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State Committee for Using the Atomic Energy of USSR

АВАРИЯ НА ЧЕРНОБЫЛЬСКОЙ АЭС И ЕЕ ПОСЛЕДСТВИЯ

THE ACCIDENT AT THE CHERNOBYL AES AND ITS CONSEQUENCES

Информация, подготовленная для совещания экспертов МАГАТЭ

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PREFACE

The information presented here is based on conclusions of the Government Commission on the causes of the accident at the fourth unit of the Chernobyl' Nuclear Power Station and was prepared by the following experts employed by the USSR State Commission Committee on the Use of Atomic Energy:

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Ministry of Health, the State Committee on Nuclear Safety, the Ministry of Defense, the Main Fire Protection Administration of the Ministry of Internal Affairs and the USSR Academy of Sciences.

INTRODUCTION

An accident occurred at the fourth unit of the Chernobyl' Nuclear Power Station on April 26, 1986, at 1:23 AM with damage to the active zone of the reactor and part of the building in which it was located.

The accident occurred just before stopping of the powerplant for scheduled maintenance during testing of the operating modes of one of the turbogenerators. The power output of the reactor suddenly increased sharply, which led to damage to the reactor and discharging of part of the radioactive products accumulated in the active zone into the atmosphere.

The nuclear reaction in the reactor of the fourth powerplant stopped in the process of the accident. The fire which broke out was extinguished, and operations were begun for containing and eliminating the consequences of the accident.

The population was evacuated from areas immediately adjacent to the area of the nuclear power plant and from a zone with a radius of 30 km around it.

In view of the extreme character of the accident which occurred at Chernobyl', an operations group headed by Prime Minister of the U.S.S.R. N. I. Ryzhkov was organized at the Politburo of the CC CPSU (Central Committee of the Communist Party of the Soviet Union) for coordinating the activity of ministries and other government departments in eliminating the consequences of the accident and rendering aid to the population. A Government Commission was formed and entrusted with studying the causes of the accident and carrying out the necessary emergency and reconstruction measures. The necessary scientific, technical and economic capabilities and resources of the country were provided.

Representatives of MAGATE were invited to the USSR and given the opportunity to familiarize themselves with the state of affairs at the Chernobyl' Nuclear Powerplant and measures for overcoming the accident. They informed the world community about their assessment of the situation.

The governments of a number of countries, many governmental, social and private organizations and individual citizens from various countries of the world appealed to various organizations of the USSR with proposals concerning participation in overcoming the after-effects of the accident. Some of these proposals were accepted.

In the thirty years of its development, nuclear power engineering has occupied an essential place in worldwide power production and, on the whole, has displayed high levels of safety for man and the environment. One cannot imagine the future of the world economy without nuclear power. However, its further development must be accompanied by still greater efforts on the part of science and engineering for ensuring its operational reliability and safety.

The accident at Chernobyl' was the result of coincidences of several events of low probability. The Soviet Union draws the proper conclusions from this accident.

Rejecting nuclear power sources would require a considerable increase in production and combustion of organic fuels. This would steadily increase the risk of human diseases and the loss of water and forests due to the continuous passage of harmful chemical substances into the biosphere.

The development of the world's nuclear power resources brings with it, in addition to gain in the area of the energy supply and the preservation of natural resources, dangers of an international character. These dangers include transfers of radioactivity across borders, especially in large-scale radiation accidents, the problem of the spread of nuclear weapons and the danger of international terrorism, and the specific danger of nuclear installations under conditions of war. All this dictates the fundamental necessity of deep international cooperation in the field of development of nuclear power systems and ensuring of their safety.

Such are the realities.

The saturation of the modern world with potentially dangerous industrial processes, in significantly intensifying the effects of military operations, places the question of the senselessness and unacceptability of war under modern conditions on a new plane.

In a speech on Soviet television on May 14, M. S. Gorbachev stated: "The indisputable lesson of Chernobyl' for us lies in the fact that under conditions of further expansion of the scientific and technical revolution, questions of the reliability of equipment and its safety and questions on discipline, order and organization take on primary importance. The strictest requirements are needed everywhere.

Furthermore, we consider it necessary to move toward a serious deepening of cooperation within the framework of the International Agency on Atomic Energy."

CHAPTER 1. DESCRIPTION OF THE CHERNOBYL' NUCLEAR POWER STATION WITH RBMK-1000 REACTORS

1.1 Design Data

The planned power of the Chernobyl's Power Station (ChAES), was 60MW, and on January 1, 1986, the power of four units of the AES was 4000MW. The third and fourth units belong to the second phase of the ChAES and to the second generation of these Nuclear Power Stations (AES).

1.2 Description of the Reactor Installation (RU) of the Fourth Unit of the ChAES

The basic design features of RBMK reactors are as follows:

1) vertical channels with the fuel and the heat-transfer agent, which permit local reloading of fuel with a working reactor;

2) fuel in the form of bundles of cylindric fuel elements of uranium dioxide in zirconium shell tubes;

3) a graphite moderator between channels;

4) a low-boiling heat-transfer medium in the forced circulation rrcirculation mode (KMPTs) with direct feeding of steam to the turbine.

These design decisions in combination condition all the basic features of the reactor and the AES, both advantages and shortcomings. The advantages include: the absence of reactor vessels, which are awkward to produce on the powerplant maximum capacity and on the production base; the absence of a complex and expensive steam generator; the possibility of continuous reloading of fuel and a good neutron balance; a flexible fuel cycle, which is easily adapted to variations in the fuel market conditions;

the possibility of nuclear superheating of the steam; high thermodynamic reliability of the thermal equipment and viability of the reactor due to the controlling of the flow rate for each channel separately, monitoring of the integrity of the channels, monitoring of the parameters and radio activity of the heat-transfer medium of each channel and replacement of damaged channels while running. The shortcomings include: the possibility of the development of a positive void coefficient of reactivity due to the phase change in the heat-transfer agent which determines the transient neutronic behavior; high sensitivity of the neutron field to reactivity disturbances of different kinds, necessitating a complex control system for stabilizing the distribution of the release of energy in the active zone; complexity of the inlet-outlet piping system for the heat-transfer agent of each channel; a large amount of thermal energy accumulated in the metal structures, fuel elements and graphite block structure of the reactor; slightly radioactive steam in the turbine.

The RBMK-1000 reactor with a power of 3200 MW (thermal) (Fig. 1) is equipped with two identical cooling loops; 840 parallel vertical channels with heat-releasing assemblies (TVS) are connected to each loop.

A cooling loop has four main parallel circulation pumps (three working pumps feeding 7000 t/h of water each with a head of about 1.5 MPa, and one back-up pump).

The water in the channels is heated to boiling and partially evaporates. The water-steam mixture with an average steam content of 14% by mass is bled through the top part of the channel and a water-steam line into two horizontal gravity separators. The dry steam (with a moisture content less than 0.1%) separated in them passes from each separator at a pressure of 7 MPa in two steam lines into two turbines with a power of 500 MW (electrical) each (all eight steam lines of the four separators are jointed by a common "ring"), and the water, after mixing with steam condensate, is fed by 12 down pipes into the intake collector of the main cooling pumps.

Condensate of the steam exhausted from the turbines is returned by feed water pumps through separators into the top part of the down pipes, creating underheating of the water to the saturation temperature at the main cooling pump inlet.

The reactor as a whole is made up of a set of vertical channels with fuel and the heat-transfer medium built into cylindric apertures of graphite columns, and top and bottom protective plates. A light cylindric housing (casing) encloses the space of the graphite block structure.

The block structure consist of graphite blocks with a square cross section with cylindric apertures along the axis assembled into columns. The block structure rests on the bottom plate, which transmits the weight of the reactor to a concrete shaft.

About 5% of the reactor power is released in the graphite from slowing down of neutrons and absorption of gamma quanta. For reducing the thermal resistance and preventing graphite oxidation, the block structure is filled with a slowly circulating mixture of helium and nitrogen, which serves at the same time for monitoring the integrity of the channels by measuring the humidity and temperature of the gas.

There are spaces under the bottom and over the top plates for placing heat carrier pipes on routes from the separator drums (BS) and distributing collectors to each channel.

A robot - a loading and unloading machine (RZM) - after removal of the appropriate section of the plating and after being moved to the coordinates of the channel links with its head, balances its pressure with the pressure of the channel, unseals the channel, removes the burned-out (fuel elements (TVS) and replaces them with a fresh one, seals the channel, uncouples itself and transports the irradiated TVS to a holding tank. While the RZM is connected to the cavity of the channel (TK), a small flow of pure water passes from it through a thermohydraulic seal into the TK, creating a "barrier" to the penetration of the RZM by hot, radioactive water from the TK.

The system for control and protection (SUZ) of the reactor is based on movement of 211 solid absorber rods in specially isolated channels cooled with water of an independent duct. The system provides: automatic adjustment to a specified power level; a rapid reduction of the power level adjustment to by both rods of automatic regulators (AR) and rods of manual regulators (RR) according to malfunction signals from the basic equipment; emergence interruption of the chain reaction by emergency protection (AZ) rods according to signals of dangerous deviations of the parameters of the unit or malfunctions of the equipment; compensation for reactivity variations in heating up and emergence at power; regulation of the distribution of the release of energy over the action zone.

RBMK reactors are equipped with a large number of independent control systems, which are being moved into the active zone at a rate of 0.4 m/s in functioning of the AZ. The low rate of movement of the control systems is compensated for by the large number of systems.

The SUZ includes subsystems for local automatic control (LAR) and local emergency protection (LAZ). Both operate according to signals of ionization chambers inside the reactor. The LAR automatically stabilizes the fundamental harmonics of radial-azimuthal distribution of the release of energy, while the LAZ provides emergency protection of the reactor against exceeding the specified power of channel cartridges in reactor individual areas. Shortened absorber rods (USP) introduced into the zone from the bottom (24 rods) are included for controlling the power fields along the height of the reactor.

The RBMK-1000 reactor includes the following basic monitoring and control systems in addition to the SUZ:

1) a system for physical monitoring of the field of the release of energy along the radius (more than 100 channels) and the height (12 channels) by means of direct charging pickups;

2) a start-up monitoring system (neutron flux monitors, start-up fission chambers);

3) a system for monitoring the water flow rate along each channel with ball flowmeters;

4) a system for monitoring the integrity (KGO) of the fuel elements based on measuring the short-time activity of volatile fission products in water-steam lines (PVK) at the outlet from each channel; the activity is detected sequentially in each channel in appropriate optimum energy ranges ("windows") with a photomultiplier, which is moved from one PVK to another by a special carriage;

5) a system for monitoring the integrity of the channels (KTsTK) by measuring the humidity and the temperature of the gas flowing in the channels.

All the data pass to a computer. The information is given out to the operators in the form of deviation signals, indications (on call) and data of recorders.

The RBMK-1000 power units operate primarily in a base-load mode (at constant power output).

In view of the great power of the unit, a full automatic shut-down of the reactor occurs only if indicators of the power level, pressure or water level in the separator pass beyond acceptable limits, in a case of a general cut-off of electric current, disconnection of two turbogenerators or two main cooling pumps at once, a drop in the feedwater flow rate by a factor of more than 2, or full cross-sectioned rupture of the main outlet pipeof cooling pumps with a diameter of 900 mm. In other cases of equipment failures, only an automatic controlled reduction in power (to a level corresponding to the power of the equipment which has remained in operation) is envisaged.

1.3. Basic Physical Characteristics of the Reactor

The RBMK-1000 nuclear power reactor is a heterogeneous thermal channel reactor, in which uranium dioxide weakly enriched in regard to uranium-235 is used as fuel, graphite is used as moderator and boiling light water is used as the heat-transfer medium. The reactor has the following basic characteristics:

Thermal power	3200 MW
Fuel enrichment	2.0%
Uranium mass in a cartridge	114.7 kg
Number/diameter of fuel elements in TVS	18/13.6 mm
Depth of fuel burnup	20 MW day/kg
Coefficient of non-uniformity of release of energy along the radius	1.48
Coefficient of non-uniformity of release of energy along the height	1.4

Calculated maximum power of channel	3,250 kW
Isotopic composition of unloaded fuel:	
uranium-235	4.5 kg/t
uranium-236	2.4 kg/t
plutonium-239	2.6 kg/t
plutonium-240	1.8 kg/t
plutonium-241	0.5 kg/t
Void reactivity coefficient at a working point	2.0×10^{-6}/vol.% steam
Fast power reactivity coefficient at a working point	-0.5×10^{-6}/MW
Coefficient of expantion fuel temperature coefficient	-1.2×10^{-5} / °C
Coefficient of expantion graphite temperature coefficient	6×10^{-5}/°C
Minimum "weight" of rods of SUZ, ΔK	10.5%
Effectiveness of rods of RR, ΔK	7.5%
Effect of replacement (on the average) of the burnup TVS with fresh	0.02%

An important physical characteristic from the point of view of control and safety of the reactor is a value called the operating reactivity margin. The operating reactivity margin means the specific number of SUZ rods plunged into the active zone which are in a region of high differential efficiency. It is determined by recalculation for fully submerged SUZ rods.

The value of the reactivity margin for RBMK-1000 reactors is generally accepted as 30 RR rods. In this case, the rate of introduction of a negative reactivity in functioning of the AZ amounts to 1 "β"/s ("β" is the proportion of delayed neutrons), which is sufficient for compensation for positive reactivity effects.

The character of the dependence of the effective breeding coefficient on the density of the heat-transfer medium in RBMK reactors is determined to a great degree by the presence of absorbers of different kinds in the active zone. In initial charging of the AZ, which includes about 240 boron-containing additional absorbers (DP), dehydration results in a negative reactivity effect.

At the same time, a small increase in the steam content at nominal power with a reactivity margin of 30 rods results in an increase in reactivity ($=2.0 \times 10^{-4}$/vol.% steam).

For a boiling water-graphite reactor, the basic parameters which define its ability to properly operate and safety in the regard to thermal equipment are: the temperature of the fuel elements, the margin before the a crisis of heat transferoccurs, and the graphite temperature.

A set of computer codes which makes it possible to conduct operating calculations on station computers for ensuring plant reliability of thermal equipment of the powerplant in a mode of continuous reloading of fuel at any position of the cut-off and control valves at the inlet to each channel has been developed for RBMK reactors. Thus the possibility of determining the physical parameters of the reactor at variable frequency of the adjustment of channel flow rates and different control criteria (based on eighter outlet steam quality or on the margin of the critical power) and also as a function of the throttling of the active zone is provided.

For defining the fields of the release of energy over the active zone of a reactor, indications of the physical monitoring system, based on measurements of the neutron flow along the radius and height of the active zone taken inside the reactor, are used. In addition to indications of the physical monitoring system, data characterizing the composition of the active zone and the energy generation of each TK, the arrangement of the regulating rods, the distribution of water flow rates along channels of the active zone and readings of gages of the pressure and temperature of the heat-transfer medium re also entered into the station computer. As a result of calculations by the PRIZMA program performed periodically by the computer, the operator receives information on a digital printing device in the form of a cartogram of the active zone, which indicates the type of loading of the active zone, the arrangement of regulation rods, the network of the arrangement of pickups inside the reactor, and the distribution of power levels, water flow rates, reserves up to critical powers and reserves up to the maximum acceptable thermal loads on the fuel elements in regard to each fuel channel of the reactor. The station computer also computes the overall thermal power of the reactor, the distribution of flow rates of the steam-water mixture among the separators, the integral generation of power, the steam content at the outlet from each TK and other parameters necessary for monitoring and controlling the installation.

The experience of operation of active RBMK reactors indicates that with the means for monitoring and control available on these reactors, maintaining temperature conditions of the fuel and the graphite and reserves before a crisis of convective heat transfer at an acceptable level causes no difficulties.

1.4. Safety Assurance Systems (Figs. 2 and 3)

1.4.1. Protective Safety Systems

The system for emergency cooling of the reactor (SAOR) is a protective safety system and is intended for providing elimination of the residual release of heat by prompt feeding of the required amount of water into reactor channels in accidents accompanied by disruption of cooling of the active zone.

Such accidents include: ruptures of large-diameter KMPTs pipelines, ruptures of steam lines, and ruptures of feedwater pipelines.

The system for protection against an excess of pressure in the main heat carrier duct is intended for providing an acceptable pressure level in the duct due to removal of steam into a perforated sprayer tank for its condensation.

The system for protection of the reactor space (RP) is intended for ensuring that an acceptable pressure is not exceeded in the RP in an emergency situation with rupture of one operating channel due to removal of the steam-gas mixture from the RP into the screen of steam-gas discharges of the sprayer tank and then into the sprayer tank with simultaneous extinguishing of the chain reaction with the AZ facilities. The SAOR and the system for cooling the reactor space can be used for introducing the appropriate neutron absorbers (salts of boron and He).

1.4.2 Localizing Safety Systems

The system for localization of accidents (SLA) realized on the fourth unit of the ChAES is intended for localizing radioactive discharges in accidents with unsealing of any pipelines of the reactor cooling duct except the PVK pipelines, the top tracts of the operating channels and that part of the down pipes which is located in the separator drum compartment, and pipelines for steam-gas discharges from the RP.

The main component of the localization system is a system of airtight compartments, including the following compartments of the reactor division:

- tightly packed cells arranged symmetrically in relation to the reactor axis and designed for an excess pressure of 0.45 MPa:

- compartments of separator group collectors (RGK) and bottom water lines (NVK); these compartments do not permit an increase in excess pressure above 0.08 MPa according to the conditions of strength of components of the reactor structure and are designed for this value.

Compartments of tightly packed cells and the steam distributor corridor are connected to the water space of the perforated sprayer condensation device by steam outlet channels.

The cut-off and sealing armature system is intended for providing airtightness of the zone of localization of accidents by cutting off communicating lines connecting the sealed and unsealed compartments.

The bubbling condensation device is intended for condensation of steam formed:

- in the process of an accident with unsealing of the reactor contour;
- in functioning of the main safety valves (GPK);
- in leaks through the GPK in a normal operating mode.

1.4.3. Security Safety Systems

The AES Power Supply

Electric power users at an AES are divided into three groups, depending on the requirements placed on the reliability of the power supply:

1) users who cannot permit interruption of the feed for fractions of a second up to a few seconds under any conditions, including conditions of a total disappearance of alternating current voltage from working and back-up transformers for system needs, and who require the obligatory presence of a power supply after functioning of the reactor AZ;

2) users who can accept a power interruption of tens of seconds up to tens of minutes under the same conditions and require the obligatory presence of a power supply after functioning of the reactor AZ;

3) users who do not require the presence of a power supply in conditions of a disappearance of voltage from working and back-up transformers for system needs and in a normal model of operation of the unit can permit interruption of the supply for the time of transfer from a working to a back-up transformer for system needs.

1.4.4. Controlling Safety Systems

Controlling safety systems are intended for automatic engagement of devices of protective, localizing and security safety systems and for monitoring of their operation.

1.4.5. The Radiation Monitoring System

The AES radiation monitoring system is a component (subsystem) of the AES automated control system and is intended for collection, processing and display of information concerning the radiation situation in compartments of the AES and in the external environment, the condition of operating facilities and ducts, and irradiation doses to personnel in accordance with active norms and legislation.

1.4.6. AES Control Points

Control of the AES is carried on at two levels: station and plant.

All the control systems which ensure safety of the AES are located at the plant level.

1.5. Description of the Area of the Chernobyl' AES and the Areas in Which It is Located

1.5.1. Description of the Region

The Chernobyl' AES is located in the eastern part of a large region known as the Belorussian-Ukrainian Alluvial Plain, on the banks of the Pripyati River, which flows in the Dnepr. This region is characterized by a relatively flat relief with very slight surface slopes in the direction of the river and its tributaries.

The total length of the Pripyati up to its flow into the Dnepr is 748 km; the area of the drainage basin at the AES site is 106 thousand km , and the width is 200-300 m. The average flow speed is 0.4-0.5 m/s, and the average water flow rate over many years is 400 m^3/s.

The water-bearing level, which is used for domestic and drinking water needs of the region in question, lies at a depth of 10-15 m in relation to the current depth of the Pripyati and is separated from Quaternary deposits by clay marls which are relatively impermeable to water.

The region of the Belorussian-Ukrainian Alluvial Plain as a whole is characterized by a low population density (before the beginning of construction of the Chernobyl' AES, the average population density in the region in question was approximately 70 people per km).

At the beginning of 1986, the total population in a 30-kilometer zone around the AES amounted to about 100 thousand people, of whom 49 thousand lived in the city of Pripyati, located west of the three-kilometer sanitary-protection zone of the AES, while 12.5 thousand lived in the regional center, the city of Chernobyl', located 15 km to the southeast of the AES.

1.5.2. Description of the AES Areas and Its Structures

The first phase of the Chernobyl' AES, composed of two power units with RBMK-1000 reactors, was built in the period of 1970-1977, and construction of two power units of a second phase was completed at the same site by the end of 1983.

Construction of another two power units with reactors of the same kind (the third phase of the AES) was begun 1.5 km southeast of this site in 1981.

To the southeast of the AES site, right in the valley of the Pripyati River, a water cooling pond was built with an area of 22 km^2; the pond provides cooling of turbine condensers and other heat exchangers of the first four power units. The normal retaining level of water in the cooling pond was adopted as 3.5 m below the grading mark of the AES site.

Two high-capacity cooling towers (a hydraulic load of 100 thousand m /h each), which can operate parallel with the cooling pond, are being built as part of the third phase of the AES.

To the west and north of the site of the first and second phases of the AES is the area of the construction base and the supply department.

1.5.3. Data on the Number of Personnel at the AES Site During the Accident

There were 176 duty operating personnel and, also, other workers of various shops and repair services at the site of the first and second phases of the Chernobyl' AES on the night of April 25 and 26, 1986.

In addition, 268 construction workers and assemblers were working on the night shift at the site of the third phase of the AES.

1.5.4. Information About the Equipment at the Site Which Operated Together With the Damaged Reactor and About the Equipment Used in the Process of the Overcoming the Accident

Construction of the Chernobyl' AES is carried out in phases, which each consist of two power units and have special water purification systems common to the two units and have special water purification systems common to the two units and auxiliary structures and the industrial site which include:

- storage for liquid and solid radioactive wastes;
- open distributor devices;
- gas equipment;
- back-up diesel generator power plants;
- hydraulic engineering and other structures.

The storage for liquid radioactive wastes, built as part of the second phase of the AES, is intended for collection and temporary storage of liquid radioactive wastes arriving in operation of the third and fourth units and for collection of water from operational flushing and its recovery for reprocessing. Liquid radioactive wastes pass from the main housing by pipelines laid on the bottom level of a scaffold, while the solid radioactive wastes come to the storage by the top corridor of the scaffold by electric trucks.

A nitrogen-oxygen station is intended for satisfying the needs of the third and fourth units of the AES.

The gas equipment is made up of compressor, electrolysis, helium and argon tank equipment intended for providing the third and fourth units of the AES with compressed air, hydrogen, helium and argon. Receivers for storing nitrogen and hydrogen are located in open areas.

A back-up diesel power plant (RDES) is an independent emergency source of electric power for systems important to the safety of each unit. Three diesel generators with a unit power of 5.5 MW were installed on each RDES of the third and fourth units. Intermediate and base diesel fuel depots, pump transfers of fuel, and emergency fuel and oil drainage tanks are included for ensuring operation of the RDES.

The source of the technical water supply for the third and fourth units is the cooling pond.

The water of the circulation pump house, which is unified for the third and fourth units, is fed into a delivery tank, from which it passes by gravity flow into the turbine condensers.

Separate water works of the third and fourth units are included for supplying technical water to important users who require an uninterrupted water supply. A back-up power supply from diesel generators is available for these water works.

All four power units of the first and second phases and auxiliary systems and industrial area facilities involved with their normal operation were working on April 25, 1986.

CHAPTER 2. CHRONOLOGY OF THE DEVELOPMENT OF THE ACCIDENT

The Chernobyl' Powerplant No. 4 was put into operation in December, 1983. By the time of stopping of the plant for a medium repair, which was planned for April 25, 1986, the active zone contained 1659 TVS with an average burnup of 10.3 MW day/kg, 1 DP and 1 unloaded channel. The main part of the TVS (75%) were cartridges of the first loading with a burnup of 12-15 MW day/kg.

Tests of turbogenerator No. 8 in a runout mode with the auxiliary consumption load only internal needs were planned just before stopping. The purpose of these tests was to experimentally verify the possibilities for using mechanical inertia energy of the rotor of a turbogenerator disconnected from steam supply, in order to generate electricity for auxiliary motors what may be required if the turbogenerator is disconnected from an electric grid. This mode is used in one of the subsystems of the high-speed system for emergency cooling of the reactor (SAOR). With the proper order of performance of the tests and additional safety measures, the performance of tests of this kind on a working AES was not prohibited.

Such tests had already been performed previously at this station. It was established at that time that the voltage on the generator busses drops much before the mechanical (inertia) energy of the rotor in running down. In the tests scheduled for April 25, 1986, the use of a special system to control regulator of the magnetic field of the generator, which was to have eliminated this shortcoming, was planned. However, the "Working Program of Tests for Turbogenerator No. 8 of the Chernobyl' AES" in accordance with which the tests were to have been conducted was not prepared and approved in the proper way.

The quality of the program proved low; the section on safety measures included in it was composed purely as a matter of form. (It pointed out only that in the process of tests, all switching is done with the authorization of the station shift director; in case of development of an emergency situation, all personnel must act in accordance with local instructions; and just before the beginning of the tests, the test leader - an electrical engineer, who is not a specialist on reactor installations - briefs the watch on duty.) In addition to the fact that the programs essentially included no additional safety measures, it prescribed disengaging the system for emergency cooling of the reactor. This meant that throughout the period of the tests, i.e., about 4 hours, the safety of the reactor appears to have been lowered significantly.

On the strength of the fact that the proper attention was not devoted to the safety of these tests, the personnel were not ready for them and did not know about the possible dangers. In addition, as one will be able to see from what follows, personnel deviated from carrying out the program, thereby creating the conditions for development of an emergency situation.

The personnel started to reduce the power output of the reactor, which had been operating at nominal parameters, at 1:00 AM on April 25, and at 1:05 PM turbogenerator No. 7 (TG No. 7) was disconnected from the grid at a reactor thermal output of 1600 MW. The electric power supply for the auxiliaries (4 main cooling pumps, 2 feed water pumps) was transferred to the busses of turbogenerator No. 8.

The SAOR was disengaged from the KMPTs at 2:00 PM in accordance with the test program. However, taking the unit out of operation was delayed according to a request from the dispatcher centre. Operation of the plant continued at this time with a disengaged SAOR in violation of the regulations.

The turbogenerator was continued at 11:10 PM. In accordance with the test program, the runout of the generator with a load of the plant auxiliaries was to be conducted at a reactor power of 700-1000 MW (thermal). However, with disengagement of the LAR (Local automatic control) system, which was necessary for operation of the reactor at a low power output, the operator was not able to eliminate the imbalance of the measurement part of the AR (automatic regulator) which developed quickly enough. As a result, the power dropped to a level below 30 MW (thermal). Only by 1:00 AM on April 26, 1986, did the personnel manage to stabilize it at a level of 200 MW (thermal). In connection with the fact that "contamination" of the reactor continued during this period, further raising of the power was rendered difficult due to the small operating reactivity margin, which was substantially below the required level by this moment.

Nevertheless, it was decided to perform the tests. At 1:03 and 1:07 AM, two more main coding pumps, one from each side were engaged in addition to the six pumps which had been operating, so that after the end of the experiment, in which four pumps were to operate to support the runout mode of operation, four pumps would remain in the forced circulation loop (KTPT) reliable cooling of the active zone.

Since the reactor power and, consequently, the hydraulic resistance of the active zone and the KMPTs were substantially below the planned level and all the eight pumps were in operation, the total flow rate through the reactor increased to (56-58) x 10 m^3/h and the rate in regard to an Individual pump increased to 8000 m^3/h, which is a violation of the operating regulations. Such a mode of operation is prohibited due to danger of interruption of the pump operation and the possibility of development of vibrations of the main feed water lines as a result of cavitation. Connection of the additional pumps and the increase in the water flow rate through the reactor caused by this resulted in a decrease in steam generation a drop in the steam pressure in the separators and changes in other parameters of the reactor. The operators tried to maintain the following basic reactor parameters manually: the steam pressure and the water level in the separators however, they were not able to accomplish this fully. Dips in steam pressure by 0.5-0.6 MPa and dips in the water level below the emergency point were observed in the separators during this period. In order to avoid shutdown of the reactor under such conditions, personnel blocked the emergency protection signals in regard to these parameters.

Meanwhile the reactivity of the reactor continued to drop slowly. At 1:22:30 AM, the operator noticed on the printout of the program for quick evaluation of the reactivity margin reserve that the operating reactivity margin was at a value requiring shutdown of the reactor. Nevertheless, this did not stop the personnel, and the tests began.

At 1:23:04, the shutdown control valves (SRK) of turbogenerator No. 8 were closed. The reactor continued operating at a power of about 200 MW (thermal). The available emergency protection for closing the SRK of the two turbogenerators No. 7 had been disengaged during the afternoon of April 25, 1986) was blocked in order to have the possibility of repeating the test, if the first attempt proved unsuccessful. Thus another departure had been made from the testing program, which did not envisage blocking the emergency protection of the reactor with respect to disengagement of two turbogenerators.

A slow increase in power began some time after beginning of the test.

At 1:23:40 the shift manager of the; plant gave the command to press pushbutton AZ-5, on a signal from which all control rods and emergency protection rods are inserted into the active zone. The rods went down, although impacts were heard, and the operator saw that the absorber rods stopped without reaching the bottom ends. Then he cut off the servodrive couplings, so that the rods fell into the active zone by their own weight.

According to the evidence of witnesses who were outside the fourth plant, two explosions were heard, one after another, at 1:24; some kind of hot fragments and sparks flew up above the fourth plant, some of which fell on the roof of the turbogenerator room and started a fire.

CHAPTER 3. ANALYSIS OF THE PROCESS OF THE DEVELOPMENT OF THE ACCIDENT ON A MATHEMATICAL MODEL

The "Skala" centralized monitoring system (STsK) of the RBMK-1000 reactor includes a program for diagnostic recording of parameters (DREG), according to which several hundred analog and discrete parameters are examined and stored periodically with a specified cycle (the minimum cycle time is 1 s).

In connection with performance of the tests, only those parameters which were important from the point of view of analysis of the results of the tests being performed were recorded with high frequency. Therefore, reconstruction of the process of development of the accident was performed by calculation on a mathematical model of the power unit with the use not only of printouts of the DREG program but also of readings of instruments and the results of questioning of personnel.

An integral mathematical model of a power unit with an RBMK-1000 reactor, realized by computer in real time, was used for providing accelerated analysis of variations and versions of the emergency situation in question. Dependences of reactance on the steam content and movement of the absorber rods were defined according to results of calculations on distributed, including three-dimensional, neutron-physics models.

In calculation reconstruction of the process of development of the accident, it was extremely important to make sure that the mathematical model of the power unit accurately describes the behavior of the reactor and the other equipment and systems under just those conditions making up the situation just before

the breakdown. As already mentioned in the previous section, the reactor was operating in an unstable manner after 1:00 AM on April 26, 1986, and the operators were introducing "disturbances" into the control object practically continuously for stabilizing its parameters. This made it possible to compare actual data recorded with adequate reliability by recording devices to data optained in numerical simulation for quite a large time interval under various effects on the reactor installation. The comparison results proved quite satisfactory, which attests to the adequacy of the mathematical model and the real object.

In order to present the effect of prehistory on the character of development of the accident more clearly, we shall analyze the calculation data beginning from 1:19:00 AM, i.e., 4 minutes before the beginning of the test with rundown of the TB (Fig. 4.). This moment is convenient in that the operator began one of the operations for replenishment of the separator drums (the second since 1:00), which introduced strong disturbances into the regulation object. At this moment, the DREG program recorded the positions of rods of all three AR; i.e., the initial conditions for the calculation were clearly recorded.

The operator began replenishment of the separator drums to avoid allowing a dip in the water level in them. He succeeded in maintaining the level in 30 s, having increased the flow rate of feedwater by a factor of more than 3. The operator apparently decided not only to maintain the water level but to raise it. Therefore, he continued increasing the water flow rate, and it exceeded the original flow rate by a factor of 4 in just about a minute.

As soon as colder water from the separating drums reached the active zone, steam generation decreased noticeably, causing a decrease in the volumetric steam content, which resulted in movement of all the AR rods upward. In about 30 s they emerged at the top ends, and the operator was forced to "help" them with manual control rods, thereby reducing the operating reactance reserve. (This operation was not recorded in the operation log, but it would have been impossible to maintain power at a level of 200 MW without it.) The operator, having moved the manual rods up, achieved recompensation, and one of the groups of AR rods was lowered by 1.8 m.

The decrease in steam generation led to a small pressure decrease. After about a minute, at 1:19:58, a high-speed reduction device (BRU-K), through which steam surpluses were released into the condenser, was closed. This promoted some decrease in the rate at which the pressure was dropping. However, the pressure continued to drop slowly up to the beginning of the test. It changed by more than 0.5 MPa during this period.

A printout of the actual fields of releases of energy and the positions of all the regulation rods was obtained on the "Skala" STsK at 1:22:30. An attempt has been made at "tying together" the calculated and recorded neutron fields by just this moment.

The overall characteristics of the neutron field at this moment were as follows: it was practically arched in a radialazimuthal direction and double-peaked, on the average, in regard to height, with a higher release of energy in the top section of the active zone. Such a field distribution is

quite natural for the situation of the reactor: a depleted active zone, almost all the regulation rods up, a volumetric steam content significantly higher in the top part of the active zone than at the bottom, contamination with ^{135}Xe higher in the central parts of the reactor than in the peripheral parts.

The reactance reserve amounted to a total of ~~6-8~~ 6-8 rods at 1:22:30. This value was at lease two time lower than the minimum acceptable reserve established by technical operating regulations. The reactor was in an unusual, nonregulation condition, and for evaluating the subsequent development of events, it was extremely important to determine the differential efficiency of rods for regulation and emergency protection in real neutron fields and the fission characteristics of the active zone. Numerical analysis indicated high sensitivity of the error in determining the efficiency of the regulation rods to the error in reconstruction of the vertical field of releases of energy. If one takes into account in addition that at such low power levels (about 6-7%), the relative field measurement error is substantially higher than under nominal conditions, the need for analyzing an extremely large number of calculation versions to ascertain the reliability or inaccuracy of some version becomes clear.

The reactor parameters were closest to stable for the time period in question by 1:23, and the tests began. A minute before this, the operator sharply reduced the feedwater flow rate, which occasioned an increase in the water temperature at the inlet to the reactor with a delay equal to the time of passage of the heat-transfer medium from the separator drums to the reactor.

At 1:23:04 the operator closed the SRK of TG No. 8 and began rundown of the turbogenerator. Due to the decrease in the flow rate of steam from the separator drums, its pressure began to increase slightly (at a rate of 6 kPa/s, on the average). The total water flow rate through the reactor began to drop due to the fact that four of the eight GTsN were working off the turbogenerator which was "running down."

The increase in the steam pressure, on the one hand, and the decrease in the water flow rate through the reactor and also in the feedwater supply to the separator drums, on the other, are competing factors which determine the volumetric steam content and, consequently, the power of the reactor. It should be emphasized in particular that in the condition at which the reactor arrived, a small change in the power results in a situation where the volumetric steam content, which directly influences reactance, increase many times more sharply than at nominal power. The competition of these factors led in the final analysis to a power increase. Just this situation could be the cause for pressing button AZ-5.

Pushbutton AZ-5 was pressed at 1:23:40. Insertion of emergency protection rods began. By this time, the AR rods, in partially compensating for the previous increase in power, were already located in the bottom part of the active zone, while the work of personnel with an unacceptably low operating reactance reserve resulted in a situation where practically all the other absorber rods were located in the top section of the active zone.

Under the conditions which had been created, the disruptions permitted by the personnel resulted in a significant decrease in

the efficiency of the emergency protection. The total positive reactance developing in the active zone began to increase. After 3 s the power exceeded 530 MW, and the runaway period came to be much less than 20 s. The positive steam effect of reactance promoted deterioration of the situation. Only the Doppler effect partially compensated for the reactance introduced at this time.

The continuing decrease in the water flow rate through the operating channels of the reactor under conditions of an increase in power led to intense steam formation and then to a crises of convective heat transfer, heating up of the fuel, its disintegration, rapid boiling of the heat-transfer agent, into which particles of disintegrated fuel were falling, a sharp increase in pressure in the operating channels, rupture of the channels and a thermal explosion, which destroyed the reactor and part of the structural components of the building and led to the release of active fission products into the environment.

Disintegration of the fuel was simulated in the mathematical model by a sharp increase in the effective heat-transfer surface area, where the specific release of energy in the fuel exceeded 300 cal/g. At just this time, the pressure in the active zone increased to the extent that a sharp decrease in the water flow rate from the GTsN occurred (the check valves closed). This can be seen clearly both from results obtained on the mathematical model and from measurement results recorded by the DREG program. Rupture of the operating channels alone led to partial reconstruction of the flow rates from the GTsN, although water passed from them into the reactor

space as well as into the surviving channels.

The steam formation and the sharp temperature increase in the active zone created the conditions for steam-zirconium and other exthermic chemical reactions. Witnesses observed their appearance in the form of fireworks of flying hot and glowing fragments.

A mixture of gases containing hydrogen and carbon monoxide capable of thermal explosion in mixing with air oxygen was formed as a result of these reactions. This mixing could occur after unsealing of the reactor space.

CHAPTER 4. CAUSES OF THE ACCIDENT

As the analysis presented above demonstrated, the accident at the fourth unit of the ChAES belongs to the class of accidents involved with introduction of excess reactance. The design of the reaction installation included protection against accidents of this type with consideration for the physical features of the reactor, including the positive steam coefficient of reactance.

The technical protection facilities include systems for control and protection of the reactor against a power excess and a decrease in the runaway period, blocking and protection against malfunctions or switching of the equipment and systems of the power unit, and a system for emergency cooling of the reactor.

Strict rules and an order for conducting the operating process at the AES, defined by power unit operating regulations, were also included in addition to the technical protection facilities. Requirements concerning the unacceptability of a decrease in the operating reactance reserve below 30 rods are among the most rules.

In the process of preparing for and conducting tests of a turbogenerator in a rundown mode with a load of system auxiliaries of the unit, the personnel disengaged a number of technical protection devices and violated the important conditions of the operating regulations in the section of safe performance of the operating process.

The table presents a list of the most dangerous violations of operating conditions committed by personnel of the fourth unit of the ChAES.

No.	Violation	Motivation	Results
1	Decrease in the operareactance reserve significantly below the acceptable value	Attempt to get out of "iodine pit"	Emergency protection of reactor proved ineffective
2	Power dip below value envisaged by testing program	Operator error in disengagement of LAR	Reactor proved to be in hard-to-control state
3	Connection of all GTsN to reactor with exceeding of flow rates established by regulations in regard to individual GTsN	Fulfillment of requirements of testing program	Temperature of heat-transfer medium of KMPTs came close to saturation temperature
4	Blocking of reactor protection on signal for shutdown of two TG	Intention to repeat experiment with disengagement of TG if necessary	Loss of possibility of automatic shutdown of reactor
5	Blocking of protection in regard to water level and steam pressure in separator drum	Attempt to conduct tests despite unstable operation of reactor	Protection of reactor in regard to thermal parameters was disengaged
6	Disengagement of system for protection against maximum theoretical failure (disengagement of SAOR)	Attempt to avoid false response of SAOR during performance of testing	Loss of possibility of reducing scale of accident

The basic motive in the behavior of the personnel was the attempt to complete the tests more quickly. Violation of the established order in preparation for and performance of the tests, violation of the testing program itself and carelessness in control of the reactor installation attest to inadequate understanding on the part of the personnel of the features of accomplishment of operating processes in a nuclear reactor and to their loss of a sense of the danger.

The developers of the reactor installation did not envisage the creation of protective safety systems capable of preventing an

accident in the presence of the set of premeditated diversions of technical protection facilities and violations of operating regulations which occurred, since they considered such a set of events impossible.

An extremely improbable combination of procedure violations and operating conditions tolerated by personnel of the power unit thus was the original cause of the accident.

The accident took on catastrophic dimensions in connection with the fact that the reactor was brought by the personnel to a condition so contrary to regulations that the effect of a positive reactance coefficient on the power build-up was intensified significantly.

5. INITIAL MEASURES TO INCREASE NUCLEAR POWER PLANT SAFETY WITH RBMK REACTORS

A decision has been made to reset terminal breakers of control rods on working nuclear power plants with RBMK reactors such that in the outermost position all rods are inserted into the core to a depth of 1.2 m. This measure increases the response efficiency of protection and precludes the possibility of the multiplication properties of the core from increasing in its lower part when the rod moves from the upper end piece. At the same time a number of absorber rods constantly in the core increases to 70 - 80; this reduces the steam void effect of reactivity to an allowable value. This is a temporary measure and in the future it will be replaced by converting RBMK reactors to fuel with initial enrichment 2.4% and placing additional absorbers in the core which ensure that positive coastdown of reactivity not exceed more than one beta for any change in coolant density.

A number of additional signallers of the cavitation reserve of reactor coolant pumps and an automatic system for computing reactivity reserve with output of an emergency reactor shutdown signal when the reserve drops below a given level are being installed. These measures have a somewhat adverse effect on economic indicators of nuclear power plants with RBMK, but guarantee the necessary safety.

In addition to technical measures organizational ones to strengthen plant discipline and increase operating quality are being implemented.

6. PREVENTING DEVELOPMENT OF AN ACCIDENT AND REDUCING ITS CONSEQUENCES

6.1 Fire Fighting on a Nuclear Power Plant

The primary task after a reactor accident was to control the fire.

As a result of explosions in the reactor an ejection of core fragments heated to high temperature onto the rooves of certain buildings of reactor section services, the deaerator, stack and turbine room more than 30 fires were started. Due to damage to individual oil lines, short circuits in electrical cables and intense thermal radiation from the reactor fire foci were formed in the turbine room above TG No. 7, in the reactor room and the partially destroyed compartments adjacent to it.

At one hour 30 minutes, fire fighting units for nuclear power plant protection from the cities of Pripyat' and Chernobyl arrived.

Due to the direct threat of the fire spreading over the cover of the turbine room to the adjacent third unit and its rapid intensification, primary measures were directed at eliminating the fire in this sector. Fires arising within compartments were fought using fire extinguishers and inside stationary fire cranes. By 2 hours 10 minutes most of the fires had been put out on the roof of the turbine room and by 2 hours 30 minutes on the roof of the reactor building. By 0500 the fire had been put out.

6.2 Estimating fuel condition after the accident

The accident led to partial destruction of the reactor core and complete destruction of its cooling system. Under these conditions, the state of the environment in the reactor shaft was determined by the following processes:

- residual heat release of the fuel due to decay of fission products

- heat release due to different chemical reactions taking place in the reactor shaft (hydrogen combustion, graphite and zirconium oxidation, etc.);

- heat discharge from the reactor shaft due to its cooling by flows of atmospheric air through holes formed in sealed (before the accident) shells surrounding the core.

To solve the problem of preventing accident development and limiting its consequences, during the first hours after the accident major efforts were devoted to estimating the fuel state and its possible change as time passed. To do this, the following analyses had to be done:

- estimate possible scales of melting (due to residual heat release) of fuel in the reactor shaft;

- study processes of the interaction of molten fuel with reactor structural materials and reactor shaft materials (metals, concrete and so forth);

- estimate the possibility of melting of construction materials of the reactor and the shaft due to heat release from the fuel.

Initially computations were done to estimate fuel state in the reactor shaft with allowance for leakage of fission products (PD) depending on time since the accident began.

Study of the dynamics of PD discharge from the reactor during the first few days after the accident showed that the fuel temperature change as time passed was nonmonotonic. It can be assumed that there were several stages in the temperature mode of the fuel. The fuel heated up at the instant of explosion. Temperature estimation from the amount of relative leakage (fraction of the isotope discharging from the fuel from its total content in the fuel at a given point in time) of iodine radionuclides showed that the effective temperature of the fuel remaining in the reactor building after the explosion was 1600 - 1800 K. During the next several dozen minutes, fuel temperature dropped due to release of heat to the graphite structure and reactor structures. This led to a drop in leakage of volatile PD from the fuel.

Here it was considered that the amount of PD discharge from the reactor shaft was determined during this time mainly by processes of graphite combustion and associated processes of migration of finely dispersed fuel and PD introduced into the graphite by the accident explosion in the reactor. Subsequently, the temperature of the fuel due to residual heat release began to rise. As a result, leakage of volatile radionuclides (inert gases, iodine, tellurium, cesium) from the fuel increased. With the subsequent temperature increase of the fuel leakage of other so-called nonvolatile radionuclides began. By 4 - 5 May, the effective temperature of the fuel remaining in the reactor unit stabilized and then began to drop.

The results of theoretical analyses of fuel state are shown in Fig. 5 which lists results which characterize residual radionuclide content in the fuel and also the temperature change of the fuel with allowance for leakage of PD from it depending on the time since the accident began.

Computations showed:

- maximum fuel temperature cannot reach its melting point;

- the PD emerges onto the fuel circuits in batches; this can lead only to local heatup on the fuel-environment boundary.

The PD escaping from the fuel fall on structural and other materials surrounding the reactor in the reactor unit according to condensation and precipitation temperatures of the fuel. Here radionuclides of krypton and zenon escape from the reactor unit almost completely, the volatile PD (iodine, cesium) to some extent and the others remain almost entirely within the reactor building.

Thus the energy of the PD is dissipated throughout the volume of the reactor unit.

As the result of these factors melting of the medium surrounding the fuel and fuel movement become of low probability.

6.3. Limiting the Accident Consequences in the Reactor Core

The potential of concentrating part of the molten fuel and establishing conditions for formation of critical mass and a self-containing chain reaction required measures against this danger. In addition, the destroyed reactor was a source of emissions of a large amount of radioactivity into the environment.

Immediately after the accident, an attempt was made to reduce the temperature in the reactor shaft and prevent combustion of the graphite structure using emergency and auxiliary feedwater pumps to supply water to the core space. This attempt was unsuccessful.

Immediately one of two decisions had to be made:

- Localize the focus of the accident by filling the reactor shaft with heat discharging and filtering materials;

- Allow combustion processes in the reactor shaft to end naturally.

The first option was taken since in the second the danger of radioactive damage to considerable areas with the threat to the health of the populations of large cities arose.

A group of specialists in military helicopters began to drop boron compounds, dolomite, sand, clay and lead onto the damaged reactor. From 27 April to 10 May almost 5000 tons of materials were dropped, most from 28 April through 2 May. As a result, the reactor shaft was covered by a layer of loose mass which intensely absorbed aerosol particles. By 6 May, the discharge of radioactivity ceased to be a major factor, having dropped to several hundred and by the end of the month dozens of curies per hour. At the same time, the problem of reducing fuel heatup was solved. To reduce temperature and oxygen concentration nitrogen from a compressor station was sent into the space under the reactor shaft.

By 6 May, the temperature increase in the reactor shaft stopped and began to drop due to formation of a stable convective air flow through the core into the free atmosphere. As insurance against extremely improbable (but possible during the first few days after the accident) failure of the lower tier of structures, it was decided to immediately establish an artificial heat discharge horizon under the building foundation in the form of a flat heat exchanger on a concrete slab. By the end of June the planned work was finished.

Experience showed that the decisions made were primarily the right ones.

From early May the situation had largely stabilized. Destroyed parts of the reactor building were in stable positions. The radiation situation following decay of the short lived isotopes improved. The exposure rate was single roentgens per hour in compartments under the reactor, in the turbine room and control panel compartments. Escape of radioactivity from the unit into the atmosphere was due mainly to wind entrainment of aerosols. The radioactivity of the releases did not exceed dozens of curies per day. Temperature conditions in the reactor shaft were stable. Maximum temperatures of various sections were several hundred degrees C with a steady trend towards dropping at a rate of roughly 0.5 degrees C per day.

The lower slab of the reactor shaft had been preserved and fuel was localized mainly (roughly 96%) in the reactor shaft an in compartments of steam water and lower steam service lines.

6.4 Measures at First-Third Blocks.

The following measures were taken on the first - third blocks after the accident on the fourth block:

- The first and second blocks were shut down at 0113 hours and 0213 hours on 27 April;

- The third block which was closely connected to the damaged fourth block but hardly suffered at all from the explosion was shut down at 0500 hours on 26 April;

- First - third blocks were prepared for prolonged cold shutdown;

- The nuclear power plant equipment following the accident was shifted into the cold reserve state.

The first - third blocks and power plant equipment were checked by on-duty personnel.

Considerable radioactiave contamination of equipment and compartments of the first-third power plant blocks was caused by entry of radioactive substances thorugh the ventilation system which continued to operate for some time after the accident.

Individual sections of the turbine room had major radiation levls since it was contaminated through the destroyed roof of the third block.

A government committee was assigned to organize decontamination and other operations on the first - third units. The objective was to prepare the units for startup and operation.

Decontamination was done using special solutions. Their composition was selected with allowance for the material to be washed (plastic compounds, steel, concrete, various coatings), the nature and level of surface contamination.

After decontamination, gamma radiation levels dropped by a factor of 10-15. Radiation dose rate for compartments of the first and second units in June was 2-10 mR/hr.

Final decontamination and stabilization of the radiation situation on the first - third units can be ensured only after completing decontamination on the nuclear power plant grounds and mothballing the damaged unit.

6.5 Monitoring and Diagnostics of the Condition of the Damaged Unit.

Diagnostic measurements made it possible to solve the following main problems:

- establish reliable monitoring of fuel movement;

- determine contamination scales on terrain adjacent to the power plant;

- estimate scales of damage and carry out dosimetry within the unit, determine the potential for working in undamaged compartments;

- determine distribution of fuel, fission products and others to generate raw data for design of mothballing facilities.

Among primary measurements monitoring of reactor state from the air was set up together with estimations of the radiation situation on the plant and around it. Radiation measurements, photographs of the damaged reactor building and its components in infrared radiation were done from helicopters and the chemical composition of gases discharged from the reactor shaft was analyzed; a number of other measurements were also taken. After it was established that compartments and equipment had survived in the lower part of the reactor building, it became possible to take initial measurements and install emergency monitoring instruments. First measuring instruments to measure neutron flux, gamma radiation dose rate, temperature and thermal flow were set up in the drained pressure suppression pool. Temperature measuring equipment was set up on a redundant basis. Evaluation of the situation in the pressure suppression pool showed the absence of any immediate danger of structures melting through. This confirmed the safe conditions for work to establish a lower protective slab.

The overall measurement strategy was as follows:

- Dosimetric and visual reconnaissance within the damaged unit;

- Radiometric and visual observation from helicopters;

- Measurement of the most important parameters (radioactivity, temperature, air flow) in surviving structures and accessible compartments.

Primary measurement efforts at the initial stage were directed at checking possible movement of fuel downward.

Solution of diagnostic problems became complicated for the following reasons:

- The regular measurement system had completely failed;

- Readings from sensors which may have survived were not accessible to personnel;

- Information on the state of compartments and the radiation situation in them was limited.

At the next stage locations of fuel discharge from the reactor shaft in the building had to be determined and its temperature and heat output conditions estimated.

To solve this problem, traditional dosimetric methods were used, and surviving pipelines for delivering measurement probes were opened. As a result, fuel distribution within the building was largely established.

The temperature in compartments under the reactor did not exceed 45 degrees C beginning in June; this indicated good heat output.

Monitoring and diagnostic methods were refined with allowance for this information.

6.6 Decontamination of the Nuclear Power Plant Site

During the accident radioactive materials were discharged over the plant grounds and fell onto the roof of the turbine room the roof of the third unit, and metal pipe supports.

The grounds of the plant, walls, and rooves of the buildings had considerable contamination due to precipitation of radioactive aerosols and radioacative dust. Contamination of the ground was non-uniform.

To reduce dispersion of radioactive dust on the grounds, roof of the turbine room building and shoulders of roads were treated with different polymerizing solutions to stabilize upper soil layers and preclude dust formation.

To establish conditions for comprehensive decontamination operations, the grounds of the nuclear power plant were divided into individual zones. Decontamination in each zone was done as follows:

- removal of trash and contaminated equipment from the grounds;

- decontamination of rooves and outside building surfaces;

- removal of 5-10 cm of soil and hauling it in containers to the solid waste storage pit of the fifth unit;

- placement of concrete slabs on the ground, if necessary, or clean soil;

- covering slabs and unconcreted grounds with fil forming compounds.

As a result of completed measures, the total gamma background in the area of the first unit was reduced to 20-30 mR/hr. This residual background was due mainly to external sources (damaged unit). This indicates the relative efficiency of decontamination of grounds and buildings.

6.7 Long Term Mothballing of the Fourth Unit

Mothballing of the fourth unit should ensure normal radiation situations on the surrounding territory and in the air as well as prevent escape of radioactivity into the environment.

To mothball the unit the following structures should be erected. (Figs. 6 - 8):

- outside protective walls along the perimeter;

- inside concrete dividers in the turbine room between the third and the fourth units, in unit "V" (Cyrillic alphabetical equivalent our "C"), and in the deaerator along the turbine room and on the side of the barrier near the tank "SAOR";

- metal divider in the turbine room betwen the second and third units;

- protective cover over the turbine room, and in addition the central hall and other reactor compartments should be sealed, the barrier near the tank "SAOR" and compartments of the northern GTsN for mothballing the barrier concreted, and protection established against radiation on the reactor unit side.

The thickness of the protective concrete walls is 1 m and greater depending on designs and the radiation situation.

There are two versions in the ventilation outline:

- open configuration with air purification using aerosol filters and discharge into the atmosphere through the existing pipe of the ventilation center;

- Closed configuration with heat discharge in a heat exchanger located in the upper part of the vented volume, while maintaining a partial vacuum in the building volume which is ensured by exhaust of air from the upper part of the volume and its discharge through filters and pipe into the atmosphere.

The aforementioned operations are carried out as follows.

1. On the grounds adjacent to the unit the surface layer of soilis removed on local sections using a special technique.

2. The grounds are concreted with the surface leveled; this allows self-propelled cranes and other equipment to move easily.

3. The rooves and walls of the building are decontaminated.

Special polymer adhesive pastes of varied compositions are used in areas with high radiation.

4. After the site is cleaned up and concreted metal frames of protective walls are installed and subsequently concreted.

5. As walls are erected work is done to set up the main structures which ensure complete mothballing the the fourth unit.

6.8. Decontamination of the 30-km zone and returning it to economic activity

Major radioactive contamination of areas adjacent to the nuclear power plant made it necessary to make a number of extreme decisions regarding the establishment of controlled zones, evacuation of population, prohibition or limitation on agricultural use of soil and so forth.

A decision was made to introduce three controlled zones: special, 10 and 30 km. Strict dosimetric monitoring of transport was set up and decontamination points deployed in them. On zone boundaries the workers were transported from one mode of transport to others to reduce transfer of radioactive substances.

The radiation situation within the 30-km zone will continue to change, especially in regions with a high gradient of contamination levels. Radionuclides will be dramatically redistributed over landscape elements according to relief characteristics. The question of re-evacuation of population can be posed only after the radiation situation has stabilized over the entire territory of the contaminated zone: burial of the fourth block, decontamination of the nuclear power plant site, and stabilization of radioactivity in areas with elevated contamination level.

Beginning in June a complex of hydraulic facilities began to be built to protect ground water and surface water in the vicinity of the Chernobyl plant from contamination, including:

- antifiltration wall in the soil along the partial perimeter of the nuclear power plant site and drawdown wells;
- curtain of the coolant pond;
- cutoff drainage curtain on the right bank of the Pripyat';

- intercepting drainage curtain in the southwestern sector of the nuclear power plant;

- drainage water purification facilities.

By this time, based on completed estimates of the situation with regard to contamination of the soil-vegetative cover of the 30-km zone, special agritechnical and decontamination measures were developed and implemented which made it possible to return the contaminated earth to agrocultural use. These measures included: changing the traditional methods of working the soil in this region, use of special compositions to suppress dust formation, changing methods of harvesting and handling the harvest and so forth.

7. Monitoring radioactive contamination of the cenvironment and the health of the population

7.1. Estimating amount, composition and dynamics of fission product release from the damaged reactor.

The following results were used as raw data for this estimate:

- systematic studies of radionuclide composition of aerosol samples collected above the damaged power plant unit from 26 April 1986;

- aerogammaphotography of the nuclear power plant region;

- analysis of precipitation samples;

- systematic data from national weather station measurements.

Discharge of radionuclides outside the damaged block of the Chernobyl plant was a long term process consisting of several stages.

In the first stage dispersed fuel from the damaged reactor was discharged. The radionuclide at this stage of escape corresponded roughly to their composition in the irradiated fuel, but enriched with volatile nuclides of iodine, tellurium, cesium, and inert gases.

In the second stage, from 26 April through 2 May 1986, the magnitude of discharge outside the damaged unit decreased due to measures taken to prevent burning of the graphite and to filter the discharge. During this period radionuclide composition in the discharge was also near their composition in the fuel. At this stage finely dispersed fuel was discharged from the reactor by a flow of hot air and by graphite combustion products.

The third stage of discharge is characterized by rapid increase in the magnitude of fission product escape beyond the reactor unit. In the predominant entrainment of volatile components was observed, in particular, iodine, and then the radionuclide composition again approached their composition in the irradiate fuel (on 6 May 1986).

This was due to heating of the fuel in the core to temperatures exceeding 1700oC by residual heat release. As a result of the temperature dependent migration of fission products and chemical transformations of uranium oxide fission products leaked from the fuel matrix and were entrained in aerosol form on graphite combustion products.

The last, fourth stage which began after 6 May was characterized by a rapid drop in discharge (Table 1). This was the result of special measures which had been taken, formation of higher melting compounds of fission products as a result of their interaction with introduced materials, stabilization andsubsequent drop in fuel temperature.

Nuclide composition of the discharge is shown in Table 2.

In air and precipitation samples fission products were found in the form of individual radionuclides (mainly volatile) and in fuel particle composition. In this case, particles (associates) were found with increased content of individual radionuclides (Cs, Ru, and so forth) formed by migration of fission products in the fuel in materials of the backfill and structures, and sorption on surfaces.

Total discharge of fission products (without radioactive inert gases) was roughly 50 megacuries; this corresponds roughly to 3.5% of the total amount of radionuclides in the reactor at the time of the accident. These data were computed for 6 May 86 with allowance for radioactive decay. Discharge of radioactive materials was essentially completed by this time.

Дата 1	Время после аварии, сут 2	q, МКи**) 3
26.04	0	12
27.04	1	4,0
28.04	2	3,4
29.04	3	2,6
30.04	4	2,0
01.05	5	2,0
02.05	6	4,0
03.05	7	5,0
04.05	8	7,0
05.05	9	8,0
06.05	10	0,1
09.05	14	~0,01
23.05	28	20.10^{-6}

Table 1. Daily discharge q of radioactive substances into the atmosphere from the damaged unit (without radioactive inert gases*)

Headings: column 1 - date; column 2 - time after accident, days; column 3 - q, megacuries**

* - error in estimating discharge + 50%. It is determined by the error of dosimetric instruments, radiometric measurements of radionuclide composition of air and soil samples, and also by the error caused by averaging precipitation over the area.

