central part of which - the slab flooring (9) - is made up of a set of blocks resting on the tops of the channel ducts. These blocks are made of iron-barium-serpentinite cement stone. The overall thickness of the covering is 890 mm. The upper flooring protects the central hall against radiation from the reactor and from the piping containing the radioactive coolant, and together with the refuelling machine container reduces the intensity of the radiation when unloading spent fuel assemblies. The peripheral part of the upper covering (8) constitutes metal cases 700 mm high filled with a mixture of pig iron shot (86% mass) and serpentinite.

In a radial direction the lateral reflector consists of four graphite blocks, with a mean thickness of 880 mm. The annular water tank (6) lying behind the reactor shroud reduces the radiation fluxes to the walls of the reactor vault (11), which are made of building concrete (density 2.2 t/m^3 and wall thickness 2000 mm). The space between the tank and the walls is filled with ordinary sand (12).

The thicknesses and composition of the materials of which the RBMK reactor shielding is made in the main directions away from the core are shown in Table 2.1.

Table 2.1. Thickness of shielding materials (in a direction away from the core centre) (mm).

Material	Direction		
	Upward	downward	radial
Graphite (reflector) (mm)	500	500	880
Steel (protective plates and sheeting of the metal structure) (mm)	290	240	45
Serpentinite filling (1.7 t/m^3)(mm)	2800	1800	-
Water (annular tank)(mm)	-	-	1140
Steel (metal structures)(mm)	40	40	30
Sand (1.3 t/m^3)(mm)	-	-	1130
Heavy concrete (4.0 t/m^3)(mm)	890	-	-
Building concrete (2.2 t/m^3)(mm)	-	-	2000

A reduction in the intensity of radiation streaming through the gas-filled channels (for temperature sensors, neutron flux detectors and ionization chambers) or channels with less effective shielding (steam-water mixture in fuel channels) is attained by inserting shielding plugs made of steel or graphite (Fig. 2.5). The annular gaps between the channels and the guide tubes are closed by means of shielding sleeves (Fig. 2.6).

The gas piping which passes through the shielding structures is made with bends (No. 13 in Fig. 2.1).

To prevent neutron streaming and gamma radiation, as well as to reduce the activation of the structures in the area below the reactor, the displacers in the CPS channels are filled with graphite (Figs 2.17 and 2.30).

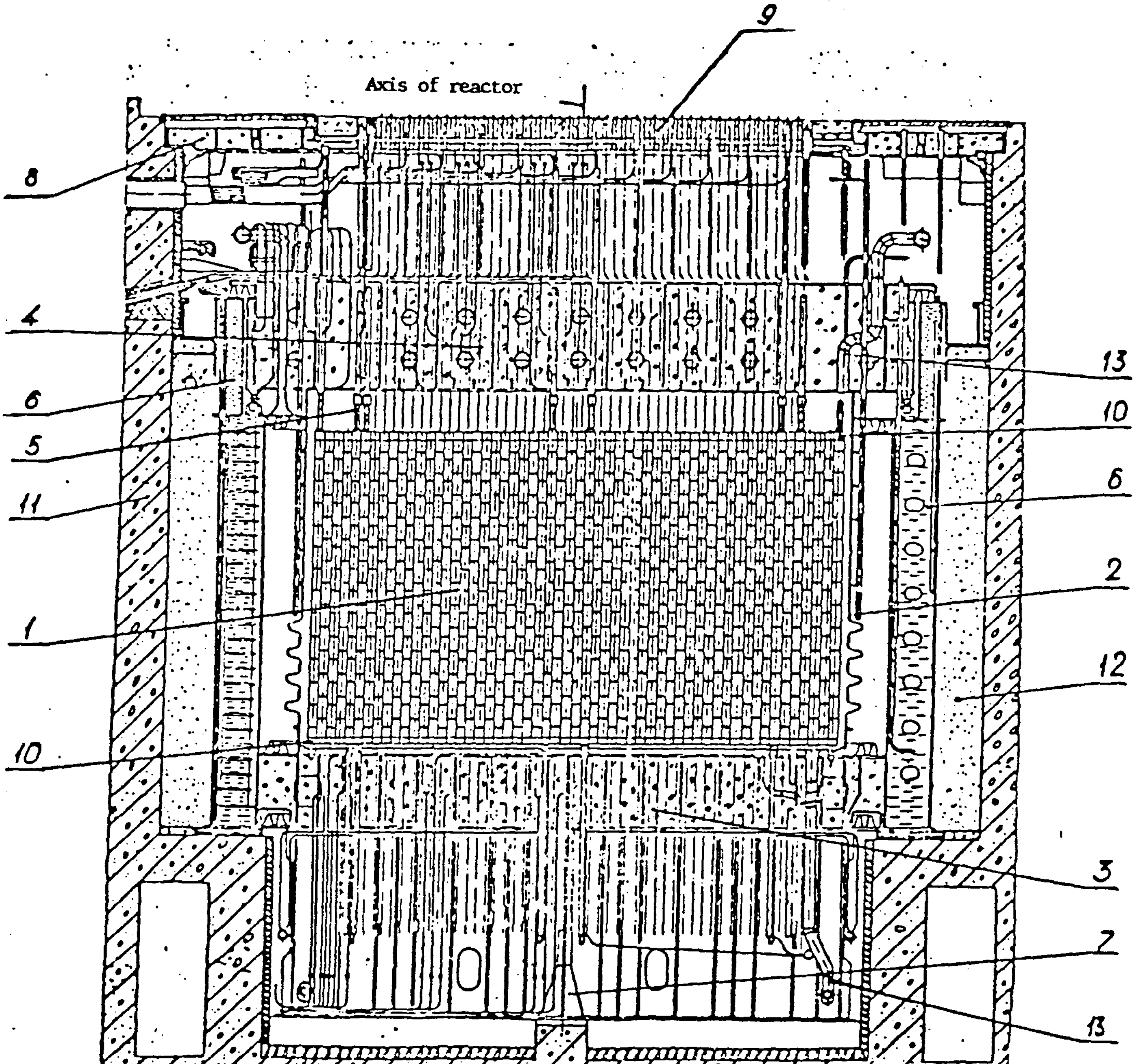


Fig. 2.1

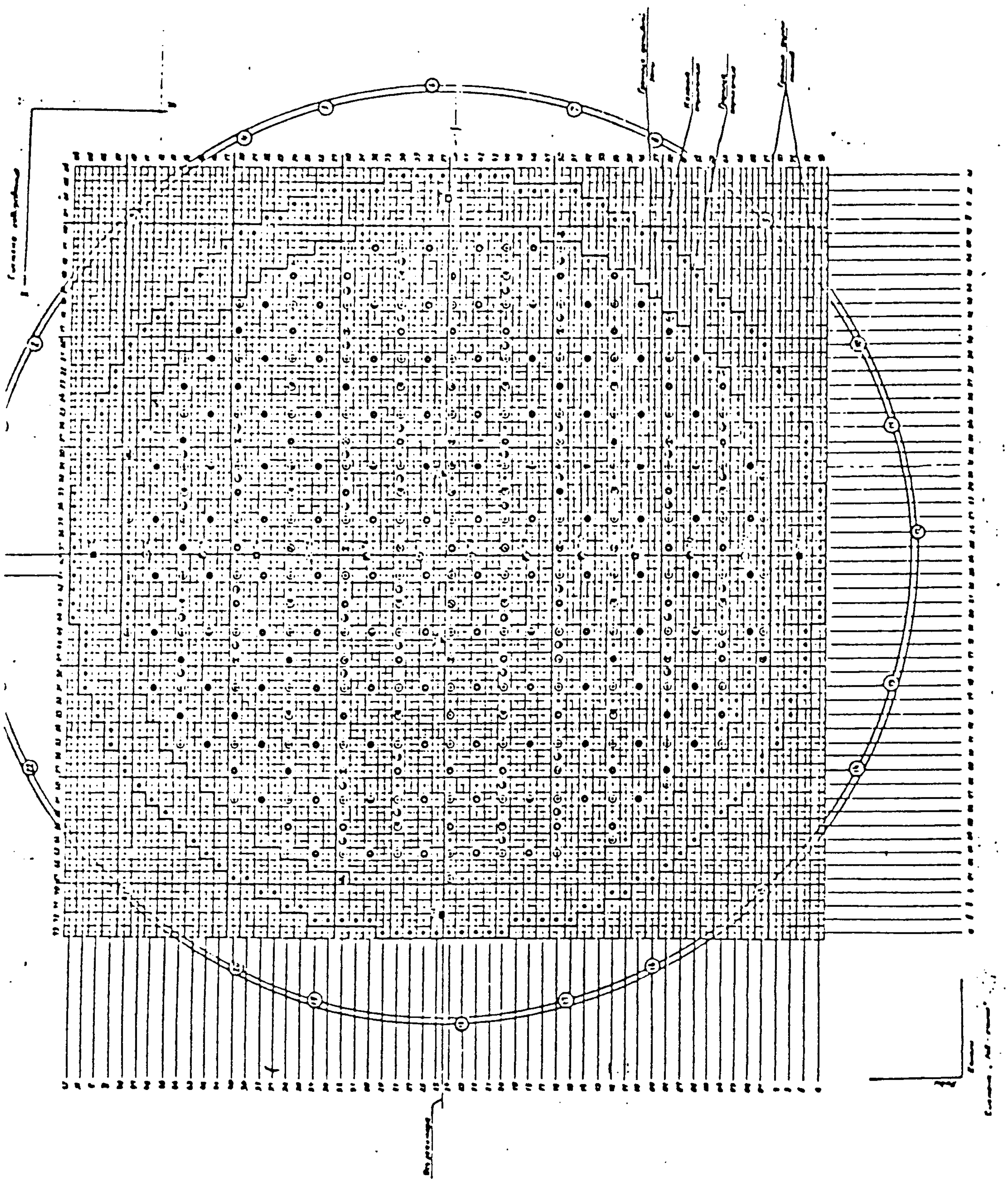


Fig. 2.Ia

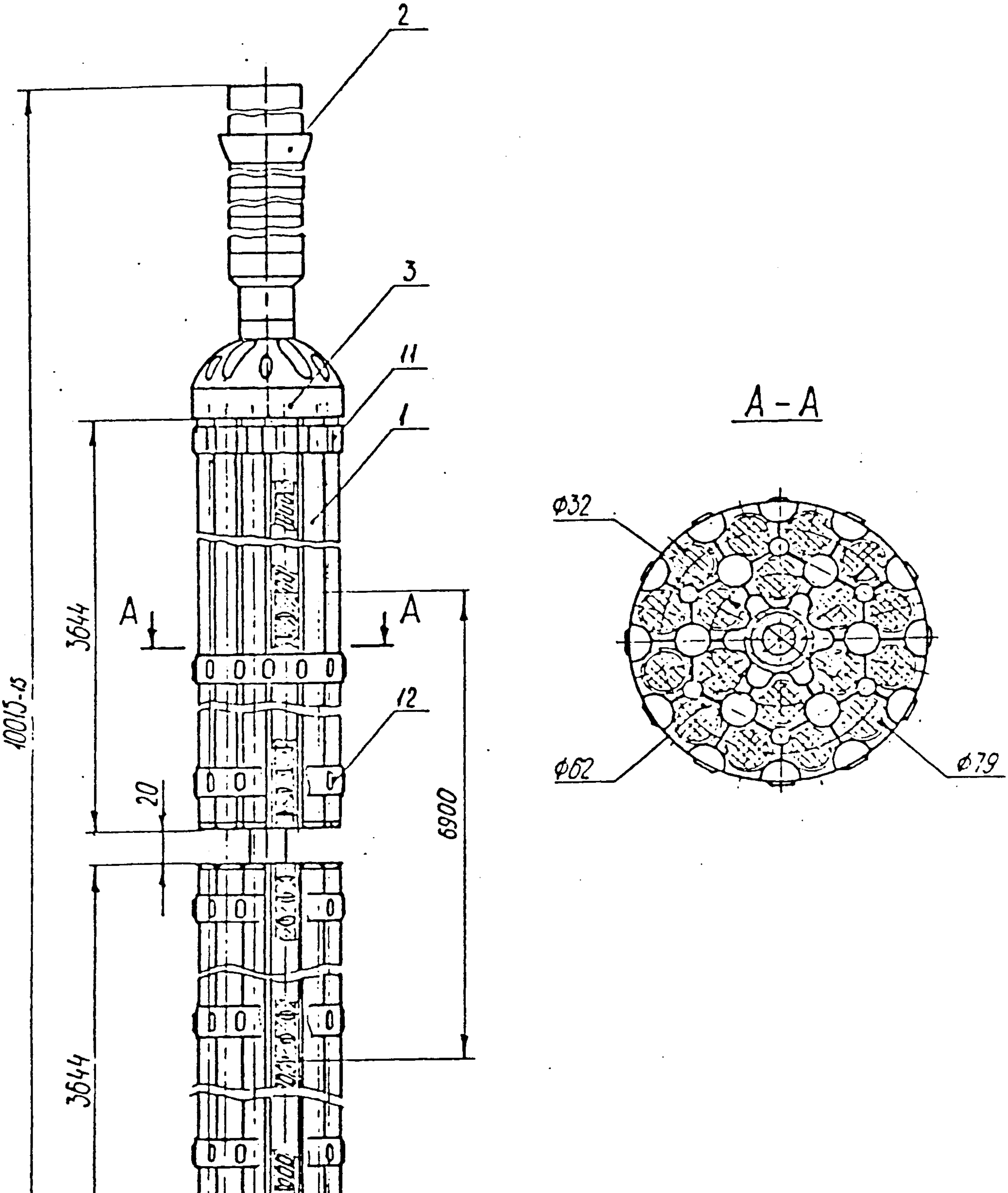


Fig. 2.2

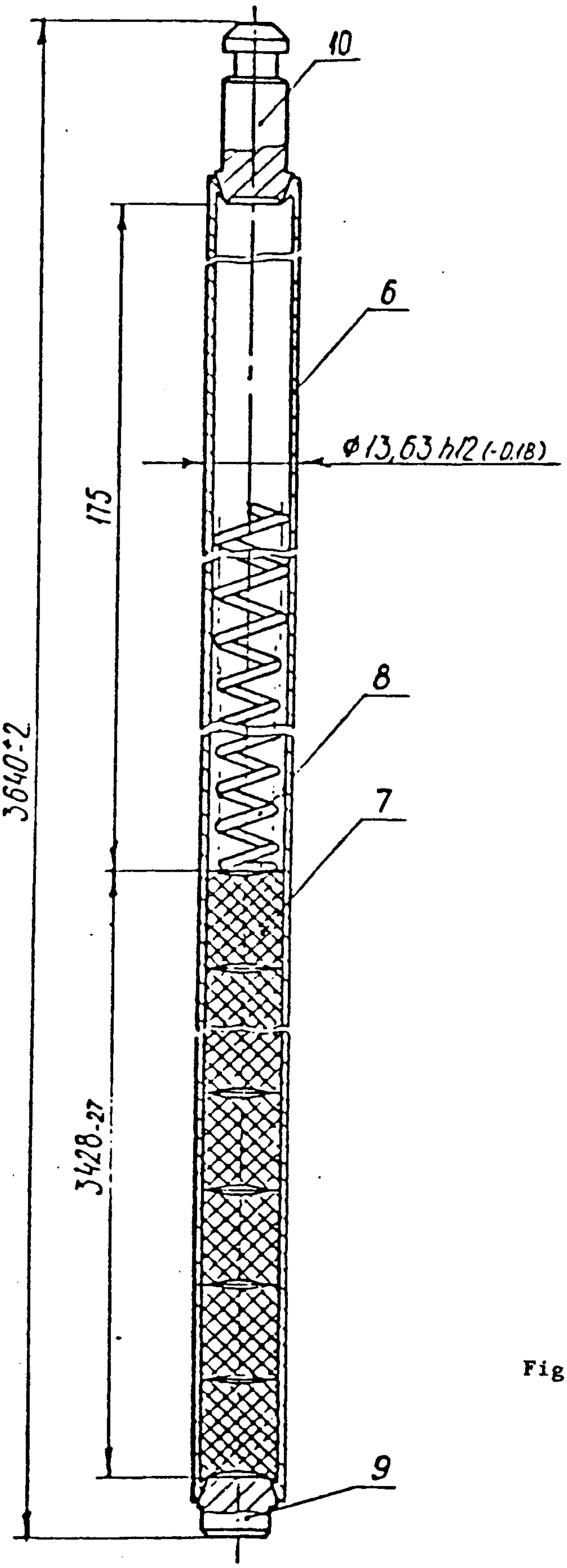


Fig. 2.2a

RBMK fuel channel

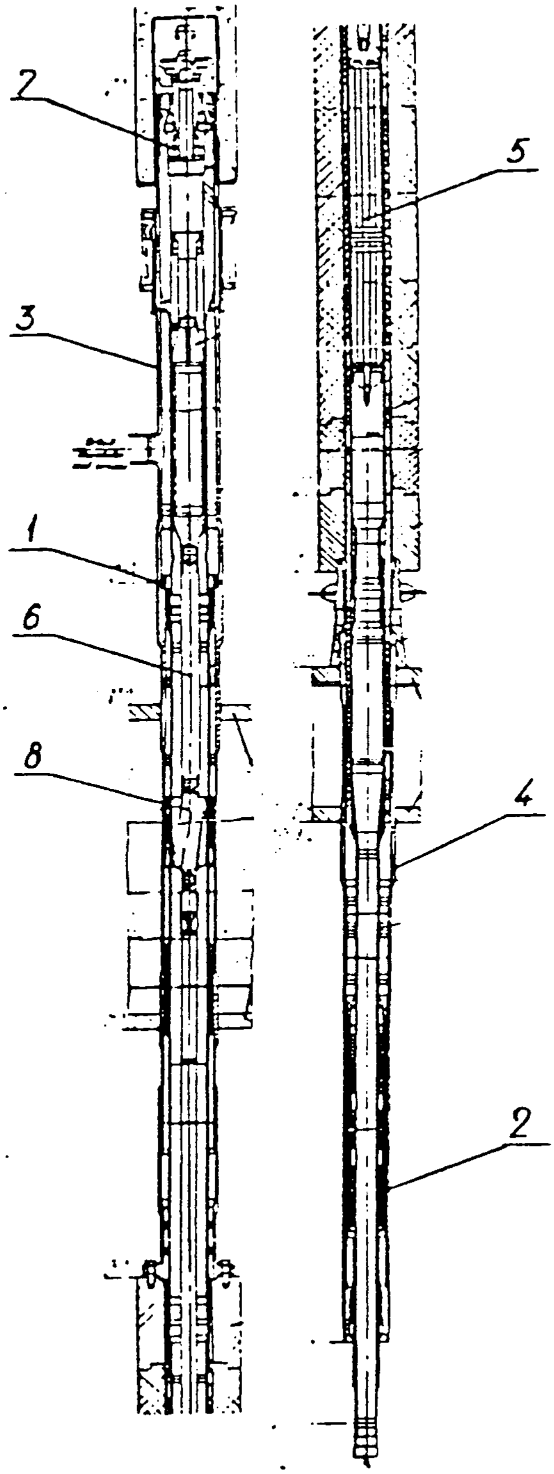


Fig. 2.3

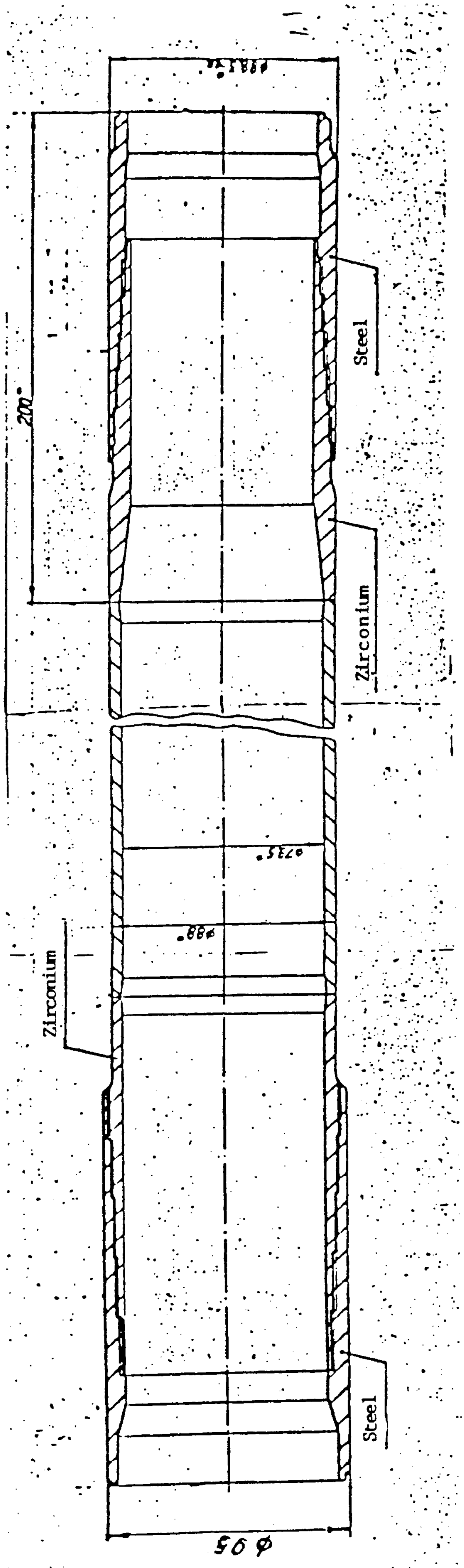


Fig. 2.3a

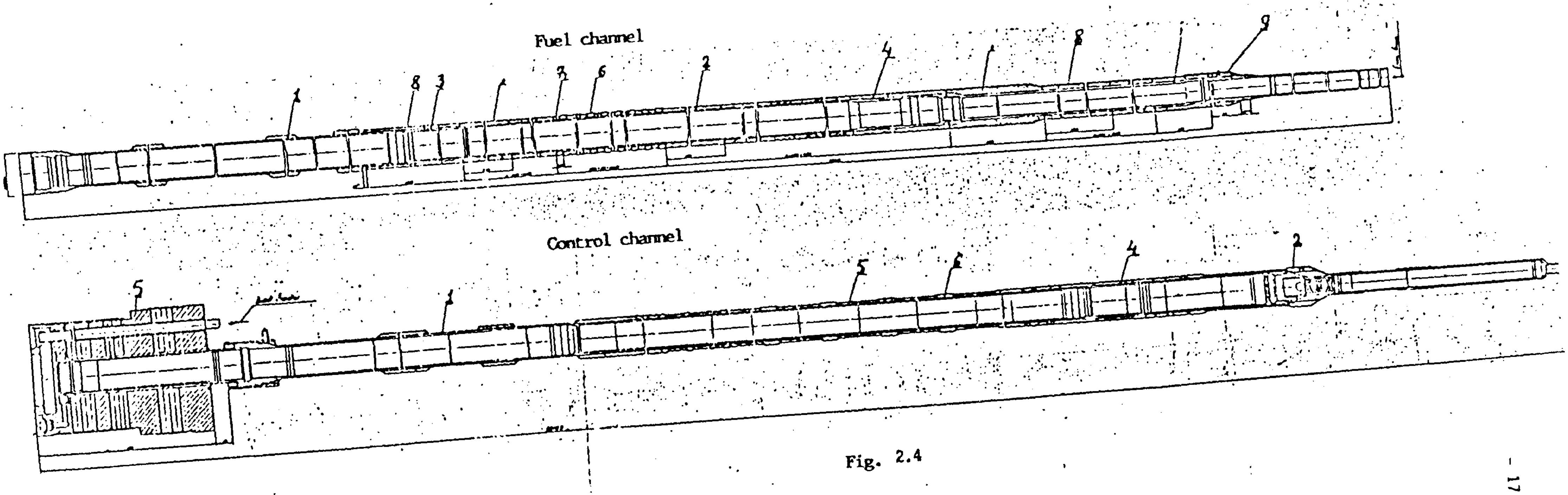


Fig. 2.4

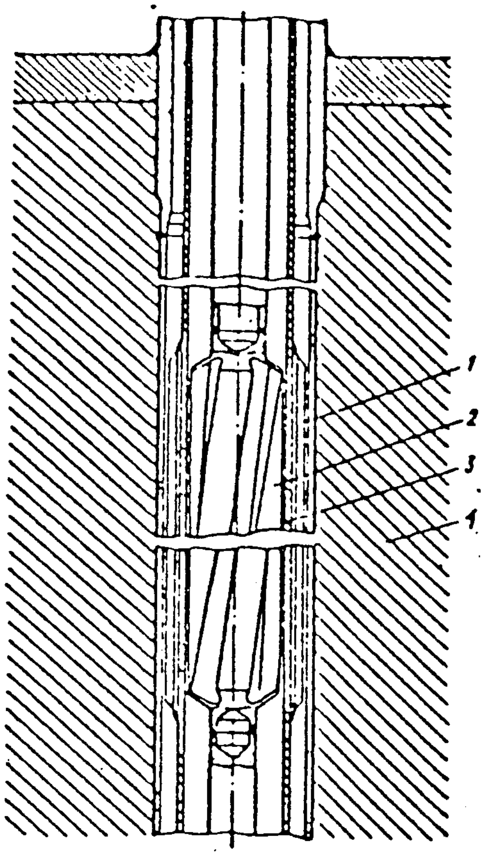


Fig. 2.5. Position of shielding plug in fuel channel: (1) steel sleeves; (2) helical steel plug; (3) channel tube; (4) serpentinite filling

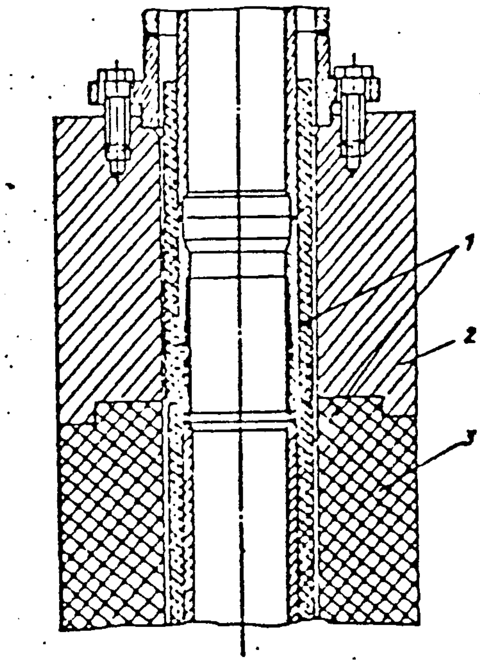


Fig. 2.6. Arrangement of shielding sleeves in upper reflector area: (1) graphite sleeves; (2) steel shielding block; (3) graphite reflector

2.2. Primary circuit (multiple forced circulation circuit)

(Fig. 2.6) [?]

The purpose of the primary circuit is to supply water to the process channels and to remove the steam-water mixture, which forms in them as a result of the heat taken up from the fuel assemblies, for subsequent separation of the steam. It consists of two loops, similar in their arrangement and equipment, which function in parallel and indendently; each removes heat from half of the reactor's fuel assemblies. A loop includes: 2 drum-type steam separators (int. diam. = 2600 mm), downpipes (325 x 16), 4 main circulation pumps, main circulation pump suction pipes (int. diam. = 752 mm) and fittings, main circulating pump pressure header (int. diam. = 900 mm), distributing headers (325 x 15 mm) with isolating and regulating valves, water lines (57 x 3.5 mm), process channels and steam lines (76 x 4 mm). (A diagram of the primary circuit fittings is shown in Fig. 2.7). [Missing from original.]

Water from the suction header (1) passes through four pipes to the main circulating pumps (2). Under normal operating conditions at normal power three of the four main circulating pumps are in operation, with one held in reserve. Water leaves the main circulating pumps at a temperature of 270°C at a pressure 82.7 kgf/cm^2 through pressure pipes, in each of which are installed in sequence a non-return valve, a gate valve and a throttle valve, and then flows into the main circulating pump pressure header (3), from where it passes through 22 lines into the distributing headers (4), which have non-return valves at their inlets, and then along individual water lines (5) into the process channel inlets (6). The flow rate through each process channel is determined by means of isolating and regulating valves in accordance with the flowmeter readings. As it passes through the process channels, the water surrounding the fuel elements is heated to saturation temperature, partially evaporates (14.5% on average) and the steam-water mixture at a temperature of 284.5°C and a pressure of 70 kgf/cm^2 (~7 MPa) flows through the individual steam lines (7) into the separators (8), where it is separated into steam and water. In order to keep the levels the same, the separators are interconnected with separate shunts for water and steam. Saturated steam passes through the steam collectors to the turbines. The water which has been separated out is mixed at the separator outlets with feed water, and flows through 12 downpipes (from each separator) into the suction header at a temperature of 270°C; this provides the cavitation margin required by the main circulating pumps.

The temperature of the water flowing into the suction header depends on the rate of steam production of the reactor unit. When this decreases, the temperature increases somewhat because of the changing ratio of water from the drum separators, at a temperature of 284°C, and feed water, at a temperature of 165°C. When the reactor is being powered down, the flow rate through the primary circuit is controlled using throttle-type control valves so that the temperature at the main circulation pump inlet maintains the necessary cavitation margin.

2.3. Special control channel cooling circuit

There is a special, independent cooling circuit for the side screen and the control channels, vertical power density monitors and the startup ionization chambers. The water circulates under gravity, i.e., because of the difference in level between the upper (storage) and lower (circulation) tanks. Cooling water at 40°C flows from the upper tank through a header along individual lines to the channel end-plugs, and continues downwards removing heat and warming up in turn to a temperature of 65°C. It then passes through a discharge header into heat exchangers, where it is cooled to 40°C, and collects in the lower tank, from which it is pumped back up into the upper tank. The mean flow rate through the control channels is 4 m^3/h and the overpressure at the channel end-plugs is 3.5 kgf/cm^2. The flow rate through each channel is controlled using isolating and regulating valves in accordance with the flow meter readings.

2.4. The gas circuit

Under normal operating conditions, a helium-nitrogen mixture flows at 200-400 nm^3/h [sic] at an overpressure on entering the reactor space of 50-200 mm head of water equivalent (0.5-2.0 kPa) through pipes which pass through the lower part of the metal structure, it is removed through the process channel failure monitoring system pipes and through special channels which remove the gas from the piping sectors of the upper part of the metal structure. The gas mixture then passes through a condenser, a three-stage scrubbing system, its flow rate is controlled by throttle and it returns to the reactor space. The gas is circulated by means of compressors.

The gas scrubbing system consists of a set of contact catalysers, scrubbing and dewatering units and cryogenic cooling system units. In the contact catalyser, hydrogenation with H_2 takes place at a temperature of ~160°C, with the formation of water vapour and combustion of CO to CO_2 and the release of heat. The reaction takes place in an oxygen atmosphere in the presence of a platinum catalyser. After passing through the contact catalyser, the gas passes through a refrigerator and dehumidifier and then on into the scrubbing and dewatering unit, which is equipped with zeolite and mechanical filters. Adsorption takes place and CO_2, H_3, C_2 and water vapour impurities are scrubbed from the helium-nitrogen gas, which then passes to the cryogenic cooling unit. Any impurities remaining in the gas are removed in this unit by dephlegmation at a temperature of -185°C.

2.5. Basic physics data

The RBMK-1000 nuclear power reactor is a heterogeneous channel-type thermal-neutron reactor which uses uranium dioxide of low enrichment in ^{235}U as fuel, graphite as moderator and boiling light water as coolant (the main characteristics of the reactor are shown in Table 2.3).

The RBMK reactor is based on experience with the design and many years of operation of uranium-graphite channel-type reactors in the USSR. Neutron physics calculation techniques which had proven themselves in operating reactor units therefore served as the basis for developing a methodology for neutron physics calculations for the RBMK reactor. Two main stages in the theoretical reactor physics studies can be identified:

(a) Calculation of the unit cell of the core and development of constants for full-scale core calculations;

(b) Full reactor calculations taking into account the details of the core structure.

For engineering design calculations in the first stage, use is made of programs which make it possible to calculate the spatial energy distribution of neutrons in a multi-group approximation in a multi-zone cylindrical cell and also in a cell with a cluster-type arrangement of fuel elements. For this, parameters such as the burnup of the uranium, the isotopic composition of fuel, the power of a channel as a function of time, the reaction rates of the isotopes of which the cell is composed and other characteristics are determined. The bulk of the calculations are performed for a one-dimensional cell with parameters averaged over the height. Constants for calculations using the reactor program are prepared in the form of a polynomial dependence on burnup and power for different average coolant densities over the reactor height.

In the second stage, full reactor calculations are performed which take into account the distribution of burnup over core channels, the actual positions of control and protection system rods and the actual power of the device. Mass calculations of states are carried out using a two-dimensional two-group program taking into account the actual field distribution over the height of the reactor obtained from sensors at different heights. Where necessary, a three-dimensional program is used for performing reactor calculations.

In addition to theoretical studies, in the design of the RBMK-1000 reactor and also during the process of operation of units already constructed, considerable attention has been paid to experimental verification and adjustment of the theoretical methodologies adopted. To this end, RBMK critical assemblies have been designed and put into operation which simulate sectors of the reactor core. At present, an extensive programme of experiments is being carried out in order to study the neutron physics characteristics of the RBMK-1000 and RBMK-1500 cores, both on assemblies and on operating units with these reactors.

With a view to achieving maximum fuel cycle economy, the RBMK-1000 reactors incorporate continuous on-load refuelling. Spent fuel is unloaded and fresh fuel loaded with the reactor operating at a specified power level using a refuelling machine. When the reactor goes into steady-state operating regime (continuous refuelling regime), all the reactor characteristics are stabilized and the fuel being unloaded from the core has an almost constant burnup, the extent of which is determined by the enrichment of the make-up fuel and also by the particular number of control rods introduced into the core which is needed for ensuring the optimum field of power output over the radius and height of the reactor. The reactivity excess for RBMK-1000 reactors adopted is 1.5-1.8% (30-36 manual control rods). With a 2% enrichment in ^{235}U in the make-up fuel, the burnup of fuel unloaded is P = 22.3 MW.d/kg. It should be particularly borne in mind, that as a result of structural materials with low absorption cross-sections and a coolant with a high steam content being used, fuel being unloaded from the RBMK-1000 reactor in continuous on-load refuelling regime has a fissionable isotope content similar to that in the wastes from enrichment plants, which practically excludes any need for it to be reprocessed for recycling.

In the design of the RBMK-1000 reactor, particular attention has been paid to demonstrating the viability of the fuel and channel element.

The main parameters determining the limiting thermal load of the fuel channel and element are the critical channel power N_c^{cr}, at which departure-from-nucleate boiling occurs on the surface of fuel elements, causing overheating of the fuel cladding, and secondly, the maximum permissible linear load to the fuel element q_l^{cr}, above which the dioxide fuel melts.

In order to estimate anticipated N_c and q_l values in the reactor, a probabilistic method for determining possible deviations was used which takes into account different factors influencing the limiting values of N_c and q_l, including the accuracy of measurement and maintenance of reactor power as a whole and its distribution over the core (the coefficients of inhomogeneity over the core radius C_r and height C_z and also over fuel elements in an assembly C_{ass} determining the maximum theoretical channel power N_c^{max} and linear load to the fuel element q_l^{max}). A Gaussian distribution was assumed for random deviations in maximum power from the most probable value of N_c^{max}. The limiting channel power is determined from the relationship:

$$N_c^{lim} = N_c^{max} \ (1 + 3\sigma_c),$$

where σ_c is the mean-square error in determination and maintenance of the channel power.

In accordance with the Gaussian distribution curve, the probability of a channel with maximum power (a channel loaded with fresh fuel in the plateau zone) exceeding a power of N_c^{lim} will be equal to (1 - 0.9987) = 0.0013.

Similarly, the limiting linear load to the fuel element is:

$$q_l^{lim} = q_l^{max} \ (1 + 3\sigma_q),$$

where σ_q is the mean-square error in determination and maintenance of the linear power of the fuel element. On the basis of calculations and operating experience, the following starting quantities have been taken for estimating N_c^{lim} and q_l^{lim}:

$$C_r = 1.48; \ C_z = 1.4; \ \sigma_c = 5.2\%; \ \sigma_q = 7.7\%.$$

In addition to economic and thermal engineering criteria, a factor of considerable importance - especially from the point of view of operating safety - is the dynamic characteristics of the core. The so-called void coefficient of reactivity α_ϕ is of particular significance. Both experimental studies on operating RBMK units and theoretical studies show that, with design parameters of a core in the refuelling regime adopted, the coefficient α_ϕ is positive and reaches a value of 2×10^{-6} units per percent steam over the volume.

However, the set of means developed for controlling the RBMK reactor includes systems which reliably ensure compensation of possible instabilities in the power output field associated with a positive reactivity feedback in terms of steam content. Specifically, the control and protection system includes local automatic control and local emergency protection sub-systems. Both operate from the signals of ionization chambers within the reactor. The local automatic control sub-system automatically stabilizes the basic harmonics of the radial-azimuthal distribution of power output, while the local emergency protection sub-system provides emergency protection for the reactor against an increase in power in individual regions of the fuel assembly in excess of the specified level. There are 24 shortened absorber rods for controlling the vertical fields, and these are introduced into the core from below.

In addition to improvements in reactor monitoring and control systems, there are other means of improving the dynamic characteristics of the RBMK core.

These include the following:

- Increase in make-up fuel enrichment to 2.4-3.0%, leading to a corresponding increase in burnup which makes it possible to reduce the void effect practically to nil;

- Increase in the amount of uranium loaded into reactor channels by using fuel compositions with a high U content.

Table 2.3 shows theoretical estimates of effects and coefficients of reactivity associated with the variation in moderator and fuel temperatures and also the "fast" power coefficient of reactivity.

Theory and experiments show that the "fast" power coefficient of reactivity is negative and near zero when the reactor is operating at nominal parameters.

Table 2.3. Basic neutron physics characteristics of the RBMK-1000 reactor.

No.	Parameters	Value
1.	Fuel enrichment, %	2.0
2.	Mass of uranium in an assembly, kg/ass	115
3.	Number/diameter of fuel elements in sub-assemblies, mm	18/13.6
4.	Burnup, MW.d/kg	20
5.	Coefficient of inhomogeneity of power output over the core radius	1.48
6.	Coefficient of inhomogeneity over the core height	1.4
7.	Limiting theoretical channel power, kW	3250
8.	Isotopic composition of fuel unloaded, kg/t	
	^{235}U	4.5
	^{236}U	2.4
	^{239}Pu	2.6
	^{240}Pu	1.8
	^{241}Pu	0.5
9.	Void coefficient of reactivity at operating point, 10^{-6}% steam	2.0
10.	"Fast" power coefficient of reactivity α^{o}_{W}, 10^{-6}/MW	-0.5
11.	Temperature coefficient of fuel α_T, $10^{-5}/^{o}C$	-1.2
12.	Temperature coefficient of graphite α_C, $10^{-5}/^{o}C$	6
13.	Minimum "weight" of control and protection system rods, ΔK	10.5%
14.	(Averaged) effect of replacing spent fuel by fresh fuel	0.02%

2.5.1. Basic thermal physics data

[See end of this section for list of symbols]

2.5.1.1. Parameters determining the possibility of reactor operation in thermal terms

For a boiling water-graphite reactor, the main parameters determining the possibility of operation and its safety in thermal terms are as follows: fuel element temperature, temperature of the graphite stack and the margin to channel power at which departure-from-nucleate boiling occurs.

The condition of the hydrodynamic stability of the fuel channels of a boiling water-graphite RBMK-1000 reactor are usually not a limiting factor since hydrodynamic instability as a rule occurs at channel powers higher than those at which departure-from-nucleate boiling occurs.

Experimental studies performed during design have confirmed that this conclusion is correct and have shown that the nominal operating parameters of the RBMK-1000 reactor are within the region of hydrodynamic stability.

If the permissible fuel temperature is exceeded or departure-from-nucleate boiling occurs, an individual fuel assembly may become defective, but after it has been exchanged the reactor's capacity for operation is restored.

Calculations of the margins to critical power and of the maximum fuel element temperature in RBMK-type reactors at steady-state power levels are performed using probabilistic statistical methods, and the same methods are used as a basis for monitoring the core condition of such reactors during operation.

In transient and accident regimes, when there are rapid changes in parameters, it is advisable to assume a higher probability of the limiting values for thermal parameters being exceeded than with operation at steady-state power levels. Experimental data and operating experience with boiling water-graphite reactors show that short-lived departure-from-nucleate boiling and increase in fuel temperature above the level permitted for steady-state regimes in these reactors do not cause fuel assemblies to become defective.

The margin to critical power and the maximum fuel temperature in transient and accident regimes with boiling water-graphite reactors are determined from the actual average values for the parameters influencing these quantities.

2.5.1.2. Thermal physics characteristics in steady-state operating regimes

The composition of the RBMK core depends on the operating cycle. During the first part of the operating cycle with these reactors, the core contains channels with fuel of low burnup and a large quantity of additional absorbers needed for compensation of excess reactivity. As burnup proceeds, the fuel inventory in the core changes continuously. During this transitional period of the cycle the core contains channels with fuel of different burnup levels, additional absorbers of different effectiveness and also channels filled with water. This transitional period of reactor operation ends after all or almost all additional absorbers have been extracted from the core and they have been replaced by fuel assemblies. The reactor is then operated in continuous on-load refuelling regime, in which a refuelling machine is used to replace spent fuel assemblies by fresh ones.

A reactor operating in continuous on-load refuelling regime can be represented as a system consisting of what can be considered "refuelling-cycle" cells. Each "refuelling-cycle" cell consists of channels loaded with fuel assemblies of different levels of burnup. At any given moment, different channels will have different powers, but the total power of all channels of a "refuelling-cycle" cell will remain approximately constant.

The design of the RBMK reactor is such that during a fuel campaign it is possible to regulate the flow of water through fuel channels by varying the aperture of the isolating and regulating valves at the inlet to each channel during operation. The purpose of regulating the water flow of each channel individually is to ensure that there is a sufficient margin to departure-from-nucleate boiling in those core channels which are under the greatest thermal stress, while maintaining the total flow of water through the reactor at a moderate level. Regulation of the water flow through a channel during a campaign is carried out in accordance with the readings of a flowmeter placed at the inlet to each reactor channel until a theoretically determined flow rate is reached; this is based on ensuring the necessary steam content at the outlet from the channel or the necessary margin to departure-from-nucleate boiling in a given channel.

A distinguishing feature of RBMK reactors is that the water flow can be measured and controlled in each fuel channel. This ensures the redistribution of water flow with changes in reactor power and in the power output field over the core radius.

In accordance with the algorithm of thermal calculations for RBMK reactors, the distribution over core fuel channels of water flow is calculated by the standard iteration method using the combined characteristics of the circulation pumps and the downcoming portion of the circulation circuit.

In order to determine the hydraulic characteristics of individual structural components and of the reactor fuel channel as a whole, experimental studies were performed on special test rigs and also on a full-scale reactor fuel channel simulator with a thermal power of up to 6 MW.

The relations for computing the relative coefficient of hydraulic resistance of a cluster of rods immersed in two-phase flux have the following form:

$$\psi = 1 + 0{,}57\left(\frac{1}{0{,}2 + \frac{W_0}{\sqrt{g d_r}}\,\frac{\rho''}{\rho'}} - 5{,}2x^2\right)x^{0{,}125}\cdot(1-x)^2$$

for the actual volume steam content in the channel:

$$\varphi = \frac{1}{1 + \frac{1-x}{x}\cdot\frac{\rho''}{\rho'}\cdot K}$$

and for the coefficient of phase transition:

$$K = 1 + \frac{0{,}6 + 1{,}5\beta^2}{\sqrt[4]{\frac{W_0^2}{g d_r}}}\left(1 - \frac{p}{225}\right),$$

where $W_0 = G/\rho'S$ is the circulation rate and β is the volume steam content flow.

Figure 2.8 compares experimental values for pressure gradients in the heated part of the full-scale test rig and data obtained theoretically.

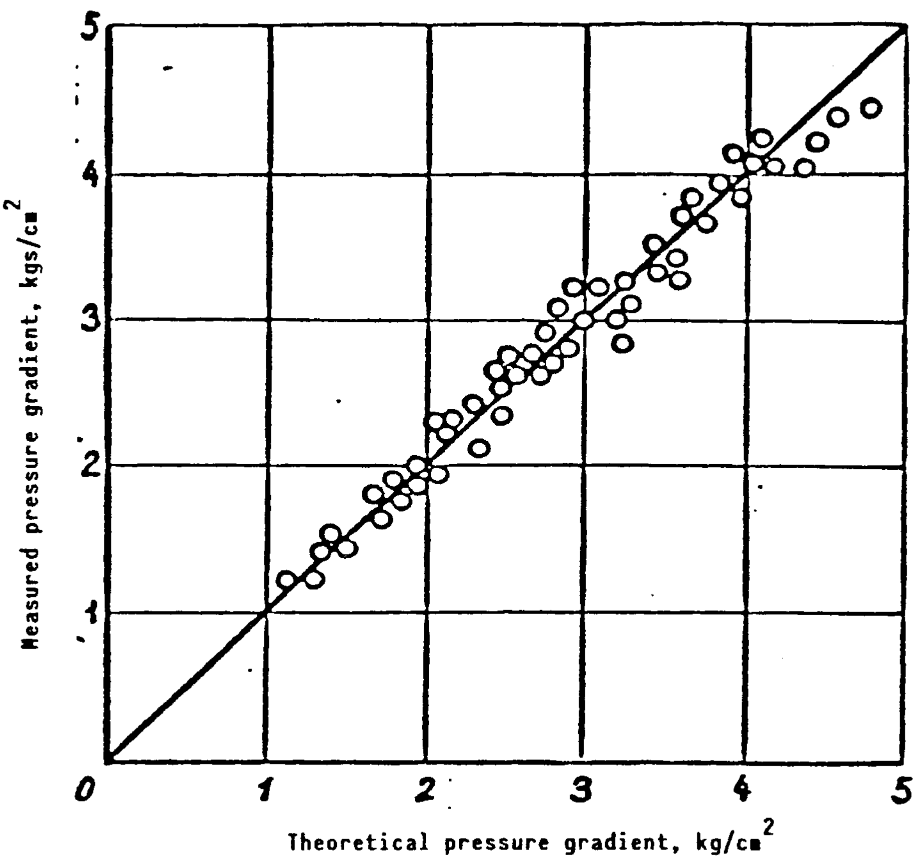


Fig. 2.8. Comparison of experimental and theoretical values for hydraulic resistance of a full-scale test rig.

O - experimental values

___ - theoretical values

It will be seen from the figure that the theoretical method satisfactorily represents the experimental data and can be used for performing thermal calculations for reactors.

At given thermal powers in each fuel channel and at a given water flow rate through it, the critical channel power N_{cr}, the minimum margin to departure-from-nucleate boiling K_m, the probability of departure-from-nucleate boiling occurring in a channel R and also the probability of all core channels operating without departure-from-nucleate boiling H are determined.

Dependences for calculating the critical power of an RBMK fuel channel were determined following analysis and processing of experimental information on departure-from-nucleate boiling in smooth clusters of heated rods and in clusters of rods with heat flux intensifiers. The experimental work was performed on test rigs with different cluster geometries (including full size) and with coolant parameters similar to operating reactor parameters.

The dependence for calculating the critical heat flux in fuel channels without heat flux intensifiers has the form:

$$10^{-6} q_{cr}(Z) = \frac{4{,}3 \cdot d_{he}^{0{,}83} \cdot (\rho W \cdot 10^{-3})^{0{,}57} + 0{,}98 \cdot 10^{-2} \cdot d_{he} \cdot \rho W \cdot 10^{-3} \cdot \Delta h}{664 \cdot d_{he}^{0{,}57} \cdot (\rho W \cdot 10^{-3})^{0{,}18} + 39{,}4 \cdot \dfrac{\int_0^Z \Phi(z)\,dz}{\Phi(Z_{cr})}},$$

where $\Phi(Z)$ is the relative distribution of power output over the channel height;

Z_{cr} is the co-ordinate of the place in which departure from nucleate boiling occurs (m);

and Δh is the heating of water to saturation at the inlet (kJ/kg).

The set of programs developed can be used for performing the thermal calculations for the RBMK reactor operating in continuous on-load refuelling regime with any positions of the isolating and regulating valves at the inlets to each "refuelling-cycle" cell channel. By the same means it is possible to determine the thermal parameters of the reactor with different frequencies of individual channel flow regulation, different regulation criteria (in accordance with outlet steam content or with the margin to critical power) and also with core flow reduced in advance to different extents.

The results of theoretical studies on the influence of the frequency of individual channel flow regulation on thermal parameters of an RBMK reactor with an electrical output of 1000 MW (RBMK-1000) operating in continuous on-load refuelling regime are shown in Fig. 2.9. It will be seen from the dependences shown that, when the frequency of individual channel flow regulation increases, the parameter H, which stands for the thermal reliability of the core, increases; this increase is most marked when the frequency of regulation is increased to twice per fuel campaign. A further increase in regulation frequency does not lead to a large increase in the parameter H. On the basis of the calculations performed during design for an RBMK-1000 reactor operating in continuous on-load refuelling regime, it was assumed that the water flow through each fuel channel would be regulated twice during a fuel assembly campaign.

For the thermal calculations for a reactor operating in the transitional period of the cycle (from the point of view of refuelling), a mathematical model has been developed with which it is possible to derive the distribution over the core channels of water flow and of margins to departure-from-nucleate boiling taking into account the specific characteristics of each individual reactor channel. In this case the reactor core is represented in the form of a system consisting of channels loaded with fuel assemblies of different levels of burnup and additional absorbers of any type. The distribution over the reactor channels of power output is determined either as a result of physics calculations for the core condition and control rod positions being considered or is transmitted to the reactor designer by means of a special automatic system for linking operating RBMK reactor units. As a result of reactor calculations for the given core condition and power output distribution over reactor channels, an optimum distribution over the channels is found for the water flow, as is the hydraulic profile of the core needed for this.

At operating RBMK reactors the margins to critical power and the temperature conditions of the fuel are monitored by a special program (PRIZMA program) using the plant's own computer. The temperature conditions of the graphite stack are monitored using thermocouples placed over the radius and height of the stack.

For calculating the distribution of power output over the reactor core, use is made of the readings of a system of physical monitoring based on in-reactor measurements of the neutron flux over the radius and height of the core. In addition to the readings of the system of physical monitoring, data on the core composition, the power output of each fuel channel, the positions of control rods and the distributions over the core channels of water flow and the readings from coolant pressure and temperature sensors are also fed into the plant's computer. After the computer has used the PRIZMA program to perform calculations on a periodic basis, the operator receives information on a digital printer in the form of core diagrams showing the type of fuel load in the core, the positions of control rods, the arrangement of the network of in-reactor sensors and the distributions of power, of water flow and of the

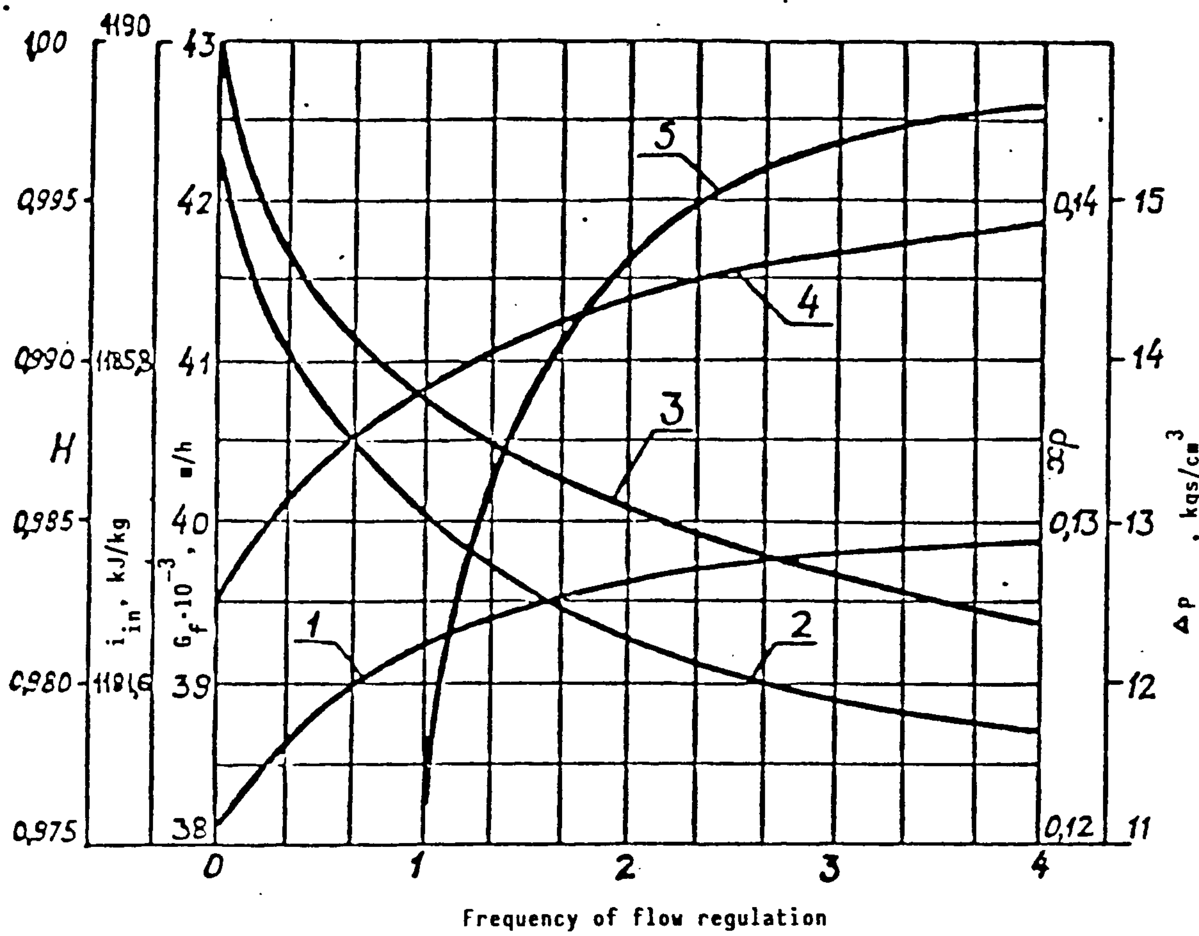


Fig. 2.9. Reactor parameters as a function of frequency of individual channel flow regulation:

1. Main circulation pump pressure (ΔP_{pump});
2. Coolant flow rate (G_f);
3. Heat content at inlet (i_{in});
4. Steam content at outlet (x_p);
5. Thermal reliability (H).

margins to critical power and to maximum permissible thermal loads to fuel elements for each fuel channel. The margins to critical power and maximum permissible thermal loads are calculated by means of a probabilistic statistical method taking into account errors in determination of the power output field over the height and radius of the reactor and errors in calculation formulae and in the accuracy of measurement and maintenance of process parameters of the plant by monitoring and measuring instruments and automatic systems. The plant's computer also calculates the overall thermal power of the reactor, the distribution over separators of the flow rates of steam-gas mixture, the integral power output, the steam content at the outlet from each fuel channel and other parameters needed for monitoring and controlling the plant.

When the reactor is operating at steady-state power levels, and also during periods of power increase or decrease, the operator monitors and controls the energy output field over the radius and height of the core using the readings from sensors of the physical monitoring system. If the field deviates by a certain amount from the specified value, a warning light begins to shine on a special panel. Also a warning is triggered off if the signals from the sensors exceed the specified absolute values of the margins to maximum permissible thermal loads to fuel elements (K_q). The operator also monitors and controls the distribution of flow rates over the core fuel channels. The distribution of flow rates is obtained on the basis of calcuations by an outside computer and by means of the PRIZMA program on the plant's computer on the basis of the distribution of margins to critical power (K_m) over the fuel channels.

The temperature conditions of the graphite stack in operating RBMK reactors are monitored by means of thermocouples placed at the corners of graphite blocks at various points over the volume of the stack. In addition to direct measurements of the graphite temperature at reference points in the stack, the PRIZMA program can be used to calculate the maximum temperature (over the height) of the graphite in the environs of any reactor fuel channel. The graphite temperature is found on the basis of readings from the thermocouples and of the distribution of power output over the core volume calculated by means of the PRIZMA program.

The temperature conditions of the graphite stack in RBMK reactors are controlled by varying the composition of the gaseous mixture in the stack (nitrogen and helium). On the basis of operating experience with Soviet water-graphite reactors, the maximum graphite temperature at which the stack does not burn up in the absence of water vapour has now been found to be 750°C.

Experience with operating RBMK reactors shows that, with the existing monitoring and control systems at these reactors, maintenance of the temperature conditions of the fuel and the graphite and of the margins to departure-from-nucleate boiling at the permissible level with steady-state power levels does not give rise to any difficulties.

Symbols used in Section 2.5.1 (Basic thermal physics data)

G - flow rate, kg/s;

S - cross-sectional area, m^2;

d - diameter, m;

g - acceleration in gravitational field, m/s^2;

x - mass steam content;

ρ - density, kg/m^3;

P - pressure, kgs/cm^3

q - thermal flux density, kW/m^2;

w - velocity, m/s.

Superscripts and subscripts

he - heated;

g - hydraulic;

cr - critical;

/ - water on saturation line;

// - steam on saturation line.

2.6. TECHNOLOGICAL LAYOUT OF THE UNIT

The technological layout consists of a single loop designed according to the "twin unit" principle, i.e. one reactor + two turbines with no steam and feedwater cross-connections.

At power, the unit operates in accordance with the following scheme (see Fig 2.1.0).

The coolant (water) of the multiple forced circulation circuit (MFCC) (primary coolant circuit) is pumped through downcomers (325 x 16 mm) from the lower part of the steam separators at a temperature of 265^{o}C and a pressure of 69 kgf/cm^2 to the intake header (1026 x 63) of the main circulation pumps. These pumps feed the water to the pressure header (1046 x 73) and then via pipes (325 x 16) to the 22 distributing group headers of the reactor. From the distributing group headers, the water is delivered individually to the reactor fuel channels through the pipes of the lower water communication lines (57 mm in diameter).

The steam-water mixture formed in the reactor passes through the pipes of the steam-water communication lines (76 mm in diameter) and is distributed to four steam separators in order to produce saturated steam to operate the turbines.

Steam is removed from the top part of each separator through 14 steam discharge pipes (325 x 19) to two steam headers (426 x 24 in diameter) which then link up in a single header (630 x 25).

Live steam is supplied via four pipes (630 x 25) to the turbines in the machine room (two pipes per turbine).

The pipe section located before the turbine main steam valves contains various steam discharge devices: eight main safety valves with a throughput of 725 t of steam per hour, four turbine condenser fast-acting steam dump stations with a capacity of 725 t of steam per hour (two per turbine plant) and six service load fast-acting steam dump stations. The purpose and mode of operation of these devices are described in Section 2.7.

The exhaust steam from the turbines is condensed in the condensers. After being completely purified in the desalination plant, the condensate from the condensers is pumped through the low pressure heaters to the deaerator at 7.6 atm (two deaerators per turbine). Five electric feedpumps (one of which is a back-up) deliver the feedwater at a temperature of 165^{o}C from the deaerators to the steam separators where it is mixed with the circulating coolant.

In addition to the MFCC, the reactor's main process systems include:

- Emergency core cooling system;

Chemical composition of primary coolant circuit materials

Material	C	Mn	Si	S	P	Ni	Cr	Cu	Nb	Mo
Steel "Krezelso" 330E	≤0,23	0,9-1,2	0,2-0,4	≤0,025	≤0,025	<0,3	≤0,4	≤0,3	-	-
Steel 1CL473Nb	≤0,05	≤0,2	≤0,75	≤0,02	≤0,035	8,5-10,5	18-20	8x% C-0,65	-	0,6

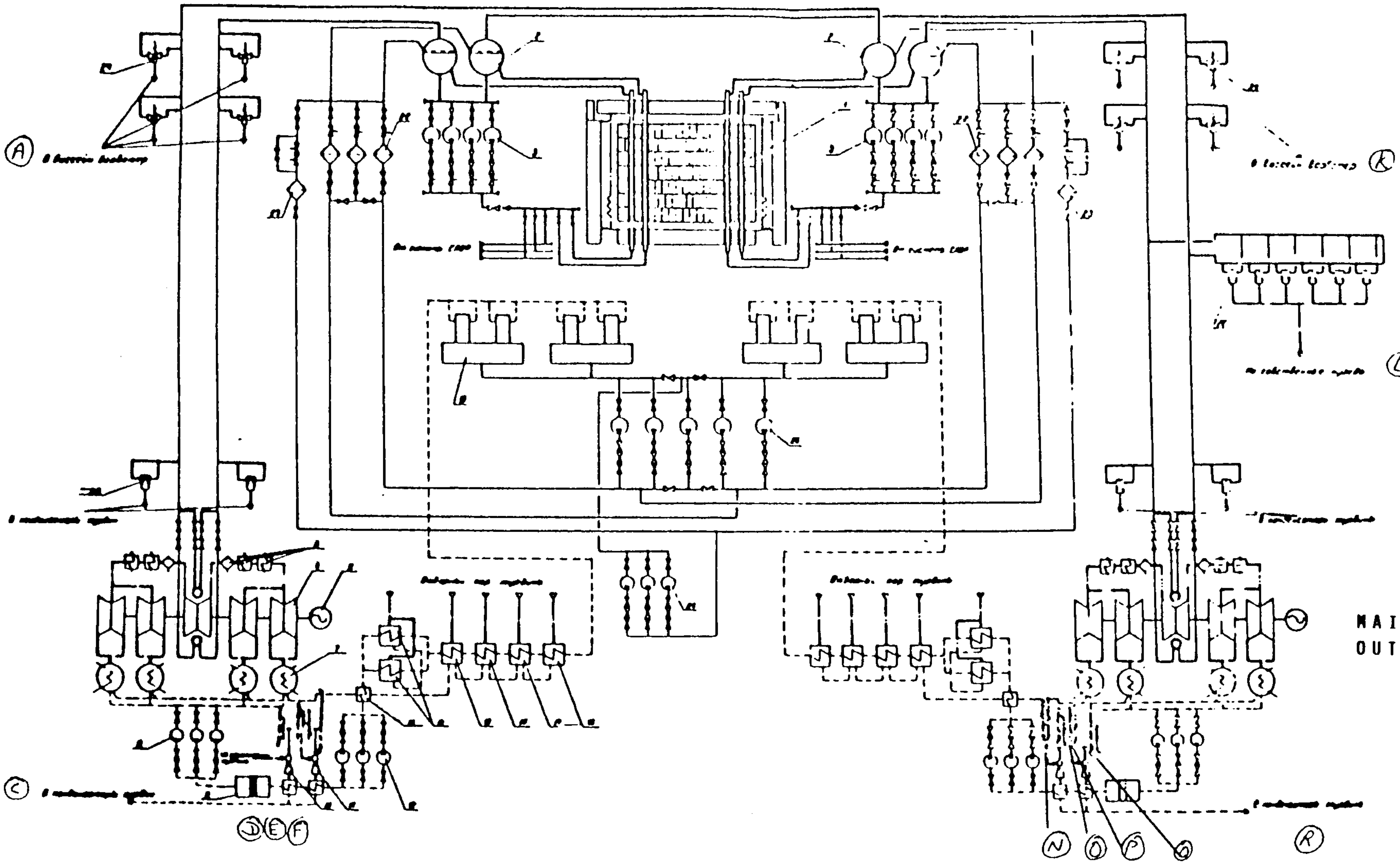
Physical and mechanical properties of the materials

Material	At temperature, °C	$\alpha \cdot 10^6$ 1/°C	E, n/mm²	σ_B kgf/mm²	$\sigma_{0,2}$ kgf/mm²	δ_5 %	ψ, %	a_{HI} kgf/cm²	a_{IV}, kgf/cm²
Steel "Krezelso" 330E	20	11,1	205200	44-60	≥22	≥20	≥48	≥7	≥4
	350	15,0	188000	≥36	≥19	≥18	≥43	-	≥3
Steel 1CL473Nb	20	16,5	193000	≥50	≥20	≥38	≥50	-	-
	350	17,5	179000	≥36	≥15	≥24	≥40	-	-
08Cr18Ni10Ti	20	16,4	205000	52	22	35	55	-	-
	350	17,6	175000	42	17	26	51	-	-

MAIN THERMAL LAYOUT OF THE UNIT

A.	To pressure suppression pool
B.	To turbine condensers
C.	To turbine condensers
D, E, F.	Original illegible
G.	From ECCS
H.	Steam from turbine
I.	From ECCS
J.	Steam from turbine
K.	To pressure suppression pool
L.	To station internal load
M.	To turbine condensers
N.	From turbine condensers
O.	Steam from de-aerators
P.	Original illegible
Q.	Steam from evaporators
R.	To turbine condensers

SECTION 2.6.



MAIN THERMAL LAY-
OUT OF THE UNIT

- Primary cooling circuit flushing and shutdown cooling system;
- Gas circuit;
- Fuel cladding failure detection system;
- Cooling pond water cooling and purification system;
- Cooling system for the water of the system "D" biological shield tanks;
- Cooling system for the control and protection system channels;
- Intermediate loop of the reactor section;
- Channel tube failure detection system.

Primary coolant circuit

The purpose of the MFCC is to provide the reactor fuel channels with a continuous supply of coolant which removes the heat produced by the reactor and to generate a mixture of steam and water which is then separated to produce saturated steam to work the turbines. The circuit consists of two identical independent loops each of which cools one half of the reactor. All the equipment of these loops is arranged symmetrically about the transverse axis of the reactor. Each circulating loop contains:

- Two steam separators;
- Water and steam connection lines between the steam separators;
- Downcomers;
- Intake header;
- Main circulation pump intake pipes;
- Four main circulation pumps (three operating, one back-up);
- Main circulation pump discharge pipes and fittings;
- Pressure header;
- Connection line between the main circulation pump intake and pressure headers and their fittings;
- Distributing group headers;
- Pipes of the lower water communication lines;

- Reactor fuel channels;

- Steam-water communication line pipes.

The downcomers, distributing group headers and the pipes of the water and steam-water communication lines are made of stainless steel 08Cr18Ni10Ti. The pressure and intake headers and pipes of the main circulation pumps are made of carbon steel 330 E surfaced with 1CL473Nb steel from a French firm Creusot-Loire.

The pump discharge pipes contain, in series, a check valve, throttle valve, gate valve with remotely controlled electric drive and an orifice meter. The presence of the gate valves on the intake and discharge pipes of the pumps enables a pump to be removed for repair while the circuit is in operation.

The throttle valve makes it possible to keep the main circulation pump capacity within the unit's steady state operating range of 5500-12000 m^3/h in transient conditions. The intake and pressure headers are linked by a connection line 750 mm in diameter, the purpose of which is to ensure natural circulation in the loop when the pumps are not operating. The connection line has a check valve which prevents the medium from flowing from the pressure header to the intake header under normal loop operating conditions, as well as a gate valve which is normally open under all operating conditions.

Inserted in the discharge nozzles of the pressure header are leak limiters in the case of pipe rupture. During the pre-start-up flushing period, mechanical filters were mounted on these. The pipes supplying water to the distributing group headers have manual gate valves. Under normal conditions these valves are locked open; they only shut when repair work is being done on the primary circuit. The distributing group header is fitted with check valves, beyond which (in terms of the direction of flow) the pipes of the water communication lines deliver water individually from the headers to the reactor fuel channels.

Primary coolant circuit flushing and shutdown cooling system

The purpose of this system is:

- Under rated conditions, to cool the flushing water of the MFCC before it is purified, heated and returned to the MFCC;

- Under shutdown cooling conditions, to remove heat from the primary coolant circuit;

- Under start-up conditions, to cool the flushing water of the circuit before it is purified, heated and returned to the circuit, and to discharge disbalance waters from the circuit when it is being heated.

The flushing and shutdown cooling system comprises a regenerative heat exchanger, large and small flushing aftercoolers, two shutdown cooling pumps, pipes and fittings.

2.6.1. Gas circuit. Condenser and filter station

In order to prevent oxidization of the graphite and to improve the transmission of heat from the graphite to the fuel channel, the gaps between the graphite blocks and rings of the reactor stack are filled with a mixture of nitrogen and helium (20 vol.% N_2 and 80 vol.% He). Impurities are removed and the nitrogen-helium ratio of the gaseous mixture is maintained by the helium purification station.

Under normal conditions the gas circuit system functions in the following manner. The nitrogen-helium mixture from the station passes through the channel tube failure detection system where channel-by-channel monitoring of the temperature and group monitoring of the humidity of the mixture being pumped through is performed.

The mixture then enters the condenser and filter station.

The purpose of the gas circuit condenser and filter station is to condense steam which gets into the nitrogen-helium mixture when the reactor channels lose their leak tightness and to remove iodine vapour from the gas mixture.

This system is designed in accordance with the principle of 100% redundancy, i.e. it has two independent subsystems one of which functions and the other is a back-up. Each subsystem contains a gas circuit condenser, an electric heater and a filtration column.

The nitrogen-helium mixture from the reactor space enters the condenser. The condensate from the condenser is removed through a water seal to the floor drain tanks via a permanently open repair valve.

Service water is supplied to the condenser at a pressure exceeding that of the steam-gas mixture both in rated and accident conditions.

After the condenser stage, the gas mixture has a humidity of about 100%. If it goes straight to the filter it is possible that the moisture will condense with the result that the filter will break down. For this reason the gas mixture is dried in the electric heater section before proceeding to the filtration column. The column purifies the mixture of solid particles and iodine in aerosol form.

The filtration column is designed to purify 1000 m^3 of the gas mixture per hour. Upon leaving the filtration column, the gas mixture goes either to the compressor intake header of the helium purification plant or to the gas activity reduction system, depending on the operating mode of the gas circuit.

The two sub-systems are housed in separate isolated compartments which enables repair work to be done on the equipment of one sub-system while the other is in operation.

2.6.2. Channel tube failure detection system

This system is equipped with sensors to monitor the integrity of the fuel channels. Its purpose is to:

- Perform group monitoring of the humidity of the gas removed from the graphite stack of the reactor and pumped through the system;

- Identify damaged reactor channels;

- Prevent moisture spreading from a damaged channel to adjacent cells;

- Dry the reactor graphite stack.

At an operating reactor, channel integrity is monitored by measuring the temperature of the gas pumped through the gaps between the channels and the graphite stack (ducts). As the amount of steam in the pumped gas increases, so does its temperature which is established by thermocouples mounted in the group valves. The graphite stack with the channels which penetrate it is conventionally divided into 26 zones, each of which contains up to 81 channels.

The impulse tubes of the channel ducts in each zone run to the corresponding group valve for that zone. Each of the 26 group valves is designated by the same number as its corresponding reactor zone.

The valve outlet nozzles of the are connected by pipes to the channel tube failure detection system ventilation and intensified extraction headers. Both these headers are linked to the process gas circuit, thus joining the channel tube failure detection system to the reactor's process ventilation circuit.

It is possible to alter the gas pumping rate through the impulse tubes leading to a given valve by switching the slide valve between the ventilation system and the intensified extraction system.

2.6.3. Helium purification plant

The purpose of the helium purification plant is to purify the gas mixture circulating through the RBMK unit closed circuit of oxygen, hydrogen, ammonia, steam, carbon oxide, carbon dioxide, methane and nitrogen impurities to a level which permits the reactor to operate normally.

The gas mixture becomes contaminated because moisture can enter the stack cavity as a result of the non-leaktightness of the fuel channels. The moisture then partially decomposes as a result of radiolysis into hydrogen and oxygen which, reacting with carbon, forms carbon oxide and carbon dioxide. The hydrogen which combines with graphite forms methane and that with nitrogen, ammonia.

The main technical characteristics of the helium purification plant are:

1.	Mixture quantity at 293 K and 101325 Pa	0.0833-0.264
	(760 mmHg), $m^3/s(m^3/h)$	(300-950)
2.	Pressure at plant inlet, MPa (mm water)	0.003 (300)
3.	Composition of unpurified mixture, % (vol.):	
	nitrogen	20
	oxygen	0.3
	methane	0.1
	ammonia	0.07
	carbon dioxide	0.02
	carbon oxide	0.1
	hydrogen	0.6
	chlorine	traces
	helium	residue
4.	Pressure at plant outlet, MPa (mm water)	0.005 (500)
5.	Temperature at plant outlet, K (°C)	308 ± 10 (35 ± 10)
6.	Composition of purified mixture, % (vol.):	
	nitrogen	10
	oxygen	0.01
	methane	traces
	ammonia	traces
	carbon dioxide and oxide	0.01
	hydrogen	0.02
	helium	residue
7.	Auxiliary products used:	
	liquid nitrogen, kg/s (kg/h)	0.039 (140)
	gaseous nitrogen, m^3/s (m^3/h)	0.097 ± 0.104 (350-500)
	gaseous oxygen, m^3/s (m^3/h)	0.0042 (15)
	cooling water, m^3/s	0.0056 (20)
8.	Duration of operating run, years	1.5
9.	Duration of startup period (s/h)	57 600 (16)
10.	Time preceding first overhaul, h (year)	43 800 (5)
11.	Life expectancy, year	30

2.6.4. Cooling and purification system for spent fuel cooling pond water

The purpose of this system is to sustain the temperature regime of the fuel assembly and pressure tube cooling pond water, which is heated by the afterheat of the spent fuel and pressure tubes. The system is designed in accordance with the principle of 100% redundancy and comprises pumps, heat exchangers, pipes and fittings. The function of the pumping station of the purification plant is to deliver the cooling pond water to the ion-exchange filters of the purification plant. There is one such water purification unit for two power units. It operates periodically.

2.6.5. Cooling system for biological shield tanks

The purpose of the pumping-heat exchanger station is to control the temperature of the water in the reactor biological shield tanks. It contains the following equipment: cooling circuit circulation pumps, heat exchanger, expansion tank, pipes and fittings.

2.6.6. Cooling system for control and protection system channel

Function and design bases

The function of the cooling system for the control and protection system, fission chamber and power density monitoring channels as well as the reflector cooling channels is to ensure that the prescribed temperature is maintained in these channels. The system must meet the following requirements:

- Maintain the prescribed temperature in the aforementioned channels in all operating regimes of the unit (startup, power operation, shutdown, disruption of normal operating conditions, accident situations);

- Satisfy the regulations regarding for water quality (chemical composition and specific activity).

The system comprises a circulation loop which operates by gravity feed; in other words, the water flows through the channels as a result of the difference in levels of the upper and lower tanks.

Water from the upper tank (known as the emergency storage tank) passes through a pipe (400 in diameter) to the pressure header and is distributed among the channels.

The volume of the emergency tank is governed by the condition that it should supply the rated flow through the channels for six minutes when the pumps are out of operation.

The cooling water from the pressure header enters the channel from above, passes down through the central tube, and then travels up out of the channel through the annular gap between the central and outer tubes to the discharge header of the reflector cooling channel. There are two discharge headers (200 in diameter).

The water from the discharge header moves along the pipe (400 in diameter) to the system's heat exchangers. Into the same pipe flows water from the reflector cooling channel discharge headers: before entering the 400 diameter pipe, the water from these headers converges in a common pipe 150 in diameter. This common pipe is equipped with a throttle device which eliminates the need for a syphon in the reflector cooling channel discharge headers.

There are six heat exchangers in the system to cool the circulating water when it leaves the reactor.

Having passed through the heat exchangers, the water enters the circulation tank through a pipe (400 in diameter) below the water level. In the circulation tank the flow is slowed down and conditions are created for hydrogen to be separated efficiently from the water. A description of the method used to keep the hydrogen concentration at a safe level in the tank is given in Section 2.7.

Water from the emergency storage tank is continuously dumped into the circulation tank via overflow pipes. The amount corresponds to the difference between what the pumps deliver and the throughput of the control and protection system, fission chamber, power density monitoring and reflector cooling channels. If two of the system's pumps are working, an overflow pipe 150 in diameter is used; if three pumps are working, a pipe 300 in diameter is used.

The system is equipped with four pumps to feed water from the circulation tank to the emergency storage tank. Two of these operate and the other two are back-up. The first back-up pump is switched on automatically, while the second is activated by the operator when required.

The pumps receive power from a category 1B secure supply from the diesel generators.

In order to maintain the required water quality in the circuit, the water is continuously purified by a bypass system at a rate of 10 m^3/h.

2.6.7. Intermediate circuit system of the reactor section

The objective of the reactor section intermediate circuit is to prevent radioactive substances from getting into the service water from the heat exchangers of systems containing radioactive coolant should these lose their leaktightness. This is achieved by keeping the pressure in the intermediate circuit lower than that of the service water.

The circuit is a closed system consisting of an expansion tank, pumps, and heat exchangers as well as shut-off, safety and regulating components.

The circuit pumps supply cooling water to the heat exchangers of the reactor section systems and remove heat from them; this in turn is absorbed by the intermediate circuit heat exchangers which are cooled by service water. An expansion tank is used to keep the pumps operating smoothly and to prime, supply make-up and to compensate for changes in the volume of intermediate circuit coolant. With regard to auxiliary reactor systems which are located at higher levels and for which cooling water cannot be supplied by the main circulation pumps, of the circuit, higher-pressure pumps have been installed to deliver water to the heat exchangers of the steam separator sampler and to the reactor refuelling machine.

Periodic or continuous cleaning of the water in the reactor section intermediate circuit by special purification plants is not necessary. The quality of the intermediate circuit water is determined by sampling. When increases in the chloride content or changes in the pH value of the medium exceed the established limits, the water of the intermediate circuit is purified by exchanging the water in the system.

Systems using the intermediate circuit are the reactor flushing and shutdown cooling system, the equipment controlled leakage system, the main circulation pump sealing water coolers, the helium purification plant and heat exchangers of the chemical monitoring sampler.

2.6.8. Water regime

The reliability, safety and economics of fuel element operation and normal radiation conditions at a nuclear power plant are governed by the water-chemical conditions of the main and auxiliary circuits. The water-chemical conditions of these systems must satisfy the following main requirements:

- Reduce the amount of contamination getting into the reactor core;

- Prevent the deposition on core components of impurities contained in the water.

A neutral water regime is used in RBMKs whereby radiolysis of the water is not inhibited and no additives are introduced to correct the pH.

In accordance with All-Union Standard 95743-79, the quality of the coolant in the MFCC should meet the following requirements:

- pH value: 6.5-8.0;

- Specific conductance: not greater than 1.0 μohm/cm;

- Hardness: not more than 10 μg-equiv./kg;

- Silicon acid: not more than 100 μg/kg;

- Chloride-ions + fluoride-ions: not more than 100 μg/kg;

- Iron corrosion products: not more than 100 μg/kg;
- Copper corrosion products: not more than 20 μg/kg;
- Oxygen: 0.05-0.1 mg/kg;
- Oil: not more than 200 μg/kg.

The quality of feedwater must meet the following requirements:

- pH value: 7.0;
- Specific conductance: not more than 0.1 μohm/cm;
- Iron corrosion products: not more than 10 μg/kg;
- Oxygen: 0.03 mg/kg;

During plant operation, the water-chemical regime prescribed for the circuit must be permanently maintained and radioactive water purified before being reused and discharged. Radioactive water at nuclear power plants goes to an active water treatment system consisting of a number of plants. These plants can be divided into main and auxiliary categories.

The main active water treatment plants include the following:

- Bypass cleaning of the primary coolant circuit flushing water;
- Cleaning of cooling pond water;
- Cleaning of cooling water for the control and protection system;
- Cleaning floor drains;
- Cleaning the controlled leakage system;
- Cleaning wash-out and resin regenerating water;
- Cleaning primary coolant circuit decontaminating solutions;
- Cleaning the pressure suppression pool water.

Auxiliary plants of the active water treatment system include the following:

- Preparation of regenerating solutions;
- Perlite preparation and deposition;
- Filter loading;

- Transfer of resin to the solid and liquid waste storage tank;

- Preparation of decontaminating solutions;

- Reuse of decontaminating solutions;

- Equipment decontamination.

Apart from the MFCC flushing water bypass purification plant and the pressure suppression pool water purification plant, the installations listed above are housed in block B on axes 35-41 at levels 0.00, 6.00 and 12.50 and serve two units.

The MFCC flushing water purification bypass plants are located in block A and block B. The pressure suppression pool water purification plant and the floor drain mechanical filter preliminary purification plant are housed in the radioactivity treatment auxiliary system block.

2.6.9. Bypass purification plant for MFCC flushing water

The purpose of this plant is to purify in the bypass mode primary circuit flushing water in order to remove corrosion products and dissolved salts. This system is the main means of maintaining the quality of the circuit water, preventing deposits forming on fuel elements and ensuring the long working life of the MFCC. It enables non-volatile fission radioisotopes to be removed from the circuit, induced activity to be reduced and, most important of all, radioactive contamination of the steam and condensate-feedwater channels to be reduced. Each unit has its own independent system of this type.

The system is designed to purify 200 t of circuit water per hour. This capacity is governed by the flushing rate with regard to corrosion products and enables the circuit water characteristics stipulated in the regulations to be maintained. Under steady-state conditions the capacity of the system may even be lower. Under transient conditions at a pressure not exceeding 16 kgf/cm^2, the MFCC decontaminating solution purification plant can be used to remove corrosion products which have accumulated under steady-state conditions. This enables the rated value for iron corrosion product flushing in the MFCC to be maintained during reactor startup and shutdown cooling.

The system components are:

1. One mechanical ion exchange filter;
2. Two mixed-bed ion exchange filters;
3. One filter trap;
4. One moisture trap.

2.7. Main equipment of the unit

Reactor

The series-produced power reactor RBMK-1000 is used as the plant's steam-generating system. The reactor and its technical characteristics are described in Section 2.2.

Turbine

The K-500-65/3000 fast turbine with underground condensers is used as the mechanical drive for the AC generator TVV-500-2UZ.

The principal rated characteristics of the turbine unit are given in the table on the next page.

Steam separator

The steam separator of the RBMK-1000 is intended for obtaining dry saturated steam from the steam-and-water mixture.

The separator is a horizontal cylindrical vessel with elliptical bottoms having 400 mm openings.

The steam-and-water mixture comes to the separator through 632 short pipes of the steam-water communication line, which are located in the cylindrical part of the lower half of the separator in four rows on each side. The kinetic energy of steam-and-water mixture is quenched and crude separation of the steam takes place at the baffles inside the separator.

Thereafter, the steam passing through the immersed plate is separated in the steam space and, passing through the perforated ceiling plate, leaves by way of 14 short pipes located in the upper generatrix of the separator.

In the body of each separator there are four nipples for monitoring steam pressure and 24 nipples for connection of water gauges.

The separator is mounted on five supports, the middle one of which is fixed, while the others are of the sliding guide type.

The major assembly parts and components of the steam separator are made of the following materials:

(a) Body and bottom: 330E steel + 1C 473 B (clad steel), made by Creusot-Loire, France (for composition and properties, see Section 2);

(b) Short outlet pipes for steam: 330E steel;

No.	Characteristic	Unit	Value
1.	Maximum turbine power, MW	MW	550
2.	Net rated turbine power	BMW	510
3.	Rated fresh steam flow, including fresh steam for the second stage of the steam heater	t/h	2890
4.	Maximum fresh steam flow, indluding fresh steam for the second stage of of the steam heater	t/h	2902
5.	Initial steam pressure	kgf/cm^2, abs.	65.9
6.	Initial steam temperature	^oC	280.4
7.	Initial moisture content of steam	%	0.5
8.	Feed water heating temperature	^oC	168
9.	Rated pressure in the condenser	kgf/cm^2, abs.	0.05
10.	Type of steam distribution	reducer type	
11.	Turbine design diagram	2 LP cylinder + HP cylinder + 2 LP cylinder	
12.	Structural formula of steam regeneration diagram	5 LP heater + de-aerator	
13.	Number of steam sampling regeneration		7
14.	Frequency of rotation	rev/min	3000
15.	Turbine load for heating (intermediate circuit graph 160/80^oC)	Gcal/h	75
	<u>Technical characteristics of condenser</u>		
16.	Quantity of steam condensed (per condenser)	t/h	441.105
17.	Cooling water temperature at condenser inlet	^oC	18
18.	Number of passes of cooling water		2
19.	Cooling area	m^2	12150
20.	Condenser hydraulic resistance	m of water column	3.63

(c) Short inlet pipes for the mixture and short downcomer pipes for circuit water: clad steel - 330E steel + 1C 473 B;

(d) Separator internals: 1C 473 B steel.

The technical data on the steam separator are:

- Steam output, t/h	1450
- Saturated steam pressure, kgf/cm^2:	
working	70
rated	75
- Moisture content of steam at separator outlet, %	not exceeding 0.1
- Steam temperature, ^oC	284.5
- Feed water pressure at steam separator inlet, kgf/cm^2	71
- Feed water temperature, ^oC	165
- Flow of circuit water, t/h	9400
- Flow of steam-and-water mixture, t/h	9400
- Average steam content in steam-and-water mixture entering the separator, %	not exceeding 15.4
- Accuracy of level control in the steam separator in relation to rated value, mm	not exceeding $\pm$ 50
- Effective water margin in the separator separator for a possible level of 100 mm below the rated value, m^3	not less than 51
- Separator service life, years	30
- Weight of the steam separator: dry, t	280
in working state, t	394
during hydraulic test, t	439

- Basic dimensions of the separator:

length, mm	30 984
inner diameter of body, mm	2600
minimum wall thickness of main metal, mm	110

De-aerator

The de-aeration unit consists of a de-aerator tank and two de-aeration columns. The tank has three supports: the outer two are of the roller sliding type, which enable the de-aerator to expand during heating, and the middle one is fixed, which restricts displacement of the central part of the de-aerator in the horizontal plane and allows movement in the vertical plane. The working pressure is 6.6 kgf/cm^2 and temperature 167.5°C. The weight of the de-aerator during hydraulic tests (completely filled with water) is 204 t.

Main circulation pump

It is a centrifugal vertical single-stage pump. The shaft has a double-end packing with insignificant supply of sealing water, precluding escape of coolant into the building.

The main characteristics of the pump are:

- Delivery	8000 m^3/h
- Head	200 m of water column
- Temperature of coolant pumped	270°C
- Pressure at pump suction opening	72 kgf/cm^2
- Minimum permissible cavitation margin	23 m
- Pump shaft power	4300 kW
- Electric motor power	5500 kW

The unit consists of a tank, a pumping part and an electric motor.

The tank, which is a welded structure, is made of the 15Kh2MFA [cr2mov] steel and has an anti-corrosion surfacing inside. It constitutes the support of the pumping part and is connected to the latter through a joint which is hermetically sealed with a packing. The pumping part contains the shaft with the working wheel, the distributor, a lower hydrostatic bearing, an end bearing and an upper thrust guide bearing, which are located in the housing. The pump is so designed that the pumping part can be partly or fully replaced.

Water is supplied to the lower hydrostatic bearing from the general pressure header of pumps through a hydrocyclone.

The thrust guide bearing has a circulating lubrication system with filtration and cooling of oil from the auxiliary oil system of each pump.

The pump allows prolonged operation at a rate in the range of 5500-12000 m^3/h. The permissible heating and cooling rate of the pump is $2^oC/mm$.

Feed pump

An electric pump unit is used for supply of feed water from the de-aerators to the steam separators.

The electric pump unit is a three-stage pump with the working wheels located on one side and with a preset screw, hydraulic journal, slit-type end packings and sliding bearings with forced lubrication. Cold condensate ($t = 40^oC$) is supplied to the pump packing.

The content of mechanical impurities in the condensate should not exceed that of the feed water in weight and volume.

The main characteristics of the unit are:

-	Delivery	1650 m^3/h
-	Head	84 kgf/cm^2
-	Feed water temperature	169^oC
-	Pressure at pump suction opening	9 kgf/cm^2
-	Minimum permissible cavitation margin	15 m of water column
-	Pump shaft power	4200 kW
-	Industrial water flow	36.5 m^3/h
-	Oil flow	3.5 m^3h
-	Cold condensate flow	21 m^3h
-	Electric motor power	5000 kW

Condensate pumps

The condensate is supplied from the condensers to the de-aerators through a low-pressure heating system by means of condensate pumps of pressure rise I and II.

The condensate pump of pressure rise I is a centrifugal vertical double-barrel electric pump with a preset wheel and end packing of interchangeable types: gland and end-face types.

The basic characteristics of this unit are:

- Delivery	1500 m^3/h
- Head	12 kgf/cm^2
- Pressure at suction opening, not more than	0.2 kgf/cm^2
- Condensate temperature	up to 60^oC
- Minimum permissible cavitation margin, not less than	2.3 m of water column
- Pump shaft power	615 kW
- Condensate flow to the end-face packing	3 m^3/h
- Flow of cooling water to pump bearings	1.5 m^3/h
- Electric motor power	1000 kW
- Weight of the unit	24 560 kg

The condensate pump of pressure rise II is a centrifugal horizontal spiral type electric pump unit with a working wheel of two-sided entry. The end packings are of two interchangeable types:

- End-face packing: for continuous operation;
- Gland packing: for trial startup operations.

The basic characteristics of the unit are:

- Delivery	1500 m^3/h
- Head	240 kgf/cm^2
- Pressure at suction opening, not more than	15 kgf/cm^2
- Condensate temperature	up to 60°C
- Minimum permissible cavitation margin, not less than	22 m of water column
- Pump shaft power	1150 kW
- Electric pump power	1600 kW
- Weight of the unit	10335 kg

Pipelines

The pressure and suction headers (average diameter 800) of the multiple forced circulation loop (MFCL) and main circulation pump inlet and outlet pipes (average diameter 800) are made of the 330E carbon steel with a surfacing of the 1C 473 B steel supplied by the firm Creusot-Loire (France).

The MFCL pipes with a diameter of 300 mm are of the 08kh18N10T [cr18ni10ti] stainless steel. The pipelines of the reactor auxiliary systems are of carbon steel. The condensate feed channel pipes are of steel 20. Fresh-steam pipes are made of the 17 GS steel.

Refuelling machine

A most important requirement which the RBMK reactor must satisfy is that it should operate with a minimum number of shutdowns. For this reason, there is provision for refuelling and control of some accident situations in the operating reactor without reduction of power. This is ensured by a special refuelling machine, which carries out the following operations:

- Loading and unloading of fuel assemblies in the operating and the cooled reactor;

- Verification of free passage through the fuel channel using a gauge simulating a standard assembly;

- Hermetic sealing of the fuel channel with a plug (normal or emergency type);

- Mechanized control of some accident situations.

Refuelling in the operating reactor is carried out at the working parameters of the fuel channel.

In 24 hours the refuelling machine can carry out five operations of fuel channel unloading and loading in an operating reactor without reduction in its power and not less than 10 such operations in a shutdown reactor.

The principal parts of the machine are a crane, a container, two detachable pressure equalization chambers (one at the machine and the other in the repair area), a frame, technological equipment, a guidance system and control organs.

The working principle of the machine in the operating reactor is described below.

The refuelling machine filled with condensate at 30°C is attached to the channel to be refuelled. A pressure equal to that in the fuel channel is created in the pressure equalization chamber and the channel is unsealed. The condensate is pumped from the pressure equalization chamber into the channel at a rate of up to 1 m^3/h. The cold condensate prevents the penetration of steam and hot water from the channel into the refuelling machine. After removal of the spent fuel assembly the channel is sealed and the pressure in the pressure equalization chamber is reduced to atmospheric pressure. The machine is disconnected from the channel and sent to the location where spent fuel assemblies are unloaded.

The refuelling machine has two systems of accurate guidance into the fuel channel: optical-TV (main) and contact (stand-by) in case of loss of visibility in the steaming channel.

The optical-TV system allows visual observation of the image of the end of the channel head on a TV screen or through the eye-piece of this system and alignment of the circle of the channel head with the dotted circle of the sight by small movements of the bridge and the carriage. The contact guidance system is a pneumatic-electromechanical device, which guides the refuelling machine on to the channel axis by means of direct mechanical contact of the system with the lateral surface of the channel head.

The refuelling machine is controlled from the operator's cabin, which is behind the end wall of the central hall.

In addition, the refuelling machine cabin has a control panel for crane movement.

The central hall includes the following service areas for the machine:

1. Parking site: an area in the central hall intended for parking the machine in the period between refuelling operations.

2. A simulator rig intended for:

    - Adjustment and checking of the machine's mechanisms;

    - Filling of the pressure equalization chamber with condensate;

    - Simulation of regular refuelling;

    - Loading of fresh fuel assemblies into the pressure equalization chamber;

    - Decontamination of the inner space of the pressure equalization chamber;

  - Replacement of the inflatable collars of the connecting sleeve.