** - values of q were computed on 6 May 86 with allowance for radioactive decay (at the time of release 26 Apr 86 activity was 20 - 22 megacuries). See Table 2 for the composition of the discharge.

Table 2. Estimation of radionuclide composition of release from damaged unit of Chernobyl nuclear power plant*.

Column 1 - Nuclide **; columns 2 and 3 - activity of release, megacuries; column 4 - percentage of radioactivity discharged from the reactor by 6 May 86. Column 4, line 1 - possibly up to 100.

Нуклид**) (1)	Активность выброса, МКи		Доля активности, выброшенной из реактора к 06.05.86, % (4)
	26.04.86 (2)	06.05.86***) (3)	
^{133}Xe	5	45	Возможно,до 100
^{85m}Kr	0,15	—	"
^{85}Kr	—	0,9	"
^{131}I	4,5	7,3	20
^{132}Te	4	1,3	15
^{134}Cs	0,15	0,5	10
^{137}Cs	0,3	1,0	13
^{99}Mo	0,45	3,0	2,3
^{95}Zr	0,45	3,8	3,2
^{103}Ru	0,6	3,2	2,9
^{106}Ru	0,2	1,6	2,9
^{140}Ba	0,5	4,3	5,6
^{141}Ce	0,4	2,8	2,3
^{144}Ce	0,45	2,4	2,8
^{89}Sr	0,25	2,2	4,0
^{90}Sr	0,015	0,22	4,0
^{238}Pu	$0,1.10^{-3}$	$0,8.10^{-3}$	3,0
^{239}Pu	$0,1.10^{-3}$	$0,7.10^{-3}$	3,0
^{240}Pu	$0,2.10^{-3}$	1.10^{-3}	3,0
^{241}Pu	0,02	0,14	3,0
^{242}Pu	$0,3.10^{-6}$	2.10^{-6}	3,0
^{242}Cm	$0,3.10^{-2}$	$2,1.10^{-2}$	3,0
^{239}Np	2,7	1,2	3,2

* - estimate error + 50%, see remarks to Table 1 for explanation.

** - data are cited for activity of main radionuclides measured in radiometric analyses.

*** - total release by 6 May 86.

The composition of radionuclides in the accident release roughly corresponded to their composition in the fuel of the damaged reactor, differing from it by the increased content of volatile iodine, tellurium, cesium and inert gases.

7.2. Monitoring system

At the time of the accident the regular system of meteorological, radiation and sanitary-hygienic monitoring began to operate according to the emergency plan. As soon as the scale of the accident became clear the monitoring system began to expand by enlisting additional groups of specialists and equipment. During the first few days after the accident primary attention was focused on immediate problems of radiation, sanitary-hygienic and medical-biological monitoring.

At the same time the monitoring system began to expand with consideration of long term problems. Organizations from Goskomgidromet of the USSR, the Ministry of Health of the USSR and union republics, the academies of sciences of the USSR, Ukrainian SSR, Byelorussian SSR, the GKAE of the USSR, Gosagroprom and others were involved in its formation.

Specialized medical facilites in Moscow and Kiev were enlisted to treat irradiated individuals.

Together with formation of the monitoring system a program of radioecological, medical-biological and other scientific problems of estimation and prediction of the effect of ionizing radiation on man, flora and fauna was set up and began to be executed.

The primary tasks of monitoring were:

- estimating the possible level of internal and external irradiation of Chernobyl power plant personnel, the population of the Pripyat' and the 30-km zone;

- estimating possible level of irradiation of the population in a number of regions outside the 30-km zone, with a level of radioactive contamination which could exceed allowable limits;

- development of recommendations for measures to protect population and personnel from irradiation above established limits.

These recommendations include:

- evacuation of population;

- prohibitition of limitation on use of food products with increased content of radioactive substances;

- recommendations for behavior of the population in houses and in open terrain.

To solve these initial problems systematic monitoring of the following was done:

- Gamma radiation level in contamination regions;

- concentration of biologically significant radionuclides in the air and water of reservoirs, in particular those used for drinking water supply;

- density of radioactive contamination of the soil and vegetation and its radionuclide composition;

- content of radioactive substances in food products, in particular iodine-131 in milk;

- radioactive contamination of special clothing, personnel clothing and footwear, transport resources and so forth;

- accumulation of radionuclides in the internal organs of individuals and so forth.

7.3. Main characteristics of radioactive contamination of the atmosphere and locale, possible ecological consequences.

Radioactive contamination of the environment as a result of the accident at Chernobyl unit No. 4 was determined by the dynamics of radioactive release and weather conditions.

The raedioactively contaminated airstream spread initially in the western and northern sectors, during the two or three days following the accident in the northern sector, from 29 April for several days in the southern sector. The contaminated air masses then spread great distances over the territory of the Byelorussian SSR, Ukrainian SSR, and the RSFSR.

On 27 April the height of the stream exceed 1200 m, the radiation level in it at a distance of 5 - 10 km from the accident site was 1000 mR/hr. The stream and radioactive trace which formed were regularly photographed by aircraft of the Goskomgidromet equipped with sampling, roentgenometric, and gammaspectrometric equipment, and in the network of weather stations.

Fission products as well as products of induced activity Np-239 and Cs-134 were detected in the air samples.

The main zones of terrain contamination following the accident formed in the westerm northwestern and northeastern directions from the power plant, and on a smaller scale in the southern direction. Radiation levels near the nuclear power plant exceeded 100 mR/hr, in the western trace maximum radiation levels 15 days after the accident were 5 mR/hr at a distance of 50 - 60 km from the accident zone (maximum distances), in the north at a distance of 35 - 40 km. In Kiev radiation levels early in May reached 0.5 - 0.8 mR/hr.

In the near zone of the trace plutonium isotopes (their propagation in the locale was insignificant) were identified (in addition to those mentioned above). In this zone fractionation of the isotopes was not significant, but in the far trace radioactive products were greatly enriched by isotopes of tellurium, iodines, and cesium.

Integration of contaminated areas made it possible to determine the total activity of precipitated radioactive materials (outside the site). In the zone of near and far precipitation in the European part of the USSR it was roughly 3.5% (see section 7.1) of the total activity of the fission products and activity accumulated in the reactor (in the near trace roughly 1.5 - 2%).

Addition of the activity of radionuclides precipitated in the near trace and determined by taking soil samples yielded a close value, i.e. from 0.8 - 1.9%.

Levels of contamination by plutonium isotopes in the aforemenetioned zones are not decisive from the point of view of decontamination efforts and making economic decisions.

Information on radioactive contamination of rivers and reservoirs was obtained by regular analysis of water samples from the Pripyat', Irpan', and Teterev rivers and the Dneprovsk water supply. Beginning from 26 April 1986 water samples were taken over the entire water area of the Kiev reservoir.

The highest concentrations of iodine-131 were found in the Kiev reservoir on 3 May 1986, i.e. 3 x 10-8 curies per liter. It must be noted that the spatial distribution of radionuclides in the water was characterized by great nonuniformity.

Monitoring of radionuclide content in bottom sediments of reservoirs both inside and outside the 30-km zone was set up from the first few days of the accident. The radionuclide concentration in bottom sediments in isolated sections of the Kiev reservoir adjacent to the accident region during the second 10 days of June was 10-7 - 10-8 curies/kg, in the water 10-10 curies/l.

Irradiation of marine organisms in the Kiev reservoir did not seriously affect the population level. Significant radiation influence on the marine eco system can occur only in the coolant pond of the Chernobyl nuclear power plant.

Water ecosystems which inhabit the cooling pond of the Chernobyl nuclear power plant were exposed to the greatest radiation burdens. For some types of water plants, dose rate of internal irradiation was 10 rad/hr, and near the bottom of the cooling pond the level of external irradiation was 4 rad/hr (at the end of May 1986).

According to estimates of specialists levels of irradiation up to 10-2 rad per day do not noticeably affect ground ~~eco-systems~~ ecosystems. Within the 30-km zone around the Chernobyl nuclear power plant higher irradiation levels were observed in isolated sections contaminated by radioactive fallout. This can lead to a noticeable change in the state of radiosensitive types of plants in these areas.

Irradiation levels outside the 30-km zone the kilometer zone around the Chernobyl nuclear power plant cannot dramatically affect species composition of plant and animal communities.

These results are of a preliminary nature. The study of the consequences of the Chernobyl accident on living organisms and ecosystems continues.

7.4. Irradiation doses to the population in the 30-km zone around the nuclear power plant.

Analysis of radioactive contamination of the environment in this zone made it possible to estimate real and predictable irradiation doses to the population of cities, towns, villages and other population centers.

Based on these estimates decisions were made to evacuate the population of Pripyat' and a number of other population centers. A total of 135,000 individuals were evacuated.

These and other measures made it possible to prevent irradiation of the population above the established limits.

Radiological consequences for the population in the next few decades were estimated. These consequences will be insignificant against a background of natural malignant and genetic diseases.

7.5. Data on irradiation of power plant and emergency service personnel. Treatment.

As a result of participation in accident control measures during the first few hours after the accident some individuals from among plant personnel received high doses (greater than 100 rem ~~ber~~) and also burns from fighting fires. First aid was rendered to all those affected. By 0600 hours on 26 April 1986 108 individuals had been hospitalized and during that day another 24 from among those examined. One patient died at 0600 hours on 26 April 1986 from severe burns and one individual from among those working on the damaged unit was not found. His work site may have been in the zone of debris and high activity.

Based on criteria of early diagnosis adopted in the USSR, by the start of the first 36 hours individuals were selected for immediate hospitalization for whom development of acute radiation sickness (OLB) was predicted with greatest probability. Clinical facilities in Kiev near the accident site and a specialized hospital in Moscow were selected for hospitalization in order to provide a maximum amount of assistance and competent analysis of observation results.

During the first two days 129 patients were sent to Moscow. From among them, during the first three days 84 were diagnosed as having OLB of II - IV degree of severity and 27 as having OLB of degree I. In Kiev, 17 individuals were diagnosed as having OLB of degree II - IV, and 55 with OLB of degree I.

Detailed information on methods and results of treating these patients is given in the appendix.

The total number of those who died from burns and OLB among personnel at the beginning of July was 28. Among the population there was no one who had received high doses leading to OLB.

8. Recommendations for increasing the safety of nuclear power engineering.

8.1. Scientific and technical aspects.

A consultation committee for coordination of scientific research in the field of nuclear plant safety approved in 1985 a "list of priority efforts" which is the foundation for planning of experimental and theoretical research on the safety of nuclear power plant engineering in the USSR aimed at more detailed validation of safety requirements, estimation of the actual safety of nuclear facilities and bringing this level for nuclear power plants started before 1975 into agreement with established requirements.

After the accident at the Chernobyl plant a revision and evaluation of the state of experimental and theoretical research on ensuring nuclear power plant safety were done and measures outlined and expand, improve and intensify it.

Theoretical programs for analysis of nuclear power plant safe behavior in all possible transition and accident modes, including those for which it is not design are being improved and the modelling systems and complexes developed.

The search continues to expand on the possibility of building reactors with passive safety systems, so-called reactors with "internally inherent" safety, with cores which cannot fail during any accidents.

Research will be intensified on quantitative-probabilistic analysis of safety, analysis of risk from nuclear power, development of conceptual and methodological principles of optimizing radiation safety and comparing the radiation hazard with other types of hazards from industrial activity.

8.2. Organizational-technical measures

The system of supervision and standard documents which exist in the USSR encompasses all main questions of ensuring nuclear power plant safety and continues to be improved. Under the aegis of Gosatomenergonadzor, a consolidated list and plan for development of rules and regulations in the field of nuclear power which coordinates and directs the activity of all the departments in development and systemization of a corresponding scientific and technical documentation was compiled in 1985 in the USSR.

Comparison of existing Soviet documents on questions of design and operation of nuclear power plants with foreign analogs does not reveal any fundamental differences. Existing standard requirements associated with safety for the most part do not require re-examination. However their practical implementation requires more careful monitoring. Quality of training and re-training of personnel must be raised, monitoring of the quality of equipment, installation, and startup efforts by builders and designers and their responsibility for subsequent efficiency and safety of nuclear power plants in operation must be intensified.

After the accident at the Chernobyl nuclear power plant organizational measures to increase power plant safety were implemented. They can be divided into two stages.

The first stage which was carried out through detailed scientific and technical analysis of the course of the accident from results of initial information from the site relates to working nuclear power plants with RBMK type reactors and includes operational measures in working nuclear power plants with RBMK developed primarily to prevent the conditions which immediately preceded the accident.

The second stage, i.e. measures developed from the results of scientific and technical analysis of the course of the accident, included measures to increase safety of all types in nuclear power plants.

These measures will ensure safe operation of nuclear power plants with RBMK type reactors.

For nuclear power plants with other types of reactors previous measures to increase safety associated mainly with new advances in science and technology, operating experience, capabilities for diagnosis of the condition of metals in piping and equipment, and devices for automatic control of industrial processes are scheduled for implementation.

To increase the level of management and responsibility for the development of nuclear power and improve operation of nuclear power plants an All-Union Ministry of Nuclear Power Engineering was formed.

A host of measures to intensify government supervision of safety in nuclear power has been outlined.

8.3. International measures

The Soviet Union, which contributes its share to international efforts in nuclear power safety and which is guided by the desire to further strengthen international safety, in light of the Chernobyl accident, came forward with initiatives for establishing an international program for safe development of nuclear power and expansion of international cooperation in this area. These suggestions were set forth by the General Secretary of the CC CPSU, M.S. Gorbachev, on 14 May and 9 June 1986.

The international system for safe development of nuclear power is a system of international legal documents, international organizations and structures, and also organizational measures and activities to protect the health of the population and the environment within the framework of peaceful use of nuclear power. Establishment of this system could be supported by international agreements, participation in corresponding international conventions, additional accords, implementation of joint coordinated scientific programs on problems of nuclear safety, exchange of scientific and technical information, and establishment of international data banks and equipment necessary for safety purposes and so forth.

With the direct participation of international organizations funds could be created for rendering immediate assistance, including immediate support with the necessary special medical preparations, dosimetric and diagnostic equipment and instruments, supply of foodstuffs, fodder, and other material aid. A system of operational warning and supply of information in the case of a nuclear power plant accident, in particular one with transnational consequences, must be set up.

Treatment of the problem of material and psychological damage in accident cases also merits attention.

There is another aspect of nuclear safety, the prevention of nuclear terrorism. The extreme importance of the problem derives from this, i.e. development of a reliable system of mesures to prevent nuclear terorism in any of its manifestations.

A major role in establishing the international system for safe development of nuclear power will be played by the MAGATE.

At present it can be noted with satisfaction that initial steps have already been taken to implement suggestions relating to establishment of the international system for safe development of nuclear power. Efforts have begun on preparation for concluding two international conventions "Operation warning of a nuclear accident" and "Assistance in the case of nuclear accidents and radiological emergency situations". Questions of expanding international cooperation, in particular research programs of the MAGATE on nuclear safety are being actively discussed.

Initiatives on establishing an international system for safe development of nuclear power are closely associated with problems of detente and nuclear disarmament. The accident at the Chernobyl nuclear power plant has again demonstrated the danger of uncontrolled nuclear power and highlighted the destructive consequences to which its military use or damage to peaceful nuclear facilities during military operations could lead. In addressing and solving problems of safe use of nuclear power it would be absurd to develop means and methods of its most dangerous and inhuman use at the same time.

9. Development of nuclear power engineering in the USSR

Due to continued development of nuclear power engineering a reduction in the increase of consumption of organic fuel by thermal power plants in the European part of the country is outlined by the energy program of the USSR. The amount of fuel oil in electric power generation should be cut in half. The nuclear power will cover most of the increased consumption of electricity by the national economy. Maximum possible use of nuclear fuel for centralized heating and industrial heat supply and establishment of nuclear-industrial complexes are planned.

The Soviet Union is a pioneer in the use of nuclear power for peaceful purposes. The first nuclear power plant in the world with a pressure tube uranium-graphite reactor has been operating for 32 years. The program for building so-called demonstration power reactors for nuclear power plants with relatively small electrical capacities which implemented at the time made it possible to select the most promising of these for further development and improvement.

The existence of three types and modifications of nuclear reactors which have been adopted in the USSR for building up nuclear power capacities allows great flexibility and reliability of energy supply, and much more complete utilization of nuclear fuel resources; it also matches the characteristics of development of the power machinery construction base to a satisfactory degree.

Nuclear power plants under construction in the USSR use reactors of types VVER, RBMK, and BN. The first two are thermoneutron reactors with cooling water. BN are fast neutrton breeder reactors with sodium coolant currently being built for industrial trials of designs which have been adopted and gradual development of a closed fuel cycle with plutonium fuel on this foundation in the future.

The basis of nuclear power engineering in the USSR is nuclear power plants with VVER and RBMK reactors. Installed capacities in the Soviet Union have reached almost 30 million kilowatts. Soviet nuclear power plants are distinguished by high operational readiness. Utilization factor of installed power in a nuclear power plant has been rather high over the last few years.

According to the "Main trends in economic and social development of the USSR for 1986 - 1990 and through the year 2000" continued development of nuclear power engineeering in the European part of the USSR and in the

Urals is planned. In 1985 nuclear power plants generated approximately 170 billion kilowatt hours of electricity and by the year 2000 this will increase by a factor of 5 - 7.

This development will allow nuclear power plants to occupy first place in terms of new capacities in power systems of the European part, having eliminated the construction of new thermal power plants using organic fuel to cover increases in the base part of the load curve.

Development of nuclear sources of heat supply based on high temperature gas cooled reactors is underway in the USSR. Construction of safe plants with these reactors will make it possible to generate high temperature heat for a number of industrial technologies.

The Soviet Union is actively involved in international cooperation in the field of nuclear power engineering and collaborates in agencies and committees of the United Nations, the MAGATE, the MIREK, and others. Nuclear power engineering in the USSR is developing in close cooperation with COMECON countries.

LIST OF ABBREVIATIONS

AZ	- emergency protection
AZ-5	- signal to insert all regulating rods and emergency protection rods into the active zone
AN	- Academy of Sciences
AR	- automatic regulator
AS	- nuclear station
AES	- nuclear power station
BN	- fast breeder reactor
BRU-K	- high-speed reduction device
BS	- water-steam separating drum
BSSR	- Belorussian Soviet Socialist Republic
GKAE	- USSR State Committee on the Use of Atomic Energy
Gosagroprom	- USSR State Agro-industry Committee
Gosatomenergonadzor	- USSR State Committee on Safe Performance of Operations in Atomic Power Engineering
Goskomgidromet	- USSR State Committee on Hydrometeorology and Monitoring of the Environment
GPK	- main safety valves
GTsN	- main circulation pump
DP	- additional absorber
DREG	- diagnostic recording of parameters (program)
KGO	- monitoring of seal of shells
KMPTs	- repeated forced circulation duct
KTsTK	- monitoring of the integrity of channel pipes
LAZ	- local emergency protection
LAR	- local automatic regulation
MAGATE	- International Atomic Energy Agency

Minzdrav SSSR - USSR Ministry of Public Health

MIREK - World Energy Congress

NVK - bottom water lines

OLB - acute radiation sickness

OON - United Nations

PVK - steam-water lines

PD - fission products

PEN - electric feeder pump

RBMK - high-power channel reactor

RGK - distribution group collector

RZM - unloading-loading machine

RDES - back-up diesel power plant

RP - reactor space

RR - manual regulator

RSFSR - Russian Soviet Federated Socialist Republic

RU - reactor installation

SAOR - system for emergency cooling of the reactor

SLA - accident localization system

SM SSSR - USSR Council of Ministers

SRK - stopper-regulating valves

SUZ - system for control and protection

SEV - Council for Mutual Economic Aid

TVS - heat-releasing assembly

TG - turbogenerator

TK - operating channel

USP - shortened absorber rods

USSR	- Ukrainian Soviet Socialist Republic
ChAES	- Chernobyl' Nuclear Power Station
EVM	- computer

LIST OF BASIC EQUIPMENT OF THE MAIN HOUSING OF THE AES

Item No.	Equipment or Product	Measurement Units	Unit Mass in Tons	Number per Power Unit
1	graphite lining	set	1850	1
2	"S" system metal components	"	126	1
3	"OR" system metal components	"	280	1
4	"Ye" system metal components	"	450	1
5	"KZh" system metal components	"	79	1
6	"L" system metal components	"	592	1
7	"D" system metal components	"	236	1
8	Water-steam separating drum	item	278	4
9	TsVN-6 Main Circulation Pump	"	67	8
10	GTsN electric motor	"	33	8
11	DU-800 main cut-off gate valve	"	5.7	8
12	intake collector	"	41	2
13	delivery collector	"	46.0	2
14	distribution group collector	"	1.3	44
15	bottom water lines (NVK)	set	400	1
16	steam-water lines (PVK)	"	450	1
17	DU-300 down pipelines	"	16	1
17a	DU-800 pipelines of MPTs duct	"	350	1
18	unloading-loading machine (RZM)	"	450	1
19	central room traveling crane Q 50/10 tf	item	121	1
20	GTsN room traveling crane Q 50/10 tf	"	176	1
21	forced-ventilation fan	"	3.5	30
22	exhaust fan	"	3.5	50

23	organized leak water tank	item	1.4	2
24	organized leak water heat exchanger	"	0.2	2
25	scheduled preventive maintenance vessels	"	25	4
26	metal components and pipelines of accident containment zone	set	270	1
27	NVK compartment check valves	"	2.5	11
28	accident containment system overflow valve	item	2	8
29	accident containment system condensers	"	3.7	36
30	container car	"	146	1
31	crane in UPAK (gas activity reduction system) compartment Q 30/5 tf	"	45	1
	pipelines of carbon steel	set	1170	1
	pipelines of stainless steel	set	760	1
	MACHINE ROOM			
32	K-500-65/3000 turbogenerator set	item	3500	2
33	SPP-500 steam superheater separator	"	15	8
34	low-pressure preheater	"	37.5	4
35	first extraction condenser pump units	"	2.5	6
36	machine room traveling crane Q 125tf	"	211	1
	pipelines of carbon steel	set	3825	1
	pipelines of stainless steel	"	1300	1
37	gas stripper	item	4.5	2

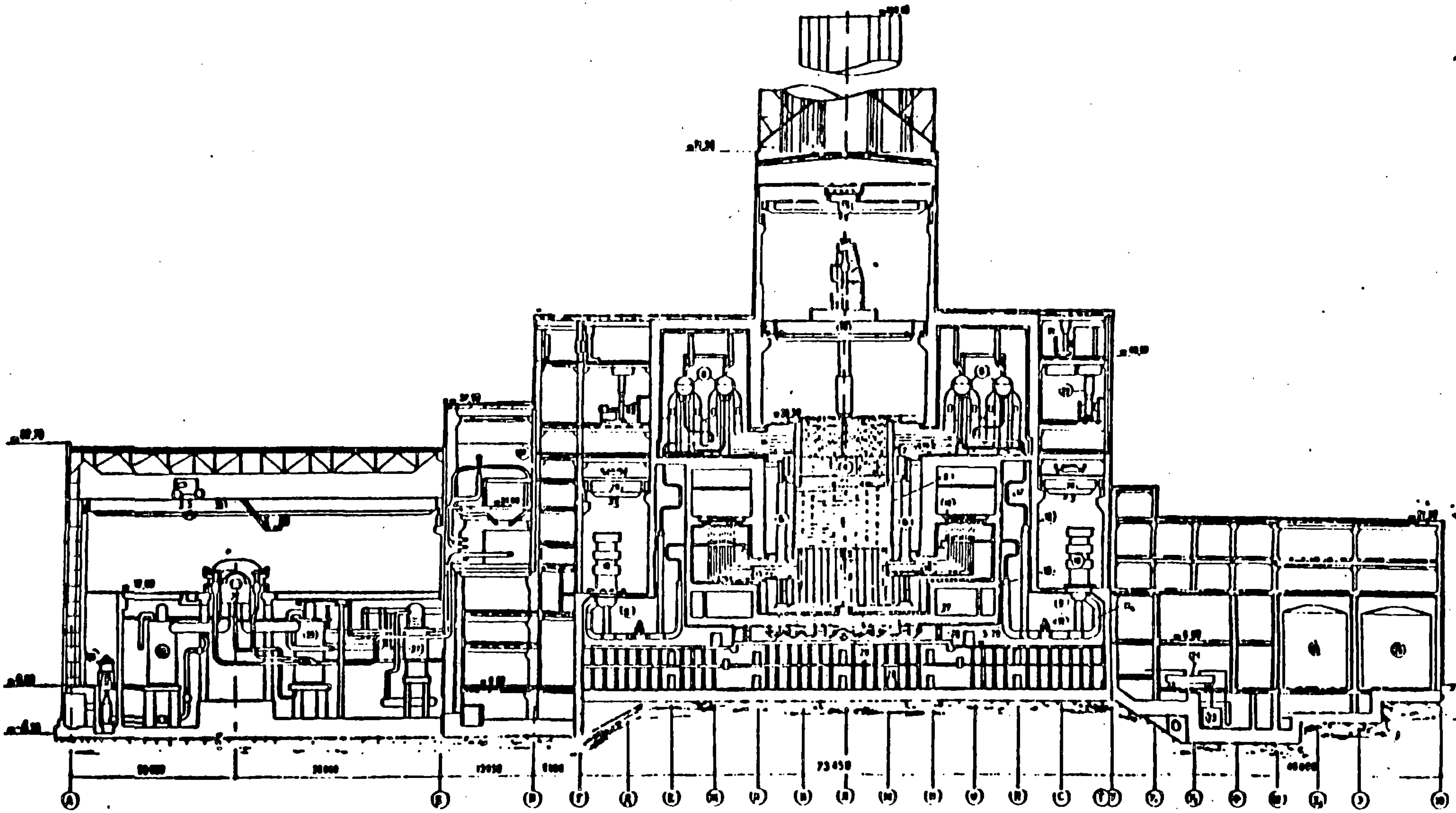

Рис. 1. Разрез по главному корпусу АЭС с РБМК-1000 (с зоной локализации)

Fig. 1. Sections along main housing of AES with RBMK-1000 (with containment zone)

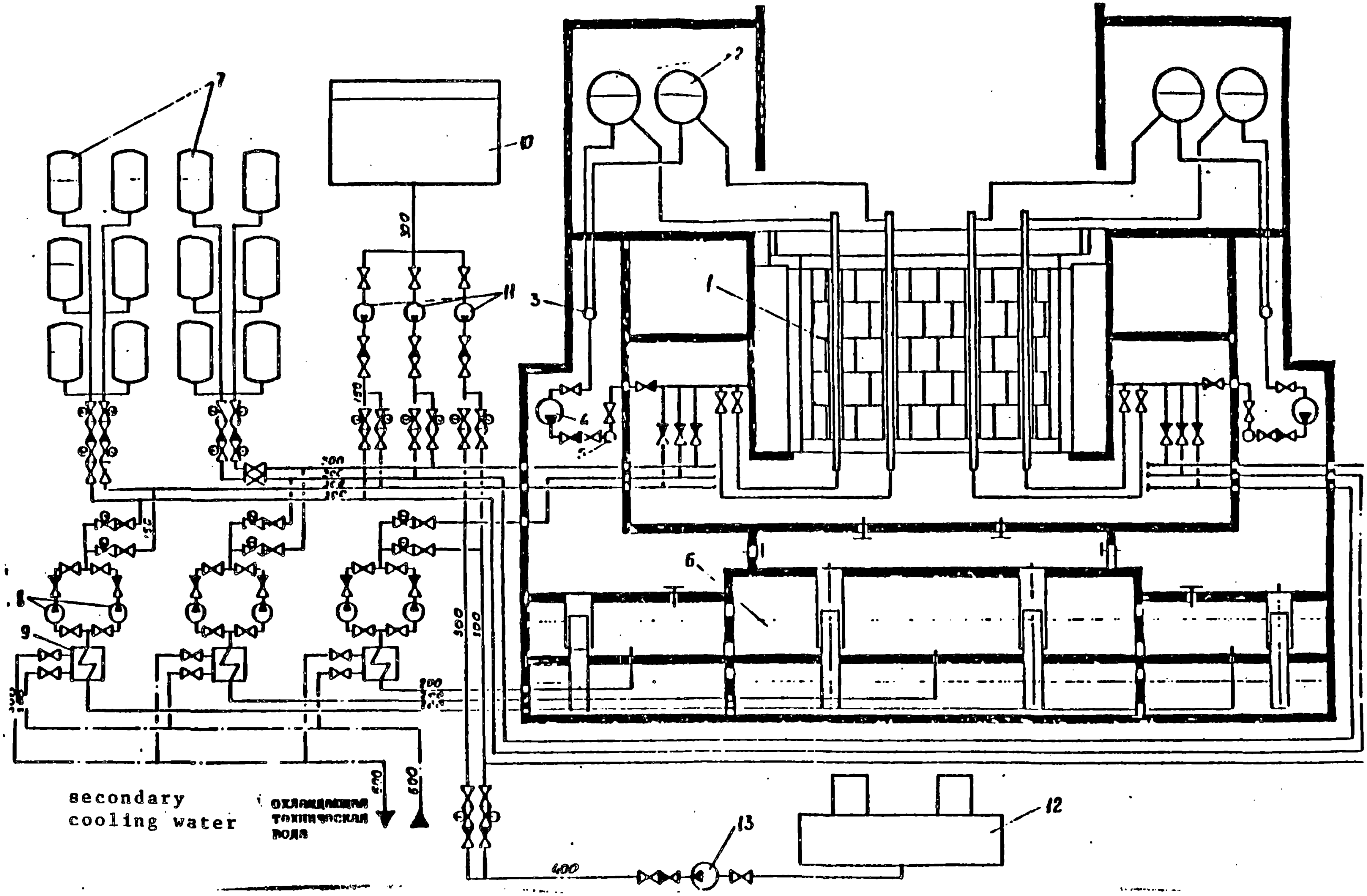

Fig. 2. Schematic diagram of system for emergency cooling of the reactor: 1 - reactor; 2 - steam separators; 3 - suction header; 4 - main circulation pump; 5 - pressure header; 6 - suppres n pool; 7 - SAOR vessels; 8 - SAOR pumps for cooling malfunctioning half reactor; 9 - heat exchanges; 10 - pure coordinate tank; 11 - SAOR

(…th containment zone)

РАЗРЕЗ ПО РЕАКТОРНОМУ ОТДЕЛЕНИЮ АЭС С РБМК-1000
(с зоной локализации)

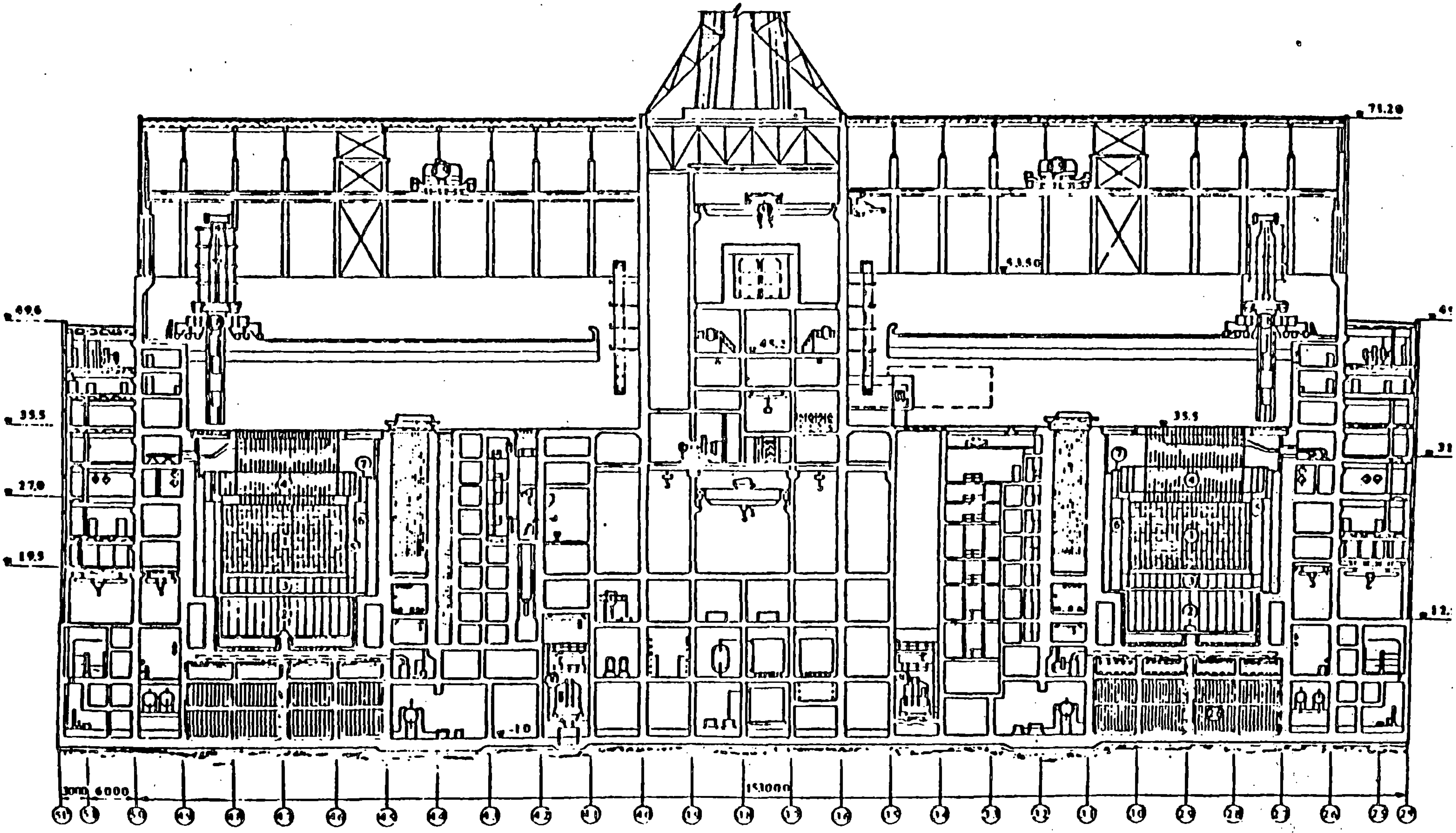

Рис. 3
Fig. 3

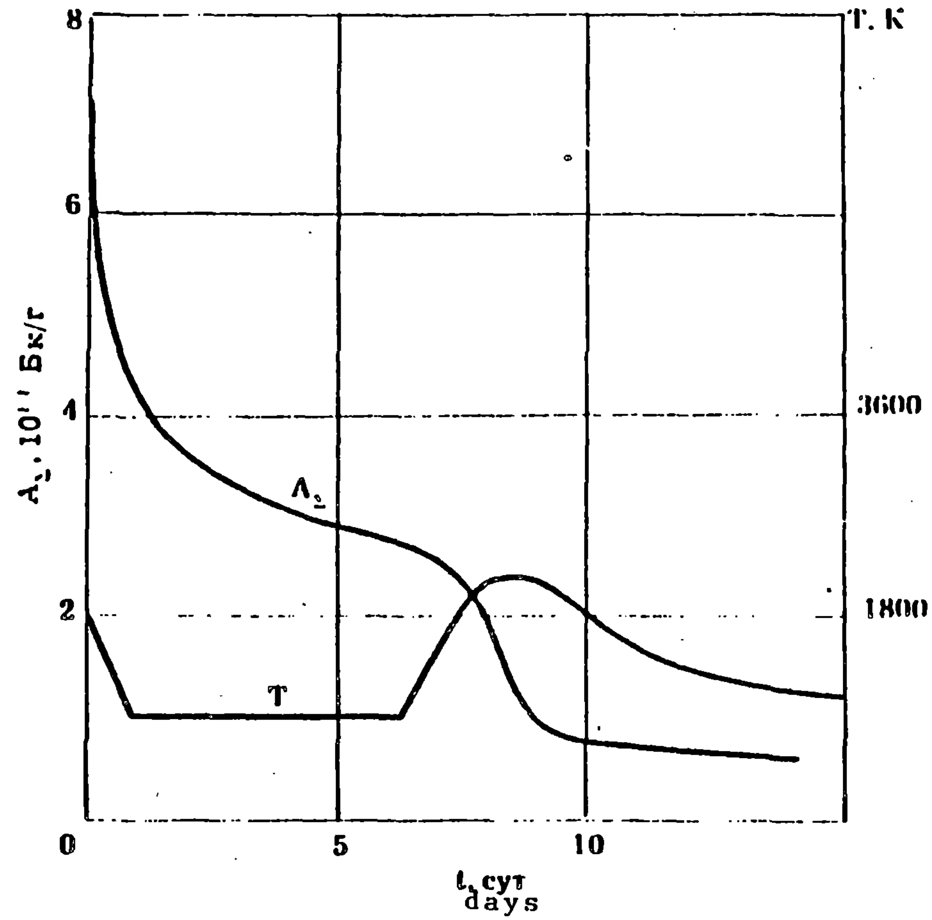

Рис. 5. Изменение активности и температуры топлива во времени

Fig. 5. Variation of fuel activity and temperature in time

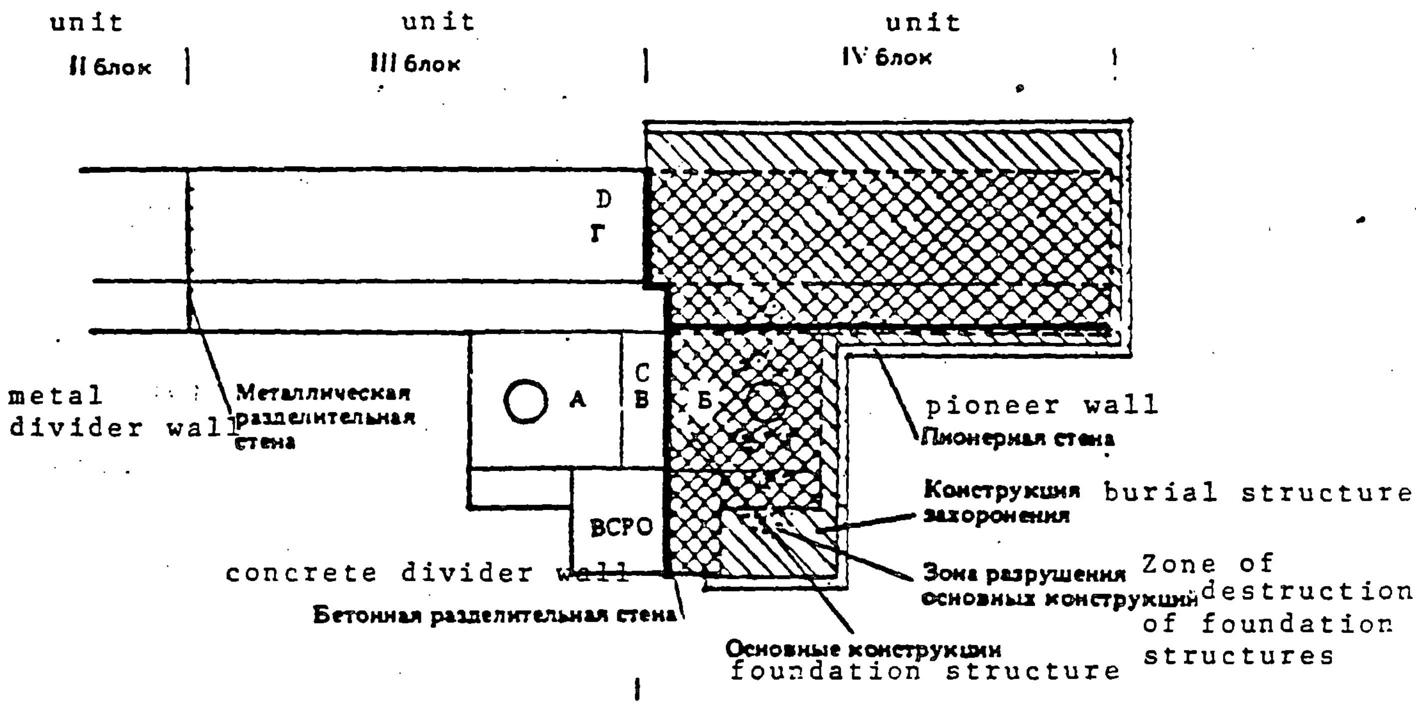

Рис. 6. Схема захоронения энергоблока № 4. Горизонтальный разрез одного из вариантов проекта

Fig. 6 Diagram of burial of power unit No. 4. Horizontal sections of one version of the plan.

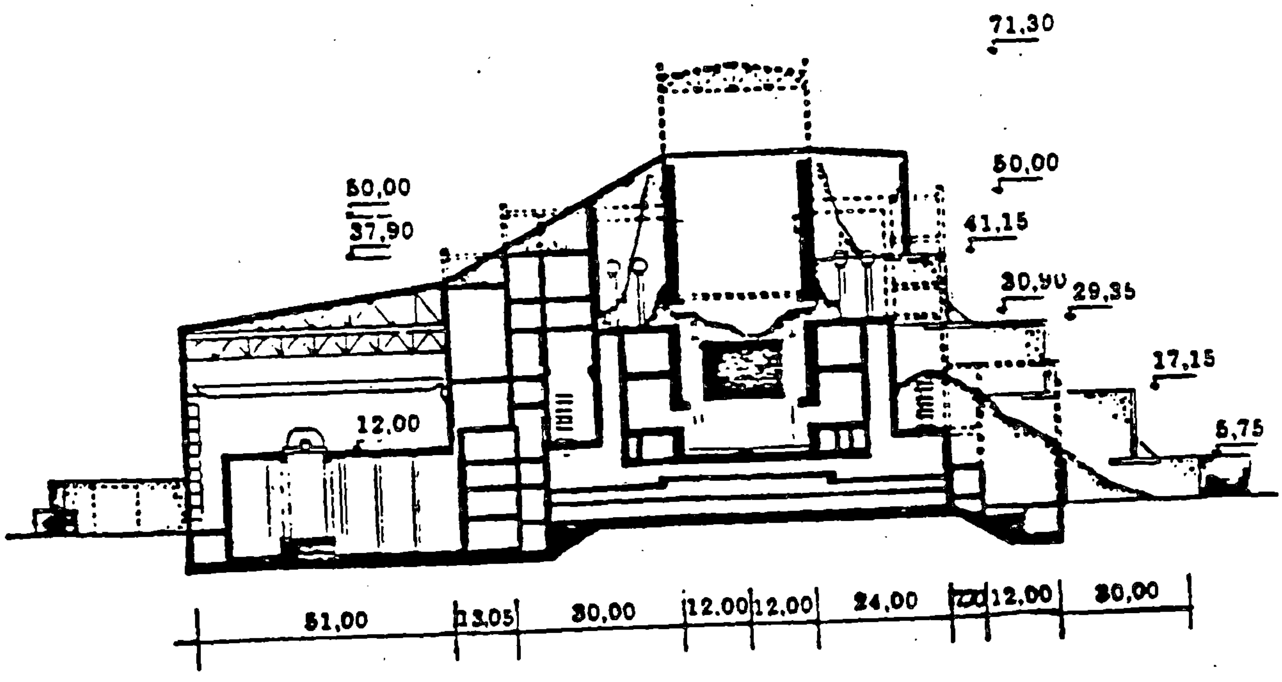

Рис. 7. Схема захоронения энергоблока № 4. Поперечный разрез одного из вариантов проекта

Fig. 7. Diagram of burial of power unit No. 4. Cross section of one version of the plan

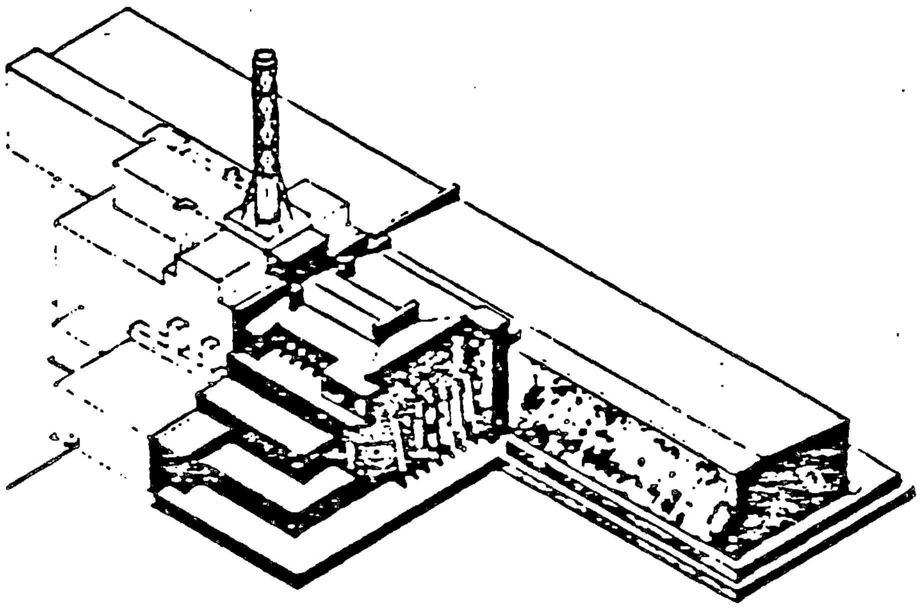

Рис. 8. Схема захоронения энергоблока № 4. Общий вид одного из вариантов проекта

Fig. 8. Diagram of burial of power unit No. 4. Overall view of one version of the plan.

CONTENTS

Appendix 1

PRESSURE-TUBE WATER COOLED GRAPHITE REACTORS AND EXPERIENCE IN OPERATING RBMK REACTORS

1. Pressure-tube water-cooled graphite reactors and experience in operating RBMK reactors.

1.1 Pressure-tube water-cooled graphite reactors use normal water as the heat carrier and graphite as the moderator. The distinguishing features of pressure-tube reactors are: the absence of a pressurized body, the relative simplicity of design, the broad potential for channel-by-channel testing and regulation, the capability of reloading fuel in a working reactor, the flexibility of the fuel cycle, and the almost unlimited potential for increasing power on the basis of standard structural elements.

The first power producing reactor in the USSR was a pressure-tube type, the water-cooled graphite reactor of the First AES with an electric power of 5 mW, which as started in June 1954 in the city of Obninsk near Moscow.

The experience accumulated in building and operating the First AES was used in designing the Beloyarsk AES (1964, 300 MW).

Further development of the water-cooled graphite reactors in the USSR resulted in the creation of a powerful pressure-tube boiling water RBMK reactor with an electric power of 1000 MW which, along with the VVER-1000 reactor, became the basic reactor for high power atomic power engineering in the USSR.

Start up of the RBMK-1000 reactor at the Leningrad AES in 1973 started the series of this type of reactor.

The broad program for building RBMK-1000 reactors unveiled in the 1970s made it possible to put on line 14 operational

Table 1.1

Таблица I.I

Характеристика (1)	РБМК-1000 (2)	РБМК-1500 (3)
Электрическая мощность (МВт) (4)	1000	1500
Тепловая мощность (МВт) (5)	3200	4800
Паропроизводительность (т/ч) (6)	5800	8800
Параметры пара перед турбинами: (7)		
давление (кгс/см^2)	65	65
температура ($^{\circ}$C)	280	280

KEY: (1) Characteristics

(2) RBMK-1000

(3) RBMK-1500

(4) Electrical power (MW)

(5) Thermal power (MW)

(6) Steam productivity (T/hr)

(7) Steam parameters in front of the turbines:

pressure (kgf/cm)

temperature (C)

reactors from 1973 through 1985 (four each at the Leningrad (LAES), Kursk (KAES), and Chernobyl AES (ChAES) and two reactors at the Smolensk AES (SAES) with a total established elecric power of 14 GW. Improvements aimed at increasing reliability and safety were introduced into each new reactor generator.

The acquired knowledge and experience in operating energy units with RBMK-1000 reactors revealed their hidden reserves and made it possible, based on this reactor, to begin designing an even more powerful reactor-the RBMK-1500 reactor with an electric power of 1500 MW, which in 1983 was placed on line at the Ignalinsk AES and by early 1985 had exceeded its designed power.

Pressure-tube reactors have advantages which facilitate solution of problems to ensure their safety.

These advantages include:

- ease of organizing individual testing of the condition of fuel elements, fuel assemblies and the integrity of the pressure-tubes;

- the capability of real time replacement of fuel assemblies which have lost their hermetic seal without shutting down the reactor;

- reduction of the danger of the consequences of explosions of the tubes of the first loop due to an increase in the number of circulation loops and a corresponding reduction in the diameter of the pipelines; and

- a design potential for increasing the unit power of the reactor without complicating the emergency cooling system for the active zone.

On the other hand, certain specific features inherent to pressure-tube graphite reactors cooled by boiling water require fundamentally new solutions in developing safety support systems. The features primarily include:

- the large steam volume in the cooling loop, which greatly moderates the rates of pressure drop in the heat carrier with accidental rupture of the pipelines;

- the possibility of manifestation of a positive steam effect of reactivity, which greatly determines the behavior of the neutron power of the reactor with accidents associated with disruptions in the circulation of the heat carrier through the active zone; and

- the large volume of thermal energy accumulated in the metal structures and the graphite lining of the reactor, which determines the drop in the thermal power after operation of emergency shielding.

The basic technical characteristics of AES with RMBP type reactors are cited in Table 1.1

1.2. Presently, the total operating time of RBMK reactors as part of AES is approaching 100 reactor-years.

Based on an analysis and correlation of the experience in operation,a constant effort is underway to modernize individual subassemblies of the reactor and to improve reactor setting systems and their modes of operation. A whole complex of measures aimed at increasing the reliability and safety of operation of the AES has been developed and introduced as a result. The most important measures are the following:

- modernization of the design of the shut-off and regulating valves, the ball sensors of the flow rate meters, and the cut off plugs in the fuel channels;

- optimization of the layout of the pipelines for steam and water communications and the discharge steam pipes for the drum-separators;

- improvement in the intrabody devices in the drum-separators;

- improvement in the main circulation pumps and their back-up systems;

- introduction of predicting programs for real-time calculations and programs for emergency recording of the condition of the equipment and for diagnosing the condition of technological systems;

- development and introduction of a system for local automatic regulation (LAR) and local emergency shielding (LAZ) which operate from intrazone sensors;

- justification and test operation in one of the units of heat releasing cartridges with an initial enrichment of 2.4%; and

- development of a system for discharge of residual heat releases, which makes it possible to perform long-term continuous repairs of equipment and subassemblies in the reactor.

These and other measures realized in energy units have ensured reliable and safe operation of AES with RBMK reactors, certain results of which are illustrated in Table 1.2, where the production of electric power at AES for 1981-1985, inclusively, and the established power utilization factor (KIUM) for 1985 are presented.

Table 1.2

Operational indicators of AES with RBMK-1000

Показатели[1]	ЛАЭС[2]	КАЭС[3]	ЧАЭС[4]	САЭС[5]
Установленная мощность[6] на 01.01.86 МВт	4000	4000	4000	2000
Выработка электроэнергии[7] за 1981-1985гг., млрд. кВт.ч	140,0	82,4	106,6	23,4
КИУМ за 1985г., %[8]	84	79	83	76

KEY: (1) Indicators

(2) LAES

(3) KAES

(4) ChAES

(5) SAES

(6) Installed power on 1 January 1986, MW

(7) Electricity production for 1981-1985, billions of kW hours

(8) KIUM for 1985, %

The maximal values of KIUM for 1985 were reached at the fourth unit of the LAES of 91%, at the second unit of the ChAES of 90%, and at the first unit of the LAES of 87%.

Correlation of the operational experience and the performed scientific research and development works revealed certain ways to increase the effectiveness of power producing units with RBMK reactors, including: increasing the power of the operating energy units, improving and optimizing the modes of operation of the unit, introduction of automatic shielding of the reactor and technological equipment, improving the conditions for performing repairs on the reactor, and increasing the repair suitability of its individual subassemblies.

As everyone knows, the extreme parameters which limit the power of the RBMK reactors are: fuel temperature, the temperature of the graphite lining and the metal structures, and the reserve to heat exchange crisis in the fuel channels. These parameters in the existing reactors are below the maximum permissible. For instance, with a nominal fuel power of the reactor, the maximal power of the fuel channel is approximately 2600 kW with a permissible of 3000 kW, a maximal temperature of the graphite lining of 923°K (650°C) with a permissible value of 1023°K (750 C), a maximal temperature of the metal structures of 573°K (300° C) with an permissible of 623K (350° C), and a reserve to heat exchange crisis of no less than 1.35. The basic equipment in the machine hall of the energy units with an RBMK-1000 reactor (the turbine generators, block transformers, deaerators, and condensate and replenishment pumps) also have a power reserve of approximately 10%.

The identified reserves made it possible to justify the possibility of the energy units operating at a high power level and the full-scale tests conducted of a number of energy units at a power up to 107% of the nominal power confirmed such a possibility.

Appendix 2

2. Reactor Plant Design

The reactor is intended to generate dry saturated steam at 70 kgf/cm-2 (about 7 MPa). The unit consists of the reactor itself with monitoring, control, and protection system, piping, and equipment for the controlled circulation loop (KMPTs).

2.1 Reactor

The RBMK power reactor is a homogeneous thermal neutron channel reactor in which the moderator is graphite, while the heat transfer agent is light water and a steam-water mixture circulating through vertical channels crossing the core.

The reactor's core (1) is shaped like a vertical cylinder 11.8 min equivalent diameter 11.8 m and 7 m high (cf. Figure 1). It is surrounded on the sides and ends by graphite reflectors 1 and 0.5 m thick respectively. The core consists of process channels (TK) with fuel assemblies (TVS), graphite moderator, channels with neutron-absorbing control rods, and monitoring system sensors. Parts of the channels located in the core are made of zirconium alloy. The graphite block structure consist of blocks assembled into columns with axial, cylindrical openings in which the process control channels are set. Process channels are placed in 1661* square-lattice cells at 250-mm intervals. Control/safety system channels (211 pc) are arranged in the same way as process channels in the central openings of the block structure (Channel layout is diagrammed in Figure 2.1a)

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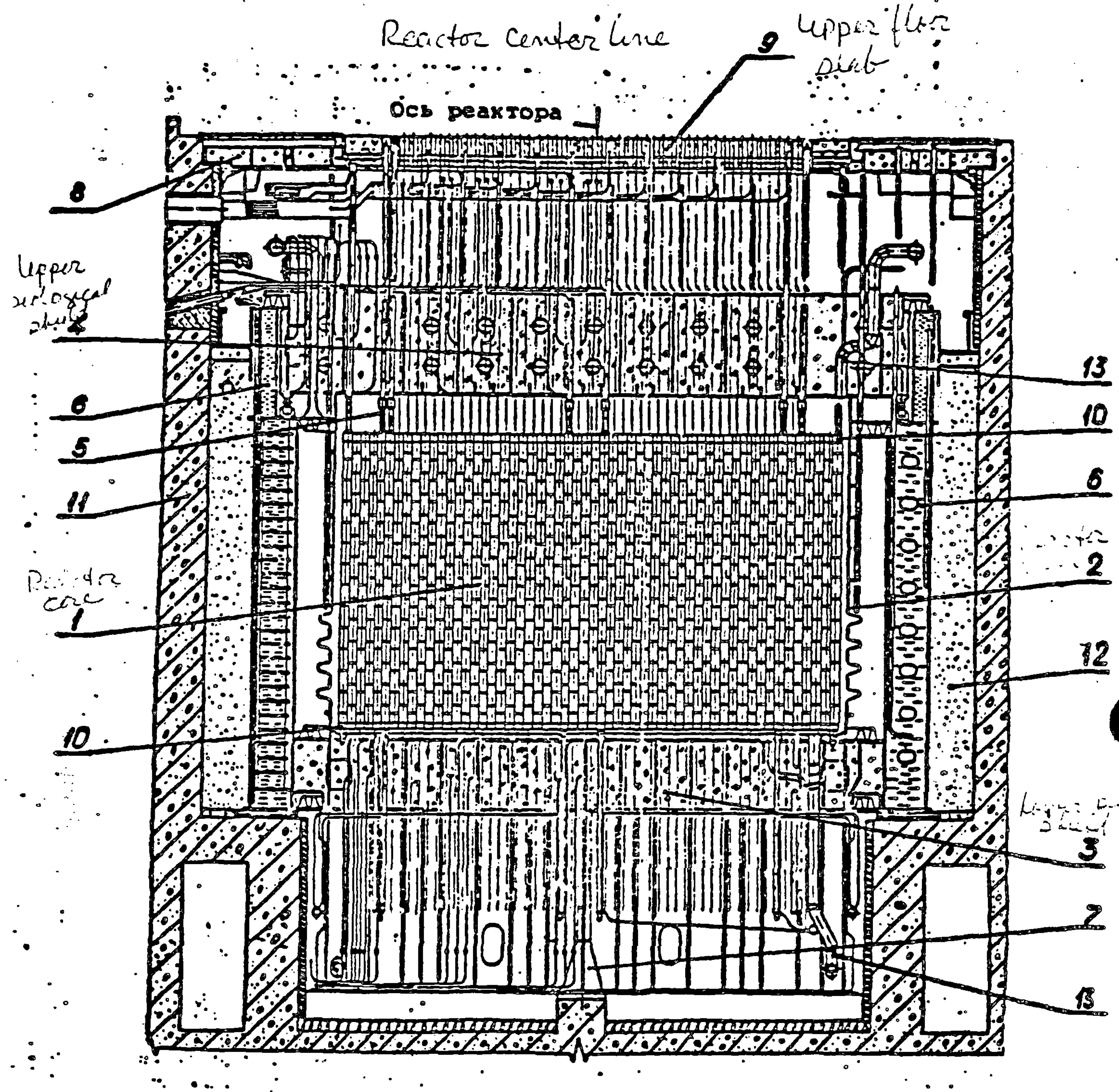
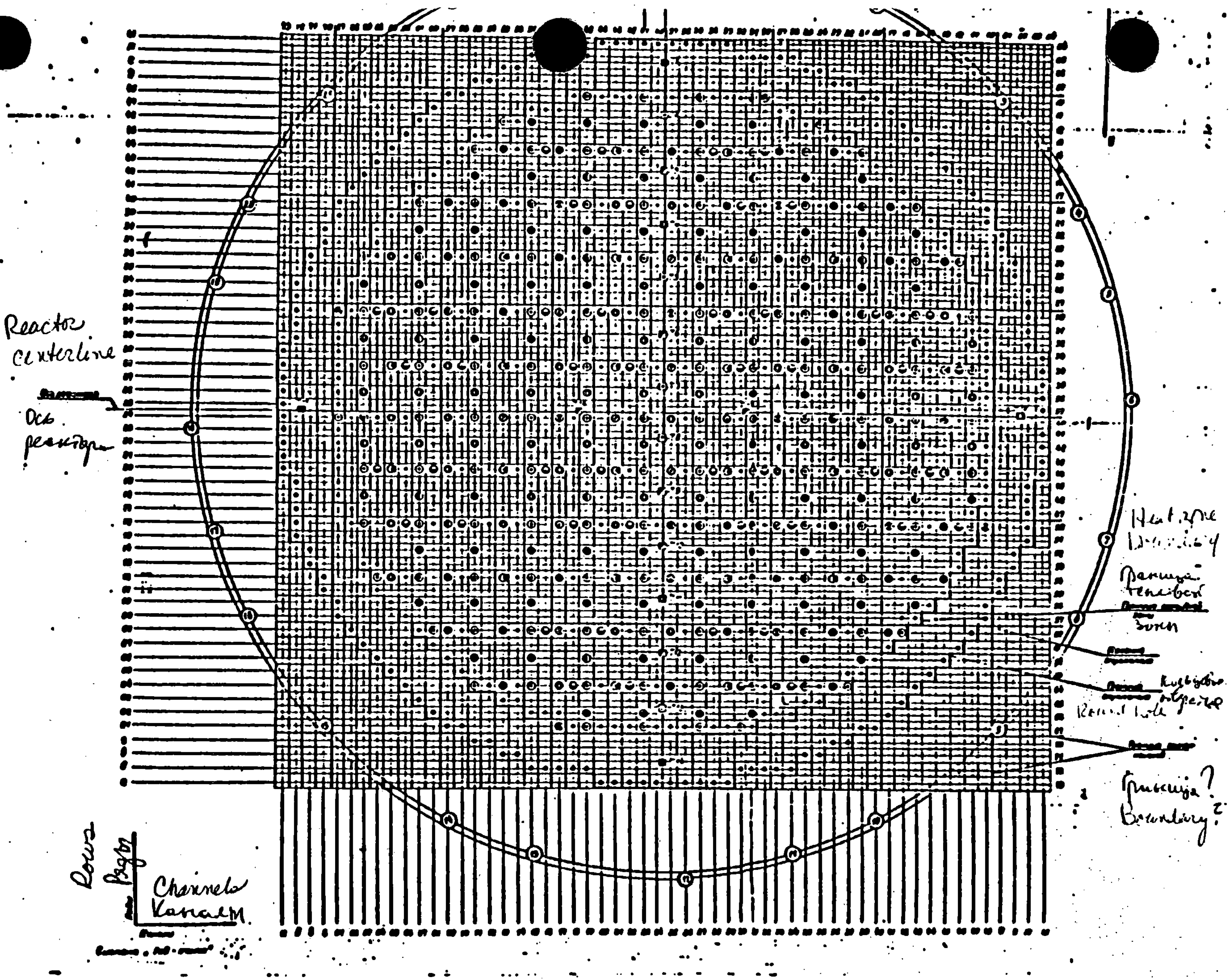
* First-generation LAES, KAES, and ChAES reactors have 1, 693 TVS's and 179 SUZ channels.