   The simulator rig has the appropriate equipment for these operations.

3. The facility that receives spent fuel assemblies is used for keeping the gauge.

4. The repair area is intended for replacement of the pressure equalization chamber if it is out of order. This area is situated in the central hall in the region of the simulator rig. A fully assembled spare pressure equalization chamber is always available in the area.

   The equipment of the safety systems is described in Section 2.9.

2.8. Control and protection system

The control and protection system (CPS) of the RBMK reactor provides:

- Control of the level of the neutron-flux-determined power of the reactor and its period under all operating regimes from 8×10^{-12} to 1.2 N_{nom};
- Startup of the reactor from the shutdown state to the required power level;
- Automatic regulating of the reactor power at the required level and changes in that level;
- Manual (from the operator's control desk) regulating of the power density distribution throughout the core and regulating of the reactivity to compensate for burnup, reflection and other effects;
- Automatic stabilization of the radial-azimuthal power density distribution in the reactor;
- Preventive protection - rapid controlled reduction of the reactor power to safe levels: protection level 1 = 50% N_{nom}, protection level 2 = 60% N_{nom};
- Emergency protection when the parameters of the reactor or generating unit change as a result of an accident (protection level 5).

The CPS (for a structural diagram see Fig. 2.11) comprises:

- Neutron flux sensors with devices (hangers) for positioning them in the reactor;
- Reactivity regulating devices (absorber rods) with drive mechanisms which move the regulating rods within the reactor channels;
- The equipment of the CPS measurement subsystem, which converts the information from the neutron flux detectors and generates discrete signals for subsequent processing in the CPS logic subsystem, as well as analog signals for the indication and recording of reactor parameters;
- The equipment of the CPS logic subsystem, which carries out the prescribed control and protection algorithms; the CPS logic system processes discrete signals from the CPS measurement and drive subsystems, from the command devices at the control desk, from the

unit's automatic process systems and from other systems; the result of this processing is the generation of a command: to move the control rods under normal and emergency conditions, to change the power level, to change the operating regimes, or to give signals;

- The equipment of the CPS drive subsystem, which controls the servo drive mechanisms in accordance with the commands from the CPS logic system;

- Output devices for indicating and recording reactor and CPS parameters at the control desk and instrumentation board;

- The CPS electrical power supply system.

2.8.1. Location of the main CPS equipment items

The hangers for the neutron flux detectors are located:

- In the tank for the water shielding around the reactor, where there are 24 ionization chamber hangers; of these, 16 have KNK-53M working-range ionization chambers and 8 have KNK-56 startup-range ionization chambers;

- During startup, 4 hangers with KNT-31 fission chambers are lowered into the reflector channels; after assured monitoring by the startup chambers has been achieved, the fission chamber hangers are withdrawn from the reflector;

- In the central openings of the fuel assemblies there are 24 in-core detectors with KTV-17 fission chambers.

All 211 CPS drive mechanisms are mounted above the CPS channels in the reactor. Their servo drives are of the channel-mounted type. The position of the CPS rods is indicated by means of a selsyn transmitter installed in the servo drive mechanism and a selsyn receiver (rod position indicator) on the CPS mimetic diagram panel in the operator's control and instrumentation board. The extreme positions of the rods are determined by cut-off switches, installed in the servo drives, which actuate the cut-off upper- and lower-end lights in the corresponding position indicators.

The equipment of the CPS drive subsystem is based in the CPS location behind the central hall wall; it consists of:

- The servo drive control panel for the manual regulating and emergency protection system and the shortened absorber rods;

- The panel for the automatic regulator servo drive control blocks, consisting of three boards with individual servo drive power control blocks;

- The control rack for the local automatic regulating system servo drives (model BA-86), containing 12 sections;

- The servo drive temperature monitoring rack.

Three boards with the automatic regulating rod synchronization system blocks are installed in the location of the non-operative part of the unit switchboard.

The CPS measurement subsystem equipment is located in the non-operative part of the unit switchboard and consists of various electronic instruments installed in 19 panels of the electronic instrumentation board, and of 2 plug-in racks holding the electronic instruments of the local automatic regulating and local emergency protection systems.

The system of output indicators and automatic recorders is installed in the operator's control desk and instrumentation board.

The CPS logic subsystem equipment is also housed in the non-operative part of the unit switchboard.

Signalling by the CPS system, which is accompanied by auditory signal and flashing lights on the signal board located in the control and instrumentation board, is carried out by devices in the CPS cupboard in the non-operative part of the unit switchboard.

The command devices (keys, buttons, etc.) which the reactor operator uses to control the CPS rods, change the reactor power, switch operating regimes, etc., are accommodated in the control desk.

2.8.2. Neutron flux monitoring

The monitoring ranges are shown in Fig. 2.12.

Neutron flux monitoring in startup regimes over the range 8×10^{-12} - $3 \times 10^{-7} N_{nom}$ is performed by four independent measurement channels with KNT-31 fission chambers. The sensitivity of the chamber to neutron flux is 0.25 pulses/1/cm^2. The secondary electronics (ISS.3M counting rate meters with KV.3M output cascades) operating from the fission chambers determine the neutron flux density on a logarithmic scale and the reactor excursion period. The information output from these channels is displayed on indicators at the control desk and recorded from a single channel selected by the operator.

At intermediate power levels in the range 3×10^{-8} - $5 \times 10^{-2} N_{nom}$ the neutron flux is monitored on the basis of signals from four KNK-56 startup current ionization chambers with an enhanced sensitivity to a neutron flux of 4×10^{13} A/1 $(cm^2.s)^{-1}$. To reduce interference from the gamma background the chamber channels are surrounded by lead screens. Additional compensation for the gamma background is achieved by regulating the negative feed voltage of the chambers' compensating electrodes. The signals from these chambers, with their secondary electronics (UZS.13 protection system amplifier with KV.2 logarithmic output cascade), determine the neutron flux density on a logarithmic scale and the reactor excursion period and generate signals to reduce the excursion period to the alarm and emergency settings (alarm signals and emergency protection system). The information output from these channels is displayed on indicators at the control desk and recorded on tape from one of the channels at the control and instrumentation board.

The discrete signals from the alarm and emergency protection systems concerning the reactor excursion period are processed in the protection system's logic circuit.

Neutron flux monitoring and recording on a linear scale in the range 8×10^{-8} - $1.0 \, N_{nom}$ is also performed by 2 KNK-53M ionization chambers with a neutron flux sensitivity of 1.45×10^{-14} A/I $(cm^2.s)^{-1}$. The secondary device in this case is a KSPV 4 high-impedance multi-range recorder.

The reactivity is measured by a ZRTA-01 reactimeter with 10 reactivity measurement ranges from 0.01 to 5 β. The reactimeter monitors the neutron flux (power) of the reactor, which is displayed on an indicator in the control desk with a scale selector and recorded by a special unit in the control board. The channel with the reactimeter operates on the signals from 2 KNK-53M ionization chambers.

2.8.3. Automatic regulating of the reactor power

The system comprises three identical sets of automatic regulators for the average reactor power. Each set consists of four ionization chambers placed around the reactor and providing information on the basis of which four automatic regulating rods are moved synchronously. The automatic regulating signal is generated by summing the relative deviations of the power from the required level, which are determined in the four individual ionization chamber measurement channels. This design principle ensures that the automatic regulator will remain functional when one ionization chamber or the instruments in one measurement channel fail.

The equipment in all three automatic regulator sets is identical.

The use of ionization chambers of different sensitivity enables these sets to work in different ranges: the low-power range from 0.5 to 10% N_{nom}

and the working-power range from 5 to 100% N_{nom}. In the low-power range there is one automatic regulator (3AR); in the working-power range there are two (1AR and 2AR).

The detector and part of each measurement channel of the automatic regulator are also used as a power overshoot protection channel: four power protection system channels in the low-power range and eight channels in the working-power range.

A structural diagram of the CPS is given in Fig. 2.11.

The detector signal in each channel is corrected by a KrT.5 current corrector. The corrected signal is compared with a reference signal from the Zd.M.5 power transducer which is common to each set of four channels. The unbalance signal is transmitted to the UZM.11 power protection amplifier and the USO.10 deviation signal amplifier. When the unbalance signal reaches the value set in the alarm and emergency protection systems for a power overshoot, the UZM.11 amplifier generates alarm and emergency signals, respectively, for further processing in the emergency protection system logic circuit. The USO.10 amplifier, whose gain can be regulated by the power transducer, produces a signal indicating relative deviation of the power from the required level. Information on the power deviation in the places monitored by the detectors is displayed on the unbalance indicator at the control desk and to some extent allows the operator to monitor the power density distortions throughout the reactor. The output signals from the USO.10 amplifiers of the four channels are summed in the USM.12 amplifier, which then transmits information about the deviation of the average power from the required level for display on the indicator at the control desk which shows when the automatic regulator is switched on. From the output terminal of the summing amplifier the signal is transmitted to the automatic regulating rods synchronization system, which synchronizes the positions of the automatic regulating rods. This synchronization system generates a relay law for power regulation. It also produces a signal indicating the average rod position for a given automatic regulator and signals indicating the deviation of the individual rods' position from the average. On the basis of the signal for the relative deviation of the average power from the required level (from the USM.12 amplifier output terminal) and of the signals for the deviation of the rod position from the average, a command is formed for the withdrawal or insertion of the automatic regulating rods into the core. These signals control the automatic regulating rod servo drives via BKS.40 power control blocks.

One of the working-range regulators is switched on, while the second is in "hot" standby. The second regulator is automatically switched on if the first regulator is switched off automatically as a result of a malfunction. In order to switch the standby regulator on smoothly, i.e. without moving the automatic regulating rods, a zero unbalance is automatically maintained at the output terminal of its summing amplifier by means of a KrU.4 automatic corrector.

The CPS ensures that identical settings are obtained from the power transducers in the working range with an accuracy no worse than 0.5% N_{nom}. The transducer settings are synchronized by a BSP.36 block and a logic circuit on the principle of stopping the transducer which has the leading setting in the direction of change of the settings.

The transducer settings are controlled by the operator from a key at the control desk. The operational rate of change in the transducer settings does not exceed:

- 0.0075% N_{nom} per second in the range 0.5-1% N_{nom};

- 0.0125% N_{nom} per second in the range 1-6% N_{nom};

- 0.15% N_{nom} per second in the range 5-20% N_{nom};

- 0.25% N_{nom} per second in the range 20-100% N_{nom}.

Under emergency conditions the settings of the working transducer settings are automatically reduced at a rate of 2% N_{nom} per second. The settings can also be lowered in an emergency by means of a button in the control desk.

The automatic regulators give a power holding accuracy for the reactor no worse than $\pm$ 1% in relation to the required level in the range 20-100% N_{nom} and no worse than $\pm$ 3% in the range 3.5-20% N_{nom}.

In addition to the monitoring of correct functioning which can be carried out on the various blocks of the system, there is also a continuous automatic monitoring of the correct functioning of the working-range automatic regulator measurement channels, including the neutron flux detectors. The BT.37 block compares the output signals of the analog channels with the signals of the channels from neighbouring detectors around the reactor. When a channel signal deviates from both its neighbours by an amount exceeding the actually possible distortions in the reactor, the channel in question is regarded by the circuit as being out of order. This type of monitoring is used at steady-state power levels and is automatically switched off in emergency conditions and transitional regimes of the generating units.

When the working-range automatic regulators are operating, the 3AR rods may be brought on-line for overcompensation of the automatic regulator which is switched on. In this case, when the rods of the switched-on automatic regulator emerge as far as the intermediate cut-off switch, corresponding to 75-100% rod insertion, the 3AR rods are automatically moved downwards, but at the intermediate cut-off switch corresponding to 25-0% rod insertion they are moved upwards.

Stabilization of the power density distribution in the reactor is achieved by the local automatic regulating and local emergency protection systems. The former is designed on the principle of independent power regulating in 12 local zones of the reactor by means of 12 regulating rods. The local automatic regulating system rods are controlled on the basis of information from two KTV.17 chambers positioned in the core around the local automatic regulating rods at a distance of 0.63 m from the rods.

The KTV.17 chamber is a current ionization chamber whose sensitive elements are coated with a ^{235}U compound and which incorporates a guard electrode to reduce loss of useful signal. The BP.119 power supply block places a negative voltage on the collecting electrode. The guard electrode receives a voltage of the same magnitude and polarity as the central collecting electrode; thus, both the guard and the collecting electrode are under an identical potential and the current losses from the collecting electrode are reduced to a minimum. The KTV.17 chamber has three sensitive elements arrayed along the height of the core.

The local automatic regulating system is switched into the automatic mode in the power range after the information received from the power density physical control system has indicated that the required power density distribution has been achieved. Before it is switched on, the output signals from the local system zones are compensated by means of the system's correction devices. Then the system, while holding the power value set before switching on in each of the 12 zones, stabilizes the power distribution in the reactor. The overall power is held by the local automatic regulating system with an accuracy no lower than that of the traditional average-power automatic regulating system. In transitional regimes, the local automatic regulating system also has considerable advantages, since it not only provides measurement and regulation of the overall power, but also smoothes out power distortions due to local perturbations in the equipment.

The local automatic regulating system is now the main system for automatic power regulating in the power range from 10 to 100% N_{nom}. The average-power automatic regulating system is used for standby and is automatically switched on when the local automatic regulating system is switched off as a result of malfunction.

The local automatic regulating system, consisting of 12 physically independent local regulators, has a high degree of "viability": when several zones of the system are switched off or malfunction, the system as a whole remains operational.

The signal from each chamber is corrected by a KT current corrector. After passing the corrector, part of the signal is transmitted to the local emergency protection system channel, where alarm and emergency signals indicating power overshoots over the required level are generated; part of the signal from each of the two chambers in the local automatic regulator zone is summed in the deviation signal amplifier, which generates the signal indicating relative deviation of the power in the local automatic regulator zone from the required level. When the values given by this unbalance are exceeded, the trigger Tg puts out signals to move the local automatic regulating rods in the corresponding zone. The speed of movement of these rods is reduced to 0.2 m/s so as not to exceed the limits laid down by the Nuclear Safety Regulations for the rate of insertion of positive reactivity when 12 rods of the local system are moved at the same time (0.7 β_{eff}/s).

There is a built-in limitation on the continuous withdrawal of the automatic regulator rods for over eight seconds.

When a power overshoot alarm signal appears in one of the channels of the local emergency protection zone, the withdrawal of the local automatic regulating rods is automatically blocked. When emergency power overshoot signals appear in both channels of the local emergency protection zone, two local emergency protection rods are lowered into this zone of the core until at least one of the emergency signals disappears. In this case the average power of the reactor is reduced by automatic lowering of the power transducer settings at their operational rate change.

The withdrawal of more than 8-10 of the manual regulating and emergency protection system or shortened absorber rods upon any malfunction (in the control desk, CPS logic, servo drive power control blocks etc.) is prevented by the "power blocking" circuit. This circuit automatically determines the number of rods in whose servo drive armature circuit a voltage for rod withdrawal is given. If this number is greater than 8-10, the circuit is automatically disconnected from the servo drive power supply source, and not a single rod can be withdrawn from the core. There are three power blocking channels which process the signals by a two-out-of-three logic.

2.8.4. Emergency protection of the reactor

The reactor is protected against emergencies by the automatic insertion into the core of all absorber rods (except for the shortened rods) from whatever initial position along the height of the core.

Twenty-four CPS rods uniformly distributed through the reactor are selected for the emergency protection mode from the total number of manual regulating and emergency protection rods by a special selector circuit installed in the CPS logic racks. When the reactor is started up, the 24 emergency protection rods are the first to be raised to the upper cut-off switches; the withdrawal from the core of any other rods is automatically prevented until the emergency system rods have been raised; the arrival in the raised position of the selected emergency protection rods is automatically verified and notified.

The reliability of the emergency protection system and the reliable functioning of the manual control system is achieved by effectively having six independent groups of 30-36 control rods each distributed uniformly through the reactor. Each CPS rod is moved by its own servo drive under the control of its individual power and logic block. The rods are connected in their six groups by the layout of the servo drive power supply and control blocks and by the design layout of the control blocks. The failure of one or even several servo drives or control blocks is not serious, since their total number is 187. Generalized reasons for the failure of several independent groups are ruled out. Since each CPS rod is surrounded in the reactor by rods of different groups, the failed rod is always surrounded by neighbouring rods in working order.

The design of the CPS drive mechanisms is such as to ensure automatic insertion of all CPS rods (except the shortened rods) into the core in a power failure. The reliability of the protection system is ensured by functional redundancy (redundant monitoring channels) for each parameter and equipment redundancy (redundant channels for logical processing of the signals).

In view of the large contribution of nuclear power plants with RBMK reactors to the general power grid, it is necessary to reduce to a minimum the outages of such plants; a differentiated approach to emergency situations in the reactor and generating unit has therefore been adopted in organizing the emergency protection system. Depending on the nature of the emergency situation, there are a number of different categories (regimes) for emergency protection:

- Emergency protection with complete shutdown of the reactor - protection level 5;

- Emergency protection acting until the emergency situation has passed - protection level 5*;

- Preventive controlled reduction of reactor power at an increased speed to safe levels: protection levels 3, 2 and 1; the safe power levels for various emergency situations and the speed of preventive power reduction are determined by calculation and confirmed experimentally.

The highest level of emergency protection is level 5, which is achieved by inserting all the CPS rods (except the shortened absorber rods) into the core up to the lower cut-off switches. This regime is entered in the following situations:

- A power overshoot of 10% N_{nom};

- A reduction in the period to 10 s;

- A drop or excess in the level in the drum separators of either half;

- A drop in the feedwater throughput;

- A pressure excess in the drum separators of either half;

- A pressure excess in the leaktight compartments, drum separators or lower water lines;

- A pressure excess in the reactor cavity;

- A fall in the level in the CPS coolant tank;

- A reduction in water throughput through the CPS channels;

- Trip of two turbogenerators, or of the only operating turbogenerator;

- Trip of three of the four operating main circulation pumps in any pump room;

- Voltage loss in the plant auxiliary power supply system, or indication of one of the protection level regimes (3, 2 or 1) without its being carried out, or order from the command units (protection level 5 button, declutching key) at the control desk and at a number of other locations in the plant.

In the event of an emergency power overshoot (power protection system) detected by the lateral ionization chambers, a partial alarm regime described as "protection level 5*" is ordered in which the insertion of the CPS rods into the core is interrupted when the original cause of the emergency has disappeared (when the power has been reduced to the appropriate level). This makes it possible to keep the unit in a power regime if the power overshoot signals have been caused by power distortions and the emergency situation can be removed by rapid partial reduction of the overall reactor power. The same is true in transitional operating regimes of the unit and in the case of significant local perturbations. The protection level 5* regime can only operate for a short time, for if the CPS rods are lowered to a significant extent into the core during a protection level 5* event, the reactor will be completely shut down just as in a normal protection level 5 regime.

The protection level 3 regime is ordered when there is an emergency load rejection by two turbogenerators, or by the only operating one.

The protection level 2 regime (reduction of N to 50%) is ordered in the following situations:

- Outage of one of two turbogenerators;

- Emergency load rejection of one of two turbogenerators.

The protection level 1 regime (lowering of N to 60%) is ordered when:

- One of the three operating main circulation pumps in any pump room is switched off;

- When the water throughput in the primary circuit falls;

- When the feedwater throughput falls;

- When the water level in the drum separators falls;

- When the group closure key for the throttle regulating valves is actuated.

In protection level 1, 2 and 3 regimes the reactor power is automatically reduced at a rate of 2% N_{nom}/s to levels of 60%, 50% and 20%, respectively, by the on-line automatic power regulating system. The emergency rate (speed) of power reduction and reactor operation stabilization at a safe power level after its reduction are obtained by automatic switching into the automatic regulating regime of the supplementary CPS rods (overcompensation and protection system) - the overcompensation regime. Signals initiating the protection level 1, 2 and 3 regimes, as well as the level 5 regime, are carried out for technical reasons in the automatic equipment system.

The generation of an emergency signal with respect to any parameter occurs upon the response of two or more detectors out of the four installed. The logic part of the emergency protection system is designed for technical reasons as two independent sets of equipment. In order to cut the protection system out during testing, there are individual keys for each parameter. When the protection is switched in, this is signalled and recorded by the "Skala" centralized control system. The protection system also has provision for signals indicating the actuation of the protection system, signals for the initial cause triggering the protection system and signals showing malfunctions in the protection system equipment.

A structural diagram of the emergency protection system for process parameters is shown in Fig. 2.13.

Both ordering the protection level 3, 2 and 1 regimes when the reactor power exceeds the safe level for such situations and carrying out the executive algorithm for levels 5, 3, 2 and 1 are the responsibility of the CPS logic circuit.

Reliability of the protection against exceeding the speed of power increase (the speed protection system) and of that against reactor power overshoot (the power protection system) is ensured in the setting-generating system for the protection level 5 regime by both functional redundancy (presence of not less than 3 monitoring channels with their sensors for each parameter) and equipment redundancy (logical processing of discrete signals by several independent channels in parallel).

A protection level 5 regime relating to speed protection is ordered when the reactor excursion period decreases to 10 s, as detected by not less than two channels out of three:

- By the startup-range speed protection system from 4×10^{-7} to 5×10^{-2} N_{nom};

- By the working-range speed protection system from 10^{-5} to 1.2 N_{nom}.

Each channel of the speed protection system consists of a UZS.13 speed protection amplifier with a KV.2 logarithmic output cascade and a KNK-56 current ionization chamber with a lead screen on the channel where it is mounted (in the startup range), and a KNK-53M current ionization chamber (in the working range).

A protection level 5 regime relating to power protection is ordered:

- By the low-power power protection system in the range from 0.005 to 0.1 N_{nom}, when a power level given by the ZM power transducer is exceeded by 0.005 N_{nom} (by 0.5% N_{nom}), as detected by not less than two out of four of the low-power protection system channels;

- By the working-range power protection system in the range from 0.06 to 1.2 N_{nom}, when the set power level is exceeded by 0.1 N_{nom} (10% N_{nom}), as detected by two out of the eight working-range power protection system channels; in this case there must be an emergency protection signal in at least one channel of each of the two groups of four working range channels.

Each working-range power protection system channel consists of:

- A UZM.11 amplifier (power protection system);
- An ionization chamber: KNK-56 in the low power range and KNK-53M in the working range;
- A BP.39 ionization chamber power supply block;
- A KrT.5 chamber current corrector.

Each group of four working-range channels has a common ZM.5 power transducer; one transducer for the low-power range and two transducers for the working range. The ionization chamber, the chamber power supply block, the chamber current corrector and the power transducer at the same time form part of the automatic regulator measurement channel of the corresponding range.

The presence of eight power protection channels in the power range with transducers distributed uniformly around the core, in conjunction with the protection system against overall power overshoot, allows the reactor to be monitored and protected against local power overshoots.

A coincidence circuit for the signals from the two independent groups (of four working-range channels each) with alternating detector locations reduces the probability of false (unjustified) reactor shutdowns when there is a malfunction of one channel or of the common element of a group - the power transducer. Dangerous failures of the emergency protection system are ruled out by the fact that the measurement and logic subsystems are designed on the principle whereby any malfunction of a block or channel is equivalent to an emergency protection signal in that channel. This design of the system makes it possible to replace any block in a single channel for repairs and preventive maintenance while the reactor is operating at power, which is particularly important for RBMK reactors with an on-load refuelling option.

The working-range power protection system is ready to respond at all times, whereas the action of the low-power system is blocked by the operator by means of a key in the control desk when the operating range of the low-power system has been passed.

A preventive power reduction is carried out by the automatic regulating system which is on-line: local automatic regulating system or 1AR or 2AR; by means of automatic lowering of the power transducer settings through protection level 3, 2 and 1 signals.

When the transducer setting is lowered, the measurement part of the automatic regulator generates power deviation (unbalance) signals. Unbalance signals for $\pm 1\%$ from the on-line automatic regulator cause the rods in that regulator to be moved, while $\pm 2.5\%$ signals actuate the overcompensation and protection system rods in protection level 3, 2 and 1 regimes. Initially, two groups of six rods each are lowered; when the rods in these groups reach the lower cut-off switches in the presence of an unbalance of +2.5%, the corresponding rods of the two next overcompensation and protection system groups are lowered. Only one group of six rods in the overcompensation and protection system is moved upwards. The ±2.5% signals are generated in the KrU.4 block of the average-power automatic regulator on the basis of a signal from the summing amplifier.

If a preventive lowering power reduction in the protection level 3, 2 and 1 regimes is ordered by an on-line local automatic regulating system, then ±2% relative unbalance signals generated in the local system trigger block cause the local emergency protection system rods to move in the corresponding zone of the local automatic regulating system. The withdrawal of the local protection system rods from the zone is allowed only after withdrawal of the local automatic regulating rods to the upper cut-off switch.

If the transducer setting of an on-line automatic regulator is not reduced at the emergency speed, or if there is no on-line automatic regulator, or if an automatic regulator has been switched off during a power reduction without another regulator being switched on, then a level 3, 2 or 1 regime is automatically converted into a protection level 5 regime.

Figure 2.11:

1. Shortened absorber rods

2. Manual regulating and emergency protection rods

3. Local automatic regulating system rods

4. Automatic regulator rods

5. Servo drives

6. Servo drive control relay contact block - shortened rods

7. Servo drive relay control contact block - manual rods

8. Servo drive control block - local automatic regulating system

9. Power control block - automatic regulator system

10. Rod synchronization system - low power automatic regulator
    - automatic regulator 1
    - automatic regulator 2

11. Individual rod control circuit

12. Shortened rods   manual rods   local automatic regulating system

13. Command devices - control desk, control and instrumentation board, standby control desk

14. Automatic process equipment

15. Signalling system

16. Protection level 5

17. Local emergency protection system

18. Local automatic regulating system on and in working order

19. Power transducer setting control

20. Low-power protection system

21. Low-power automatic regulator on and in working order

22. KNK-53M

23. KNK-56

24. KNT-31

25. KNK-53M

26. KV.2

27. KV.5M

28. BP.38

29. BP.38
ZRTA reactimeter

30. Power recorder (N)

31. Reactivity recorder (ρ)

32. Working-range speed protection system, channel 1
channel 2

33. BP.38
Working-range speed protection system, channel 3
UZS.13

34. Current indicator   Period indicator

35. Startup-range speed protection system, channel 1
channel 2
channel 3

36. BP.38   startup-range speed protection system
TsU.1   UZS.13

37. Period indicator

38. Current indicator (g)

39. Power recorder (N) on logarithmic scale

40. Period indicator

41. Speed indicator

42. BP.38M — Counting rate meter, channel 1 ISS.3M

43. Counting rate meter, channel 2
channel 3
channel 4

44. BP.39

45. Power recorder (N) at standby control desk

46. Working-range speed protection system

47. Startup-range speed protection system

48. Automatic regulator 1 on and in working order

49. Power protection system

50. Automatic regulator 2 on and in working order

51. Indicators at control desk and board

52. CPS logic

53. Indicators at control desk and board

54. Power transducer setting

55.)
56.) Unbalance, power protection amplifier

57. Unbalance, deviation signal amplifier

58. Local automatic regulating system on

59. BT_{Σ}

60. Synchronized drive block

61. Trigger block

62. BT_{Σ}

63. Power transducer

64. Synchronized drive block

65. Local automatic regulating system, zone 1, 3, 5, 7, 9, 11

66. Local automatic regulating system, zone 2, 4, 6, 8, 10, 12

67. Power protection amplifier

68. Current corrector

69. KtV.17

70. Deviation signal amplifier

71. Control key

72. Power supply block

73. Unbalance, power protection amplifier

74. Unbalance, deviation signal amplifier

75. Power transducer setting

76. BT.37

77. ZM.9

78. Low-power automatic regulator, channel 1
    USO.10
    UZM.11
    KrT.5
    BP.38

79. USM.12

80. Low-power automatic regulating system, channel 2, 3, 4

81. KNK.56, KNK.53M

82. Automatic regulator 1, channel 1, 2, 3

83. USM.12

84. Unbalance, deviation signal amplifier
    Unbalance, power protection amplifier

85. Power transducer setting

86. Unbalance, deviation signal amplifier
Unbalance, power protection amplifier

87. BT.37

88. USO.10
UZM.11
KrT.5
BP.39
Automatic regulator 1, channel 4

89. KrU.4

90. ZM.9

91. BSP.36

92. KrU.4

93. ZM.9

94. Automatic regulator 2, channel 1
USO.10
UZM.11
KrT.5
BP.39

95. Automatic regulator 2, channel 2, channel 3, channel 4

96. USM.12

97. KNK.53M

98. Alarm and emergency protection systems

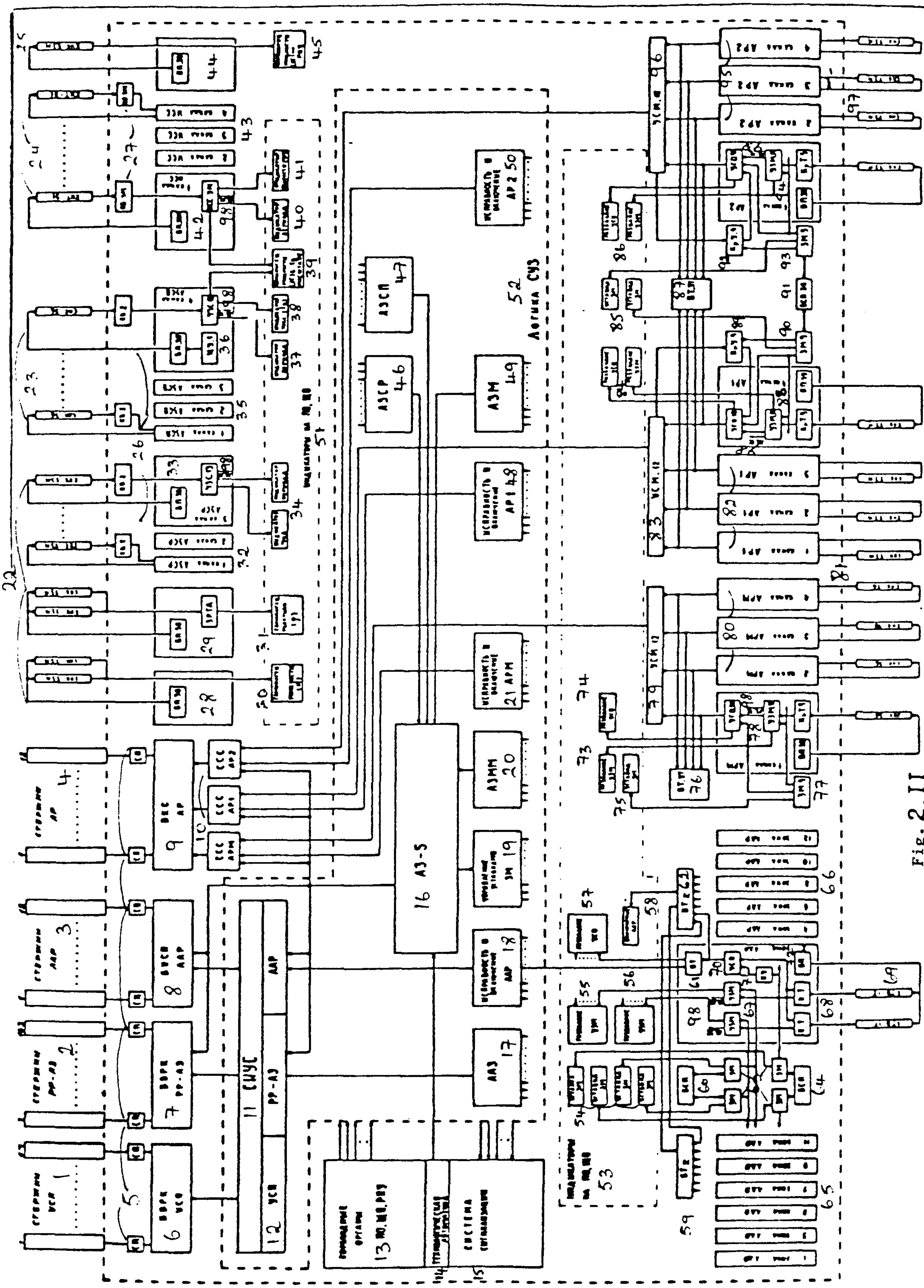
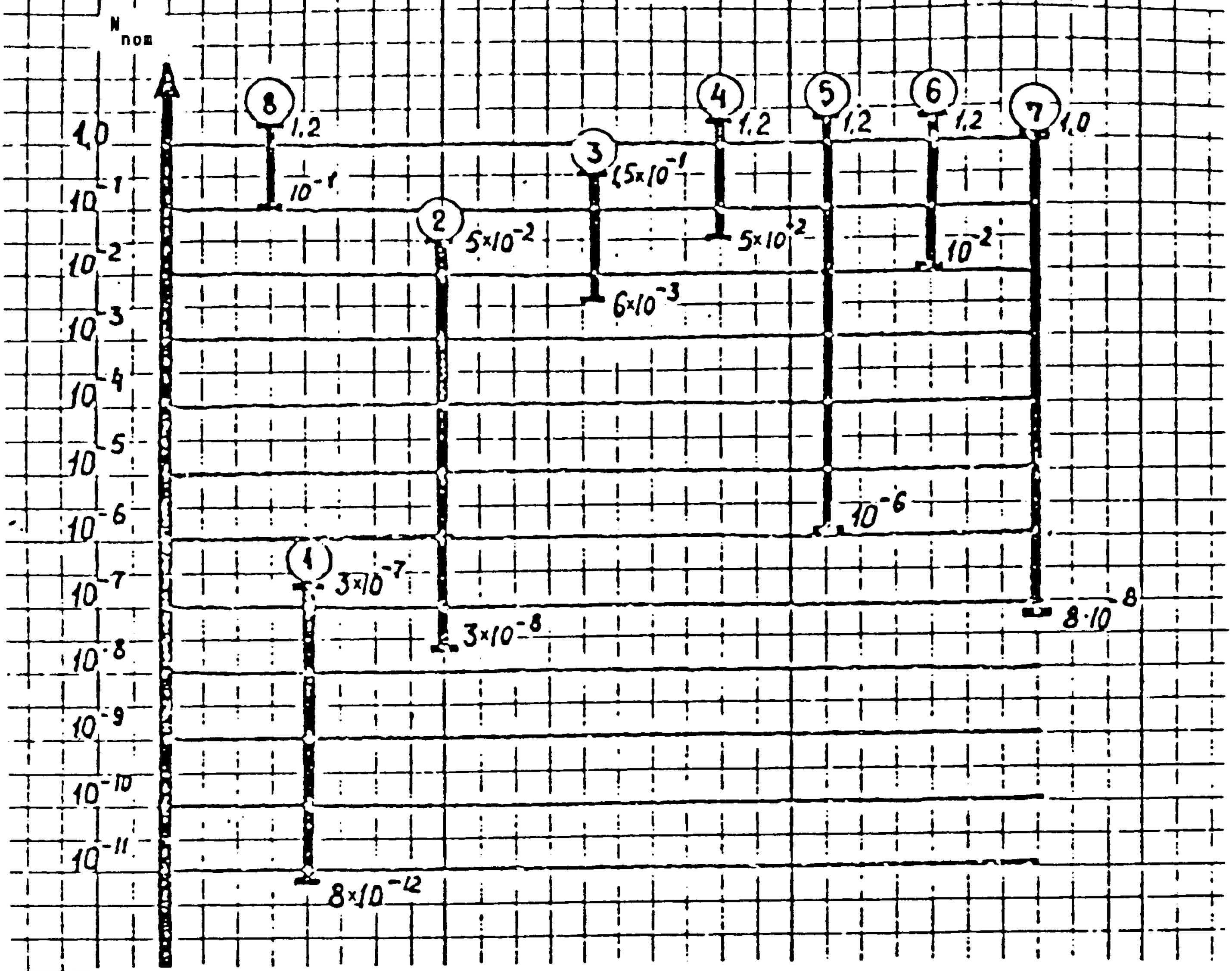


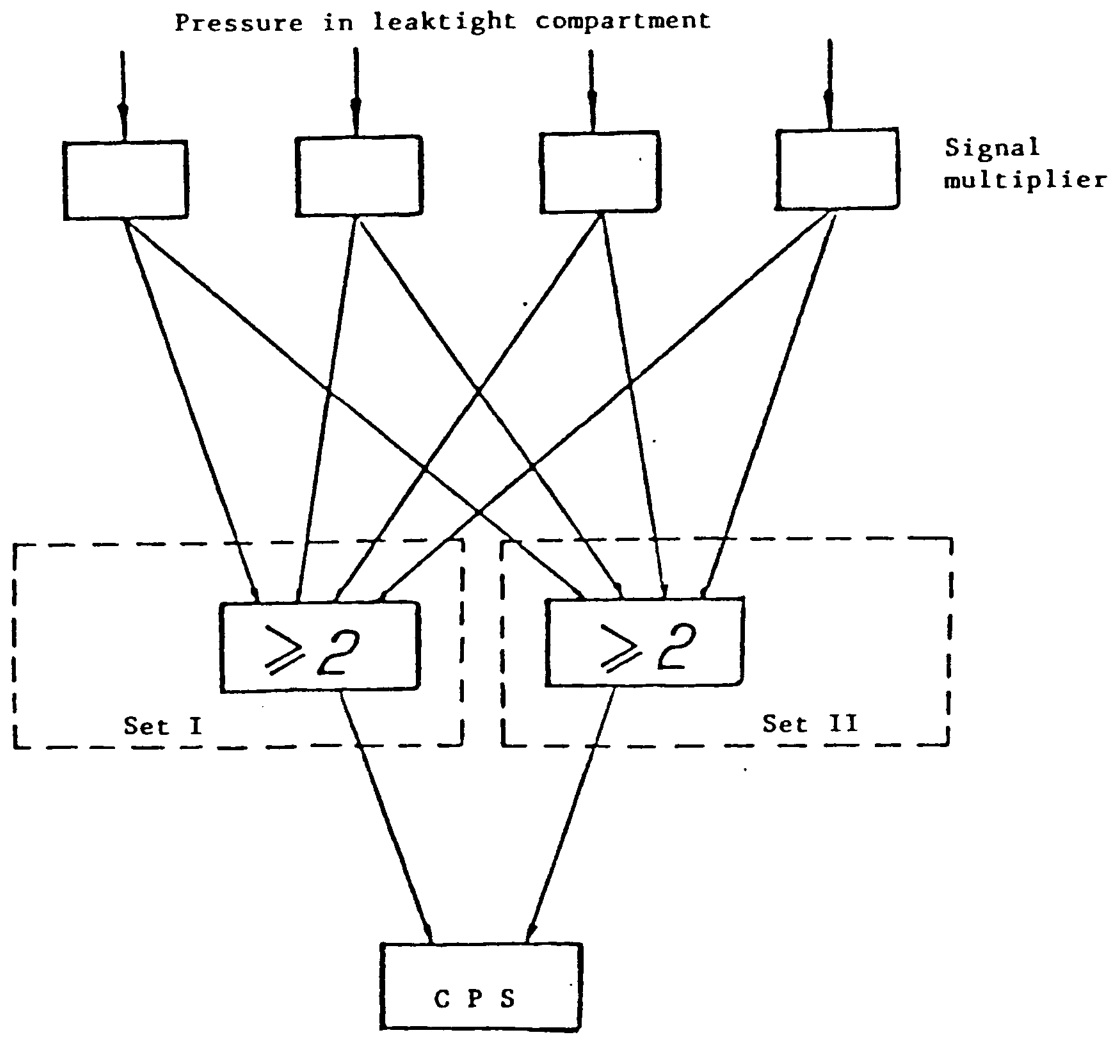
Fig. 2.II



Neutron flux monitoring ranges.

(1) Range of power monitoring on logarithmic scale by ISS.3M units with KNT-31 chambers.
(2) Range of power monitoring on logarithmic scale by UZS.13 (speed protection in startup range) units with KNK-56 chambers (with lead screen).
(3) Range of power monitoring by low-power automatic regulator using KNK-56 chambers.
(4) Range of power monitoring by automatic regulator 1 (2) using KNK-53M chambers.
(5) Range of power monitoring on logarithmic scale by UZS.13 (speed protection in working range) units with KNK-53M chambers.
(6) Range of power monitoring on linear scale by automatic recorder at standby control desk using KNK-53M chamber.
(7) Range of power monitoring on linear scale by automatic recording potentiometer at control and instrumentation board using KNK-53M chambers.
(8) Range of power monitoring by local automatic regulator channels using KtV.17 chambers.

Fig. 2.12.



Shown here is a diagram of the protection system for increasing the pressure in the leaktight compartments.
The diagrams of the protection systems for other process parameters are analogous.

Fig. 2.13 Structural diagram of protection system for process parameters.

2.9. Reactor process monitoring system

The reactor process monitoring system provides the operator with information in visual and documentary form on the values of the parameters which define the reactor's operating regime and the condition of its structural elements: the process channels, the control channels, reflector cooling, graphite stack, metal structure and so on.

The following systems relate to the process monitoring system:

- Channel-by-channel coolant flow rate monitoring in the process and control channels;
- Temperature monitoring of the graphite stack and the metal structure;
- Channel integrity monitoring from the temperature and humidity of the surrounding gas;
- Physical power density monitoring system;
- Fuel cladding failure detection;
- "Skala" central monitoring system.

Information from the PIC system is collected and processed by the "Skala" central monitoring system, and by individual instruments or independent systems (channel failure detection, physical power monitoring system, fuel cladding failure detection) for the more important parameters.

The reactor has the following monitoring points:

- Fuel channel flow rate measurement: 1661 points;
- Control channel flow rate measurement : 227 points;
- Temperature measurement of the metal structure and biological shielding: 381 points;
- Measurement of the graphite stack and plates: 46 points
- Radial and vertical power measurement: 214 points;
- Gas temperature measurement: 2044 points;
- Measurement of coolant activity: 1661 points.

2.9.1. Coolant flow rate monitoring

Flow rates in all the reactor channels are measured using tachometrical ball flowmeters. The flowmeters includes the primary ball sensor, a magnetic induction transducer and an electronic transistor unit. For measuring the flow rates in the process channels, flowmeters are used with a range up to 50 m^3/h and up to 8 m^3/h in the control channels. The flowmeters in the process channels operate in a temperature range between 20 and 285°C and pressures up to 10 MPa, and at temperatures between 20 and 80°C and pressures of 5 MPa in the control channels. The flowmeters are accurate to 1.5%, and have a positive systematic error component due to temperature, which is determined on a high-temperature flowmeter metrology rig and is automatically corrected by the "Skala" central monitoring system. The response of the flowmeters is six seconds or less.

The coolant flow rate in each process channel and in the control channels is monitored by the computing unit. Channel-by-channel flow rates are compared with norms which are set as a function of the characteristics of the channels and their position within the reactor, and which can alter when the operating conditions of the station change. When the computer unit detects a breach in the limits set for coolant flow rate, it sends an error signal to the channel mimic board and the group error mimic board, registers on the teletype the fact that an error has appeared and blocks the CPS system when the water flow rate in the control channel falls below the permitted level.

Regular diagnostic checks are carried out on the condition of the primary sensors and magnetic induction transducers, and decisions are regularly taken as to whether it is possible for them to continue in use or whether prophylactic replacement should be carried out. Diagnostic checks of the primary sensor are carried out periodically by displaying the signals from the magnetic induction transducer on an oscilloscope, determining the signal's amplitude and period ratio and comparing that ratio with a given criterion. The magnetic induction transducer undergoes periodic diagnostic testing through monitoring the resistance of its magnetic coil.

Section 2.9.2.

Temperature Monitoring

The temperatures of the graphite core and metal structures are monitored using mass-produced Chromel-Alumel cable-type thermoelectric transducers.

For monitoring the temperatures of the graphite stack and the upper and lower metal plates, an assembly of thermocouple units is used. The thermocouple assemblies are situated along the longitudinal and lateral axes of the reactor in 17 cells at the points where the corners of the graphite blocks meet. The temperature of the graphite stack is measured using 12 three-zone assemblies (of which four are in the reflector), while five two-zone assemblies measure the temperatures of the support and upper shielding slabs. A thermocouple assembly contains cable heat sensors and a support structure consisting of a biological shielding plug, graphite bushes and connecting tubes. (Fig. 2.36.). In the three-zone assemblies, the functional junctions of the temperature sensors are positioned in the central cross-section of the core and at 2800 mm below and 2700 mm above the central cross-section. The temperature sensors are manufactured from cable with an external diameter of 4.6 mm and a casing of carbonization-resistant high-nickel alloy. The cable has four cores with magnesia insulation and contains two Chromel and two Alumel thermoelectrodes moulded into a single functional junction. Each heat sensor thus contains two thermocouples with a common functional junction.

The systematic error component in the measurement of the graphite core temperature, caused by heat released internally within the thermocouple assembly elements, equals 2.2% of the measured value and is allowed for by correcting the measurement output in the "Skala" central monitoring system.

The thermal response time of the assembly is within acceptable limits at 90 seconds, and is much faster than the thermal response time of the graphite core, which is between 30 and 40 minutes. In the two-zone thermocouple assembles, the functional junctions of the thermocouples are level with the upper and lower plates.

The temperatures of the other metal structures are monitored using cable-type Chromel-Alumel heat sensors made from thermocouple cable 4 mm in diameter inside hermetic steel sleeves (Fig. 2.37). The sleeves are designed not only to protect the thermocouples, but also to act as guide elements when less accessible locations are monitored. In this way, it is also possible to replace thermocouples which have become unserviceable. The temperature monitoring of the metal structures is designed to determine their condition in stationary and transient conditions. For the upper and lower metal structures, which are more complicated, contain a larger number of structural elements and are acted on by significant thermal stresses, the maximum number of control points is 30. The temperatures are monitored of the external surfaces of the fuel channel ducts and control channels, element fins, roller supports, expansion joints and the upper and lower plates.

The reactor casing temperature is monitored at four points along a single vertical generatrix. The metal support structure is monitored at six points along a single radius. For the metal structures of the upper covering in the central hall, the temperatures of the undersides of the beam casings are monitored (8 points). In addition, the temperature of the water in the water biological shielding tanks is monitored using headed Chromel-Alumel temperature probes (16 points) (Fig 2.38).

The temperature of the water at the control channel discharges is monitored using six cable-type Chromel-Alumel temperature sensors at reference points.

One hundred and fifty-six Chromel-Copel heat sensors are used to monitor the water temperature in the reflector cooling channels.