Reactor center line
Upper flow slab
Ось реактора
9
8
Upper biological shield
4
6
5
11
Reactor core
1
10
13
10
6
2
12
3
7
13

Fig ~~Рис.~~ 2.1



Reactor centerline
Ось реактора
Heat zone boundary
Граница тепловой зоны
Кубовое отверстие
Round hole
Граница?
Boundary?
Rows
Ряды
Channels
Каналы

~~Рис.~~ 2.Ia
Fig

The graphite block structure is in a hermetically sealed cavity (reactor space RP), formed by cylindrical enclosure (2) and the metal structure's upper (4) and lower (3) slabs. To prevent graphite oxidation and improve heat transfer from the graphite to the process channels, the reactor space is filled with a mixture of helium and nitrogen with a volumetric composition of 85-90% He and 15-10% N2. To prevent a possible helium leak from the RP, the inner cavities of the metal structure and the space surrounding the enclosure are filled with nitrogen at a pressure higher than that in the RP at 50-100 mm H2O (about 0.5-1.0 kPa).

The process control channels are set in routes welded onto the metal structure (5). The top and bottom metal structures and the circular water-filled vessel surrounding shell of annular tank (6), act as biological shield for the areas surrounding the reactor. The heat transfer agent, water, is carried downward by individual pipes to each individual channel. Ascending and washing the fuel elements, water is heated and partially evaporated, and the steam-water mixture is also removed from the top of the channels by individual pipes.

The nuclear fuel is recharged without reducing reactor output by fueling machines (RZM).

In steady-state operation, the rate at which fuel is recharged at rated output is 1-2 cartridges per day.

The reactor is fitted with a control/safety system (SUZ) and a process monitoring system which issues information on the status of the active core, the performance of individual assemblies, and required signals to the SUZ and the emergency signalling system.

Main Reactor Characteristics

Heat transfer agent flow rate through reactor	37.6 x 103
Steam pressure in separator, kgs/cm2	70
Pressure in grouped pressure headers, kgs/cm2	82.7
Average steam content at exit from reactor, %	14.5
Heat transfer agent temperature, oC	
at entry	270
at exit	284
Maximum channel power with regard for 10% power bias, kV	3,000
Heat transfer agent flow rate in maximum-power channel tonne/hr	28
Maximum steam content at exit from channel, %	20.1
Minimum reserve to critical power	1.25
Core height, mm	7,000
Core diameter, mm	11,800
Process lattice spacing, mm	250x250
Number of process channels	1,661

2.1.1 Cartridge and Fuel Element Design

The RBMK-1000 reactor's fuel cartridge consists of the following components (cf. Figure 2.2):

- two fuel element assemblies (TVS)	1;
- carrier rod	2;
- shank and end cap	(3), (4);
- nuts	(5);
Cartridge length	10,015 mm.

The fuel element assembly consists of 18 rods, shell, and 18 clamping rings.

The cartridge's fuel element consists of can (6), fuel column (7), spring lock (8), plug (9), and end caps (10).

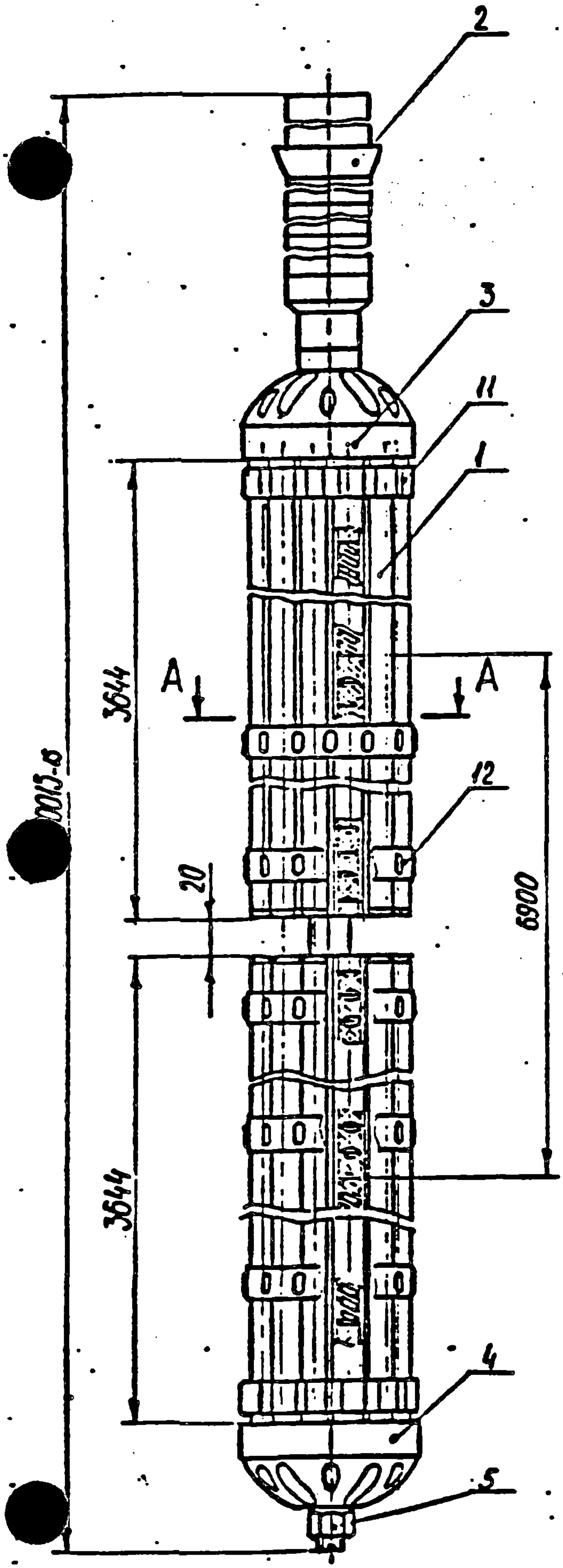
The can and end parts are made of zirconium alloy with 1% niobium (alloy 110). The lock is made of zirconium alloy Ts2M. The fuel element can diameter is 13.6 mm; its minimum thickness, 0.825 mm.

The fuel is pellets of sintered uranium oxide. Pellet diameter -- 11.5 mm; height -- 15 mm. To reduce thermal expansion in the fuel column, pellets on the ends have spherical indentations. The average mass of fuel in an element is 3,600 g; minimum pellet density -- 10.4 g/cm3; diametral clearance between fuel and shell -- 0.18-0.38 mm.

The fuel elements are sealed by contact butt-welding an end cap onto one end of the casing tube and a plug on the other end.

The original medium under the can is helium at -1 kg/cm2 (0.1 MPa). The fuel column is held in the element by a spring lock with a compressive force of about 15 kg.

Figure 2.2



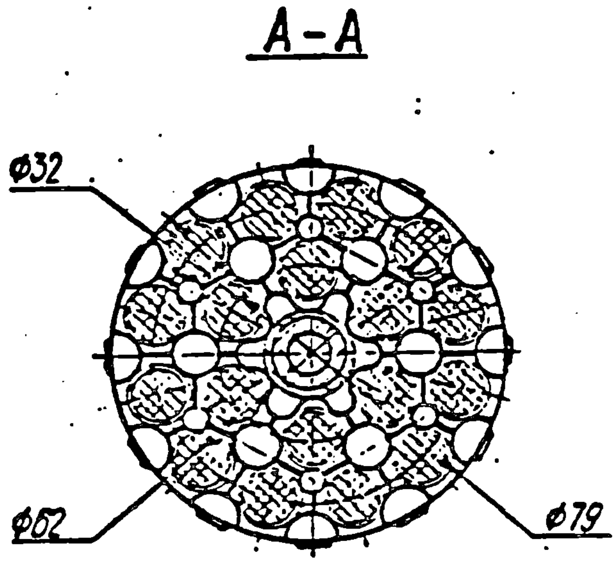
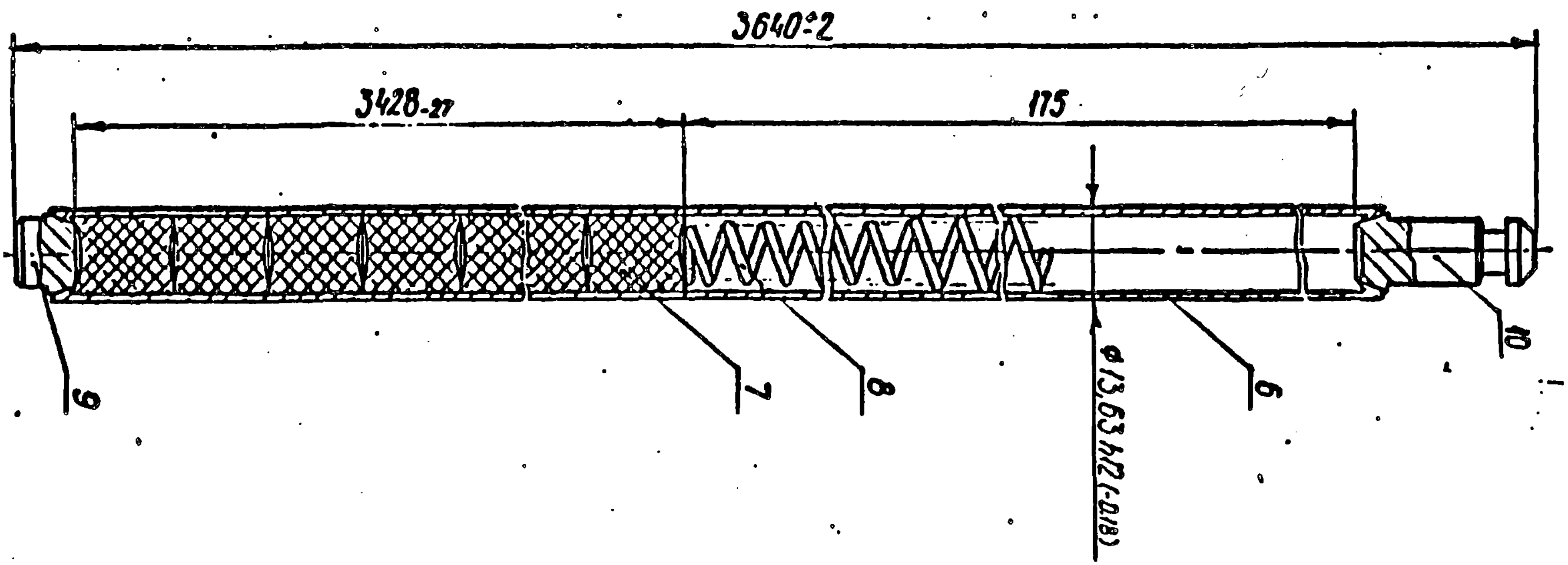


Figure 2.2a



The framework consists of a central tube 15 mm in diameter with walls 1.25 mm thick, one end lattice (11) and 10 spacer lattices (12). The central tube and end lattice are made of zirconium alloy with 2.5% niobium (alloy 125); the spacer lattices, of stainless steel.

The central tube is connected by two expanders to the end lattice to prevent axial play in the connection and so that the lattice can turn relative to the tube. For orientation and to prevent the TVS's from turning relative to one another, the shell tubes have special grooves.

Spacer lattices are attached every 360 mm on the central tube. Each lattice is fastened by inserting the projecting end of the central sleeve into two grooves on the tube so that it can move along the tube if there is slight azimuthal play.

A spacer lattice is assembled from individual shaped cells (12 cells on the outside and 6 on the inside row), a central sleeve and a cover rim. Lattice parts are connected by spot contact welding.

The opening in the lattice for the fuel element is 13.3 mm in diameter. The lattice rim has projections which facilitate loading the cartridge into the channel. The diameter along the rim projections is about 78.8 mm.

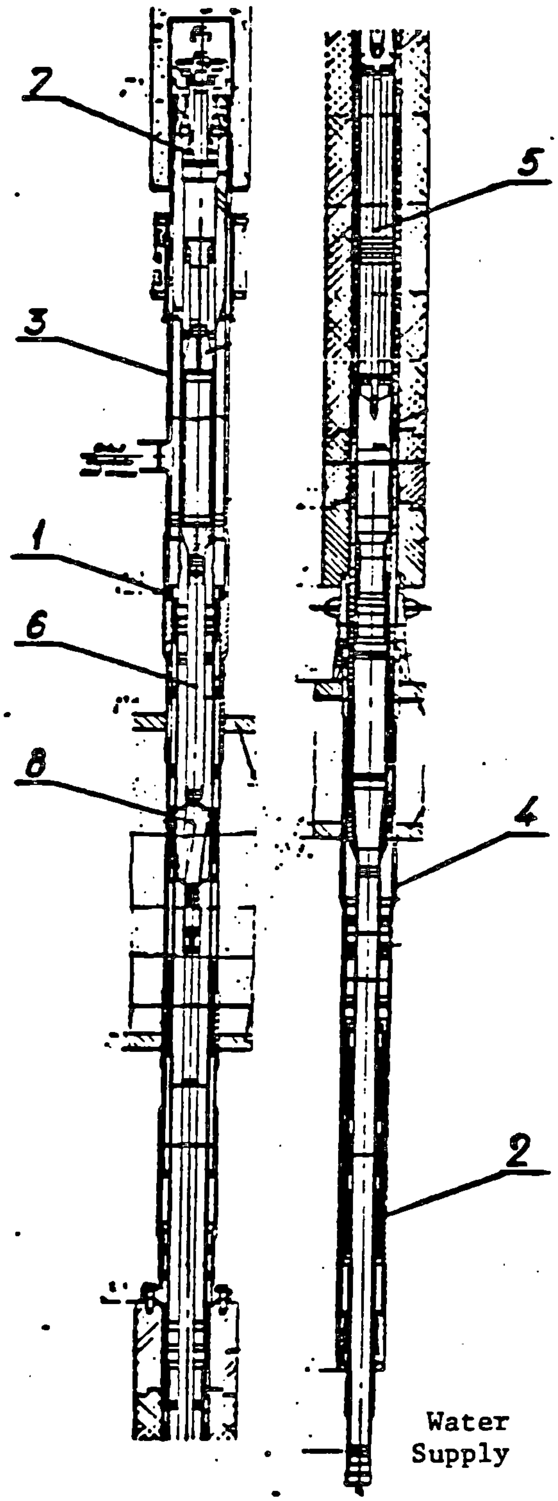
Cells are made of tube with 0.35-mm-thick walls; the central sleeve is a tube with 0.5-mm-thick walls; the rim, a tube with 0.3-mm-thick walls.

Fuel elements are attached to the end lattice by clamping rings made of stainless steel. The fuel elements are permanently attached, since clamping rings are deformed when the elements are attached.

The designs of the two TVS's are identical.

Figure 2.3

RBM-K Process Channel



When the cartridge is assembled, an end cap, two TVS's, and shanks which are held by a nut are assembled on the central rod. The nut is locked by a pin.

Two types of cartridges are installed in the reactor: a working cartridge and a cartridge to monitor energy release (across the core's radius). The latter differs from the working cartridge in the design of the carrier rod. It is hollow and consists of tube with an outside diameter of 12 mm and a wall thickness of 2.75 mm and plugs made of zirconium alloy (alloy 125), a steel-zirconium adapter, and an extension tube made of stainless steel.

2.1.2 Process Channel

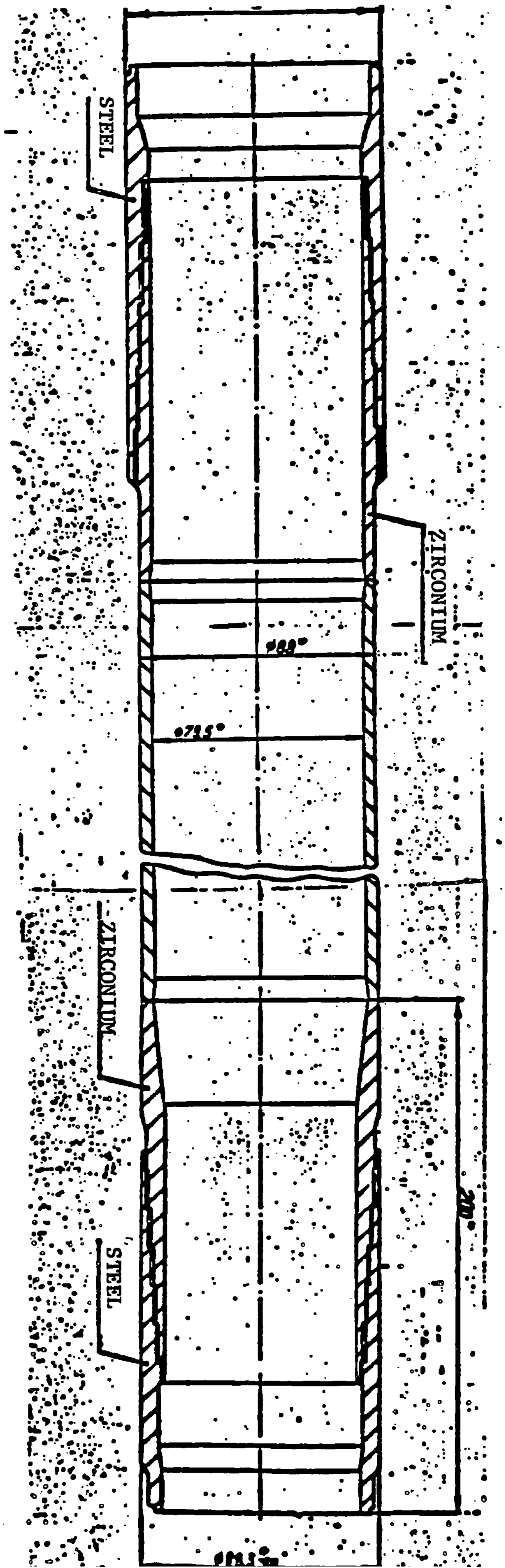
(Figure 2.3)

The process channel is intended to hold TVS's with nuclear fuel and to organize the flow of heat transfer agent. The channel housing is a welded structure consisting of a middle and ends. The middle (2), made of zirconium alloy (Zr + 2.5% No) is a tube with inside diameter of 88 mm and wall thickness of 4 mm; top (1) and bottom (5) ends are made of corrosion-resistant tube (steel 08Kh18N10T). The middle is joined to the ends by special steel-zirconium transition pieces (3, 4).

The corrosion-proof steel/zirconium alloy transition pieces are produced by diffusion vacuum-welding (Figure 2.3 a).

The transition pieces were designed to obtain programmed configurations and a stressed state in the joint area to guarantee strength and reliability under operating conditions. The interior of the joint is made of zirconium

FIG. 2.3a



alloy; the outer cover is made of corrosion-proof steel. During diffusion welding, a thin layer of products of mutual diffusion forms on the contact surface of the parts to be joined. The quality of the diffusion weld is monitored by ultrasonic defectoscopy and metallography. As part of the process channels, the transition piece undergo helium seal and hydrostatic testing.

Channel tubes and zirconium transition piece parts are joined by electron beam welding. To increase corrosion properties, the welded joints are subject to additional hardening and heat treatment.

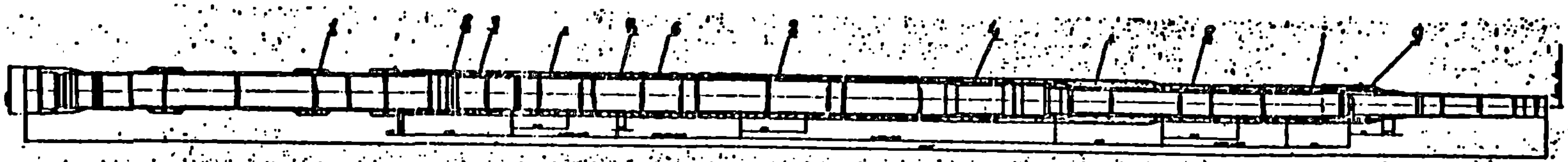
Steel transition piece parts are connected to the top and bottom of the process channel by argon welding. An aluminum metallization coating is applied to the outer surfaces of steel channel parts to protect them from corrosion.

Graphite split rings 20 mm high are set in the middle to improve heat release from the graphite module to the channel. These rings are set along the top of the channel tight against one another so that the side of every other ring is in direct contact either with the tube (7) or with the inside surface of a block (6), and so that their ends touch.

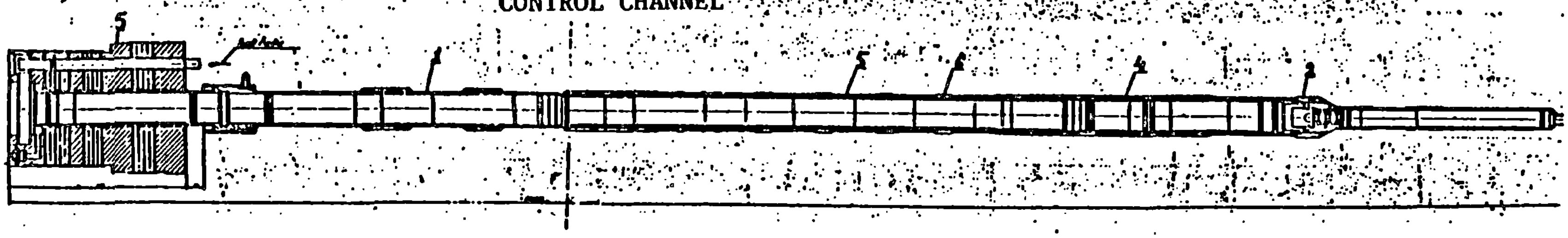
Minimum gaps between the ring and channel -- 1.3 mm -- and between the ring and the block -- 1.5 mm are determined to prevent the channel from becoming wedged in the stack due radial thermal shrinkage during reactor operation.

The channel housing in the reactor is set in pipe routes (3, 4) welded to the top and bottom of the support structure (Figure 2.4).

PROCESS CHANNEL



CONTROL CHANNEL



It is firmly attached in the bottom route by a collar and a thin argon arc weld (1). The bottom of the housing is welded to the metal work route channel through a bellows compensator (2), which makes is possible to compensate for the difference in TK and metal work thermal expansions and provides reliable reactor space seal. The channel housing is rated for 30 yr of trouble-free operation. However, if necessary, a defective channel housing may be removed from an inoperative reactor and replaced with a new one.

The cartridge with fuel elements (5) is set inside the channel on a suspension support (6) which keeps it in the core and permits a spent cartridge to be replaced by the RZM without stopping the reactor.

The suspension support is fitted with a plug (7) which is set in a holder in the upper route. The plug hermetically covers the route space with a ball lock with a packing gland. Depressurization processes during fueling are performed remotely with the RZM.

2.1.3 Control Channels (Figure 2.4)

The channels are intended to hold control rods and energy release sensors and ionization chambers. The middle of the channel (3) is made of zirconium alloy (Zr + 2.5% No) and is a tube with outside diameter of 88 mm and wall thickness of 3 mm. The top (1) and bottom (4) ends are made of corrosion-proof tube (steel 08Kh18N10T). The middle is connected to the end tubes by steel-zirconium adapters similar to those of the process channels. The channels are firmly attached to the upper tube route by a collar and thin weld; to the lower route through a bellows compensator. SUZ channels have caps (5) at the top for attaching actuators and to feed cooling water to the channel. Graphite sleeves (6) are mounted on the channel to ensure required temperatures in the graphite stack. The bottom of the channel has a constrictor to ensure that the channel is completely filled with water.

Placing the control channels in graphite stacks separate from process channels guarantees their safe keeping and, consequently, the working capacity of the control members they contain during emergencies due to process channel ruptures.

2.1.4 Reactor Metal Structure

(Figure 2.1)

The biological shielding vessel (6) is a cylindrical, round-section tank with outside diameter 19 m and inside diameter 16.6 m made of 30-mm-thick low-alloyed plate steel in perlite class 10KhSND. The interior of the tank is divided into 16 vertical hermetically sealed compartments filled with water, from which heat is removed by the cooling system.

The upper metal structure (4) is a cylinder 17 m in diameter, 3 m high. The upper and lower slabs of the cylinder are made of steel 10KhN1M 40 mm thick are welded to the side rim by hermetic welds and to each other through vertical stiffeners. Pipe routes (5) for the process channels and control channels are welded into openings drilled in the top and bottom plates. The space between the tubes is filled with serpentine (a mineral containing bound crystal water). The metal structure is mounted on 16 roller supports attached at the projection of the circular side biological shielding block, and it absorbs forces from the weight of the loaded channels, the central hall floor and piping from the upper steam-water and water lines.

The lower metal structure (3), 14.5 m in diameter and 2 m high, is identical to the upper in design. The structure is loaded by the graphite stacking which is set on it along with support assemblies and lower water lines. The number and arrangement of the lower pipe routes for process and control channels welded to the upper and lower bottom of the metal structure are the same as those in the upper metal structure. Its internal cavity is filled with serpentine. The support structure on which the lower metal work is set consists of plates with stiffeners 5.3 m high (7) which cross along the center of the reactor.

The cylindrical shell (2) is welded, and has an 14.52-m inside diameter; it is 9.75 m high and made of 16-mm-thick 10KhN1M plate steel. The shell has a lens compensator to compensate for longitudinal thermal expansion. The shell and the lower and upper metal structure form the closed reactor space.

The metal structure of the upper span (8) in the central hall has an opening for installation of process and special channels. The opening is covered by removable flooring (9) consisting of separate slabs. The floor provides the the central hall with biological shielding from radiation and with thermal insulation. The slab floor consists of upper and lower slabs and blocks resting on process and control channels routes. Slabs and blocks constitute the metal works which are filled with iron-barium-serpentine aggregate (ZhBCTsK).

Air is sucked from the central hall through openings in the floor into ventilation ducts. The floor cools the air and eliminates the possibility of radioactive effluence into the central hall from steam-water line areas.

2.1.5 Graphite Block Structure (Figure 2.1)

Graphite block structure (1) is set on the lower metal works inside the reactor space. It is a vertical cylinder consisting of 2,488 pieces in columns assembled from graphite blocks with a density of 1.65 g/cm3. The blocks are in the shape of parallelipipeds with a cross section of 250 x 250 mm and are 600 mm high. The stacking weighs 1,700 tonnes. Holes 114 mm in diameter are made along the axis of the blocks to form routes in the columns for process and control and monitoring channels. Each graphite column is set on a steel support plate (1) which in turn rests on a liner welded to the upper plate of the lower metal works. The graphite block structure is kept from moving radially by bars placed in the peripheral columns of the side reflector. The bottom of the rod is welded to the support liner; the top is joined by a flexible connection to the pipe route welded to the lower plate of the upper metal works. A reflector block cooling channel is set into the hollow bar, which is made of corrosion-proof steel 08Kh18N10T pipe. Heat released in the stacking is removed primarily to process channels and partially to SUZ channels. The presence of a solid contact ring in the channels and the fact that the ring-channel and block-ring spaces are filled with helium-nitrogen mixture ensures that structure temperature will be kept below 700 oC.

The maximum temperature zones for the graphite blocks are found in the blocks' ribs; the minimum, on the outside surface of the vertical holes in which the process and other channels are located. Blocks halfway up the central part of the core have the highest temperature.

The greatest temperature drop for the rib-inside surface of the opening occurs in the block with the process channel and is about 150 oC.

2.1.6 Biological Shielding

The biological shielding of Reactor No. 4 at the Chernobyl AES was designed in accordance with current USSR requirements, "Regulations for Radiation Safety NRB-76" and "Health Regulations for Designing and Operating AES SP-AES-79."

The intensity of the doses in terms of external exposure in the central hall and service areas adjacent to the reactor shaft does not exceed 2.8 x 10-2 mSv/hr (2.8 mrem/hr). During recharging, when a spent TVS is removed through the flooring of the central hall, the intensity of gamma radiation near the RZM quickly increases to 0.72 mSv/hr. In the lower waterline area below the reactors, the shielding ensures a reduction in neutron flow density to levels at which there will be no noticeable activation of piping and structures. Entry into this space is possible only when the reactor is inoperative.

Shielding from emissions from the heat transfer agent in the primary loop piping and equipment makes it possible to perform repair and adjustment operations, e.g. adjustment of heat transfer agent flow rate in individual channels using the multipurpose fitting mounted on group collectors, repair of GTsN electric motors, etc. Thermal radiation release is reduced to levels at which the temperature of the metal support structure (upper, lower, container) and reactor shell does not exceed 300 oC, which makes it possible to used low-alloyed steel.

The flow of fast neutrons with energy above 0.1 MeV to the reactor shell and metal works plates close to the core has not exceeded 10 neutrons/cm2 in 30 years of operation.

The shielding developed was implemented as follows (Figure 2.1).

Steel blocks (10) (lower, 200 mm thick; upper, 250 mm thick) are mounted on each graphite column between the 500-mm-thick end reflectors and the upper and lower metal works. These steel blocks are intended to reduce fast neutron flow to the load-bearing metal structure and to reduce energy release in it.

The space between pipes in the upper and lower metal works is filled with serpentine (3, 4), which made it possible to reduce the length of process channels and overall building dimensions.

A shield (reactor room floor) is set above the steam-water lines. Its middle -- floor plate (9) -- is a set of blocks resting on the tops of the pipe routes. These blocks are made of ZhBCTsK. The total thickness of this shield is 890 mm. The upper floor protects the central hall from radiation from the reactor, from piping with radioactive heat transfer agent and, together with the RZM container, helps reduce radiation intensity during removal of a spent TVS. The periphery of the upper span (8) comprises metal ducts 700 mm high filled with iron grit (86% by mass) with serpentine.

Radially, the side reflector consists of 4 graphite blocks with an average thickness of 880 mm. The circular vessel with water (6), located outside the reactor shell, reduces radiation flow to the walls of the reactor shaft (11) which are filled with structural concrete (density 2.2 tonne/m3, wall thickness 2,000 mm). The space between the vessel and the reactor shaft walls is filled with ordinary sand (12).

Table 2.1 presents the thickness and composition of the shielding materials for the RBMK reactor in primary directions from the core.

Table 2.1

Thickness of Shielding Materials (from the Center of the Core), mm

Материал (1)	Направление (2)		
	вверх (3)	вниз (4)	радиальное (5)
(6) Графит (отражатель), мм	500	500	880
(7) Сталь (защитные плиты и лист металлоконструкции), мм	290	240	45
(8) Засыпка из серпентинита (I,7 т/м^3), мм	2800	I800	-
(9) Вода (кольцевой бак), мм	-	-	II40
(10) Сталь (металлоконструкции), мм	40	40	30
(11) Песок (I,3 т/м^3), мм	-	-	I300
(12) Тяжелый бетон (4,0 т/м^3), мм	890	-	-
(13) Строительный бетон (2,2 т/м^3), мм	-	-	2000

KEY: (1) Material

(2) Direction

(3) Upward

(4) Downward

(5) Radial

(6) Graphite (reflector) mm

(7) Steel (shielding plates and sheet of metal structure), mm

(8) Serpentite back filling (1.7 T/m^3), mm

(9) Water (an annular tank), mm

(10) Steel (metal structures), mm

(11) Sand (1.3 T/m^3), mm

(12) Heavy concrete (4.0 T/m^3), mm

(13) Structural concrete (2.2 T/m^3), mm

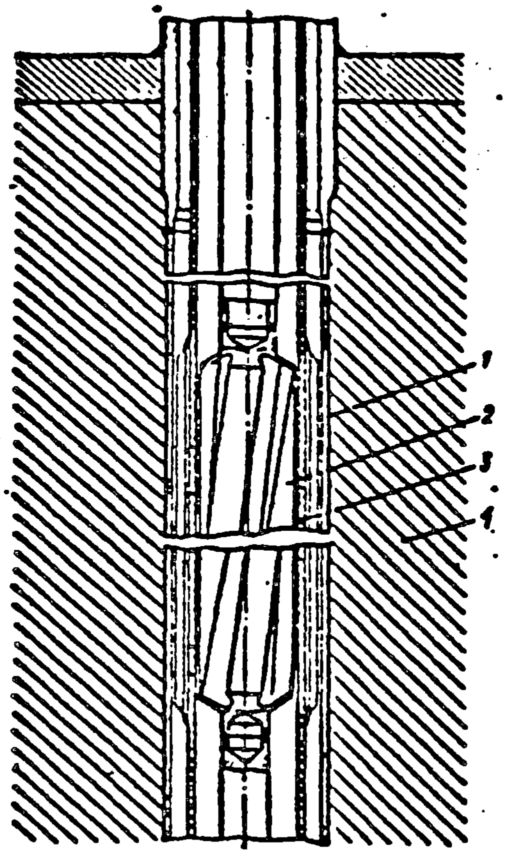


Figure 2.5 Arrangement of safety plug in process channel
1 - Steel sleeve; 2 - Spiral steel plug; 3 - Channel pipe; and 4 - Serpentine fill.

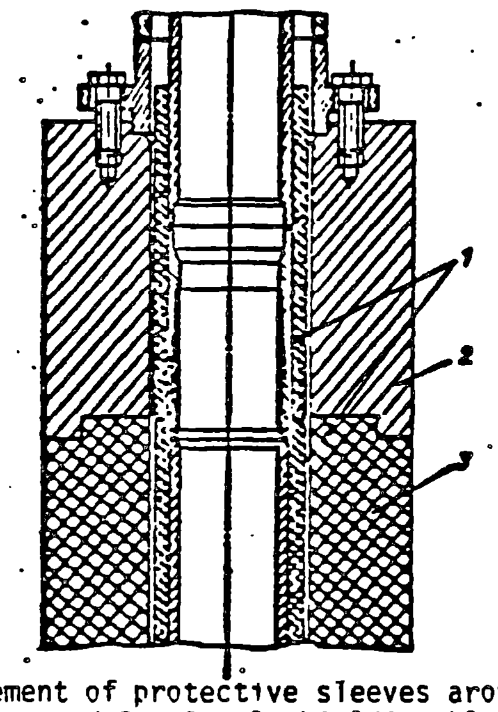


Figure 2.6 Arrangement of protective sleeves around the upper reflector:
1 - Graphite sleeves; and 2 - Steel shielding block; 3 - Graphite reflector.

The intensity of the radiation stream in channels filled with gas (temperature-sensor channels, neutron flow detector channels, ionization chamber channels) or ineffective shielding (steam-water mixture in a process channel) is reduced by installing protective steel or graphite plugs in these channels (Figure 2.5). The annular gap between channels and guide tubes are covered with protective sleeves (Figure 2.6).

Gas piping passing through protective structures is bent (Figure 2.1, item 13).

To prevent neutron and gamma radiation streams, as well as to reduce activation of structures in the sub-reactor area, displacers in SUZ channels are filled with graphite (Figure 2.17, 2.30).

2.2 Controlled Circulation Loop

(Fig. 2.6)

The MPTs loop is intended to supply water to the process channels and to remove the steam-water mixture formed in it as a result of heat removal from the TVS, followed by separation of steam from it.

It consists of two autonomous loops, operating in parallel and similar in arrangement and equipment makeup. Each loop removes heat from half of the reactor fuel cartridges. The loop includes: two steam drums ($I\overset{D}{\emptyset}$=2,600 mm), downpipes (325 x 16), four main circulating pumps (GTsN), GTsN intake pipes (IO=900 mm); GTsN pressure headers ($I\overset{D}{\emptyset}$=900 mm); grouped distribution headers (RGK) (325 x 15 mm) with multipurpose valves; waterlines (57 x 3.5 mm); process channels; steam-water lines (76 x 4 mm). (KMPTs fitting diagram, fig. 2.7)

Water from the intake header (1) travels over four pipes to the main circulation pumps (GTsN) (2).

Under normal operating conditions at rated power, three of the four GTsNs are running; the fourth is in reserve. After the GTsN, water at 270 oC and 82.7 kgf/cm2 travels over pressure pipes on which are installed consecutively a check valve, stop gate valve, and throttle valve, to the GTsN pressure header (3), from which it travels over 22 pipes to grouped distribution header (4) at whose entry check valves are installed, and then over individual waterlines (5) to the entry to the process channels (6).

Flow rate through each process channel is set using multipurpose valves on the basis of flow meter readings. Moving along process channels, water, washing the fuel elements, is heated to saturation temperature and partially (14.5% on the average) evaporated. The steam-water mixture at 284.5 oC and 70 kgf/cm2 (about 7 MPa) travels over individual steam-water pipes to separators (8), where it is separated into steam and water. To maintain a uniform level, the separators are connected by cofferdams. Saturated steam moves through steam headers to turbines. Separated water at exit from the separators is mixed with feed water, and at 270 oC (which ensures the required margin in terms of GTsN suction head), travels to the intake header over 12 down pipes (from each separator).

The temperature of the water sent to the intake header depends on the steam-generating capacity of the reactor plant. As steam-generating capacity drops, temperature rises slightly due to changing ratios between the amounts of water collected from the drums at 284 oC and feed water at 165 oC. If reactor output drops, flow rate over the KMPTs loop is adjusted by throttle-adjusting valves so that the temperature level at the GTsN intake ensures the required positive suction head.

2.3 Special Control Channel Cooling Loop

A special autonomous circulation loop was created. To cool control channels, energy-release sensors, and startup ionization chambers and to cool the side reflector. Water is circulated by gravity, i.e. because of the difference in the levels of the upper (service) and lower (circulation) tanks. Cooling water at 40 oC from the upper tank travels through the header over individual pipes to the channel caps, and moving downward, removes heat from them, itself heating to a temperature of 65 oC. Then it travels through gully water drainage header to heat exchangers, where it is cooled to 40 oC and collected in the lower tank, from which it is pumped to the upper tank. Average water flow rate through the control channel equals 4 m3/hr, pressure at the channel heads is 3.5 kgf/ cm (excess). Flow rate through each channel is set with multipurpose valves according to flow meter readings.

2.4 Gas Loop

Under normal operating conditions, the helium-nitrogen mixture travels at a flow rate of 200-400 nm3/hr at pressure at entry of 50-200 mm H2O excess (0.5-2.0 kPa) to the reactor space over pipes passing through the lower metal works. It is removed through pipes in the process channel integrity monitoring system and special channels which remove gas from the pipe routes of the upper metal works. Then the gas mixture passes through the condenser and the three-stage cleaning system. It is throttled and returned to the reactor space. The gas is circulated by compressors.

The gas cleaning system consists of catalytic reactors, cleaning and drying modules, and deep-cleaning system modules.

The H2 is hydrogenized in the catalytic reactor at about 160 oC, forming water vapor and CO is burned to CO2, releasing heat. The reaction takes place in the presence of a platinum catalyst in an oxygen medium. From the catalytic reactor, the gas, passing through coolers and a dehumidifier, travels to a cleaner-dryer, which is made up of zeolite and mechanical filters. Through adsorption, CO2, H3, C2 and water vapor are removed from the helium-nitrogen gas, which then goes to deep cleaning. The impurities remaining in the gas are removed in this unit by fractionation.

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2.5 Fundamental Data on Physics

The RBMK nuclear power reactor is a heterogeneous thermal neutron channel reactor in which uranium oxide weakly enriched with U235 is used as fuel; graphite as moderator; and boiling light water as heat transfer agent (main reactor characteristics appear in table 2.3).

The development of the reactor was based on experience designing and many years' operating uranium-graphite channel reactors in the USSR. Therefore, methods of neutron-physics analysis tested in working units were the basis for developing a procedure for neutron-physics analysis of the RBMK reactor. There are two basic stages in the research on reactor physics:

a) Calculation of an elementary core cell and preparation for full-scale calculation of the core;

b) Complete calculation of the reactor with regard for the detailed structure of the core.

Programs which permit calculation of the three-dimensional energy distribution of neutrons in a multigroup approximation in a multizone cylindrical cell, as well as in a cell with cluster fuel element arrangement, were used for design engineering calculations in the first stage. Parameters such as uranium burnup depth, fuel isotope composition, and channel power as a function of time, rates of reactions with isotopes incorporated into the cells, along with other characteristics, were determined. Most of the calculations were done for a one-dimensional cell with parameters averaged in terms of height. Constants for calculation on the basis of the reactor schedule are prepared in the form of a polynomial equation for the relationship between burnup and power for various average heat transfer agent density in terms of reactor height.

In the second stage, the reactor is completely calculated with regard for burnup distribution over core channels, actual SUZ rod decay, and the actual power of the equipment. Large-scale status calculations are performed according to a two-dimensional two-group program with regard for actual distribution of the field in terms of the reactor height, obtained from upper-level sensors. If necessary, the reactor is calculated with a three-dimensional program.

When the RBMK-1000 reactor was designed, as well as when already constructed units were operated, in addition to analytical research, a great deal of attention was given to experimental checks and precise definition of the calculating procedures used. Critical RBMK stands which model fragments of the reactor's core were designed and commissioned. At present there is an extensive program of experiments to study the neutron physics characteristics of the RBMK-1000's and the RBMK-1500's core both on the stands and in working units with these reactors.

A system for "continuous" refueling during reactor operation was created for maximum fuel cycle economy. Removal of spent fuel and refueling fuel during reactor operation at the assigned power is accomplished by the fueling machine (RZM). When the reactor attains steady-state operating mode (continuous fuel mode), all reactor characteristics stabilize, and the fuel removed from the reactor's core has approximately constant burnup depth, the magnitude of which is determined by fuel enrichment replenisher, as well as by the assigned number of control rods introduced into the zone necessary to form the optimum energy release field over the radius and height of the reactor. The reactivity margin for RBMK-1000 reactors is 1.5+1.8% (30+36 rods). Fuel burnup depth at 2% replenishment with U235 is P=22.3 MW-day/kg. Note that, because construction materials with low absorption sections and a high heat transfer agent steam content are used, the fuel discharged from the RBMK-1000 reactor during continuous refueling is close in terms of fissionable isotope content to enrichment plant dumps, which virtually eliminates the need for further processing for return to the fuel cycle.

When the RBMK-1000 was designed, a great deal of attention was given to substantiating channel and fuel element working capacity.

The main parameters which determine the limit thermal load for the channel and the fuel element are critical channel power Ncrc (when this level is reached a boiling crisis develops on the surface of the fuel elements, causing overheating of the fuel element can), and maximum permissible linear load on the fuel element q_l^{kp} (when this is exceeded, the oxide fuel melts.)

To evaluate the anticipated N_c and q_l in the reactor, probability method of determining possible deviations was used. This procedure accounts for various factors which affect limit N_c and q_l, including the accuracy with which overall reactor power is measured and maintained and its distribution over the core (coefficient for nonuniformity across the core radius K_r and along the height K_z, as well as over the fuel element in the cartridge K_{car}, which determine maximum design channel power N_c^{max} and linear load on the fuel element q_l^{max}. It was believed that random deviations in maximum power from the most probable value, N_c^{max}, follow a normal distribution. Limit channel power is determined from the equation

$$N_k^{Lim} = N_k^{max} (1 + 3\sigma_k)$$

where σ_k is the standard error in determining and maintaining channel power.

In accordance with the Gaussian distribution curve, the probability of a channel with maximum power (freshly fueled channel in the plateau zone) exceeding N_c^{Lim} will equal (1-0.9987)=0.0013.

Similarly, the limit linear load on the fuel element

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

where σ_q is the standard error in determining and maintaining linear fuel element power.

On the basis of calculations and operating experience, the following initial values are used to evaluate N_c^{lim} and q_l^{lim}:

$$K_r = 1.48;\ K_z = 1.4;\ \sigma_K = 5.2\%;\ \sigma_q = 7.7\%.$$

Along with economic and technological indicators, the core's dynamic characteristics are of particular importance, especially from the standpoint of operating safety. The so-called steam reactivity coefficient α_ϕ is of special significance. Experimental radiation at operating RBMK units, as well as analytical studies, show that coefficient α_ϕ is positive at design core parameters in stable conditions and reaches 2x10-6 units per percent of steam by volume.

However, the set of equipment developed to control the RBMK reactor includes a system to ensure reliable compensation for possible energy release field instabilities due to positive reactor feedback in terms of steam content. Specifically, the control/safety system (SUZ) includes local automatic adjustment (LAR) and local protection (LAZ) subsystems. Both operate from signals from the ionization chambers inside the reactor. The LAR automatically stabilizes the primary harmonics of the radial-azimuthal distribution of energy release, while the LAZ ensures protection against exceeding the assigned cartridge power in its individual regions. To adjust the upper-level fields, there are truncated USP absorber rods (24) which are to be introduced into the zone from below.

Besides improved reactor monitoring and control equipment, there are other means for improving the dynamic characteristics of the RBMK reactor's core.

These include:

Increasing replenisher fuel enrichment to 2.4-3.0% and, correspondingly, the depth of fuel burnup, which makes it possible to reduce the steam effect to almost zero;

Increasing the uranium charge to reactor channels by using fuel compositions with increased U content.

Calculations of the effects and reactivity coefficients due to a change in moderator and fuel temperature, as well as the size of the "fast" power reactivity coefficient are shown in table 2.3

Calculations and experiments show that the "fast" power reactivity coefficient when the reactor operates at rated parameters is negative and close to zero.

Table 2.3

Main Neutron-Physics Characteristics of the RBMK-1000 Reactor

Fuel enrichment	2.0%
Uranium mass in a cartridge	114.7 kg
Number/diameter of fuel elements in TVS	18/13.6 mm
Depth of fuel burnup	20 MW day/kg
Coefficient of non-uniformity of release of energy along the radius	1.48
Coefficient of non-uniformity of release of energy along the height	1.4

Table 2.3 Cont.

Calculated maximum power of channel	3,250 kW
Isotopic composition of unloaded fuel:	
uranium-235	4.5 kg/t
uranium-236	2.4 kg/t
plutonium-239	2.6 kg/t
plutonium-240	1.8 kg/t
plutonium-241	0.5 kg/t
Void reactivity coefficient at a working point	2.0×10^{-6}/vol.% steam
Fast power reactivity coefficient at a working point	-0.5×10^{-6}/MW
Coefficient of expantion fuel temperature coefficient	-1.2×10^{-5} / °C
Coefficient of expantion graphite temperature coefficient	6×10^{-5}/°C
Minimum "weight" of rods of SUZ, ΔK	10.5%
Effectiveness of rods of RR, ΔK	7.5%
Effect of replacement (on the average) of the burnup TVS with fresh	0.02%

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2.5.1 Main Data on Reactor Thermal Physics

2.5.1.1 Parameters Defining the Thermotechnical Working Capacity of the Reactor

The main parameters which define the working capacity and safety of the boiling water-graphite reactor in the thermal engineering sense are: element fuel temperature, graphite block structure temperature, and margin to channel power at which a heat exchange crisis begins.

The hydrodynamic stability of fuel channels in a RBMK-1000 boiling water-graphite reactor is usually not a limiting factor, since hydrodynamic instability usually occurs at channel powers greater than those at which a heat exchange crisis occurs.

Experimental studies performed during design confirmed this conclusion and showed that rated parameters for RBMK-1000 reactor operation lie in the hydrodynamic stability zone.

If the permissible fuel temperature is surpassed or a heat exchange crisis occurs, an individual fuel element may fail. However, after it is replaced, the reactor's working capacity is restored.

Calculations of the criticality margins and determination of the maximum element fuel temperature in RBMK-type reactors at stable power levels are performed using statistical probability methods, and the status of the core of these reactors during operation is monitored on the basis of these same methods.

In transient and emergency modes, when parameters change rapidly, it is reasonable to assume that the probability that the definitive thermal engineering parameters will exceed the limit values is higher than during operation at stable power levels. As experimental data and operating experience with boiling water-graphite reactors show, a momentary heat exchange crisis, and a rise in temperature beyond permissible for stable modes in these reactors will not cause the TVS's to fail.

The criticality margin and maximum fuel temperature in transient and emergency modes for boiling water-graphite reactors are determined from current average values of parameters affecting these values.

2.5.1.2 Thermophysical Characteristics in Stable Reactor Operating Modes

The makeup of the RBMK reactor's core depends on the operating period. The initial operating period for these reactors is characterized by the presence in the core of channels with fuel which has a low burnup depth and large number of additional absorbers required to compensate for excess reactivity. As the fuel is burned up, the load on the core varies continuously. In this transient period, the channels in the reactor core contain fuel at various burnup depths, additional absorbers of varying efficiency, and channels filled with water. The transient period of reactor operation ends after all or almost all of the additional absorbers are removed from the core and are replaced by new fuel cartridges. The reactor is operated further in continuous refueling mode, when the fueling machine replaces spent cartridges with fresh one.

A reactor operating in continuous refueling mode can be represented as a system consisting of so-called "periodicity" cells. Each "periodicity" cell consists of channels filled with fuel cartridges with at various burnup depths. At any given moment, different channels have different powers, but the total power of all channels in the "periodicity" cell remains nearly constant.

The RBMK reactor's design provides the possibility of adjusting water flow rate over the fuel channels during a fuel operating period by changing the level at which the multipurpose valves installed at entry to each channel are opened during reactor operation. Channel-by-channel adjustment of water flow rate is provided to ensure sufficient margins to heat exchange crisis in the most thermally stressed core channels when the total water flow rate through the reactor is moderate. The flow rate of water through the channels is adjusted during an operating period on the basis of readings from a flow meter mounted at entry to each reactor channel to a level determined by calculation to ensure required steam content at exit from the channel or the required margin to heat exchange crisis in a given channel.

The possibility of measuring and adjusting water flow rates in each RBMK reactor fuel channel is a distinguishing feature of these reactors and ensure redistribution of water flow rates if reactor power and the energy release field across the core radius change.

According to the thermal analysis algorithm for the RBMK reactors, the distribution of water flow rates over core fuel channels is calculated by ordinary iterative method using the total characteristics of circulating pumps and the down path of the circulation loop.

To determine the overall hydraulic characteristics of individual structural elements and fuel channel in the reactor, experiments were conducted on special model stands and on half-scale fuel channel simulator stands for a reactor with a thermal capacity to 6 MW.

The equations for calculating relative hydraulic resistance factors for a bundle of rods washed by a two-phase flow takes the form:

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

for the actual volumetric steam content in a channel:

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

and for the phase slippage coefficient:

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

where

$$W_0 = \frac{G}{\rho' S}$$

is circulation rate, β daily volumetric steam content.

Figure 2.6 is a chart comparing experimental values for pressure drop at the heated part of the full-scale stand and calculations. The graph shows that the calculation technique satisfactorily describes experimental data and can be used in thermal calculations for reactors.

If the thermal power of each fuel channel and the water flow rate through it are known, critical channel power, Ncr, minimum margin to heat exchange crisis, Kz, the probability of a channel entering a heat exchange crisis mode, R, and the probability of crisis-free operation of all core channels, H, are calculated.

Functions for calculating critical RBMK fuel channel power were determined as a result of analysis and processing of experimental material on a heat exchange crisis in smooth bundles of heated rods and in bundles of rods with heat exchange intensifiers. Experiments were conducted on stands with various (including full-scale) geometries

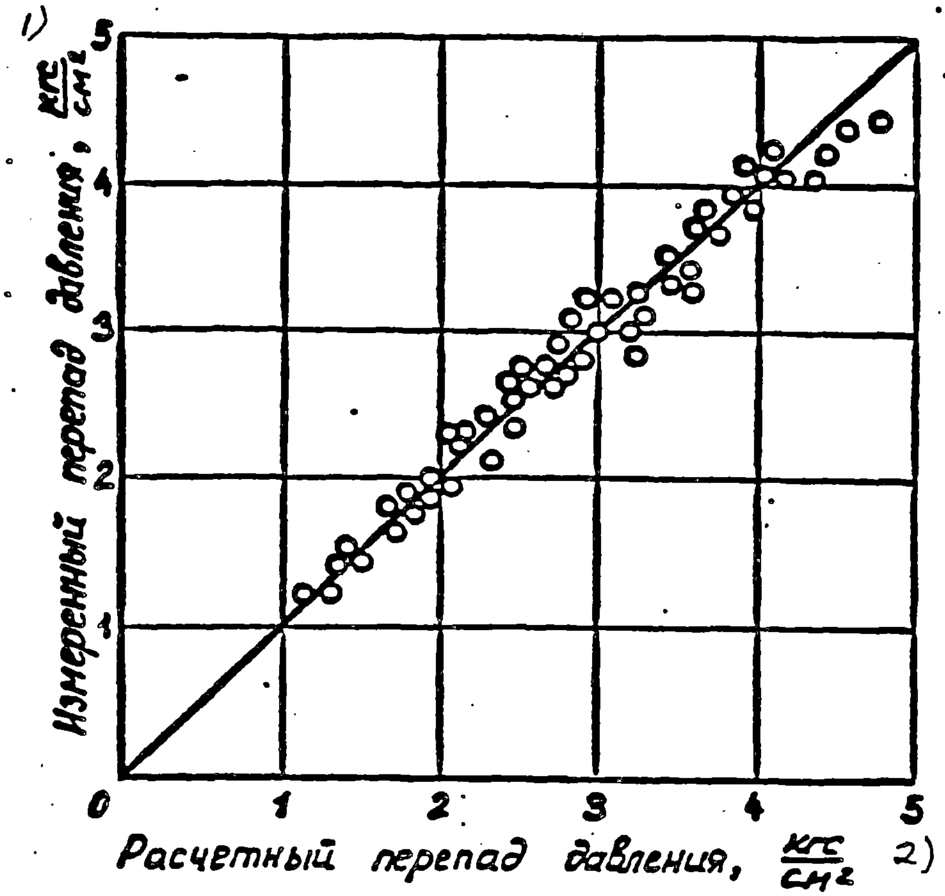


Figure 2.8. A comparison of the experimental and calculated values of the hydraulic resistance of a full scale testbench: O are the experimental values and ——— are the calculated values.

KEY: 1) Measured pressure differential, kgf/cm^2

2) Calculated pressure differential, kgf/cm^2

of bundles at heat transfer agent parameters close to working reactor parameters.

The equation for calculating critical heat flow in fuel channels without heat exchange intensifiers takes the form:

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

where $\Phi(z)$ is the relative distribution of energy release along the channel's height;

$z_{кр}$ ~~zcr~~ is the coordinate of the crisis site, m;

Δh ~~h~~ is water heating to saturation at entry, kJ/kg.

A set of calculating programs which was developed makes it possible to perform a thermal calculation of an RBMK reactor operating with continuous refueling regardless of the position of the multipurpose valves at entry to each "periodicity" cell channel. Thus it is possible to determine the thermal engineering parameters for the reactor at various per-channel flow rate adjustment rates, various adjustment laws (on the basis of exit steam content or on the basis of margin to critical power), and at various levels of core pre-throttling.

The results of the calculation of the effect of channel flow rate adjustment rate on the thermal engineering parameters of an RBMK reactor with an electric output of 1,000 MW (RBMK-1000) with continuous refueling appear in Figure 2.9. As the equations shown there show,

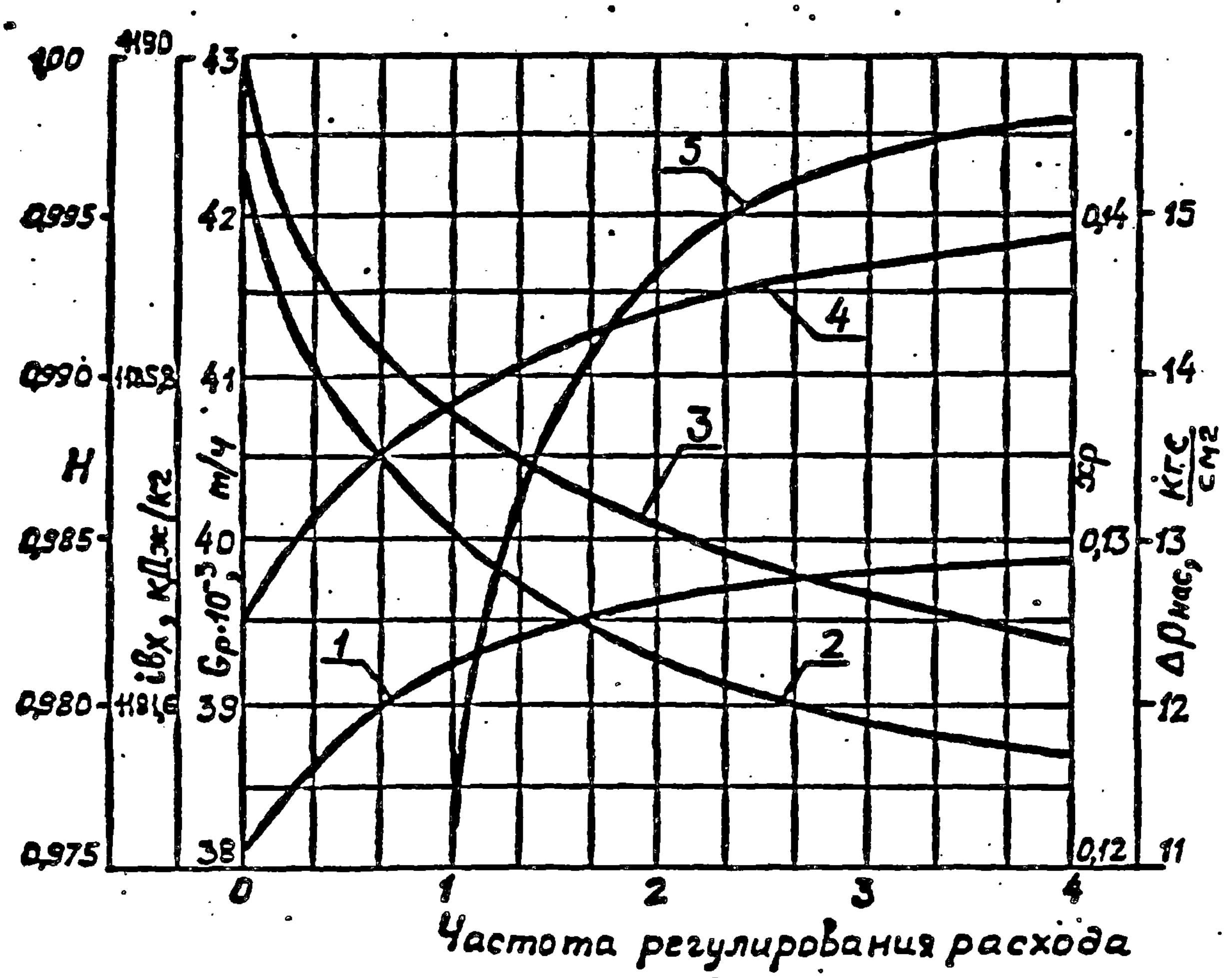


Fig. 2.9 Reactor parameters as a function of per-channel flow rate adjustment:

1 - GTsN head

2 - Heat transfer agent flow rate (Gp);

3 - Thermal content at entry;

4 - Steam content at exit;

5 - Thermal engineering reliability (H).

if the rate at which the flow rate of each channel is adjusted increases, parameter H, which describes the thermal engineering reliability of the core, rises, and this increase is more noticeable if the rate of adjustment doubles over the fuel operating period. A further increase in adjustment rate does not cause a significant increase in H. On the basis of calculations performed during the planning of the RBMK-1000 reactor operating with continuous refueling, a double adjustment was used for water flow rate through each fuel channel over the operating period of a fuel cartridge.

To perform a thermal calculation of a reactor operating in the transient (from the standpoint of TVS overload) operating period, a mathematical model was developed to allow distribution of water flow rates and margins to heat exchange crisis in core channels with regard for the specific characteristics of each individual reactor channel. The reactor core in this case is represented as a system consisting of channels filled with fuel cartridges with various burnup depths and with additional absorbers of any type. The distribution of energy release over reactor channels is determined either as a result of physical calculation for the subject core status and the position of the control rods or is sent to the reactor developed by a special automated communications system from operating power units with RBMK reactors. As a result of calculating reactors for a given core status and energy release distribution over reactor channels, the optimum distribution of water flow rates in channels and the hydraulic profiling of the core required to ensure this are determined.

In operating RBMK reactors, criticality margins and fuel temperature regime are monitored by a special program (PRIZMA) using a resident computer. The temperature regime of the graphite block structure is monitored by thermocouples placed across the radius and along the height of the structure.

The distribution of energy release in the reactor core is calculated using readings from a physical monitoring system based on measurements of neutron fuel across the radius and along the height of the core taken directly in the reactor. In addition to physical monitoring system readings, the computer also receives data describing the makeup of the core, the energy output of each fuel channel, the position of control rods, the distribution of water flow rates in core channels, and heat transfer agent pressure and temperature sensor readings. As a result of calculations with the PRIZMA program performed regularly by the computer, the operator obtains information on a printer in the forms of core cartograms which indicate the type of core fill, the position of control rods, the network of sensors inside the reactor, the distribution of powers, water flow rates, criticality margins and margins to permissible thermal loads on fuel elements for each reactor fuel channel. Criticality margins and margins to limit permissible thermal loads are calculated using statistical probability methods with regard for error calculating the energy release field along the reactor height and radius, errors in calculating formulas and the precision with which technological parameters for installing the instrumentation and automation system are measured and maintained.

The resident computer also calculates the reactor's total thermal power, the distribution of steam-and-water mixture flow rates in separators, integral energy output, steam content at exit from each fuel channel and other parameters required to monitor and control the plant.

When the reactor operates at stable power levels, and when power is raised or lowered, the operator monitors and controls the energy release field across the radius and along the height of the core, using readings from physical monitoring system sensors. If the field deviates from the assigned value by a certain amount, a light signal is activated on a special display. At the same time, signalling is provided if sensor signals exceed given absolute values for margins to limit permissible thermal loads on fuel elements (Kq). The operator also monitors and controls distribution of flow rates in core fuel channels. Flow rates are distributed on the basis of calculating the criticality margins (Kz) in fuel channels on an external computer and with the PRIZMA program on the resident computer.

The temperature regime of graphite block structures in operating RBMK reactors is monitored using thermocouples mounted at the corners of graphite blocks at various points along the structure. In addition to direct measurements of graphite temperature at structure support points, the PRIZMA program can be used to calculated maximum (in terms of height) graphite temperature near any reactor fuel channel. Graphite temperature is calculated on the basis of thermocouple readings and the distribution energy release over the core volume calculated with the PRIZMA program.

The temperature of the RBMK reactors' graphite block structure is adjusted by changing the composition of the gas mixture in the structure (nitrogen+helium). At present, on the basis of experience operating domestic water-graphite reactors, maximum graphite temperature at which the structure will not burn up in the absence of water vapor has been set at 750 oC.

Experience operating RBMK reactors shows that, given the monitoring and adjustment equipment available on these reactors, there is no difficulty maintaining the temperature behavior of fuel and graphite and margins to heat exchange crisis at the permissible level when power is at stable levels.

For section 2.5.1 Basic Data on Reactor Thermophysics

Notation G - flow rate, kg/sec; S - Cross section area, sq. m; d - diameter, m; g - gravitational acceleration, m/sec-sec; x - mass steam content;

- density, kg/cu m; p - pressure, kgf/sq cm; q - thermal flow density, kW/sq.m; W - velocity, m/sec.

К разделу 2.5.I.

Основные данные по теплофизике реактора

Условные обозначения

G - расход, кг/с;

S - площадь поперечного сечения, $м^2$;

d - диаметр, м;

g - ускорение в поле тяготения, $м/сек^2$;

x - массовое паросодержание;

ρ - плотность, $кг/м^3$;

P - давление, $\frac{кгс}{см^2}$;

q - плотность теплового потока, $кВт/м^2$;

W - скорость, м/с.

Subscripts

ob - heated;

(g) - hydraulic;

kp - critical;

/ - water at saturation line;

// - steam at saturation line.

Индексы

об - обогреваемый;

г - гидравлический;

кр - критический;

/ - вода на линии насыщения;

// - пар на линии насыщения.

2.6 Unit Process Flow Diagram

The process flow diagram is a single-loop scheme based on the modular principle of a double unit: reactor - two turbines without transverse steam and feedwater connections.

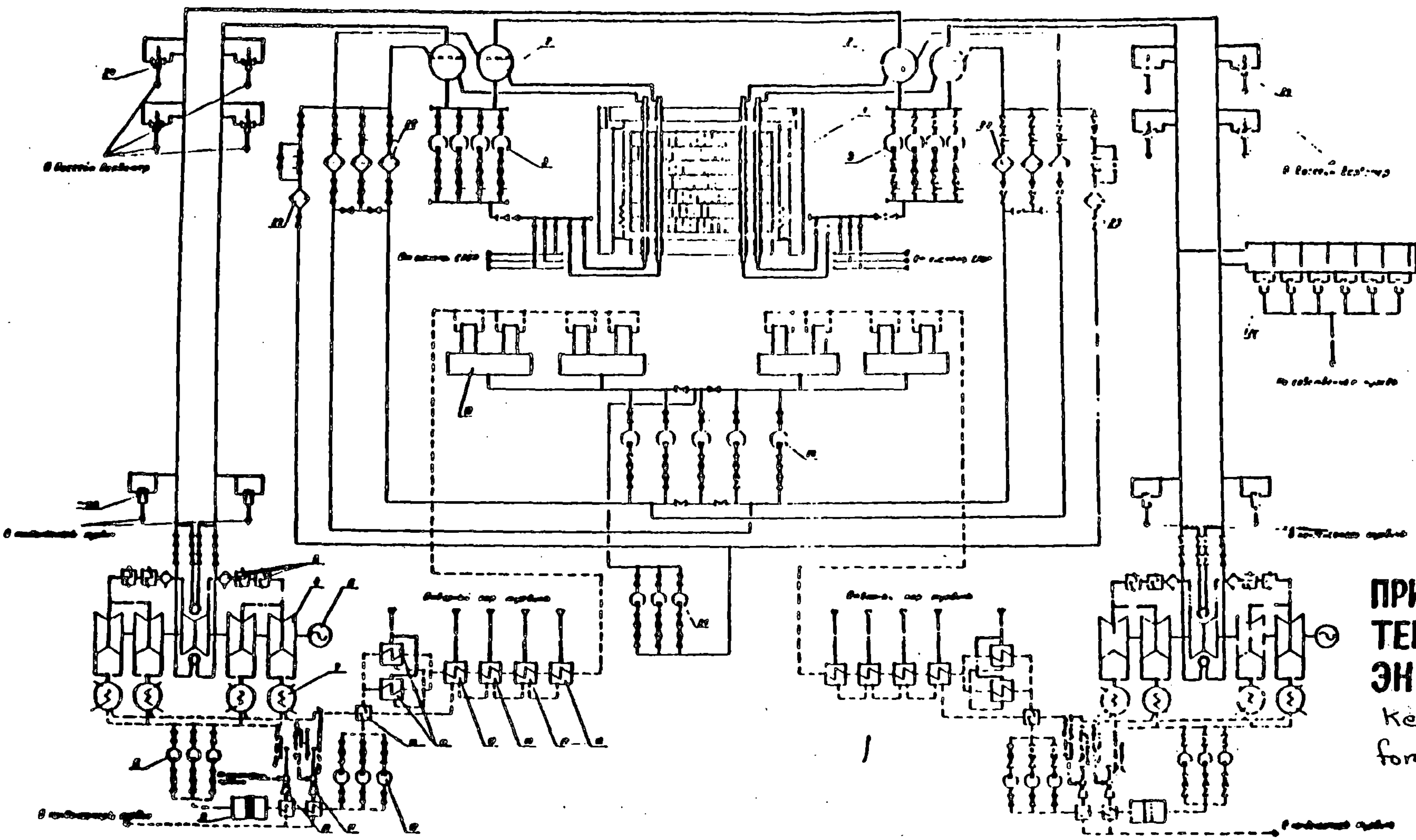
The power unit operates according to the following scheme (cf. Figure 2.1.0):

The controlled circulation loop's heat transfer agent (water) travels over 325 x 16-mm diameter downpipes from the lower part of the steam separator at 265 oC and 69 kgf/sq cm to the main circulation pumps' (GTsN) 1026 x 63 mm diameter intake header. The main circulation pumps send the water to the 1046 x 73 mm diameter pressure header and then along 325 x 16 mm diameter pipes to 22 grouped distribution header (RGK). From the RGK, lower waterlines (NVK) 57 mm in diameter individually carry water to reactor fuel channels.

The steam-and water mixture formed in the reactor travels along steam-and-water lines (PVK) 76 mm in diameter to be divided among four steam separators to produce saturated steam to run the turbine.

Steam is bled from the upper part of each separator along 14 steam bleed tubes 325 x 19 mm in diameter to two steam headers 426 x 24 mm in diameter, which are then connected to one header 630 x 25 mm in diameter.

Fresh steam travels over four pipes 630 x 25 mm in diameter to the turbines in the machinery room (two pipes per turbine).



ПРИНЦИПИАЛЬНАЯ ТЕПЛОВАЯ СХЕМА ЭНЕРГОБЛОКА

Key Diagram for Power Unit

50a

Steam discharge devices are located at the steam line area before the turbines' main steam gate valves (GPZ): eight main relief valves (GPK) with a throughput of 725 tonne/hr, four fast-action turbine condenser reducers (BRU-K) with a capacity of 725 tonnes of steam per hour (two per turbine set), and six fast-action reducers for internal needs (BRU-SN). The purpose and operating mode for these devices is described in section 2.7.

The steam spent in the turbine, is condensed in the condensers. The condensate from the condensers, 100% distilled, is sent by condensate pumps through low-pressure heaters to a 7.6-atm deaerator (2 deaerators per turbine). From the deaerator, feedwater at 165 oC is pumped by five electric feed pumps, one of which is a backup, travels to steam separators, where it is mixed with the circulating heat transfer agent.

The primary reactor process systems besides the controlled circulation loop include the following:

- Emergency reactor cooling system (CAOP);
- KMPTs blowdown and afterheat system (CP and R);
- Gas loop;
- Fuel can seal monitoring system (KGO);
- Holding tank water cooling and treatment system;
- Scheme "D" biological shielding tank water cooling system;
- SUZ channel cooling system;
- Intermediate reactor section loop;
- Process channel integrity monitoring system (KTsTK).

MPTs Loop

The controlled circulation loop is intended for continuous delivery to the reactor's fuel channel of heat transfer agent which removes heat generated in the reactor and for generation of the steam-and-water mixture and its separation to obtain saturated steam to run the turbine.

The MPTs loop consists of two independent, identical loops, each of which cools the corresponding part of the reactor. All loop equipment is arranged symmetrically to the reactor's transverse axis. Each circulation loop includes:

- 2 steam separators;
- Cofferdams between steam separators for water and steam;
- Downpipes;
- Intake header;
- GTsN intake pipes;
- 4 main circulation pumps (3 working, one backup);
- GTsN pressure pipes with fittings;
- Pressurized header;
- Cofferdam between GTsN intake and pressure headers with fitting;
- Group distribution header (RGK);
- Lower waterlines;
- Reactor fuel channels;
- Steam-and-waterlines.

Downpipes, RGK, and water- and steamlines are made of stainless steel 08Kh18N10T. Pressurized and intake collectors and GTsN pipes are made of carbon steel 330E with a surface buildup of steel 1CL473N6 from Creusot-Loire of France.

Chemical Composition of KMPTs Materials

Материал MATERIAL	C	Mn	Si	S	P	Ni	Cr	Cu	Nb	Mo
STEEL Сталь Крезелсо 330E	≤0,23	0,9-I,2	0,2-0,4	≤0,025	≤0,025	≤0,3	≤0,4	≤0,3	-	-
STEEL Сталь 1CL473N6	≤0,05	≤0,2	≤0,75	≤0,02	≤0,035	8,5-I0,5	I8-20	8x% C-0,65	-	0,6

Материал MATERIAL	Temp. При температуре, °C	$\alpha \cdot 10^6$ 1/°C	E, N н/мм²	σ_B kgf кгс/мм²	$\sigma_{0,2}$ kgf кгс/мм²	δ_5 %	ψ, %	a_{HI} кгс/см²	a_{U}, кгс/см²
STEEL Сталь Крезелсо 330E	20	II,I	205200	44-60	≥22	≥20	≥48	≥7	≥4
	350	I5,0	I88000	≥36	≥I9	≥I8	≥43	-	≥3
STEEL Сталь 1CL473N6	20	I6,5	I93000	≥50	≥20	≥38	≥50	-	-
	350	I7,5	I79000	≥36	≥I5	≥24	≥40	-	-
08XI8HI0T	20	I6,4	205000	52	22	35	55	-	-
	350	I7,6	I75000	42	I7	26	5I	-	-

Stress-Strain Properties of Materials

A relief valve, throttle valve, stop valve with remote control electric drive, and a measuring membrane are installed sequentially on the pumps' pressure piping. The presence of stop valves on the pumps' intake and pressure pipes makes it possible to sideline a pump for repair while the loop is operating.

The throttle valve makes it possible to keep the GTsN's capacity in the stable unit operation zone from 5,500 to 12,000 cu m/hr in transient modes. Between the intake and pressurized collectors is a cofferdam 750 mm in diameter intended to ensure natural circulation in the loop if a pump stops. The cofferdam has a check valve which prevents overflow from the pressurized collector to the intake collector in normal system operating mode and a stop valve which is normally open in all operating modes.

The outlet branches of the pressure header have flow limiters in case of piping rupture. During pre-startup flushings, mechanical filters are attached to them. The pipe supplying water to the RGK have manual stop valves. In normal conditions, these gate valves are open and sealed and are closed only during repair of the MPTs system. The RGK has check valves, past which (in the direction of flow) waterlines distribute water from the RGK to individual reactor channels.

KMPTs Blowdown and Afterheat Cooling System

The blowdown and afterheat cooling system is intended for the following:

- In rated mode — to cool blowdown water from the MPTs system before cleaning, after which the water is heated before it is returned to the MPTs system;

- In shut-down cooling mode — to remove heat from the KMPTs;

- In startup mode — to cool system blowdown water before cleaning, after which it is heated before return to the system and to discharge unbalanced waters from the loop when it is heated.

The blowdown and afterheat cooling system includes a regenerator, large and small blowdown precoolers, 2 consumption pumps, piping and fittings.

2.6.1 Gas Loop. Condenser and Filter Unit

To prevent oxidation of graphite and improve heat transfer from the graphite to the process channel, the spaces between graphite blocks and block structure sleeves are filled with a nitrogen-helium mixture (20 vol.% N2 and 80 vol.% He). Impurities are removed and the nitrogen-helium balance maintained in the gas mixture by a helium cleaner (OG).

In normal mode, the gas loop system operates as follows. The nitrogen-helium mixture leaving the apparatus passes through the KTsTK system, where the temperature of each channel is checked and the overall humidity of the mixture being pumped is monitored.

Then the mixture travels to the condenser and filter unit.

The condenser and filter unit for the gas loop is intended to condense water vapor entering the nitrogen-helium mixture when the reactor channels are depressurized and to remove iodine vapors from the gas mixture.

The system is based on the 2 x 100% principle, i.e. it consists of two independent subsystems, one of which is the working system; the other, the backup.

The nitrogen-helium mixture travels from the RP to the condenser. The condensate is removed from the condenser through a hydroseal to intermediate tanks for drainage water through a permanently open repair valve.

Service water travels to the condenser

at a pressure exceeding that of the steam-gas mixture, both in rated and in emergency modes.

After the condenser, the gas mixture has about 100% humidity. If it is not immediately sent to the filter, the moisture may condense, which would cause the filter to fail.

Therefore, before the gas mixture is sent to the filtering column, it is dried in the electric heater section.

In the column, solid particles and iodine aerosol are removed from the mixture.

The filtering column is rated to clean 1,000 cu m per hr of gas mixture. After the filtering column, the gas mixture, depending on gas loop operating conditions, goes either to the intake header of the helium cleaner's compressors or to the activity suppressor (UPAK).

Each of the two subsystems is laid out in individual boxes, which makes it possible to repair equipment in one subsystem while the other is operating.

2.6.2 KTsTK System

The KTsTK system houses sensors to monitor channel integrity. The KTsTK is intended for:

- group monitoring of the humidity of the gas bled from the graphite block structure and pumped through the system;

- determination of reactor channel damage;

- blocking dissemination of moisture from a damaged channel to adjacent cells;

- drying the reactor's graphite block structure.

The integrity of channels in a working reactor is monitored by measuring the temperature of the gas pumped in the spaces between the channels and the graphite structure (routes). To increase the amount of water vapor in the gas being pumped, its temperature is increased, and this is recorded by thermocouples mounted in group channels. The reactor's graphite block structure, with the channels permeating it, is divided into 26 zones, each of which contains up to 81 channels.

Pulse tubes from channel routes in each zone lead to the corresponding group valve in this zone. Each of the 26 group channels is identified by the same number as the reactor zone corresponding to it.

Outlet branches of a valve are connected by tubes to ventilation and increased suction headers of the KTsTK system. Both these headers are connected to the reactor's process gas loop, thus closing the KTsTK system into the reactor's process ventilation system.

Switching a valve gate can change the amount of gas pumped through impulse tubes leading to a given valve by connecting it either to the ventilation system or to the increased suction system.

2.6.3 Helium cleaner

The helium cleaner is intended to remove oxygen, hydrogen, ammonium, water vapor, carbon monoxide, carbon dioxide, methane, and nitrogen from the gas mixture circulating in the closed loop through the RBMK apparatus and reduce them to a concentration which permits normal reactor operation.

The gas mixture is contaminated if moisture enters the block structure cavity through defects in process channels; then this moisture partially decomposes due to radiolysis into hydrogen and oxygen, which, reacting with carbon, forms carbon monoxide and carbon dioxide gas. Hydrogen, combining with the graphite, forms methane; with nitrogen, ammonium

The primary specifications for the helium cleaner are:

1. Amount of mixture at 293 K and 101,325 Pa (760 mm Hg), cu m/sec (cu m/hr)	0.0833-0.264 (300-950)
2. Pressure at cleaner inlet, MPA (mm Hg)	0.003 (300)
3. Composition of uncleaned mixture, % (vol.)	
Nitrogen	20
Oxygen	0.3
Methane	0.1
Ammonium	0.07
Carbon dioxide	0.02
Carbon monoxide	0.1
Hydrogen	0.6
Chlorine	Trace
Helium	Rem.

4.	Pressure at exit from cleaner, MPa (mm Hg)	0.005 (500)
5.	Temperature at exit from cleaner, K (deg C)	308+/-10 (35+/-10)
6.	Cleaned mixture composition, % (vol.):	
	Nitrogen	10
	Oxygen	0.01
	Methane	Trace
	Ammonium	Trace
	Carbon dioxide and monoxide	0.01
	Hydrogen	0.02
	Helium	Rem.
7.	Secondary products required:	
	Liquid nitrogen, cu m/sec (cu m/hr)	0.039 (140)
	Gaseous nitrogen, cu m/sec (cu m/hr)	0.097+/-0.104
	Gaseous oxygen, cu m/sec (cu m/hr)	0.0042 (15)
	Cooling water, cu m/sec	0.0056 (20)
8.	Length of working life, yr	1.5
9.	Length of startup period, sec (hr)	57,600 (16)
10.	Time to first major overhaul, hr (yr)	43,800 (5)
11.	Service life, yr	/30

2.6.4 Spent Fuel Holding Tank Water Cooling and Treatmention System

The system is intended to maintain the temperature of water in cartridge holding tanks and process channels heated by residual heat release from spent fuel and by process channels. The system is based on the 2 x 100% principle and consists of pumps, heat exchangers, piping and fittings.

The treatmen plant's pumping station is intended to supply holding tank water to the ion-exchange filters in the treatment plant. The holding tank water treatment system is one of two power units. The operating mode is periodic.

2.6.5 Biological Shielding Tank Cooling System

The pump-heat exchanger unit (NTU) is intended to maintain the temperature of the water in the reactor's biological shielding tanks.

The system includes the following equipment: cooling loop circulation pumps, heat exchangers, expansion tank, piping, and fittings.

2.6.6 Control and Protection System Channel Cooling System

Purpose and Design Basis

The SUZ, KD, DKE and reactor reflector cooling channel cooling system is intended to provide the assigned temperature in these channels.

The following requirements are imposed on the system:

- To maintain the assigned temperature in SUZ, KD, DKE and KOO channels in all unit operating modes (startup, operation to power, shut-down, disturbance of normal operating procedure, emergencies);

- To maintain assigned water quality standards in terms of chemical composition and specific activity.

The system is a circulation loop operating on gravity, i.e. water flows through channels due to the difference in the level of the upper and lower tanks.

Water from the upper, so-called emergency water supply, tank travels over piping with a nominal diameter (ND) of 400 mm to the pressure collector and is distributed to channels.

The capacity of the emergency tank is set to ensure rated flow rate through channels for 6 minutes when pumps are not working.

Cooling water from the pressure header enters a channel from above, moves down along the central tube and ascends along the annular gap between the center and outer tubes to the KOO drainage collector. There are two drainage headers (ND=200).

Water from the drainage header travels over piping (ND=400) to the system's heat exchangers. Water from the KOO drainage header comes to this same pipe. (This header is connected to a common pipe (ND=150) before entry to the pipe (ND=400).) A throttling device which eliminates the siphon in the KOO drainage headers is mounted on the common pipe (ND=150).

There are six heat exchangers to cool the circulating water past the reactor.

Past the heat exchangers, water travels over pipe (ND=400) to the circulation tank below water level. The flow slows in the circulation tank and conditions for efficient removal of hydrogen from the water are created. A description of the method for ensuring safe hydrogen concentration in the tank appears in section 2.7.

Some of the water from the emergency supply is constantly discharged in overflow pipes to the circulation tank. This water represents the difference between pump capacity and SUZ, KD, DKE, and KOO channel throughput. If two pumps are working in the system, overflow proceeds along piping (ND=150); if three pumps are working, along piping (ND=300).

The system has four pumps to supply water from the circulation tank to the emergency supply tank. Two of these are working pumps; two are backups. The first backup pump is activated automatically; the second backup pump is switched on by the operator if necessary.

The pumps are powered from a category 1B reliable supply net with diesel generators.

To maintain the required water quality in the loop, there is constant bypass treatment at 10 cu m/hr.

2.6.7 Intermediate Reactor Area Loop

The intermediate reactor area loop is intended to prevent radioactive media from entering service water from the heat exchange equipment of systems with radioactive heat transfer agents if their impenetrability is disrupted. This is achieved because pressure in the intermediate loop is below the pressure of the service water.

The loop is a closed system including an expansion tank, pumps, heat exchangers, and stop, safety, and control fittings. The loop's pumps supply cooling water to the heat exchange equipment in the reactor area systems, and remove heat from it. This heat is then absorbed by intermediate loop heat exchangers cooled by service water. The expansion tank maintains stable pump operation, filling, replenishment and compensation for a change in the volume of intermediate loop heat transfer agent. For secondary reactor systems which are at higher elevations and for which cooling water cannot be supplied by primary loop circulation pumps, there are pressure-boosting pumps which feed water to steam separator sampler heat exchangers and to the reactor's fueling machine.

Regular or continuous treatment of water in the intermediate reactor area loop with special water treatment plants is not necessary. The quality of the intermediate loop water is determined by sampling. If the chloride content is exceeded or the medium's pH exceeds established norms, the water in the intermediate loop is purified by exchanging water in the system.

Intermediate loop consumers are the reactor blowdown and afterheat cooling system, the system which organizes leaks from the fitting equipment, GTsN sealing water chillers, the helium cleaner, and chemical monitoring sampler heat exchangers.

2.6.8 Water Regime

The reliability, safety, and economy of fuel element operation, and normal radiation condition at an AES are determined by the water-chemical regime of the primary and secondary loops.

The following requirements are imposed on the system's water-chemical regime:

- Reduced entry of contamination into the reactor core;

- Preventing water-containing impurities from building up on the core elements.

The RBMK uses a neutral water regime without suppression of water radiolysis and without introduction of corrective additives to adjust pH.

Quality of the heat transfer agent in the loop by Goct 95743-79 must follow requirements.

- pH - 6.5 - 8.0
- electroconductivity less than 1.0microSi/cm.
- impurity less than 10 microgram equivilant/kg
- Si acid less than 100 microgram/kg
- chloride + chloride ions less 100 micrgram/kg
- iron oxides less than 100 microgram/kg
- copper oxides less than 20 microgram/kg
- oxygen - 0.05 -0.1 milligram/kg
- oil less than 200 microgram/kg

Feedwater must follow requirements below :

- pH -7.0
- electroconductivity less 0.1 microSi/cm
- ion oxides less than 10 microgram/kg
- oxygen 0.03 milligram/km

At the time of the operation of the nuclear power plant, all the time must be organized, required chemical regime of the coolant in loop of circulation.
The radioactive water

must be cleaned before repeated use or dumping.
Radioactive water transported to the special water purification station

which

consists of a number of components. Components can be selected on the main and support components.

The main components of the special water purification system are:

- Bypass cleaning of the blowoff water from the loop :
- Purification of water of the spent fuel storage
- Purification of the water of the control roads cooling system.

- Gully water treatment;
- Organic seepage treatment;
- Washing water and loosening water treatment;
- Treatment of deactivating solutions for the controlled circulation loop;
- Treatment of bubbling pond water.
- Auxiliary SVO units include:
- Preparation of recovery solutions;
- Perlite preparation and precoat;
- Charging filters;
- Pumping tars to KhZhTO;
- Preparation of deactivating solutions;
- Reuse of deactivating solutions;
- Equipment deactivation.

These units, in addition to the MPTs loop blowdown water bypass treatment plant and the bubbling pond water treatment plant are located in block, "B" on axes 35-41 at elevations 0.00; 6.00; and 12.50, and are intended for 2 blocks.

The MPTs loop blowdown water bypass treatment plants are located in block "A" and block "B". The bubbling pond water treatment plants and the plant which pre-treats drainage waters with mechanical filters are located in the VSRO block.

2.6.9 KMPTs Blowdown Water Bypass Treatment Plant

The plant is intended for bypass treatment of circulating loop blowdown water to remove products of corrosion and dissolved salts. The plant is the primary means by which loop water quality is maintained to prevent deposits on fuel elements and ensure continuous KMPTs operation. It can remove fragmentary nonvolatile radioisotopes from the loop, reduce induced activity and, most important, reduce radioactive contamination of steam and condensate-feed routes. Each block has its own independent plant.

The plant is rated to treat 200 tonne/hr of loop water. This capacity is based on the extent of blowdown in terms of products of corrosion and makes it possible to maintain standardized MPTs loop water indices. In stable modes, the plant's capacity may be lower. In transient modes, if pressure is no greater than 16 kgf/sq.cm, products of corrosion built up in stable regime can be removed by the KMPTs deactivating solution treatment plant, which makes it possible to ensure design MPTs loop blowdown in terms of products of iron corrosion during reactor startup and shut-down cooling.

Elements in the system:

1. Mechanical ionite filter - 1
2. Combined action ionite filter - 2
3. Trap filter - 1
4. Moisture trap - 1

2.7 Main Block Equipment

Reactor

The steam-generating plant at the station uses a series-built RBMK-1000 power reactor. The reactor and its technical characteristics are described in section 2.2.

Turbine

The mechanical drive for the TVV-500-2U3 AC generator is a K-500-65/3000 high-speed turbine with underground condenser.

Main Design Characteristics for the turbine set appear in the following table:

Основные расчетные характеристики турбоагрегата приведены в таблице:

№№ пп (No)	Наименование (Item)	Размерность (Unit of Measure)	Величина (Amount)
I	2	3	4
I.	Maximum turbine output, MW Максимальная мощность турбины, МВт	MW МВт	550
2.	Rated turbine output, net Номинальная мощность турбины, нетто	MW МВт	5I0
3.	Rated fresh steam flow rate, including fresh steam flow rate to 2nd stage Расход свежего пара номинальный, включая расход свежего пара на вторую ступень п.п.	т/ч tonne/hr	2890
4.	Max. fresh steam flow rate, including to 2nd stage Расход свежего пара максимальный, включая расход свежего пара на вторую ступень п.п.	"	2902
5.	Initial steam pressure Начальное давление пара	k f/cm^2 abs. кгс/см^2 абс.	65,9
6.	Initial steam temperature Начальная температура пара	^{o}C	280,4
7.	Initial steam humidity Начальная влажность пара	%	0,5
8.	Potable water heating temperature Температура подогрева питательной воды	^{o}C	I68
9.	Design pressure in condensor Расчётное давление в конденсаторе	кгс/см^2 абс. k f/cm^2 abs.	0,05
I0.	Type of steam distribution Тип парораспределения	дроссельное	
II.	Turbine design diagram Конструктивная схема турбины	2ЦНД+ЦВД+2ЦНД 2LP+HP+ZLP	
I2.	Structural formula for regeneration Структурная формула схемы регенерации	5ПНД+Д 5LP+P	
I3.	Number of regenerative steam bleed-offs Число регенеративных отборов пара	шт. PC	7
I4.	Speed Частота вращения	об/мин	3000
I5.	Turbine central heating lead (intermediate loop curve 160/80^{o}C) Теплофикационная нагрузка турбины (график промежуточного контура I60/80 ^{o}C)	Gcal/hr Гкал/час	75

1	2	3	4
	Condenser specifications Техническая характеристика конденсатора		
16.	Amount of steam to be condensed (per condenser) Количество конденсируемого пара (на один конденсатор).	tonne/hr т/ч	441,105
17.	Cooling water temp. at condenser entry Температура охлаждающей воды на входе в конденсатор	°C	18
18.	Number of cooling water passes Число ходов охлаждающей воды	pc шт.	2
19.	Cooling surface Поверхность охлаждения	$м^2$	12150
20.	Hydraulic condenser resistance Гидравлическое сопротивление конденсатора	m H_2O м.в.ст.	3,63

Steam Separator

The RBMK-1000 steam separator is intended to produce dry saturated steam from a steam-and-water mixture.

The separator is a horizontal, cylindrical vessel with elliptical bottom with manholes 400 mm in diameter.

The steam-and-water mixture comes to the separator over 432 PVK branches which are arranged in the cylindrical part of the lower half of the separator, four rows on each side. The kinetic energy of the steam-and-water mixture is suppressed and the steam rough-separated by recoil plates inside the separator.

Then the steam, passing over a submerged plate, is separated in the steam space, and, crossing a perforated ceiling plate, exits through 14 steam branches arranged at the upper generatrix of the separator.

Steam pressure in the separators is monitored by 4 connecting pipes on the housing of each separator and 24 connecting pipes for connecting water gages.

The separator is mounted on five supports; the middle support is fixed, the rest are sliding guides.

The materials in the primary steam separator assembly units and parts separator are:

a) Shell and bottom - steel 330E + IC 473 B (clad steel), Creusot-Loire, France (see section 2 for composition and properties);

b) Steam bleed pipes - steel 330E;

c) Mixture feed pipes and loop water downpipes -- clad steel: steel 330 steel + IC 473 B;

d) Interior separator devices - steel IC 473 B.

RBMK-1000 reactor steam separator technical data:

- Steam-generating capacity, t/hr - 1,450
- Saturated steam pressure, kgf/sq.cm:
- Working - 70
- Rated - 75
- Steam humidity at exit from separator - no more than 0.1%
- Steam temperature, deg C - 284.5
- Feedwater pressure at entry to the steam separator, kgf/sq.cm - 71
- Feedwater temperature, deg C - 165
- Loop water flow rate, tonne/hr - 9,400
- Steam-and-water mixture flow rate, tonne/hr - 9,400
- Average steam content in steam-and-water mixture going to separator - no more than 15.4%

- Steam separator level adjustment drop from rated, mm, no more than +/- 50

- Effective water supply in separator with possible level position 100 mm below nominal, at least 51 cu m.

- Steam separator service life, yr - 30
- Steam separator weight: dry, t - 280
- In working condition, t - 394
- During hydrotesting, t - 439
- Main steam separator dimensions:

  Length, mm - 30.984

  Inner housing diameter, mm - 2,600

  Minimum base metal wall thickness, mm - 110

Deaerator

The deaeration plant is a deaerator consisting of a deaerator tank and two deaeration columns. The deaeration tank has three supports; the two outer supports are sliding rollers which permit for the deaerator to expand during heating; the middle one is fixed to restrict horizontal movement of the central part of the deaerator and to permit its vertical movement. The deaerator's working pressure is 6.6 kgf/sq.cm; temperature, 167.5 deg C. During hydrotests (when completely filled with water) it weighs 204 tonnes.

Main Circulation Pump

The pump is a centrifugal, vertical single-stage unit. The shaft has a double end seal with supply of a small amount of sealing water to keep the heat transfer agent from leaking into the space.

Main pump characteristics:

- Capacity - 8000 cu m/hr
- Head - 200 m H20
- Temperature of heat transfer agent being pumped - 270 deg. C

- Pressure at pump intake - 72 kgf/sq.cm
- Minimum permissible positive suction head - 23 m
- Power per pump shaft - 4,300 kW
- Electric motor power - 5,500 kW

The unit consists of a tank, cavity, and electric motor.

The pump tank is welded of steel 15Kh2MFA and is coated inside with anticorrosive surface coating. It supports the hollow part of the pump and connects it with a socket sealed by a gasket. The recess contains the shaft and rotor, guide, lower hydrostatic bearing, O ring and upper thrust-guide bearing, which are in the housing. The pump's design permits partial or total replacement of its cavity.

Water is pumped to the hydrostatic bearing from the common pressure header through a hydrocyclone.

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

The pump permits continuous operation at flow rates from 5,500 to 12,000 cu m/hr. Pump heating and shut-down cooling is permitted at a rate of 2 deg C/min.

Feed Pump

An electric pump assembly is used to ensure supply of feedwater from deaerator to steam separators.

The electric pump assembly is a three-stage unit with unilateral arrangement of rotors and prerotation propeller, hydraulic pivot, slit-type end seals and sliding bearings with forced lubrication. Cold condensate (t=40 deg. C) is delivered to the pump seals.

The amount of mechanical impurities in the condensate being delivered should not exceed the amount of mechanical impurities in feedwater, both in terms of weight and in terms of volume.

Main assembly characteristics:

- Capacity - 1,650 cu m/hr
- Head - 84 kgf/sq.cm
- Feedwater temperature - 169 deg. C
- Pressure at pump inlet - 9 kgf/sq. cm
- Minimum permissible position suction head - 15 m H2O
- Power per pump shaft - 4,200 kW
- Service water flow rate - 36.5 cu m/hr
- Oil flow rate - 3.5 cu m/hr
- Cold condensate flow rate - 21 cu m/hr
- Electric motor power - 5,000 kW

Condensate Pumps

Condensate is carried from the condensor through the low-pressure heater system to deaerators by condensate pumps I and II.

Condensate pump I is an electric pump unit -- centrifugal, vertical, double-housing sectional type -- with interchangeable prerotation propeller and end seal: packing gland and end.

Main unit characteristics:

- Capacity - 1,500 cu m/hr
- Head - 12 kgf/sq.cm
- Pressure at inlet, no more than - 0.2 kgf/sq. cm
- Condensate temperature - to 60 deg. C
- Minimum permissible positive suction head, at least 2.3 m H20

- Power to pump shaft - 615 kW
- Condensate flow rate to end seal - 3 cu m/hr
- Cooling water flow rate to pump bearings - 1.5 cu m/hr
- Electric motor output - 1,000 kW
- Assembly weight - 24,560 kg

Condensate pump II is an electric, centrifugal, horizontal spiral pump with bilateral inlet prerotation propeller. The end seals come in two interchangeable versions:

- End seal for continuous operation;
- Packing gland seals for startup and adjustment.

Main assembly characteristics:

- Capacity - 1,500 cu m/hr
- Head - 240 kgf/sq. cm
- Inlet pressure, no more than - 15 kgf/sq. cm
- Condensate temperature - to 60 deg. C
- Minimum permissible positive suction head, at least - 22 m H20
- Electric motor output - 1,600 kW
- Assembly weight - 10,335 kg

Piping

Pressurized and intake collectors for the KMPTS (ND=800) and piping (ND=800) to and from the GTsN are made of carbon steel 330E with a surface buildup of steel 1C 473 B supplied by Creusot-Loire of France.

KMPTs pipe with diameter up to 300 mm is made of stainless steel grade 08Kh18N10T. Reactor auxiliary system piping is made of carbon steel. Condensate-feedwater conduit piping is made of steel 20.

Fresh steam piping is made of steel 17 GS.

Fueling Machine

The most important requirement imposed on the RBMK reactor is that it operate with a minimum number of shutdowns. Therefore, it can be refueled and certain emergency situations eliminated in an operating reactor without reducing output. This is done by a special fueling machine (RZM) which performs the following operations:

- Reloads TVS in operating reactors and reactors shut down to cool;
- Tests the passability of the process channel route with a gauge which simulates a standard cartridge;
- Seals the process channel with a plug (process or emergency);
- Mechanically eliminates certain emergency situations.

Fuel is reloaded into a working reactor at working process channel parameters.

The fueling machine performs five operations per day to refuel process channels in a working reactor without reducing its output and at least 10 operations to refuel channels in a shutdown reactor.

The machine's main components are a crane, container, two interchangeable pressure housing (one on the machine, the other in the repair zone), a truss, process equipment, guidance system, and controls.

The machine's operating principle in a working reactor is as follows.

The RZM, which is filled with condensate at 30 deg. C is butted against the channel to be refueled. Pressure in the pressure housing is set equal to the pressure in the process channel, and the channel is depressurized. Condensate is pumped from the pressure housing at a flow rate of 1 cu m/hr. Cold condensate blocks penetration of steam and hot water from the process channel to the RZM. After removal of the spent cartridge, the channel is sealed and pressure in the pressure housing brought to atmospheric. The machine separates from the channel and moves to the next site to remove spent cartridges.

The RZM has two systems for precise guidance to the process channel: optical-TV (main) and contact (backup) in case visibility is lost due to steam from the channel.

The optical-TV system makes it possible to visually observe the image of the end of the process channel cap through a TV or eyepiece and to match the circumference of the process channel cap to the viewfinder's circumference by small movements of the bridge and carriage. The RZM's contact guidance system is a pneumoelectromechanical device intended to guide the RZM to the channel axis by means of direct mechanical contact between the system and the side of the process channel cap.

The RZM is controlled from the operator's area which is behind the end wall on the reactor side of the central hall.

In addition, the RZM's cabin has a crane travel control panel.

The central hall has the following machine service areas: 1. Stopping point - a zone in the central hall intended to stop the machine during reactor refueling.

2. Training stand, intended for:

- Adjusting and checking machine mechanisms;
- Filling the pressure housing with condensate;
- Simulating standard refueling;
- Charging a fresh cartridge into the housing;
- Deactivating the inner space of the housing.
- The training stand has appropriate equipment to perform these operations.

3. Spent cartridge receiver to receive the gauge.

4. The repair zone is intended for replacement of a pressure housing which has malfunctioned. It is located in the central hall near the training stand. A completely assembled spare pressure housing is always in the zone.

The safety system equipment is described in section 2.9.

2.8 The Reactor Control and Safety System

The control and safety system of an RBMK reactor makes it possible to: monitor the level of neutron power of the reactor and the period of its increase under any operating conditions, namely from 8 times 10exp-12 to 1.2 N nominal;

startup of the reactor from a down condition to a specific level of power

automatic control of the power of the reactor at a specific level and changing a specific level of power;

manual (from the control panel) adjustment of the distribution of energy release over the active zone and control of reactivity to compensate for burnout, poisoning, and so forth;

automatic stabilization of the radial-azimuthal distribution of energy release in the reactor;

preventive protection, namely a rapid controlled reduction in reactor power to safe levels: AZ-I = 50% N nominal, AZ-2 = 60% N nominal;

emergency protection in the event of an accidental change in reactor or generation unit parameters (AZ-5)

The control and safety system includes (a diagram is given in Figure 2.11):

neutron flux sensors with devices (mounts) for setting them in the reactor

reactivity controls (absorbers) with actuator mechanisms which move the controls within the reactor channels

the system's instrumentation, which converts information from the neutron flux sensors and generates digital signals for subsequent processing in the system's logic components and analog signals for indicating and recording reactor parameters;

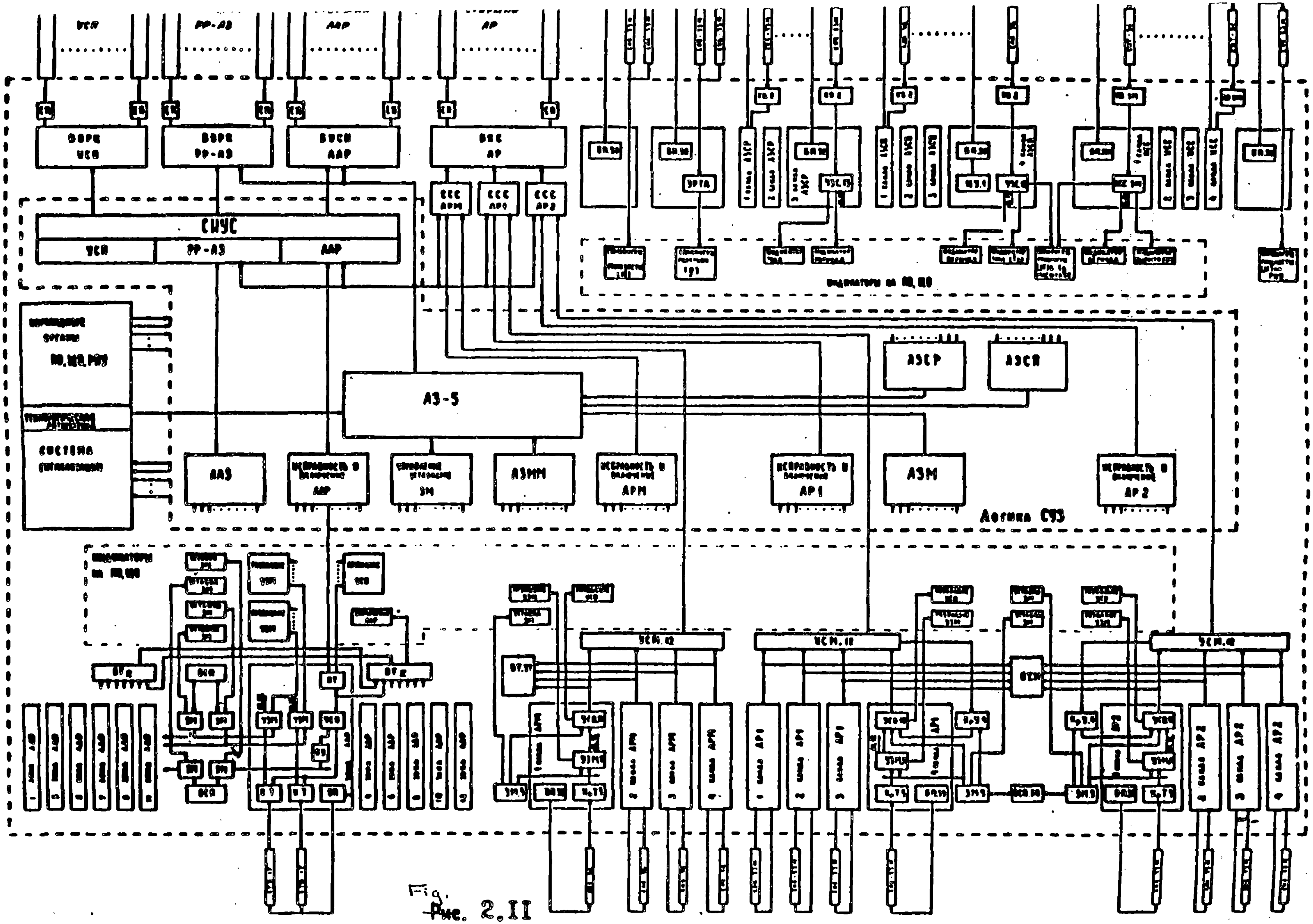


Fig.
~~Рис.~~ 2.II

the components of the control and safety system's logic system, which implements specific control and safety algorithms and processes digital signals from the instrument and actuator components of the system, from command instruments at the operator control panels, from the automation system of the generation unit, and other systems; processing generates commands to shift the control rods under normal and emergency conditions, adjust the power level, change operating conditions, and trigger alarms and signals;

the actuator equipment of the control and safety system, which controls the servodrives of the actuator mechanisms of the system on the basis of instructions from the system's logic components;

display instruments for indicating and recording reactor and control and safety system parameters at the reactor operator panel and board;

2.8.1. The Location of the Primary Equipment of the Control and Safety System

Neutron flux sensor mounts are installed in the following places:

24 IK mounts in the water shielding tank around the reactor, including 16 mounts with KNK-53M working ionization chambers (RIK mounts) and 8 mounts with KNK-56 (PIK) startup ionization chambers;

4 mounts with KNT-31 fission chambers (KD) are lowered into the reflector channels during the startup period and are removed after the PIK sensors are operating properly;

in the center holes of the TVS (?) are 24 intrareactor sensors with KTV-17 type fission chambers.

All of the system's 211 actuator mechanisms are mounted on control and safety system channels in the reactor. The servodrives of the mechanisms are channel type drives. The locations of the control rods are indicated by means of a synchro sensor mounted in the servodrive and a synchro receiver (rod position indicator) mounted on the control system mimic panel on the operator's board. The extreme positions of a rod are indicated by means of limit switches which include upper and lower end lights built into the corresponding position indicators.

The instrumentation of the actuator component of the control and safety system is located in the control and safety system room behind wall Ts 3 and includes:

a control panel for the RR-3 and USP servodrives;

a board of control units for the AR servodrives, which consists of three panels with separate BKS units;

a control bay for the LAR (type BA-86) servodrives, which includes 12 sections;

a servodrive temperature monitoring bay.

Three panels with components of the AR rod synchronization system are installed in room BShchU-N.

The components of the instrumentation portion of the system are located in a separate section of the BShchU-N control board and consist of individual control and safety system electronic instruments mounted on 19 panels of the electronic instrument board and two LAR-LAZ bays with electronic components made as plugin sections.

A set of indicators and recorders are mounted on the reactor operator panel and board.

The logic components of the control and safety system are also located in the BShchU-N room.

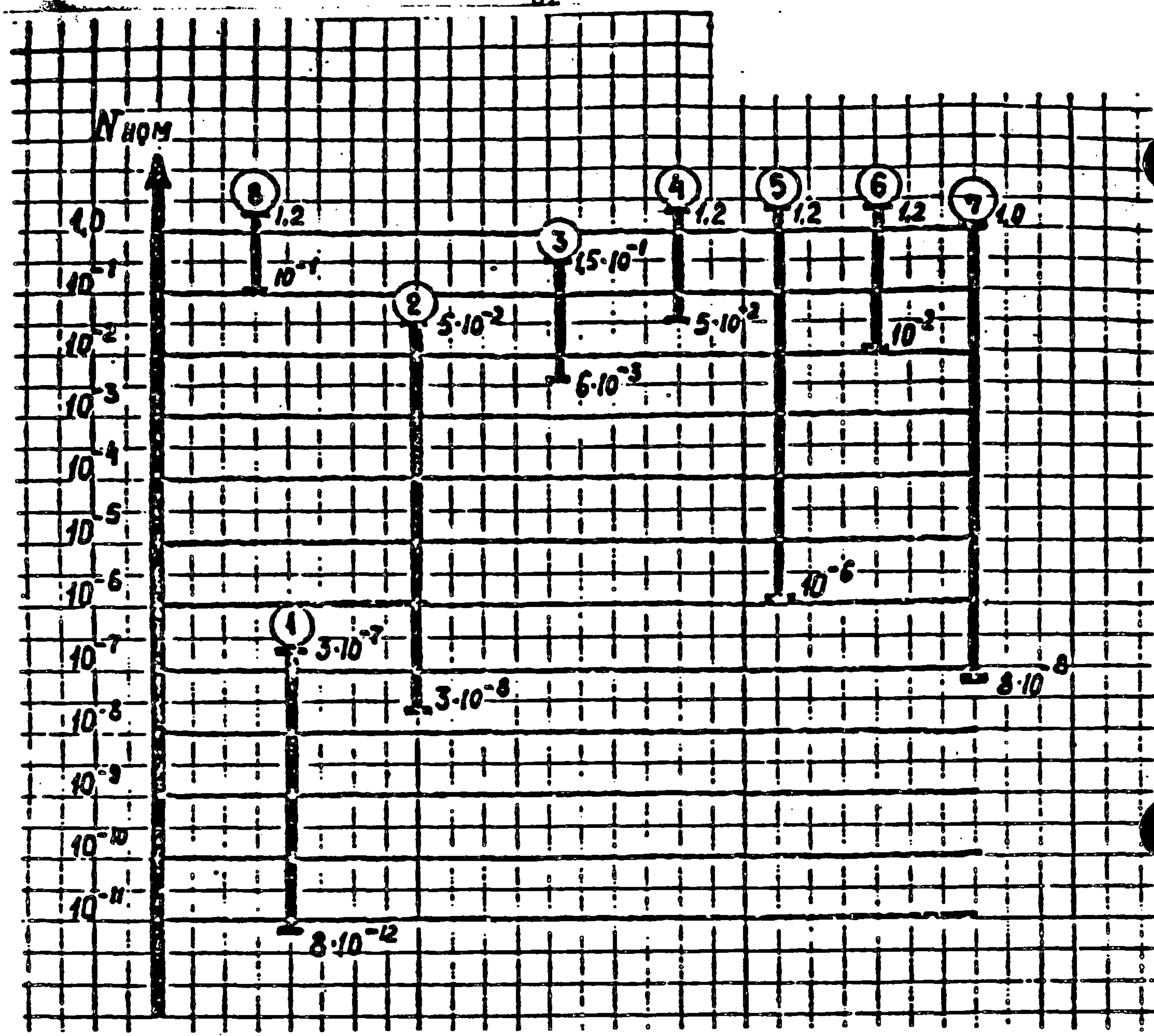
The reactor's alarm system, which includes sound alarms and light alarms located on the reactor operator's board, involves the use of equipment contained in SSZ cabinets in the BShchU-N room.

The controls (switches, buttons, and so forth), which the operator uses to control the rods, vary reactor output, change operating modes, and so forth, are located on the operator's panel.

2.8.2 Neutron Flux Monitoring

Under startup conditions (8 times 10exp-12 to 3 times 10exp-7 N nominal), neutron flux is monitored on four independent measurement channels with KNT-31 fission chambers. The sensitivity of a chamber to neutron flux is 0.25 pulses/1/square centimeter. Secondary electronic instruments (ISS.ZM count rate meters with KV.ZM stages), which run off the fission chambers, are used to determine neutron flux density on a logarithmic scale and reactor rideup time. The output information of these channels is displayed on indicators on the operator panel and may be recorded from a channel of the operator's choice.

At intermediate power levels of 3 times 10exp-8 to 5 times 10exp-2 N nominal, neutron flux is monitored by means of signals from four KNK-56 startup current ionization chambers characterized by enhanced sensitivity to neutron flux (4 times 10exp-13 A/1/(square centimeters per second). In order to reduce the influence of the gamma-background, the chambers' channels are surrounded by lead shields. Additional compensation for the gamma background is provided by adjustment of the negative supply voltage for the chambers' compensation electrodes. The signals of these chambers and secondary instruments (UES.13 emergency protection amplifiers with KV.2 log taking stages) are used to determine neutron flux density on a logarithmic scale and reactor rideup time and to generate signals for reducing rideup time to the warning and alarm points. The output information of these channels is displayed on gauges on the operator panel, while information from any one channel may be recorded by the tape of a recorder at the operator's board.



1. range of power monitoring on a logarithmic scale by means of ISS. ZI instruments with KNT-31 chambers
2. range of power monitoring on a logarithmic scale by means of UZS.13 (AZSP) instruments with KNK-56 chambers (with lead)
3. range of power monitoring by means of ARM instruments with KNK-56 chambers
4. range of power monitoring by means of 1 (2) AR instruments with KNK-53M chambers
5. range of power monitoring on a logarithmic scale by means of UZS-13 (AZSR) instruments with KNK-53M chambers
6. range of power monitoring on a linear scale by means of a recorder with a PRU and KNK-53M chamber
7. range of power monitoring on a linear scale by a KSVP recorder at the operator's board with KNK-53M chambers
8. range of power monitoring by LAR channels with KNV.17 chambers

Figure 2.12. Neutron Flux Monitoring Ranges

Digital warning and alarm signals on the reactor rideup period are processed in the safety logic circuitry.

Neutron flux is monitored and recorded on a linear scale in the 8 to 10exp-8 to 1.0 nominal range by means of two KNK-53M ionization chambers with a neutron flux sensitivity of 1.45 times 10exp-14 A/1/(square centimeter/sec). A KSVP 4 high resistance multirange recorder is used as a secondary instrument.

Reactivity is measured by means of an ERTA-01 reactimeter which has 10 reactivity measurement ranges within limits of 0.01 to 5 betas. The reactimeter is used to monitor the neutron flux (output) of the reactor along with a gauge at the operator's panel with a scale switch and a special recorder on the operator's board is used to record reactivity. The reactimeter channel operates according to signals from two KNK-53M ionization chambers.

2.8.3. Automatic Reactor Power Control

The system includes three identical sets of AR automatic average reactor power controllers. Each AR set includes four ionization chambers situated around the reactor, and the information from these chambers is used to move four AR control rods in sync. A control signal is generated by means of adding the relative deviations of power from a specified level generated in four separate measurement channels. This principle makes it possible to keep the automatic controller functional if one instrument channel or its instruments should fail.

The instrumentation of all three automatic controller sets is the same.

The use of measurement channels of different sensitivities makes it possible for these sets to operate in different ranges, namely a low power range of 0.5 to 10% nominal and a working power range of 5 to 100% nominal. The low power range includes one ARM automatic controller (ZAR);

in the working power range there are two automatic controllers (1AR and 2AR).

The sensor and part of each automatic controller measurement channel are also used as a power overshoot safeguard channel: four AZMM channels in the low power range and eight AZM channels in the working range respectively.

A diagram of the control and safety system is given in Figure 2.11.

The signals of the sensors in each channel are correlated by means of a KrT.5 corrector. The corrected signal is compared with a reference signal from a Zd.M.5 output set point device which is common for one set of four channels. A discrepancy signal proceeds to a UZM.11 power emergency safeguard amplifier and a USO.10 deviation signal amplifier. If a discrepancy signal should reach the warning and alarm set points, the UZM.11 amplifier will generate warning and alarm signals respectively for further processing in the safeguard logic circuitry. In the USO.10 amplifier, whose gain is controlled by a power set point device, a relative power deviation signal is generated. Information on power deviations at places monitored by the sensors is displayed on a discrepancy indicator on the control panel and to a certain extent makes it possible for the operator to monitor energy release differentials in the reactor. The output signals of the USO.10 amplifiers of the four channels are summed in a USM.12 amplifier, and data on average power deviations are transmitted to an automatic controller activation indicator on the control panel. Signals proceed from the output of the summing amplifier to a control rod synchronization system employed to make the movement of the control rods synchronous. The instruments of the synchronization system generate relay power control actions. The synchronization system generates a signal of the average position of the rods of a given automatic controller and signals of the deviations of the positions of specific control rods from the average.

The average power deviation signal (from the output of the USM.12 amplifier) and rod deviation signals are used to generate commands to remove and insert automatic control rods into the zone. These signals are transmitted via BKS.40 power control units to control the servodrive of a control rod.

One of the working range controllers is activated, while the other is on "hot standby". This controller will be activated automatically in the event that the first controller shuts off automatically due to malfunctions. In order to ensure "shock free" activation of the backup controller (no rod movement), a KrU.4 automatic corrector is used to maintain zero unbalance at the output of its summing amplifier.

The control and safety system provides for identical set point values from the power set point devices in the working range with an accuracy of at least 0.5% N nominal. The set point values of the sensor are kept in sync by means of a BSP.36 unit and a logic circuit on the basis of the principle of stopping the set point device whose set point value has run ahead in the direction of the variation of the set point values.

The operator can control the set point values of the set point devices by means of keys on his control panel. The working rate of change of set point values is no greater than:

0.0075% of N nominal per second in the 0.5 to 1% N nominal range;

0.0125% N nominal per second in the 1 to 6% N nominal range;

0.15% of N nominal per second in the 5 to 20% N nominal range;

0.25% of N nominal per second in the 20 to 100% nominal range;

Under emergency conditions the set values of the working set point devices will decrease automatically at a rate of 2% N nominal per second. A button on the operator's panel may be used to reduce set point values in an emergency.

The automatic controllers maintain reactor power with an accuracy of at least ± 1 relative to a specific level in the 20 to 100% N nominal range and of at least ± 3% in the 0.5 to 20% N nominal range.