The temperature measurement equipment used has a relatively fast response time: the thermal response time of the cable-type thermocouples is in the order of 5 seconds, and 60 seconds when they are installed in an additional protective sleeve for measuring the temperatures of load-bearing metal structures. Instrument error is in the order of 2% of the temperature measurement range.

Temperature information is periodically printed by the "Skala" central monitoring system, and it is also possible to call up any of the parameters using the call-up feature on the numerical display unit of the "Skala" central monitoring system and on the redundant instrument set.

2.9.3. Process channel integrity monitoring

The process channel integrity monitoring system is part of the reactor ventilation system and, in general, is designed to carry out the following functions:

Detection of non-hermetic reactor channels;

Containment of the spread of humidity from the damaged channel;

Ventilation of the reactor space.

The main outlines of the process channel integrity monitoring system are shown in Fig. 2.3.9. The process channel integrity monitoring system for RBMK reactors relies on measuring the parameters of the gas (temperature and humidity) as it is pumped round the graphite stack of the reactor through the gas ducts formed by the stack and the process channels. In this way, there is individual monitoring of the temperature of the gas drawn off and group monitoring of its humidity. The gas circulates through the graphite core from bottom to top. Temperatures are measured by short Chromel-Copel heat sensors installed at each process channel integrity monitoring impulse tube. Temperature signal information from the thermocouples is passed for processing to the "Skala" system which, when it has detected the channel (or group of channels) with a temperature overshoot, sends a signal to the channel mimic boards on the reactor unit's control panels.

Humidity monitoring in each of the 26 zones and detection of zones where there is excess humidity are carried out using humidity indicators. A humidity indicator consists of eight humidity sensors and one eight-channel humidity measuring unit. The sensing element of the humidity sensor is of the sorption type and is designed to operate at temperatures of between 40 and 100°C and relative humidities from 50 to 100%. The relative humidity unit gives readings in steps of 5%. When the humidity indicator operates, it sends signals to the "Skala" system which are reflected on the humidity board on the reactor unit control panel. The relative humidity of the gas in the reactor space is continuously monitored within a range of 0 to 100% by means of a hygrometer, which consists of a sorption-type primary sensor, measuring unit and recorder. In order to increase the reliability of the system for determining process and CPS channel integrity, there is a system for draining the sylphon bellows cavities of the CPS channels and measuring the temperatures of the drainage pipes. The humidity which appears in the reactor space when a loss of integrity occurs evaporates and, as it condenses on the nearest "cold" control channels, settles in part in the sylphon bellows cavity and from there passes to the lower part of the duct into the drain pipe. When this occurs, the temperature of the drain pipe, which passes through the lower water communication line housing and is at the same temperature when there is no flow, decreases, and this is picked up by a thermocouple. The region in which the search for the burst is carried out is defined by the thermocouple readings, which give temperature values approximately 100°C below the temperature of the lower water communication line housing. The temperatures of 126 control channel drainage ducts are measured and processed through the "Skala" central monitoring system for periodic print-out.

2.9.4. Fuel element cladding integrity monitoring

The physics and design characteristics of an RBMK power station - channel-type reactor with boiling coolant - determine the structure of the system for identifying and locating fuel assemblies with burst fuel elements within the core while the reactor is in operation. The physical system for monitoring fuel cladding integrity comprises:

- A sampling system for monitoring the activity of gaseous fission products in the separated steam in each drum separator; this makes it possible for the condition of the fuel elements in a quarter of the fuel assemblies in the core to be observed continuously;

- A non-sampling channel-by-channel system for periodically monitoring total gamma activity of the coolant in each steam-water communication line, the secondary part of which is electronic and compensates for the background component of the signal in order to distinguish the gamma activity of fission products emanating from burst fuel elements.

2.9.5. Monitoring of the multiple forced circulation circuit

The physical monitoring system for the primary coolant (multiple forced circulation) circuit is designed to ascertain the condition and operating modes of its basic elements: the drum separators, main circulation pumps and the suction and pressure headers. It includes monitoring of drum separator level and pressure, drum separator metal temperature, equalizing tank temperatures, main circulating pump flow rates and drum separator steam out and feed water in flow rates.

Platinum resistance thermometers are used in the suction headers to measure the coolant temperature in order to determine the cavitation margin. Pressure is monitored in the main circulation pump suction and pressure headers. The flow rate through the main circulating pumps is measured using a differential manometer, for which the pressure drop is created on the constriction principle. The primary coolant circuit parameters are monitored by the "Skala" central monitoring system.

2.9.6. The "Skala" central monitoring system

The "Skala" computerized central monitoring system is designed to carry out monitoring of the processes in the basic equipment of RBMK-1000 nuclear power station units, and to provide calculations and logic analysis of the units' process conditions in finished form for the operating staff. A diagram of the "Skala" system's structure and of its links with external systems (CPS, processed channel integrity monitoring system, physical monitoring system and so on) is shown in Fig. 2.40. The basis of the system is a two-processor computing unit which is designed to be able to capture information from the source and transmit it to the output devices using either of the two processors (functional back-up). Information on the condition of the unit coming from the process monitoring system sensors through the individual signalling channels or through the computer unit and is passed by the operator to the display and digital instruments, the mimic diagram, channel mimic board and the individual error board and is also registered on recorders, teletypes and high-speed printers. The information the operator needs to work is accessed in the "Skala" system by means of a number of input-output devices. The operation of the system as a whole is organized by the system control unit. The basic technical features of the "Skala" system are as follows:

(1) Number of monitoring signals:

Analogue signals	7200
Discrete signals	6500

Signals are accepted from:

Chromel-aluminium [Alumel?] and Chromel-Copel heat sensors;

Platinum and copper resistance thermometers;

Tachymetric ball flowmeters;

Sensors: diffusion manometers with standardized outputs of 0-5 mA, selsyns, on-off sensors, independent physical power monitoring system and the average control rod position signals.

(2) Monitoring periods:

Mass parameters: 1-5 min.

Calculated parameters: 30 min.

(3) Functions:

Measurement of the parameters input through the group and individual information capture channels, and also, when commanded by the staff, on the group, individual and digital display instruments.

Signalling on the mimic diagram, processe channel mimic board, group error board and CPS mimic board of the conditions of mechanisms, fittings, generator sets, process parameters and correct equipment function.

Monitoring of directly measured errors and errors in calculated parameters with results shown on output devices and also recorded.
Process calculations on a periodic basis and on request.
Print-out of any of the measured and calculated process parameters on a periodic basis and on request, with record made of the run-up to and development of accident situations.

(4) Operation time to failure:

- Monitoring functions: 1×10^4 hours;

- Calculating functions: 2×10^3 hours.

(5) Electric power required: 95 kW.

2.9.7. System for physical monitoring of the power density distribution

Purpose and structure of the system

This system is intended to measure and record signals from the power density monitors which characterize the energy release in the reactor. By initially processing the signals coming in from the monitors and then comparing them with preset maximum values, the system makes recommendations to the reactor operator for regulation of the power density distribution. The light and sound signals emitted by the system are used for operative flattening of the power density distribution. For additional correction of the density distribution use is made of the link between the physical monitoring system and the "Skala" computer system, where on the basis of the signals from the monitors, results of the physical calculation and other requisite information there is periodic calculation and recording of the power and the maximum permissible power margin for each fuel assembly, as well as calculation of other parameters for different assemblies and the reactor as a whole.

For operative monitoring of the thermal power of the reactor between the minimum verifiable level and the nominal level, use is made of the monitoring system's automatically recording potentiometer, which records the total monitor current over the reactor radius and has a scale graduated in megawatts (0-4000 MW). A doubling instrument is used for the same purpose.

According to its functional purpose the system for physical monitoring of the power density distribution is divided into three systems: a system for physical monitoring of the radial power distribution, a system for physical monitoring of the vertical power distribution, and an auxiliary system for periodically checking the monitors.

The radial power density distribution monitoring system is intended for measurement and recording of signals from 130 in-core detectors monitoring the power density over the reactor radius, for preliminary processing of the signals, for transmitting them to the "Skala" computer system, for comparing the signals with three set levels and emitting light and sound signals indicating that the power density values in the fuel assemblies fitted with monitors have overshot the prescribed limits. The maximum power of the assemblies with radial monitors is determined by the "Skala" system computer on the basis of the requirement of flattening the power density distribution and ensuring the safety of the given and neighbouring assemblies.

The vertical monitoring system is designed to measure and record signals from 12 in-core seven-section power monitors in a vertical direction, for preliminary processing of the signals, transmission of them to the "Skala" computer, comparison of the signals with three set values and emission of light and sound signals indicating that the local power density in neighbouring assemblies with monitors has overshot the prescribed limits. The maximum values of the signals from different sections of the vertical monitor are determined by the "Skala" computer on the basis of the requirement that there should be stabilization of the axial power density distributions and safe operation of the assemblies without overshooting the maximum local thermal loads.

Periodic checking of the monitors is intended for routine calibration of the sensitivity of the radial and vertical monitors, as well as for determining the error involved in calculating the fuel assembly power with the "Skala" computer system.

The system for physical monitoring of the radial power density distribution includes:

130 detectors for monitoring the radial power density of the reactor;

measuring devices of the power density monitoring equipment;

automatic recording potentiometer (system power recorder) and doubling indicating instrument;

scanning instruments of the power monitoring equipment.

The system for physical monitoring of the vertical power density distribution includes:

12 seven-section detectors for monitoring the vertical power density of the reactor;

measuring device of the power density monitoring equipment;

scanning instruments of the monitoring equipment.

The measuring devices have a block-type structure and are serviced by a common multichannel recorder belonging to the power monitoring equipment, which records on a numerical printing device the detector signals exceeding the relevant maximum levels, time of occurrence and the detector co-ordinates.

The layout of the radial and vertical monitors and the CPS control rods ensuring monitoring and control of the power density distribution in the reactor is shown in Fig. 2.1.a. For purposes of monitoring and controlling the power distribution the reactor design makes provision for the arrangement of ~ 310 suspensions and assemblies with a dry central supporting tube (casing). Of these 130 are intended to hold the radial monitors and 48 of them to hold the local emergency protection and control systems, and at least 130 remain free (assemblies for scanning the power) and are used for periodic calibration of the radial monitor sensitivity. Assemblies of this type are placed alongside the assemblies fitted with radial monitors.

The measuring devices and multichannel recorder are positioned in the non-operative part of the unit control panel. The devices for light signals indicating that the radial and vertical signals have deviated from the norm are located on the CPS-physical monitoring system mimetic board on the operator's control panel. The unit that switches on the light signals on the mimetic board in response to commands arriving from the measuring devices of the power monitoring equipment is part of the "Skala" system.

The power recorder of the power density physical monitoring system is located on the reactor operator's control panel, while the scanning instrument doubling its readings is on the control desk. The scanning instruments of the power monitoring equipment are located on the control panel.

The monitor verification system includes:

calibration detectors of the radial power monitor type;

tri-axial calibration fission chambers;

annular ionization chamber;

measuring equipment of the monitor verification system.

The installation and removal of the calibration detectors from the reactor is remotely controlled by means of the crane in the central hall. The measuring equipment is situated in the central hall or in the crane operator's room in the central hall.

Radial power density monitors

The radial power density monitors are enclosed in dry central supporting zirconium cases with an internal diameter in the core of 6.5 mm and arranged along the axis of the assembly (all the way along). The design of the radial monitor is shown in Fig. 2.41. It consists of a sensitive element in a leaktight casing (4) made of corrosion-resistant steel with an outside diameter of 6 mm; a leaktight join (2), a cable (3) inside a leaktight protective sheath, and the elements of the biological shielding (1). The casing is filled with inert gas (argon) to protect the envelope of the sensitive element against corrosion.

The overall length of the monitor is 16 167 mm, and the length of the sensitive element is 8500 mm.

As the sensitive element, use is made of a beta emission detector with a silver emitter (5). It takes the form of a high-temperature cable (type KDMS(S)) with an outer diameter of 3 mm, central silver filament 0.65 mm in diameter, envelope made of corrosion-resistant steel and insulation made of 0.8 mm thick magnesium oxide. The cable is manufactured using the industrial technology normally adopted for high-temperature cables and thermocouples. Its sensitivity is $\sim 5 \times 10^{-20}$ A.cm^2.s/n per metre of length. The maximum current of the radial monitor at nominal reactor power is about 15°μA. The maximum temperature of the sensitive element as a result of radiation heat-up is greater than the coolant temperature in the assembly and amounts to ~ 350°C.

The mean ratio between the power of a non-spent assembly with a radial monitor and the current of a non-spent radial monitor is 0.2 MW/μA. Variations in this ratio for each radial monitor due to its individual sensitivity and neutron spectrum are taken into account by periodically calibrating the monitor during operation of the reactor. The mean square spread of the radial monitor's sensitivity to the neutron flux is, according to experimental data, 4%. At the same time, the mean square spread of the sensitivity to the power of the fuel assembly is greater and amounts to 6%, which is explained by the difference in neutron spectrum in the different assemblies with radial monitors. This effect may be taken into account on the basis of the measured distributions over the reactor of the spectral characteristics, but in the practice of operating RBMK reactors the method adopted has been direct periodic calibration of each radial monitor from the power of the fuel assembly by scanning the assemblies with a hollow central casing in an operating reactor, using beta emission detectors of the radial monitor type or tri-axial fission chambers.

The theoretical-experimental dependence of the radial monitor's sensitivity ξ_D to the neutron flux on the integral radial monitor current I_i is an extremely efficient measure of the neutron fluence as a weak function of the neutron spectrum and temperature of the monitor.

The ratio ξ_{TD} of the power of an assembly with monitor to the neutron flux density at the site of the monitor is a function of the integral energy yield of the assembly Ei.

When the reactor is in operation the power of the assembly with monitor is calculated from the following equation:

$$W_i = K_{\text{гр}\,i} \cdot \xi_D(I_i) \cdot \xi_{TD}(E_i) \cdot J_i , \qquad (1)$$

where: K_{rpi} is the individual calibration coefficient of the i-th radial monitor;
J_i is the current of the i-th radial monitor.

When the monitor is replaced in a spent fuel assembly, there is a change of the present value of ξ_D in the "Skala" computer system with variation in the integral radial monitor current in the computer memory. Design calculations showed that the error associated with the use of Eq.(1) when replacing a spent monitor by a fresh one was not more than 1%. Generalized experience gained in operating RBMK reactors shows that the indicated calculation of the burnup of the radial monitor and assemblies does not create errors of more than 1% in determining the power density of the reactor.

The radial power density monitor (without the cabling) is installed in the central casing of the fuel assembly with the aid of the central hall crane. The cables are laid when the reactor is being assembled. Replacement of failed monitors is carried out by remote control either with the reactor shut down or in operation after the monitors have been disconnected from their cables. The laying of new cables is only possible with the reactor shut down.

The radial power density monitor is designed to operate throughout the service life of the fuel assembly. Experience with RBMK reactors has shown the high reliability of these monitors. The mean time to failure, according to operational data, is 9.7×10^4 h.

The radial power density monitor is considered to have failed in the following instances:

- The emitter breaks off and as a result there is no current at the monitor connecting joint;

- The monitor readings in the "Skala" computer system are rejected when making operative calculations with the "Prizma" program;

- There is a drop in the monitor's sensitivity, allowing for burn-up, of more than 15% between two calibrations;

- There are rapid fluctuations in the monitor's signal that are not confirmed by the neighbouring monitor readings;

- The resistance of the monitor's insulation drops below 100 kohm.

Vertical power density monitors

To monitor the vertical power density distribution in the reactor use is made of 12 sets of monitors uniformly arranged in the core in the area of the radial distribution plateau. Each set contains seven beta emission detectors, with a silver emitter, arranged in a uniform vertical pattern, and made like the radial monitors, in the form of a cable (type KNMS)(S). Each sensitive element (section) of the vertical monitor is a spiral made from cable with an outside diameter of 62 mm and height of 105 mm. The overall length of cable in the spiral is 2.6 m. The centres of the top and bottom sections are shifted 500 mm towards the centre with respect to the core boundaries.

The design of the vertical monitor is shown in Fig. 2.42. Seven sensitive elements are housed in a dry leaktight casing made of corrosion-resistant steel, which is mounted in a channel similar to the one intended to hold the control rods. On the outside the casing is cooled by stream of flowing water 7 mm thick, with a temperature at the reactor outlet of not more than 70°C. A central tube running along the axis of the casing is intended for periodic calibration of the sensitivity of the monitor sections with the aid of a tri-axial fission chamber moving vertically up the monitor. In the non-working position the fission chamber may be left in the central tube of the monitor since its sensitive volume will then lie below the lower boundary of the core.

The sensitive elements are joined by high-temperture cables (type KNMS (S)) to leaktight joints at the point where the casing comes out in the central hall. The same cable, enclosed in a protective sheath of corrosion-resistant steel, is used to connect the sensitive elements via the joints to an outside terminal block. The cable route is designed to ensure the best immunity from interference. It is not permitted, for example, to lay vertical monitor cabling together with the cables feeding the control rod drives.

The inside of the casing is filled with an argon-helium mixture to reduce radiation heat-up of the vertical monitors; the maximum temperature of the sensitive elements does not therefore exceed 150°C.

To protect the space above the reactor from ionizing radiation coming from the core and steam-water communication lines, the vertical power monitor is fitted with two steel shielding plugs located at the top of the monitor casing. Furthermore, the top of the monitor makes provision for the mounting of a special protective cap, the function of which is to protect the monitor joints from mechanical damage at the same time.

When the reactor is working at nominal power the currents from the various sensitive sections of the vertical monitor may vary between a few μA and 15 μA, depending on their position in the core.

The design of the set of monitors and the channel makes it possible to replace them while the reactor is in operation as well as when it is shut down. This is done by remote control using the crane of the central hall. The cables are laid while the reactor is being assembled, and replacement is only possible with the reactor shut down.

While the reactor is in operation, the signal from each section of the vertical monitor permits calculation of the neutron flux density at its point of location:

$$(nv_0)_{ij} = K_{rg\,ij} \cdot \xi_D(I_{ij}) \cdot J_{ij} \,, \tag{2}$$

where: n is the neutron density;

v_o = 2200 m/s;

K_{rgij} is the individul calibration coefficient of the i-th section of the j-th vertical monitor;

$\xi_D(I_{ij})$ is a correction for emitter burn-up, identical to the one used in the radial monitor, and depending on the integral current of the i-th section of the j-th vertical monitor I_{ij};

J_{ij} is the current of the i-th section of the j-th vertical monitor.

The assumed lifetime of a vertical power monitor is 2.5 years. Experience gained in operating RBMK reactors shows the satisfactory reliability of the monitor. The mean time to failure derived from operational data is 4.0 x 10^4 h.

The vertical power monitor is considered to have failed if there are two defective sections side by side in it or if any three sections are defective.

A section of the monitor is considered to have failed in the following instances:

There is a drop in the sensitivity of the section, allowing for burnup, of more than 15% between two calibrations;

There are rapid fluctuations in the section signal not confirmed by the readings of neighbouring sections;

The resistance of the section insulation drops below 100 kohm.

Power density monitoring equipment

Purpose and composition of equipment

The power density monitoring equipment is designed in the form of four racks on which are placed the main functional units and systems of control and monitoring. The equipment also consists of scanning instruments for operative monitoring of the power density distribution in the reactor; a recorder and an indicating instrument for monitoring the thermal power; signal setting control desk, switching devices and numerical indicators showing the co-ordinates of the monitor called on for a signal to the scanning instruments, and rigs for testing and adjusting the main functional units.

In terms of its purpose the power density monitoring equipment can be divided into two parts: the equipment of the system for physical monitoring of the radial power density distribution and the equipment for the system for physical monitoring of the vertical power density. The principal functional units in both parts are different in design: in the first case there are two measuring devices, and in the second only one such device. All three measuring devices are served by one multi-channel recorder which records on a numerical printer the values of the monitor output signals exceeding the "emergency" level, the time of occurrence and the monitor's co-ordinates.

The measuring devices perform the main functions of shaping the information signals and operatively monitoring the power density distribution in the reactor. These functions are carried out by separate circuits for each radial and vertical monitor (monitoring lines). The measuring devices of the system for physical monitoring of the radial power density process signals received from 130 radial detectors mounted in the reactor, and there is a possibility of switching in another 14 radial monitors (in all 144 monitoring lines). The measuring device of the vertical power density monitoring system processes signals from 12 seven-section vertical monitors (84 monitoring lines), but has the possibility of processing signals from 12 eight-section vertical monitors (in all 96 monitoring lines).

The radial and vertical monitor signals reach the inputs of the individual amplifiers with adjustable negative feedback, which then transform the monitor currents (input signals) into dc voltage signals (output signals). These signals are fed to the inputs of the device for operative monitoring of the power density distribution (signalling device), to the inputs of the switching device (scanning instruments), to the unit averaging the monitor signals, to the "Skala" computer system and, through the contacts of the actuator relays, to the multi-channel recorder.

The signalling device compares the radial and vertical monitor output signals with the set limit values of the signals. The comparison is made at three levels (signal thresholds) termed the "undershoot", "warning" and "emergency" levels. When the radial and vertical monitor output signals deviate from the given levels, the relevant signal system is triggered in the power density monitoring equipment and colour signals light up on the CPS-power density monitoring system mimetic board. A green light means that the radial and vertical monitor signal is equal to or less than the "undershoot" level. Absence of a light means that the signal is greater than the "undershoot" level, but less than the "warning" level, i.e. it lies within the accepted limits. A red light means that the radial or vertical monitor signal is equal to or greater than the "warning" level, but has not yet reached the emergency level, while a blinking red light means that the signal has attained or overshot the "emergency" level. In the latter case a sound signal is given along with the blinking light.

To present the information more clearly, the CPS-density monitoring system mimetic board takes the form of a mnemonic representation of a horizontal cross-section of the reactor in which there are indicators showing the position of the control rods and the signal elements of the radial and vertical monitors. The arrangement of the position indicators and signalling elements on the mimetic board matches the arrangement of the control rods and the radial and vertical monitors in a radial plane of the reactor. The systems for displaying the monitoring information enable the operator to see clearly the area in which there has been deviation of the monitor signal from the set level, to determine from the type of signal how to actuate the control rods (upwards or downwards) in order to eliminate the deviation, and to select the rod required for that purpose.

Furthermore, the radial equipment provides for the possibility of altering, within $\pm 15\%$, the "undershoot" and "warning" levels for all the radial monitors at the same time from the operator's control desk; this enables operators to detect promptly of the areas where the power density is close to maximum.

The radial monitoring equipment provides for two modes of operation: comparison of the radial monitor output signals with the floating levels (thresholds) of the "undershoot" and "warning" signals and with the fixed levels (thresholds) of the "emergency" signal; and a mode by which the radial monitor output signals are compared with the fixed thresholds for all three levels.

In the first mode of operation the "undershoot" and "warning" signal levels for each radial monitor vary in proportion to the arithmetic mean of the radial monitor ouput signals, i.e. in proportion to the present power of the reactor, while the "emergency " level is fixed at a level selected on the basis of operational requirements. When the arithmetic mean of the radial output signals attains a pre-set maximum (set level of power at the given stage of reactor operation), the "undershoot" and "warning" signal thresholds are fixed (limited) and the radial power density monitoring equipment automatically switches to the second operational mode.

The vertical monitoring equipment operates in a mode where it compares the output signals of the vertical monitor sections with the floating thresholds for all three signal levels, i.e. it only carries out relative monitoring of the vertical power density of the reactor. Here, the signal thresholds for each vertical monitor vary in proportion to the arithmetic mean of the signals from the sections of the given vertical monitor.

To check the working capacity of the radial and vertical monitors the equipment makes provision for devices by which to determine the resistance of the insulation of any detector by switching an additional resistor R_{gob} (100 kohm) into the input circuit of an individual amplifier of the monitoring line. From the relative decrease in monitor output signal the resistance of the monitor insulation can be computed:

$$R = R_{go\delta} \frac{U'/U}{1 - U'/U}, \qquad (3)$$

where: v is the monitor output signal up to the moment when the R_{gob} is switched in;
v' is the monitor output signal after the R_{gob} has been switched in.

Principal specifications of the equipment

The maximum value of the signals at the output of the individual amplifiers of the monitoring line is 5 V, while the polarity of the signals is negative. The range of adjustment of the coefficients for transformation of individual amplifiers ranges from 0.26 to 0.78 V/µA. The input amplifier resistance is not more than 100 ohm.

The principal relative error in transforming the input signals (in percentage) is not more than:

$$\delta = \pm \left[0{,}5 + 0{,}14 \left(\frac{J_{\text{макс}}}{J} - 1 \right) \right], \qquad (4)$$

where: J_{max} = 19 µA is the maximum input current;
J is the present value of the input current (µA).

The permissible capacitance of the monitor together with the cabling should not exceed 0.05 µF.

The permissible resistance of the load on the output terminals of the amplifiers is at least 2 kohm.

The equipment ensures that eight monitor signals are displayed simultaneously on indicating instruments (M1830A): one of the radial monitor output signals and seven output signals from the sections of the vertical monitor selected. Calling for signals on the indicating instruments is effected by switching devices which store the radial and vertical monitor address in a code for the co-ordinate grid of the reactor channels.

The equipment shapes a signal equal to the arithmetic mean of the radial monitor output signals (reactor power signal) and displays it on an indicating instrument (0-100 µA scale) and on a recording instrument (0-100 mV scale; scale run-through time 10 sec). Provision is made for the introduction of a correction to the arithmetic mean of the signals that takes into account the absence of signals at the amplifier input. The mean relative error in shaping the averaged signal, given signals of at least 2.5 V at the individual amplifier output and a number of average signals ranging from 70 to 130, does not exceed ±0.5%

The equipment shapes four power signals for the quadrants of the reactor equal to the arithmetic mean of the radial monitor output signals of the corresponding quadrant; these are displayed on four indicating instruments.

The equipment shapes for each vertical monitor a signal equal to the arithmetic mean of the output signals of its sections. Provision is made for the introduction of a correction to the arithmetic mean of the signals that takes into account the absence of signals at the amplifier input. The principal relative error involved in averaging, given signals of at least 2.5 V and a number of average signals between 4 and 8, does not exceed ±0.5%.

To monitor the radial power density distribution, a fixed "emergency" signal threshold is established for each monitoring line:

$$U_{\text{ав}i} = \frac{1}{K_{\phi i}} \cdot \frac{\delta_{\text{ав}}}{100} \cdot U_{100}, \tag{5}$$

and a "warning" signal threshold:

$$U_{\text{пр}i} = \frac{1}{K_{\phi i}} \cdot \delta_{\text{пр}} \cdot \overline{U}, \tag{6}$$

and an "undershoot" signal threshold, which vary in proportion to the reactor power:

$$U_{\text{зон}i} = \frac{1}{K_{\phi i}} \delta_{\text{зн}} \cdot \overline{U}, \tag{7}$$

where: $K_{\phi i}$ is the coefficient of transmission for the comparator unit amplifier for the i-th monitoring line and is fixed smoothly within the limits 0.6-2.5;
δ_{ab} is the coefficient of transmission for the "emergency level %" divisor and is established discretely in percentage of the nominal power between 0 and 100%;
υ_{100} is the voltage corresponding to nominal power level and is regulated smoothly from 1 and 5 V;
δ_{np} is the relative "warning" signal level and is established smoothly between 0.65 and 1.25;
δ_{3aH} is the relative "undershoot" signal level is established smoothly between 0.65 and 1.25;
$\bar{\upsilon}$ is the arithmetic mean of the radial monitor output signals.

The equipment provides for the possibility of simultaneous, remote-control alteration of the "warning" and "undershoot" signal thresholds by steps of 1%, up to ±15% of the set values.

Provision is made for limiting the "warning" and "undershoot" signal thresholds to the levels:

$$\left(U_{np\,i}\right)_{\text{макс}} = \frac{1}{K_{\phi i}} \cdot \delta_{np} \cdot U_{\text{огр}\,np}, \tag{8}$$

$$\left(U_{\text{зан}\,i}\right)_{\text{макс}} = \frac{1}{K_{\phi i}} \cdot \delta_{\text{зан}} \cdot U_{\text{огр.зан}}. \tag{9}$$

The limiting levels $\upsilon_{\text{orp.np}}$ (warning) and $\upsilon_{\text{orp.3aH}}$ (undershoot) are set smoothly within the limits of 1-5 V.

To monitor the vertical power density distribution, in each monitoring line of the j-th vertical monitor signal thresholds are set which vary in proportion to the arithmetic mean of the output signals from the sections of that monitor:

"Emergency" - $$U_{ae\,ij} = \frac{1}{K_{\phi ij}} \cdot \delta_{ae} \overline{U_j}, \tag{10}$$

"Warning" - $$U_{np\,ij} = \frac{1}{K_{\phi ij}} \cdot \delta_{np} \cdot \overline{U_j}, \tag{11}$$

"Undershoot" - $$U_{3aH\,ij} = \frac{1}{K_{\phi ij}} \cdot \delta_{3aH} \cdot \overline{U_j}, \tag{12}$$

where: $K_{\phi ij}$ is the transmission coefficient for the comparator unit amplifier in the i-th monitoring line of the j-th vertical monitor and is fixed smoothly between 0.6 and 2.5;
δ_{ab}, δ_{np}, δ_{3aH} are the corresponding relative signal levels and are set smoothly between 0.75 and 1.85;
$\bar{v}$ is the arithmetic mean of the output signals from the sections of the j-th vertical monitor.

To establish the thresholds at which the signals are triggered in the monitoring lines of the equipment, the "Skala" computer system calculates the transmission coefficients $K_{\phi i}$ and $K_{\phi ij}$ for every operative calculation on the basis of the "Prizma" program. The remaining parameters in Eqs (5-7 and 10-12) determining the triggering levels are the set constants or current values of the average signals. The transmission coefficients are adjusted in accordance with the new calculated values in all cases in which the transmission coefficients displayed in the equipment and recalculated differ by 5% or more even for one monitoring line.

The principal relative error involved in signal triggering at monitor output signal levels of at least 2.5 V does not exceed ±2%.

The theoretical time to failure of a monitoring line of the equipment is at least 15 000 h. The lifetime of the equipment is at least 6 years.

2.9.8. Special software for operating the reactors of the Chernobyl' nuclear power station

1. Functions to be carried out and structure of calculations

The special software for the reactors of the Chernobyl' nuclear power station is intended to perform the following functions:

- Calculation of the power in each fuel assembly;
- Calculation of the power margin up to critical heat flow in each assembly;
- Calculation of the graphite temperature inside the core;
- Calculation of the reactor power by the heat balance method;
- Calculation of the steam content at the outlet from each channel;
- Calculation of the thermal reliability of the reactor;
- Calculation of the energy release of each assembly and the whole reactor;
- Calculation of the settings for the in-core radial and vertical power density monitors;
- Calculation of some of the characteristics of vertical power density distribution;
- Calculation of the operative reactivity reserve;
- Calculation of the recommendations for regulating the flow of water through the fuel channels;
- Calculation of the overall reactor parameters: radial non-uniformity of the power density distribution; distribution of power and flow through the halves of the reactor and drum separators etc.;
- Calculation of recommendations on reloading fuel channels.

Of the enumerated functions the last one is performed by an outside computer and the calculation data are transmitted to the nuclear power plant through a communication channel, together with the results of the neutron physics calculation.

The remaining functions are carried out with the "Skala" station computer in the form of a multifunctional "Prizma" program.

The input data for the "Prizma" program are:

- Signals from the in-core monitors of the system for physical monitoring of the radial and vertical power density distribution;
- Signals from the control rod position indicators;
- Signals from the flowmeters for each reactor channel, water temperature in the pressure headers, pressure in the drum separators, feedwater flow etc.;
- Signals from thermocouples measuring the graphite temperature;
- Results of the neutron physics calculation of power density distribution.

2.9.8.1. Periodicity and accuracy of calculations

The periodicity with which the main "Prizma" program calculations are carried out is once in 5-10 minutes. Once per day there is calculation of the energy yield of each fuel assembly and of the variation in the sensitivity of the monitors through the burnup of their emitters and the fuel burnup.

The accuracy attained in calculating the relative power of each fuel assembly is ≈ 3%. At the Chernobyl' nuclear power station this has been confirmed by a special experiment in which the calculation data for assembly power were compared with the measurements obtained with a calibration detector.

2.9.8.2. Basic theoretical relationships

Calculation of the power in each assembly (calculation of the power density distribution) is a basic element of computation in operative information processing in the "Skala" centralized monitoring system. The procedure used for this calculation takes as initial data the results of the neutron physics calculation of the power density distribution and the readings of the in-core monitors.

The result of the neutron physics calculation of the power density - the power of each fuel assembly $q^{(o)}_{pi}$, $i = 1, 2, \ldots, N_{TBK}$ (N_{TBK} is the number of fuel channels in the reactor) - is corrected for refuelling of the fuel channels from the relationship:

$$q_{pi} = q^{(o)}_{pi} \frac{\xi_T(R, E')}{\xi_T(R, E)}, \tag{13}$$

in which the correction coefficients ξ_T are different for different types of refuelling and are tabulated as a function of the distance R between the considered and refuelled channels and the energy yield of the unloaded (E) and loaded (E') assemblies. The correction coefficients are obtained from a theoretical analysis of the refuelling by means of the neutron physics calculation program. The correction is carried out with the station's own computer directly before refuelling or immediately afterwards.

The power density distribution is calculated from the following relationships.

An empirical correction is added to the results of the neutron physics calculation of the density distribution that takes into account variation in the power of each fuel assembly on account of movement of the control rods:

$$q'_{pi} = q_{pi} \prod_k \frac{\xi_{pc}(R_{ik}, h'_k)}{\xi_{Rc}(R_{ik}, h_k)}, \qquad (14)$$

where the tabulated coefficients ξ_{pc} depend on the distance R_{ik} between the i-th fuel assembly and the k-th rod and the depth of immersion of the k-th rod at the moment the given calculation (h'k) is made and at the moment corresponding to the neutron physics calculation. The correction coefficients have been derived on the basis of a theoretical analysis of the effect of the rods on the fuel channel power using the neutron physics calculation program.

The signals J_j from the in-core monitors of the physical power density monitoring system are converted into the following values:

$$q_j = J_j K_{Tpj} \xi_{Dj} \xi_{TDj}, \; j = 1, 2, \ldots, N_D, \qquad (15)$$

where: N_D is the number of properly working monitors;
K_{Tpj} is the calibration coefficient for the j-th monitor;
ξ_{Dj}, ξ_{TDj} are coefficients taking into acount burnup of the monitor emitter and the fuel in the monitor channel.

For assemblies with monitors one calculates the ratios:

$$V_j = \frac{q_j}{q'_{pj}}, \qquad (16)$$

the radial-azimuthal distribution of which is approximated by the expression:

$$\tilde{V}(r_i, \varphi_i) = \sum_{e} a_e f_e(r_i, \varphi_i) \tag{17}$$

for all assemblies in the reactor. In this expression r_i, y_i are the co-ordinates of the i-th assembly; $f_e(r_i, \varphi_i)$ is a set of radial-azimuthal functions. The coefficients a_e are determined by the method of least squares. This approximation means that one can take into account possible deformations of the power density distribution through transient xenon and temperature processes as well as the part of the neutron physics calculation error that is due to the random spread of the physical characteristics of the assemblies and other elements of the core, together with the methodological error in the neutron physics calculation.

For assemblies with monitors the values:

$$\overset{\circ}{V}_j = \frac{V_j}{\tilde{V}(r_j, \varphi_j)} - 1, \tag{18}$$

are calculated.

The readings from the j-th monitor or control rod position indicator close to the monitor are considered credible if:

$$\overset{\circ}{V}_j^{\,2} > \chi^2 D_{\overset{\circ}{V}}, \tag{19}$$

where χ is a normal distribution quantile corresponding to a given probability of non-acceptance of the true measurement;

$$D_{\overset{\circ}{V}} = \sum_{j=1}^{N_g} \overset{\circ}{V}_j^{\,2} / (N_g - 1). \tag{20}$$

The readings of these monitors are not used in the calculations; information about them is automatically fed to the printer for use by the operating staff.

Finally, the power of each assembly, including the assemblies with monitors, are calculated from the equation:

$$W_i = \tilde{V}(z_i, \varphi_i)\, q'_{pi} \left(1 + \sum_{j=1}^{4} b_{ij} \overset{\circ}{V}_j\right). \tag{21}$$

Summing is conducted over the four monitors closest to the i-th assembly. The weight coefficients b_{ik} are determined by solving a set of four linear equations compiled with the requirement of minimum error. These coefficients depend on the distance between the assembly and monitor, and the statistical characteristic values $\overset{\circ}{v}_j$ of the error in calibrating the monitors.

As well as the calculation of the power W_i of each assembly, there is also calculation of the dispersion D_i of the error in it.

Calculation of the power margin coefficient

The maximum permissible power of the RBMK reactor fuel assembly is taken to be the power at which the probability of the assembly experiencing critical heat flow attains a preset value constant in time and identical for all assemblies. In accordance with this definition the power margin up to critical heat flow for the i-th assembly is:

$$K_{zi} = \frac{W_{kpi} - \sqrt{W_{kpi}^2 - c_{1i} c_{2i}}}{c_{1i} c_{2i}}, \tag{22}$$

where: W_i is the power of the fuel assembly;
W_{kri} is the power of the assembly at which there is onset of critical heat flow. This value is determined from the tables as a function of the water flow through the channel, pressure in the drum separator and water temperature in the pressure header;

$$\begin{aligned} c_{1i} &= 1 - \varkappa^2 (D_i + D_{\tau n 2}); \\ c_{2i} &= W_{kpi}^2 (1 - \varkappa^2 D_{\tau n 1}) - \varkappa^2 D_{\tau n 3}; \end{aligned} \tag{23}$$

where: χ is the normal distribution quantile corresponding to the set probability that the assembly will experience critical heat flow;
D_i is the dispersion of the relative error determining the assembly power;
D_{Tni} is the dispersion of the relative error in determining the critical power of the assembly (methodological);
D_{Tn2} is the dispersion of the error in determining the reactor power;
D_{Tn3} is the dispersion of the determination of W_{kri} through errors in measuring the flow, pressure and temperature.

Calculation of the graphite temperature

The temperature of the graphite is calculated on the basis of assembly power calculations given for the vertical power distribution and signals from thermocouples mounted in the stack.

The graphite temperature is calculated for each k-th join in the graphite columns from the following relationship:

$$t_{\Gamma K} = t_T + \alpha \overline{W_K} \varphi_K , \tag{24}$$

where t_T is the mean temperature of the coolant in the reactor;

$$\overline{W_K} = \frac{1}{m} \sum_{i=1}^{m} W_{i(k)} \prod_{i=1}^{m} \gamma_{i(k)}, \tag{25}$$

where: $W_{i(k)}$ is the assembly power in the channel adjoining the k-th join;
m is the number of these assemblies ($m \leq 4$);
$\gamma_{i(k)}$ is a coefficient taking into account the effect on heat removal from the graphite by channels with different loadings (with fuel, absorbers, and control rods);
φ_k is the relative neutron flux density at the level where the thermocouples are located;
α is a coefficient of proportionality between the thermocouple signal and the power as determined by the least square method.

The thermocouple readings which stand out strongly are processed.

Calculation of the in-core monitor settings

The setting for the radial monitor is calculated from the equation:

$$U_j = J_j \max_{i \in \omega_j} \left\{ \frac{q(r_i)}{W_i} \right\} \cdot C \tag{26}$$

where: J_j is the signal from the j-th monitor;
W_i is the power of the i-th assembly in the region W_j close to the j-th monitor (this region is a square 5 x 5 reactor cells in size, in the centre of which the j-th monitor is positioned);
$q(r_i)$ is the relative regulation value of the power of the i-th assembly;
C is the normalizing constant corresponding to the prescribed maximum assembly power.

The setting for the vertical monitor is calculated from the requirement that the prescribed maximum linear load on the fuel element and the assemblies close to the monitor should not be exceeded.

From the design values of the settings for the in-core monitors are calculated and printed out, on request, the transmission coefficients of the amplifiers of the comparator units in the physical power density monitoring system, which are then fed into the equipment.

Calculation of the operative reactivity reserve

The operative reactivity reserve on the control rods is calculated from the equation:

$$\rho = \sum_{k=1}^{N_{pc}} C_k \int_0^{h_k} \Phi_k^2(z)\,dz \Big/ \sum_{k=1}^{N_{pc}} \int_0^{H} \Phi_k^2(z)\,dz \tag{27}$$

where: C_k is the relative "weight" of the rod and depends on its type;
N_{pc} is the number of regulating rods;
$\Phi_k(z)$ is a value proportional to the vertical distribution of the absolute neutron density flux at the point where the k-th rod is situated, and is calculated by the use of the values of the assembly power W_i computed from the vertical monitor readings averaged over the reactor.

2.9.9. Presentation of information on the calculation results

All the calculation results are transmitted, upon call by the staff, to printing devices in the form of charts (cartograms) and summary tables. The latter show, in particular, lists of 60 channels in which the channel power is maximal, 60 channels with maximum graphite temperatures, and 60 channels with the least margin coefficients.

Transmitted automatically to the printout device are the time, the co-ordinates of the rejected detector, the rejection constant, the time, the channel co-ordinate and the power (in the case where the prescribed value is overshot).

The mimetic board for the channels shows the channels in which the power is greater than the value set by the operator; the channels in which the coefficient is below the operator's setting, and so forth.

The values of any quantity computed can be shown on the digital indicating instruments if called for.

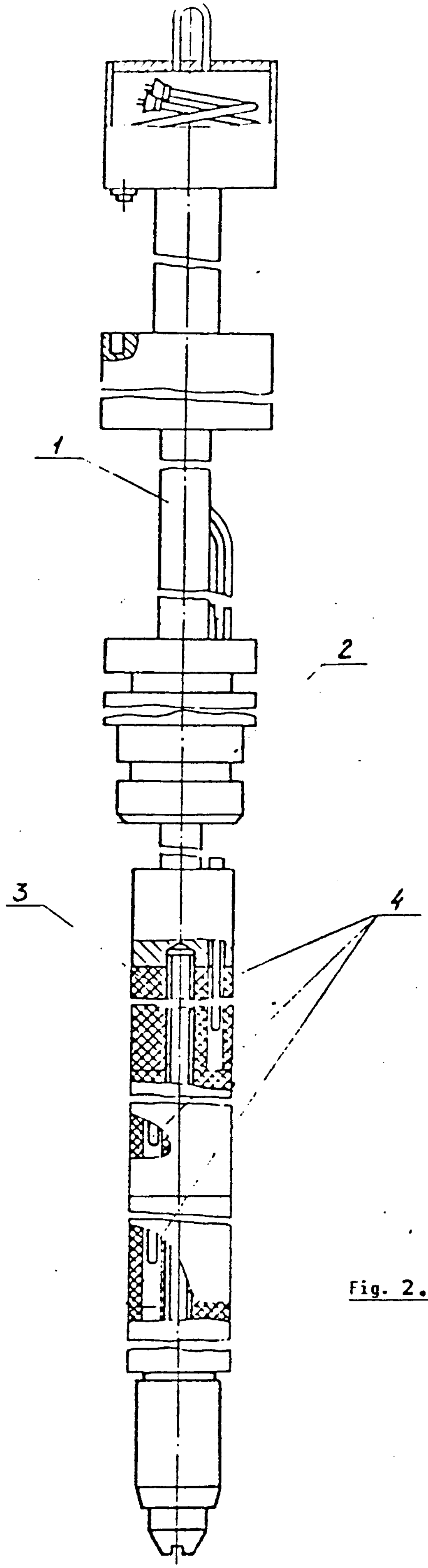


Fig. 2.36. Thermocouple unit and assembly:

1. Tube; 2. Rod; 3. Graphite bush;
4. Chromel-Alumel heat sensor.

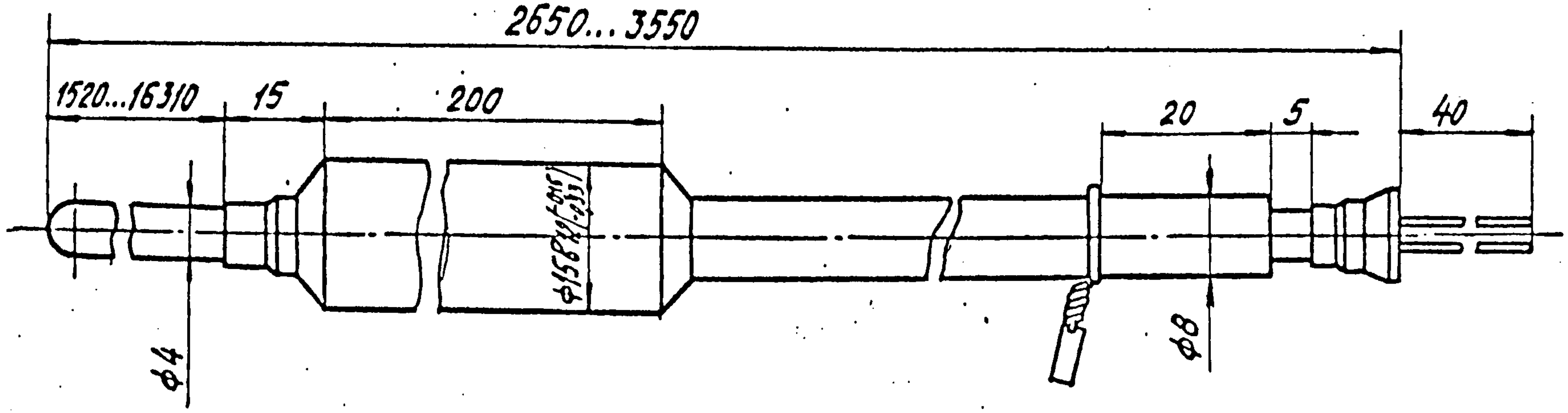


Fig. 2.37. Outline drawing of Chromel-Alumel cable-type thermocouple.