In addition to the functional monitoring incorporated in certain units of the system, the functioning of the measurement channels of the working range automatic controllers , including the neutron flux sensors, is continuously monitored. The BT.37 unit compares the output signals of similar channels, and the comparison is made with the signals of channels adjacent with respect to the locations of the sensors around the reactor. If the signal of a channel differs from those of both adjacent channels by an amount which exceeds possible differences in the reactor, the channel in question will be considered malfunctioning by the logic circuitry. This type of monitoring is employed at steady power levels and is automatically halted in accident safety modes and transient operation modes.

In the operation of the working range automatic controllers, the ARM rods may be involved in an overcompensation mode of an activated controller. In this case, when the rods of an activated controller move to an intermediate terminus corresponding to 75 to 100% insertion of the rods, the ARM rods will automatically move downward, and when the first rods move to an intermediate terminus corresponding to 25 to 0% insertion, the ARM rods will automatically move up.

The LAR-LAZ, or local automatic control and local safety system, is included to stabilize the distribution of energy release in the reactor. The LAR system is designed on the basis of the principle of independent control of power in twelve local reactor zones by means of twelve control rods. Data from two KTV.17 chambers situated in the active zone of the LAR rod at a distance of 0.63 meters from the rod are used to control LAR rods.

A KTV.17 chamber is a current ionization chamber with a coating of sensitive elements with a composition of U235 whose design includes a protective electrode for the purpose of reducing legitimate signal leaks. Negative voltage is applied to the collector electrode from a BP.119 power supply.

Voltage of the same amplitude and polarity as that applied to the central collector electrode is applied to the protective electrode. In the process the protector and collector electrodes are under the same potential and leakage currents are minimized. The KTV.17 chamber has three sensitive elements distributed over the height of the active zone.

The LAR is put into an automatic mode in the power generation range after the required distribution of energy release has been ensured on the basis of information from the SFKRE system. Prior to activation, the output signals from the LAR zones are compensated by means of correctors included in the system. Subsequently, the LAR, by maintaining specific power values in each of twelve zones prior to activation, stabilizes energy distribution in the reactor. The LAR system maintains overall power with an accuracy at least as good as that of a traditional medium power automatic control system. The LAR system also has significant advantages in transient modes, because it not only makes it possible to vary and control overall power, but also eliminates power unbalances caused by local technical perturbations.

At present, the LAR is the basic automatic power control system in the 10 to 100% N nominal generation range. A automatic average power control system is used as a backup system which is automatically activated in the event of failure of the LAR.

The LAR system, which consists of 12 physically independent local controllers, is characterized by a high level of viability. In the event that several zones malfunction, the system as a whole will remain operable.

Signals from each chamber are corrected by means of a KT current corrector. After it leaves the corrector, part of the signal is sent to the LAZ (local emergency safety system) channel, which generates power overshoot alarm and warning signals; part of the signals from each of two chambers of an LAR zone are summed in a USO amplifier, which generates a signal of the relative deviation of power in an LAR zone. If this deviation exceeds specific levels, the flip flop generates signals to move the LAR control rod of the zone in question.

The rate of movement of the LAR control rods has been reduced to 0.2 meters per second for the purpose of not exceeding the maximum permissible rates of introduction of positive reactivity (Nuclear Safety Regulations) when twelve LAR rods are moved simultaneously (0.07 β_{eff} ~~betaeff~~/s).

A feature which limits the continuous extraction of automatic control rods to eight seconds has been incorporated in the design.

In the event that a power overshoot warning signal appears in one of the LAZ channels of a zone, removal of the LAR rod is automatically prevented. In the event that power overshoot alarm signals appear in both channels of an LAZ zone, two LAZ rods are inserted into the active zone until at least one AZ signal disappears. In the process the average output of the reactor is reduced by means of automatic reduction of the set point values of the power set point devices at the working rate of change.

The withdrawal of more than eight to ten RR-AZ or USP rods in the event of any malfunction (operator panel, control and safety system logic, SP control power units, and so forth) is prevented by a "power lock" system. The power lock system automatically determines the number of rods to whose power control armature circuits has been applied voltage to remove the rods. If this number is greater than 8 to 10, the circuit from the control power source is automatically broken, and no rods may be removed from the zone. There are three power lock channels which operate on the principle of "2 out of 3" voting logic.

Emergency Protection of the Reactor

Emergency protection of the reactor is effected by the automatic insertion of all (except for the USP) absorber rods from any initial vertical position into the active zone.

24 control and safety rods evenly distributed over the reactor are selected from the RR-AZ rods and put into an emergency protection mode by means of a special selection circuitry included in the logic bays. In the process of starting up the reactor, 24 safety rods are the first to be raised to VK; the removal of any other rods from the active zone is prohibited prior to the raising of the safety rods; and the raising of the safety rods is automatically monitored and signalled.

The reliability of accident safeguards and the reliability of manual control of the reactor are ensured by 6 independent groups of 30 to 36 control rods evenly distributed over the reactor. Each control and safety rod is moved by its own servodrive controlled by a separate power and logic unit. The rods are combined into six groups on the basis of providing power supplies for the servodrives and control units and the layout of the control units. The failure of one or even several servodrives is insignificant when there are a total of 187 present. The failure of several independent groups is practically impossible. Because every control and safety rod is surrounded by rods of other groups, a "sick" rod will always have several "healthy" neighbors.

The design of the actuator mechanisms of the control and safety system ensures automatic insertion of all (except for the USP) rods into the active zone in the event of power failure. The reliability of emergency protection is ensured by functional redundancy (redundant monitoring channels) for each parameter and hardware redundancy (redundant signal processing channels).

Due to the fact that nuclear power plants with RBMK reactors make such a great contribution to total power supplies and that down time must be minimized, the designers of accident safety systems took a differential approach to emergency situations in the reactor and generation unit. There are several categories of emergency protection, depending on the nature of the situation:

emergency protection with total reactor shutdown: AZ-5 emergency protection until the emergency is taken care off: AZ-5

preventive controlled reduction of reactor output at an increased rate to safe levels: AZ-3, AZ-2, AZ-1; safe levels of output for different emergency situations and the rates of preventive output reduction have been calculated and determined experimentally.

The most serious accident safety condition , AZ-5, involves the insertion of all the control rods (except for the USP) into the active zone to the bottom ends. This condition occurs if:

output increases to 10% more than nominal;

a decrease in the period to 10 seconds, a reduction and increase in level in the BS of any half;

a reduction in feed water flow

an increase in pressure in the BS of any half

an increase in pressure in the PPB room, BS, or NVK

an increase in pressure in the reactor space

a reduction in the level of the control and safety system cooling tank

a reduction in water flow into the control and safety system channels

shutoff of two turbogenerators or one turbogenerator operating alone

the shutoff of three or four operating main circulation pumps in any pump station

if voltage disappears in the inhouse power supply or if conditions AZ-1, AZ-2, or AZ-3 have been declared and cannot be handled either by means of the controls (AZ-5 button, KOM switch) on the operator panels and a number of other power plant rooms

In the event of an emergency power overshoot recorded by side measurement channels, a partial AZ-5 condition is declared in which control rods are no longer inserted in the active zone once the original reasons for the condition have disappeared (that is, when power has returned to the proper level). This makes it possible to keep the generator unit in operation if the power overshoot signals were due to power unbalances and the emergency situation can be corrected by a quick partial reduction in total reactor output. The same applies to transient modes of reactor operation and when significant local perturbations occur. A partial AZ-5 condition is short term; if a large number of control rods are inserted into the zone during the condition, the reactor will be completely shut down, as in an AZ-5 condition.

An AZ-3 condition is declared in the event of an accidental load shedding by two turbogenerators or one operating alone.

An AZ-2 condition is declared (reduction in N to 50%) in the event that:

one of two turbogenerators shuts down

one of two turbogenerators accidentally sheds its load

An AZ-1 condition (reduction in N to 60%) is declared if:

one of three operating main circulation pumps in any pump station shuts down

there is a reduction in water flow in the KMPTs

there is a reduction in feed water flow

there is a reduction in the water level in the BS

when the switch to close all the throttle and control valves of the DRK is thrown

In conditions AZ-1, 2, and 3 reactor output is automatically reduced at a rate of 2% nominal to levels of 60, 50, and 20% by the engaged automatic power control system. Emergency rates of power reduction are provided and the operation of the reactor is stabilized at a safe output level by means of automatic engagement of additional control rods in an emergency protection mode. Signals which initiate conditions AZ-1, 2, and 3 and AZ-5 are for technical reasons generated in the process automation system.

The generation of an emergency protection signal with respect to any parameter takes place when two or more sensors of the four installed operate. For technical reasons, the logic component of emergency protection consists of two independent sets of hardware. Individual switches for each parameter are installed to take off the protection during tests. The introduction of protection is indicated and recorded by the SKALA STsK. The safety system includes alarms which announce the operation of the system, the reasons for its activation, and safety equipment malfunctions.

Figure 2.13 illustrates the structure of the system on the basis of process parameters.

AZ-3, 2, and 1 conditions, if reactor output is greater than the safe level for these situations, and the actuator algorithm for conditions AZ-5, 3, 2, and 1 are implemented in the control and safety system's logic.

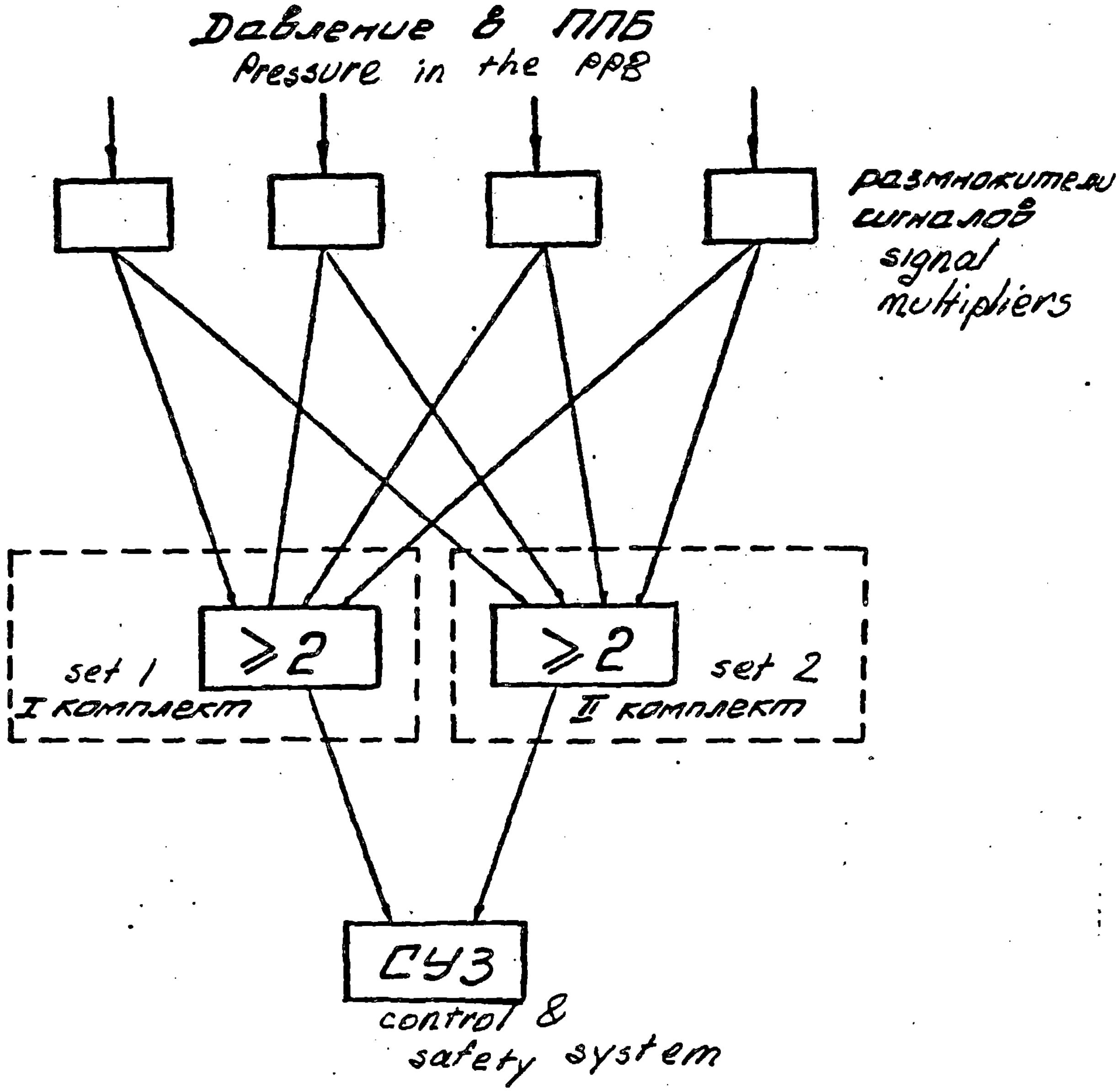
The reliability of the accident safety system with respect to excessive increases in nuclear power plants and reactor power and declaring condition AZ-5 is ensured by functional redundancy (the presence of at least three monitoring channels with its own sensors for each channel) and hardware redundancy (parallel processing of digital signals by several independent channels).

Condition AZ-5 is declared for when reactor rideup time decreases to below 10 seconds and is recorded by at least two channels out of three:

In the AZSP startup range from 4 times 10exp-7 to 5 times 10exp-2 N nominal;

in the ASZP working range from 10exp-5 to 1.2 N nominal.

Each power increase rate emergency protection system channel consists of AZS.13 rate emergency protection amplifiers with a separate KV.2 log taking scale and a KNK-56 current ionization chamber with a lead shield on the channel at which it is installed (in the startup range) and a KNK-53M current ionization chamber (in the working range).



Показана схема АЗ по повышению давления в ППБ.
Схема АЗ по другим технологическим параметрам аналогична.

Figure 2.13 Diagram of a Process Parameter Safety System

Рис.2.13 Структурная схема АЗ по технологическим параметрам

An AZ-5 condition is declared for power overshoot protection: in the low power range: from 0.005 to 0.1 N nominal, when power exceeds nominal by 0.5% and is recorded by at least two power overshoot protection channels out of four; in the working power range: from 0.06 to 1.2 nominal, when the power overshoot is 10% of nominal and is recorded by two of eight emergency power protection channels; in the process there should be an emergency signal in at least one channel of each of two groups consisting of four emergency power safety channels.

Each power safety channel consists of a UZM.11 power safety amplifier;

ionization chamber: a KNK-56 in the AZMM channel and a KNK-53M in the AZM channel;

a BP.39 measurement channel power supply;

a KrT.5 chamber current corrector

A group of four power safety channels has a common ZdM.5 power set point device; one set point device in the AZMM (low power range and two set point devices in the AZM (working power range). The ionization chamber, chamber power supply, chamber current corrector, and power set point device are at the same time part of the measurement channel of the automatic controller of a corresponding range.

The presence of eight power safety channels in the generation range with sensors evenly distributed around the active zone makes it possible to provide protection from overall power overshoots and monitor and protect the reactor from local power overshoots.

A coincidence circuit for signals from two independent groups (of four power safety channels) with an alternating sensor configuration reduces the probability of unjustified reactor shutdowns in the event of a malfunction in one channel or malfunction of the common component of a group, namely the power set point device. In the process dangerous malfunctions of the emergency safety system are prevented by the fact that the measurement and logic components of the system were designed on the following principle: any malfunction of a unit or channel is equivalent to an emergency signal in this channel.

This design makes it possible to replace any unit in a channel for repairs or maintenance when the reactor is generating power, which is particularly important for RBMK reactors operating in a continuous fuel transfer mode.

The power overshoot protection system is always ready to operate, while the low power protection system can be stopped by the operator by means of a switch on the panel after the end of its working range has been reached.

Preventive reductions in power are effected by an activated automatic control system, namely the local automatic controller, 1AR or 2AR, or by means of an automatic reduction in the set point values of the power set point devices upon AZ-3,2, or 1 signals.

When the set point value of a set point device is reduced, power unbalance (deviation) signals are generated in the measurement component of the automatic controller. When an activated automatic controller generates unbalance signals of plus or minus 1%, the rods of this controller will move, while the PK-AZ rods will move upon unbalance signals of plus or minus 2.5%. Initially two groups of six rods will move downward, and then, when they reach the lower terminus, the corresponding rods of the next two PK-AZ groups will move downward upon an unbalance signal of plus or minus 2.5%. Only one group of six PK-AZ rods will move up. Plus or minus 2.5% signals are generated in the KrU.4 unit of the average power automatic controller upon signals from the summing amplifier.

If a preventive AZ-3,2, 1 power reduction is effected by an activated local automatic controller, relative unbalance signals of plus or minus 2% generated in the flip flop unit of the local automatic controller will cause the local safety rods of the appropriate local automatic control zone to move. Removal of the local safety rods from the zone is permitted only after the local automatic control rods have been removed to the VK.

If the set point value of the set point device of an activated automatic controller does not drop at an emergency rate or no automatic controller has been activated or if an automatic controller shuts off in the process of reducing power and an alternate has not been activated, AZ-3,2, and 1 conditions automatically turn into AZ-5 conditions.

2.9 The Reactor's Process Monitoring System

The reactor's process monitoring system provides the operator with visual and written information on the values of the parameters which define the operation of the reactor and the condition of its parts, namely process channels, control channels, reflector cooling channels, the graphite stacking, metal structures, and so forth.

The process monitoring system includes the following systems:

--individual channel monitoring of the flow of heat transfer agent through process and control channels

--temperature monitoring of the graphite stacking and metal structures

--monitoring the integrity of the channels on the basis of the temperature and moisture content of the gas flowing over them on the outside

--physical monitoring of energy release (SFKRE)

--the "Skala" centralized monitoring system

Data from KIP systems is gathered and processed by the Skala centralized monitoring system, while information on the most critical parameters is gathered and processed by individual instruments or self contained systems (the KtsTK, SFKRE, KGO).

The extent of monitoring in the reactor is as follows:

--measurements of process channel flows: 1661 check points

--flow measurements in control channels: 227 check points

--measurements of in metal structure and biological shield temperatures: 381 points

--measurements of the graphite stacking and blocks: 46 points

--measurements of energy release by radius and height: 214 points

--measurements of gas temperature: 2044 points

--measurements of heat transfer agent radioactivity: 1661 points

2.9.1. HEAT TRANSFER AGENT FLOW MONITORING

Flow in all reactor channels is measured by means of tachometric ball flow meters. The flow meter includes a primary ball transducer, a magnetic induction transducer, and transistor components. Flow meters with a measurement range of up to 50 cubic meters per hour are used to measure flow in process channels, while meters with ranges of up to 8 cubic meters per hour are used in the control channels.

Process channel flow meters measure the flow of a medium with a temperature of 20 to 285C and a pressure of up to 10 megapascals, while control channel flow meters measure medium with temperatures of 20 to 80C and pressures of 5 megapascals.

The basic error of the flow meters is 1.5%. The flow meters have a positive systematic error due to temperature, which is determined by a high temperature flow meter tester and is adjusted for automatically by means of introducing appropriate correction factors to measurement results in the Skala system.

The lag of the flow meter does not exceed 6 seconds.

Heat transfer agent flow in each process and control channel is monitored by a computer system. The values of channel flows are compared with set point values specified on the basis of the characteristics of the channels and their location in the reactor. The set point values may be changed if there is a change in the operational mode of the reactor.

If heat transfer agent flow goes outside the limits defined by the set point values, the computer system will issue a shutoff signal to the channel mimic panel and the group shutoff mimic panel; a shutoff notice will appear on the teletype, and the control and safety system will be locked on if water flow in the control channel is below limits.

The primary and magnetic induction transducers should be diagnosed on a regular basis and decisions should be made as to whether they may be used further or replaced on a preventive basis.

The primary transducer should be diagnosed by taking oscillographs of signals from the magnetic induction transducer, determining signal amplitude and period ratios, and comparing the values of the ratios with set criteria. The magnetic induction transducer is diagnosed periodically by means of checking the resistance of its magnetic coil.

2.9.2 TEMPERATURE MONITORING

The temperatures of the graphite stacking and metal structures are monitored by means of commercially manufactured KhA thermoelectric cable transducers.

A thermocouple assembly (BT) is used to monitor the temperature of the graphite stacking and the upper and lower metal plates.

Thermocouple assemblies are placed on the longitudinal and transverse axes of the reactor in 17 cells situated at the joints of the corners of the graphite blocks. 12 trizonal assemblies are used to measure the temperature of the graphite stacking (four of which are in the reflector), while 5 bizonal assemblies are used to measure the temperature of the base and upper shielding plates.

A thermocouple assembly includes thermoelectric cable transducers and a bearing structure consisting of a biological shielding plug, graphite sleeves, and connecting tubes. See Figure 2.36.

In the trizonal assemblies the working junctions of the thermoelectric transducers are situated in the central section of the active zone and 2800 millimeters below and 2700 millimeters above the central zone. The transducer is built around a cable with an outside diameter of 4.6 millimeters with a sheath made of a carbonization resistant high nickel alloy. The cable has four strands with magnesial insulation, and has two chromel and 2 alumel thermoelectrodes formed into a single working junction. Thus, each thermoelectric transducer includes two thermocouples with a common working junction.

The systematic component of graphite stacking temperature measurement error due to internal heat release in the elements of the thermocouple assemblies may reach 2.2% of measured temperature and is taken into account by means of adjustments to measurement results in the Skala system.

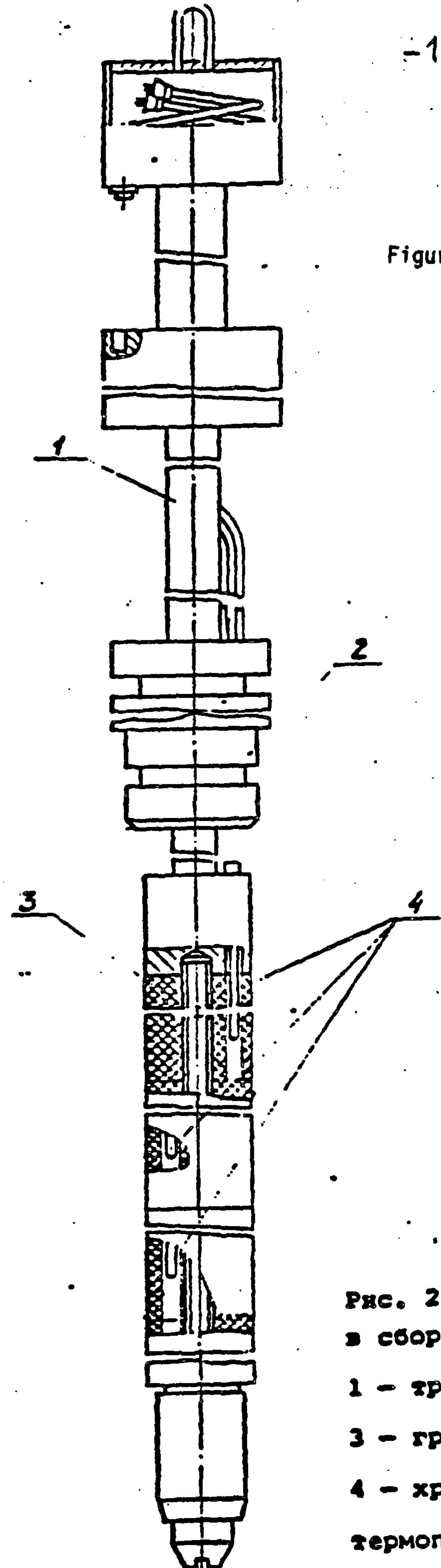


Figure 2.36. A thermocouple assembly

Key:
1. tube
2. rod
3. graphite sleeve
4. chromel-alumel thermoelectric transducer

Рис. 2.36. Блок термопар в сборе:

1 – труба, 2 – стержень,

3 – графитовая втулка,

4 – хромель-алюмелевый термопреобразователь

The heat lag of the assembly is acceptable and amounts to 90 seconds, which is much lower than the heat lag of the graphite stacking, which sometimes reaches 30 to 40 minutes.

In bizonal thermcouple assemblies the working junctions of the thermocouples are situated at the levels of the upper and lower plates.

The temperatures of other metal structures are monitored by means of cable chromel alumel thermoelectric transducers built around a thermocouple cable 4 millimeters in diameter and inserted into sealed steel sleeves (Figure 2.37). The sleeves both protect the thermocouples and serve as guide elements in monitoring hard to reach places. The temperatures of metal structures are monitored for the purpose of determining their condition under static and transient conditions.

In the upper and lower metal structures, which are very complex structures which include a large number of structural members under high temperature stresses, there may be as many as 30 checkpoints. The outside surfaces of the process and control channels, the edges of the structural members, roller supports, expansion joints, and upper and lower plates are monitored.

The temperature of the reactor vessel is monitored on a single generatrix at four points by height. The bearing metal structure is monitored at six points on a single radius. The bottom sides of the beams in the ceiling of the central room are monitored at eight points.

In addition, the temperature of the water in the biological shielding tanks is monitored by means of chromel alumel thermoelectric transducers with heads at 16 points (Figure 2.38).

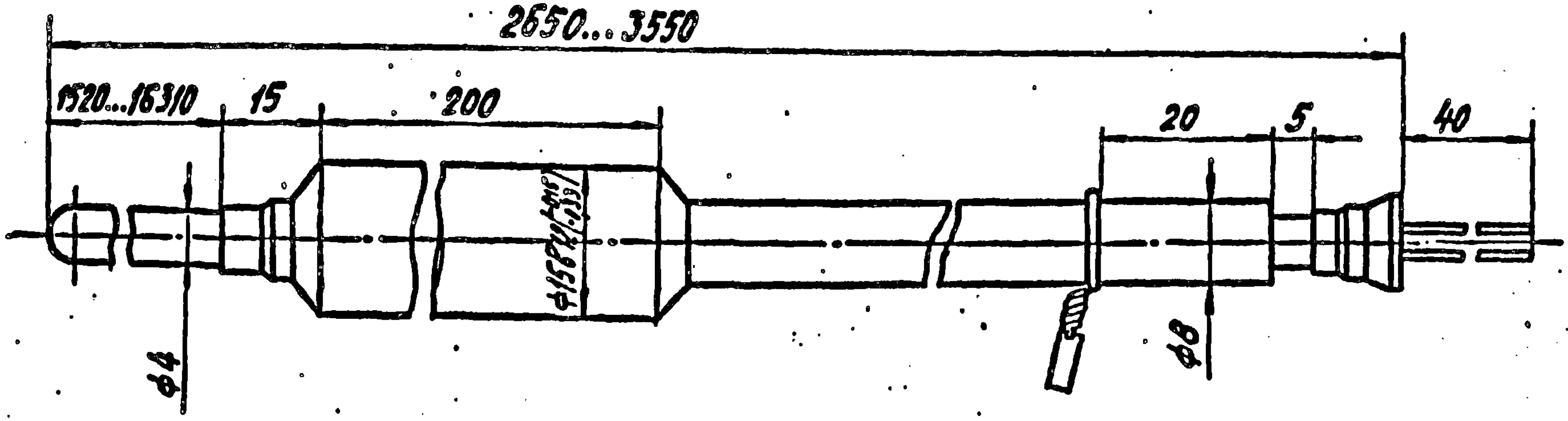


Figure 2.37 A dimensional drawing of a chromel alumel cable thermocouple

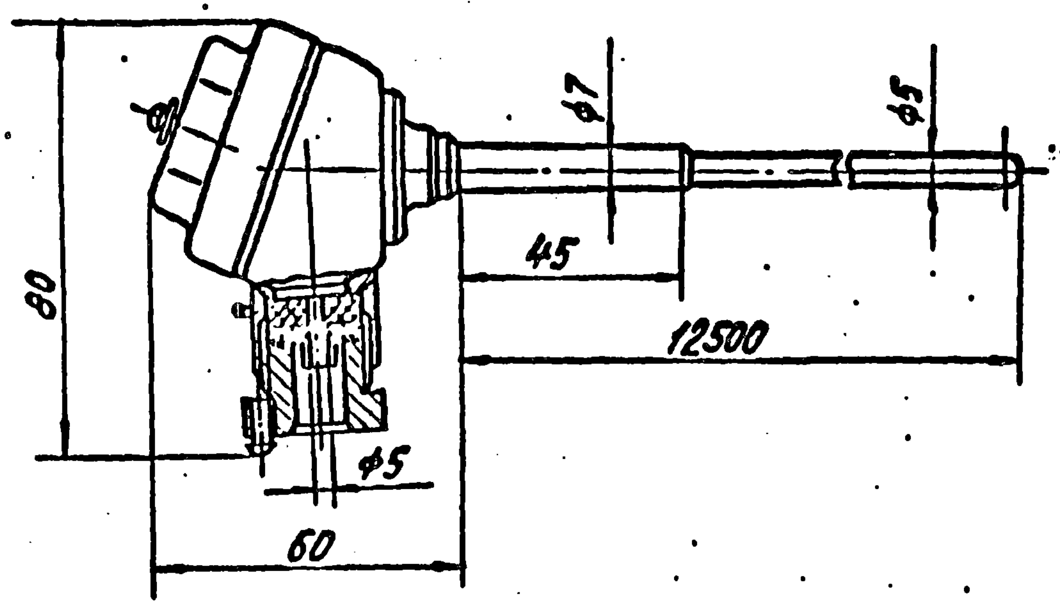


Figure 2.38 A dimensional drawing of a chromel alumel thermocouple with a head

Six cable chromel alumel thermoelectric transducers installed at benchmarks are used to monitor water temperature at the control channel drain.

156 chromel alumel thermoelectric transducers are used to monitor water temperature in the reflector cooling channels.

The temperature instruments used are classified as low lag instruments; the heat lag of cable thermocouples is approximately 5 seconds, but when they are installed in an additional protective sleeve for measuring the temperature of metal bearing structures, the heat lag is approximately 60 seconds.

Instrument error is approximately 2% of the measurement range.

Temperature information is periodically printed out by the SKALA system and any parameter may be called for by code on the SKALA's digital display and backup group instruments.

2.9.3. PROCESS CHANNEL INTEGRITY MONITORING

The process channel integrity monitoring system (KTsTK) is part of the reactor ventilation system and is generally designed to solve the following problems:

find leaky reactor channels

ventilate the reactor space

A diagram of the process channel integrity monitoring system is given in Figure 2.39.

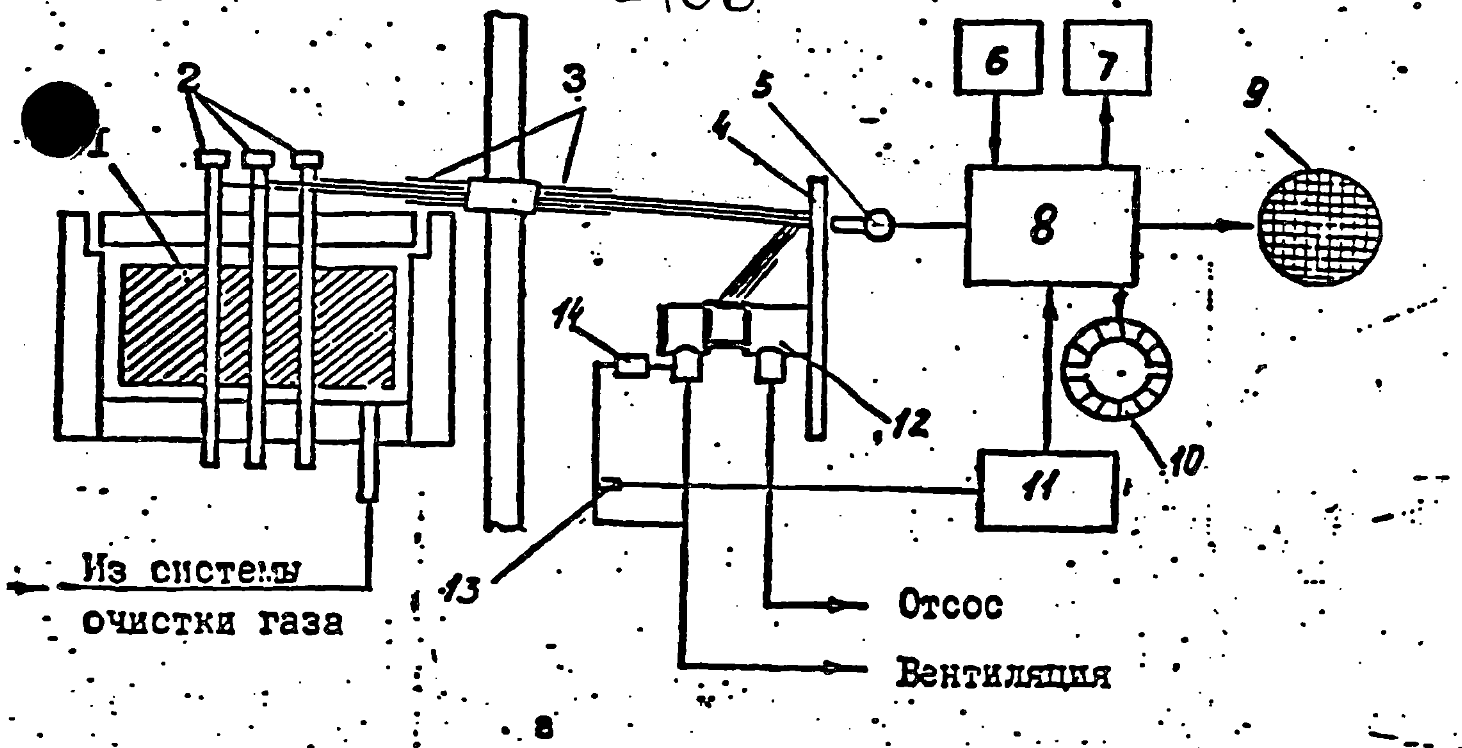
The operation of a process channel integrity monitoring system is based on measuring the parameters of gas (temperature and moisture content) as it circulates through the graphite stacking in gas lines formed by the graphite stacking and the process channels.

Temperature is individually monitored, while moisture content is collectively monitored.

Temperature is measured by means of short chromel copel thermoelectric transducers mounted on each pulse tube of the system.

Information on temperature signals is transmitted from the thermocouples to the SKALA system for processing, which in turn, after; determining the channel or group of channels in which temperature exceeds the set point, transmits a signal to a channel mimic panel located on the generation unit control board.

Moisture alarms are used to monitor moisture in 26 zones and determine wet zones. The moisture alarm consists of 8 moisture sensors and an eight channel moisture measurement unit. The sorption type sensitive element of the moisture sensor is designed to operate at temperatures of 40 to 100C and relative humidities of 50 to 100%.



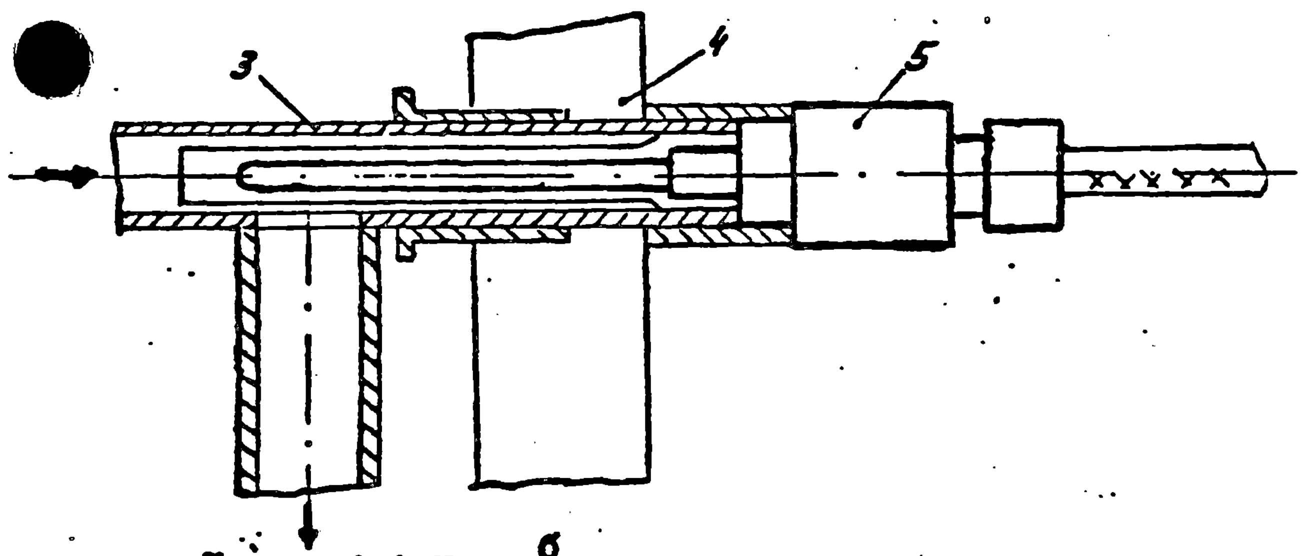


Figure 2.39. The process channel integrity monitoring system (a) and installation of the chromel copel thermoelectric transducer (b) Key: 1. reactor 2. process channel 3. pulse tubes 4. panel 5. transducer 6. call device 7. digital reader 8. Skala system 9. channel mimic panel 10. moisture panel 11. moisture alarm 12. group valve 13. moisture sensor 14. gas blower

Relative humidity can be set at intervals of 5%. When the moisture alarm operates, it transmits signals to the SKALA system, which displays them on a moisture panel located on the unit control board.

A hygrometer which includes a sorption type primary transducer, a measurement unit, and a recorder, is used to continuously monitor gas moisture content in the reactor space over a range of 0 to 100%.

The bellow cavities of the control and safety system channels are drained and the temperature of the drainage piping is measured in order to improve the reliability of determining process and control channel integrity.

The moisture which appears in the reactor space during a leak will evaporate, and by condensing on the nearest "cold" control channels, will partially settle in the bellow cavity and later run down into the drainage piping. In the process the temperature of the drainage piping, which runs through the bottom water line room and has the same temperature if there are no leaks, will drop, a fact which will be recorded by a thermocouple.

Where to look for leaks is determined by the readings of the thermocouples, which yield temperature values approximately 100C lower than the temperature of the bottom water line room.

Temperature measurements are taken on 126 control channel drain lines and are periodically printed out by the SKALA system.

2.9.4. MONITORING THE FUEL CAN FOR LEAKS

The design philosophy and physical characteristics of a nuclear power plant with RBMK reactors (channel type reactors with boiling heat transfer agents) determine the structure of the system for identifying and locating leaky fuel cans during the operation of the reactor.

Fuel cans are checked for leaks by a system which includes:

--a sample taking system for monitoring the activity of gaseous fission products in the separated steam of each separator drum, which makes it possible to make continuous observations of the condition of the fuel elements of the fourth part of the fuel element assembly in the active zone.

--a non-sample channel by channel system for periodically checking the total gamma activity of the heat transfer agent in each steam and water line, the secondary electronic component of which is used to compensate for the signal background in order to isolate the contribution of the gamma activity of fission products leaking out of leaky fuel elements.

2.9.5. MONITORING THE MULTIPLE FORCED CIRCULATION LOOP

The multiple forced circulation loop is monitored to determine its condition and the operation of its basic components, namely the separator drums, the main circulation pumps, and the intake and pressure collectors. Monitoring involves checking level and pressure in the BS, the temperature of the metal of the BS, the surge tanks, the flow and pressure of the main circulation pumps, and the flow of steam from the drum and feed water into the drum.

Platinum resistance thermometers which measure the temperature of the heat transfer agent are used to determine the cavitation condition of the intake collectors. Pressure is monitored in the intake and pressure collectors of the main circulation pumps.

The flow of water through the main circulation pumps is measured by means of a differential pressure gauge with a constrictor to generate a pressure differential. The parameters of the multiple forced circulation loop are monitored by the SKALA system.

2.9.6. THE SKALA CENTRALIZED MONITORING SYSTEM

The SKALA integrated automated centralized monitoring system is designed to monitor the primary equipment of the generation units of nuclear power plants with RBMK-1000 reactors and to perform calculations and logic analysis of the operating conditions of the units and provide processed information to plant operators and staff. Figure 2.40 illustrates the structure of the system and its ties to other systems (the control and safety system, the process channel integrity monitoring system, and so forth).

The primary system is a two processor computer system which can receive information from the plant and output information to displays by means of any of its processors (functional redundancy).

Information on the condition of the generation unit is transmitted from process monitoring system sensors via individual channels or VK by the operator to reading and digital instruments, mimic panels, channel mimic panels, and individual deviation panels and is recorded by recorders, teletypes, and fast printers. An operator can get the information he needs from the system by means of a set of input and output devices. The system has its own control system to manage its operation.

The basic technical characteristics of the system are given below:

1. extent of monitoring: 7200 analog signals and 6500 digital signals

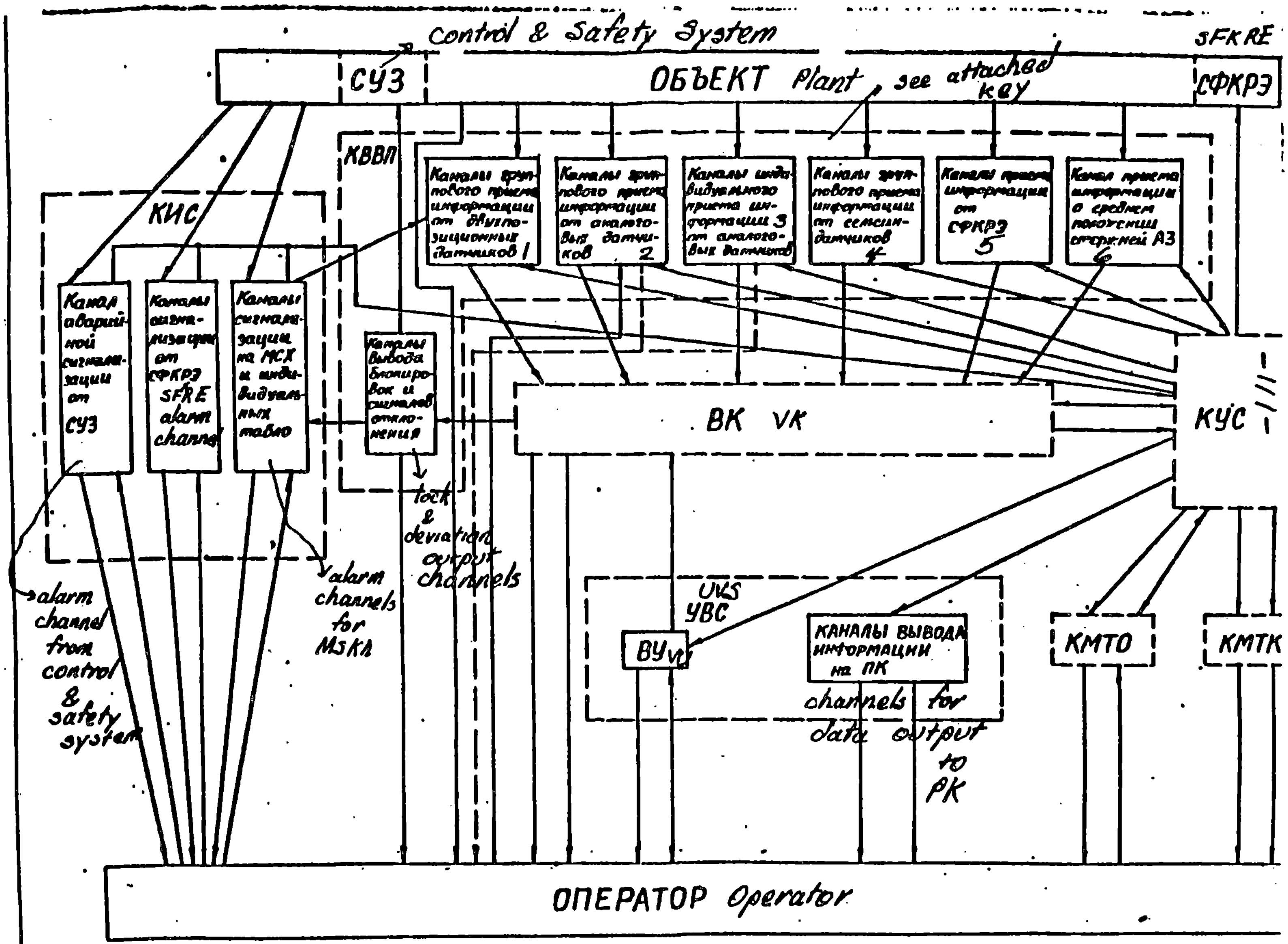


Рис. 2.40 Структурная схема системы „Скала"
Diagram of the SKALA System

Notes 1-6
Page 111a

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1. channels for collective reception of information from flip flop sensors
2. channels for collective reception of information from analog sensors
3. channels for individual reception of data from analog sensors
4. channels for group reception of data from synchro sensors
5. channels for receiving data from SFKRE
6. channel for receiving data on average control rod position

signals received: from chromel aluminum and chromel copel thermoelectric

platinum and copper resistance thermometers

tachometric ball flow meters

sensors: differential pressure gauges with a standardized 0 to 5 milliamp

output, synchro sensors, flip flop sensors, the SFKRE independent system,

and signals of the average position of the control rods

2. Monitoring periods:

mass parameters 1 to 5 minutes

analysis parameters 30 minutes

Functions:

Measuring parameters input via collective and individual data reception channels and upon calls from staff to collective, individual, and digital reading instruments.

Indicating the condition of machinery, valves, aggregates, and process parameters on mimic panels, MTK, collective deviation panels, and control and safety system panels.

Monitoring the deviations of directly measured and calculated parameters and outputting the results to data displays and recording them. Periodic process calculations at the request of the staff.

Periodic printout of any measured or calculated process parameters at the request of the staff and recording the prehistory and development of an emergency.

4. Mean time until failure:

monitoring functions 1 times 10exp4 hours

calculation functions 2 times 10exp3 hours

5. power consumption 95 kilowatts

-ENERGY RELEASE (SFKRE)

The Purpose and Composition of the SKFRE

The SKFRE is designed to measure and record signals from energy release monitoring detectors which characterize energy release in the reactor. By providing primary processing of signals from the energy release detectors and comparing them with preset limit values, the SFKRE gives recommendations to the operator for energy release distribution control. The light and sound alarms issued by the SFKRE are used to balance energy release distribution. The interface between the SFKRE and the computer of the SKALA system is used to provide additional adjustment of energy release. The computer uses signals from the energy release detectors, the results of physical analyses, and other necessary information for periodic calculations and recordings of power and safety factors for each fuel element canister and to calculate a number of other canister and reactor parameters.

The SFKRE's recording potentiometer is used for online monitoring of the thermal output of a reactor from the minimum monitorable level to nominal. The potentiometer records the total current from the energy release detectors over the radius of the reactor and has a scale calibrated in megawatts (a scale of 0 to 4000 megawatts). A backup instrument is used for the same purpose.

With respect to functional purpose the SFKRE may be divided into three systems: a system for physical monitoring of the distribution of energy release over the radius of the reactor, a system for physical monitoring of the distribution of energy release over the height of the reactor, and an auxiliary system for periodic followup calibration of the detectors.

The SFKRER is designed to measure and record the signals of 130 intrazone detectors for monitoring energy release over the radius of the reactor, preliminary processing of these signals, transmission to the computer of the SKALA system, comparing the signals with three set levels, and providing light and sound alarms when energy release in the canisters equipped with detectors goes off limits. The limit values of the output of canisters with energy release detectors are determined by the computer of the SKALA system so as to balance energy release and ensure the safety of the canister in question and its neighbors.

The SFKREV is designed to measure and record signals from 12 intrazone seven section detectors for monitoring energy release over the height of the reactor, for preliminary processing of these signals, transmitting them to the SKALA computer, comparing the signals with three set levels, and providing light and sound alarms when local energy release levels in canisters adjacent to the detectors go off limits. Limit signal values for individual sections are determined by the SKALA computer so as to stabilize axial distributions of energy release and ensure the safe operation of the canisters without going over maximum permissible local heat loads.

The auxiliary system (SPD) is designed for periodic followup calibration of the energy release detectors and to determine fuel can output calculation errors for the SKALA computer.

The SFKRER includes 130 radial energy release detectors, energy release monitoring instruments, a recording potentiometer (SKFRE power recorder), a backup reading instrument, and instrument displays.

The SFKREV includes 12 seven section detectors for monitoring energy release over the height of the reactor, energy release instruments, and energy release instrument displays.

The instruments are modular in design and are serviced by one common multichannel recorder which utilizes a digital recorder to record detector signal values exceeding limit values, overshoot time, and the coordinates of the detectors.

Figure 2.2a illustrates the configuration of the radial and height energy release detectors and control and safety system rods used to monitor and control energy release in the reactor. For the purpose of monitoring and controlling energy release distribution, the reactor's designers included about 310 mounts and containers with dry bearing tubes (sleeves) in their centers. 130 of them are designed to hold radial energy release detectors, 48 are designed to hold local automatic control and safety system detectors, and at least 130 are left free (power scanning containers) and are used for periodic followup calibration of the detectors. These containers are placed next to the containers with the radial detectors.

The instruments and multichannel recorder are placed in the unit control board. The energy release light alarms are located on the SUZ-SFKRE mimic panel on the operator's board. A unit which activates the light alarms on the mimic panel on commands from the energy release monitoring instruments is included in the SKALA system.

The SFKRE's power recorder is located on the operator's board, and a backup instrument is located on the operator's panel. The energy release monitoring displays are located on the operator's board.

The SPD includes DKER type calibration detectors, triaxial calibration fission chambers, a ring ionization chamber, and instruments.

Calibration detectors are moved in and out of the reactor by means of the crane in the central room. The instruments may be located in the central room or the crane operator's room in the central room.

Radial Energy Release Detectors

Radial energy release detectors are kept in dry central zirconium bearing sleeves with inside diameters in the active zone of the reactor of 6.5 millimeters located along the axis of a container (over their entire length). The design of a radial detector is illustrated in Figure 2.4.1. A radial detector consists of a sensitive element in a sealed housing 4 made of corrosion resistant steel with an outside diameter of 6 millimeters, a sealed connector assembly, a cable line in a sealed protective housing, and biological shielding components. The housing of the detector is filled with an inert gas (argon) to protect the casing of the sensitive element from corrosion.

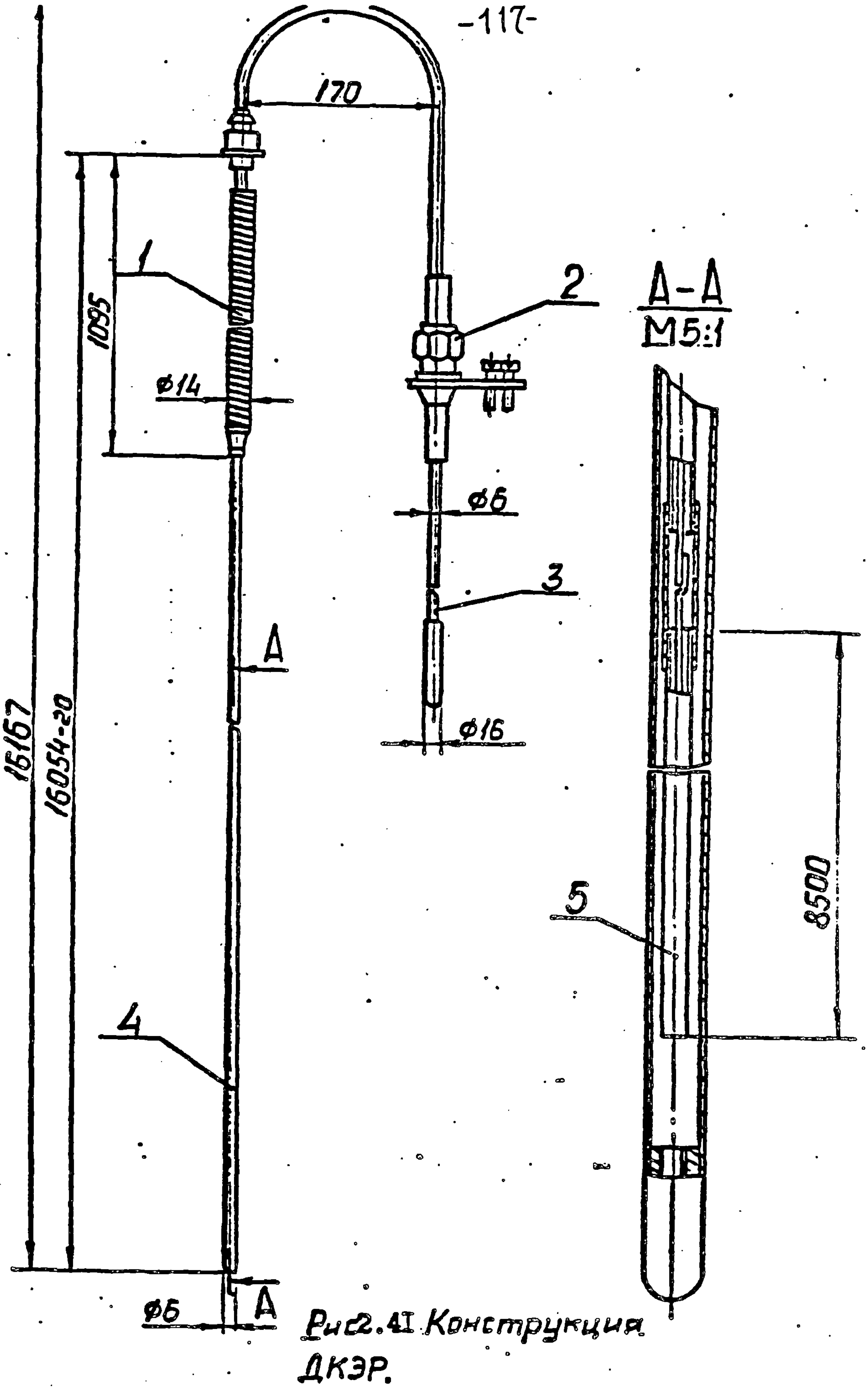


Рис2.41 Конструкция ДКЭР.

The total length of the detector is 16167 millimeters and the length of the sensitive element is 8500 millimeters.

A beta emission detector with a silver emitter 5 is used as the sensitive element. It is a high temperature KDMS(S) cable with an outside diameter of 3 millimeters, a middle strand of silver 0.65 millimeters in diameter, a sheath of corrosion resistant steel, and magnesium oxide insulation 0.8 millimeters thick. The cable is manufactured by an industrial technology used for high temperature cables and thermocouples. The sensitivity of the detector is approximately 5 times 10exp-20 A per square centimeter per second per neutron per meter of length. The maximum current of the sensitive element at nominal reactor power is about 15 microamperes. The maximum temperature of the sensitive element of the detector is greater than that of the heat transfer agent in the container due to radiation heating and amounts to approximately 350C.

The average ratio of the power of an un-burnt out can with an energy release detector to the current of an un-burnt out detector is 0.2 megawatts per microampere. The variations of this ratio for each detector due to its individual sensitivity and neutron spectrum are taken into account by means of periodic calibrations of the detectors during the operation of the reactor. The mean square variation of the sensitivity of a detector to neutron flux is 4%, according to experiment data. At the same time, the mean square variation of the sensitivity of a detector to can power is greater and amounts to 6%, which can be explained by

This effect may be taken into account on the basis of measured distributions of spectrum characteristics over the reactor, but in practice, a method involving direct periodic calibration of each detector on the basis of the power of each fuel can and scanning containers with hollow central sleeves in an operating reactor by means of DKER type beta emission detectors or triaxial fission chambers has been adopted.

The mathematical and experimental relationship of the neutron flux sensitivity detector to the integral current of a detector is a quite effective measure of neutron fluence which is quite independent of the neutron spectrum or the temperature of the detector.

The ratio of the power of a can with a detector to neutron flux density at the point of installation of the detector depends on the integral energy generation of the can Ei.

During the operation of the reactor can power is calculated by the formula

$$W_i = K_{\tau p i} \xi_D (I_i) \cdot \xi_{TD} (E_i) J_i \qquad (I)$$

where $K_{\tau p i}$ is the individual calibration factor of the ith detector and Ji is the current of the ith detector.

When a detector in a burnt out can is replaced, the current value of (I) in the SKALA computer is changed by changing the integral current of the differences in the neutron spectrum among different cans with radial energy release detectors.

detector stored in the computer's memory. Calculations have shown that the error associated with the use of formula 1 in replacing a burnt out detector with a fresh one does not exceed 1%. In general the experience of operating RBMK reactors has shown that the aforementioned allowance for the burnout of detectors and cans does not lead to errors greater than 1% in determining energy release in the reactor.

The radial energy release detector (without its cable line) is installed in the center sleeve of the container by means of the crane in the central room. Cable lines are laid when the reactor is built. Malfunctioning detectors may be replaced when the reactor is down or when it is going after they are disconnected from the cable lines.

The detector is designed to have the same service life as the fuel can. Radial detectors have proven to be highly reliable in RBMK reactors. According to service records, the average mean time to failure of a detector is 9.7 times 10exp4 hours.

A radial energy release detector is considered to have failed if:

--the emitter breaks and there is no current at the emitter's connector

--the readings of the detector are rejected by the SKALA computer when it is performing calculations on the Prizma program.

--the sensitivity of the detector drops, allowing for burnout, by more than 15% between calibrations

--rapid fluctuations in the signals of a detector which do not occur with adjacent detectors

--a drop in the resistance of the detector's insulation below 100 kiloohms

Height Energy Release Detectors

12 detector assemblies evenly distributed over the active zone of the reactor in the radial distribution plateau region are used to monitor energy release distribution by height. Each assembly includes seven beta emission detectors with silver emitters made in the form of KDMS(S) cable like those of the radial energy release detectors. These detectors are evenly distributed over the height of the reactor.

Each sensitive element (section) is a coil of the same cable with an outside diameter of 62 millimeters and a height of 105 millimeters. The total length of cable in the coil is 2.6 meters. The centers of the upper and lower sections are offset from the boundaries of the active zone towards the center by 500 millimeters.

The design of a height energy release detector is illustrated in Figure 2.42. Seven sensitive elements are enclosed in a dry sealed sleeve made of corrosion resistant steel and mounted in a channel similar to that designed to hold the control rods. On the outside the sleeve is cooled with a layer of running water 7 millimeters thick with a temperature of no more than 70C when it leaves the reactor. Along the axis of the sleeve is a center tube designed for periodic calibration of the detector sections by means of a triaxial fission chamber which can be shifted vertically over the detector. In the idle position the fission chamber may be left in the center tube of the detector, because its sensitive space will lie below the lower boundary of the reactor's active zone.

The sensitive elements are connected by high temperature KNMS(S) cables to sealed connectors located at the outlet from the sleeve into the central room. The same cable, which is enclosed in a protective sheath made of corrosion resistant steel, connects the sensitive elements by the connectors to an outside terminal. The cables are routed so as to optimize noise immunity. In particular, one may not lay detector cables and the power cables of the control rod drives together.

The inside of the sleeve is filled with a mixture of argon and helium in order to reduce radiation heating of the detector; as a result, the maximum temperature of the sensitive element will not exceed 150C.

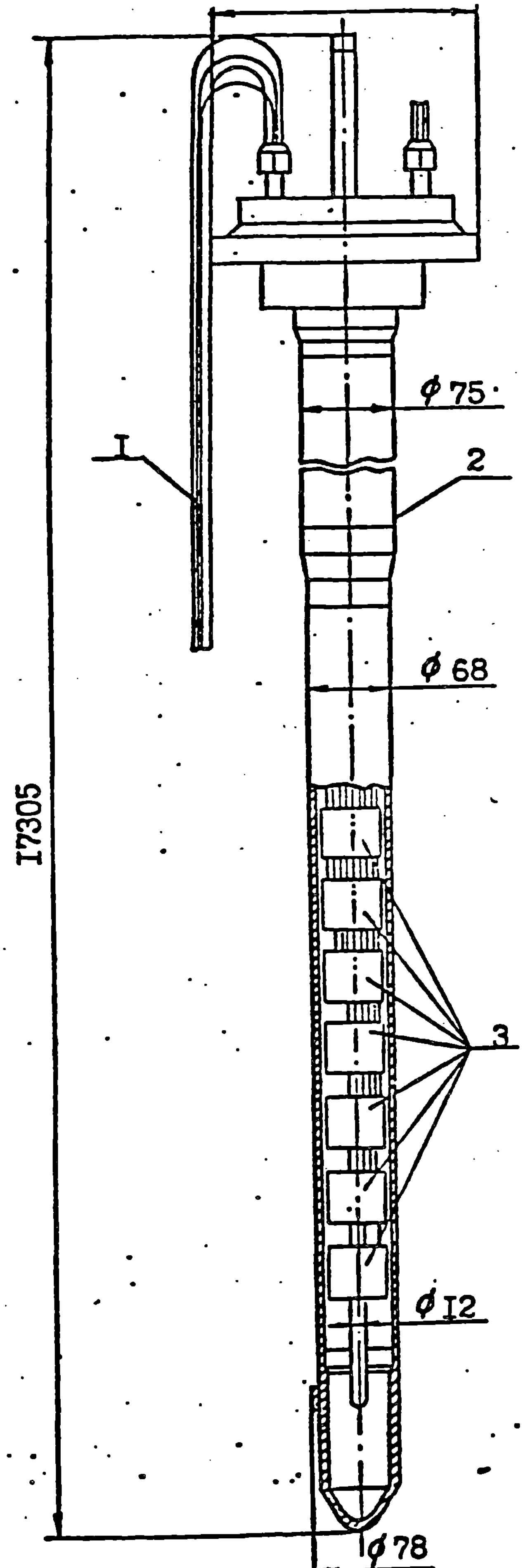


Рис. 2.42 Конструкция ДКЭВ.
I-кабель; 2-герметичная труба; 3-чувствительные элементы.

Figure 2.42 The Design of the Height Energy Release Detector

Key: 1. cable 2. sealed tube 3. sensitive elements

In order to shield the space above the reactor from the ionizing radiation of the active zone and the steam and water lines of the reactor, the detector is equipped with two steel shielding plugs situated in the top part of the sleeve. In addition, there is a special shielding cap in the top part of the detector which at the same time serves to protect the connectors of the detector from mechanical damage.

When the reactor is operating at nominal power, the currents of individual sensitive sections may vary from several microamps to 15 microamps, depending on the location of the section in the active zone.

The design of the detector assembly and panel makes it possible to replace a detector when the reactor is running or shut down. Detectors are replaced by remote control by means of the central room's crane. Cables must be laid during the construction of the reactor and can be replaced only when the reactor is shut down.

During the operation of the reactor, signals from each section of the detector are used to calculate neutron flux density at the place where it is installed:

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

where n is neutron density, v0 is 2200 meters per second, Kgpij is the individual calibration factor of the ith section of the jth detector, xi(D)(Iij) is a correction factor for emitter burnout which depends on the integral current of the ith section of the jth detector Iij and is identical to that used for a radial energy release detector, and Jij is the current of the ith section of the jth height energy release detector.

The service life of a height energy release detector is assumed to be two and a half years. The experience of operating RBMK reactors has shown that these detectors are satisfactory in terms of reliability. The average mean time until failure of a height energy release detector, according to service records, is 4.0 times 10exp4 hours.

A height energy release detector is assumed to have failed in the following cases: --there is a reduction in the sensitivity of a section of more than 15% between calibrations, allowing for burnout. --rapid fluctuations in the signals from a section not present in the readings from other sections --a reduction in the resistance of a section's insulation below 100 kiloohms.

Energy Release Monitoring Instruments

The Purpose and Composition of the Instruments

Energy release monitoring instrumentation is physically made in the form of four bays which contain basic functional units and control and testing equipment. The instrumentation also includes displays for checking energy release distribution in the reactor, a recorder, and a reading instrument for monitoring the thermal power of the reactor, a control panel for alarm set points, switching equipment, and digital indicators of the coordinates of a detector called by the display, and equipment for testing and adjusting the basic functional units.

Energy release monitoring instrumentation may be divided into two groups on the basis of its purpose: SFKRER and SFKREV equipment. The basic functional units of the groups differ in design: the first group includes two measuring instruments, while the second includes one.

All three measuring instruments are serviced by one multichannel recorder which employs a digital printer to record detector output signals which exceed safe limits, excess time, and the coordinates of the detector.

The measuring instruments perform the basic functions of generating data signals and online monitoring of energy release distribution in the reactor. These functions are performed separately for each radial and height energy release detector by components of the system (instrumentation lines). SFKRER instruments can process signals from 130 radial detectors mounted in the reactor and an additional 14 radial detectors can be hooked up (a total of 144 instrument lines). An SFKREV instrument can process signals from 12 seven section height energy release detectors (84 instrument lines), but it can handle signals from 12 eight section detectors (a total of 96 instrument lines).

Detector signals proceed to the inputs of individual amplifiers with controlled negative feedback, which in turn convert detector currents (the input signals)into DC voltage signals (output signals). These signals proceed to the inputs of an online energy release distribution monitor (alarm device), to the inputs of a switching device (displays), to a detector signal averaging device, to the inputs of the SKALA computer, and, by way of the contacts of actuator signal relays, to a multichannel recorder.

The alarm device compares the output signals of detectors with the specified limit values of these signals.

The comparison is carried out at three levels (alarm thresholds), called "too low", "warning", and "alarm". In the event that detector output signals deviate from the aforementioned levels, the appropriate alarms will be activated in the energy release monitoring instruments and on the SUZ-SFKRE mimic panel blinking colored warning lights will be activated. A green light means that detector signals are equal or less than the "too low" level. The absence of light means that a detector signal is above the "too low" level but below the "warning" level, that is, is normal. A red light means that the detector signals are equal to or higher than the "warning" level, but have not yet reached the "alarm" level, while a blinking red light means that the signals have reached or surpassed the "alarm" level. In this case, a sound alarm will be activated in addition to the blinking red light.

In order to provide a clear presentation of information, the SUZ-SFKRE mimic panel is made in the form of a model of the horizontal section of the reactor, where the control rod position indicators and detector alarm elements are installed. The location of the position indicators and alarm elements on the mimic panel corresponds to the location of the control rods and detectors in the radial plane of the reactor. The SFKRE displays make it possible for an operator to get a clear view of the area where a detector signal deviation occurred, use the signal to determine whether to move the control rods up or down, making it possible to correct this deviation, and select the necessary rod for this purpose.

In addition, the SFKRER equipment provides the possibility of changing the "too low" and "warning levels" by as much as 15% in either direction for all the detectors simultaneously from the reactor operator's panel, which makes it possible for the operator to determine areas where energy release is close to the limit ahead of time.

The SFKRER instrumentation includes two operational modes, namely a mode of comparison of the output signals of the radial detectors with floating levels (thresholds) of "too low" and "warning" alarms and fixed levels (thresholds) of "alarm", and a mode of comparing the output signals of the radial detectors with fixed thresholds for all three levels.

In the first mode, the "too low" and "warning" levels for each detector will vary proportionally to the mean arithmetic value of the output signals of the detectors, that is, in proportion to current reactor power, while the "alarm" level will be fixed at a level chosen on the basis of operational requirements. When the mean arithmetic values of the output signals of a detector reach some specified limit value (a specific level of power at a given phase of reactor operation), the "too low" and "warning" thresholds will be fixed (limited) and the SFKRER instrumentation will automatically switch to the second mode.

SFKREV instruments operate in a mode of comparing the output signals of the height detector sections with floating thresholds for all three alarm levels, that is, carry out only relative monitoring of energy release distribution by height of the reactor. Alarm thresholds for each height energy release detector will vary in proportion to the mean arithmetic values of the signals of the sections of a given detector.

To check the serviceability of the detectors, the instrumentation includes devices which make it possible to determine the resistance of the insulation of any detector by a method of connecting an additional resistor (100 kilometers) to the input circuit of an individual amplifier in the instrument line. The relative reduction in the output signal of the detector can be used to determine the resistance of the insulation mathematically:

$$R = R_{\text{доб}} \frac{U'/U}{1 - U'/U},$$

where U is the output signal of the detector prior to the connection of the resistor $R_{\text{доб}}$ and U' is the output signal of the detector after the resistor $R_{\text{доб}}$ has been connected.

The Basic Technical Data and Characteristics of the Instruments

The maximum output signal value of the individual amplifiers of the instrument lines is 5 volts, and signal polarity is negative. The range of control of the conversion factors of the individual amplifiers is 0.26 to 0.78 volts per microamp. The input resistance of the amplifiers is no greater than 100 ohms.

The basic relative input signal conversion error, in percentage points, is no greater than:

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

where Jmax = 19 microamps, or the maximum value of input current, J is the current value of input current in microamps.

The maximum capacitance of the detector and the cable line should not exceed 0.05 microfarads.

The load resistance on the output terminals of the amplifiers should be at least 2 kiloohms.

The instruments can output eight detector signals to M1830A reading instruments simultaneously: one of 130 output signals of the radial detectors and seven output signals from the sections of a selected height detector.

Calls to the reading instruments are made by means of the switching devices, which remember the address of a radial or height detector in reactor channel coordinate code.

The instruments generate signals equal to the mean arithmetic values of radial detector signals (reactor power signals) and output it to a reading instrument (scale of 0 to 100 microamps) and a recording instrument (scale of 0 to 100 microamps, 10 seconds to cover the scale). In the instruments adjustments may be made to signal mean arithmetic values to allow for the absence of signals at the inputs of some of the amplifiers. The basic relative error of average signal generation, given an individual amplifier output signal amplitude of at least 2.5 volts and 70 to 130 averaged signals, does not exceed plus or minus 0.5%.

The instruments generate four power signals, one for each quarter of the reactor, equal to the mean arithmetic value of the output signals of the radial detector of a corresponding quarter, which are output to 4 reading instruments.

For each height detector, it generates a signal equal to the average arithmetic value of the output signals of its sections. Adjustments may be made to the mean arithmetic values to account for the absence of signals at the inputs of certain amplifiers. Basic relative averaging error, given signal amplitudes of at least 2.5 volts and 4 to 8 averaged signals, does not exceed plus or minus 0.5%.

A fixed "alarm" threshold is set in each instrument line to monitor the radial distribution of energy release in the reactor:

$$U_{ABi} = \frac{1}{K_{\phi i}} \cdot \frac{\delta_{AB}}{100} \cdot U_{100},$$

along with a "warning" threshold which varies in proportion to reactor power:

$$U_{npi} = \frac{1}{K_{\phi i}} \cdot \delta_{np} \cdot \overline{U},$$

and a "too low" threshold which varies in the same way:

$$U_{3AHi} = \frac{1}{K_{\phi i}} \delta_{3AK} \cdot \overline{U},$$

where Kphii is the gain factor of the amplifier of the comparison unit of the ith instrument line which can be adjusted continuously from 0.6 to 2.5 deltaab is the gain factor of the "alarm level %" divider, which can be adjusted discretely in percentage points from the nominal level from 0 to 100%, U100 is voltage corresponding to the nominal level of reactor power and can be adjusted continuously within a range of 1 to 5 volts, deltanp is the relative "warning" level and can be adjusted continuously within a range of 0.65 to 1.25, deltazak is the relative "too low" level, which can be adjusted continuously within limits of 0.65 to 1.25, and U(overhead bar) is the mean arithmetic value of the output signals of the radial energy release detectors.

The instruments make it possible for an operator to change the "warning" and "too low" levels by as much as 15% in either direction in percentage points simultaneously.

The instruments make it possible to limit the "warning" and "too low" thresholds at the levels:

$$(U_{np\,i})_{макс} = \frac{1}{K_{\phi i}} \cdot \delta_{np} \cdot U_{огр.np},$$

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

The limit levels Uogr.np and Uogr.zan (warning and two low) can be adjusted continuously within a range of 1 to 5 volts.

Alarm thresholds which vary in proportion to the mean arithmetic value of the output signals of a height energy release detector are set in each instrument line of a jth height energy release detector:

--"alarm" – $U_{ав\,ij} = \frac{1}{K_{\phi ij}} \cdot \delta_{ав} \overline{U_j},$

--"warning" – $U_{зан\,ij} = \frac{1}{K_{\phi ij}} \cdot \delta_{зан} \cdot \overline{U_j},$

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

--"too low" where Kphiij is the gain factor of the amplifier of the comparison unit of the ith instrument line of the jth height energy release detector and can be adjusted continuously within a range of 0.6 to 2.5.

~~beta~~δab, ~~beta~~δnp, and ~~beta~~δzan are the respective alarm levels and can be adjusted continuously within a range of 0.75 to 1.85

U(overhead bar)j is the mean arithmetic value of the output signals of the sections of the jth height energy release detector.

The gain factors Kphii and Kphiij are calculated to set alarm operation thresholds in the instrument lines by the SKALA computer in every online run performed with the Prizma program. The other parameters in formulas (5-7) and (10-12), which define alarm operation thresholds, are either set constants or the current values of averaged signals. Gain factors must be adjusted in accordance with the new calculated values in all cases where set values and newly calculated values vary by 5% or more in even one instrument line.

Given detector output signal levels of at least 2.5 volts, basic relative alarm operation error does not exceed plus or minus 2 percent.

The calculated mean time until failure of an instrument line is at least 15,000 hours. The service life of the instrumentation is at least six years.

2.9.8. Special Operating Software for the Reactors at the Chernobyl Plant

I. functions performed and the structure of calculations

The special software for the Chernobyl reactors is designed to perform the following functions: --calculate power in each fuel can --calculate the power safety factor standing between a critical heat removal situation in each can --calculate the temperature of the graphite in the active zone --calculate reactor power by a heat budget method --calculate steam content at the outlet of each channel --calculate the thermotechnical reliability of the reactor --calculate the energy output of each can and the reactor --calculate settings for the energy release detectors in the active zone of the reactor --calculate several characteristics of height energy distribution --calculate the effective reserve of reactivity --devise recommendations for controlling water flow through process channels --calculate general reactor parameters, namely the coefficient of radial unevenness of energy release distribution, power distribution, and flow rates for reactor halves, drum separators, and so forth. --devise recommendations with respect to process channel transfers

This last job is done by an outside computer center and the results are transmitted to the nuclear power plant on a communications line together with the results of neutron physical analyses.

The other functions are performed by the plant's SKALA computer using the multifunctional Prizma program.

Initial data for the Prizma program include:

--the signals of the energy release detectors inside the reactor's active zone --signals from the control rod position indicators --signals from the flow meters of each reactor channel, water temperature in the pressure collectors, pressure in the drum separators, feed water flow, and so forth;

--signals from the thermocouples used to measure the temperature of the graphite --the results of neutron physical analysis of energy release distribution

2.9.8.1. Calculation Schedules and Accuracy

Basic calculations are performed with the Prizma program once every five to ten minutes. The energy output of each can and the change in the sensitivity of detectors due to burnout of emitters and fuel are calculated once a day.

The accuracy of calculations of the relative power of each can is approximately 3%. At the Chernobyl Plant this accuracy was confirmed by a special experiment which involved comparing the results of can power calculations and measurements taken with a calibration detector.

2.9.8.2. Basic Analytical Formulas

Calculations of power in each fuel can constitute the basic analytical component of on line data processing in the SKALA centralized monitoring system. The procedure for this calculation uses the results of neutron physical analysis of energy release distribution and the readings of intrareactor detectors as initial data.

The results of neutron physical calculations of energy release distribution, namely the powers of each fuel can q(0)pi, i = 1, 2,...,Ntvk (Ntvk is the number of fuel cans in the reactor) are adjusted when fuel transfers are made in the process channels by the formula

$$q_{pi} = q_{pi}^{(0)} \frac{\xi_T(R, E')}{\xi_T(R, E)},$$

in which the correction factors ξt, which differ for different types of transfers, are defined in tables in relation to the distance R between the channel in question and the channel where the transfer is underway and the energy output of the can being unloaded and the can being loaded (E and E' respectively). The correction factors were obtained on the basis of a mathematical analysis of transfers according to the neutron physical analysis program. The aforementioned correction is made in the plant computer directly prior to a transfer or immediately after it.

Energy release distribution is calculated by the following formulas.

An empirical correction is made to the results of neutron physical analysis of energy release distribution to allow for the change in the power of each can due to movement of the control rods:

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

where the coefficients ξ_{pc} defined in the table depend on the distance Rik between the ith can and the kth rod and the depth of insertion of the kth rod at the moment the calculation was made (h'k) and the moment corresponding to the time when neutron physical calculations were made. The correction factors were obtained on the basis of a mathematical analysis of the effect of rod movement on can power using the neutron physical analysis program.

The signals $\mathcal{J}_j$ of the intrareactor detectors are converted into the quantities

$$q_j = \mathcal{J}_j \, K_{rpj} \, \xi_{\partial j} \, \xi_{T\partial j}, \quad j = 1, 2, \ldots, N_\partial,$$

where N_∂ is the number of serviceable detectors, Krpj is the calibration factor of the jth detector, and $\xi_{\partial j}$, $\xi_{T\partial j}$ are coefficients which allow for the burnout of the emitter of the detector and fuel in the channel containing the detector.

For containers with detectors the following ratios are calculated

$$V_j = \frac{q_j}{q'_{rj}},$$

the radial azimuthal distribution of which can be approximated by the formula

$$\tilde{V}(z_i, \varphi_i) = \sum_{\ell} a_\ell \, f_\ell(z_i, \varphi_i)$$

for all cans in the reactor. In this formula Zi and Yi are the coordinates of the ith can, $f_\ell(z_i\, y_i)$ is a system of radial azimuthal functions. The coefficients ae are determined by the smallest squares method. This approximation makes it possible to allow for possible deformations of energy release distribution due to transient xenon and temperature processes and the component of neutron physical analysis error due to random variations of the physical characteristics of the cans and other elements of the active zone and the methodological error of neutron physical analysis.

For cans with detectors the following quantities are calculated:

$$\overset{\circ}{V}_j = \frac{V_j}{\tilde{V}(z_j, y_j)} - 1.$$

The readings of the jth detector or the position indicator of the control rod located in the vicinity of this detector are considered unreliable if

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

where χ is a quantile of normal distribution corresponding to a specific probability that an unreliable measurement will be taken

$$D_{\overset{\circ}{V}} = \sum_{j=1}^{N_9} \overset{\circ}{V}_j{}^2 / (N_9 - 1).$$

The readings of these detectors are not used in calculations and information on them is automatically printed out for staff use.

Finally the power of each can, including those with detectors, is calculated by the formula

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

Summation is carried out over the four detectors closes to the ith can. The weighting factors β_{ik} are determined by means of solving a system of four linear equations written to minimize calculation error. The values of these coefficients depend on the distance between the can and the detector, the statistical characteristics of the quantities Vj, and detector calibration error.

At the same time that the power Wi of each can is calculated, the dispersion Di of the error of this value is also calculated.

Safety Factor Calculations

The maximum permissible power of an RBMK reactor can is considered the power at which the probability for a can to be in a critical heat removal situation will reach a certain value which will be constant over time and identical for all cans. According to this definition, the power safety factor for the ith can is equal to

$$K_{3i} = \frac{W_{\kappa pi} - \sqrt{W_{\kappa pi}^2 - C_{1i} C_{2i}}}{C_{1i} C_{2i}},$$

where Wi is can power, Wkri is the can power at which a heat removal crisis will occur. Wkri is determined from tables on the basis of water flow through the channel, pressure in the separator drum, and water temperature in the pressure collector;

$$C_{1i} = 1 - \varkappa^2 (D_i + D_{T\Pi 2});$$

$$C_{2i} = W^2_{\kappa p i}(1 - \varkappa^2 D_{T\Pi 1}) - \varkappa^2 D_{T\Pi 3};$$

$\varkappa$ is the quantile of normal distribution corresponding to a specific probability that a can will suffer a heat removal crisis; Di is the dispersion of the relative error of determination of can power Dtpl is the dispersion of the relative error of determination of critical can power (methodological) Dtp2 is the dispersion of the error of determination of reactor power Dtp3 is the dispersion of the determination of Wkp due to flow, pressure, and temperature measurement errors.

Graphite Temperature Calculations

Graphite temperature is calculated on the basis of calculations of can power, data on the height distribution of energy release, and signals from thermocouples installed in the graphite stacking.

Graphite temperature is calculated for every kth joint of the graphite columns on the basis of the following formula:

$$t_{rk} = t_T + \alpha \overline{W_k} \varphi_k ,$$

where t_T ~~tt~~ is the average temperature of the heat transfer agent in the reactor;

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

Wi(k) is the power of the can in the channel adjacent to the kth joint
m is the number of such joints

$\gamma_{i(k)}$ is a coefficient which allows for the effect of channels containing different objects on heat abstraction from the graphite (fuel, absorbers, control rods)

φ_{K} is the relative density of neutron flux at the level where the thermocouples are located

α is the factor of proportionality between the signals of the thermocouples and power determined by the smallest squares method.

Very abnormal thermocouple readings are discarded.

Calculating Intrazonal Detector Settings

The setting of a radial monitoring detector is calculated by the formula:

$$U_j = J_j \max_{i \in \omega_j} \left\{ \frac{q(z_i)}{W_i} \right\} \cdot c$$

where J_j is the signal of the jth detector, Wi is the power of the ith can located in area wj near the jth detector (this area is a square of five by five reactor cells, at the center of which is the jth detector), and $q(i)$ is the relative stipulated power of the ith can, and C is a normalizing constant corresponding to a specific maximum can power.

The setting of a height monitoring detector is calculated so that a specific maximum linear load on the fuel elements in the cans closest to the detector is not exceeded.

On the basis of the calculated settings, the computer calculate and prints out, on request, the gain factors of the amplifiers of the comparison units of the monitoring instruments, which are then set in the instruments.

Calculations of Effective Reactivity Reserves

The effective reactivity reserve on the control rods is calculated by the formula

$$\rho = \sum_{K=1}^{N_{PC}} C_K \int_0^{h_K} \Phi_K^2(z)\,dz \Big/ \sum_{K=1}^{N_{PC}} \int_0^{H} \Phi_K^2(z)\,dz$$

where Ck is the relative "weight" of the rod depending on its type, Npc is the number of control rods, $\Phi_K(z)$ is a quantity proportional to the height distribution of absolute neutron flux density at the location of the kth rod calculated using calculated can power values and height monitor detector readings averaged over the reactor.

2.9.9. Display of Information on Calculation Results

All calculation results may be printed out on a printer at the request of the

channels with the highest can power, the 60 channels with the highest graphite temperatures, and the 60 channels with the lowest safety factors.

The printer automatically gives the time and the coordinates of a down detector, the rejection constant, the time and coordinates of the channel, and power (if out of limits).

The channel mimic panel indicates the channels in which power is above the set point given by the operator and channels whose safety factors are below those set by the operator, and so forth.

The values of any calculated quantity may be output to a digital reader on request.

2.10 Safety Assurance Systems

2.10.1. Protective Safety Systems

2.10.1.1. The Reactor Emergency Cooling System

The reactor emergency cooling system (Figure 2.43) is a protective cooling system and is designed to provide for the removal of residual heat release (after the chain reaction has been suppressed) by means of the timely delivery of the required amount of water into the reactor channels during accidents accompanied by disruption of the active zone's cooling system.

Such accidents include breaks in the large diameter piping of the multiple forced circulation loop, breaks in the fresh steam lines, and breaks in the feed water lines.

In addition, the emergency cooling system may be used for emergency delivery of water into reactor channels in situations which do not involve piping breaks but make it impossible to deliver water by standard systems (for example, steaming of the PEN and APEN).

The designers of the emergency cooling system took into account the following requirements which the system must meet:

1. It should deliver water to the affected and unaffected halves of the reactor at rates which are at least those indicated in Figure 2.44 so as to prevent meltdowns, massive overheating, and unsealing of the fuel elements.

2. The system should be automatically activated on a maximum design emergency signal, which should distinguish the affected half from the unaffected half and be generated when the following symptoms are present:

a) an increase in pressure in the rooms containing multiple forced circulation loop piping (a sign that a pipeline has broken);

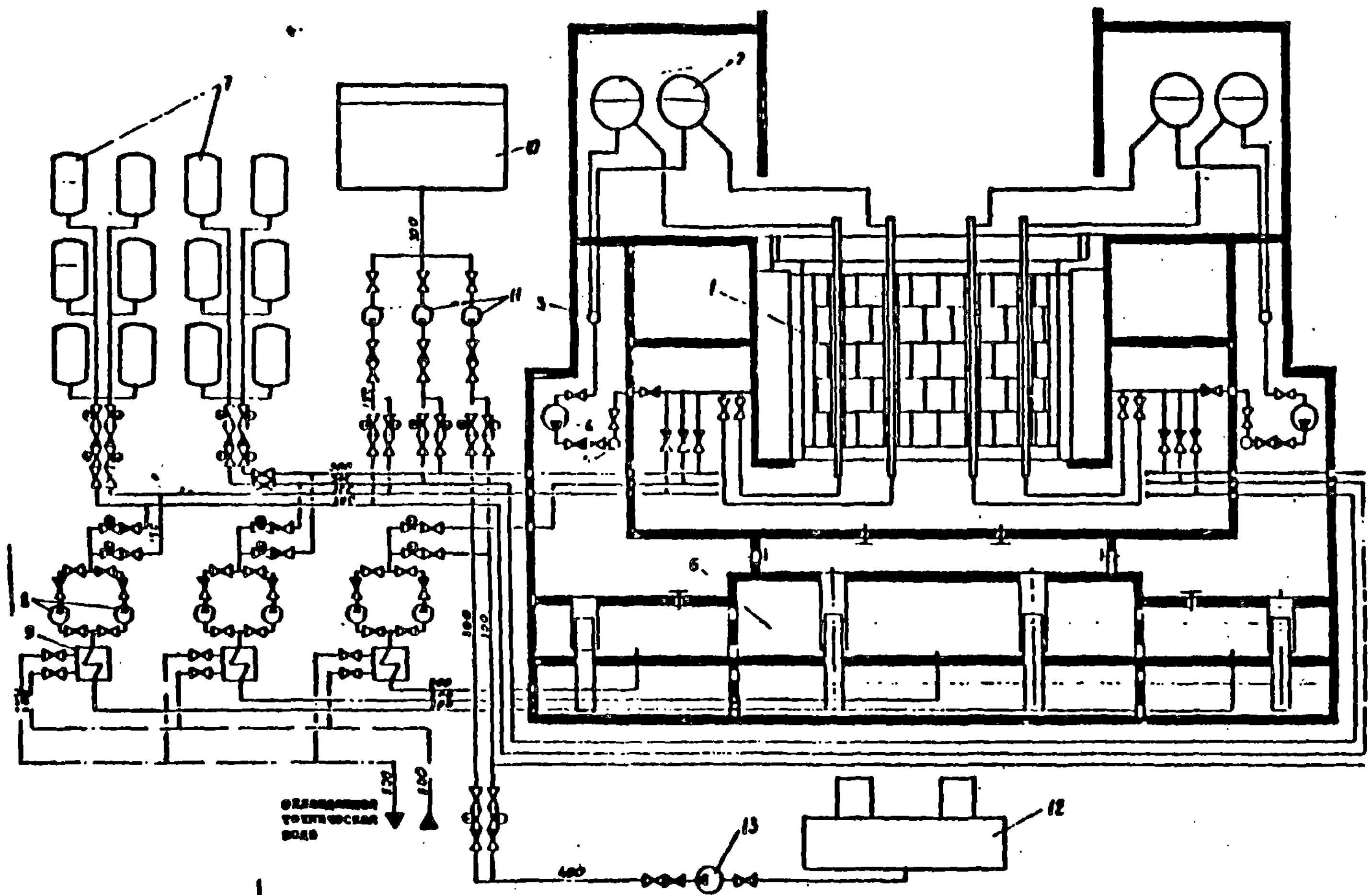


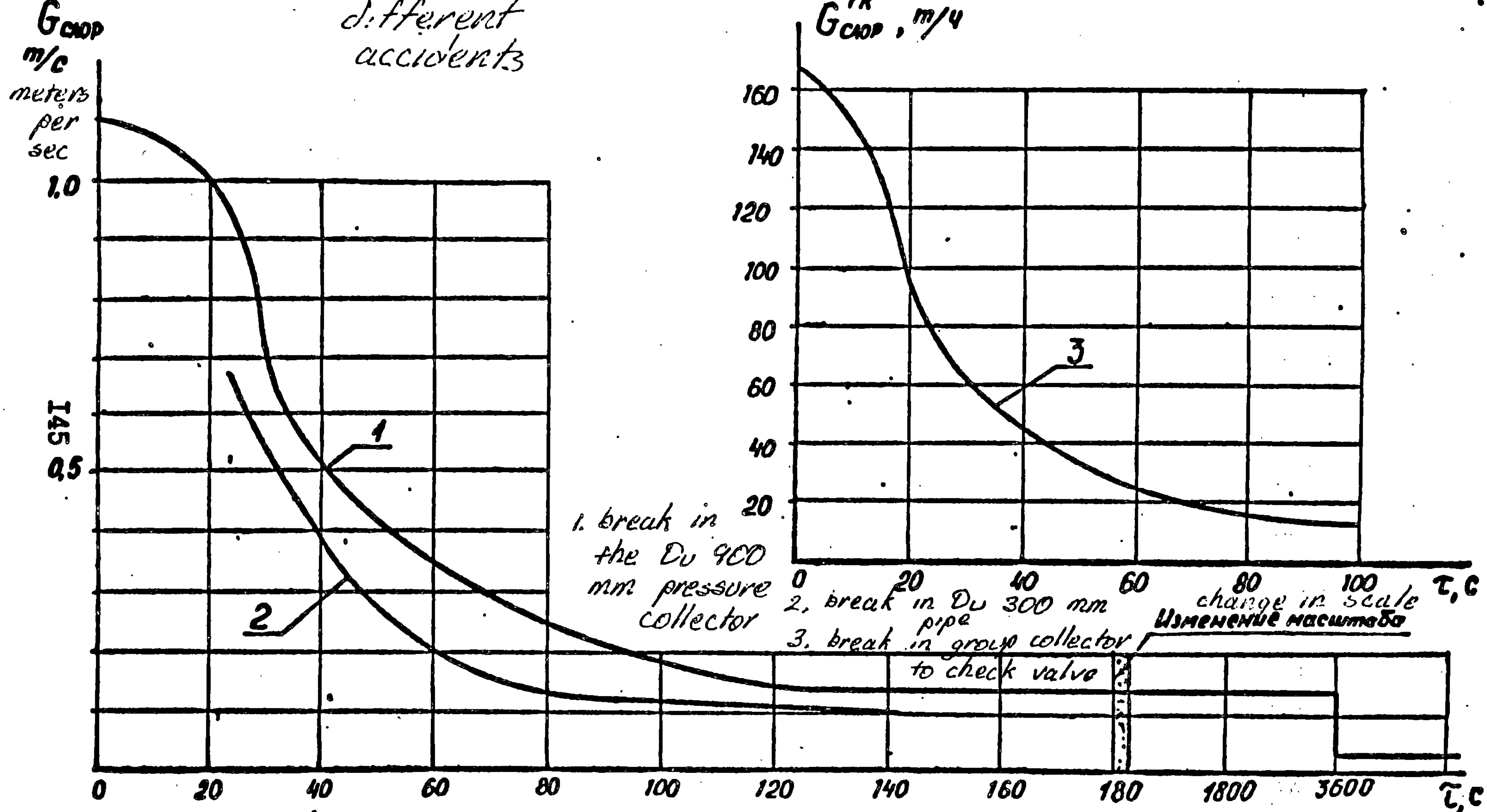
Рис. 2.43 Принципиальная схема системы аварийного охлаждения реактора

1 – реактор; 2 – сепараторы пара; 3 – всасывающий коллектор; 4 – главный циркуляционный насос; 5 – напорный коллектор;
6 – бассейн-барботер; 7 – ёмкости САОР; 8 – насосы САОР охлаждения аварийной половины реактора; 9 – теплообменники;
10 – бак чистого конденсата; 11 – насосы САОР охлаждения неаварийной половины реактора; 12 – деаэратор; 13 – питательный насос

Figure 2.4.3. Design of the Emergency Cooling System
1. reactor 2. steam separator 3. intake collector
4. main circulation pump 5. pressure collector 6. bubbling tank 7. system tanks 8. pumps for cooling the affected half of the reactor 9. heat exchangers
10. clean condensate tank. 11. pumps for cooling nonaffected half 12. deaerator 13. feed pump

Потребный расход воды в аварийную половину реактора при различных авариях

required water flow to the affected half of the reactor for different accidents.



Типы аварий:
type of accident

1 – при разрыве напорного коллектора Ду 900 мм
2 – при разрыве опускной трубы Ду 300 мм
3 – при разрыве группового коллектора до обратного клапана

Рис. 2.44

b) when it coincides with one of the following two signals

--a reduction in the level of the steam separators of the affected half of the reactor;

--a reduction in the pressure differential between the pressure collector of the main circulation pump and the steam separators of the affected half of the reactor.

3. Its speed should be such that any interruptions in the delivery of water to the affected half of the reactor in the event of a maximum design emergency should not last longer than 3.5 seconds.

4. Reductions in water flow to the reactor channels due to unproductive discharge through the site of the break into the room should not occur.

5. The system should perform its safety functions regardless of any failures unrelated to the initial event of the following elements of the system: active or passive elements with moving mechanical parts.

6. The system should consist of several independent channels (subsystems) and should be effective even if any one channel (subsystem) fails.

7. Nitrogen should be kept from getting into the reactor from the system's tanks when they are emptied.

8. The emergency cooling system should perform its role even if a maximum design emergency coincides with an internal power failure.

In accordance with the initial requirements listed above, the system consists of three independent channels (subsystems), each of which has a capacity of at least 50% of that required.

Each channel (subsystem) includes a fast acting part and a long term cooling part.

The fast acting part delivers the required amount of water to the channels of the affected half of the reactor during the initial phase of an accident.

The fast acting parts of two system channels (the tank parts) consist of a system of tanks (filled with water and nitrogen under pressure of 10.0 megapascals) connected by piping and collectors to the forced multiple circulation loop RGK.

A Du 400 valve, which makes it possible to obtain the required flow into the affected half of the reactor in 3.5 seconds, is used as a fast acting valve to activate the tank part of the system. The power supply for the valves is classified in the highest category of reliability and is provided by storage batteries (See Section 2.7.3)

Each of two tank parts consists of 6 tanks with volumes of 25 cubic meters each. The total initial volume of water in one tank part is about 80 cubic meters, while it contains about 70 cubic meters of nitrogen. Each tank part ensures a water flow into the affected half of the reactor of at least 50% of requirements for at least 100 seconds. The duration of the system's operation depends on the extent of heat transfer agent leakage from the forced multiple circulation loop during an accident.

The tanks are configured symmetrically so as to reduce the collector effect during discharge.

Nitrogen is kept from entering the reactor from the tanks by means of automatic closure of two sequentially installed valves on piping from the tank portions to the RGK upon receiving a minimum level signal.

The fast acting part of the third system channel is a unit for delivering water from the PEN, which provides a flow of at least 50% of that required into the affected half of the reactor.

In the event that a maximum design emergency coincides with an internal power failure, the flow of water from the PEN should be maintained for 45 to 50 seconds by operating the PEN off the turbogenerator.

Backup power for the actuators of the fast acting valves is provided by independent noninterruptible supplies (batteries).

The long term cooling part cools both the affected and unaffected halves of the reactor. It should be activated as soon as the fast acting part of the system ceases to operate.

The long term cooling parts of each channel consist of two groups of pumps:

cooling pumps for the affected half of the reactor

cooling pumps for the unaffected half of the reactor

The pump part for cooling the affected half of the reactor consists of two parallel connected pumps and delivers water at a rate of about 500 tons per hour, that is, no less than 50% of requirements during a maximum design emergency for the affected half.

Intake water is provided by the bubbling tank of the accident containment system and is cooled by industrial water in a heat exchanger installed on the common intake line of the two pumps and is delivered to the collectors of the system through pressure piping.

Flow restrictor inserts are installed on the pressure piping of the pumps and are designed to ensure the stable operation of the pumps during emergencies characterized by sharp drops of pressure in the circulation loop of the reactor when a pipeline breaks (flow is limited by the ebullition of water in the narrow section of the insert).

The pump part for cooling the unaffected half of the reactor includes one pump and provides a flow of about 250 tons per hour, that is, at least 50% of that required for the unaffected half in a maximum design emergency.

Intake water is taken from pure condensate pumps and is fed to the collectors of the tank part downstream of the fast acting valve through pressure piping.

The flow limiter inserts in the pressure piping perform the same role that they do in pumps for cooling the affected half of the reactor.

Backup power for the electric motors of the pumps and the valve actuators is provided by diesel generators.

2.10.1.2. The System for Guarding Against Overpressure in the Main Heat Transfer Agent Loop

The system is designed to keep pressure under control in the loop by means of removing excess steam into a bubbling tank, where it is completely condensed. It includes pulse safety devices and a system of piping and collectors to remove the steam into the bubbling tank of the accident containment system. The pulse safety device consists of pulse valves and main safety valves.

The system meets the following basic requirements:

--ensures that overpressure in the loop will not exceed working pressure by more than 15%, allowing for a single failure of any active or passive element in the system with moving mechanical parts.

--is characterized by high operational reliability when pressure in the loop reaches the operation set points

--has highly reliable main safety valves which close promptly once the closure set points are reached

--has the required service life under cyclical dynamic loads accompanying the operation of the main safety valves --provides for the entry of steam into the water of the bubbling tank at speeds close to the speed of sound, even if only one main safety valve operates (This is required for shock free steam condensation).

Figure 2.45 illustrates the design of a system for collecting steam from the main safety valves in the bubbling tank.

The system consists of eight main safety valves with a total capacity of 5800 tons per hour at nominal pressure in the loop, that is, a capacity equal to the nominal steam capacity of the reactor plant.

Each main safety valve with a capacity of 725 tons per hour is controlled by means of a direct action pulse valve (arm plus weight type) equipped with an solenoid type actuator for opening and closing it.

Steam is discharged from the main safety valve into the bubbling tank below the water level through submerged nozzles with outlet diameters of 40 millimeters each (1200 nozzles in all).

In order to prevent the formation of a vacuum in the discharge piping, and as a consequence, the ingress of water into them, and to ensure shock free condensation of possible small leaks of steam through closed main safety valves, steam air ejectors are used.

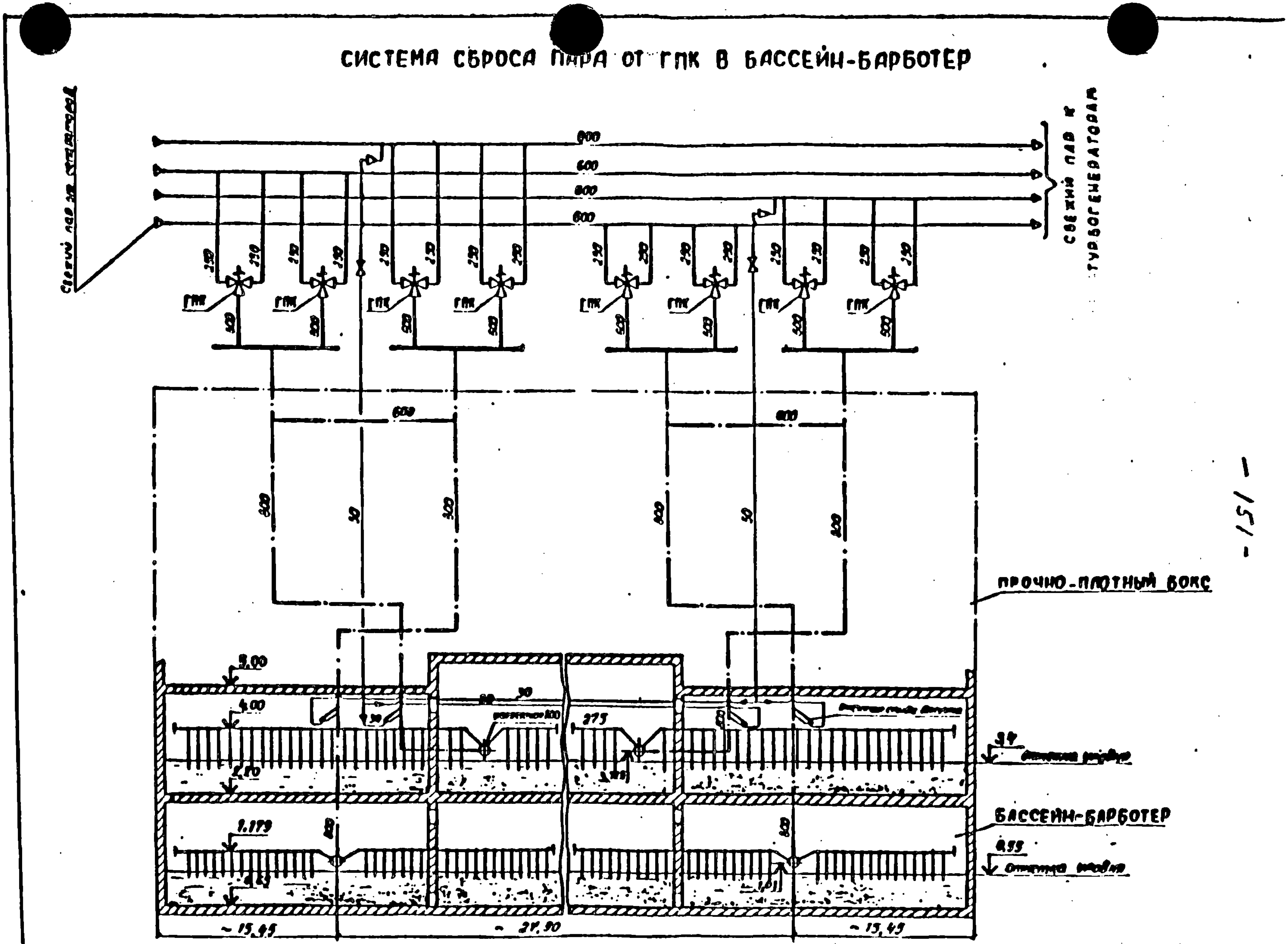


Рис. 2.45

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Fig. 2.45 Steam Dumping system from GPK into pressure suppression pool.

Steam dumping systems are equipped with:

- monitor of the absence of water level in BB headers;

- monitor of the temperature conditions of the outside pipe surface behind each GPK and pipes within the pressure suppression pool.

In the normal operating mode of the unit the GPK are closed and the system is in the waiting mode.

The system automatically engages only when the pressure in the MPTs loop exceeds the following values:

- 76 kgs/cm2 - 1 GPK triggers

- 77 kgs/cm2 - 2 GPK trigger;

- 78 kgs/cm2 - 1 GPK triggers;

- 81 kgs/cm2 - 4 GPK trigger.

Operating personnel can also force GPK open from BShchU and RShchU.

The main components of the system which protects the loop from excess pressure were experimentally checked on test stands during development. The system as a whole underwent comprehensive testing for agreement with design indicators during startup operations.

2.10.1.3. System for protecting reactor space (RP) from excess pressure.

The system is designed to ensure that allowable pressure in the RP is not exceeded in an emergency with rupture of one fuel channel due to escape of the gas-steam mixture from the RP into the steam-gas dumping compartment of the pressure suppression pool and then into the pressure suppression pool itself.

The system satisfies the following main requirements:

- Ensures the pressure in the RP is not exceeded by more than 1.8 kgs/cm2 (ABS) for a total transverse rupture of one fuel channel with allowance for single failure in the passive element system with moving mechanical parts (active elements absent in the system);

- Prevents water from the steam-gas dumping compartment of the pressure suppression pool from reaching the RP in the maximum conceivable accident;

- Reliably isolates the RP from the atmosphere.

Figure 2.46 shows a schematic of the system which protects the reactor space from excess pressure.

The reactor space is permanently connected to the steam-gas dumping compartment of the pressure suppression pool of the emergency localization system by eight pipelines Du 300 (four pipes above and four pipes below the RP which are then combined into two Du 600 pipes).

Each of the Du 600 pipes leads into its own stage of the compartment and is immersed 2 m below the water level, i.e. in the normal operating mode of the block the reactor space is cut off from the atmosphere by a hydraulic seal 2 m high.

The height of the vertical sections of steam dumping pipes Du 600 from the reactor to the water level in the compartment exceeds 28 m; for this reason pipe Du 600 which combines four pipes Du 300 below the RP is specially raised to mark 28.8 and then drops into the compartment.

This design is necessary to prevent water or the steam-air mixture from the compartment from entering the RP in accidents with rupture of KMPTs piping up to MPA.

The volume of water in the compartment is selected and maintained such that it is enough to fill the steam dumping pipes in the aforementioned situation with some reserve.

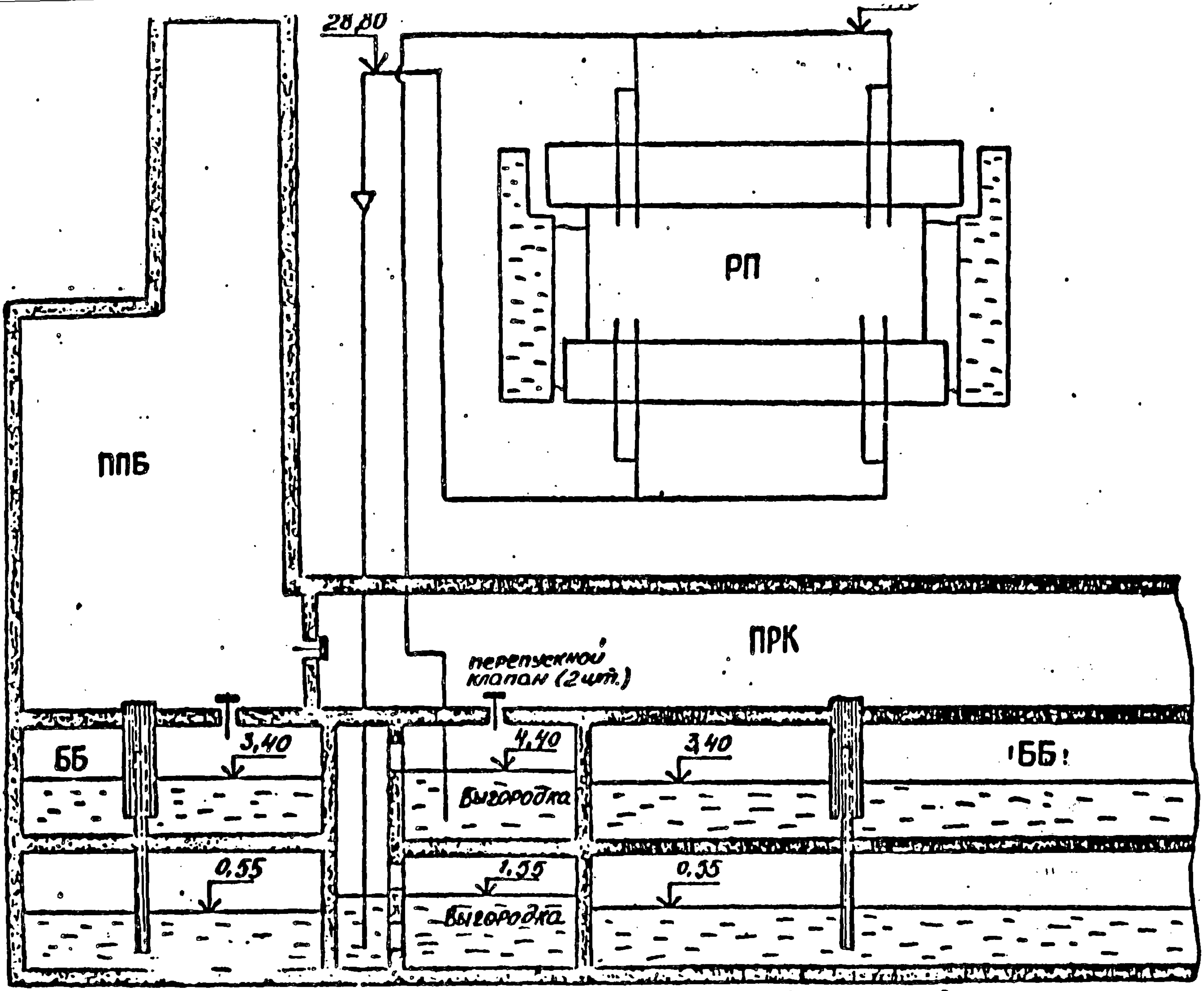


Рис. 2.46 Система защиты РП от превышения давления

Figure 2.46. System for protect[redacted]he RP from excess pressure.

Check (relief) valves which allow steam to be dumped from the compartment into the pressure suppression pool and which prevent backflow are used as an additional second barrier which prevents water or the steam-air mixture from the compartment from reaching the RP.

To prevent monitored spread of solid radioactive discharges throughout the water volume of the pressure suppression pool the gas-steam dumping compartment is reliably (by three barriers) cut off from the water volume of the pressure suppression pool during rupture of TK.

When the pressure in the RP rises to 1.2 kgs/cm2 (ABS) the hydraulic seal in the compartment is forced out and the gas-steam mixture enters the compartment through the steam dumping pipes. When the pressure in the space above the water in the compartment reaches 1.1 kgs/cm2 (ABS) check (relief) valves open and the gas-steam mixture enters the steam distribution lane, and from there through steam dumping pipes to below the water level of the pressure suppression pool. Steam formed in the RP during TK rupture first condenses completely in the water volume of the compartment and after its accumulating capacity is exhausted in the pressure suppression pool. As it bubbles through the water layer in the compartment in the pressure suppression pool gas from the RP cools and is kept in the spaces of the accident localization zone from which it is released into the atmosphere by the hydrogen removal system after required holding and purification.

Maximum pressure in the RP at all accident stages does not exceed 1.8 kgs/cm2 (ABS).

The protection system is equipped with the following:

- monitoring of pressure (rarefaction) in the RP;

- monitoring of the level in the steam-gas dumping compartment;

- reliable drainage of steam dumping pipes.

Monitoring of technological parameters and control of active system components (cutoff fittings) are done by operators from BShchU and RShchU.

2.10.2. Localizing safety systems.

The accident localization system (SLA) on the four blocks of the Chernobyl power plant is designed to localize radioactive emissions during accidents with failure of the seals of any piping of the reactor cooling system, except for the piping of the steam-water service lines (PVK), the upper runs of the fuel channels and that part of the downcomers of the drum type steam separators. Figure 2.47 shows a schematic of the system.

2.10.2.1. System of sealed compartments

The main component of the localization system is the system of sealed compartments which includes the following compartments of the reactor section:

- sealed volumes (items 1 and 2 on Fig. 2.47) arranged symmetrically relative to the reactor axis and designed for a gauge pressure of 0.45 MPa;

- RGK-NVK compartments (items 3 and 4) also symmetrical relative to the reactor axis and separated from one another by the reactor support.

crosspiece with leaks of total area 5 m2. In terms of strength conditions of reactor structural elements these compartments prevent pressure from rising above 0.08 MPa and are designed for this quantity. All components of the reactor loop are concentrated in the sealed volumes and compartments of the RGK-NVK. The system is designed for an accident with damage to these elements;

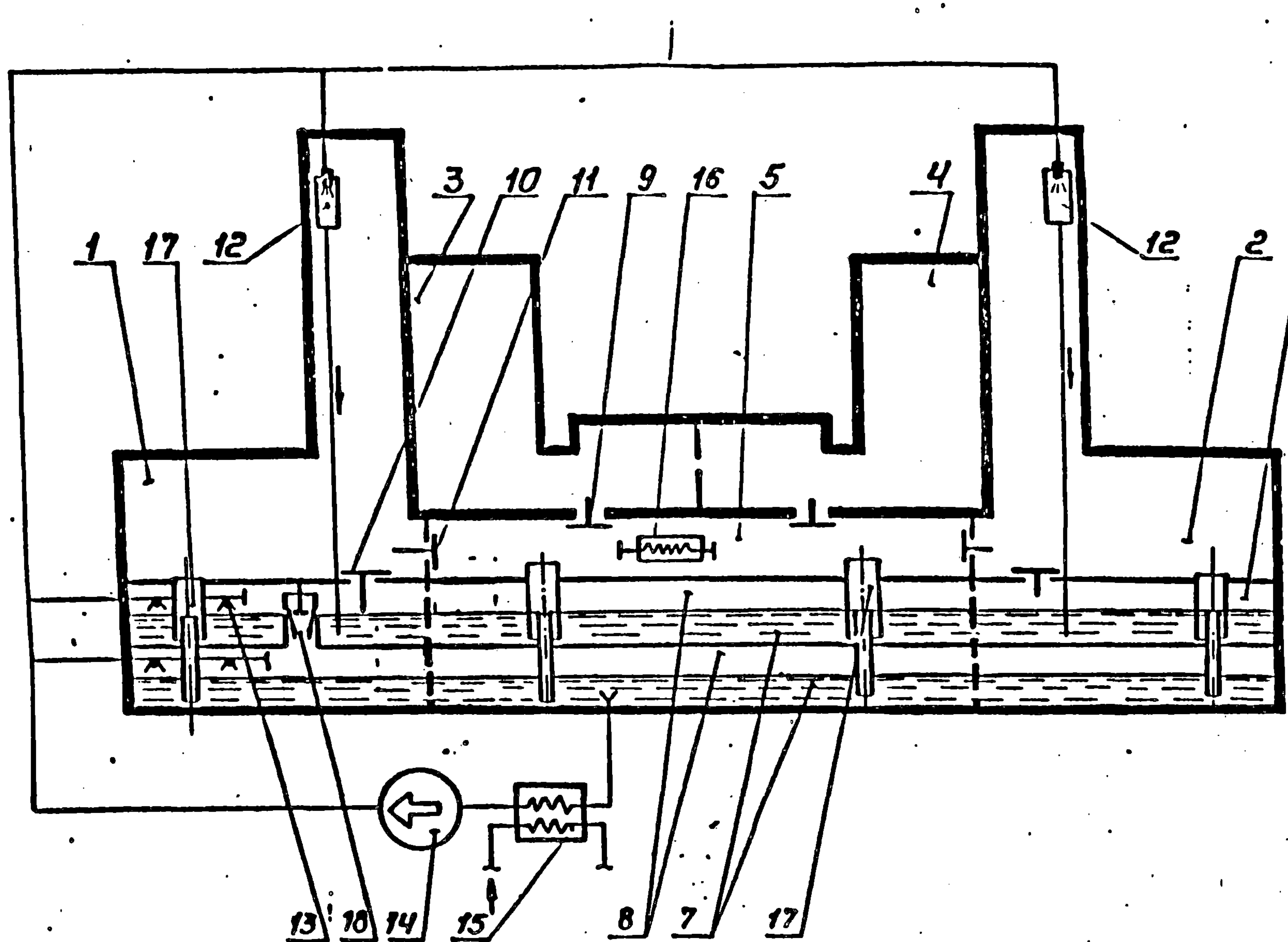


Рис. 2.47 Принципиальная схема системы локализации

Figure 2.47. Schematic of localization system

- the compartment of the steam distribution lane (item 5);

- the compartment of the two story pressure suppression pool - condensation device (BKU), some of which is filled with water (item 7) and the remainder with air (item 8).

  The sealed compartments are interconnected by valves of three types:

- check valves (item 9) mounted in the openings of the overlap which separates the RGK-NVK compartment and the steam distribution lane;

- relief valves (item 10) mounted in the openings of the overlap which separate the air space of the pressure relief tank and the sealed volumes;

- check valve panels (item 11) mounted in barriers which separate the steam distribution lane and sealed volumes.

The compartments of the sealed volumes and steam distributing lane are connected by steam discharge channels to the water volume of the pressure suppression pool - condensation device (item 17).

During normal operation the system of sealed compartments and the pressure suppression pool - condensation device operate in the waiting mode.

In emergency situations the system works as follows. With loss of integrity of the reactor loop component in one of the sealed volumes boiling coolant begins to enter it. Steam formation leads to pressure rise in the accident compartment. The check valves of the panels which connect the accident half of the sealed volume to the steam distributing lane (item 11) open when the pressure in them changes to exceed 0.002 MPa. When the pressure in the accident half of the volume reaches a value sufficient to discharge the liquid column from the steam discharge channels, a steam-air mixture begins to enter both stages of the condensation device at the same time. As it boils through the water layer. the steam condenses and air collects in the air volume of the condensation device compartment; when the pressure there exceeds 5 kPa, relief valves open which connect the air space of the condensation device compartment to the nonaccident sealed volume, and some of the air flows into this volume. Thus its volume in an emergency is used to reduce pressure in the accident half of the sealed volume. During this emergency check valves (item 9) remain closed.

If the seal of the reactor loop fails in the "RGK-NVK" compartment the pressure rise in it leads to opening of the check valves which connect the RGK-NVK compartments and the steam distributing lane (for pressure change exceeding 0.02 MPa). The steam-air mixture front he lane travels through the steam dumping channels into the water volume of the center part of the condensation device located under the steam distribution lane. The pressure raise in the air space of the condensation device causes the relief valves to open which connect the air space of the condensation device to two sealed volumes. Under conditions of this emergency situation the spaces of the two sealed volumes are used to reduce pressure in the accident compartment, and the valves of the panels (item 11) remain closed.

All sealed compartments of the system except for the BKU are lined with a shell of steel VSTZKP2 4 mm thick and are checked for local and integral seal. The BKU is lined with a shell of steel 08Kh18N10T 4 mm thick.

Results

of computations of pressure change in sealed compartments in an accident with rupture of the pressure header of the GTsN (Du 900 mm) in a sealed volume and in an accident with rupture of distributing group header (Du 300 mm) in the RGK-NVK compartment are shown in Figs. 2.48 and 2.49 respectively. As the graphs in these figures show the gauge pressure in the accident sealed volume does not exceed the maximum allowable value 0.25 MPa, and the gauge pressure in the accident RGK-NVK compartment does not exceed the maximum allowable value 0.08 MPa.

The system operates under conditions of a single failure of any passive element with moving parts (no active elements in the system).

2.10.2.2. Penetrations, doors

To prevent escape of radioactivity from sealed compartments the sealed barrier of the emergency localization system (walls, overlaps) at the site of its intersection with piping or electrical cable is equipped with a special sealed penetration.

Piping penetrations are designed to react to a jet emerging from a pipeline when it fully ruptures. Here the integrity of the penetration is not violated.

The design of the penetrations makes it possible to check their integrity both during installation anbd operation. The penetrations ensure integrity at a gauge pressure in the emergency localizations compartments up to 45 kPa, temperature up to 150oC and relative humidity up to 100%.