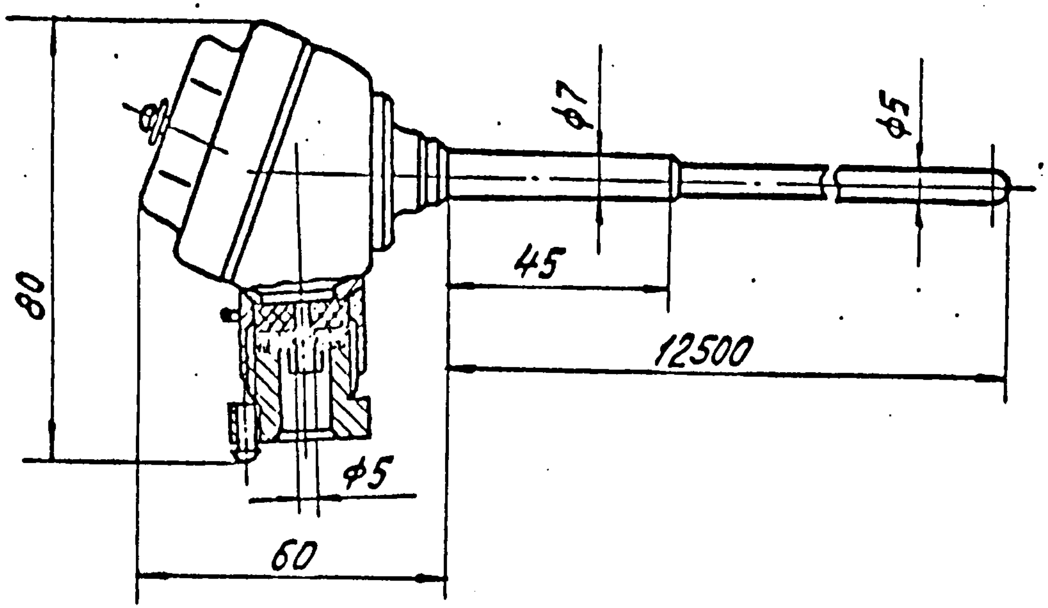
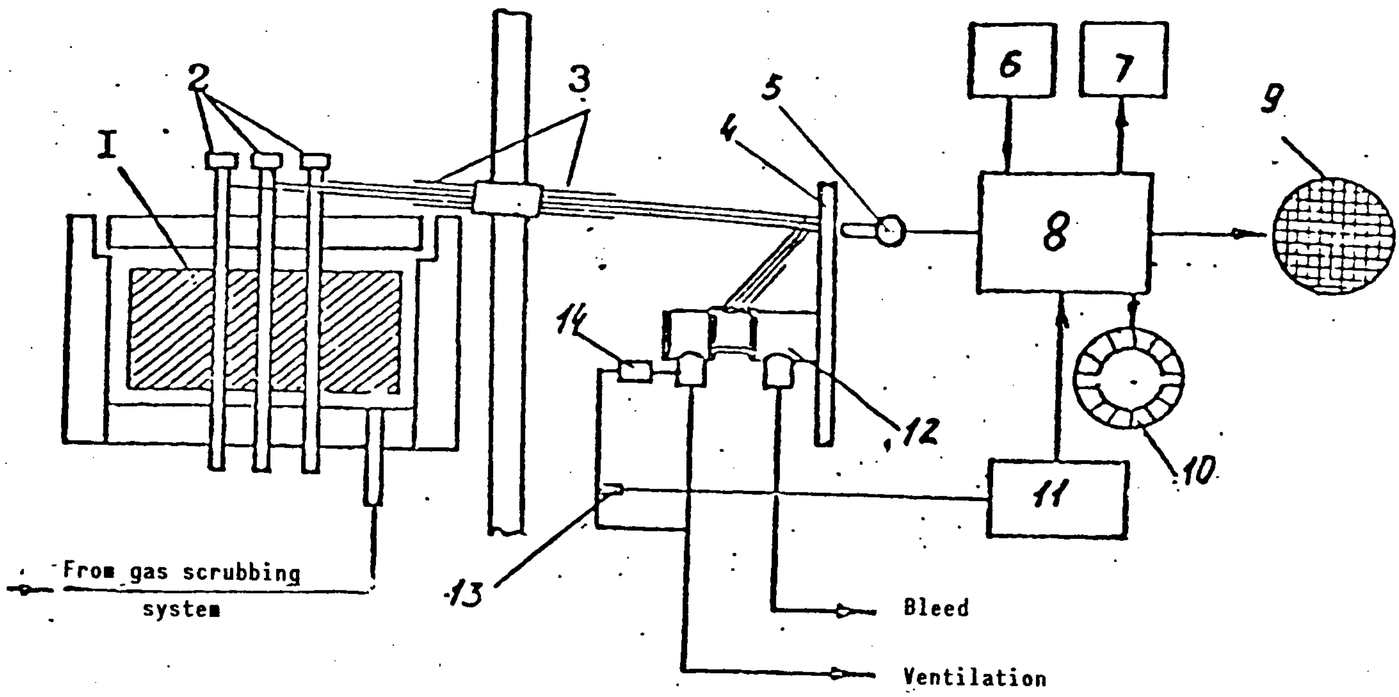
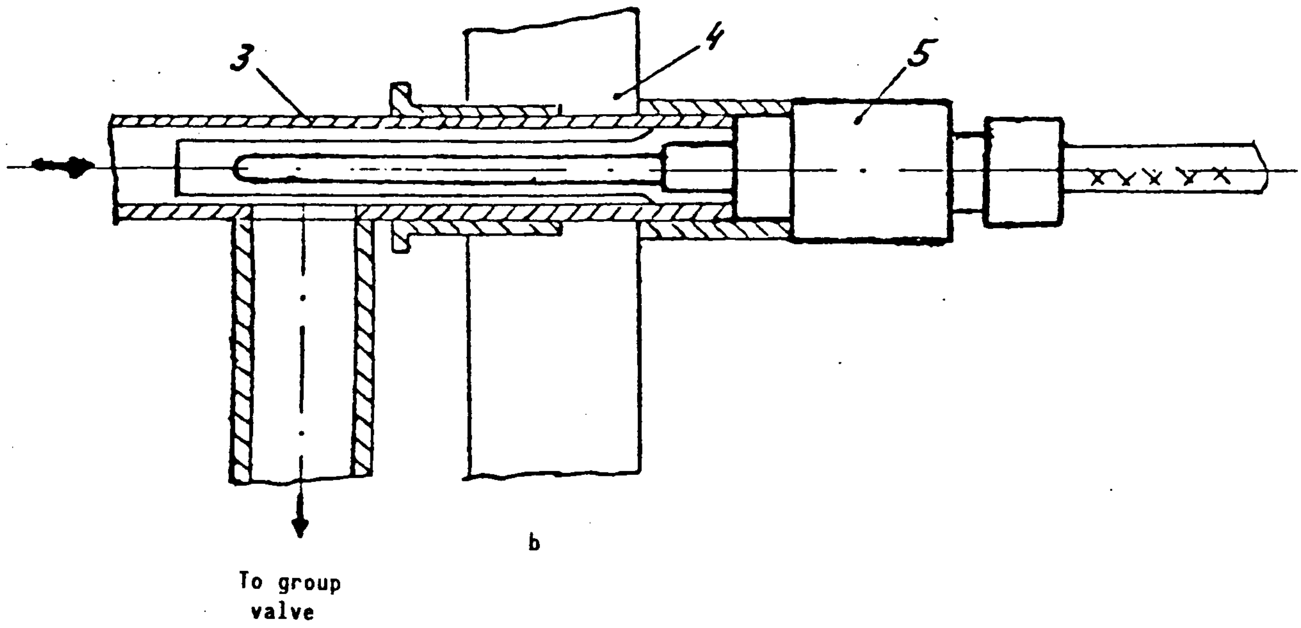


Fig. 2.38. Outline drawing of headed Chromel-Alumel cable-type thermocouple.



a



b

Fig. 2.39. (a) Process channel integrity monitoring system;
(b) Chromel-Copel temperature sensor unit.

1. Reactor; 2. Process channel; 3. Impulse tubes; 4. Panel; 5. Temperature sensor; 6. Retrieval device; 7. Digital display equipment; 8. "Skala" system; 9. Channel mimic board; 10. Humidity board; 11. Humidity indicator; 12. Group valve; 13. Humidity sensor; 14. Gas blower.

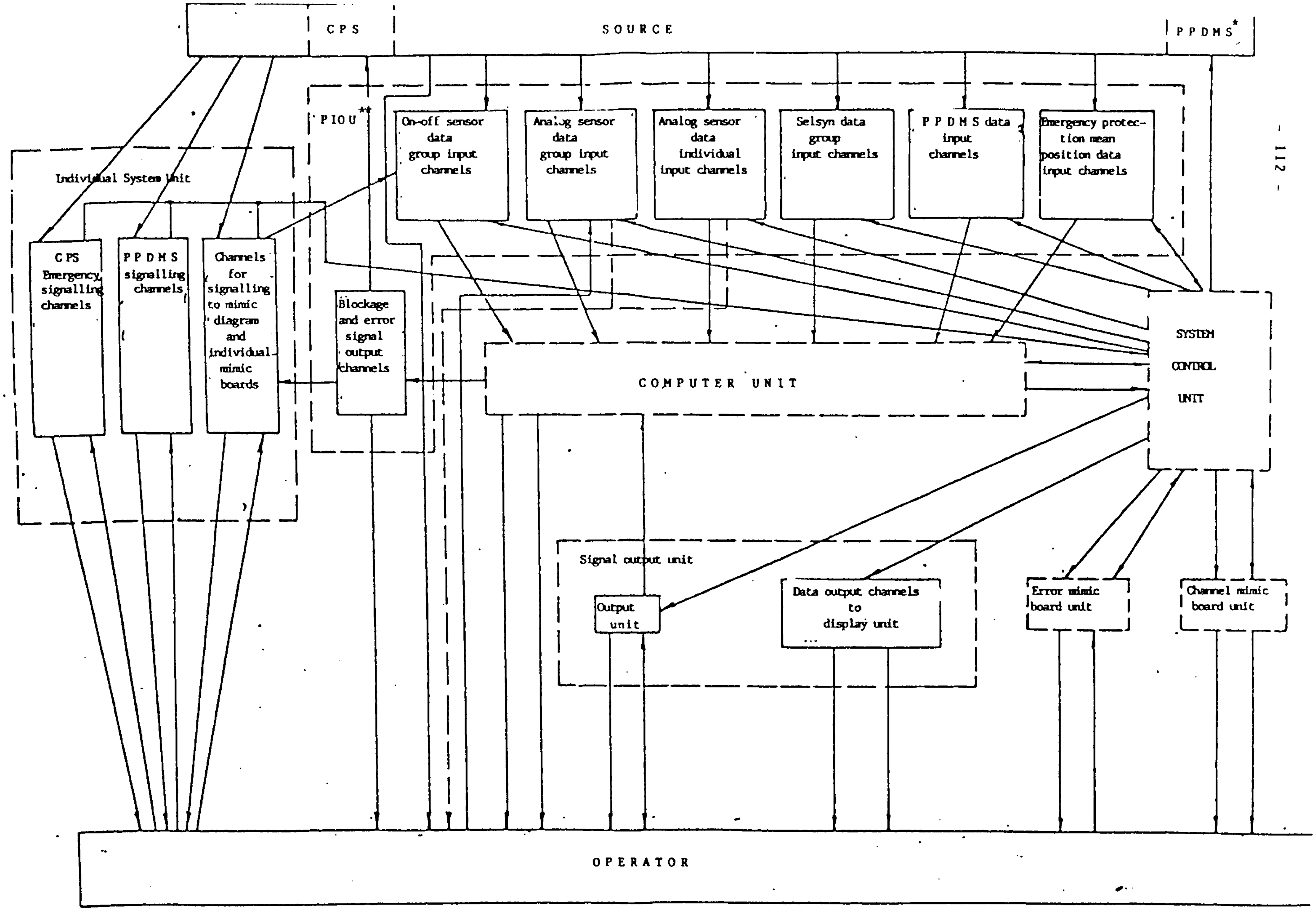


Fig. 2.40 STRUCTURAL DIAGRAM OF THE "SKALA" SYSTEM

* P P D M S = Physical power distribution monitoring system

** P I O U = Parameter input-output unit

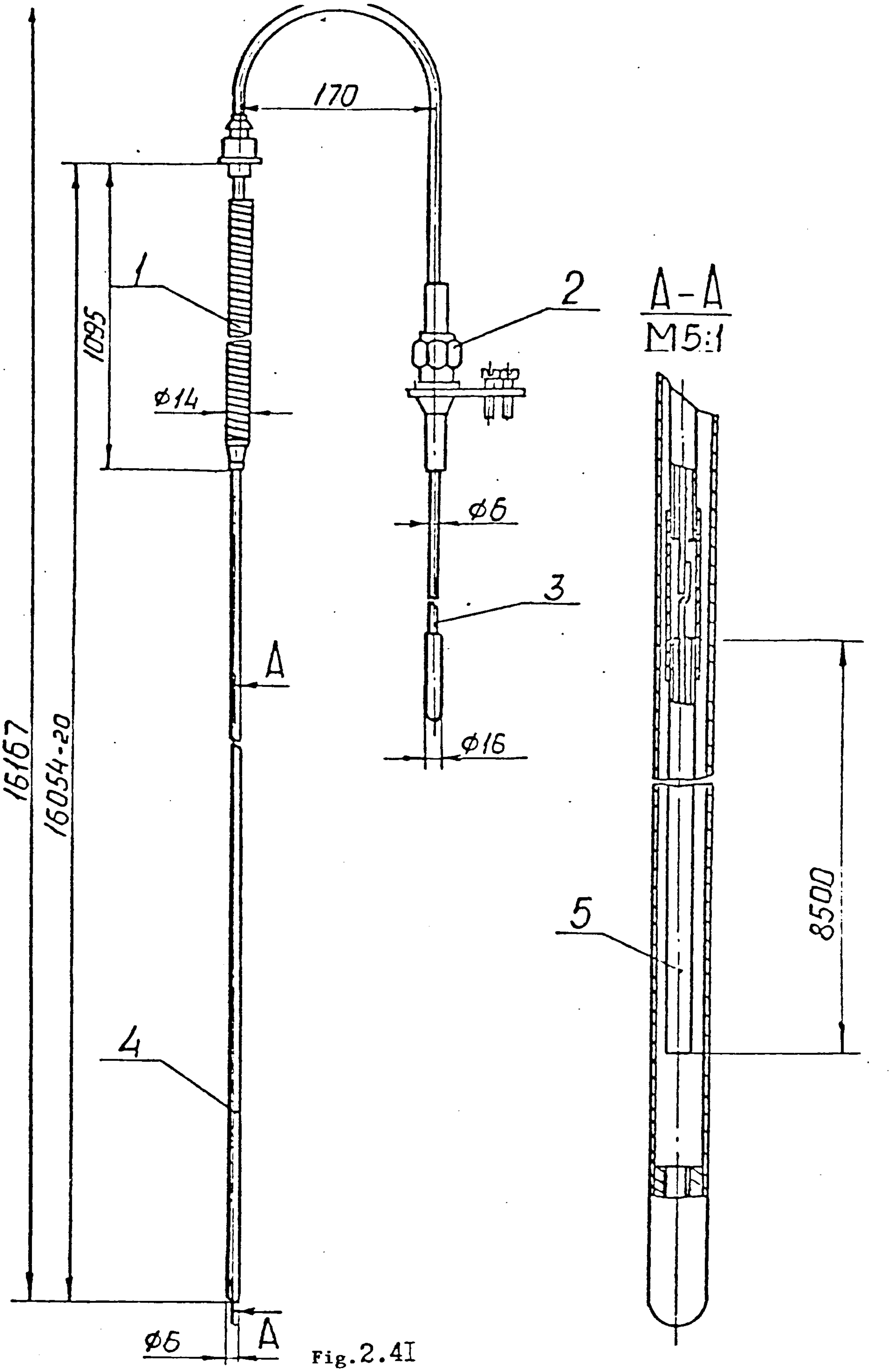


Fig.2.41
Design of the radial power density monitor

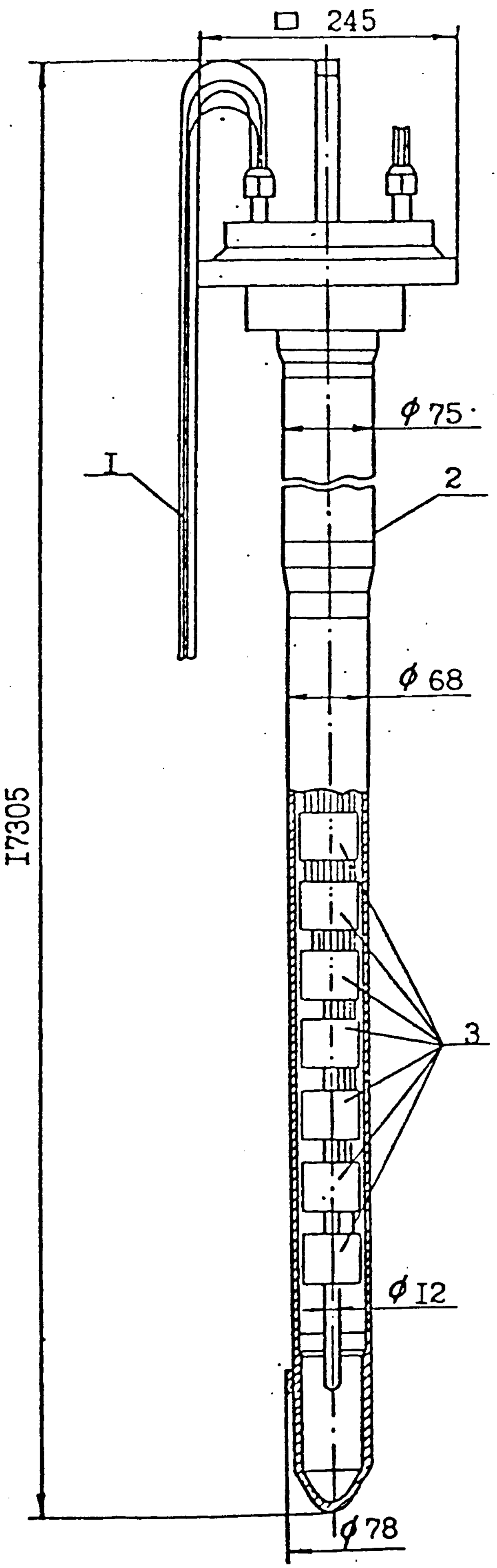


Fig. 2.42 Design of the vertical power density monitor:
(1) cable; (2) leaktight tube; (3) sensitive elements

2.10. Safety systems

2.10.1. Protective safety systems

2.10.1.1. Reactor emergency cooling system

The emergency core cooling system (ECCS), shown in diagram 2.43, is a protective safety system designed to draw off the residual heat release (after suppression of the chain reaction) through the timely feeding of the appropriate volume of water into the reactor channels in the event of accidents accompanied by damage to the core cooling system.

Associated with such accidents are: ruptures in the large diameter multiple forced circulation circuit (primary coolant circuit) (MFCC) pipelines, ruptures in the fresh steam pipelines and ruptures in the feedwater pipelines.

In addition to this, the ECCS may be used for the emergency feeding of water into the reactor channels in situations which are not connected with ruptured pipelines, but which make it impossible to supply the water through regular systems (for example, steam in the electric feed pumps).

The ECCS was designed with the following requirements in mind:

1. It must ensure a water supply to the damaged and undamaged halves of the reactor in quantities not less than those shown in diagram 2.44, thereby preventing melting, massive overheating and seal failure in the fuel elements;

2. The ECCS must come into operation automatically on receipt of the "maximum design-basis accident signal", which must distinguish between the damaged and undamaged halves and be formed on the basis of the following indications:

   (a) An increase in pressure in compartments containing MFCC pipelines (indication of pipeline rupture);

   (b) Coincidence with either of the following two signals (showing selection of the damaged half):

      - Drop in level in the steam separators of the damaged half of the reactor;

      - Decrease in the pressure differential between the main circulation pump pressure header and the steam separators of the damaged half of the reactor;

3. The speed of operation of the ECCS is such that the break in water supply to the damaged half of the reactor in the event of a maximum design-basis accident does not exceed 3.5 sec.;

4. There must not be an unacceptable reduction in water supply to the reactor channels as a result of its unproductive loss through the point of rupture in the compartment;

5. The system must perform its safety functions in the event of any failure, occurring independently of the source event, in any of the following parts of the system: any active or passive element having moving mechanical parts;

6. The system must comprise a number of independent channels (subsystems) and must function with the required effectiveness in the event of a failure, occurring independently of the source event, in any one channel (subsystem) of this system;

7. In the event of drainage of the ECCS vessels, nitrogen from them must not be allowed to reach the reactor;

8. The ECCS must operate as intended in the event of a maximum design-basis accident coinciding with a loss of own-requirement power from the power unit.

In order to comply with the above essential requirements, the ECCS comprises three independent channels (subsystems), each of which ensures not less than 50% of the required output.

Each channel (subsystem) includes a fast-acting part and a part providing prolonged afterheat removal.

The fast-acting part supplies the required quantity of water to the channels of the damaged half of the reactor during the initial stage of the accident.

The fast-acting parts of two ECCS channels (the cylinder parts) take the form of a system of vessels (filled with water and nitrogen at a pressure of 10.0 MPa), connected by pipelines and headers to the distributing group header of the primary coolant system.

A Du 400 gate valve is used as a quick-closing armature for bringing into operation the cylinder parts of the ECCS; by this means the required supply can be brought to the damaged half of the reactor in the space of 3.5 sec. The power supply for the gate valves is supplied under the maximum reliability category by the accumulator batteries (see section 2.7.3).

Each of the two cylinder parts comprises six vessels of 25 m^3 volume. The total initial volume of water of one cylinder part is approximately 80 m^3, and of nitrogen, approximately 70 m^3. Each cylinder part supplies not less than 50% of the required quantity of water to the damaged half of the reactor over a period of not less than 100 sec. The period of operation depends on the scale of the coolant leak from the primary coolant system in the event of an accident.

The configuration of the cylinders is symmetrical, in order to reduce the "collector effect" when discharging.

The nitrogen from the ECCS vessels is prevented from reaching the reactor through the automatic closing of two gate valves installed in series on the pipelines from the cylinder parts to the distributing group header on receipt of a signal indicating the minimum level in the cylinders.

The fast-acting part of the third ECCS channel is a unit for supplying water from the electric feed pump, which ensures a supply of not less than 50% of the required amount of water to the damaged half of the reactor. In the event of a maximum design-basis accident coinciding with a loss of power to users of the own-requirement supply from the power unit, the supply of water from the electric feed pump is assured for a period of 45-50 sec. as the pump runs down in tandem with the turbogenerator.

The reserve power for the drive units of the fast-acting gate valves is supplied by the independent sources of uninterrupted power (the accumulator batteries).

The prolonged afterheat removal part provides cooling both to the damaged and undamaged parts of the reactor. It comes into operation no later than the moment at which the fast-acting part of the ECCS ceases to operate.

The long-term afterheat removal part of each of the three ECCS channels comprises two groups of pumps:

- the cooling pumps of the damaged half of the reactor;

- the cooling pumps of the undamaged half of the reactor.

The pump section of the cooling system for the damaged half of the reactor of each of the three ECCS channels consists of two pumps connected in parallel. Its function is to ensure a supply of water at a rate of approximately 500 t/h, that is, not less than 50% of the required rate for the damaged half in the event of a maximum design-basis accident.

The water is drawn by the pumps from the pressure suppression pool of the accident localization system, is cooled by the service water in the heat exchanger mounted on the common intake line of the two pumps, and reaches the ECCS headers through the discharge lines.

Flow-limiting inserts are mounted on the discharge lines of the pumps; they are designed to ensure the steady functioning of the pumps in emergency situations characterized by a sharp drop in pressure in the reactor's coolant circuit through a ruptured pipe (flow limitation is achieved by boiling water in the narrow cross-section of the insert).

The pump section of the cooling system for the undamaged half of the reactor of each of the ECCS channels contains one pump and supplies water at a rate of approximately 250 t/h, that is, not less than 50% of the flow required for the undamaged half in a maximum design-basis accident.

Water is drawn by suction from the tanks containing clean condensate and, by means of the discharge line, reaches the headers of the cylinder section behind the quick-closing armature.

The flow-limiting inserts in the discharge lines of the pumps perform the same functions as for the pumps of the cooling system of the damaged half of the reactor.

Stand-by power for the electric motors of the pumps and armature drive units of the pump sections of the damaged and undamaged parts of the reactor is supplied by the diesel generators.

2.10.1.2. System to protect against excess pressure in the main coolant circuit

This system is designed to ensure that the permissible pressure level in the circuit is not exceeded; it does this by drawing off the unbalanced steam into the pressure suppression pool, where it is completely condensed. The system includes a pulsed safety device and a system of pipes and headers which conduct the steam into the pressure suppression pool of the accident localization system. The pulsed safety device comprises the pulse valves and the main safety valves.

The system satisfies the following main requirements:

- It ensures that the pressure in the circuit is not exceeded by more than 15% of the working pressure, taking into account a single failure in the system of an active or passive element with moving mechanical parts;

- It comes very reliably into operation when the pressure in the coolant circuit reaches the minimum operating value;

- It is very reliable in closing the main safety valves when the close value is reached;

- It is capable of working adequately under conditions of alternating dynamic loads upon operation of the main safety valves;

- It introduces steam into the water of the pressure suppression pool at speeds that are close to that of sound, even when one main safety valve is in operation (this is necessary for shock-free steam condensation).

A schematic drawing of the system for discharging steam from the main safety valves into the pressure suppression pool of the accident localization system is shown in diagram 2.45.

The system comprises eight main safety valves with a total output of 5800 t/h, under nominal circuit pressure, i.e. an output which is equal to the nominal steam output of the reactor installation. The control of each main safety valve with an output of 725 t/h is effected using a directly acting pulse valve (lever-gravity type), equipped with an electromagnetic drive unit for opening and closing.

Steam from the main safety valves is discharged into the pressure suppression pool beneath the water level through submersible nozzles, each with an exit diameter of 40 mm (approximately 1200 nozzles in all).

In order to prevent the formation of a vacuum in the discharge pipelines and the consequent entry of water into them, and also to ensure the shock-free condensation of possible small flows of steam through the closed main safety valves, steam-air ejectors are used.

The steam discharge systems include:

- Control of the absence of water level in the pressure suppression pool headers;

- Control of the temperature conditions of the external surface of the pipes for each main safety valve and pipes situated inside the pressure suppression pool.

When the unit is working normally, the main safety valves are closed and the system is in waiting mode.

The system comes automatically into operation only when there is excess pressure in the primary coolant circuit; it does this as follows:

76 kgf/cm^2 - 1 main safety valve operates;

77 kgf/cm^2 - 2 main safety valves operate;

78 kgf/cm^2 - 1 main safety valve operates;

81 kgf/cm^2 - 4 main safety valves operate.

It is possible for the operative staff of the unit control room and reactor control room to forcibly open the main safety valves.

The main elements of the system to protect the circuit from excess pressure during operation have undergone experimental bench tests. The system as a whole underwent comprehensive and direct testing in respect of design requirements during the period of commissioning operations.

2.10.1.3. System to protect the reactor space from excess pressure

The purpose of this system is to ensure that the permissible pressure in the reactor space is not exceeded in an accident situation involving the rupture of one fuel channel. It achieves this by drawing the steam and gas mixture from the reactor space into the steam and gas discharge compartment of the pressure suppression pool and subsequently into the pressure suppression pool itself.

The system satisfies the following main requirements:

- It ensures that the excess pressure in the reactor space does not exceed 1.8 kgf/cm^2 (abs.) in the event of a total cross break of one fuel channel and taking into account a single failure in the system of a passive element with moving mechanical parts (there are no active elements in the system);

- It prevents water from the steam and gas discharge compartment of the pressure suppression pool from entering (overflowing into) the reactor space in the event of a maximum design-basis accident;

- It ensures that the reactor space is reliably isolated from the atmosphere.

A schematic drawing of the system for protecting the reactor space from excess pressure is shown in Fig. 2.46.

The reactor space is constantly connected to the steam and gas discharge compartment of the pressure suppression pool of the accident localization system by eight Du 300 pipes (four pipes below and four pipes above the reactor space, which then join to become two Du 600 pipes).

Each of the Du 600 pipes goes to its own level of the compartment and is immersed in 2 m of water, that is, under normal unit operating conditions, the reactor space is separated from the atmosphere by a hydroseal 2 m high.

The height of the vertical sections of the Du 600 steam discharge pipes from the reactor to the water level in the compartment is more than 28 m, and for this the Du 600 pipe, which joins together the four Du 300 pipes beneath the reactor space, rises specially to the 28.8 mark and then immediately drops into the compartment.

Such an arrangement is necessary in order to prevent water or a steam-air mixture from the compartment from entering the reactor space in the event of accidents involving the rupture of MFCC pipes, right up to a maximum design-basis accident.

The volume of water in the compartment is selected and maintained in such a way as to ensure a sufficient reserve to fill the steam discharge pipes in the situation indicated above.

A second and supplementary barrier, designed to prevent water or steam-air mixture from the compartment from entering the reactor space, is in the form of non-return (escape) valves, which make it possible to discharge steam from the compartment into the pressure suppression pool and to prevent its reverse flow.

In order to prevent the [un-?] controlled spread of solid radioactive wastes throughout the water content of the pressure suppression pool, the steam-gas discharge compartment is reliably cut off (by three barriers) from the water content of the pressure suppression pool in the event of a fuel channel rupture.

In the event of a rise in pressure in the reactor space to 1.2 kgf/cm^2 (abs.), the hydroseal in the compartment pops open and the steam and gas mixture enters the compartment through the steam discharge pipes. Should the pressure in the above-water part of the compartment reach 1.1 kgf/cm^2 (abs.), the non-return (release) valves open and the steam and gas mixture enters the steam distribution corridor; it then enters the water of the pressure suppression pool by means of the steam discharge pipes. The steam formed in the reactor space in the event of a fuel channel rupture is fully condensed, initially in the water in the compartment, and, when the storage capability of this is exhausted, in the pressure suppression pool. The gas from the reactor space, bubbling through the layer of water in the compartment/pressure suppression pool, is cooled and maintained in the compartments of the accident localization zone, after which, following the necessary period of holding and cleaning, it is discharged into the atmosphere by the hydrogen disposal system.

The maximum pressure in the reactor space at all stages of the accident sequence does not exceed 1.8 kgf/cm^2 (abs.).

The protection system includes:

- Pressure control (underpressure) in the reactor space;

- Control of the level of steam-gas discharges in the compartment;

- Reliable drainage of the steam discharge pipes.

Monitoring of the process parameters and control of the active elements of the system (cut-off armature) is carried out by operative staff in the unit control-room and reactor control-room.

2.10.2. Safety based on confinement systems

The accident confinement system built in the fourth unit is designed to confine radioactive releases during accidents involving failure of any piping of the reactor cooling circuit, except the steam-water lines, upper fuel channels and the part of the downcomers which is located in the drum separator area. The schematic diagram of the system is given in Fig. 2.47.

2.10.2.1. System of sealed locations

The basic component of the confinement system is the system of sealed locations comprising the following locations of the reactor part:

- Strong leak-tight compartments (1 and 2 in Fig. 2.47) arranged symetrically in relation to the reactor axis and designed for an overpressure of 0.45 MPa;

- Locations of the distributing group header and lower water communication lines (DGH-LWCL) (3 and 4), which are also symetrical in relation to the reactor axis and are separated from each other by the reactor's supporting cross-piece having a total non-tight area of 5 m^2. According to the strength specifications for the reactor structural elements, these locations do not tolerate a pressure rise above 0.08 MPa and are designed for this value. The strong leak-tight compartments and the DGH-LWCL locations contain all those reactor circuit elements that may be damaged in accidents for which the system is designed;

- Location of the steam distribution corridor (5);

- Location of the two-storey condensation-type pressure suppression system, a part of which is filled with water (7) and the rest with air (8).

The sealed locations are connected with each other by means of valves of three types:

- Non-return valves (9), installed in the openings of the cover separating the DGH-LWCL location and the steam distribution corridor;

- Release valves (10), installed in the openings of the cover separating the air space in the pressurizer relief tank and the strong leak-tight compartments;

- Panels of non-return valves (11) installed in the partitions separating the steam distribution corridor and the strong leak-tight compartments.

The locations of the strong leak-tight compartments and steam distribution corridor are connected with the water volume of the condensation-type pressure suppression system by steam outlet channels (17).

In normal operation the system of sealed locations and the condensation-type pressure suppression system operate in the stand-by mode.

In emergency situations the system functions in the following manner. If a reactor circuit component in one of the strong leak-tight compartments fails, the boiling coolant begins to flow into that compartment. Steam formation leads to a rise in pressure in the accident location. The non-return valves of the panels connecting the damaged half of the compartment with the steam distribution corridor (11) open at a pressure difference of > 0.002 MPa. When the pressure in the damaged half of the compartment attains a value sufficient for displacing the liquid column from the steam outlet channels, the steam and air mixture begins to flow into both stories of the condensation system at the same time. By bubbling through the water layer the steam condenses and the air is collected in the air space of the condensation system; when the pressure there reaches > 5 kPa, the release valves connecting the air space of the condensation system with the undamaged strong leak-tight compartment open and part of the air flows to that compartment. Thus, its volume is used in this emergency situation to reduce pressure in the damaged half of the leak-tight compartment. In the course of such an emergency the non-return valves (9) remain closed.

If the reactor circuit failure occurs in the DGH-LWCL location, the pressure rise there opens the non-return valves connecting it and the steam distribution corridor (at a pressure difference of > 0.02 MPa). From the corridor via the steam discharge channels the steam-air mixture goes into the water volume of the condensation system's central part situated under the corridor. Pressure rise in the air space of the condensation system opens the release valves connecting that space with the two strong leak-tight compartments. In this kind of emergency situation the volumes of both compartments are used to reduce pressure in the damaged location, while the panel valves (11) remain closed.

All the sealed locations of the system, except the condensation-type pressure suppression system, have a 4 mm-thick lining of the VST3KP2 steel and are subjected to control tests for local and integral leak-tightness. The condensation-type pressure suppression system has a 4 mm lining of the 08Kh18N10T [cr18ni10ti] steel. Figures 2.48 and 2.49, respectively, show the results of calculation of pressure changes in the sealed locations during an accident involving a rupture of the main circulation pump pressure header (average diameter 900 mm) in the strong leak-tight compartment and those during an accident with rupture of the distributing group header (average diameter 300 mm) in the DGH-LWCL location. As will be seen from the graphs in these figures, the overpressure in the damaged leak-tight compartment does not exceed the maximum permissible value of 0.25 MPa, while the overpressure in the damaged DGH-LWCL location does not exceed the maximum permissible value of 0.08 MPa.

The system carries out its functions under conditions of a single failure of any passive component having moving parts (the system has no active components).

2.10.2.2. Penetrations and doors

To prevent the spread of activity outside the sealed locations the sealed barriers of the accident confinement system (walls, floors and ceilings) are equipped with special sealed penetrations at the places where they are traversed by pipes or electrical cable.

The pipe penetrations are designed to withstand the action of jets from a pipeline during its complete rupture. In such a case, the sealing of the penetration is not destroyed.

The design of the penetrations is such as to allow checking of their tightness during both assembly and operation. The penetrations ensure leak-tightness at an overpressure of up to 45 kPa in the accident confinement locations, at a temperature of up to 150°C.

The sealed pipe penetrations intended for the passage of "hot" lines are equipped with a water or air cooling system to prevent overheating of the concrete in the penetration area.

Sealed doors are intended to provide access of the service personnel into the locations of the accident confinement system when the reactor is shut down and to ensure sealing of those locations when the reactor is operating.

The sealed doors of the condensation-type pressure suppression system ensure the necessary leak-tightness and operability after elimination of each accident situation, including a design basis accident (DBA).

There are two sluice-gate-type entrances into the above pressure suppression system, each entrance having two successive sealed doors.

2.10.2.3. Cut-off and sealing valves

The cut-off and sealing valve system ensures isolation of the accident confinement area by cutting off the communication lines between the sealed and non-sealed locations.

The system's design is based on the following major principles:

- All communication lines traversing the sealing circuit which have to be shut off at the time of an accident in order to prevent escape of radioactive substances outside the sealed locations are equipped with three successive cut-off devices;

- Each pipeline which is not connected directly with the primary circuit or with the space of the sealed locations is equipped with one cut-off device mounted outside the sealing circuit;

- The positions of the shut-off valves which seal the locations under accident conditions are indicated on the unit's control board (safety panels) and stand-by control board, from where they can be remote-controlled by the operator if necessary;

- The drives of the cut-off valve installed on one main line are powered from independent sources of the reliable power supply system of category 1A.

The special fast-acting (10-15 s) cut-off valves and non-return valves are used for isolating and sealing the accident confinement locations.

Pre-operational tests are carried out in the manufacturing factories.

The testing of isolation valves during the operation of the nuclear power plant is performed only when the unit is shut down. The entire system of isolation valves is verified. The tests include verification of their efficiency and tightness.

The valves are closed automatically by DBA signals.

The system of cut-off and sealing valves is so designed that any single failure in the system will not disrupt its functions.

2.10.2.4. Condensation-type pressure suppression system

The purpose of this system is to condense steam formed:

- during an accident involving reactor circuit failure;

- during the actuation of the main safety valves;
- during leaks through the main safety valves under normal operating conditions.

The system is a two-storey reinforced concrete tank with a metal lining inside. The space in each storey is divided by longitudinal partitions into four corridors and by transverse partitions into three sections: two lateral (under the leak-tight compartments) and one central (under the steam distribution corridor). The longitudinal and transverse walls have the necessary openings for water and air. The lower part of each storey is filled with water. The thickness of the water layer in each storey is 1200 mm. The total volume of water in the two storeys is 3200 m^3 and the volume of the air space is 3700 m^3.

Steam goes into the water volume through the steam discharge channels, which are located uniformly over the whole area of the leak-tight compartments and the steam distribution corridor. Each steam discharge channel is in the form of a pipe-within-pipe-type block, which ensures simultaneous and uniform delivery of steam to both the storeys. The number, diameter and spacing of the steam distribution pipes and their depth under water are determined from tests on a large-scale model and ensure full condensation of the steam in the water volume of the condensation system, its uniform heating and rapid reduction of temperature in the damaged sealed location during accidents involving reactor circuit failures.

The upper storey of the system has the necessary number of special vertical overflow pipes with a diameter of 800 mm (28 in Fig. 2.47). The purpose of these pipes is to maintain the necessary level in the upper storey and to equalize pressure in the air spaces of both storeys.

There is continuous monitoring of the water level in both storeys and of the temperature and chemical composition of water. The required chemical composition of the water is ensured by the by-pass purification unit.

The confinement system also includes a system of heat removal from the condensation system and from sealed locations and a system of hydrogen removal.

Heat removal from the sealed locations of the confinement system is carried out by two systems:

1. Sprinkler cooling system;
2. Surface-type condensers located in the steam distribution corridor.

The sprinkler cooling system carries out the following functions:

- Cooling and purification of for air in the leaktight compartments and in the air space of the condensation system under both normal operating and accident conditions;

- Cooling of the water volume in the condensation system.

The main components of the sprinkler cooling system are shown in Fig. 2.47. Water is collected from the condensation system and is sent via three lines of pipe (each accounting for 50% of the system's throughput) to the heat-exchangers (15), where it is cooled by industrial water, and then through pumps (14) to all users of the system:

- to the jet coolers (12) in the leaktight compartments;

- to the nozzles (13) located in the air space in the two storeys of the condensation system.

The jet coolers form part of the sprinkler cooling system and ensure circulation of air in the leaktight compartments, cooling of the air and removal of radioactive aerosols and steam from it.

The air from the upper (hottest) part of the leaktight compartments is collected, cooled by water jets and sent to the lower part of the compartments. After its contact with the air the cooling water returns to the condensation system. The jet coolers work continuously both under normal operating and accident conditions.

The sprinkler nozzles located in the air space of the condensation system ensure spraying of the cooling water and mixing and cooling of the air. The necessary pressure difference at the nozzles is created by the reducer discs at the feeders supplying cooling water. The nozzles work continuously both under normal operating and accident conditions.

The purpose of the surface-type condensers (16) in the steam distribution corrider is to remove heat from the sealed locations during accidents involving reactor circuit failure by condensing the part of steam entering to the corridor. The cooling medium is industrial water. In normal operation the surface-type condensors work in the standby mode and go into operation on receiving the DBA signal.

During the development period the efficiency of the system was confirmed by tests on a large-scale model.

The system performs its functions when there is a single failure of any active component or passive component with moving parts.

2.10.2.5. Hydrogen removal system

The purpose of this system is to create a negative pressure in the accident confinement locations, to measure the concentration of hydrogen, which may enter these locations with uncontrolled leaks from the multiple forced circulation loop and also during steam discharge from the main safety valves and during accidents associated with pipe failures in the main forced circulation loop, and to remove the hydrogen upon its occurrence.

Under normal operating conditions of the unit, hydrogen may enter the accident confinement system locations with coolant leaks, the magnitude of which is taken as 2 t/h, and with possible leaks of steam through the closed safety valves.

Hydrogen may also enter under conditions of short-time steam release during actuation of the main safetey valves and under conditions of pipe failure.

The highest quantity of hydrogen may enter the location under DBA conditions (the hydrogen accumulated in the coolant and also that formed during the accident by radiolysis and by the reaction of zirconium with water).

Figure 2.50 gives the total amount of hydrogen entering under these conditions.

Against the existing standard of 4% (by volume) for the lower explosive limit of hydrogen in air, 0.2% concentration (by volume) was taken as the design value in the project. It is necessary to evacuate air from the accident confinement locations at the rate of 800 m^3/h in order to maintain this concentration under the most unfavourable conditions. This rate is adopted also for all other operating conditions of the unit. The hydrogen removal system (Fig. 2.51) comprises: an electric heater, a contactor, a condenser, a moisture separator and a gas blower. This equipment is divided into three sub-systems, each of which is located in a separate compartment. Each sub-system is provided with a stand-by active component-gas blower. The protective isolation valves are located in a separate compartment. The principle of 3 x 100% redundancy is envisaged.

Under normal operating conditions the gas-air mixture passes through the electric heater, contactor (in the presence of hydrogen), condenser and moisture separator and, by means of the gas blower, through the filtration plant and is discharged into the atmosphere.

By a DBA signal the protective isolation valve closes and the equipment of the hydrogen removal system is disconnected. After 2-3 hours (as hydrogen accumulates) the operator opens the protective isolation valve and switches on the gas blower of the hydrogen removal system. The control is exercised from the latter system's control board. The mode of post-accident operation of this system is identical with operation under normal conditions except that the mixture is discharged through the gas activity reduction system (GARS). In addition, there is provision for recirculation of the mixture.

If necessary (according to gas analyser readings), intensified evacuation from any location of the accident confinement system can be organized by switchibng on the stand-by gas blower or the stand-by sub-system.

Nitrogen supply is provided for cleaning the hydrogen removal system equipment and for fire fighting.

Gas analysers carry out continuous automatic monitoring of hydrogen concentration in all locations of the accident confinement system. The control board of the hydrogen removal system and the unit's control board (operational part) have warning signals (acoustic and luminous) for rises in hydrogen concentration in the accident confinement system locations. There is also provision for control measurements of hydrogen concentration in these locations by manual sampling on chromatographs. In the hydrogen removal system proper the flow rates, temperature and radioactivity are measured. All the data are displayed on the system's control board.

The monitoring and control system has three independent channels. The devices of the hydrogen removal system receive their power supply from the sources of the corresponding safety sub-systems.

The sprinkler system supplies cooling water to the condensers of the hydrogen removal system.

2.10.3. Power-supply safety systems

2.10.3.1. Power supply for the plant's own requirements (house power supply)

Own-requirement users at the plant (house users) can be divided up into the following groups depending on the extent to which they need a reliable power supply:

- The first group comprises users which cannot tolerate an interruption in power supply or can tolerate interruptions of between fractions of a second and several seconds in any regime - including the regime of total loss of alternating current from working and reserve own-requirement transformers - and for which a power supply is absolutely essential after a scram (tripping of the emergency protection system);

- The second group comprises users which can tolerate interruptions in the power supply of between tenths of a second and tenths of a minute in the same regimes and for which a power supply is absolutely essential after a scram;

- The third group comprises users which do not require a power supply in the event of total loss of power from working and reserve own-requirement transformers and which during normal reactor operating conditions can tolerate an interruption in power supply for the time needed for switching from the working own-requirement transformer to the reserve transformer.

For own-requirement users at the plant there are two independent and interchangeable (mutually redundant) power supplies: a normal working supply and a reserve supply from the working and reserve own-requirement transformers.

For users of the first and second groups there is an additional supply from a third independent emergency source.

The emergency power sources consist of the following:

(a) A storage battery with static transformers for users of the first group;

(b) Automatic diesel generators for users of the second group.

The circuit diagram of own requirements is shown in Fig. 2.52.

2.10.3.2. Circuit diagram of own requirements of 6 kV for users of the third group

The third group of users comprises the main circulation pumps, the feed pumps, the first- and second-stage condensate pumps, the mechanisms of reactor auxiliary systems and of the machine room and other systems involved in reactor operation in normal condition.

For providing third-group users with power, the plant has 6 kV and 380/220 V 50 Hz circuits which are supplied by the working own-requirement and reserve transformers.

In normal operating regime the 6 kV supply for own-requirement users is provided by two 63 MV working own-requirement transformers at a voltage of 20/6.3-6.3 kV with two separated windings. The own-requirement transformer has a dead connection between two switches in the unit generator circuit which are switched in series. Each own-requirement unit transformer is connected to two 6 kV working sections for supplying own-requirement users.

The fact that the circuit of each generator has two switches means that the working own-requirement transformers can be used for starting up the unit and for shutting it down if the generator circuit is defective, and that the own-requirement power supply will be maintained if the unit is switched off for technical reasons, and also in the event of any electrical breakdowns at the reactor higher up than the generator switches, in particular, short-circuiting in the unit transformers.

The two-switch circuit is such that the run-down of the turbogenerator can be used for supplying power to the feed pumps, which provide the water supply to the core during the first 45 seconds from the beginning of a design-basis accident in the event of loss of own-requirement power supply from the higher-voltage (external-requirement) circuit. The 20 kV switches are disconnected from the unit transformer by the separate operation of the turbogenerators when they are running down.

In this case the generator voltage varies in proportion to its rpm by means of a special "run-down unit" which is connected to the turbogenerator excitation regulator; this "run-down unit" ensures that the rotor current in the generator is maintained constant with the decrease in frequency.

The "run-down unit" is switched on when the design-basis accident signal is given and the turbine shut-off valves close.

Redundancy is achieved with the own-requirement unit transformers by means of a 63 MVA reserve transformer connected by an open-air line to the 330 kV outdoor switching station.

The 6 kV users of each turbogenerator are connected to the corresponding sections of the own-requirement unit transformer, and the reactor-part and whole-unit users are distributed evenly between the sections of the two own-requirement unit transformers; the electric motors of interchangeable (mutually redundant) mechanisms are connected to different sections.

2.10.3.3. Emergency power supply system

Users of the first and second groups receive power from the emergency power supply system, the power sources for which, in addition to the working and reserve own-requirement transformers, are independent (storage batteries with static transformers and diesel generators).

The users of the first and second groups are subdivided into users from safety-related process systems and "whole-unit users" for which a power supply is absolutely essential, even when the plant's own-requirement power supply has been totally cut off.

2.10.3.4. Circuit diagram for the 0.4 kV emergency power supply system for the first group and for the direct current circuit for safety systems

Safety system users of the first group include the isolating mechanism for the accident localization (containment) system and hydrogen removal system, the fast-acting valves and gate valves on emergency core-cooling system (ECCS) lines and monitoring, protection and automatic control devices of safety systems.

Three independent power supply sources (storage battery with static inverter transformers and 6 kV and 0.4 kV own-requirement sections) are foreseen for supplying power for users of the first group of each safety sub-system.

The direct-current distribution panel of the safety sub-system receives power from a rectifier connected to the 0.4 kV section of the emergency power supply for the second group (NNBS),and when power is lost in this section from the storage battery operating in the "buffer" regime.

Users of 0.4 kV alternating current of the first group are connected to a 0.4 kV section (NNAS) which receives power from the direct-current distribution panel through static inverter transformers.

In normal reactor operating regime the direct-current distribution panel and the NNAS 0.4 kV section of each safety sub-system are connected to the monitoring and control devices and automatic control systems of the corresponding safety sub-system, and in design-basis accident regime they have to cope with an additional load, that of the electrical drives of gate valves and other valves of the ECCS and the accident localization (containment) system. In order to prevent overloading of the inverter transformers above the permissible levels with the current for starting up the electrical drives for gate valves, the drives are actuated in stages following the design-basis accident signal.

2.10.3.5. Circuit diagram for the 0.4 kV whole-unit emergency power supply system for the first group and for the direct-current circuit

Whole-unit users of the 0.4 kV emergency power supply system for the first group are the "SKALA" central monitoring system, the control and protection system, the dosimetric monitoring system, monitoring and measuring instruments and automatic control systems of the reactor, turbine and generator and the fast-acting pressure-reducing mechanism.

In order to supply power to users of the whole-unit emergency power supply and direct-current system, there are two whole-unit emergency power facilities, each of which include the following: power sources (storage battery and static inverter transformers), direct-current distribution panel, first-group 0.4 kV emergency power supply distribution panels and own-requirement 6 kV and 0.4 kV sections.

The direct-current distribution panel of each whole-unit emergency power supply facility receives power from a rectifier connected through a 6/0.4 kV transformer to the 6 kV section of the second-group emergency power supply and, if power in this section is lost, from a storage battery operating in the "buffer" regime.

First-group 0.4 kV users are linked to NNA sections through TKEO thyristor commutator systems. NNA sections receive power through static inverter transformers from the direct-current distribution panel.

Each user of the whole-unit circuit of the emergency power supply system has two power sources. For the second source either the circuit is used or another inverter transformer.

For users which cannot tolerate an interruption in power supply of more than 10-20 ms (the "Skala" central monitoring system and control and protection systems), the change-over to reserve power source is performed by a TKP thyristor switching commutator, which changes the power supply to the user over from one source to another in 10 ms. For users which can tolerate an interruption in power supply of up to 100-200 ms, there are relay-contact switching devices.

Systems for which redundancy is foreseen (the "A" and "B" feeder, the "Skala" central monitoring system, the 1000 Hz and 400 Hz "Skala" transformers, the emergency protection system control panel units, etc.) are supplied by one of the emergency power supply facilities.

The power supply to systems for which redundancy is not foreseen (control and measurement instruments, automatic control systems, regulation systems, etc.) is provided by two emergency power supply facilities.

Direct-current users (emergency protection system control panels, warning systems, etc.) receive their power from both direct current distribution panels. Switching over from one distribution panel to another is performed manually.

2.10.3.6. Circuit diagram for the 6 kV and 0.4 kV own-requirement emergency power supply systems for the second group

Users from the second group of safety systems are mechanisms of the ECCS and the accident localization (containment) system.

Whole-unit users of the second group are mechanisms of the auxiliary turbogenerator systems, certain auxiliary reactor systems (intermediate circuit, cooling systems of the fuel cooling pond, blowdown and cooling system, and so on).

For supplying power to users of the second group, there are three emergency power supply sections of 6 kV and 0.4 kV (in accordance with the number of safety sub-systems). Whole-unit users are distributed over the safety sub-system sections.

Diesel generators with a capacity of 5500 kW were used as an independent power supply for the 6 kV emergency power supply sections at the fourth unit of the Chernobyl' nuclear power plant. The startup time of the diesel generator is 15 seconds.

The diesel generators take up the load in stages. The time for each stage to be taken up is 5 seconds.

The diesel generator is started up automatically, with the load taken up in stages, upon receipt of the design-basis accident signal or current loss signal.

When one of these signals is received by the circuit for automatic startup of the diesel generator with take-up of the load in stages, the following commands are issued:

- Startup of diesel generator;

- Switching off of both section switches linking the working own-requirement 6 kV section with the emergency power supply section;

- Switching-off of the load on the 6 kV emergency power supply section ("clearing of the section");

- Blocking, by automatic stand-by startup of mechanisms connected to the particular emergency power supply section.

After the diesel generator has been started up and connected to the section, automatic switching-on of the own-requirement mechanism switches takes place in stages at 5-second intervals in accordance with the schedule of load take-up in stages (Fig. 2.53).

Depending on the signal received, the circuit automatically switches on in stages the corresponding mechanisms needed for a design-basis accident or for loss of current by own-requirement users.

For the 0.4 kV users of the second group there are 0.4 kV sections of the safety systems (NNBS) and whole-unit 0.4 kV sections which are independent of them. Each section receives power from the corresponding 6 kV section of the safety system through a 6/0.4 kV transformer.

The number of NNBS 0.4 kV safety system sections corresponds to the number of operating safety sub-systems.

Emergency power supply 6/0.4 kV transformers for the second group are a stage of load take-up by diesel generators that cannot be switched off.

No mutual redundancy is foreseen between the 6 kV and 0.4 kV sections of the second group since there is redundancy of the users themselves.

2.10.4. Controlling safety systems

The controlling safety systems are designed to switch on automatically devices of the protection, localization (containment) and power-supply safety systems and to monitor their operation.

For each of the three safety sub-systems there is an independent controlling safety system.

A controlling safety system issues the design-basis accident signal if the pressure in the containment, lower water line or drum separator enclosures rises to 5 kPa with confirmation of the decrease in the level of the separator by 700 mm from the nominal level or decrease in the gradient between the pressure header of the main circulation pumps and the drum separator to 0.5 MPa.

In order to increase their reliability, all three controlling safety systems have been constructed independently from one another, i.e. each of these controlling safety systems has its own engineered structures and electricity supply, and separate areas for engineered structures and cable conduits. Four sensors are provided for issuing the signal indicating high pressure in the containment, drum separator and lower water pipe enclosures. If two or more of the sensors are triggered, a signal is issued.

The signal indicating a pressure decrease in the drum separator and also the signal indicating a decrease in the pressure gradient between the pressure header and the drum separator are issued when any two of the sensors provided are triggered.

The design-basis accident signal is issued independently for either half of the reactor.

When the design-basis accident signal is issued, the controlling safety system issues instructions for output actions for switching over the corresponding safety system devices and for switching on the diesel generators and mechanisms for taking up the load in stages.

The design provides for the possibility of remote control of the safety system, for which the operational part of the reactor control panel has control switches for each controlling safety system.

In this case the half of the plant in which there is an accident is selected automatically, for which use is made of the information part - which is independent of the controlling safety system - from the emergency protection system triggering circuit for process reasons.

For purposes of monitoring the correct functioning of the controlling safety system there are warning lights and acoustic signals indicating that instruments are defective.

The safety system is monitored and controlled from the safety panels set up in the area of the operational circuit of the reactor control panel and on a redundant control panel.

The safety panels contain devices for controlling the pumps of the emergency core cooling system (ECCS), the accident localization (containment) system, the safety system equipment, instruments for monitoring the flow of ECCS water into the reactor, etc.

Figure 2.54 shows a flow chart of the controlling safety system.

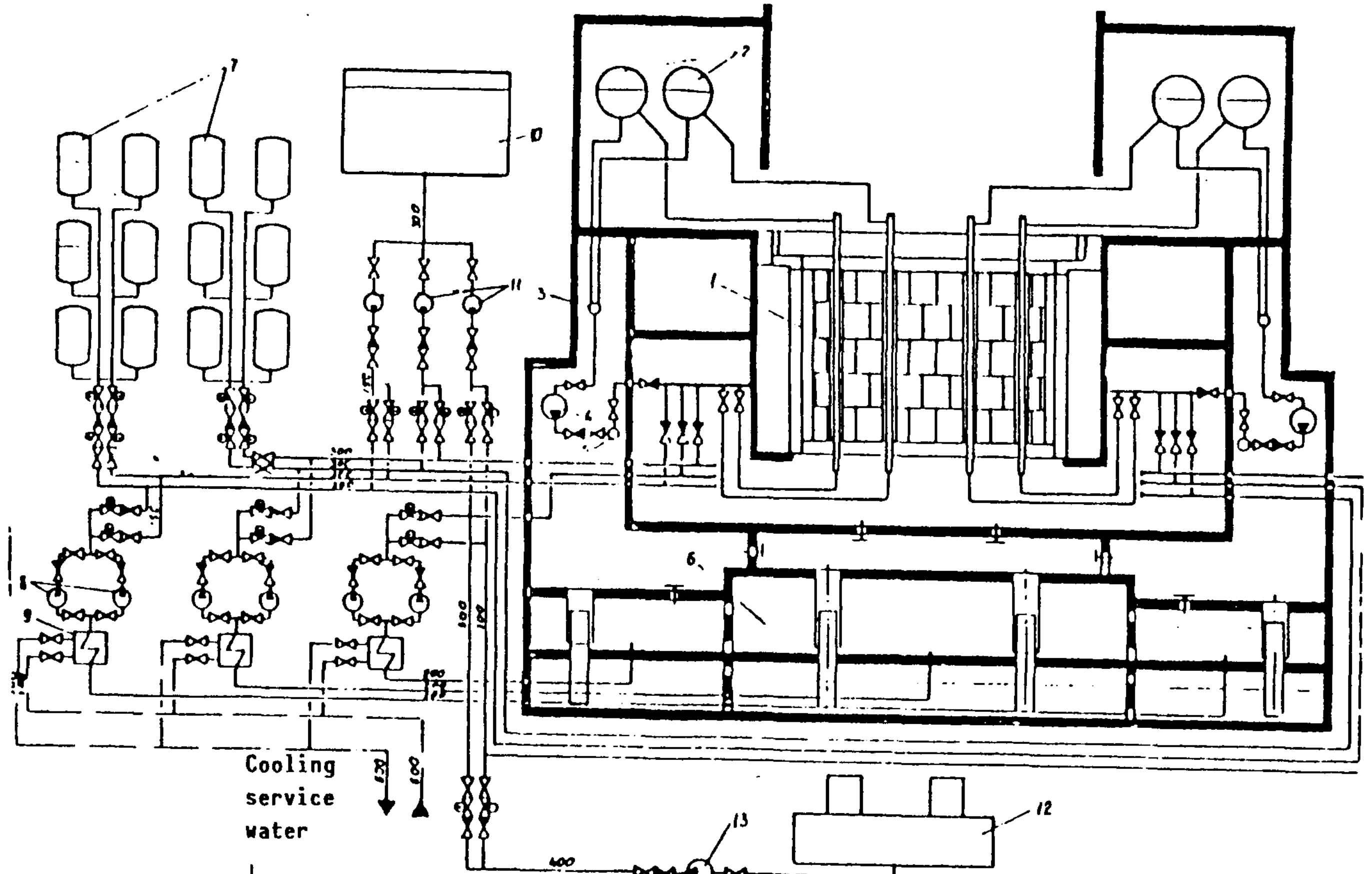
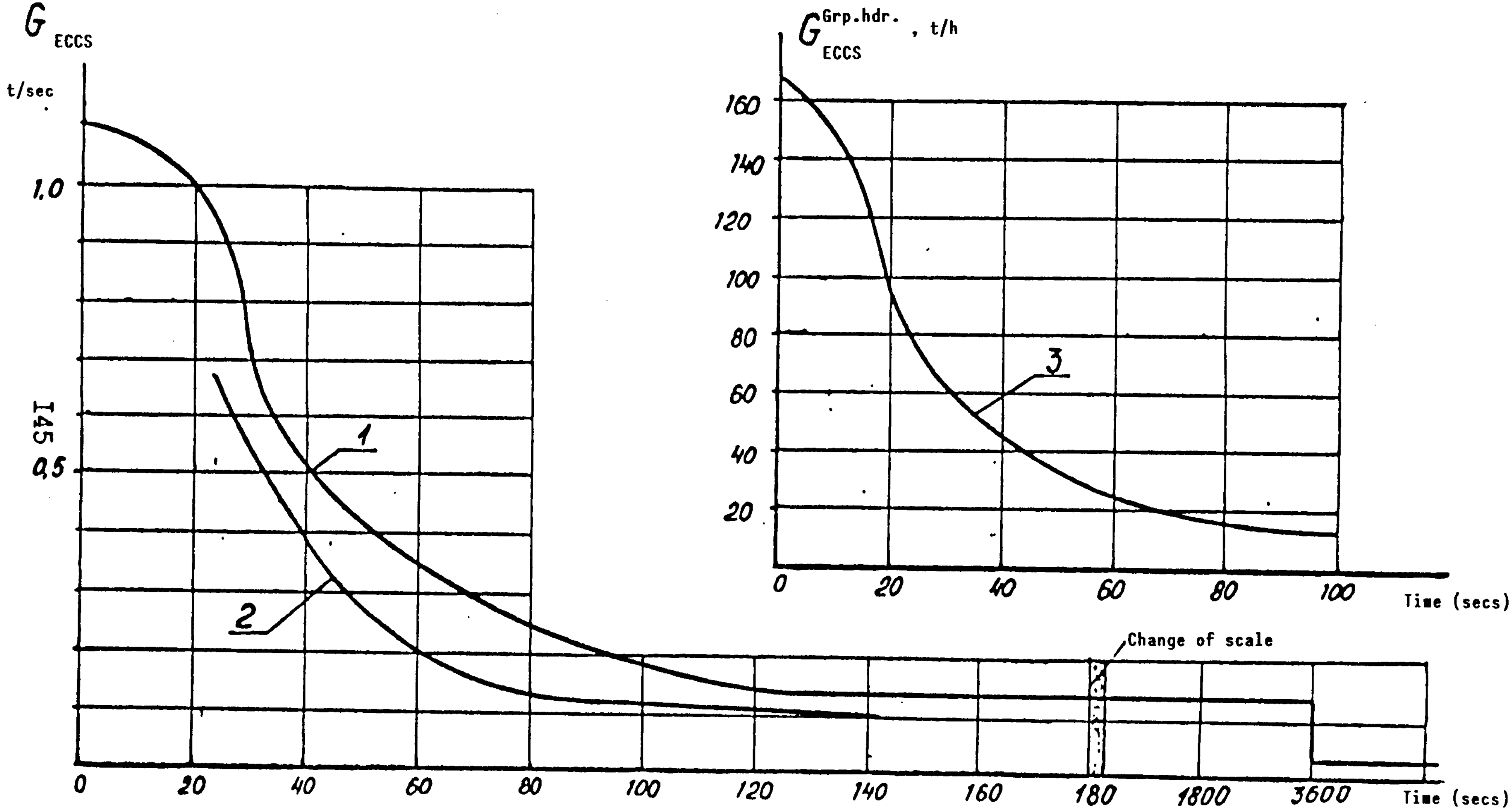


Fig. 2.43 Schematic drawing of the reactor's emergency cooling system.

1. Reactor 2. Steam separators 3. Suction header 4. Main circulation pump
5. Pressure header 6. Pressure suppression pool 7. ECCS vessels
8. ECCS pumps for cooling the damaged half of reactor 9. Heat exchangers
10. Clean condensate container 11. ECCS pumps for cooling the undamaged
half of the reactor 12. De-aerator 13. Feed pump.

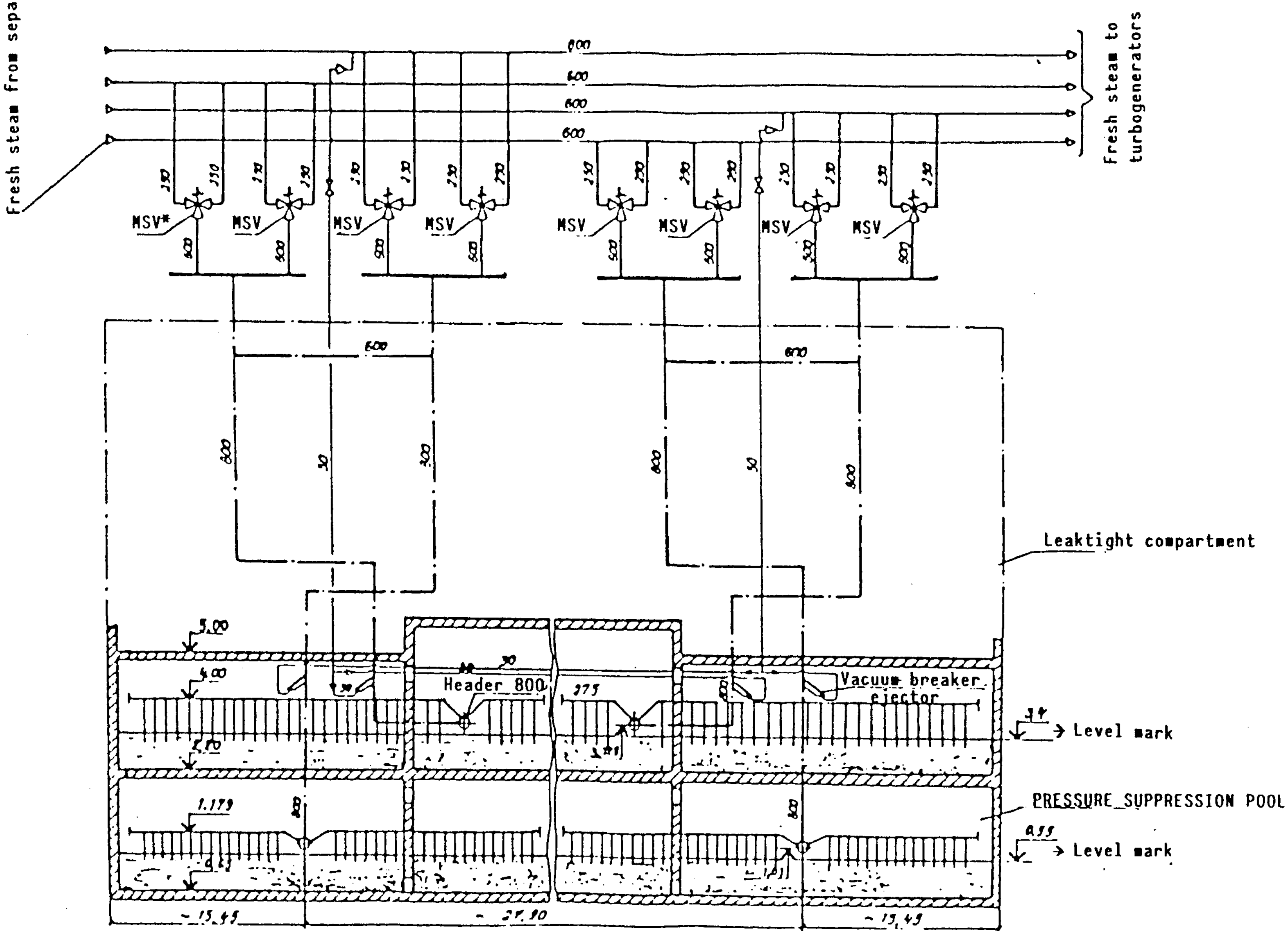
Required supply of water to the damaged half of the reactors in different accident situations



Type of accident: 1. Rupture of a Du 900 mm pressure header
2. Rupture of a Du 300 mm down pipe
3. Rupture of a group header as far as the non-return valve

Fig. 2.44

SYSTEM FOR DISCHARGING STEAM FROM THE MAIN SAFETY VALVES INTO THE PRESSURE SUPPRESSION POOL



*/ MSV = main safety valve.

Fig. 2.45.

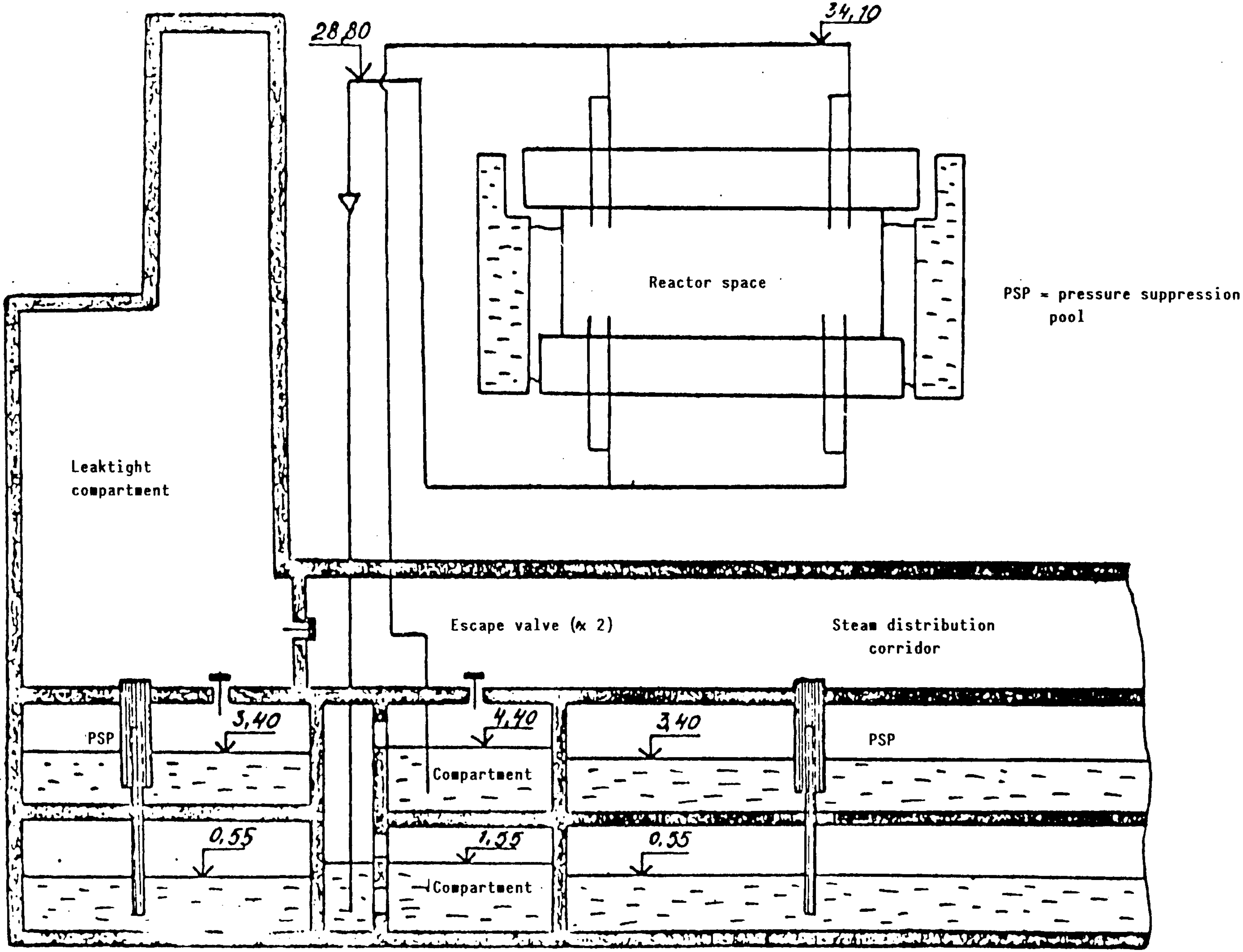


Fig. 2.46 : System to protect the reactor space from excess pressure

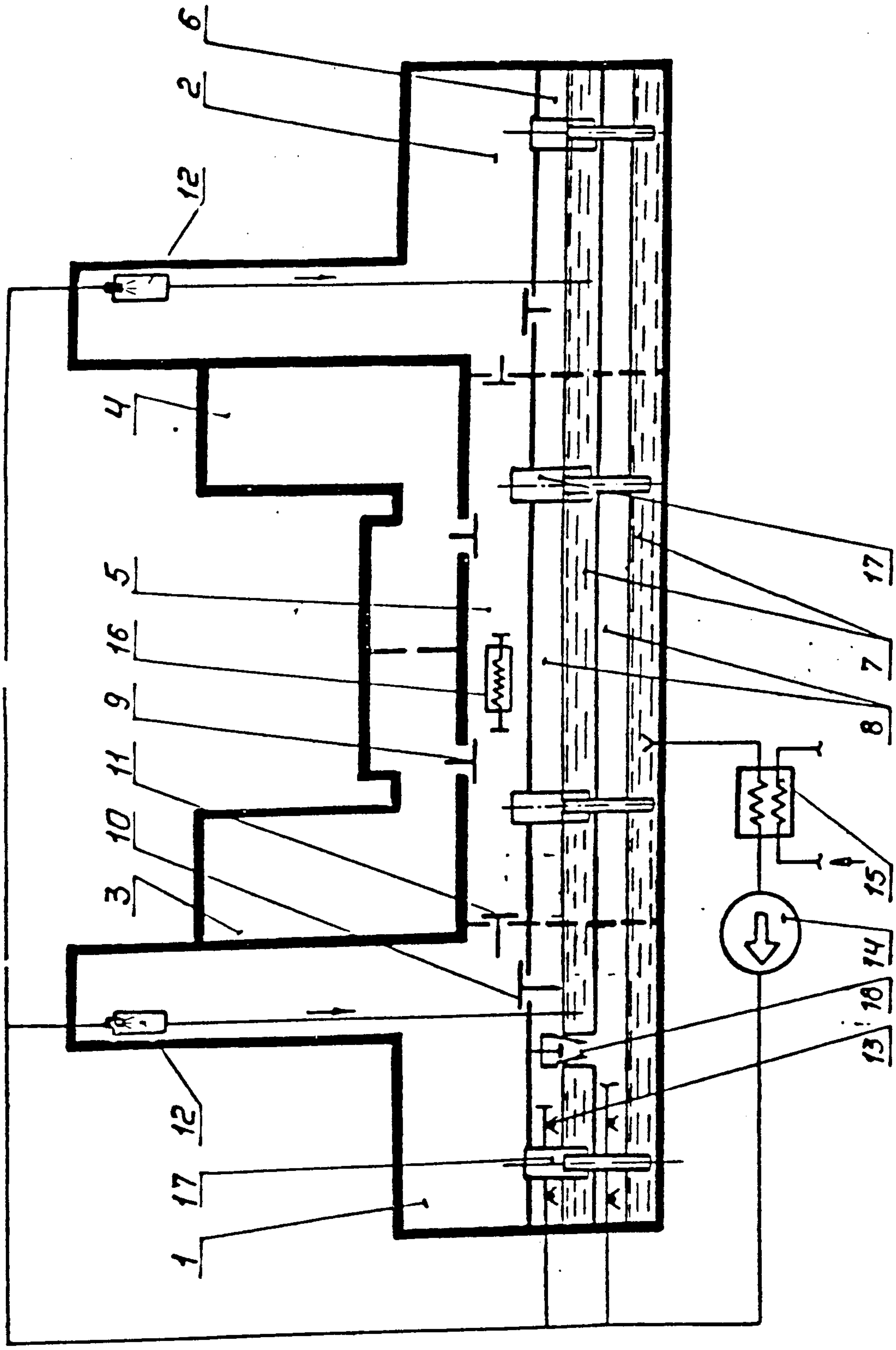
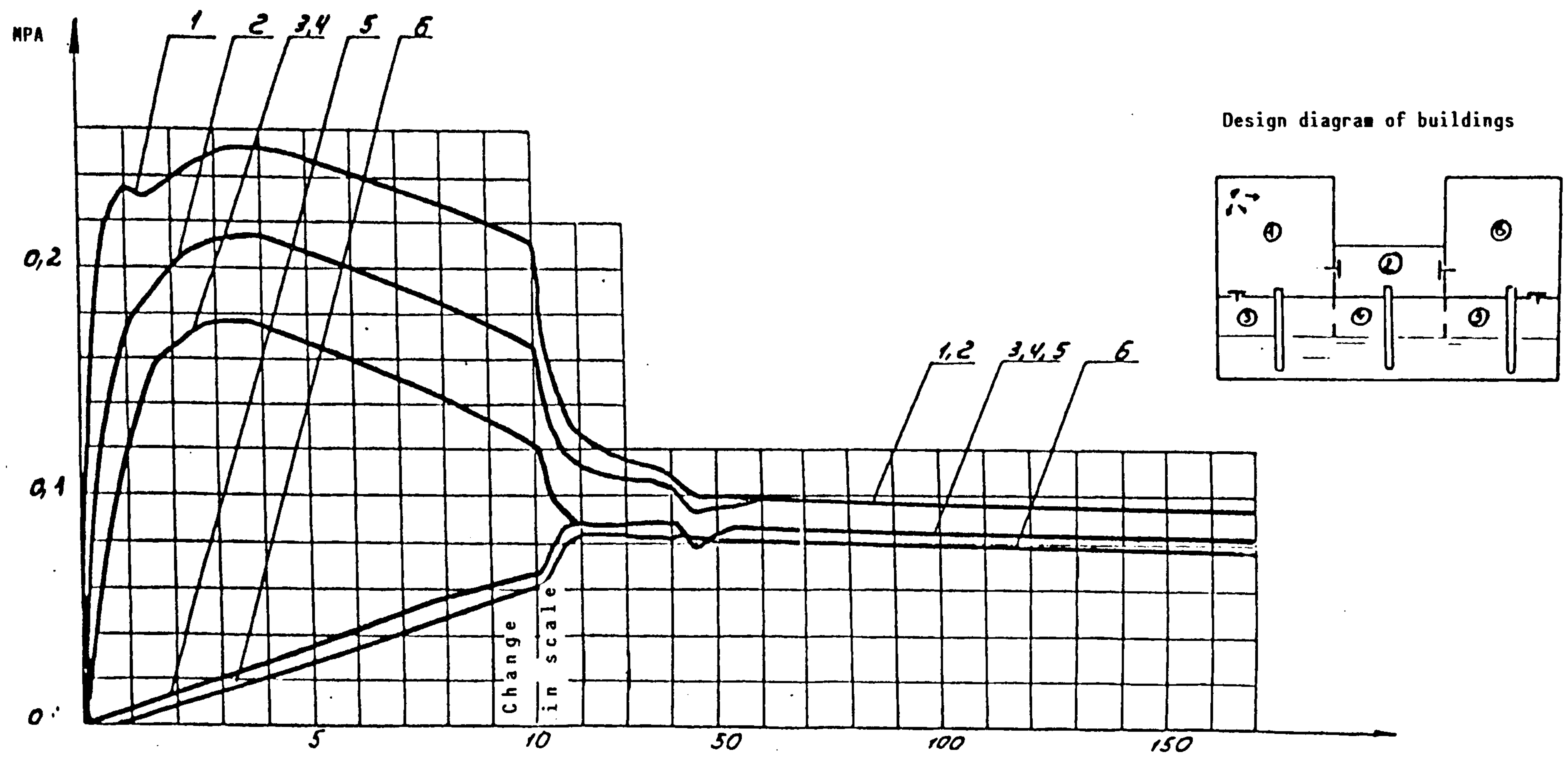


Fig 2.47 Schematic diagram of the confinement system

Fig 2.48 Pressure changes in the buildings of the second unit of the Smolensk nuclear power station and the fourth units of the Kursk and Chenobyl' nuclear power stations during pressure header failure



2.48

Design diagram of buildings

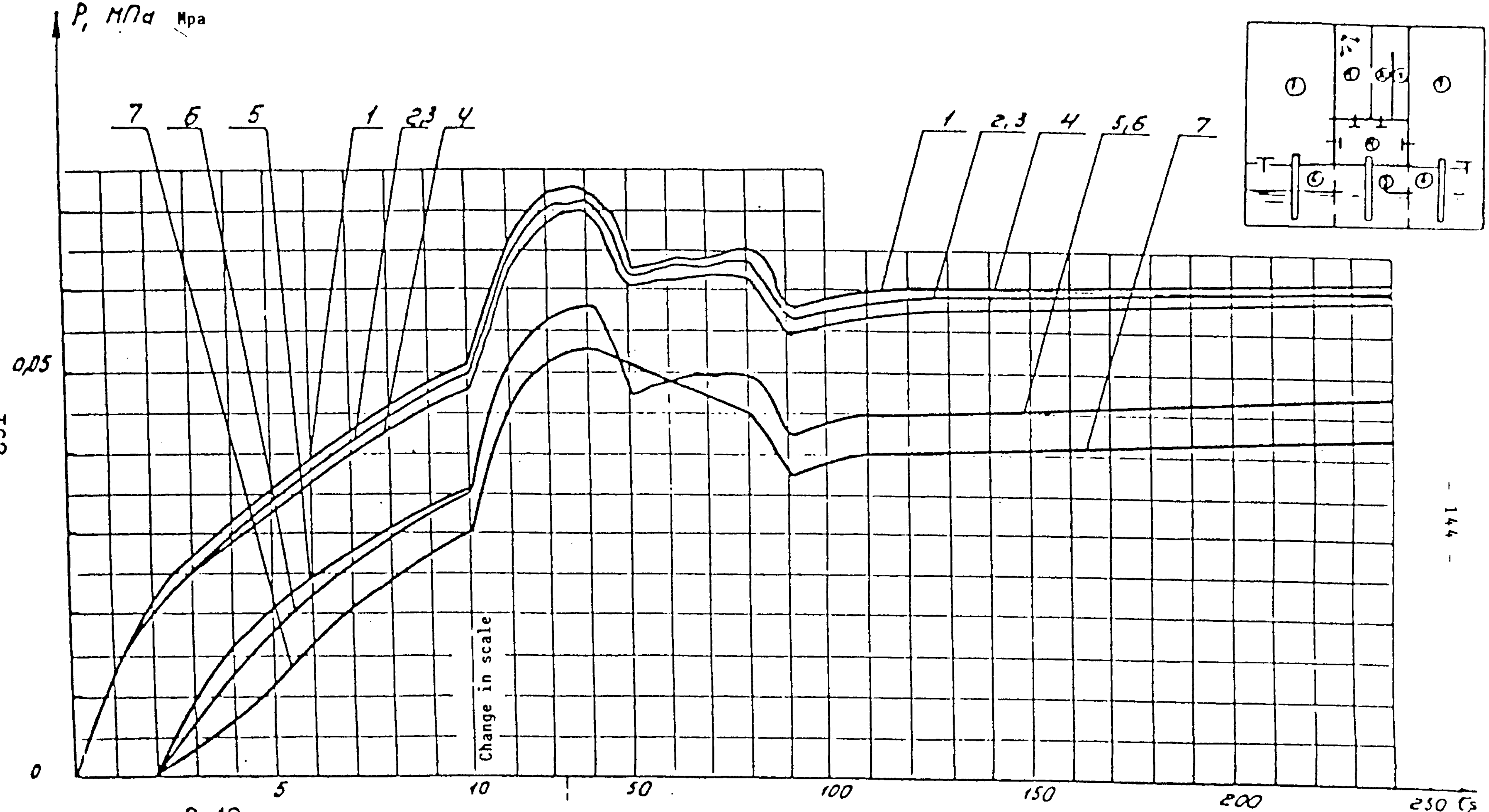


Fig 2.49 Pressure changes in the buildings of the second unit of the Smolensk nuclear power station and the fourth units of the Kursk and Chernobyl' nuclear power stations during failure of the disturbing group header

Entry of H_2 into the accident confinement area

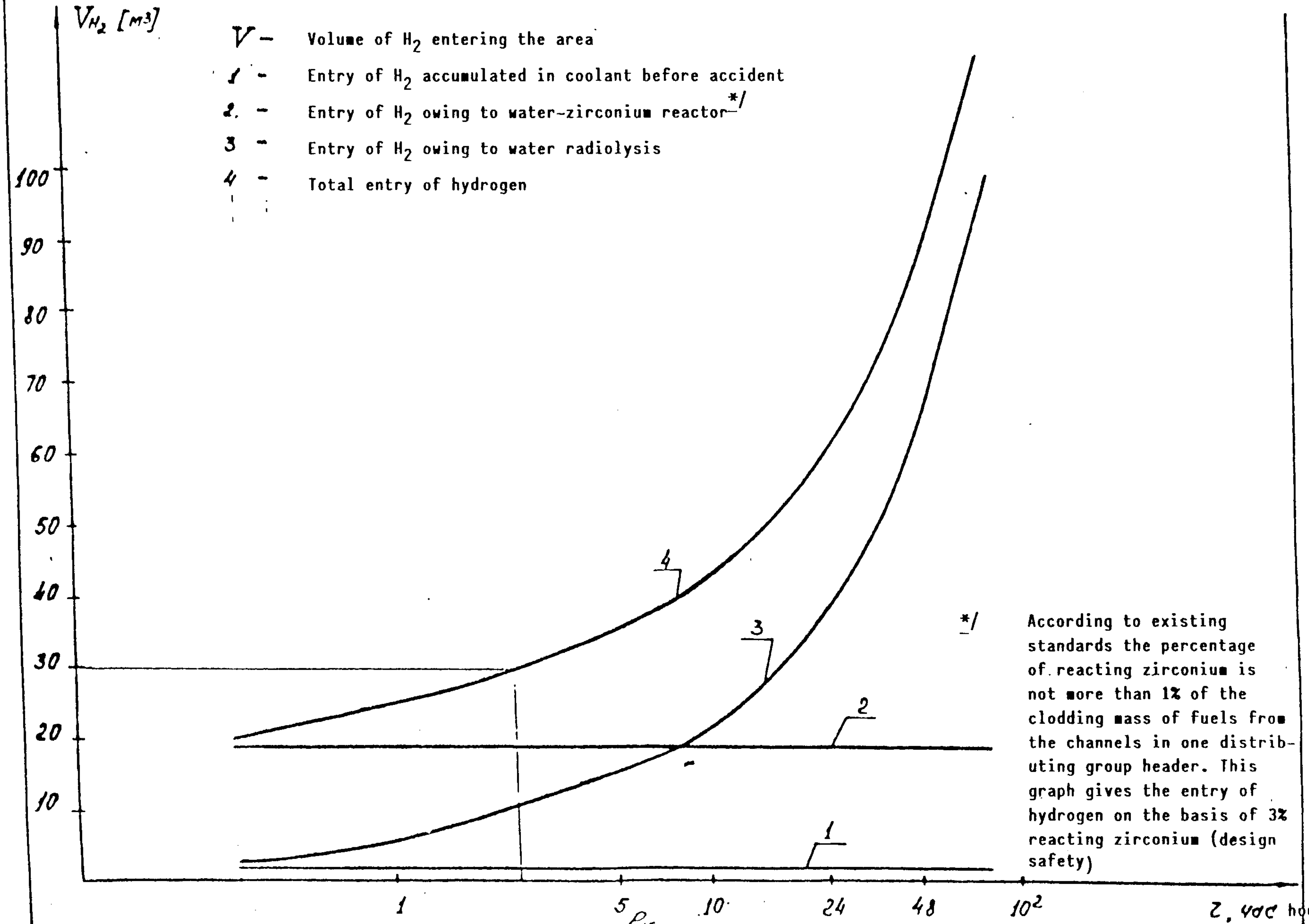


Fig 2.50

LEGEND TO FIG 2.51

HYDROGEN REMOVAL SYSTEM

1. Nitrogen supply
2. Discharge to vent flue via V-1, A
3. Discharge into GARS or to vent flue without GARS
4. To trap
5. Water outlet - water inlet
6. Air suction
7. Air suction
8. Sub-System No. 1
9. Sub-System No. 2
10. Sub-System No. 3
11. Steam-gas discharge component
12. Downcomer shaft
13. Pressure suppression pool
14. Downcomer shaft
15. Discharge for gas analysers
16. Measurement of hydrogen concentration

Hydrogen removal system

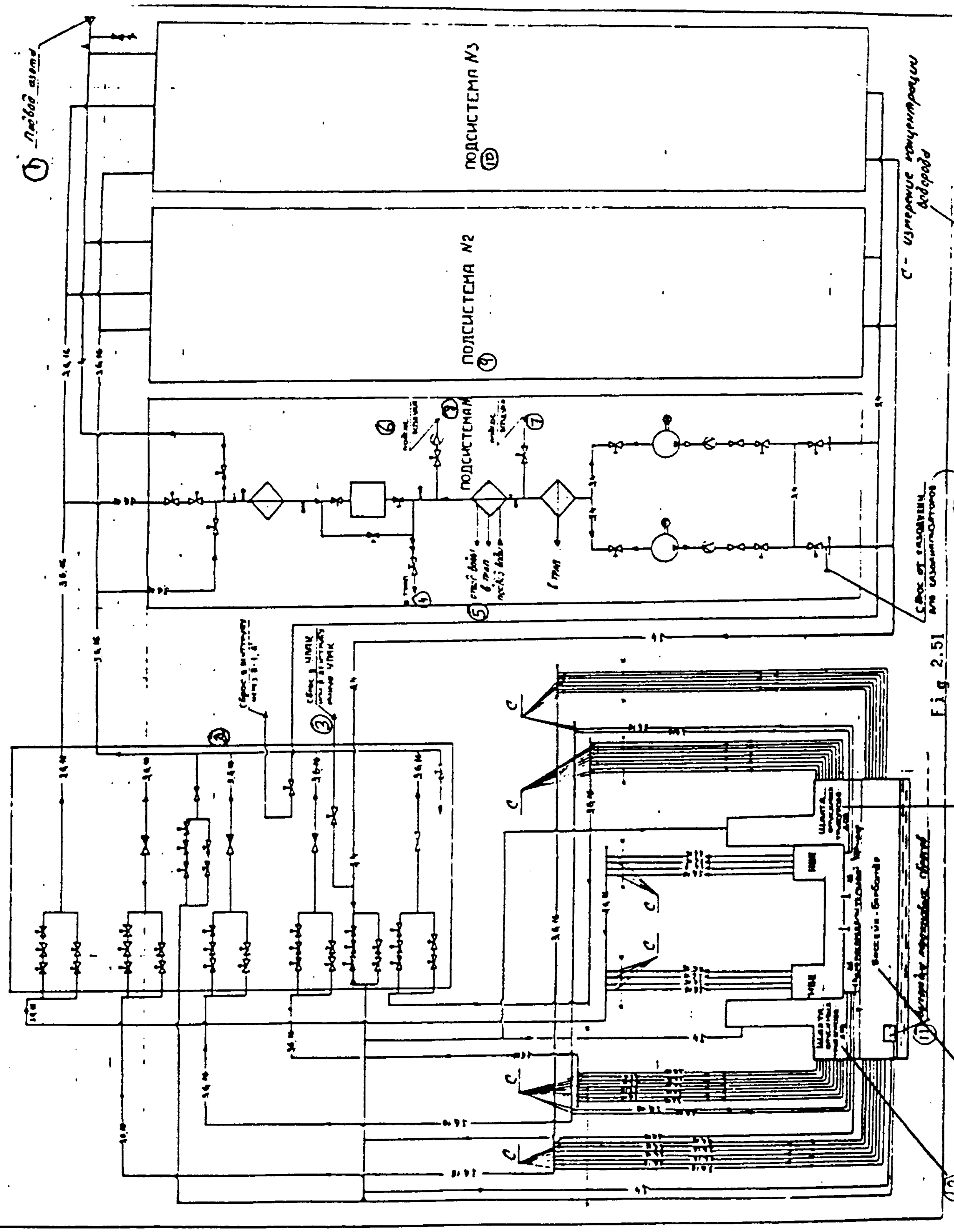


Fig. 2.51

Key to Fig. 2.52 [This figure has been translated to the extent that the quality of the original permits]

Figure caption: Own-requirement circuit diagram for the fourth unit at the Chernobyl' NPP

1. 63 MVA 350/6.3-6.3 kV reserve transformer
2. To 330 kV outside switching station
3. 63 MVA 20/6.3-6.3 kV working own-requirement transformer
4. Unit transformer
4a. To 750 kV outside switching station
5. 20 kV
6. 6 kV reserve busbar
7. 6 kV sections of the third emergency power supply group
8. 7RB
9. 7RA
10. 8RB
11. 8RA
12. Main circulation pumps
13. First-stage condensate pump
14. Second-stage condensate pump
15. Feed pump
16. Other mechanisms
17. Working own-requirement transformers
18. Other own-requirement transformers
19. To 6 kV section of the third unit
20. 6/0.4 kV reserve transformer
20a. 0.4 kV reserve busbar
21. 6 kV sections of the third emergency power supply group
22. Diesel generator 2
22a. Diesel generator 1
23. Diesel generator 3
24. 7RNB
25. 7RNA
26. 8RNA
27. Process water pump
28. Non-accident-half ECCS pump
29. Accident-half ECCS pump
30. Emergency protection system channel cooling circuit pump
31. Sprinkler-cooling system pump
32. Emergency supply transformer 82 TNP
33. 0.4 kV sections of the second emergency supply system group
34. 2NNBC
35. 72NNB
36. 71NNB
37. 92NNB
37a. 91(?)NNB
38. 3NNBC