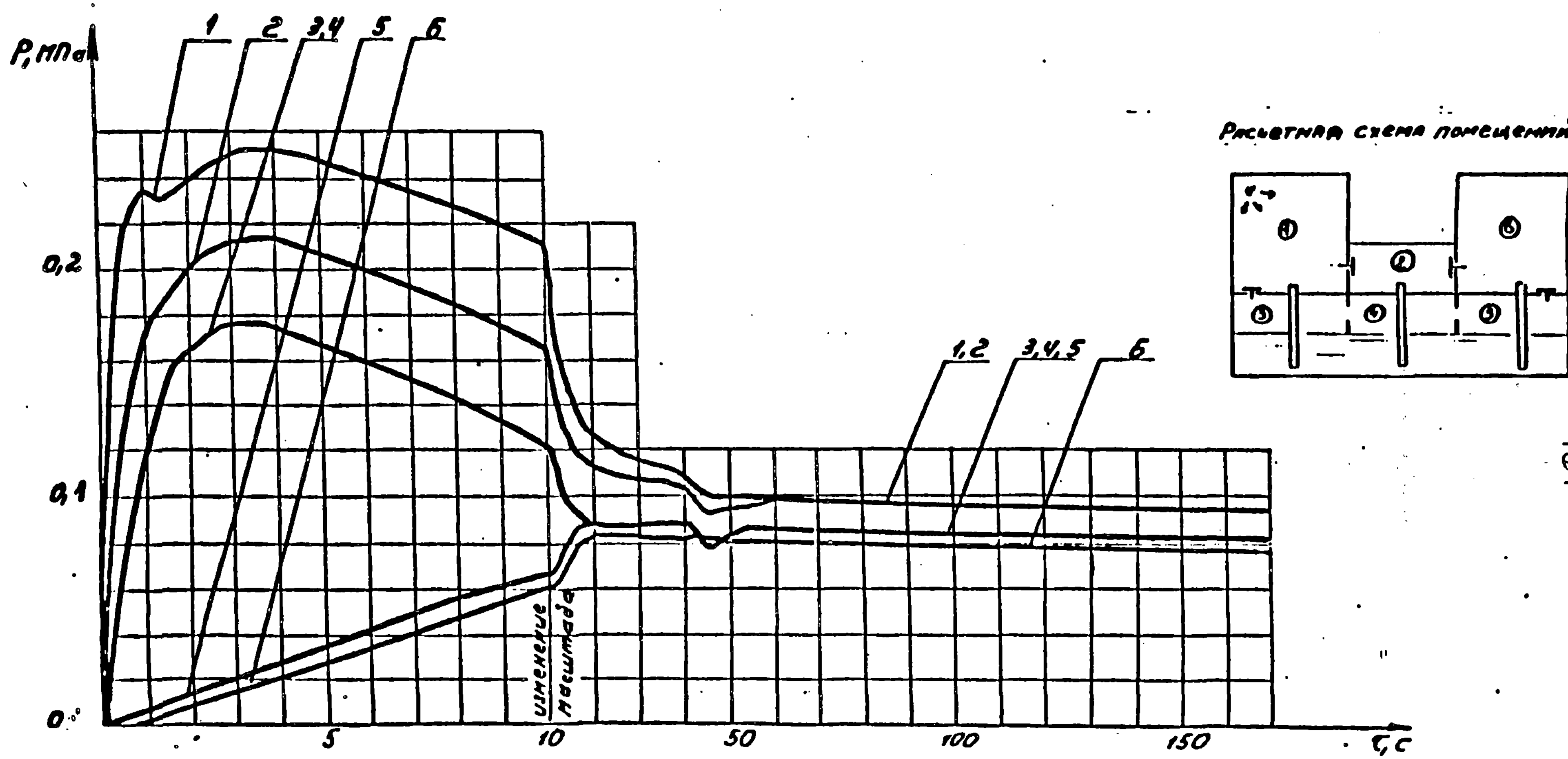


Figure 2.48. Pressure change in compartments of block 2 of the SAES and the four blocks of the Ku AES and Chernobyl with rupture of a pressure header.

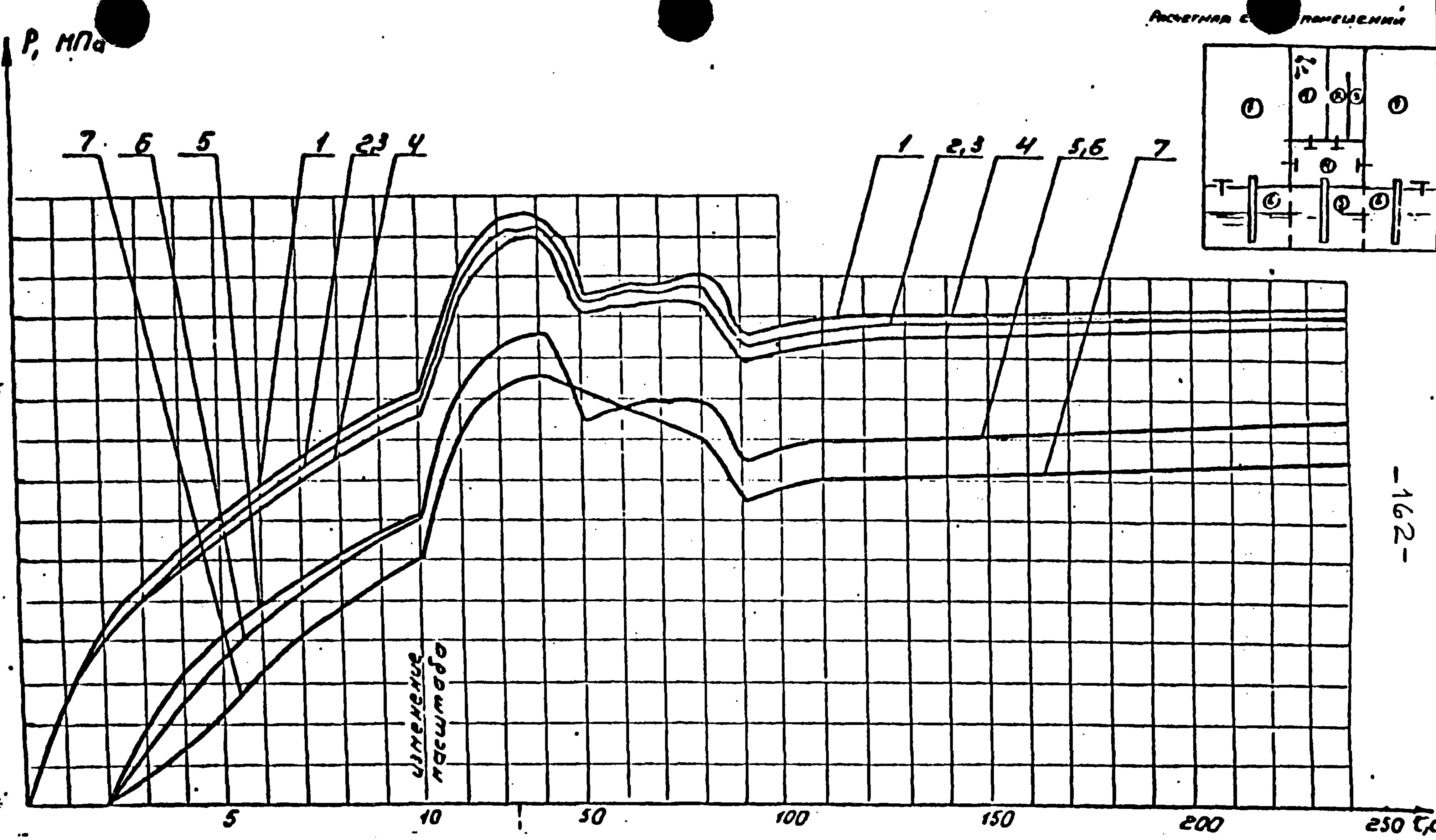


Figure 2.49. Pressure change in compartments of block 2 of the SAES and four blocks of the Ku AES and Chernobyl plant with rupture of RGK.

Sealed pipe penetrations designed to carry "hot" pipelines are equipped with a water or air cooling system to prevent overheating of concrete at the penetration site.

The sealed doors are designed to allow access of personnel to compartments of the emergency localization zone with the reactor shutdown and to ensure integrity of the compartments of the emergency localization zone with the reactor operating.

The sealed doors of the BKU ensure the required seal and operating capacity after any emergency situation, including MPA.

2.10.2.3. Cutoff and sealing fittings.

The system of cutoff and sealing fittings is designed to ensure that the emergency localization zone is sealed by cutting off service lines which connect sealed and unsealed compartments.

The design of the system is based on the following main principles:

- all service lines which intersect the sealing loop and which should be closed at the time of the accident to prevent escape of radioactive materials from sealed compartments are equipped with three successive cutoff elements;

- each pipeline not directly connected to the primary loop or the space of the sealed compartments is equipped with one cutoff element located outside the sealing loop;

- The location of the isolation valves which seal the compartments under emergency conditions is fixed on the BShchU (safety panel) and the RShchU from which they are remote controlled of necessary by the operator;

- the drives of the cutoff fittings mounted on a single line are powered from independent sources of category 1A reliable power supply system.

Special quick acting (10 - 15 s) isolating valves and check valves are used as cutoff and sealing fittings for the emergency localization compartments.

A preoperational check is carried out at the valve manufacturer's.

The isolating fittings are checked during power plant operation only with the unit shut down. All isolating valves are checked. Tests include checking serviceability and integrity.

Valves are closed automatically on maximum conceivable accident signals.

The system of cutoff and sealing fittings is designed such that any single failure in the system not lead to disruption of its functions.

2.10.2.4. Pressure suppression pool - condensation device.

The pressure suppression pool - condensation device is designed to condense steam formed:

- during an accident with loss of reactor loop integrity;
- when the main safety valves trigger;
- in case of leaks through the GPK under normal operating conditions.

In design terms the pressure suppression pool - condensation device is a two story reinforced concrete volume lined inside with a metal liner. The volume of the condensation device on each story is divided by longitudinal bulkheads into four lanes and by transverse bulkheads into three compartments: two side (under sealed volumes) and center (under the steam distribution lanes). The transverse and longitudinal walls of the pressure suppression pool have the necessary openings for water and air. The lower part of each story of the condensation device is filled with water. The thickness of the water layer on each story is 100 mm. The total water volume on both stories is 3200 m3, the volume of the air space 3700 m3.

Steam enters the water volume of the condensation device through steam dumping channels arranged uniformly over the entire area of the sealed volumes and a steam distributing lane. Each steam dumping channel is built in the form of a block of the pipe-in pipe type which ensures simultaneous and uniform delivery of steam to both stories of the condensation device. The number, diameter, and spacing of the steam distribution pipes and distance under the water surface were determined by tests on a large scale model and ensure complete steam condensation in the water volume of the condensation device, uniform heating of it and rapid pressure drop in the emergency sealed compartment during accidents with loss of integrity of the reactor loop.

The upper story of the condensation device is equipped with the necessary number of special vertical overflow pipes with a diameter of 800 mm (Fig. 2.47, item 28). The overflow pipes are designed to maintain the necessary level on the upper story and equalize pressure in the air spaces of both stories of the condensation device.

The water levels on both stories of the condensation device, temperature and chemical composition of the water are constantly monitored. The required chemical composition of the water is ensured by a bypass purification unit.

The localization system also includes a system for discharging heat from the BKU and sealed compartments and a hydrogen removal system.

Heat is removed from the sealed compartments of the localization system by two systems:

1. Sprinkler-cooling system;

2. Surface type condensers located in the steam distribution lane.

The sprinkler-cooling system is designed to do the following:

- cool and purify water in sealed volumes and the air space of the condensation device both during normal operation and in an accident;

- shutdown cooling of the water volume of the condensation device.

The main components of the sprinkler-cooling system are shown in Figure 2.47. Water is collected from the condensation device and sent over three legs of piping (each of which ensures 50% of system capacity) to heat exchangers (item 15) where it is cooled by service water and subsequently through pumps (item 14) to all system consumers:

- to ejection coolers (item 12) mounted in sealed volumes;

- to nozzles (item 13) located in the air volume on both stories of the condensation device.

The ejection coolers, a component part of the sprinkler-cooling system, are designed to circulate air in sealed volumes, cool it and remove radioactive aerosols and steam from it.

Air is collected from the upper (hottest) part of the sealed volumes, cooled by water jets, and sent to the lower part of the volumes. After contact with the air the cooling water is returned to the condensation device. The ejection coolers work continuously both during normal operation and accidents.

The nozzles of the sprinkler system located in the air space of the condensation device spray the cooling water, mix and cool the air. The pressure gradient on the nozzles is achieved by installing choke washers on cooling water feed headers. The nozzles operate continuously both during normal operation and an accident.

Surface type condensers (item 16) mounted in the steam distribution lane are designed to remove heat from sealed compartments during an accident with loss of reactor loop integrity by condensing some of the steam entering the steam distributing lane. Service water is the cooling medium. During normal operation the surface type condensers operate in the waiting mode and are engaged on MPA signal.

The serviceability of the system is tested on a large scale model during development.

The system begins to operate during a single failure of any active or passive component with moving parts.

2.10.2.5. Hydrogen removal system (SUV)

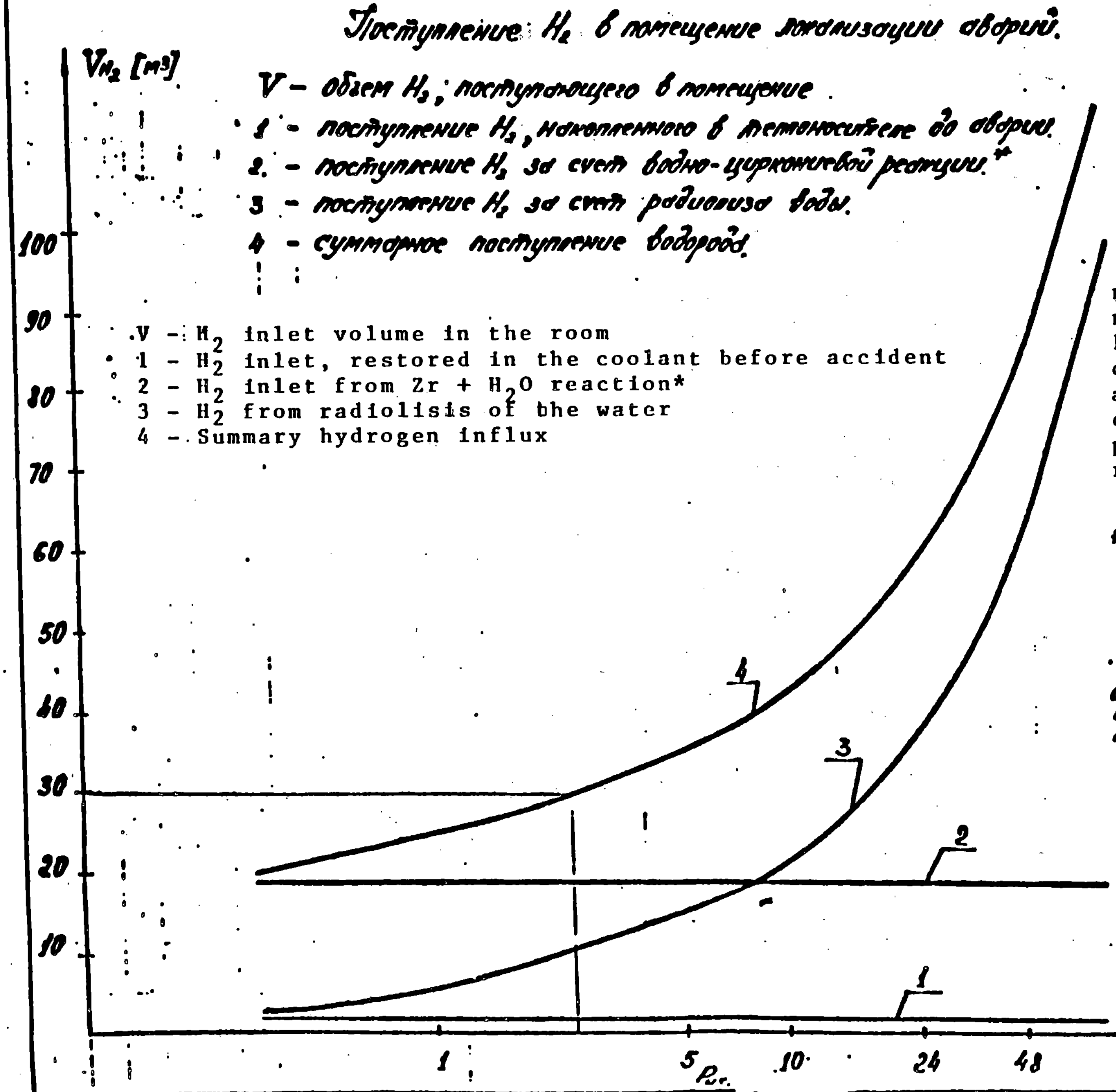
The system is designed to generate partial vacuum in emergency localization compartments, to measure the concentration of hydrogen which may enter these compartments with uncontrolled leaks from the KMPTs, and also under conditions of dumping steam from the GPK and during accidents associated with rupture of KMPTs pipelines, and removal of hydrogen when it is found.

During normal operation of the block hydrogen can enter SLA compartments with coolant leaks with a magnitude of 2 t per hour and with possible steam leaks through closed safety valves.

Hydrogen can also enter under conditions of short term discharge of steam when the GPK triggers and under piping rupture conditions.

The largest amount of hydrogen can enter the compartments under MPA conditions (hydrogen which has accumulated in the coolant and also formed during the accident due to radiolysis and reaction of zirconium with water). The total influx of hydrogen under these conditions is shown in Fig. 2.50.

With the existing standard of the lower limit of hydrogen explosibility in air - 4% (by volume), 0.2% (by volume) was adopted in the design as the reference value. To maintain this concentration under least favorable conditions 800 m3/hr air must be exhausted from the SLA compartments. This flow rate was also adopted for all other operating conditions of the block. The SUV (Fig. 2.51) includes the following: electric heater, contact apparatus, condenser, moisture separator, circulator.



*In regard with requirements, part of reacted Zr less than 1% of the weight of cladding of fuel affected group header channels. This figure presented with 3% reacted (for a margin)

* Согласно существующим нормативам доля прореагировавшего циркония составляет не более 1% массы оболочек твэлов каналов одного раздаточного группового коллектора. На данном графике приведено поступление водорода исходя из количества прореагировавшего циркония в размере 3% (в запас расчета).

Рис.2.50

Figure 2.50. Influx of H2 into [illegible]gency localization compartment.

This equipment is divided into three subsystems, each located in a separate volume. There is a backup of the active element - circulator in each subsystem. The protective-cutoff fittings are located in the individual volumes. A 3 x 100% backup principle is provided.

The gas-air mixture in the rated mode passes through the electric heater, contact apparatus (in the presence of hydrogen), condenser, moisture separator and by means of the circulator it is discharged into the atmosphere through the filter station.

On the MPA signal the protective-cutoff fittings close and the SUV equipment is turned off. After 2 - 3 hours (as hydrogen accumulates) the operator opens the protective-cutoff fittings and turns on the SUV circulators. Control is effected from the SUV panel. The post-accident operation configuration of the SUV is identical to operation in the rated mode, however the mixture is dumped through the radioactivity suppression unit. In addition the mixture can recirculate.

If necessary (according to gas analyzer readings) intensified exhaust from any SLA compartment can be set up by turning on the backup circulator or backup subsystem.

Nitrogen feed is provided to flush the SUV equipment and extinguish fires.

The hydrogen concentration in all SLA compartments is automatically monitored on a continuous basis by the gas analyzers. Forced signalling (acoustic and light) regarding a hydrogen concentration rise in SLA compartments appears on the SUV and BShchU - O panel. In addition test measurements of hydrogen concentration in SLA compartments by manual sampling using chromatographs is possible. Flow rates, temperature and radioactivity are measured in the hydrogen removal system itself. All readings appear on the SUV panel.

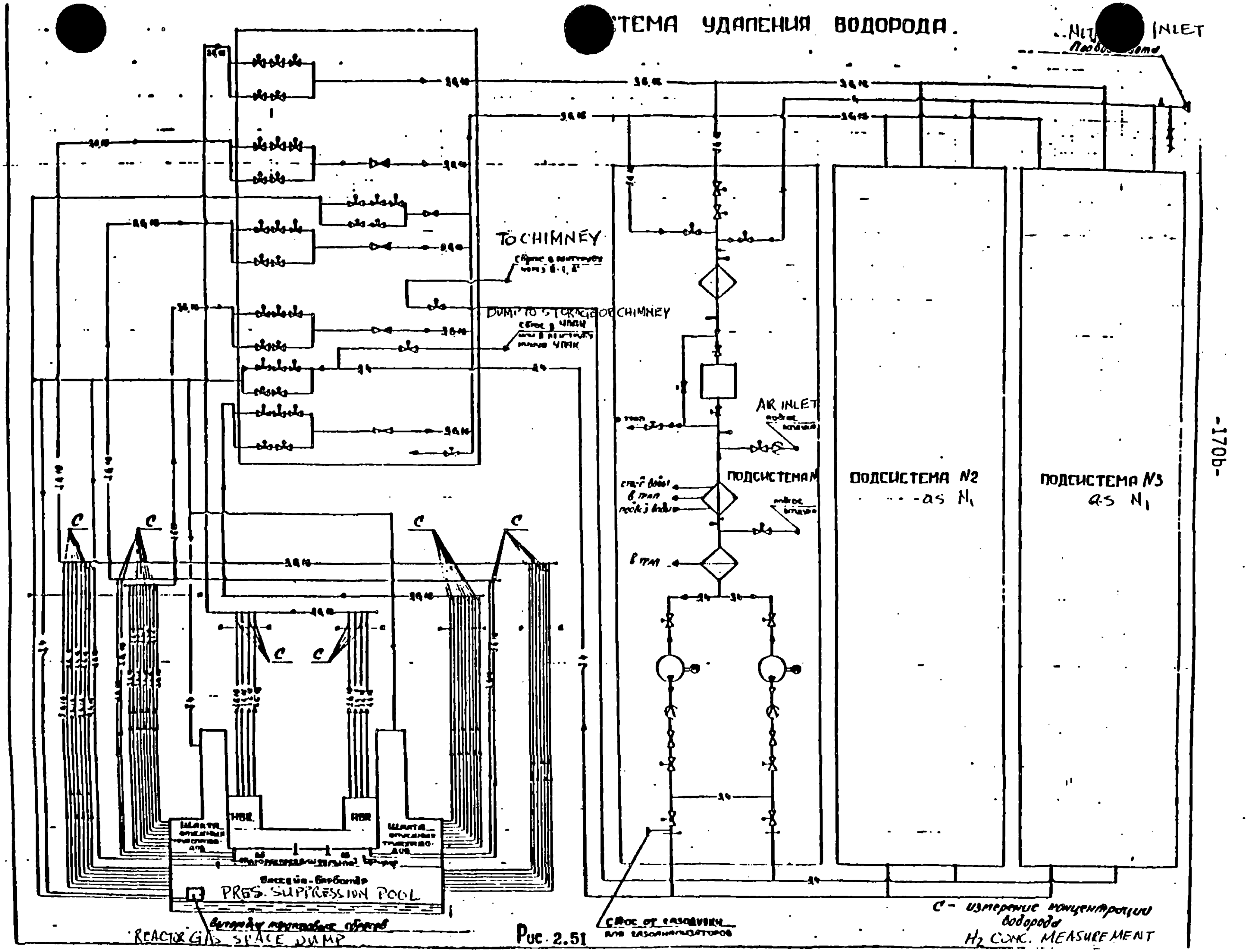


Figure 2.51. Hydrogen removal system.

The control and monitoring system is executed over three independent channels. Power is supplied to SUV equipment from the power supply sources of the corresponding safety subsystems.

Cooling water is sent from the sprinkler-cooling system to the SUV condensers.

2.10.3. Support safety systems.

2.10.3.1. Electric power supply system for power plant in-house needs

Power plant in-house consumers are divided into the following groups depending on requirements imposed on power supply reliability:

- first group - consumers which do not permit interruption of the power supply or allow interruption of the power supply from fractions of a second to several seconds under any conditions, including complete disappearance of AC voltage from working and backup transformers for in-house needs and which require that power be available after triggering of reactor AZ;

- second group - consumers who allow interruption of power supply from dozens of seconds to dozens of minutes under the same conditions and which require that power be available after triggering of reactor AZ;

- third group - consumers which do not require that power be available under conditions of disappearance of voltage from working and backup transformers for in-house needs and during normal block operation which allow interruption of power supply for the period of transition from working to backup transformer for in-house needs.

There are two independent power supply sources which back up one another to supply consumers of in-house power plant needs; normal working and backup power from working and backup transformers for in-house needs.

There is additional power from a third independent emergency source for consumers of the first and second groups.

The following are provided as emergency power supply sources:

a) storage battery with static converters for consumers of the first group;

b) automatic diesel generators for consumers of the second group.

Figure 2.52 shows a diagram of in-house needs.

2.10.3.2. Schematic of in-house needs of 6 kV consumers of the third group.

The consumers of the third group include reactor coolant pumps (GTsN), feedwater pumps (PN), the first and second condensate extraction pumps (KN1, KN2), mechanisms of auxiliary systems of the reactor and turbine room and other systems which allow normal operation of the block.

There are 6 kV and 380/220 V, 50 Hz networks which are used to supply electricity to consumers of the third group in the nuclear power plant; working and backup transformers for in-house needs are installed for this purpose.

Six kV in-house consumers of the unit during normal operation are supplied from two working transformers for in-house needs of 63 MV each, voltage 20/6.3 - 6.3 kV with two split windings. The in-house transformer is closed coupled between two series connected breakers to the generator voltage circuits of the unit. Two working sections of 6 kV each for supplying in-house consumers of the unit are connected to each unit transformer for in-house needs.

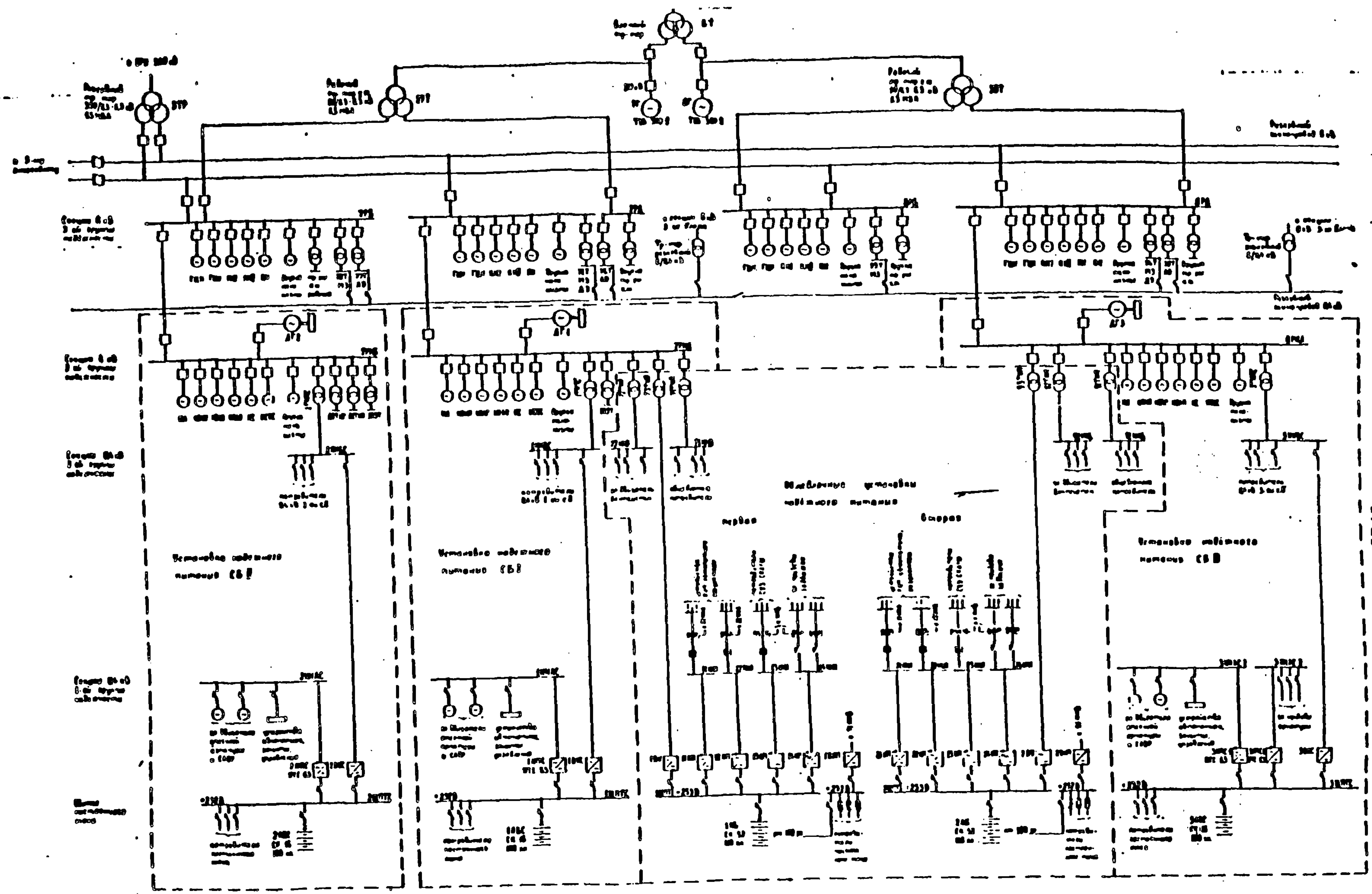


Схема собственных нужд 4-го энергоблока Чернобыльской АЭС

Figure 2.52. Diagram of in-house needs of the fourth block of the Chernobyl power plant.

The presence of two breakers in the circuit of each generator makes it possible to use the working transformers for in-house needs to start and stop the block in case of malfunctions in the generator circuit, maintain electricity supply for in-house needs when the block is shut down for technical reasons, and also for all types of electrical accidents in the unit above the generator breakers, in particular short circuit in the unit transformers.

The circuit with two breakers makes it possible to use rundown of the turbogenerator to supply power to feedwater pumps which ensure water supply to the reactor zone for the first 45 seconds from the start of a maximum conceivable accident in case of loss of power for in-house needs from the high voltage network. Separate rundown of turbogenerators is ensured by turning off the 20 kV breakers from the unit transformer side.

In this case the voltage on the generator changes in proportion to its rpm by connecting a turbogenerator of a special "rundown unit" to the excitation regulator which keeps rotor current constant on the generator when frequency drops.

The rundown unit is engaged on the maximum conceivable accident and the turbine check valve fit signal.

The unit transformers for in-house needs are backed up by a 63 MVA backup transformer connected by an overhead line to the 330 kV ORU.

The 6 kV consumers of teach turbogenerator are connected to the corresponding sections of the unit transformer for in-house needs and the consumers of the reactor section and general unit consumers are distributed uniformly between sections of the two unit transformers for in-house needs; electric motors of mechanisms which back up one another are connected to different sections.

2.10.3.3. Emergency power supply system

Power is supplied to consumers of the first and second groups from the emergency power supply network for which self-contained sources, i.e. storage batteries with static converters and diesel generators are used as power supply sources in addition to working and backup transformers.

Consumers of the first and second groups are divided into two consumers of plant safety configurations and "general unit consumers" who require that power be available even in complete cutoff of in-house power plant needs.

2.10.3.4. Schematic diagram of 0.4 kV emergency power supply system of the first group and DC networks of the safety systems

Consumers of the first group of safety systems include cutoff fittings of the emergency localization system and hydrogen removal system, quick acting valves and gates on lines of the reactor emergency cooling system (SAOR), and emergency system automatic protective and monitoring devices.

There are three independent power supply sources (storage battery with static inverter converters, 6 kV and 0.4 kV sections for in-house needs) to supply electricity to the consumers of the first group of each subsystem. The DC panel of the safety subsystem (ShchPTS) is powered from a rectifier connected to a 0.4 kV emergency power supply section of the second group of the NNBS, and when voltage disappears on this section from a storage battery operating in the "buffer" mode.

The AC 0.4 kV consumers of the first group are connected to the 0.4 kV section of the NNAS which is powered from the DC panel via static inverter converters.

During normal unit operation monitoring, automatic and control devices of the corresponding safety subsystem are connected to the DC panel (ShchPTS) and the 0.4 kV section (NNAS) of each subsystem, and under conditions of the maximum conceivable accident an additional load is connected, i.e. electric drives of the SAOP slides and valves and emergency localization system. To prevent overloading of the inverter converters above allowable limits by the starting currents of the electric drives of the slides the latter are connected by stages on the MPA signal.

2.10.3.5. Configuration of the general unit emergency power supply 0.4 kV network of the first group and direct current

The general unit consumers of the 0.4 kV emergency power supply network of the first group include the "SKALA" system, SUZ, dosimetric monitoring, the KIP equipment and automatic reactor systems, turbines and generator, and the quick acting reducing valves.

There are two general unit emergency power supply systems to supply power to consumers of the general unit emergency power supply and DC network; each of the units includes: power supply sources-storage battery and static inverter converters, DC panel (ShchPT), 0.4 kV emergency power supply system of the first group (NNA), the six kV and 0.4 kV in-house supply section.

The DC panel (ShchPT) of each general unit emergency power supply unit is supplied from a rectifier connected via 6/0.4 kV transformer to the 6 kV emergency power supply section of the second group, and when voltage disappears on this section from a storage battery operating in the "buffer" mode.

The 0.4 kV consumers of the first group are connected to the NNA sections via thyristor switching devices TKYeO. The NNA sections are supplied via static inverter converters from the ShchPT.

Each consumer of the general unit emergency power supply network has two power supply sources. Either the network or another inverter converter is used as the second source.

For consumers which do not allow a power supply interruption longer than 10 - 20 ms ("SKALA", SUZ) the transfer to the reserve power supply source is done by a thyristor switching commutation device TKP which allows transfer of consumer power supply from one source to another in 10 ms. For consumers who allow power supply interruption up to 100 - 200 ms, relay contactor switching devices are provided.

Devices with technological backup (feeder "A" and "B", "SKALY", 1000 Hz converters and 400 Hz "SKALY", AZ panel sets and so forth), are supplied from one of the emergency power supply units.

Devices without technological backup (KIP devices, automatic units, control devices and so forth) are supplied with power from the two emergency power supply units.

DC consumers (protection, signalling panels and so forth) are supplied from two DC panels.

Power is switched from one panel to the other manually.

2.10.3.6. Schematic of 6 kV and 0.4 kV emergency power supply in-house needs of the second group

Consumers of the second group of safety systems include mechanisms of the reactor emergency cooling system and the accident localization system.

General unit consumers of the second group include mechanisms of auxiliary turbogenerator systems and selected reactor plant systems (plant loop, cooling system of the fuel cooling pond, the flushing and shutdown cooling system and so forth).

There are three 6 kV and 0.4 kV emergency power supply sections (according to the number of safety subsystems) to supply power to consumers of the second group. General unit consumers are distributed by sections of safety subsystems.

5500 kW diesel generators were installed as a self-contained power source for the 6 kV emergency power supply sections on the IV unit of the Chernobyl power plant. The start time of the diesel generators was 15 seconds.

The diesel generators were loaded in stages. The start time of each stage was 5 seconds. The diesel generators (GD) with step load pickup are started automatically on the MPA (maximum conceivable accident) signal and on the current cutoff signal.

When one of these signals is generated commands are sent by the automatic start circuit of the DG with step load pickup to

- DG start;

- disconnect both section breakers which connect the working 6 kV in-house needs section to the emergency power supply section;

- disconnect the load on the 6 kV emergency power supply section ("clear the section");

- prohibit AVR mechanisms connected to this emergency power supply section.

After turning of the DG and connecting it to the section the breakers of the in-house needs mechanisms are automatically engaged in stages after 5 seconds according to the adopted stepped load pickup graph (Fig. 2.53).

Depending on the incoming signal the circuit automatically carries out stepped connection of corresponding mechanisms necessary in MPA or when in-house consumers are cut off from power.

For 0.4 kV consumers of the second group there are 0.4 kV sections of the safety systems (NNBS) and 0.4 kV general unit sections independent of them (NNB).

Each section is powered from the corresponding 6 kV section of the safety system via the 6/0.4 kV transformer.

The number of 0.4 kV NNBS sections corresponds to the number of plant safety subsystems.

The 6/0.4 kV emergency power supply transformers of the second group are an undetachable stage of the load on the diesel generators.

There is no mutual backup between the 6 kV and 0.4 kV sections since the consumers themselves have a backup.

2.10.4. Safety control systems

The safety control systems are designed for automatic engagement of devices of the protective localizing and support safety systems and monitoring of their operation.

№ п/п	Наименование потребителей надежного питания	Подключено на одну подсистему безопасности (шт)	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)	Расчетные величины: Расчетная мощность (кВт)	Пусковая мощность (кВА)	Скольжение $S_н$ (о.е.)	cos $φ_{пуск}$	Количество пусковых двигателей (шт)
1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18
1	Трансформаторы надежного питания и с.н. ДЭС	4	-			100	200						800	1600			
2	Насос технической воды аппарата - НА	1	АВ-18-89-16	1250	1100	160	830	94,1	0,8	5,2	0,8	370	1170	1900	0,015	0,22	1
3	Насос САОР низкого давления - НСНД	1	2АЗМ 800/6000	800	720	88	460	95,5	0,91	5,2	1,1	2970	755	1540	0,01	0,3	1
4	Насос спринклерной охладительной системы - НСОС	1	А-113-4М	250	241	28	144	92	0,89	5,0	1,3	1480	241	405	0,013	0,34	1
5	Насос САОР аварийной подпитки - НОАП	2	2АЗМ 800/6000	800	720	88	459	95,5	0,91	5,2	1,1	2970	755	1540	0,01	0,3	2
6	Насос аварийного охлаждения бассейна 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	1
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	1
11	Трансформатор надежного питания - ТНП	1	ТСЗС-1000	800		68	136						560	1120			1
12	Насос расхолаживания бассейна выдержки - НР	1	2АЗМ 500/6000	500	204	54	330	95,5	0,92	6,0	1,2	2980	223	1050	0,01	0,31	–
13	Насос борного регулирования - НБОРГ	1	А-113-4М	250	200	28	144	95	0,89	5,0	1,3	1480	210	405	0,013	0,34	–

№ п/п	Режим максимальной проектной аварии. 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	Количество пускаемых двигателей
1	19	20	21	22	23	24	25	26	27	28	29	30	31	32	33
1	1600/200											800/100			
2		2700/930										1970/260			1
3			3510/270									2225/348			–
4				3130/482								2966/377			1
5					6096/1298							4476/558			–
6						4831/698						4703/583			1
7							—					—			–
8								—				—			1
9									—			—			1
10										5108/1211	–	4917/611,9			1
11											6037/747	5437/679			1
12															–
13															–

№ п/п	Режим обесточивания собственных нужд. 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	Примечания
1	34	35	36	37	38	39	40	41	42	43	44	45	46	47	48
1	1600/200											800/100			
2		2700/930										1930/260			
3			—									—			
4				2333/404								2211/289			
5					—							—			
6						2616/413						2438/318			
7							—					—			
8								2843/461				2648/346			
9									4188/807			1403/415			
10										3808/379		3612/464			
11											4937/599	4177/512			
12															[illegible]
13															В ступенчатом пуске не участвует

т. г

Данная таблица составлена для одной подсистемы безопасности. Для двух других подсистем таблица – аналогична

Figure 2.51.

Each of the three safety subsystems has its own independent control safety system (USB).

The USB forms the MPA signal when the pressure rises in the PPB compartment, in the NVK compartment or in the BS department to 5 kPa with confirmation of a level drop in the BS to 700 mm from the nominal level or a drop in the gradient between the pressure header of the GTsN and the BS to 0.5 MPa.

To increase reliability, all three USB are built independently of one another, i.e. each USB has its own equipment and power supply, individual compartments for equipment and cable routes.

To form the pressure rise signal there are four transducers each in the PPB, BS, and NVK compartments. A signal is formed when two or more transducers trigger.

The level drop signal in the BS and the signal which indicates a pressure gradient drop between the pressure header and the BS is formed when any of these two transducers trigger.

The MPA signal is formed for each half of the reactor independently.

When the MPA signal appears the USB generates output actions to switch the corresponding valves of the safety system, and to turn on the diesel generators and mechanisms for step load pickup.

The design provides for the possibility of remote control of the safety system; for this reason control switches for each USB are mounted on panel BShchU-0.

In this case the emergency half is selected automatically; to do this an information part independent of the USB consisting of an emergency protection shaping circuit is used for technological reasons.

To monitor serviceability of the USB there is light-acoustic signalling of instrument malfunction.

The safety system is monitored and controlled from safety panels located in the zone of the operational BSchchU loop and on the backup control panel.

The pump control elements of the SAOR, the SLA, the safety system valves, the devices which monitor SAOR water flow rate into the reactor and others are located on the safety panels.

Figure 2.54 shows a structural diagram of the USB.

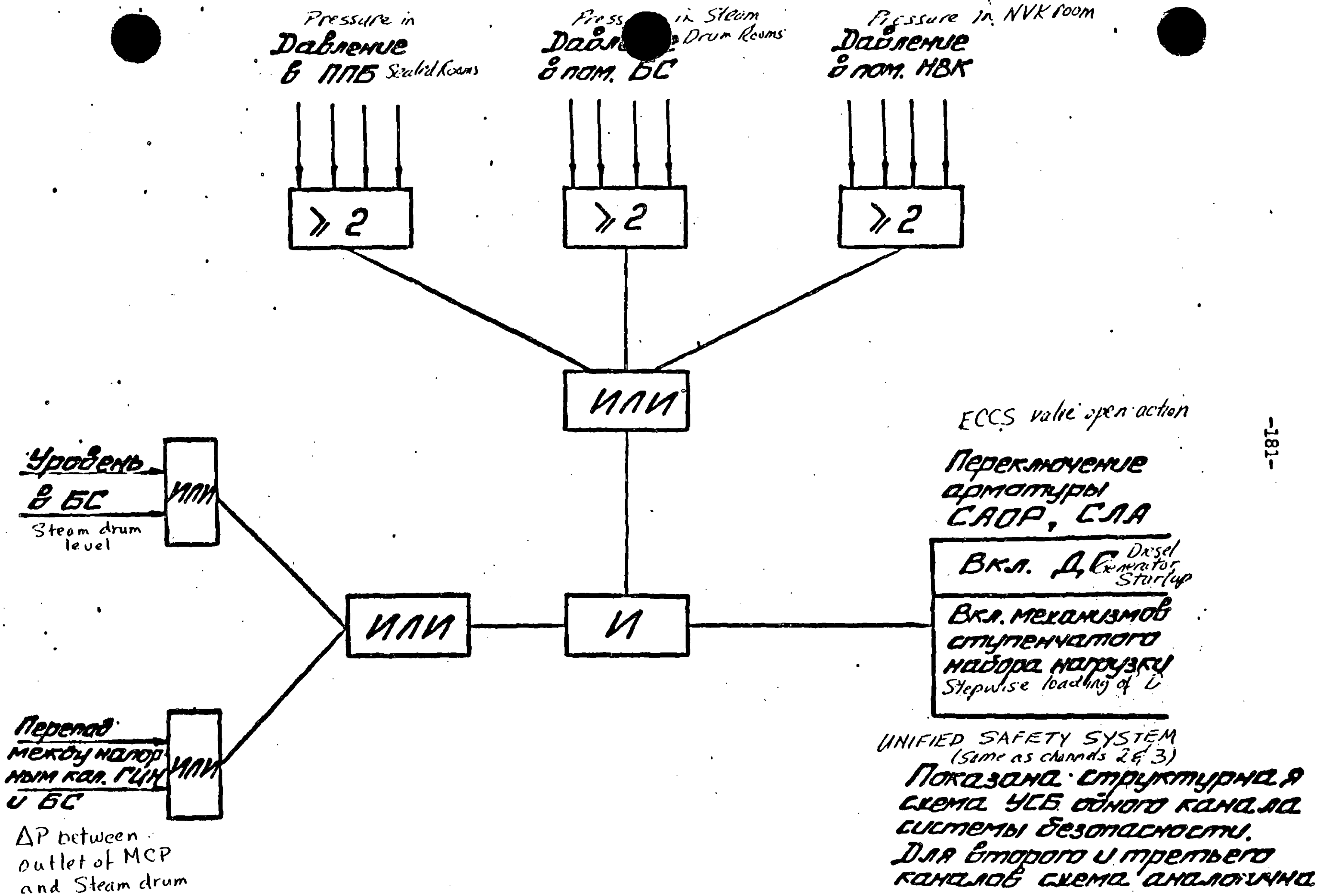


Рис. 2.54

Figure 2.54. Structural diagram of the USB

2.11. Other systems important for safety

2.11.1. MPTs loop

The MPTs loop and its main parts are described in sections 2.6 and 2.7.

2.11.2. SUZ channel cooling system

The SUZ cooling system is designed to maintain temperature conditions of these channels, control elements and SUZ servo drives.

The system performs the following functions:

- maintains a temperature of 40°C at the cooling water input into the control channels;

- discharges 28.1 MW of heat from the channels of the control elements and SUZ servo drives;

- ensures cooling of the channels of the control elements and SUZ servo drives by a nominal flow rate for about 6 minutes when the pumps are not operating;

- maintains an explosion-proof hydrogen concentration under all operating conditions;

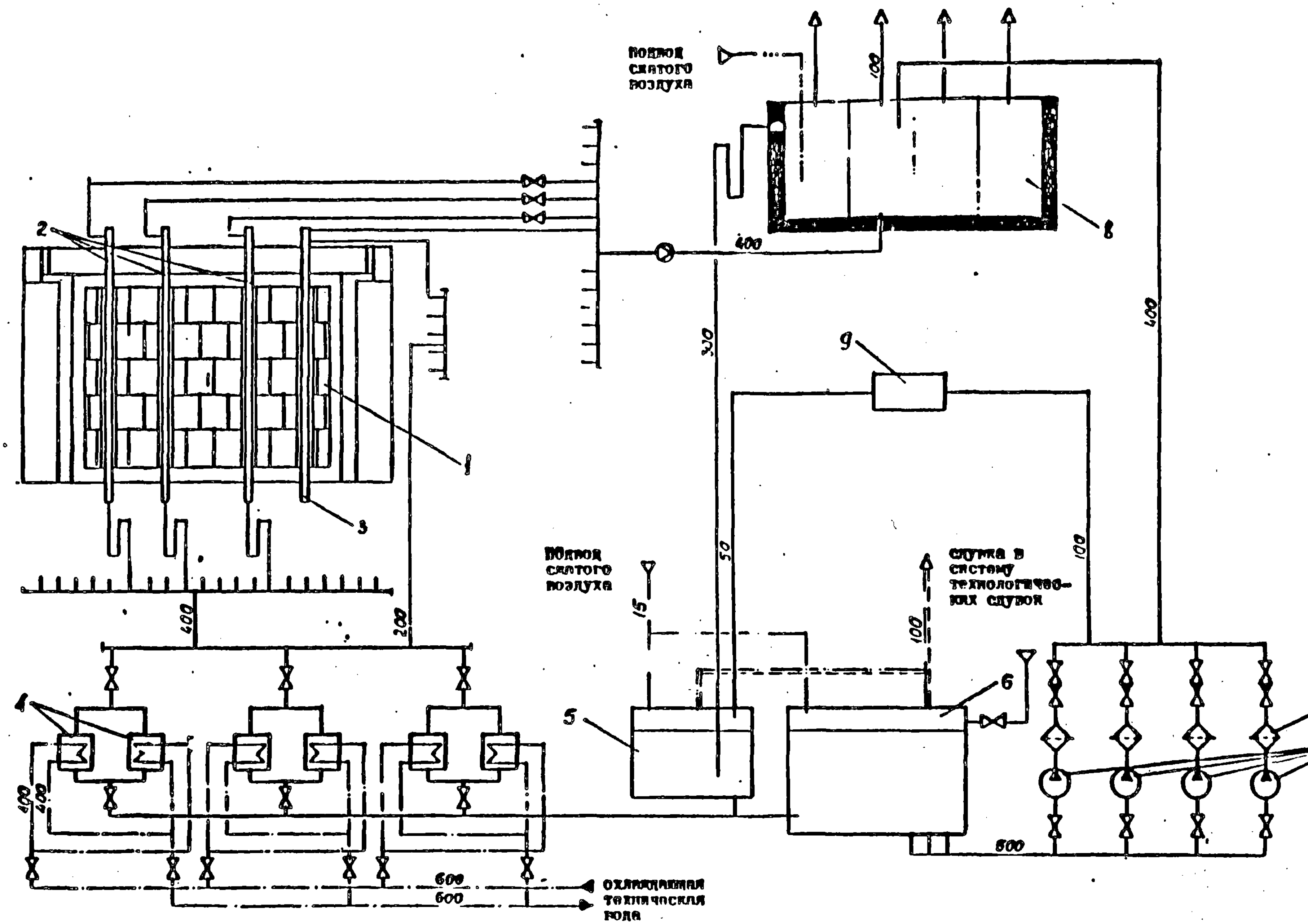
- maintains the necessary quality of water cooling the channels and SUZ servo drives;

- ensures emergency protection of the reactor when problems occur in the cooling system.

These functions are performed with allowance for a single failure in the system of an active or passive component with moving or mechanical parts.

Figure 2.55 shows a schematic of the SUZ channel cooling system.

The system is a circulation loop which operates by gravity



LEGEND ON Pg 183a

Рис. 2.55 Принципиальная схема системы охлаждения каналов СУЗ, КД, ДКЭ и КОО.

1 – реактор; 2 – каналы СУЗ, КД, ДКЭ; 3 – каналы КОО; 4 – теплообменники СУЗ; 5 – бак сливной; 6 – бак циркуляционный; 7 – насосы СУЗ; 8 – бак аварийный СУЗ; 9 – установка байпасной очистки; 10 – фильтры

Figure 2.55. Schematic of the channel cooling system of the SUZ, KD, KE, and KOO.

Legend: 1 - reactor; 2 - SUZ, KD, DKE channels; 3 - KOO channels; 4 - SUZ heat exchangers; 5 - drain tank; 6 - circulation tank; 7 - SUZ pumps; 8 - SUZ emergency tank; 9 - bypass purification unit; 10 - filters.

Water from the upper emergency reserve tank flows by gravity into the pressure (distributing) header and is distributed through the channels. The channels contain components of the reactor protection and control system, sleeves with fission chambers and energy release monitoring transducers. Some of the channels are used to set up a flow of water which cools the graphite of the side reflector.

As it passes through the SUZ channels, the water of the cooling loop gives off heat to the service water and the system heat exchanger. Depending on the service water temperature and the degree of fouling of the heat exchanger surface two heat exchangers ensure heat discharge with two in reserve.

Following the heat exchangers water flows into the lower tanks of the system. A level which ensures stable operation of pumps under stationary and transient conditions is automatically maintained in the tanks. The total volume of the lower tanks makes it possible to accept the entire volume of water in the system with the pumps stopped.

Four pumps are used to feed water from the lower tanks to the emergency reserve tank. The output of each pump is roughly 700 t/hr with a head of roughly 0.9 MPa. Two of them are working, two are backup.

Measures are provided to reduce the probability of all pumps failing for a common reason (pumps are located in different compartments, have independent power sources, and so forth).

The capacity of the working pumps exceeds the capacity of the cooling system; therefore some of the water is continually dumped into the lower tanks of the system from the emergency reserve tank (the level in the emergency reserve tank is kept at the overflow mark).

Hydrogen is released from the water of the SUZ cooling system due to radiolysis in the reactor core.

To preclude formation of an explosive hydrogen concentration the space above the water in the upper and lower tanks is constantly vented with monitoring of hydrogen content of the water in the SUZ cooling system and the space above the water in the system tanks.

The emergency reserve water tank is linked to the atmosphere by four breather pipes from its upper points. In addition the space above the water in the tank is continuously flushed with compressed air. During a failure in the compressed air feed system, the space above the water is vented by ejecting air by means of debalanced water which is continually dumped from the emergency reserve tank into the lower tanks through an overflow pipe. When the system stops all water from the emergency reserve tank drains into the lower tanks.

The lower tanks of the system are constantly flushed with compressed air; air ejected from the emergency reserve tank also flows into them. In addition since the tank is under a slight vacuum, air from the compartment flows into them through a special line with a valve. Negative pressure in the tanks and discharge of air supplied for flushing are accomplished by a special tank vent system.

When any of the flushing (vent systems) fail, those still in operation maintain a safe hydrogen concentration.

To maintain the necessary quality, water in the cooling system is constantly purified. Water is supplied for purification from the pressure header of the pumps and is returned to the lower tanks.

When failures occur in the cooling system (drop in the level in the emergency reserve tank, decrease of water flow rate), signals for emergency reactor shutdown are formed.

Monitoring of technological parameters and control of the SUZ channel cooling system are accomplished by operator personnel from the BShchU.

The system was fully checked during startup operations and unit operation.

2.11.3. Purge and shutdown cooling system

The purge and shutdown cooling system (SPiR), Fig. 2.56, is designed to cool the KMPTs purge water collected for purification with subsequent heating before returning to the KMPTs under nominal conditions; and for reducing the temperature of the KMPTs water to the required level in the shutdown cooling mode.

In the nominal mode KMPTs coolant at a rate of 200 t/hr (100 t/hr from each KMPTs loop) under the GTsN head is sent to the regenerative heat exchanger, where it is cooled from 285 to 68 degrees C by discharging heat to the cold backflow, and then further cooled by water of the closed cooling water system in a purge supercooler down to 50oC and is sent to the loop water purification system. As it passes through the regenerative heat exchanger in the opposite direction the purified water is heated from 50 to 69oC and returned to the steam separators through mixers on feedwater pipelines. It should be noted that either of the two supercoolers in the SPiR can be operating in this mode.

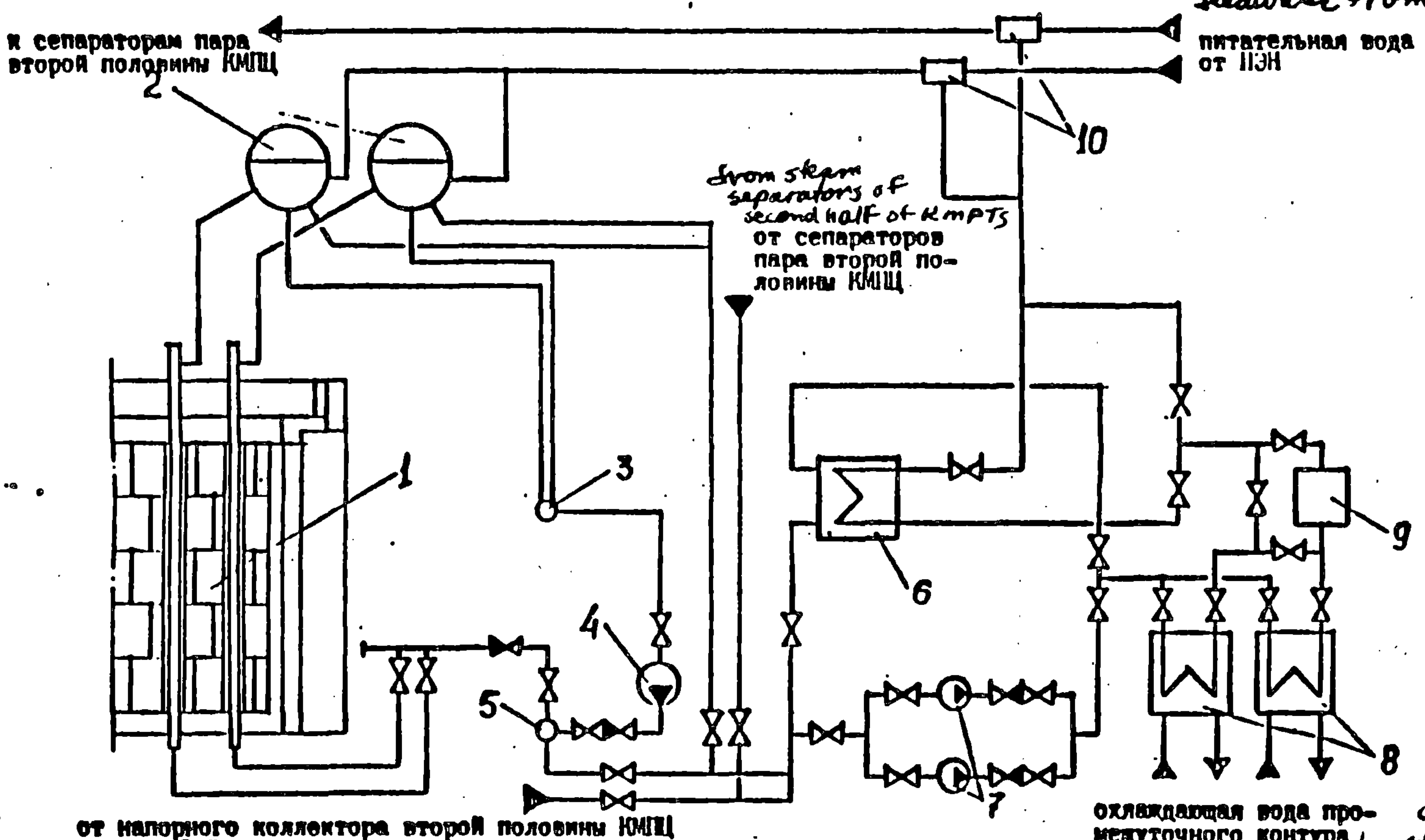
In the unit shutdown cooling mode the SPiR reduces the water temperature in the KMPTs from 180oC to the value required by unit repair conditions. Circulation is accomplished over the steam separator - shutdown cooling pumps - large supercooler - steam separators route. The SPiR can also be used to discharge residual reactor heat when power to satisfy in-house unit needs fails. The operating configuration in this mode is the same as in the shutdown cooling mode.

Figure 2.56. Schematic of purging and shutdown cooling system

Legend: 1 - reactor; 2 - steam separator; 3 - intake header; 4 - reactor coolant pump; 5 - pressure header; 6 - regenerative heat exchanger; 7 - shutdown cooling pumps; 8 - purge superheaters; 9 - unit for purifying the water of the reactor circulation loop; 10 - purge and feed water mixers.

to steam separators of second half of KMPTs

feedwater from PEN



cooling water of intermediate loop

from pressure header of second half of KMPTs

Рис. 2.56. ПРИНЦИПИАЛЬНАЯ СХЕМА СИСТЕМЫ ПРОДУВКИ И РАСХОЛАЖИВАНИЯ

1 – реактор; 2 – сепараторы пара; 3 – всасывающий коллектор; 4 – главный циркуляционный насос; 5 – напорный коллектор; 6 – регенератор; 7 – насосы расхолаживания; 8 – доохладители продувки; 9 – установка очистки воды циркуляционного контура реактора; 10 – смесители продувочной и питательной воды

2.11.4. Gas loop system

Figure 2.57 shows a schematic of the system.

In the nominal mode the gas loop system operates as follows: a nitrogen-helium mixture leaving the reactor masonry passes through the system which monitors the integrity of the fuel channels (KTsTK) where the temperature is monitored channel by channel and the moisture content of the nitrogen-helium mixture is monitored group by group.

As it passes through the KTsTK system the mixture travels through the condensers, heaters and filters in which iodine vaporous precipitate to the intake of the helium purification unit compressor where the hydrogen, oxygen, methane, carbon dioxide gas, carbon monoxide and ammonia are removed from the mixture down to the concentration which allows normal reactor operation.