39. 0.4 kV users of the second safety system
40. 0.4 kV users of the first safety system
41. Electric motors of ventilation systems
42. Whole-unit users
43. Whole-unit emergency supply system facilities:
44. Emergency supply system facility for second safety system (SD II)
45. Emergency supply system for first safety system (SB I)
46. First
47. Second
48. Emergency supply system facility for third safety system (SB III)
49. Monitoring and control instruments, automatic control systems, regulators
50. Emergency protection system and "Skala" central monitoring system users
51. Electric drives of gate valves
52. 21NNA
53. 22NNA
54. NNB
55. 11NNA
56. 12NNA
57. NNB
58. 11NNA
59. 12NNA
60. 13NNA
61. 14NNA
62. 21NNA
63. 22NNA
64. 23NNA
65. 24NNA
66. 0.4 kV sections of first emergency power supply group
67. 2NNAS
68. 1NNAS
69. 3NNAS-1
70. 3NNAS-2
71. Electric motors of isolating mechanism and ECCS
72. Automatic control, protection and regulation devices
73. 2 MPS
    PTS-63
74. 2 VUS
75. 1 MPS
    PTS-63
76. 1 VUS
77. 1 VU
78. 11 MP
79. 12 MP
80. 13 MP
81. 14 MP
82. 1 VUP
83. 91 NNB

84. 21 MP
85. 22 MP
86. 23 MP
87. 24 MP
88. 2 VU
89. 2 VUP
90. 91 NNV
91. 3 MPS-1
    PTS-63
92. 3 MPS-2
    PTS-63
93. 3 VUS
94. Direct current distribution panel 3 S (3ShchPTS)
95. Direct current distribution panels
96. = 232 V
97. Direct current distribution panel 2 S (2 ShchPTS)
98. = 232 V
99. Direct current distribution panel 1 S (1ShchPTS)
100. Direct current distribution panel 1 (1ShchPT) = 253 V
101. = 232 V
102. Direct current distribution panel 2 (2ShchPT) = 253 V
103. = 232 V
104. = 232 V
105. Direct current users
106. 3ABC
     SK-16
     108 elec.
107. 1ABC
     SK-16
     108 elec.
108. 1AB
     SK-52
     118 elec.
109. From 108 elec.
110. Emergency power supply transformer 2 S (2TNPS)
111. Emergency power supply transformer 1 S (1TNPS)
112. Emergency power supply transformer 3 S (3TNPS)
113. Emergency power supply transformer 72
114. Emergency power supply transformer 73
115. Emergency power supply transformer 71
116. Emergency power supply transformer 93
117. Emergency power supply transformer 92
118. Emergency power supply transformer 91
119. Emergency power supply transformer 81 [?]
120. Transformer 225 T

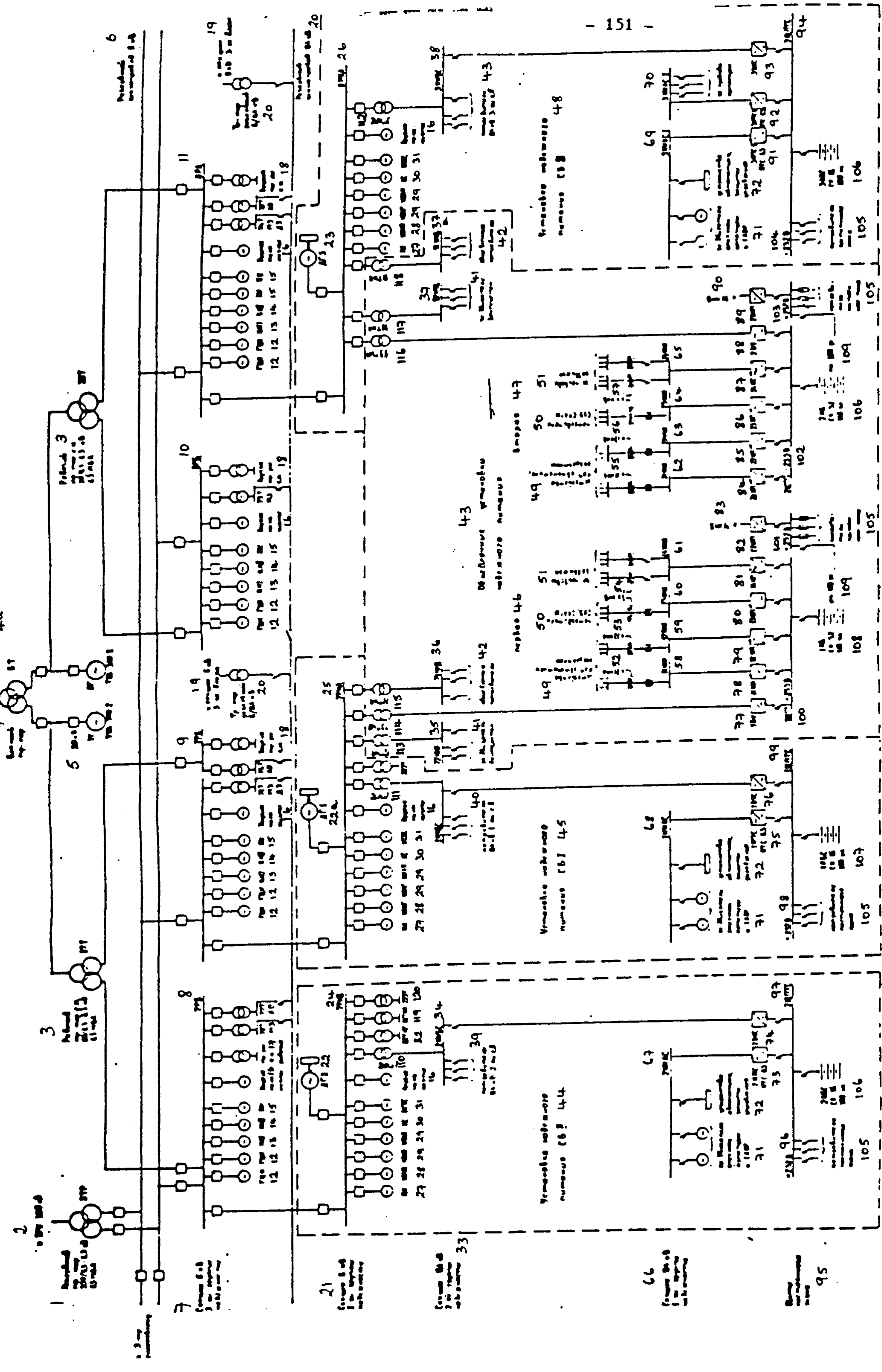


Fig. 2.52

Key to Fig. 2.53

Horizontal captions (see column numbers on figure)

1.	No.
2.	Name of emergency power supply user
3.-13.	Starting data
3.	Number attached to one safety sub-system
5.-6.	Power (kW)
5.	Rated
6.	Consumption
7.	Nominal current, I_n (A)
8.	Startup current, I_{st} (A)
13.	rpm
14.-17.	Theoretical values
14.	Theoretical power (kW)
15.	Startup power (kW)
16.	Slippage (S_{nom})
17.	Cosϕ startup
18.-32.	Design-basis accident regime
19.-29.	Switching stages
18.	Number of motors started up
19.	15 s I
20.	20 s II
21.	25 s III
22.	30 s IV
23.	35 s V
24.	40 s VI
25.	45 s VII
26.	50 s VIII
27.	55 s IX
28.	60 s X
29.	65 s XI
30.-32.	$\frac{P_{rated}\ (kW)}{I_{rated}\ (A)}$
30.	Per stage
31.	Up to 10 min
32.	Up to 30 min and beyond
33.-47.	Loss of current by own-requirement users
33.	Number of motors started up
34.-35.	Switching stages
34.	15 s I
35.	20 s II
36.	25 s III
37.	30 s IV
38.	35 s V

39. 40 s VI
40. 45 s VII
41. 50 s VIII
42. 55 s IX
43. 60 s X
44. 65 s XI
45.-47. $\underline{P_{rated}\ (kW)}$
I_{rated} (A)
45. Per stage
46. Up to 10 min
47. Up to 30 min
48. Notes: [opposite vertical columns 12, 13:] Not started up in stages

Caption at bottom right of figure

This table was compiled for one safety sub-system. The tables for the other two sub-systems are similar.

Vertical captions (see line numbers on figure)

1. Transformers for emergency supply system and plant's own requirements
2. Process water pump
3. Non-accident-half ECCS pump
4. Sprinkler-cooling system pump
5. Accident-half ECCS pump
6. Emergency protection system channel cooling circuit pump
7. Reserve
8. Clean condensate pump
9. Emergency feed pump
10. Fire pump
11. Emergency supply system transformer
12. Circuit cooling pump
13. Tank pump

№ п/п	Наименование потребителей надежного питания	Количество на одну подсистему безопасности	Тип электродвигателя	Мощность (кВт): Установленная	Мощность (кВт): Потребляемая	Номинальный ток $I_н$ (А)	Пусковой ток $I_п$ (А)	η %	cos $φ_н$	$К_п = \frac{I_п}{I_н}$	$К_м = \frac{M_п}{M_н}$	Скорость вращения (об/мин)	Расчетная мощность (кВт)	Пусковая мощность (кВА)	Скольжение $S_н$ (о.е.)	cos $φ_{пуск}$
1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17
1	Трансформатор надежного питания и с.н. АЭС	4				100	200						800	1600		
2	Насос технической воды ... - НА	1	АВ-13-39-16	1250	1100	160	830	93,1	0,8	5,2	0,8	370	1170	1900	0,015	0,22
3	Насос САОР низконапорный - НСНА	1	2АЗМ 800/6000	800	720	88	459	95,5	0,91	5,2	1,1	2970	755	1540	0,01	0,3
4	Насос спринклерной системы - НСОС	1	А-113-4М	250	241	28	144	92	0,89	5,0	1,3	1480	241	405	0,013	0,34
5	Насос САОР высоконапорный - НОВП	2	2АЗМ 800/6000	800	720	88	459	95,5	0,91	5,2	1,1	2970	755	1540	0,01	0,3
6	Насос охлаждения приводов СУЗ - НС	1	А-113-4М	250	216	28	144	92,0	0,89	5,0	1,3	1480	227	405	0,013	0,34
7	Резерв															
8	Насос чистого конденсата - НЧК	1	А-113-4М	250	200	28	144	92	0,89	5,0	1,3	1480	210	405	0,013	0,34
9	Аварийный питательный насос - АПН	1	2АЗМ 800/6000	800	720	88	459	95,5	0,91	5,2	1,1	2970	755	1540	0,01	0,3
10	Пожарный насос - ПЖН	1	А-113-4М	250	204	28	144	92	0,89	5,0	1,3	1480	214	405	0,013	0,34
11	Трансформатор надежного питания - ТНП	1	ТСЗ-1000	800		68	136						560	1120		
12	Насос аварийного борирования - НБ	1	2АЗМ 500/6000	500	204	54	330	95,5	0,92	6,0	1,2	1485	223	1050	0,01	0,31
13	Насос баков ППР - НППРГ	1	А-113-4М	250	200	28	144	93	0,89	5,0	1,3	1485	210	405	0,013	0,14

Режим максимальной проектной аварии

№ п/п	Количество пускающих двигателей (шт)	Включение по ступеням $P_п$ (кВт)/$I_п$ (А): I 15 сек	II 20 сек	III 25 сек	IV 30 сек	V 35 сек	VI 40 сек	VII 45 сек	VIII 50 сек	IX 55 сек	X 60 сек	XI 65 сек	$P_{уст}$ (кВт)/$I_{уст}$ (А): на каждой ступени	до 10 минут	до 30 минут и более
	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32
1		1600/200											800/100		
2	1		2700/930										1910/260		
3	1			3510/470									2725/348		
4	1				3730/492								2966/373		
5	2					6046/1298							4476/558		
6	1						4821/634						4703/583		
7								—					—		
8	—								—				—		
9	—									—			—		
10	1										5408/721		4917/616,9		
11	1											6031/747	5437/679		
12	—														
13	—														

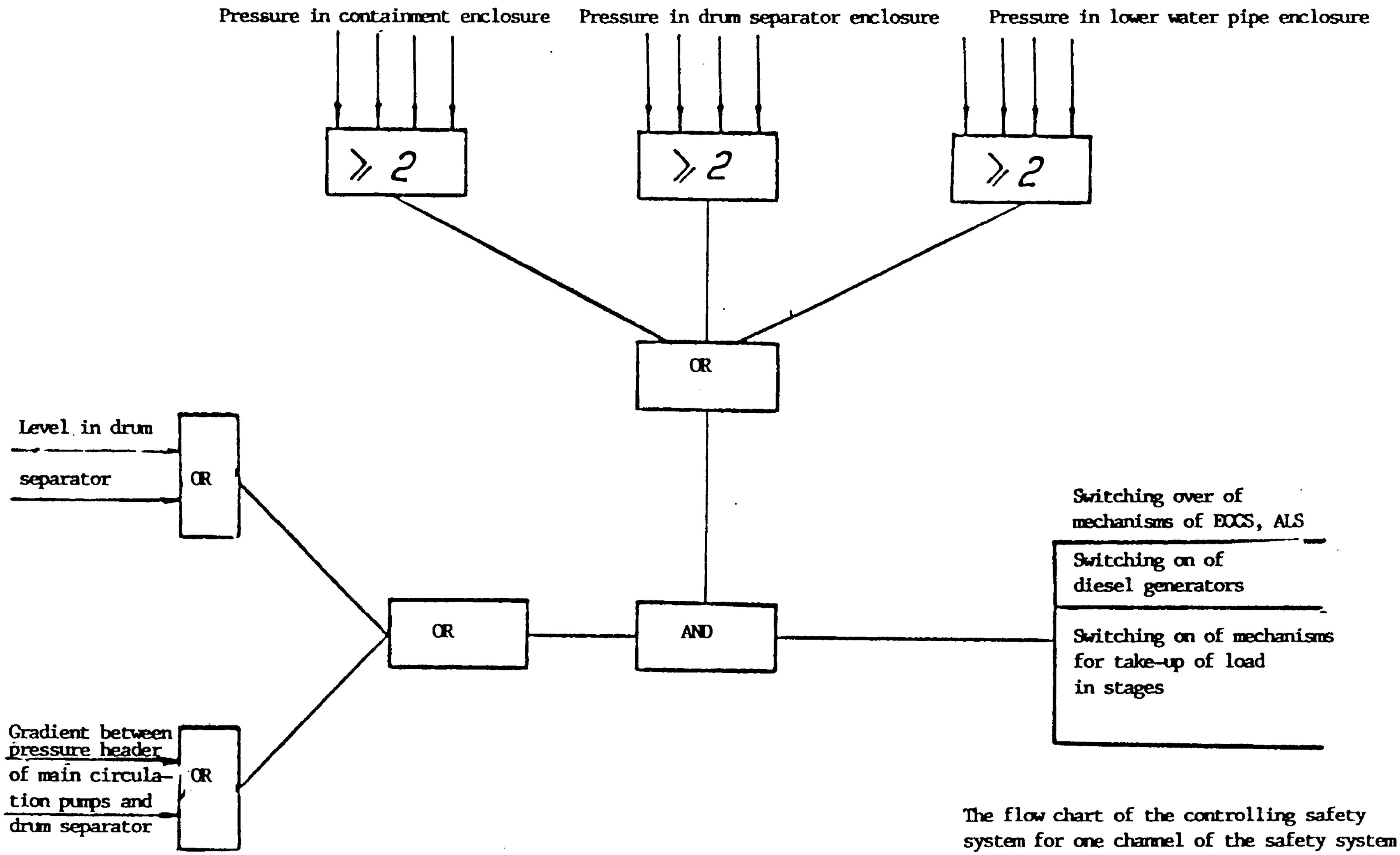
Режим обесточивания собственных нужд

№ п/п	Количество пускаемых двигателей	Включение по ступеням $P_п$ (кВт)/$I_п$ (А): I 15 сек	II 20 сек	III 25 сек	IV 30 сек	V 35 сек	VI 40 сек	VII 45 сек	VIII 50 сек	IX 55 сек	X 60 сек	XI 65 сек	$P_{уст}$ (кВт)/$I_{уст}$ (А): на каждой ступени	до 10 минут	до 30 минут	Примечание
	33	34	35	36	37	38	39	40	41	42	43	44	45	46	47	48
1		1600/200											800/100			
2	1		2700/930										1910/260			
3	—			—									—			
4	1				2375/404								2211/289			
5	—					—							—			
6	1						2816/433						2438/318			
7	—							—					—			
8	1								2843/461				2648/346			
9	1									4188/807			3403/435			
10	1										3808/379		3617/464			
11	1											4137/599	4137/512			
12	—															[illegible]
13	—															[illegible]

Данная таблица составлена для одной подсистемы безопасности. Для двух других подсистем таблица - аналогична.

Рис. 2. 51

Fig. 2.53



The flow chart of the controlling safety system for one channel of the safety system is shown. The charts for the second and third channels are similar.

2.54

Flow chart of controlling safety system

2.11. Other safety-related systems

2.11.1. Multiple forced circulation circuit (MFCC).

A description of the multiple forced circulation circuit (primary coolant circuit) and its main components is given in sections 2.6 and 2.7.

2.11.2. CPS channel cooling system

The system for cooling the CPS (control and protection system) channels is designed to ensure the requisite temperature conditions for these channels together with the control elements and servodrives of the CPS.

The system performs the following functions:

- It maintains a temperature of 40^{o} at the coolant water inlet to the control channels;

- It removes a thermal output of 28.1 MW from the channels for the CPS control elements and servodrives;

- It ensures cooling of the CPS control element and servodrive channels at a nominal flow rate for ~6 minutes when the pumps are not functioning;

- It maintains a nonexplosive concentration of hydrogen under all working conditions;

- It maintains the requisite amount of water cooling the channels and CPS servodrives;

- It ensures that there is emergency protection of the reactor if the cooling system is disrupted.

These functions are performed with allowance for single failure in the system of an active element or a passive element with moving mechanical parts.

A schematic diagram of the CPS channel cooling system is shown in Fig. 2.55.

The system constitutes a circulation loop operating by gravity.

Water from the top emergency supply tank flows by gravity into the pressure (distributing) header and is distributed through the channels. The channels contain the elements of the control and protection system and tubes containing the fission chambers and power density monitors. Some of the channels are used to control the flow of water cooling the graphite of the lateral reflector.

After flowing through the CPS channels the cooling loop water transfers its heat to the service water in the heat exchangers of the system. Depending on the temperature of the service water and degree of contamination of the heat exchange surface, the required heat removal is ensured by two heat-exchangers, two others being redundant.

After the heat-exchangers the water flows into the lower tanks of the system, in which there is automatically maintained a level ensuring stable functioning by the pumps under steady-state and transient conditions. The total volume of the lower tanks is such that they can hold all the water in the system to be received if the pumps stop.

There are four pumps for feeding the water from the lower tanks to the emergency supply tank. The delivery rate for each pumps is ~700 t/h at a pressure head of 0.9 MPa. Two of these are in operation, while two are redundant.

Provision is made for reducing the probability of all pumps failing for the same reason (the pumps are located in different rooms, have independent power supplies and so forth).

The output of the working pumps exceeds the throughput of the cooling system, hence some of the water is always being discharged from the emergency supply tank into the lower tanks of the system (the level of the water in the emergency tank is kept at the overflow mark).

Radiolysis in the reactor core causes the generation of hydrogen from the water in the CPS cooling system.

To prevent the formation of an explosive concentration of hydrogen there is continuous ventilation of the space above the water in the top and bottom tanks, together with monitoring of the hydrogen content in the CPS cooling water as well as in the space above the water in its tanks.

The emergency water supply tank is connected to the atmosphere by four breather pipes which lead off from the top of it. In addition to this the space above the water in the tank is constantly blown through with compressed air. If there is a failure in the compressed air supply system, the space above the water is ventilated by air ejection, using the excess water continuously drained from the emergency tank into the lower tanks through the overflow. If the system stops, all the water from the emergency tank is drained into the lower tanks.

The lower tanks of the system are continuously blowndown with compressed air and they also receive the air ejected from the emergency tank; furthermore, since the tanks are at reduced pressure, they receive air from the building through a special line with a valve. Reduced pressure in the tanks and removal of air blown through them is ensured by a special tank ventilation system.

If any of the blowdown (ventilation) systems should fail, those still functioning keep the hydrogen concentration at a safe level.

To maintain the requisite quality of the water in the cooling system, it has to be constantly purified. Water is fed for clean-up from the pump pressure header and returned to the lower tanks.

If there are disruptions in the cooling system (reduced level in the emergency supply tank or a fall in the water flow rate), signals are sent for emergency shutdown.

Parameter monitoring and control of the CPS channel cooling system is carried out by the operating staff from the unit control panel. The system was thoroughly checked during the startup and adjustment operations and during operation of the unit.

2.11.3. Blowdown and cooling system

The blowdown and cooling system shown in Fig. 2.56 is intended to cool the blowdown water of the MFCC bled off for clean-up, followed by reheating before it is returned to the MFCC under nominal conditions, and to reduce the temperature of the circulation water to the required level under cooling conditions.

Under nominal conditions the MFCC coolant flowing at 200 t/h (100 t/h from each loop) is pumped by the main circulation pumps to a regenerator where it cools from 285°C to 68°C through heat removal to a cold counterflow, and is then further cooled down to 50°C by the water of the intermediate circuit in the blowdown afterheater, from where it enters the circuit water clean-up system. As it passes through the regenerator in the opposite direction, the cleaned water heats up from 50°C to 269°C and is recycled to the steam separators through mixers in the feed water piping. It should be pointed out that either of the two after-heaters in the blowdown and cooling system may operate in this mode.

When cooling the unit the blowdown and cooling system reduces the temperature of the water in the MFCC, starting at 180°C, down to the temperature required for repairs to the unit. Circulation takes place along the line: steam separators - cooling pumps - larger after-heater - steam separators.

The blowdown and cooling system may also be used to remove residual heat from the reactor when there loss of current for the power unit's own needs. In this mode the operational system is the same as for the cooling mode.

2.11.4. Gas circuit system

The main layout of the system is shown in Fig. 2.57.

Under nominal operating conditions the gas circuit system works in the following way: the nitrogen-helium mixture, when it emerges from the reactor, passes through the fuel channel integrity monitoring system, where a channel-by-channel temperature check is made and the moisture content of the nitrogen-helium mixture is monitored for groups of channels.

Having cleared the fuel channel integrity monitoring system, the mixture passes through a series of condensers, air heaters and filters in which iodine vapours are deposited, and reaches the compressor intake of the helium scrubbing unit, in which apparatus hydrogen, oxygen, methane, carbon dioxide, carbon monoxide and ammonia impurities are removed from the mixture, down to a concentration permitting normal reactor use.

Removal of radioactive argon-41 takes place in cooling tanks [at -195°C].

After passing through the scrubbing system, the mixture is returned to the reactor pile. A hydraulic seal is fitted to the pipe which introduces the mixture into the pile to prevent the pressure from rising above permitted levels, i.e. higher than 1-3 kPa.

In order to reduce leakage of helium from the reactor pile, nitrogen (99.9999% pure) is introduced into the metal structures of the reactor at a pressure of 2-5 kPa. A hydraulic seal is fitted to the feed pipe.

In the gas circuit system there exists the possibility of flushing the reactor pile with nitrogen. In this event the nitrogen is dumped via the activity reduction system.

In the gas circuit system measurements are made of flow rate, impurity concentration, moisture content, temperature and pressure of the nitrogen-helium mixture, and the circuit is monitored for radiation. All results are displayed on the gas circuit control panel.

The system is controlled from the gas circuit control panel.

2.11.5. Cooling of spent fuel storage ponds

The pond cooling system is designed to stabilize the temperature of the water ponds, which is heated by the decay heat from the spent fuel, in all operating modes including a total power failure affecting in-house requirements. The system maintains the temperature of the water in the cooling ponds:

- Under normal operating conditions - at no more than 50°C, where the maximum decay heat is 1800 kW;

- During simultaneous unloading into the pond of 5% of the fuel assemblies from damaged channels - at no more than 70°C, with maximum decay heat in the pond of no more than 3000 kW;

- When heat removal ceases as a result of any departures from normal operating conditions or loss of electric power to the system (- the water temperature should rise to no more than 80°C in the 20 hours after heat removal has ceased.)

During this time measures must be taken to restore the functioning capacity of the system.

Water quality in the cooling ponds is maintained by a bypass purification system. Arrangements exist to exclude the possibility of accidental drainage of the ponds. The space above the water is ventilated.

The main layout of the cooling pond system is shown in Figure 2.58.

The cooling of the ponds is achieved by means of a closed loop. Water, heated in the ponds by decay heat production, passes from the upper regions of the ponds to heat exchangers, where the heat is transferred to service water. The required heat reduction is achieved with one heat exchanger, while a second stands in reserve. Having passed through the heat exchangers the water is returned to the pond by one of two pumps at a rate of ~ 160 m^3/hr and at a pressure of ~ 20 m water column (the second pump is a reserve pump).

The cooling water outflow and return pipes are so arranged that if they were to rupture the level in the ponds would not fall below the permissible minimum.

To prevent overfilling of the ponds each has an overflow.

To prevent the formation of an explosive concentration of hydrogen in the space above the water in the ponds, constant ventilation is provided by air taken from the central hall. Should the pond ventilation system suffer a malfunction, the flow cross-section of the ducts which connect the ponds with the central hall is such that one may view them as a single compartment with a volume of 40 000 m^3, ventilated by an independent ventilation system.

The water in the ponds is purified by means of a loop which is independent of the cooling system.

The following parameters are monitored:

- Water levels in the ponds;
- Temperature of water in the ponds;
- Flow rate of cooling water etc.

Operating personnel control the system and monitor its process parameters from the unit control room.

The system underwent comprehensive and direct testing against design specifications during commissioning operations and while the power unit was in use.

2.11.6. Ejection cooling system

The ejection cooling system (Fig. 2.59) is designed to remove heat from the leaktight compartments. For each of the two leaktight compartments there are four groups of coolers set at level 5.0 in the housing of the main circulation pump tanks. Each group consists of four coolers, and each cooler has a capacity of 2500 m^3/hr. Each group has an independent air supply. As regards their water supply, the coolers are divided into two independent sub-systems of eight coolers each and connected to different feedback Sylphon pumping systems.

Air at maximum temperature is drawn off from the upper regions of the downcomer shafts through four pipes, fed to each group of coolers where it is cooled by jets of water down to 35°C in summer, and to 18°C in winter, and then passes into the compartment containing the main circulation pump tanks. The cooled air removes heat from the mechanical equipment and unplanned coolant leakages. To prevent escape of dispersed moisture with the air, separators are installed at the outlet from the coolers. Apart from cooling the air and eliminating excess moisture, the ejection coolers also remove aerosols, including radioactive iodine.

The ejection cooling system is compact and contains no active elements which require maintenance or control while it is operating.

2.11.7. Radiation monitoring system

The nuclear power station radiation control system is a component part (sub-system) of the automated station control system and is designed to gather, process and present information on radiation conditions in the station compartments and outside, on conditions in the process media and circuits and on irradiation doses to personnel and individuals from the population in accordance with the norms and statutes in force.

The radaition monitoring system as a whole can be divided into two: the process radiation monitoring and the radiation dosimetry systems. The purpose of the process radiation monitoring system is process optimization, and also to monitor the condition of the protective barriers against the spread of radionuclides. The purpose of the radiation dosimetry system is to monitor the radioecological factors arising from the operation of the facility and, in the final account, to determine the internal and external irradiation doses received by staff and individuals in the population.

The off-site dosimetric monitoring system is distinct from the radiation dosimetry system, and:

- Determines the activities and nuclide compositions of radioactive substances in the atmosphere;
- Monitors gamma radiation dose exposures in the area;
- Monitors radioactive fallout;
- Monitors ground water activity in test bore-holes;
- Determines the content of radioactive substances in soil, vegetation, locally-produced feedstuffs, food products and so on.

A structural diagram of the radiation dosimetry system is shown in Fig. 2.60.

The following are used for radiation monitoring:

(1) The combined AKRB-06 unit, which includes detection units and equipment, informtion processing equipment, surface contamination monitoring units and units and dosemeters for monitoring the station personnel;

(2) Individual, portable and wearable devices;

(3) Laboratory equipment and instruments.

The radiation monitoring structure takes the form of a data and measurment system with a large number of dispersed information sources and capture devices, arranged in a radial-annular manner; under this system, the detection units, the UNO-06r information storage and processing unit and the local units for signalling when established values are exceeded are linked radially. The UNO-06r system's internal links and links with the UNO-01r monitoring and data exchange device are arranged in a ring.

The AKRB-06 thus monitors continuously the readings from the detecting units and devices, transmits information on all channels to the computer, signals failures of its components and controls the shut-off equipment on the sampling lines. The devices for displaying information (display, console, signalling units) are located on the radiation monitoring board.

The detection units feeding into the AKRB-06 measure:

- The gamma exposure rate within the range 10^{-5} to 10^{3} R/h (BDMG-41, BDMG-41-01, UDMG-42, UDMG-41-02);

- The activity concentration of gamma emitters in liquid process media and circuits within the range 5×10^{-11} to 10^{-3} Ci/L (UDZhG-04r, UDZhG-05r, UDZhG-14r1);

- The activity concentration of iodine vapours in air within the range 10^{-11} to 10^{-6} Ci/L (BDAD-06);

- The activity concentrations of aerosols with dispersion phases containing beta emitters within the range 10^{-13} to 10^{-9} Ci/L (BDAB-05);

- The beta activity concentration of inert gases in the air and process media within the range 10^{-9} to 1.4×10^{-4} Ci/L (UGDB-08) and 10^{-5} to 0.3 Ci/L (UDGB-05-01);

- The activity of long-lived beta-emitting aerosols in the gas-aerosol releases to the vent stack within the range 3×10^{-14} to 3×10^{-10} Ci/L;

- The activity of short-lived beta-emitting aerosols in the gas-aerosol releases to the vent stack within the range 1.5×10^{-12} to 1.5×10^{-8} Ci/L;

- The activity of beta-emitting inert gases in gas-aerosol releases to the vent stack within the range 8×10^{-9} to 8×10^{-5} Ci/L;

- The activity of the gamma-emitting vapours in the gaseous phase of the gas-aerosol releases to the vent stack within the range 3×10^{-13} to 3×10^{-10} Ci/L.

Gas-aerosol releases to the vent stack are measured using RKS-03-01 and RKS-2-02 radiometers. The airflow through the vent stack is measured using a partial flowmeter with a metal-polymer sensing element. The measurement data are processed by a computer.

Each power station unit has a total detector complement of 490 units, of which approximately 400 are in production areas with a continuous or restricted staff presence.

The surface contamination monitoring unit notifies staff when contamination exceeds the following established threshold levels:

- For the skin of the hands: beta emitters within the range 10 to 2000 counts/min./cm^2 (RZG-05-01, SZB-03, SZB-04);

- For the skin of the body or basic protective clothing, beta emitters within the range 5 to 2000 counts/min./cm^2 (RZB-04-04);

- For means of transport, on leaving the station in order to detect objects to be investigated in detail using other means, gamma radiation within the range 2.78×10^{-2} to 0.278 μR/s (RZG-05);

- For personnel, on leaving the station, for detection and subsequent detailed investigation, gamma radiation within the range 1.4×10^{-2} to 0.14 μR/s (RZG-04-01).

The personnel irradiation monitoring unit continuously monitors external irradiation. To this end, the basic items of equipment used are:

- Sets of individual dosimetric photomonitors to measure the total exposure to gamma radiation within the range 0.05 to 2 R at energies of 0.1 to 1.25 MeV (IFKU-1);

- Sets of thermoluminescence dosemeters to measure exposures to X-ray and gamma radiation in the energy range 0.06 to 1.25 MeV within limits of 1.0 to 1000 R and 0.1 to 1000 R (KDT-02);

- Gamma exposure rate dosemeter-indicators in the range 0.1 to 9.9 R/h; there is also a range of other dosemeter variants with similar characteristics.

The internal irradiation recording equipment measures the whole-body burden of ^{137}Ce and ^{60}Co nuclides and the thyroid ^{131}I burden (MSG-01). In addition, DGDK-type semiconductor detectors and their analysis and processing equipment are used to identify a range of radionuclides in the human body. There is a wide range of portable and wearable dosemeters and radiometers in the total stock. For example, the following dosemeters are used:

- For measuring exposures to gamma and X-ray radiation in the energy range between 15 keV and 25 MeV with a measurement range between 0.1 μR/s and 11 R/s (DRG, DKS);

- For measuring neutron dose equivalent rates between 0.05 and 5000 μrem/s (KDK-2);

- For express measurements of the specific activities of samples (RKB4-1 beta radiometer) within the range 2 x 10^{-12} to 10^{-7} Ci/L;

- For measuring nuclide activity concentration in liquids and air for alpha, beta and gamma radiation over various energy ranges (RZhS-05, RGA-01, MKS-01 and others).

Off-site dosimetric monitoring is carried out in the area of the station within a radius of approximately 35 km. It is carried out by the off-site dosimetry service of the plant and is designed to obtain the information required to evaluate the external and internal doses to individuals in the population. The monitoring equipment is located at 38 posts, and includes total gamma dosemeters, vials for collecting atmospheric fallout and seven aspiration sets.

Samples are analysed using semiconductor detectors, spectrometers and analysers with microcomputers. On the basis of the data on releases into the atmosphere through the power station vent stack and by means of automatic measurement of the meterological parameters, a forecast is made using a microcomputer of the radiation situtation in the power station area.

2.11.8. NPP control

The NPP is controlled on two levels:

- As a station;

- By unit. (See Fig. 2.6.1 "Basic structural diagram of station control sytem"). Control over all plant safety systems is carried out at unit level.

Station-level control

At the station level, operational control is effected from the central control board. At the station level, the operating staff are responsible for:

- Control over the electrical equipment in the main electrical connection circuit (750-kV line interrupters, unit transformers, autotransformer and so on, 20 kV generator interrupters, 330 kV autotransformer interrupters and interrupters for the 6 and 330 kV back-up medium voltage transformers);

- Distribution of active and reactive power;

- Co-ordinating the work of the operating staff at the unit control panels and in separate installations on the site.

On the central control board there are remote control keys for the aforementioned interrupters, and also audible and visual signalling of accident and fault conditions, and visual signalling of the condition of the switching apparatus (interrupter in or out) on the mimic diagram.

The protection relay equipment, anti-accident automatics and telemechanics are housed in the protection relay buildings for the corresponding 750 kV and 330 kV distribution equipment. Microchip integrated circuits form the basis of the 750 kV line protection relay equipment; they monitor the function of each separate channel and make it possible to test them. Mass-produced electromechanical relays are also used in the protection relay control devices and anti-accident automatics.

Unit-level control

The process installations and structures of a given unit are controlled at unit level:

- the reactor and its supply facilities (main circulating pumps, feed pumps, emergency feed pumps and so on);

- turbogenerators and auxiliary equipment;

- normal and back-up medium voltage supplies and so on;

- separate installations on the site: diesel power station, process water supply pumps and so on.

The above are controlled from the unit control board, which includes the control board and display. The operator control circuit of the unit control panel is split into control areas:

- reactor control;

- steam generator set control;

- turbine, generator and medium voltage supply control.

On the operator circuit of the unit control board are located the operators' work stations and the control panels for the:

- senior reactor control engineer;

- senior unit control engineer;

- senior turbogenerator control engineer.

These control panels contain the following:

- control apparatus;

- monitoring system instruments;

- "Skala" central monitoring system, call-up devices and display units;

- communications apparatus.

On the unit control panel operator circuit board are the following:

- reactor channel mimic board;

- CPS mimic board;

- mimic diagram of the thermal and electrical parts of the unit;

- individual instruments for the monitoring system signalling equipment.

The "Skala" central monitoring system monitors the main bulk of the parameters. The most important parameters required for correct process operation also have their own individual monitoring instruments. These include instruments showing reactor output, drum separator level and pressure, steam flow rate ex drum separator, feedwater flow rate into drum separator, measurements from the physical power distribution monitoring system and the CPS and so on.

Electricity is supplied to the unit control panel and the "Skala" central monitoring system from the secure power supply so that even when there is a power loss at the medium voltage buses, the operator does not lose information on the conditions of the process parameters. On the unit control panel operating circuit are located the controls for the process protection and monitoring devices. Overall operative control of the unit is carried out from the controller's board, which has telephone apparatus and loudspeaker links.

Included in the operator circuit of the unit control panel there are in addition special safety panels for each of the three safety sub- systems; on these the back-up medium voltage power supply (diesel generators) and the emergency reactor cooling and accident containment systems are controlled and monitored.

A back-up control board is provided for the eventuality that the reactor cannot be shut down and maintained in sub-critical condition from the unit control board. On the operator circuit of the back-up control board are located control panel, operator circuit panels and safety panels. On the control panel are located the AZ-5 emergency protection system button, CPS coupling disconnection switch, signalling board and so on. On the operator circuit panels are the recorders for neutron power, drum separator pressure and so on. The safety panels of the back-up control board are analogous to those on the unit control board.

Local control panels are provided for the range of systems which operate independently of the main processes; these include the gas circuit, active waste treatment, radiation monitoring system, ejector gas sorption scrubbing unit and the turbines.

Local boards are also provided for a range of units involved in the main process (main circulation pump, electric feed pump, emergency electric feed pump and so on), and these are installed with the equipment itself.

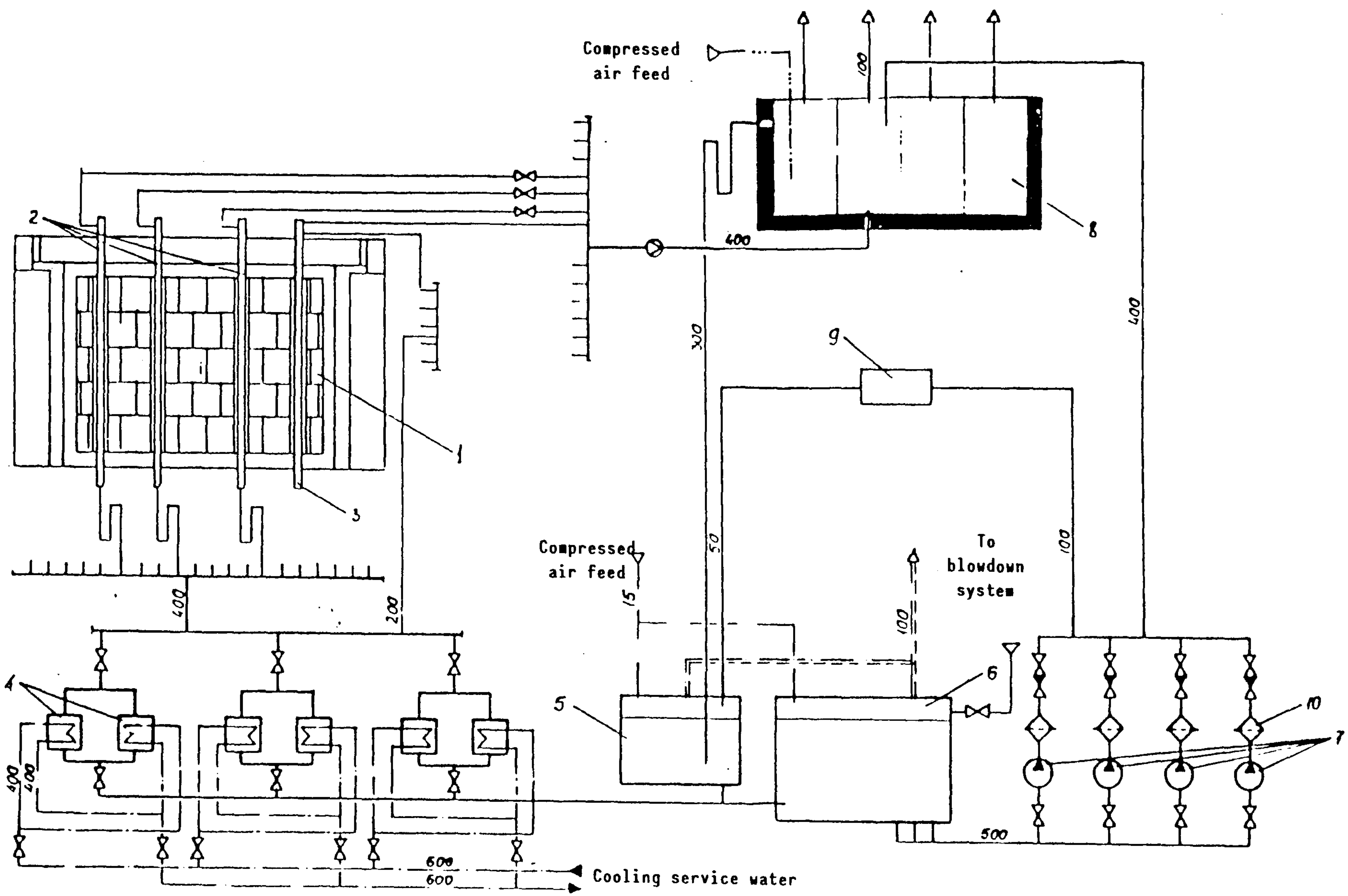


Fig. 2.55 Schematic representation of CPS fission chamber, power density monitor and failure detection channel cooling system: (1) Reactor; (2) CPS, fission chamber and power monitor channels; (3) Failure detection channels; (4) CPS heat-exchangers; (5) Drainage tank; (6) Circulation tank; (7) CPS pumps; (8) CPS emergency tank; (9) Bypass cleaning unit; (10) Filters.

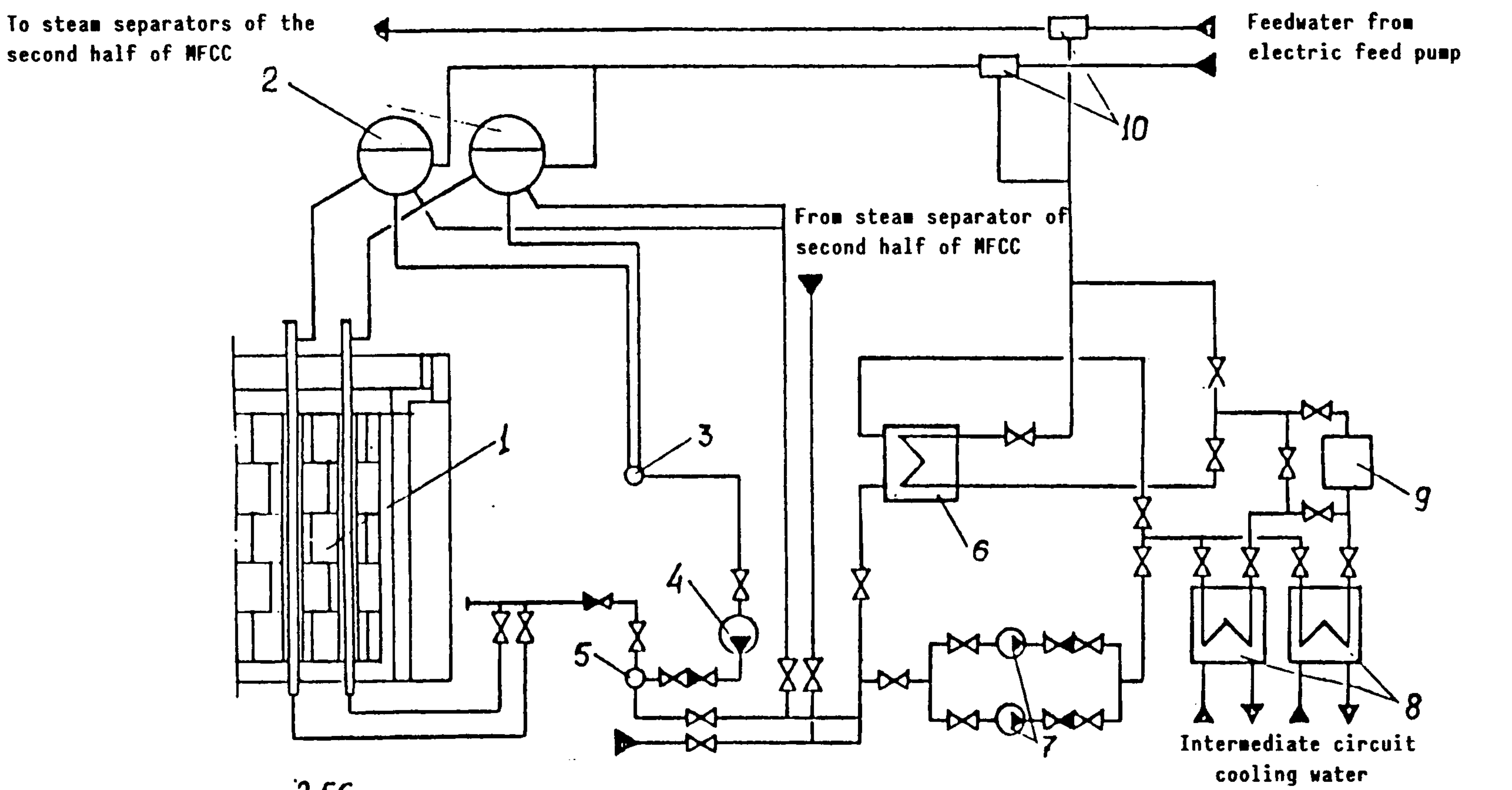


Fig. 2.56. From pressure header of second half of MFCC.
Schematic representation of blowdown and cooling system: (1) Reactor; (2) Steam separators; (3) Intake header; (4) Main circulation pump; (5) Pressure header; (6) Regenerator; (7) Cooling pump; (8) Blowdown afterheater; (9) Plant for cleaning circuit water; (10) Blowdown and feedwater mixers.

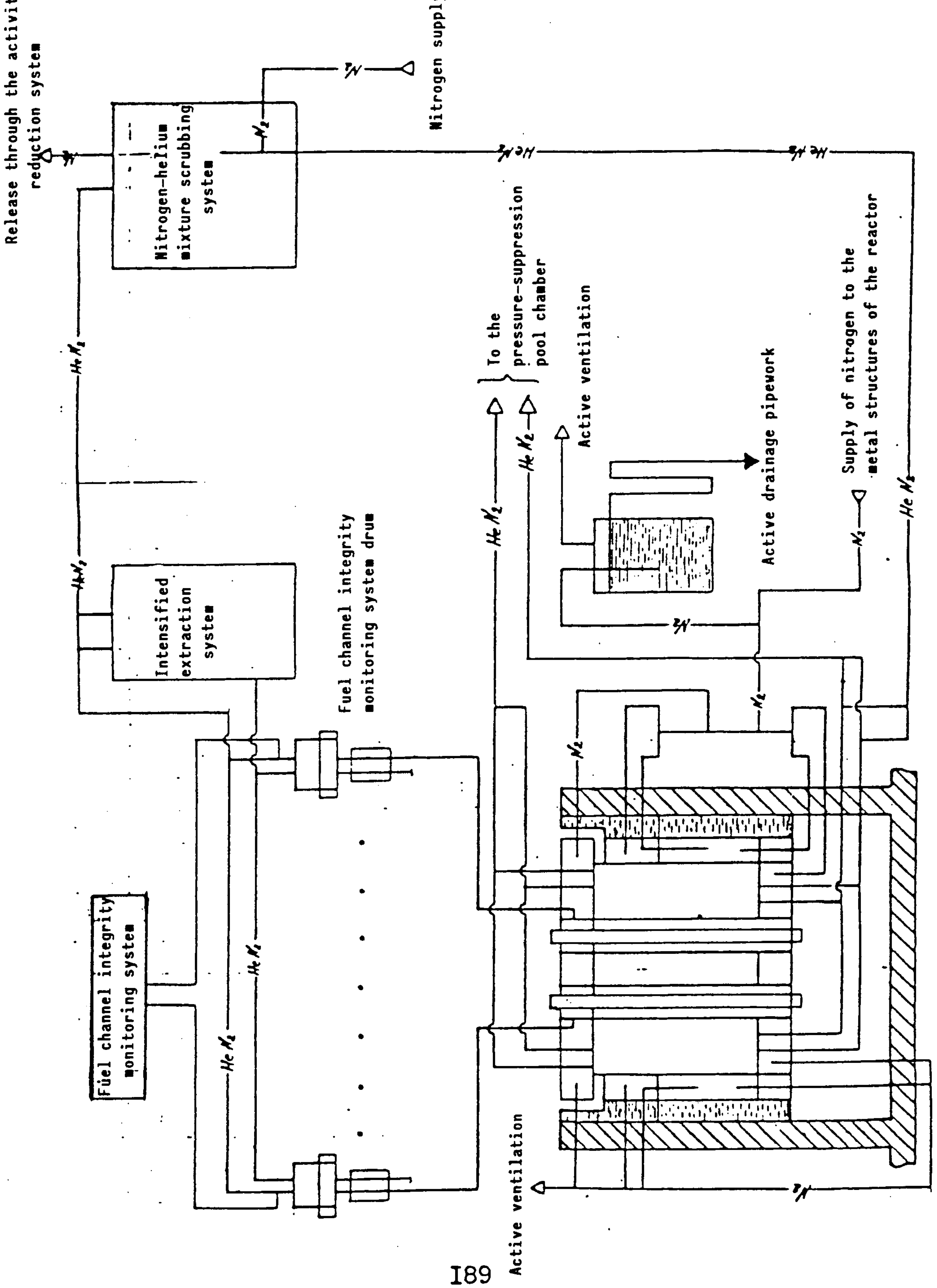


Fig. 2.57 Gas Circuit System.

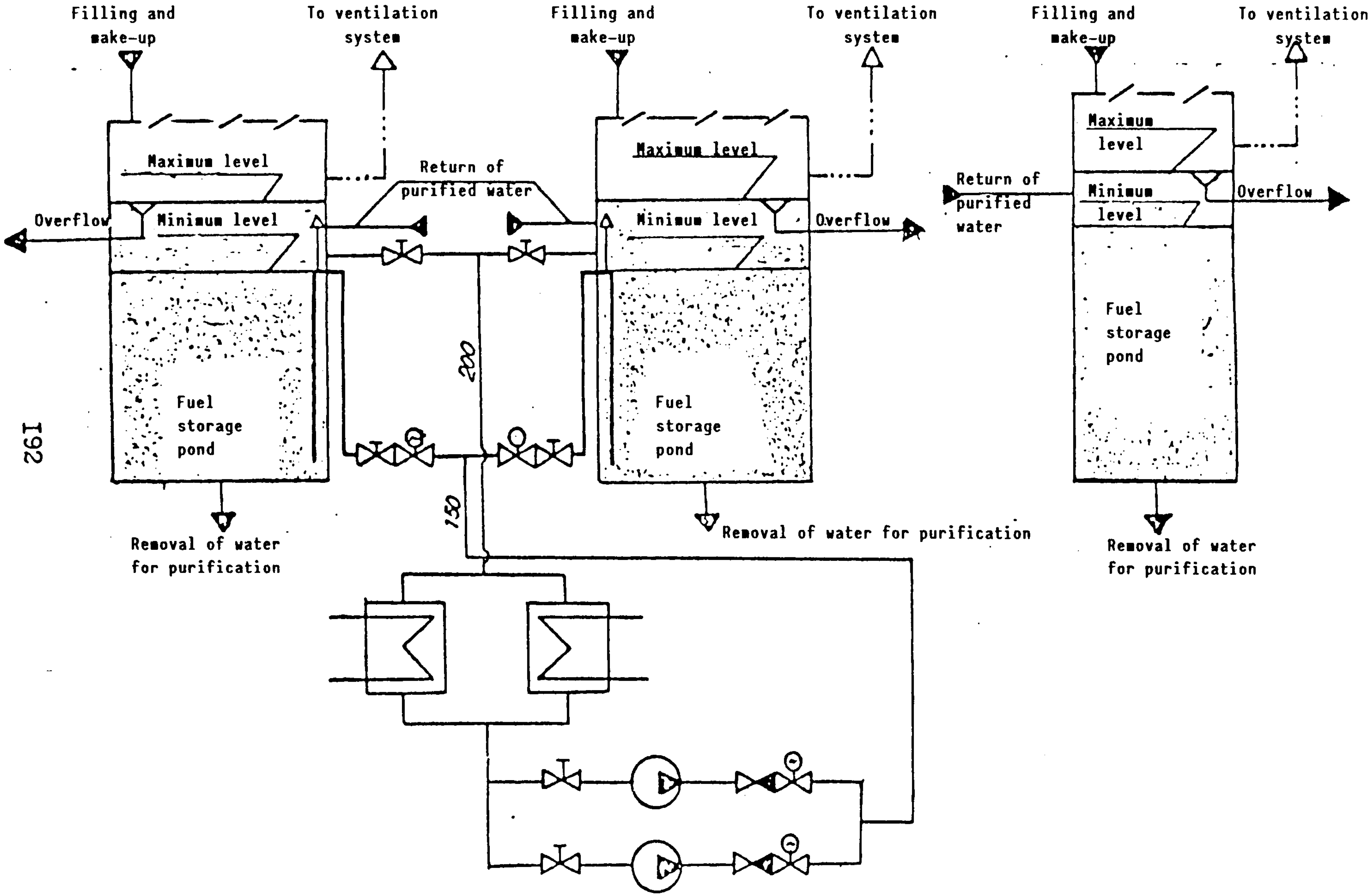


Fig. 2.58 System for Cooling Spent Fuel Storage Ponds

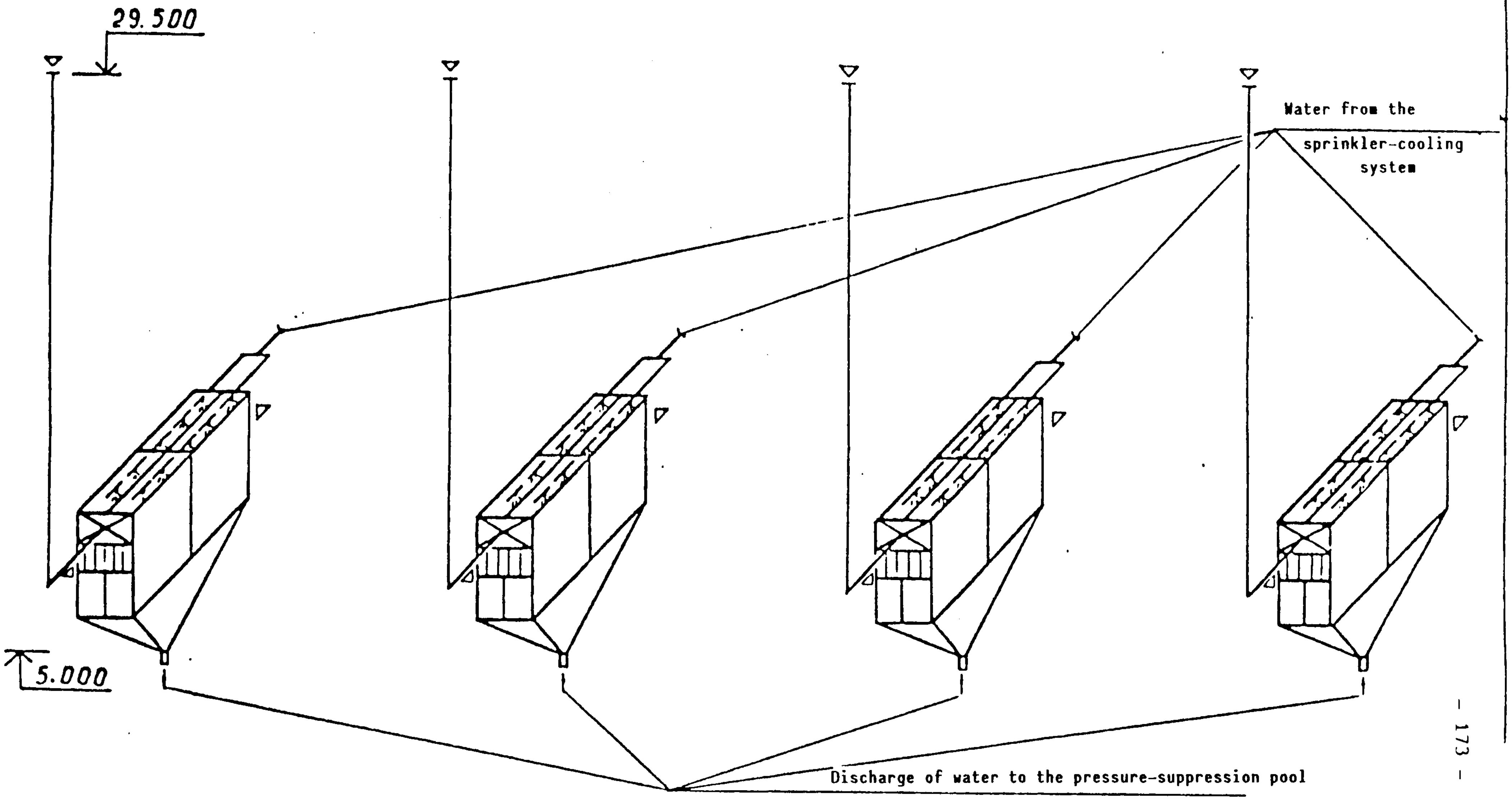


Fig. 2.59 Layout of the Ejection Cooling System.

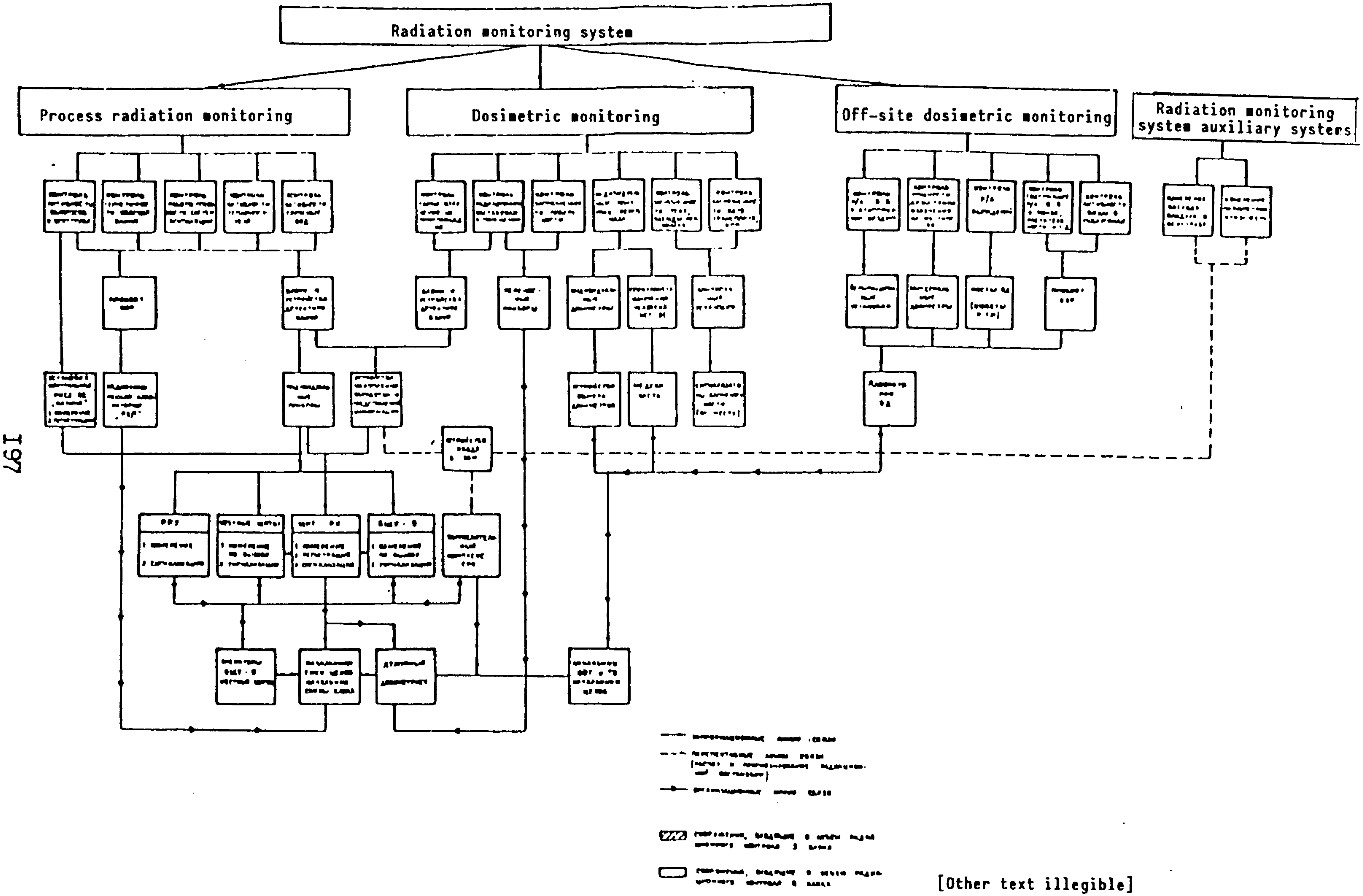

Radiation monitoring system
Process radiation monitoring
Dosimetric monitoring
Off-site dosimetric monitoring
Radiation monitoring system auxiliary systems
[Other text illegible]

Fig. 2.60

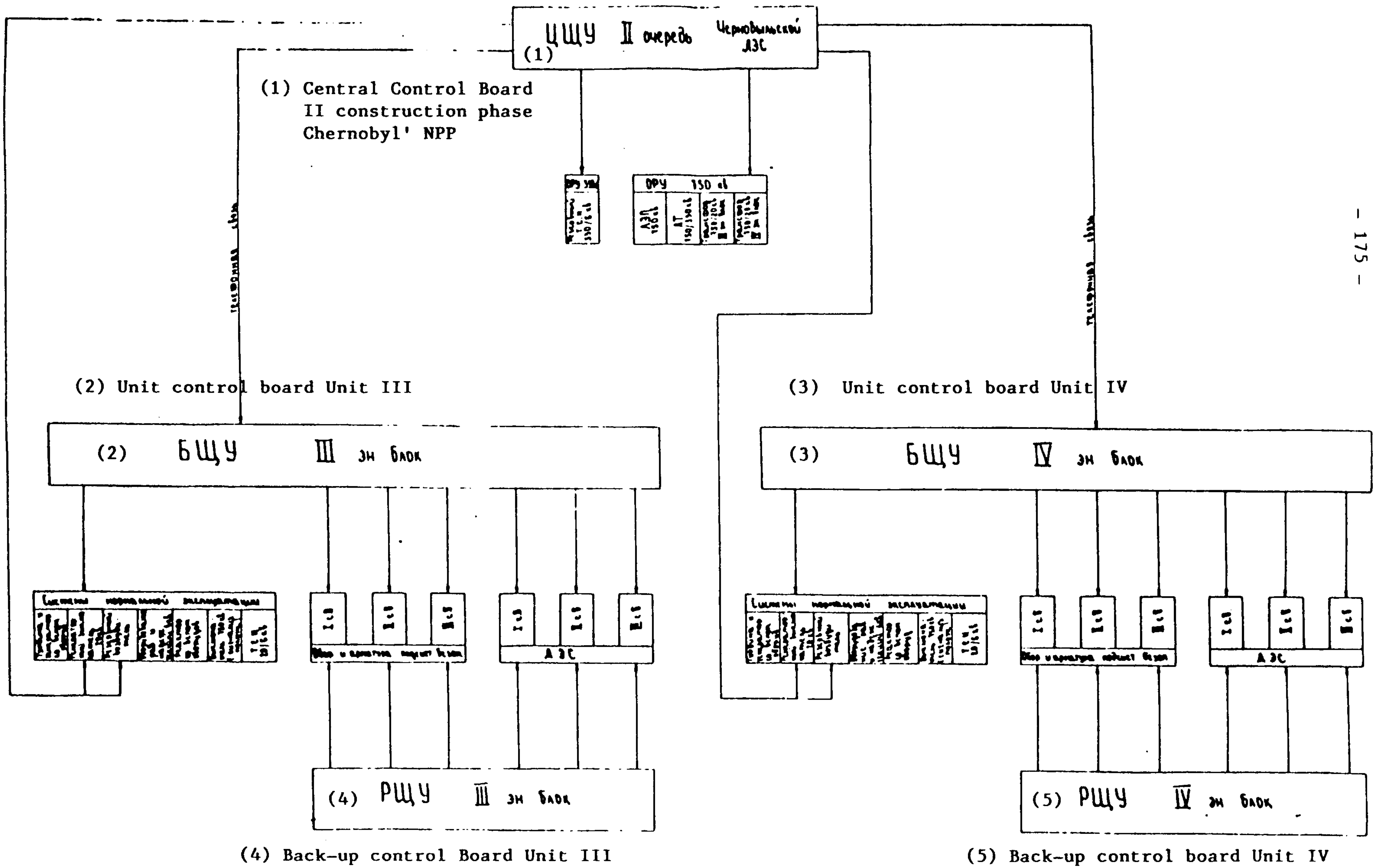


Fig. Рис. 2.61

Принципиальная структурная схема управления станции

Basic structural diagram of station control system

[Other text illegible]

2.12. Reactor and unit operating modes

2.12.1. Normal operating modes

The reactor and unit operating modes can be divided into normal operating modes and transient modes associated with equipment failure. Normal operating modes consist of unit startup and shutdown, unit operation at power, reactor cooling modes during equipment maintenance (maintenance repair modes).
Reactor startup and shutdown

RBMK power units are started up with the main circulation pumps in operation, at a "sliding" pressure and at a separator water level selected by the operator within a given range. The required cavitation margin at the main circulation pump intake is ensured by reducing pump delivery using the throttle-regulating valves installed at the pump discharge. Under these conditions the cooling water flow rate in all the fuel channels of the core is continuously monitored and reactor safety is thereby ensured. Initial heating of the unit is carried out at a "sliding" pressure in the separators, i.e. the pressure is not constant but increases as the temperature rises.

During the startup and initial heating of the unit the circulation loop is fed by the emergency feed pumps. Reactor power during startup and initial heating is maintained at an average level of 2-3% of nominal capacity. The thermal power of individual fuel channels during this process can be as much as 6% of nominal because of the non-uniformity of power density distribution in the core.

Reactor power ascension and initial heating of the circuit can take place with one, two or three of the main circulation pumps (capacity 6000-7000 m^3/h each) operating on each side of the reactor. At this pump capacity it is possible to monitor the water flow rate through each fuel channel and at the same time to ensure an adequate pump cavitation margin. At a reactor power of 2-3% of nominal, the circuit installations are heated to a temperature of about 200°C. The circuit is heated at a rate of about 10°C per hour, the limiting factor being the thermal stresses in the reactor metal structures.

At a pressure of 2-4 kgf/cm^2, the de-aerators begin to heat up.

A vacuum begins to build up in the condensers of the turbine being started at a separator pressure of about 15 kgf/cm^2. Once the vacuum has been created, the turbine starts up and begins to build up speed. The turbogenerator is normally synchronized and connected to the grid when the pressure in the separators is about 50 kgf/cm^2. Further increase in the parameters up to rated values takes place in parallel with the build-up of electric load.

Figure 2.6.2 gives an example of the evolution of the main reactor parameters from the time the reactor reaches the minimum power level that can be monitored until the turbogenerator is synchronized and connected to the grid.

The main circulation pumps remain in operation during scheduled shutdown and cooling of RBMK units. Before the onset of shutdown cooling, the reactor power is run down to the after-heat level and the unit turbogenerators are disconnected from the grid and shut off. When reactor power is reduced to the 20% level, the capacity of the main circulation pumps in service should be cut to 6000-7000 m^3/h. The circuit is cooled down to a temperature of 120-130°C by gradually lowering circuit pressure by discharging steam in a controlled manner from the separators to the turbine condensers or to the process condenser. To achieve a greater degree of cooling, a special shutdown cooling system is employed which consists of pumps and heat exchangers.

The factor limiting the cooling rate, and also the heating rate, is the thermal stresses in the reactor metal structures. Since during shutdown cooling the rate of temperature reduction in the circuit is determined principally by the rate of controlled steam discharge from the separators, it is not difficult to keep the cooling rate at the prescribed level under these conditions.

Unit operation at power

During power operation of the unit, reactor safety is ensured by keeping its critical parameters within the permissible range.

Up to the 500 MW(t) power level, the coolant is circulated through the reactor by the main circulation pumps operating at 6000-7000 m^3/h. At a power of 500 MW(t), the throttle-regulating valve is opened and the main circulation pump capacity increases to 8000 m^3/h. At power levels above 500 MW(t) up to the rated level, the unit operates at a constant main circulation pump capacity. When the power level exceeds 60% of rated, no fewer than three main circulation pumps should be operating on each side of the reactor. The hydraulic distribution of an RBMK reactor core is such that, when rated capacity is reached, the throttle-regulating valves are fully open and the total flow through the reactor is 48 000 m^3/h.

Maintenance/repair modes

The main requirement when inspecting or servicing any item of reactor equipment is that the core must be safely cooled throughout this period. Also, the reactor design and the organization of maintenance work should be such as to ensure that all the circuit equipment can be serviced.

From the standpoint of maintenance work, the primary coolant circuit is split into four sections: the discharge (delivery-side) section, which extends from the discharge valves of the main circulation pumps to the channel isolating and regulating valves; the fuel channel ducts from the isolating and regulating valves to the separators; the separators and downcomers to the intake valves of the main circulation pumps; and the section between the intake and discharge valves which includes the circulation pumps and the associated fittings.

The maintenance of equipment and pipes located in the section between the intake and discharge valves of the main circulation pumps does not pose any difficulties, and in theory can be carried out while the reactor is operating.

To do this, it is necessary to close the isolating and intake gate valves on the pipes of the main circulation pump in question; once the coolant has been drained, the pump itself and the discharge and intake pipe sections adjacent to it as far as the gate valves are accessible for servicing. In this instance the coolant is circulated through the reactor by the other main circulation pumps of the relevant loop.

To repair structural elements of fuel channels, the fuel assembly is withdrawn from the channel under repair, the isolating and regulating valve at the channel inlet is closed and the water level in the separators is lowered to below the level at which the steam-water communication pipe of this channel is connected to the separator casing. The remaining channels in the core are cooled either by forced or natural coolant circulation.

To repair the separators, downcomers and intake valves of the main circulation pumps, the discharge valves of these pumps are closed and the level in the fuel channels is lowered. To ensure safe cooling of the core under these conditions, a special maintenance tank is connected to the main circulation pump pressure header; the channels are fed from this tank and the steam which forms in them is evacuated to the separators. To allow inspection and repair of the separators, a system has been installed which draws off steam from the separators to the process condenser.

During repair work on the equipment of the discharge section, this section is cut off from the core by closing the isolating and regulating valves, and the residual heat is removed by water fed into the channels from the separators. This mode of fuel channel cooling (the bubbling mode of cooling) was studied on special test units during the reactor design stage. It was established experimentally that, when the isolating and regulating valves are closed, safe cooling of the fuel channels in the bubbling mode is ensured when the following conditions are met:

- The water level in the primary circuit is higher than the levels at which the steam-water communication pipes connect with the separator;

- The pressure in the separator is atmospheric;
- The after-heat in the fuel assembly is not greater than 25 kW;
- The water temperature in the separator is not more than 80-90°C in order to prevent water hammers in the steam-water communication pipes.

The most complicated repair operation relates to the channel flow meters and the isolating and regulating valves. To do this work, a technique is used whereby the water is frozen in the inlet pipes of the fuel channels. When this technique is employed, the fuel assemblies are cooled in the same manner as when repairs are being conducted with the isolating and regulating valves at the fuel channel inlet closed.

The water is frozen in the vertical sections of the inlet pipes by means of group and single refrigerating chambers attached to these pipes. The refrigerant is air at a temperature of -100°C which is supplied from the nitrogen-oxygen station. While the freezing operation is taking place the isolating and regulating valves remain closed and the fuel assemblies are cooled by the bubbling mode. Once ice plugs about 0.5 m high have formed in the pipes, the isolating and regulating valves and the flow measurement detectors are accessible for repair. This freezing method has repeatedly been used successfully at the Leningrad and Chernobyl' nuclear power plants.

2.12.2. Transient modes resulting from equipment failures

Because of the large unit power of RBMK boiling-water, graphite-moderated reactors and their extreme importance in energy systems, the control and protection system (CPS) of such reactors provides for rapid controlled power reduction at a prescribed rate to safe levels in the event of the failure of certain types of equipment. When a signal is transmitted indicating a fault in the process installations, emergency protection systems of three kinds (AZ1, AZ2, AZ5) are triggered.

The following algorithm for the operation of emergency protection systems has been developed for the CPS of existing RBMK-1000 reactors:

- AZ1 is triggered when one of the six main circulation pumps shuts off, the feedwater flow rate decreases and the level in the separators is reduced. At the AZ1 signal, reactor power is reduced to the 60% level;
- AZ2 is activated in the case of emergency load shedding or the failure of one of the two operating turbogenerators. At this signal, the reactor power drops to the 50% level;

In other accident situations caused by equipment failure the AZ5 emergency protection system is activated and triggers an uncontrolled power reduction to complete shutdown;

In order to study emergency conditions at RBMK units, a mathematical model of the plant was developed at the design stage which contains kinetic, hydrodynamic and heat-exchange equations and a description of the algorithms for the operation of the equipment and systems which automatically regulate NPP parameters. A subsequent comparison of the theoretical results with data on individual dynamic regimes actually experienced at operating nuclear power plants indicated that the mathematical model developed provides a satisfactory description of unit dynamics. Transient regimes mainly associated with transition to natural circulation of the coolant have been studied on special mock-up test stands.

Operating experience from units in service has shown that the measures and systems foreseen guarantee the safety of RBMK reactors in all modes resulting from equipment failures.

A great deal of research has been done to demonstrate the safety of reactor operation in the power reduction mode when the AZ5 emergency protection system is activated since this mode is accompanied by major changes in the process parameters and, in particular, by a reduction in the water level in the separators.

The behaviour of the main reactor parameters under transient conditions due to the activation of the AZ5 protection system is shown in Figure 2.6.3.

A loss of power plant internal load is one of the most severe accident situations that can occur at the unit. When internal load is lost, the coolant is circulated through the core at the start of the accident by the running down main circulation pumps and thereafter by natural circulation. The transient mode resulting from the loss of the internal load of the unit is shown in Figure 2.6.4.

This figure shows that, in the initial phase of the process, the decrease in the water flow rate is somewhat higher than the rate at which the reactor thermal power decreases; this results in a brief increase in steam content and reduction in the departure from nucleate boiling (DNB) ratios. More detailed studies have shown that under such conditions the reduction in DNB ratios - even in those channels which are under greatest thermal stress - is insignificant and poses no danger to the reactor, since in the initial phase of an accident the reactor is safely cooled by the running down main circulation pumps.

The running down pumps have a significant effect on coolant circulation through the reactor only for the first 30-35 seconds of the transient regime. Thereafter the core is cooled by natural circulation. The reliability and degree of natural circulation depends to a large extent on a number of factors such as the primary coolant circuit design, pressure behaviour in the circuit, the change in feedwater flow rate and temperature and so on.

Experimental research on natural circulation regimes has been conducted both on heat engineering mock-up rigs of the reactor primary coolant circuit and directly on operating reactors at the Leningrad and Kursk nuclear power plants. The experiments at the test rigs established, and those at reactors confirmed, the safety of cooling the core by natural circulation both in steady state and in dynamic regimes, given a constant pressure in the circuit. At the operating reactors, the tests under steady state conditions were conducted at a power level of 5 and 10% of nominal, while under dynamic conditions the main circulation pumps were switched off at a power of 25% and 50% of nominal. When the pressure drops as a result, for example, of safety valves opening and then not closing tightly, the coolant boils, the level in the separators increases and, as a result, the steam-water mixture is removed from the circuit. It was established at the test rig that, in the case of pressure reduction to a certain level, partial removal of the steam-water mixture and of the water from the circuit does not reduce the "levelling" head or stop coolant circulation. Overheating of the experimental channel fuel elements was observed only when the pressure in the separators dropped below 35 kgf/cm^2.

To ensure reactor safety follwoing a loss of internal load of the unit and a sharp drop in pressure, the emergency core cooling system is activated and feeds water to the fuel channels.

The safety of natural circulation regimes at RBMK units has been confirmed by accident situations which have occurred under real operating conditions at nuclear power plants. For example, at one unit of the Kursk NPP in January 1980 a total loss of station internal load occurred. During the transient conditions, readings from the thermocouples of the fuel assemblies and from the flowmeter at the inlet to one of the reactor fuel channels were recorded. During the entire transient regimes, no increase in the temperature of the fuel element cans was registered and the flow through the channel recorded under natural circulation conditions was not less than 20% of the flow rate at nominal capacity. The normal system for monitoring fuel can integrity showed no increase in coolant activity when the reactor power was subsequently increased. Experimental data on natural circulation regimes was correlated and compared with the results of calculations from the theoretical programs developed. In view of the good agreement between the results, theoretical predictions were made which showed that reliable and safe operation of RBMK-1000 units under natural circulation conditions is possible at power levels up to 35-40%.

The reactor loss of feedwater and the separator level protection system ensures the safe operation of these units at all power levels. When the AZ-5 loss of feedwater emergency protection system is triggered, not only are the emergency systems activated but the main circulation pumps are disconnected after a certain time lag. This is done to stop the level in the separator dropping too much and to prevent cavitation disruption of the main circulation pumps, i.e. to ensure optimal conditions for effective natural circulation. As indicated above, the safety of disconnecting the main circulation pumps and of reactor shutdown cooling by natural circulation has been confirmed by numerous experiments and by operating experience from nuclear power plants.

Figure 2.6.5 shows the theoretical transient regime following total instantaneous cut-off of feedwater flow.

The safety of the reactor following accidents in the feedwater supply system has also been confirmed by operating experience at RBMK units.

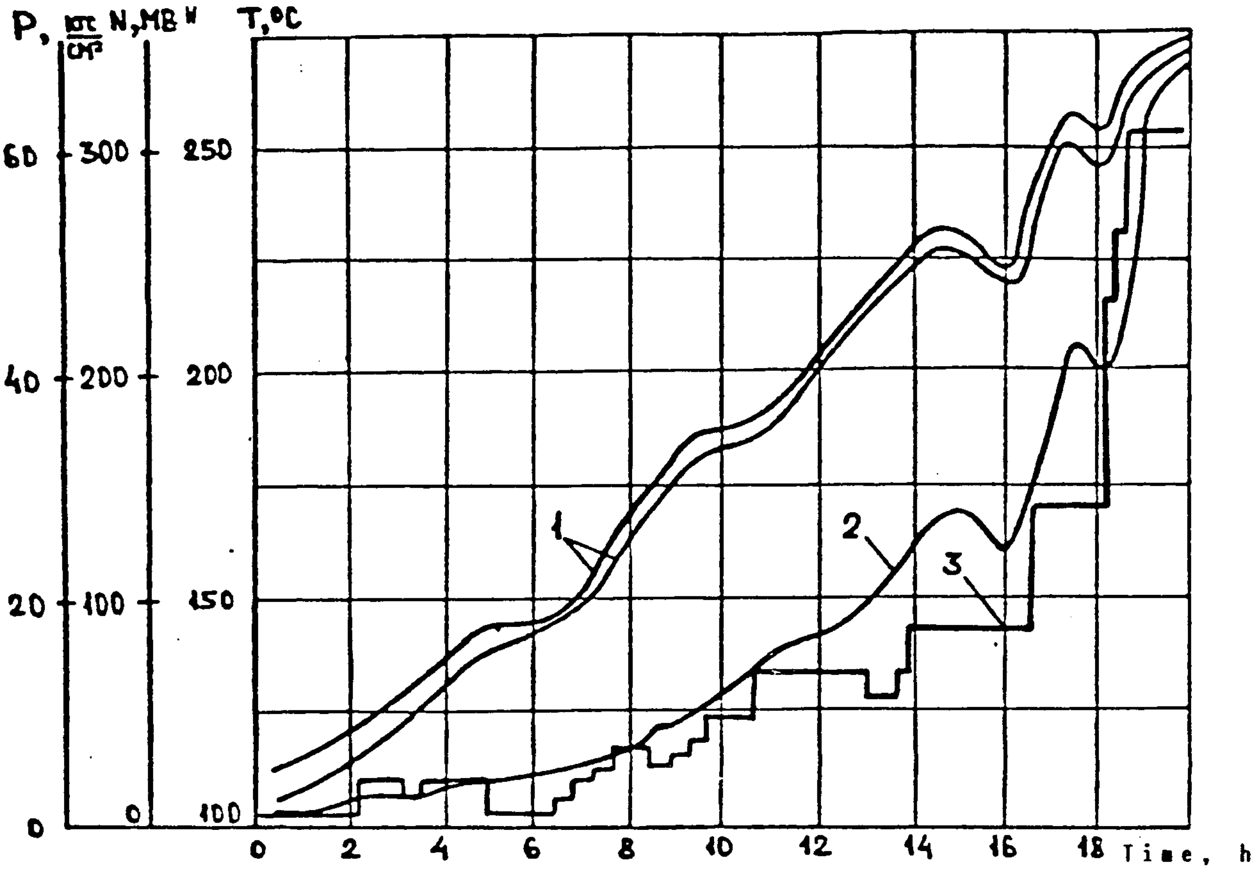


Fig 2.6.2. Evolution of reactor parameters during startup

I - Water temperature (T) in reactor circulation loops

2 - Pressure (P) in separators

3 - Thermal power (N) of reactor

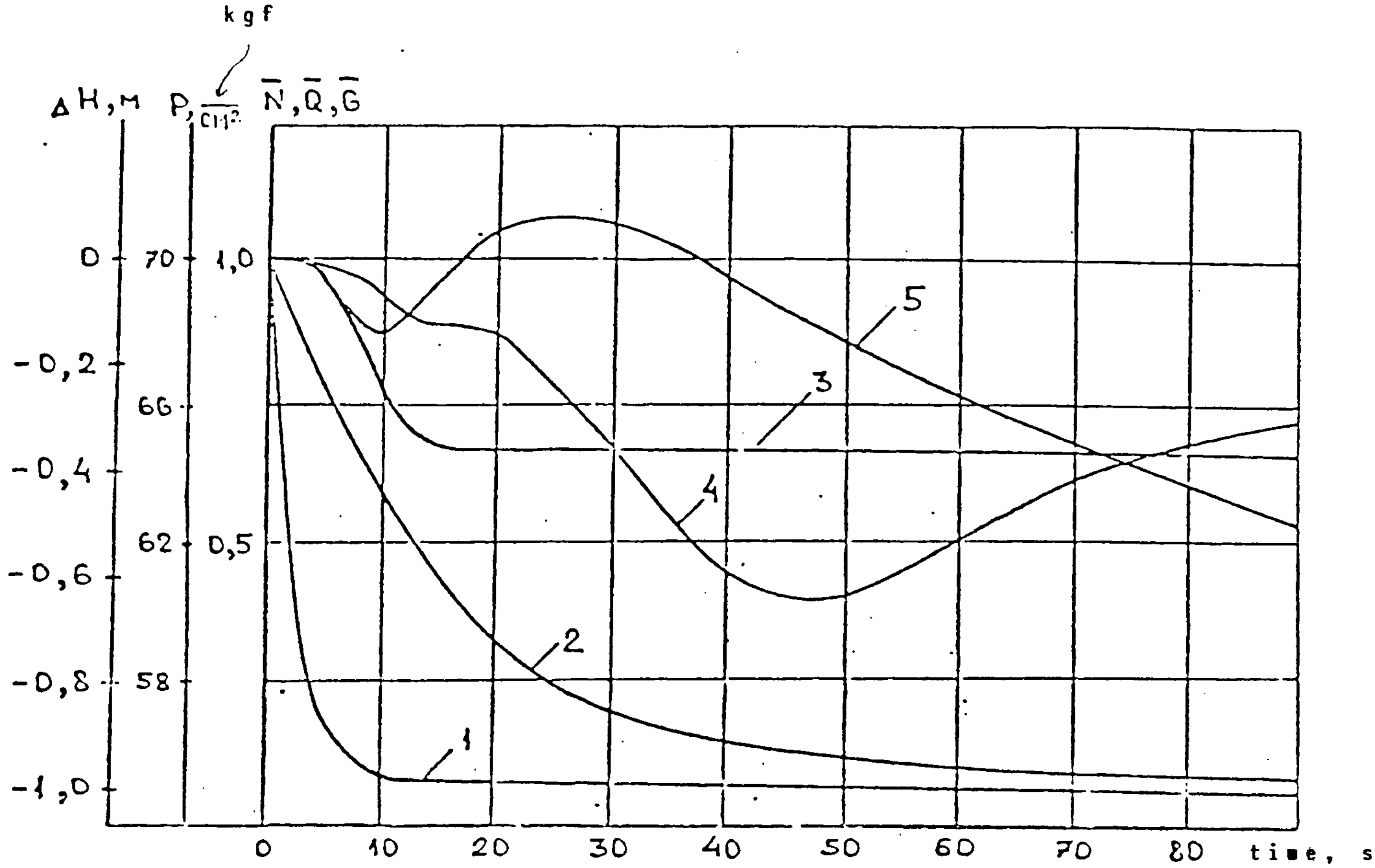


Fig 2.6 3. Behaviour of reactor parameters following activation of emergency protection system AZ-5

I - Nuclear power ($\bar{N}$)

2 - Thermal power ($\bar{Q}$)

3 - Circulating water flow rate ($\bar{G}$)

4 - Change in the level in the separators (ΔH)

5 - Pressure in separators (P)

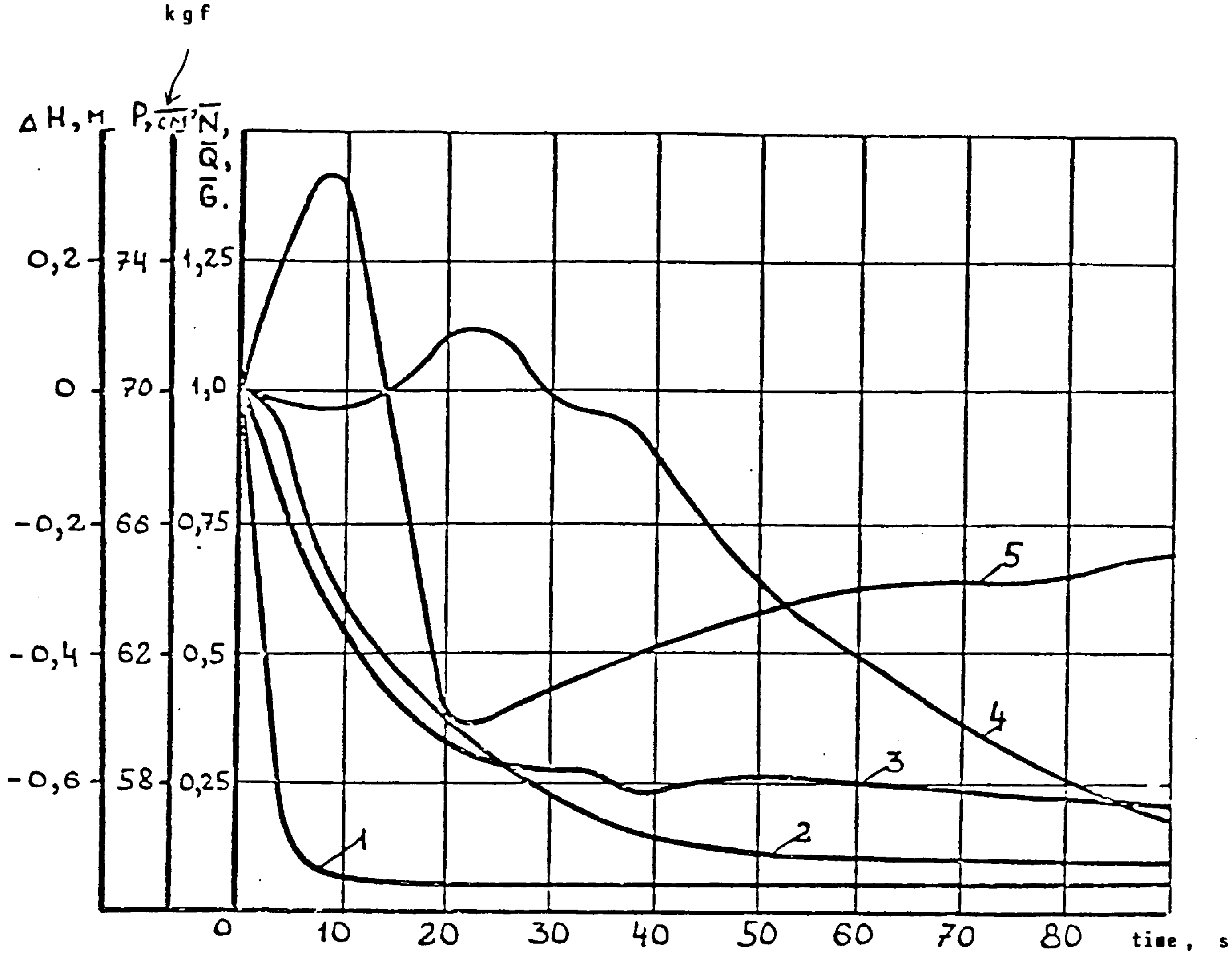


Fig 2.6.4. Behaviour of reactor parameters following loss of unit internal load.
For legend, see Fig 2.6.3.

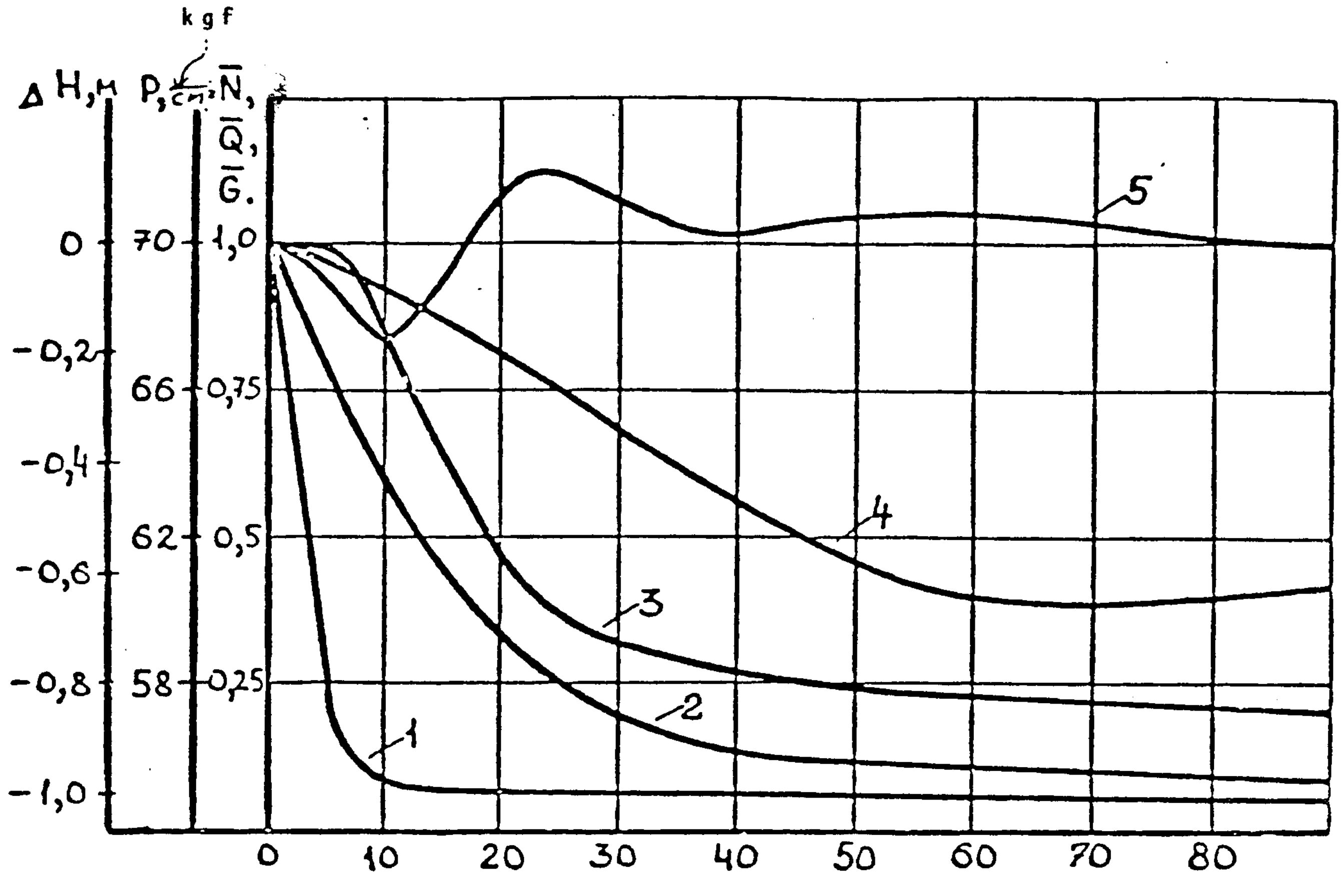


Fig. 2.6.5. Transient process following a total instantaneous cut-off of feedwater. For legend, see Fig. 2.6.3.