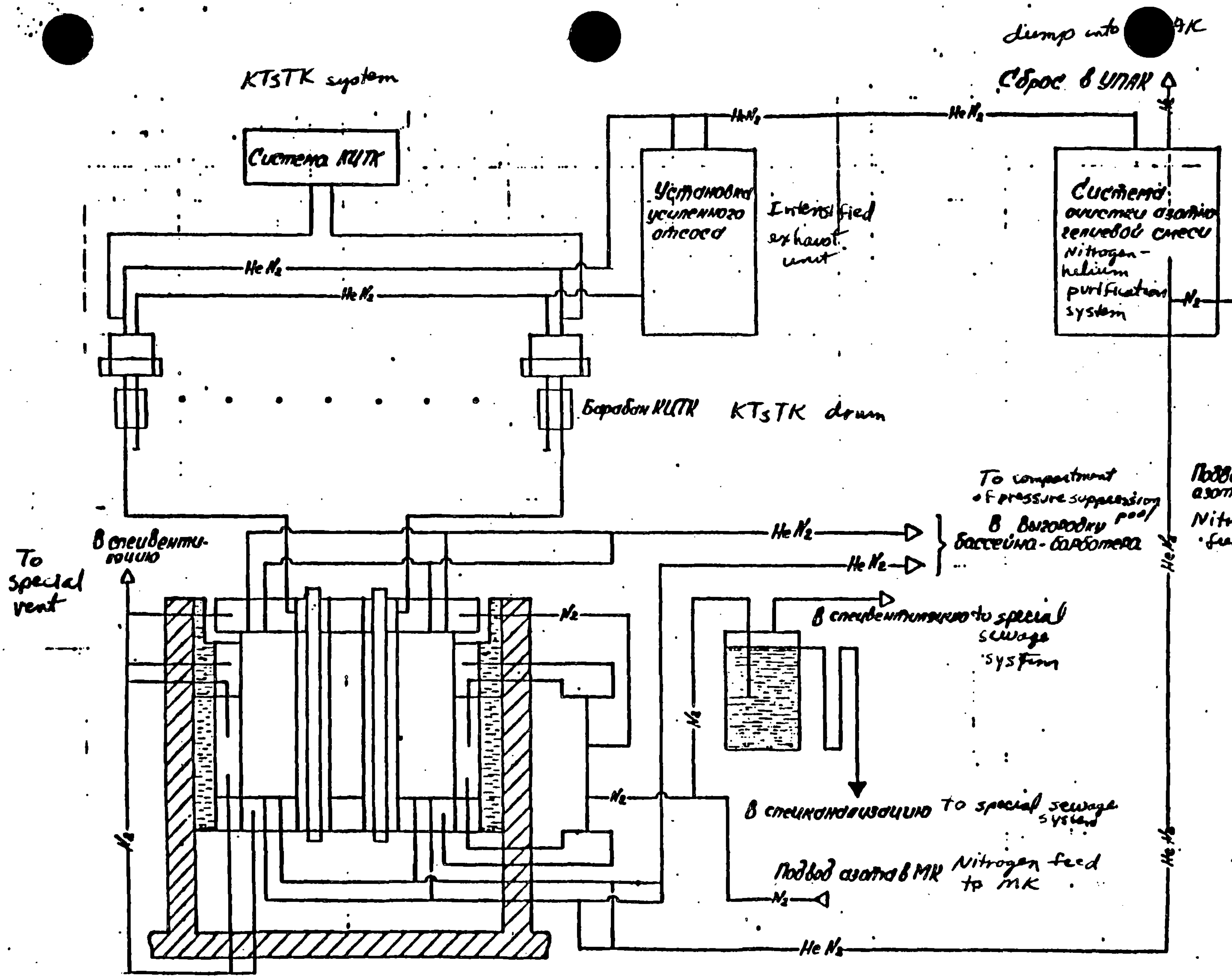


Fig ~~Рис~~ 2.57 Система газового контура
Gas Loop System

~~(text cut off~~ . Radioactive argon-41 is removed in holding tanks.

Following the purification system the mixture is returned to the reactor masonry. A hydraulic seal which prevents the pressure rising above an allowable level, i.e. above 1 - 3 kPa is mounted on the mixture inlet line into the masonry..

To reduce helium leaks from the reactor masonry into the reactor steelwork, nitrogen (99.9999% pure) is supplied at a pressure of 2 - 5 kPa. There is a hydraulic seal on the feed pipeline.

The gas loop system is capable of purging the reactor masonry with nitrogen. Here the nitrogen is released through the activity suppression system.

In the gas loop system flow rates, impurity concentration, moisture content, temperature and pressure of the nitrogen-helium mixture are measured and radiation is monitored. All readings appear on the gas loop panel.

Valves and equipment are controlled from the gas loop panel.

2.11.5. The cooling pond cooling system

The cooling pond cooling system is designed to maintain temperature conditions of the water heated by residual heat release from spent fuel in the ponds under all conditions, including complete loss of power to in-house consumers of the unit. The system keeps water temperature in the ponds:

- under normal operating conditions not greater than 50°C, at maximum residual heat output of 1800 kW;

- with simultaneous unloading of 5% of fuel assemblies from emergency channels into the pond - not greater than 70°C; in this case residual heat output in the pond is not greater than 3000 kW.

- when heat dissipation ceases due to disruption of normal operating conditions or loss of power to the system (a rise in water temperature not greater than roughly 80oC in 20 hours from the start of termination of heat dissipation).

During this time measures should be taken to restore system operation.

Water quality in the spent fuel cooling ponds is maintained by a bypass purification system.

Measures are provided to eliminate the possibility of accidental evacuation of the cooling ponds.

Measures are also provided to vent the air space above the basins.

Figure 2.58 shows a schematic of the cooling pond cooling system.

The ponds are cooled over a closed loop. Water heated in basins by residual heat release travels from the upper parts of the pond to the heat exchangers where it gives up its heat to the service water. Required heat release is ensured by a single heat exchanger with a second in reserve. After passing through the heat exchangers, water is returned to the pond by one of two pumps with a capacity of roughly 160 m3/hr with a head of roughly 20 m water column (second pump in reserve).

The cooling water return and collecting pipes are designed such that when they fail the level in the ponds does not drop below a minimum allowable value.

There is an overflow in each of the ponds to prevent overfilling.

To preclude formation of an explosive hydrogen concentration,

Система охлаждения бассейнов выдержки

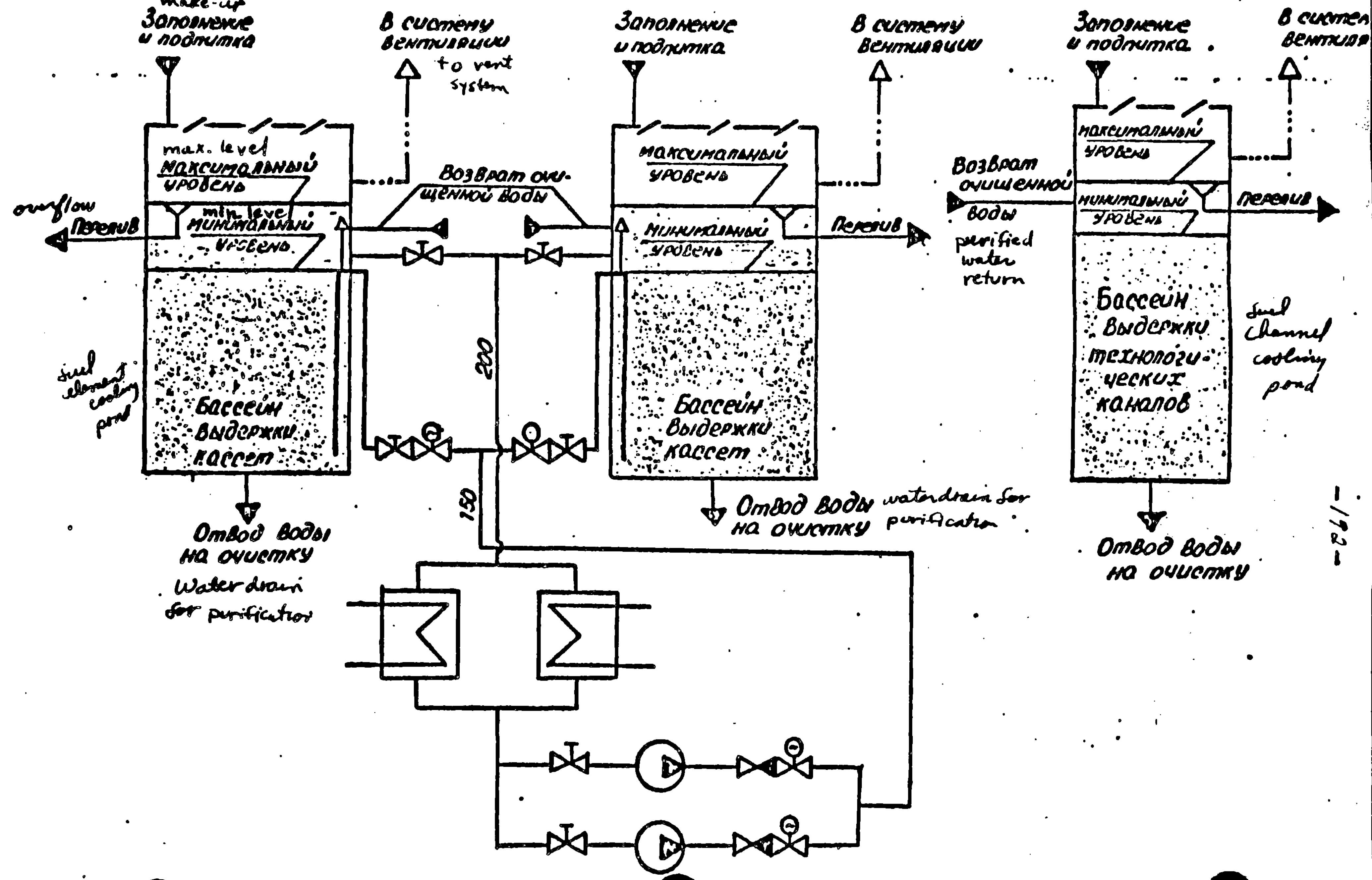


Fig. ?.2.58 Cooling Pond System

It is continuously vented in the space above the ponds (air collected from the central room is sucked in). When the pond vent system fails, the flow section of the openings which connect the cooling ponds to the central room is such that they can be regarded as a compartment with a volume of 40,000 m3 which is vented by an independent venting system.

Pond water is purified over a loop independent of the cooling system. The following are monitored:

- water level in the ponds;
- water temperature in the ponds;
- cooling water flow rates and so forth.

The system is controlled and its technological parameters monitored by operator personnel from the BShchU.

The system was comprehensively checked for agreement with design indicators during startup operations and unit operation.

2.11.6. Ejection-cooling unit

The ejection-cooling unit (Fig. 2.59) is designed to dissipate heat from the sealed volume compartment.

The ejection-cooling unit for each of the two sealed volumes consists of four groups of coolers mounted at the 5.0 mark in the GPsN tank compartment. A group consists of four coolers. The capacity of one cooler is 2500 m3/hr. Each group is filled with air independently. In terms of water the coolers are divided into two independent subsystems of eight coolers each and are connected to the different pump systems of the SOS.

Air at maximum temperature is collected from the upper zones the downcomer shafts by four pipes, sent to

СХЕМА УСТАНОВКИ ЭЖЕКЦИОННЫХ ОХЛАДИТЕЛЕЙ

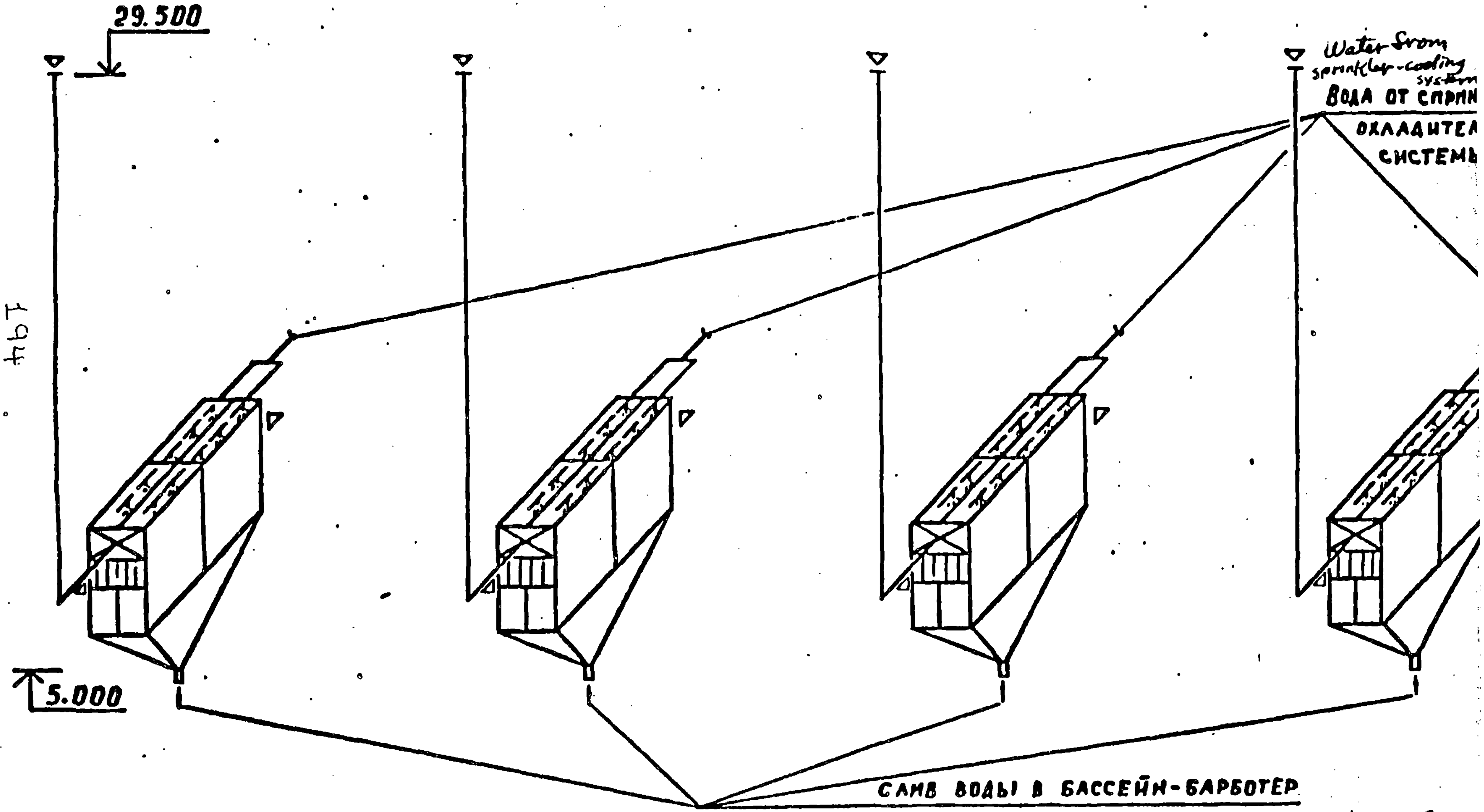


Fig ~~Рис~~. 2.59 Ejection Cooling System

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to each group of coolers where is cooled using water jets in summer to 35oC, in winter to 18oC, and is sent to GTsN tank compartment. The cooled air dissipates heat from plant equipment and accidental coolant leaks. To preclude entrainment of dispersed moisture with air there are separators at the cooler output. Ejection coolers not only cool air and remove excess moisture from it, but also remove aerosols, including radioactive iodine.

The ejection-cooling system is compact and does not include active elements which require servicing or control during operation.

2.11.7. Radiation monitoring system

The radiation monitoring system of a nuclear power plant is a component part (subsystem) of the automated control system of the power plant and is designed to collect, process and output data on the radiation situation in compartments of the power plant and in the external environment, the status of process media and loops, on exposure of personnel and individuals in the population according to current standards and legislation.

The entire radiation monitoring system is divided into two systems: the radiation profess monitoring (RTK) system and the radiation dosimetric monitoring system (SRDK).

The purpose of the radiation process monitoring is to optimize technological processes and also monitor the condition of protective barriers on radionuclide propagation paths.

The purpose of the radiation dosimetric monitoring system is to monitor radioecological factors engendered by nuclear power plant operation and ultimately to determine internal and external radiation doses received by personnel and individuals in the population.

External dosimetric monitoring is separated from the radiation dosimetric monitoring system; the former makes it possible to determine:

- activity and nuclide composition of radioactive substances in atmospheric air;

- monitoring the gamma radiation dose rate in the locality;

- monitor radioactive fallout;

- monitor ground water activity in test wells;

- determine content of radioactive substances in soil, plants, locally produced fodder, in foodstuffs and sc forth.

Figure 2.60 shows a structural diagram of the SRDK.

The following are used for radiation monitoring:

1) a complex of AKRD-06 gear which includes detection units and devices, data processing equipment, units for monitoring contamination of surfaces by radioactive substances, units and dosimeters for monitoring irradiation of power plant personnel;

2) individual portable, carried instruments;

3) laboratory equipment and instruments.

The structure of radiation monitoring consists of an information-measuring system with a large number of concentrated data sources and receivers set up in a radial-annular configuration; detection units with data collector and processor UNO-06R are linked radially to units for local indication of set values being exceeded.

The UNO-06R are linked to one another and the data exchange and monitoring device (UNO-01R) over annular communications lines.

The AKRB-06 also allows continuous monitoring of readings

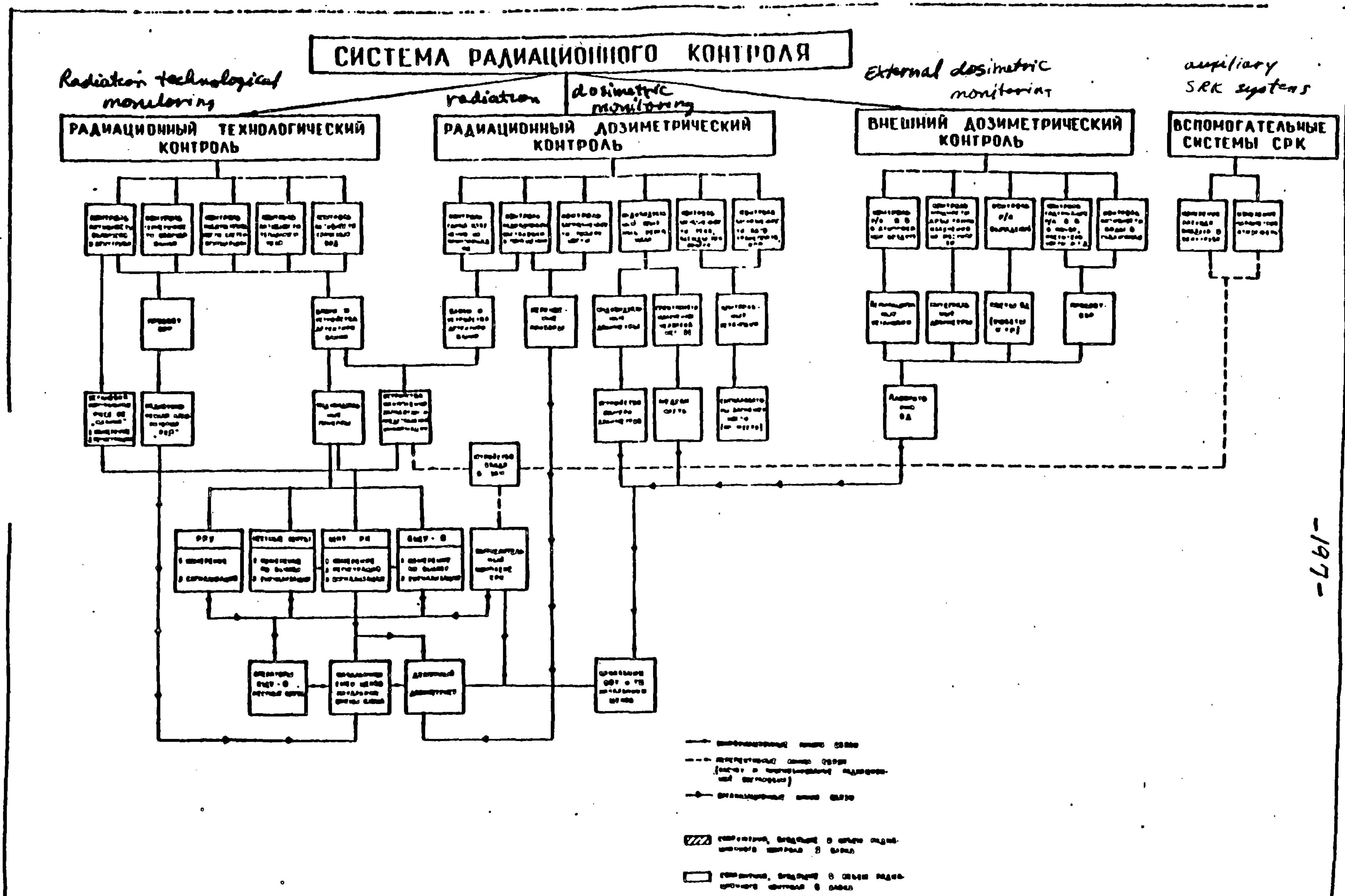


Fig 2.60 Radiation Monitoring System

of detection units and devices, transmission of information over all channels to the computer, output of signals regarding component part malfunction, and control of isolating valves mounted on sampling lines.

Data display devices (displays, consoles, signalling units) are located on the radiation monitoring panel.

The detection units which make up the AKRB-06 make it possible to measure the following:

- intensity of gamma radiation exposure dose in the range from 10-5 - 103 R/hr (BDMG-41, BDMG-41-01, UDMG-42, UDMG-41-02);

- volumetric activity of gamma emitting nuclides in liquid process media and loops in the range 5 x 10-11 - 10-3 curies/l (UDZhG-04R, UDZhG-05R, UDZhG-14R1);

- volumetric activity of iodine vapors in air in the range 10-11 - 10-6 curies/l BDAB-06);

- volumetric activity of aerosols with a dispersed phase containing beta emitting nuclides in the range 10-13 - 10-9 curies/l (BDAB-05);

- volumetric beta activity of inert gases in air and process media in the ranges 10-9 - 1.4 x 104 curies/l (YDGB-08) and 10-5 - 0.3 curies/l (UDGB-05-01);

- the activity of long lived beta aerosols in gas aerosol releases into the vent pipe in the range 3 x 10-14 - 3 x 10-10 curies/l;

- activity of short lived beta active aerosols in gas aerosol releases into the vent pipe in the range 1.5 x 10-12 - 1.5 x 10-8 curies/l;

- activity of beta active inert gases in gas aerosol releases into the vent pipe in the range 8 x 10-9 - 8 x 10-5 curies/l;

- activity of gamma active gas vapors in gas aerosol releases into the vent pipe in the range 3 x 10-13 - 3 x 10-10 curies/ℓ.

Gas aerosol releases into the vent pipe are monitored by radiometers RKS-03-01 and RKS-2-02. Air flow rate in the vent pipe is measured by a partial flowmeter at the base of a metal-polymer sensing element. Measurement data results are computer processed.

For each unit of the nuclear power plant the total number of protection units is 490, of which roughly 400 measurements are taken in process compartments where personnel may be located on a permanent or limited basis.

Surface contamination monitoring units warn personnel when established contamination threshold levels are exceeded:

- skin of the hands by beta active substances in the range 10 - 2000 beta particles/(min.cm2) (RZG-05-01, SZB-03, SZB-04);

- skin of the body or primary protective clothing by beta active substances in the range 50 - 2000 beta particles/(min.cm2) (RZB-04-04);

- transport when leaving the nuclear power plant in order to detect objects for detailed examination by other means for gamma radiation in the range 2.78 x 10-2 - 0.278 microroentgens/s (RZG-05);

- personnel when leaving the nuclear power plant for protection and subsequent detailed examination of the subject for gamma radiation in the range 1.4 x 10-2 - 0.14 microroentgens/s (RZG-04-01).

Devices which monitor personnel irradiation doses constantly monitor external irradiation. To do this the following are mostly used:

- individual dosimetric film badges to measure integral gamma radiation exposure doses in the range 0.05 - 2 R at energy 0.1 - 1.25 MeV (IFKU-1);

- thermoluminescent dosimeter sets to measure x-ray and gamma radiation exposure dose in the energy range 0.06 - 1.25 MeV with measurement limits 1.0 - 1000 R, 0.1 - 1000 R (KDT-02);

- Dosimeters-signallers of the intensity of gamma radiation exposure dose in the range 0.1 - 9.9 R/hr. There are also a number of dosimeter modifications with similar characteristics.

The instrument for recording internal human irradiation allows measurement of the content of radioactive nuclides cesium-137 and cobalt-60 uniformly distributed in the human organism and iodine-131 in the thyroid gland (MSG-01). Also semiconductor DGDK type detectors are used at the nuclear power plant with corresponding analysis and processing equipment which makes it possible to identify a number of other radionuclides in the human organism.

The pool of portable and carried instruments constitutes a wide range of dosimeters and radiometers, for example:

- to measure gamma and x-radiation dose rates in the energy range 15 keV - 25 MeV with measurement ranges 0.1 microroentgens/s - 11 roentgens/s (DRG-DKS);

- to measure equivalent neutron dose rates from 0.05 - 5000 micro ber/s (KDK-2);

- for express measurements of individual sample activities (beta radiometer RKB4-1) with range 2 x 10-12 - 10-7 curies/1;

- for measuring volumetric activity of nuclides in liquids, gases, in air for alpha, beta and gamma radiation in different energy ranges (RZhS-05, RGA-01, MKS-01, and others).

External dosimetric monitoring is done in the region of the nuclear power plant to a radius of about 35 km. It is done by the external dosimetry service of the nuclear power plant and is designed to obtain information necessary to estimate internal and external irradiation doses to individuals in the population.

Monitoring equipment is located at 38 stations which include integral gamma dosimeters, cells for collecting atmospheric precipitation and also seven suction units.

Samples are analyzed using semiconductor detectors, spectrometers and analyzers with microcomputers.

The radiation situation in the vicinity of the nuclear power plant is predicted using microcomputers from data on releases into the atmosphere through the vent pipe of the nuclear power plant and automatic measurement of weather parameters.

2.11.8. AES management centers

AES management is performed on two levels:

- plant level
- unit level

(see Figure 2.61 the fundamental management layout of the plant).

Management of all of the devices which ensure the safety of the AES is performed at the unit level.

Plant management level

Real-time management of an AES at the plant level is performed from a central control panel (TsShchU).

At the plant level the operational personnel are responsible for:

- managing the electric equipment of the main circuit for electrical connections (circuit breakers for the 750 kV lines, the unit transformers, the autotransformers and others, circuit breakers for the 20-kV generators, circuit breakers for the 330-kV autotransformers, and circuit breakers for the 6-kV and 300 kV reserver transformers s.n.);
- distribution of the active and reactive power; and

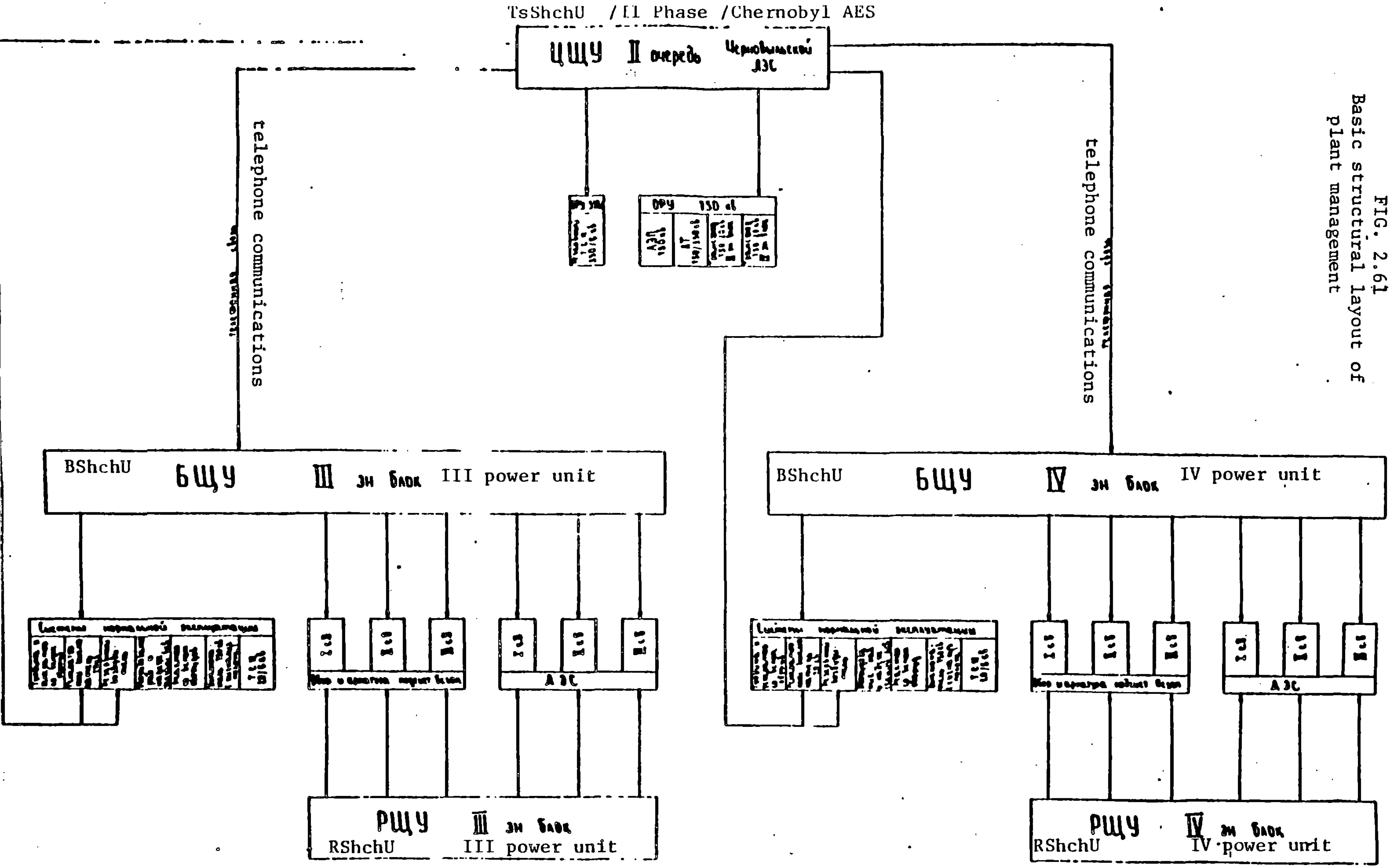


FIG. 2.61 Basic structural layout of plant management

- coordination of the work of the operational personnel stationed at the unit control panel and at individual structures of the commercial site.

Keys for controlling the above cited circuit breakers are provided at the central control panel, as well as sound and light signaling of an emergency and breakdown and light signaling of the status of the commutation devices (the circuit breaker is open or closed) on a graphic panel.

The devices for relay protection, accident preventing automatic devices, and telemechanics are located in the buildings of the relay panels of the corresponding distribution devices (ORU): ORU-750 kV and ORU 330 kV.

Devices built on the basis of integral microcircuits are used for relay protection of the 750 kV lines. These devices can troubleshoot individual channels and make it possible to perform tests.

Series produced electromechanical relays are also used in the devices for controlling the relay protection and the accident preventing automatic devices.

Unit management level

Management of technological objects and structures included in that particular energy unit is performed at the unit level:

- the reactor with the support installations (GTsN; PN; APN and others);
- turbine generators with auxiliary equipment;
- operational and reserve power sources s.n. and others; and
- individual structures of the commercial site: diesel power plants, commercial water supply pumps, and others.

Management of the cited equipment is performed from the unit control panel (BShchU), which includes the panel and a control panel. The operational circuit of the BShchU is divided into management zones:

- reactor management;
- steam generator management; and
- turbine, generator and s.n. power source management.

The work spaces of the operators- the control panels are positioned in the operational circuit of the SChchU:

- for the senior reactor management engineer;
- for the senior unit management engineer; and
- for the senior turbine generator management engineer.

The control panels include:

- management equipment
- system testing instruments;
- calling devices and indicator complexes from the "SKALA" centralized control system; and
- communication equipment.

The panel of the operational circuit of the SCHchU includes:

- the mimic panel of the reactor;
- the mimic panel of the SUZ;
- the graphic panel of the thermotechnical and electrical parts of the unit;
- individual instruments of the test system; and
- signaling systems.

The basic volume of the parameters are controlled by the "SKALA" central control system (STsK).

The most important parameters required for correct performance of technological processes are also controlled by individual instruments.

These include the reactor power, the pressure and level in the BS, the flow rate of steam after the BS, the flow rate of the replenishment water into the BS, and measurements of the SFKRE and SUZ and others.

Power supply to the BShchU and the "SKALA" STsK is implemented from a reliable power system in such a way that if the s.n busbars are shut off, the operator does not lose information about the status of the technological parameters.

Technical systems for controlling technological protection and testing systems are positioned in the operational circuit of the BShchU.

Overall operational management of the unit is performed from a dispatcher post equipped with a telephone and a loudspeaker.

The zone of the operational circuit of the BSHchU is also provided with special panels - safety panels (PB) for each of the three safety subsystems, to which management and testing of the reserve s.n. power sources (diesel generators) and systems for emergency cooling of the reactor and localization of the accident) are shifted.

A reserve control panel (RShchU) is provided for shutting down the reactor and maintaining it in a subcritical state when it is impossible to perform this operation from the BShchU.

The operational circuit of the RShchU includes a control panel, operational circuit panels, and safety panels.

An AZ-5 button, an SUZ coupling cut off key, a signaling panel and others are installed on the control panel. Recorders of the neutron power, the pressure in the BS, and so on are installed on the panels of the operational circuit. The PB RShchU panels are analogous to the PB BShchU panels.

For a number of technological systems which are not associated with the basic technological process, local control panels are provided for: the gas loop, the special water purification; radiation control systems; for installations for sorptive purification of ejector gases, and turbines.

Local panels are also provided for a number of units of the basic technology (GTsN, PEN, APEN, and others), which are delivered in a complex with the equipment.

2.12 Operating regimen of the reactor and block

2.12.1 Normal operating modes

The operating regimens of the reactor and block may be characterized as normal operating regimens and transitional regimens during equipment downtime. Normal operating modes are startup and shutdown of the block, operation of the block at power-producing capacity, reactor cooling regimens during equipment maintenance (maintenance regimens).

Startup and shutdown of the reactor

The startup of the power block with an RBMK reactor is conducted with operating reactor coolant pumps , with "sliding" pressure and with a free water level in the separators. The required anticavitation supply during coolant pump suction is provided by reducing the output by the restricting control valves that are mounted on the head of the pumps. The continuous control of cooling water consumption is carried out in this mode over all of the fuel channels of the core and by this token assures the safety of the reactor. Heating of the block is conducted under "sliding" pressure in the separators, i.e. the pressure is not constant, but rises as the temperature increases.

Injection of the circulation loop during the startup and heating of the block is carried out by emergency injection pumps.

The capacity of the reactor during startup and heating is maintained on the average at a level of 2-3% of nominal value. Additionally, thermal capacity of individual fuel channels due to unequal distribution of energy release along the core may comprise up to 6% of nominal.

It is permissible to conduct increases in reactor capacity and heat the circuit using one, two, and three operating reactor coolant pumps on each side of the reactor, having a capacity of 6,000-7,000 cubic meters per hour each. With such a capacity, the reactor coolant pump has the capacity of controlling water consumption through each of the fuel channels, as well as assuring an adequate supply prior to pump cavitation. With a 2-3% reactor capacity from nominal capacity, heating of the circulation loop is carried out until a temperature of about 200 degrees Celcius. The heating rate of the circuit is about 10 degrees C per hour and is limited by temperature stresses in metal structures of the reactor.

At a pressure of 2-4 kg per square cm the heating of the deaerators is initiated.

Vacuum is initiated in the condensers of the startup turbine at a pressure of about 15 kgs per square cm in the separators. After vacuum is reached, the turbine is jolted and accelerates. Synchronization of the turbogenerator and connection with the network is usually accomplished when pressure in the condensers is about 50 kgs per square cm. Continued increase in parameters until nominal values is achieved parallel with increase in the electric load.

Figure 2.62 gives an example of the changes of basic parameters of a reactor installation from the moment the reactor is brought to a minimally controlled capacity level, and until synchronization and connection of the turbogenerator into the network.

Regular shutdown and cool down of a power block with an RBMK reactor is conducted when reactor coolant pumps are in operation. Prior to cool down the capacity of the reactor is reduced to the level of residual heat release, while the

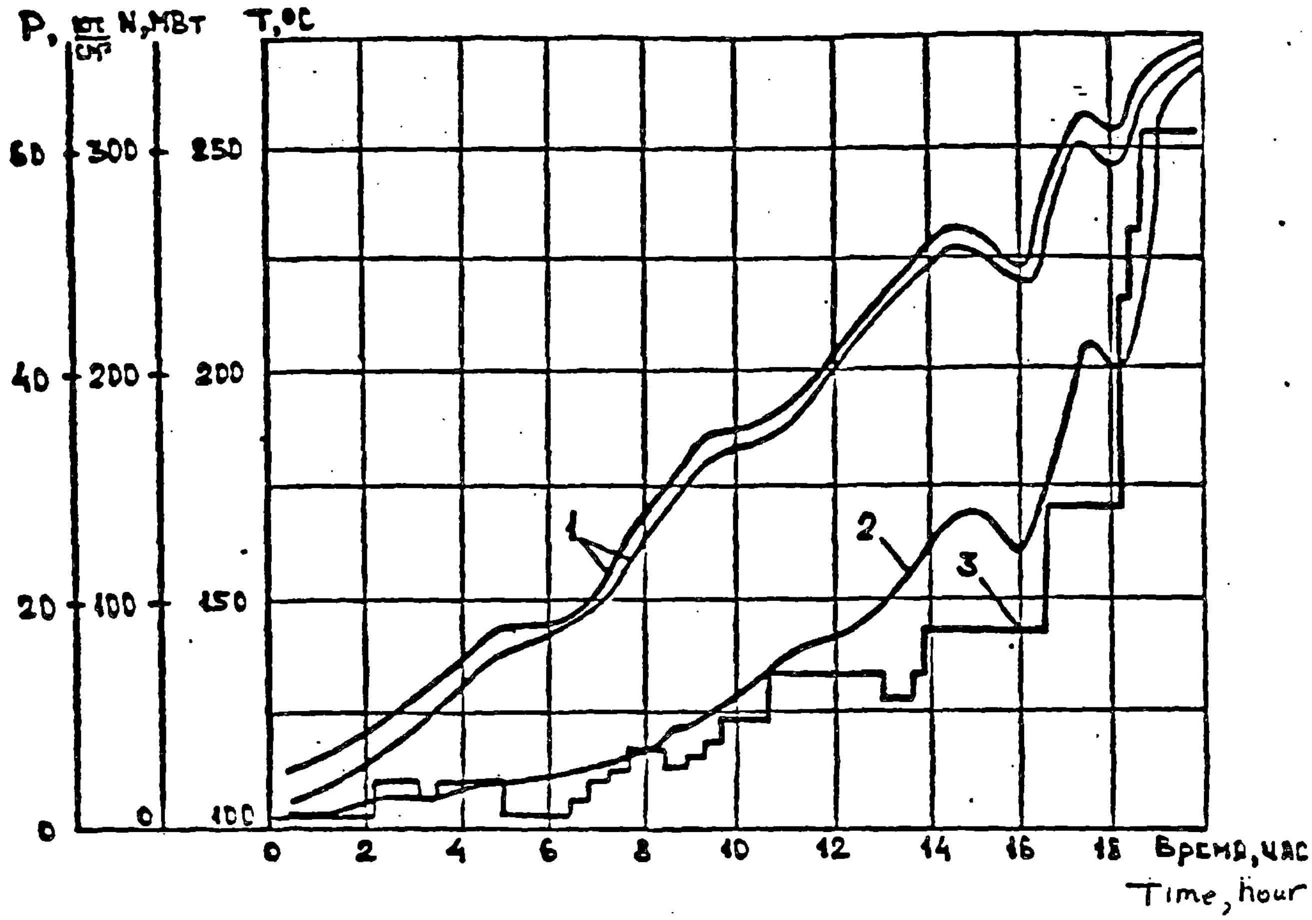


Рис. 2.6.2. Изменение параметров реактора при пуске:

Figure 2.6.2. Changes in reactor parameters during startup:

1. Water temperature (T) in the circulation loops of the reactor;

2. Pressure (P) in the separators;

3. Thermal capacity of the reactor (N).

turbogenerators of the block are switched off the network and are stopped. When reactor capacity is reduced to 20%, capacity of the operating reactor coolant pumps must be reduced to 6,000-7,000 cubic meters per hour. Cool down of the circulation loop to a temperature of 120-130 degrees C is carried out by a gradual decrease of pressure in the loop via a controlled release of steam from the separators into the turbine condensers or the industrial condenser. For a more thorough cool down a special cooldown system is utilized that consists of pumps and heat exchangers.

As during heating, the factor that limits the cooldown rate are the temperature stresses in the reactor's metal structures. Since during cooldown the rate of temperature reduction in the circuit is determined primarily by the extent of discharge of steam from the separators, then maintaining the rate of cooldown at a given level is not a complex problem in this mode.

Operation of the block at power-generating capacity

When the block is operating at power-generating capacity, reactor safety is achieved by maintaining the guide parameters within allowable limits.

Until a capacity of 500 MWt(t) the circulation of heat carrier through the reactor is achieved by the reactor coolant pumps that operate at a capacity of 6,000-7,000 cubic meters per hour. At a capacity of 500 MWt(t), the capacity of the reactor coolant pumps is increased to 8,000 cubic meters per hour as a result of opening of the DRK (expansion unknown). At capacities above 500 MWt(t) and up to nominal level, the block operates with a steady capacity of the reactor coolant pumps. When capacity is increased above 60% from nominal no less than three reactor coolant pumps must be in operation on each side of the reactor. Hydraulic profiling of the core of anRBMK reactor is carried out in a way so that upon getting to nominal capacity, the DRK would be fully open, with the total consumption through the reactor being equal to 48,000 cubic meters per hour.

Maintenance modes

The principal requirement for carrying out inspections or maintenance of any of the reactor's equipment is the assurance of reliable cooling of the core during this period. At the same time, the construction of the reactor and organization of maintenance regimens much be such that all of the equipment of the circulation loop be assured of maintenance.

From the point of view of conducting maintenance operations, the entire circulation loop is broken down into four areas: pressure section from the pressure valves of the reactor coolant pumps to the channel pressure damper valves, the fuel channel tracts from the ZRK to the separators, separators and downcomer pipelines to the suction gate valves of the reactor coolant pumps, and the section between the suction and pressure gate valves, which includes circulation pumps and appropriate equipment.

Maintenance of equipment and pipes that are situated in the section between the suction and pressure reactor coolant pump gate valves does not present any difficulty, and in principle, could be carried out when the reactor is in operation.

For this purpose the pressure and suction slide valves on the pipelines of this particular GTsN must be closed, and after draining the heat carrier, the GTsN itself and the sectors of the pressure and suction pipelines adjacent to it up to the slide valves are accessible for maintenance. Circulation of heat carrier through the reactor in this instance is carried out by other reactor coolant pumps of the given circulation loop.

In order to carry out maintenance on the structural elements of fuel channels, the heat-carrying assembly is removed from the channel that is being serviced, the ZRK mounted at the channel inlet is closed, the level of water in the separators is lowered below the notch marking the water and steam communications of this channel into the body of the separator. Cooling of the remaining channels of the core is carried out either under forced or under natural circulation of the heat carrier.

In order to service separators, downcomers and suction gate valves, or the suction gate valves of the reactor coolant pump, the pressure gate valves of the reactor coolant pump are closed, and the level in fuel channels is dropped. In order to conduct a danger-free cooling of the core during such modes, the connection of a special servicing vat to a pressure collector of the reactor coolant pump is foreseen; this vat is used to feed the channels and the steam that forms in them is evacuated into the separators. In order to be able to examine and service the separators, a system for suctioning steam from the separators into a technological condenser has been foreseen.

When servicing pressure equipment of the section, this section, due to the fact that the ZRK is closed, is cut off from the core, while the removal of residual heat is done by water that is fed into the channels by the separators. Such a cooldown regimen of the fuel channels (bubbling cooling regimen) was studied during the design of the reactor, using special test benches. Experimentally it was shown that when damper-control valves are shut, a reliable cooling of fuel channels is assured with bubbling cooling when the following requirements are fulfilled:

-- the level of water in the circulation loop is above the notch mark of the steam-and-water communications (PVK) in the separator;

-- atmospheric pressure in the separator;

-- capacity of residual heat release in the fuel assembly is no higher than 25 kWt;

-- water temperature in the separator is not below 80-90 degrees Celsius, in order to avoid hydraulic shocks in the PVK pipes.

The most complicated maintenance procedure is servicing of the channel flowmeters and the ZRK. In order to carry out this operation, used are

2.12.2. Transition modes involved with equipment failures

In view of the high unit power of RBMK boiling water - graphite reactors and their relatively high importance in power systems, the system for control and protection (SUZ) of these reactors in malfunctions of individual types of equipment includes a rapid, controlled decrease in power at a predetermined rate to safe levels. Emergency protection of three types (AZ1, AZ2, and AZ5) operates on signals of malfunctions of the operating equipment.

The following algorithm for operation of emergency protection measures has been realized in the SUZ of active RBMK-1000 reactors:

- AZ1 functions in stopping one of six main circulation pumps, a decrease in the feedwater flow rate on a decrease in the level in the separators. At the AZ1 signal, the reactor power decreases to a 60% level;

- AZ2 functions in an emergency discharge of the load or stopping of one of two operating turbogenerators. The reactor power drops to a 50% level at the AZ2 signal.

In other emergency situaitons involved with equipment failures, emergency protection AZ5 operates; in functioning of AZ 5, an uncontrolled reduction of power occurs up to full shutdown of the reactor.

A mathematical model of the installation, including equations of kinetics, hydrodynamics and heat transfer and a description of algorithms of operation of the equipment and systems for automatic regulation of the

AES parameters, has been developed for studying the malfunction conditions of power units with

RBMK reactors. The comparison of results of calculations with data on individual dynamic modes which occur at working AES which was performed later indicated that the mathematical model which had been developed satisfactorily describes the dynamics of the power unit. Transition modes involved mainly with the transfer to natural circulation of the heat-transfer medium were studied on special test models.

The experience of operation of active power units indicated that the measures and means envisioned fully ensure the safety of the RMBK reactor under all conditions involved with equipment failures.

A great deal of research was done for substantiating the safety of reactor operation under conditions of a decrease in power in functioning of emergency protection AZ 5, since the occurrence of this mode is accompanied by deep changes in operating parameters: in particular, a decrease in the water level in the separators.

The behavior of basic reactor parameters in a transition mode in functioning of protection AZ5 is shown in Fig. 2.63.

The loss of power for system auxiliaries is one of the most severe emergency situations in a power unit. In loss of power for system auxiliaries, circulation of the heat-transfer medium through the active zone at the beginning of the malfunction is provided by GTsN which are running down, after there is only natural circulation. The transition process in a loss of power for system auxiliaries of a unit is shown in Fig. 2.64.

One can see in the figure that in the initial stage of the process, the rate of the drop in the water flow rate is somewhat higher than the rate of the decrease in thermal power of the reactor, which leads to a brief increase in the steam content and a decrease in reserves before a crisis of convective heat transfer. More detailed research indicated that the decrease in reserves before the crisis even in the channels under the greatest thermal stress in this mode is slight and is safe for the reactor, since at the beginning of a malfunction, the reactor is cooled reliably due to circulation created by the GTsN which are running down.

The GTsN running down have a noticeable effect on the level of circulation of the heat-transfer medium through the reactor only in the first 30-35 s of the transition process. After this cooling of the active zone is effected due to natural circulation. The stability and intensity of natural circulation depend to a great extent on a whole series of factors, such as the design of the circulation duct, the behavior of the pressure in the duct, variation in the feedwater flow rate and temperature, etc.

Experimental research on conditions of natural circulation were performed both on thermotechnical testing units - models of the reactor circulation duct - and directly on working reactors on the Leningrad and Kursk AES. The reliability of cooling of the active zone in natural circulation in both stationary and dynamic modes with preservation of a constant pressure in the circulation duct was established on the testing units and confirmed on the reactors. In stationary modes, tests were conducted on working

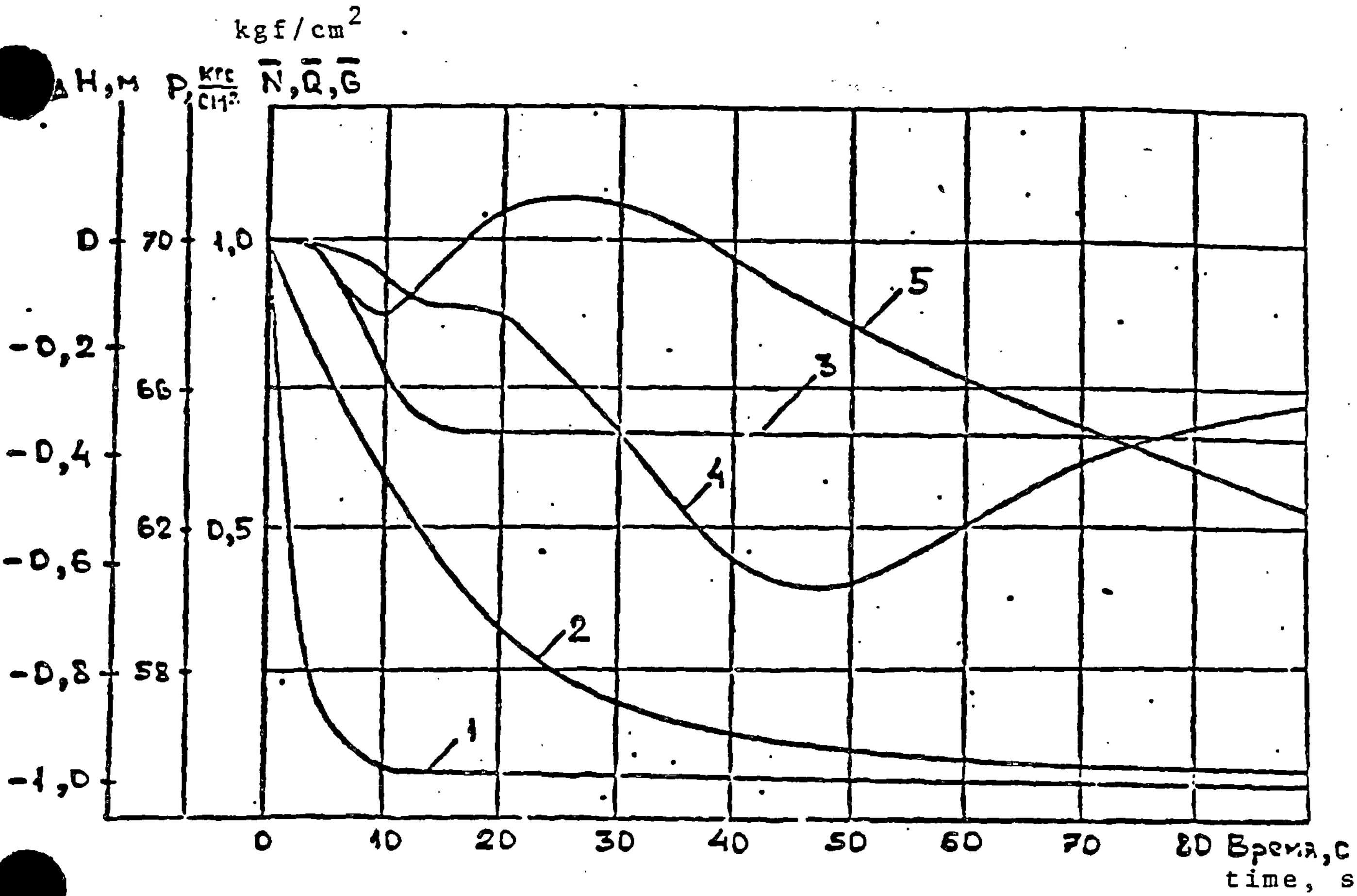


Рис. 2.6 3. Поведение параметров реактора в режиме срабатывания АЗ5:

I - нейтронная мощность ($\bar{N}$);

2 - тепловая мощность ($\bar{Q}$);

3 - расход циркуляционной воды ($\bar{G}$);

4 - изменение массового уровня в сепараторах (ΔH);

5 - давление в сепараторах (P).

Fig. 2.6.3 Behavior of reactor parameters under condition of AZ5 functioning:

1. neutron power ($\bar{N}$);
2. thermal power ($\bar{Q}$);
3. circulating water flow rate ($\bar{G}$);
4. variation of mass level in separators (ΔH)
5. pressur in separators (P).

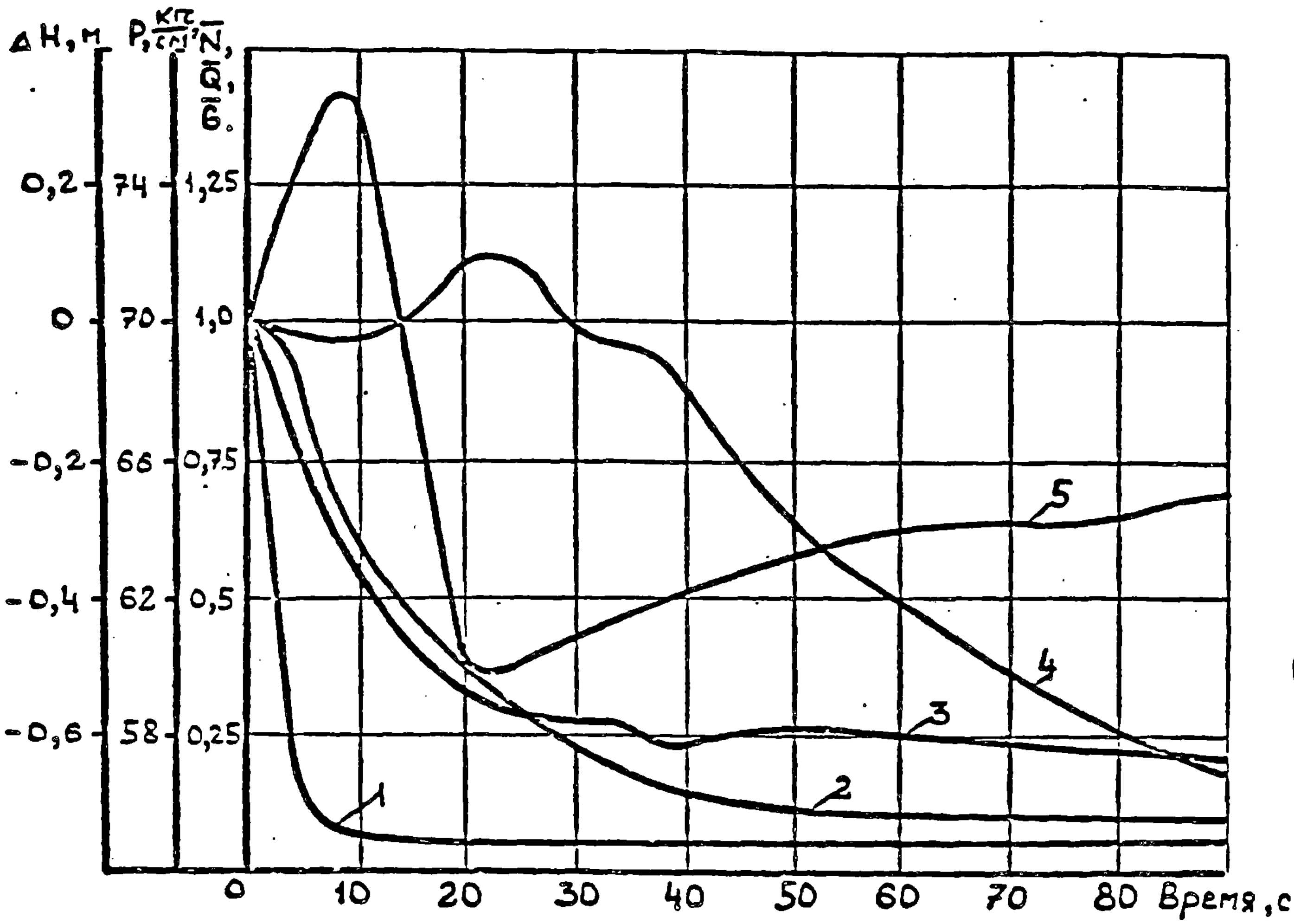


Рис. 2.6.4. Поведение параметров реактора в режиме обесточивания собственных нужд блока.

Обозначения приведены на рис. 2.63

Fig. 2.6.4 Evaluation of the reactor parameters during onsite power loss. Legend shown in Fig. 2.63

reactors at powers of 5 and 10% of the nominal power; disengagement of the GTsN was performed at powers of 25 and 50% of the nominal in dynamic modes. With a decrease in pressure, which can be caused, for example, by opening and subsequent poor closing of safety valves, boiling up of the heat-transfer medium in the drop, expansion of the level in the separators and, as a result, loss of the steam-water mixture form the duct. It was established on a testing unit that with a decrease in pressure to a certain level, partial loss of the steam-water mixture and water from the duct does not result in a decrease in the level heap or interruption of circulation of the heat-transfer medium. Overheating of the fuel elements of an experimental channel was observed only with a decrease in pressure in the separators below 35 kgf/cm .

For ensuring reactor safety in a loss of power for system auxiliaries of the power unit and a deep pressure drop, the system for emergency cooling of the reactor is engaged and feeds water to the fuel channels.

The safety of conditions of natural circulation on power units with RBMK reactors has been confirmed by accidents which have occurred under real conditions at working AES. For example, a full loss of power for system auxiliaries occurred at unit I of the Kursk AES in January, 1980. Readings of thermocouples of thermometric fuel cartridges and a flowmeter installed at the inlet to one of the fuel channels of the reactor were recorded during the transition process. No temperature increase of the fuel element shells was recorded throughout the transition processd, and the recorded flow rate through the channel in natural circulation was at least

20% of the flow rate which occurred at nominal power. The standard system for monitoring the airtightness of the fuel element shells recorded no increase in the activity of the heat-transfer medium when the reactor subsequently went on power. Experimental data on conditions of natural circulation were summarized and compared to results of calculations by the calculation programs which have been developed. In view of the good agreement of the results, predictive calculations were preformed and indicated the possibility of stable and safe operation of RBMK power units in a natural circulation mode at power levels up to 35-40%.

Protection of the reactor in relation to a decrease in the feedwater flow rate and the level in the separators ensures safe operation of these power units at all power levels. Along with introduction of protection systems, disengagement of the GTsN in performed in functioning of the AZ5 emergency protection according to the decrease in the feedwater flow rate with a time lag. This is done for reducing the magnitude of the level drop in the separator and preventing cavitation failure of the GRsN, i.e., for providing optimum conditions for development of stable natural circulation. The safety of disengagement of the GTsN and cooling of the reactor in a mode of natural circulation, as indicated above, has been confirmed by numerous experiments and by the experience of AES operation.

Figure 2.65 shows the calculated transition process in full instantaneous stoppage of the feedwater flow.

Reactor safety in malfunctions in the feedwater supply system has also been confirmed by the experience of operation of active RBMK power units.

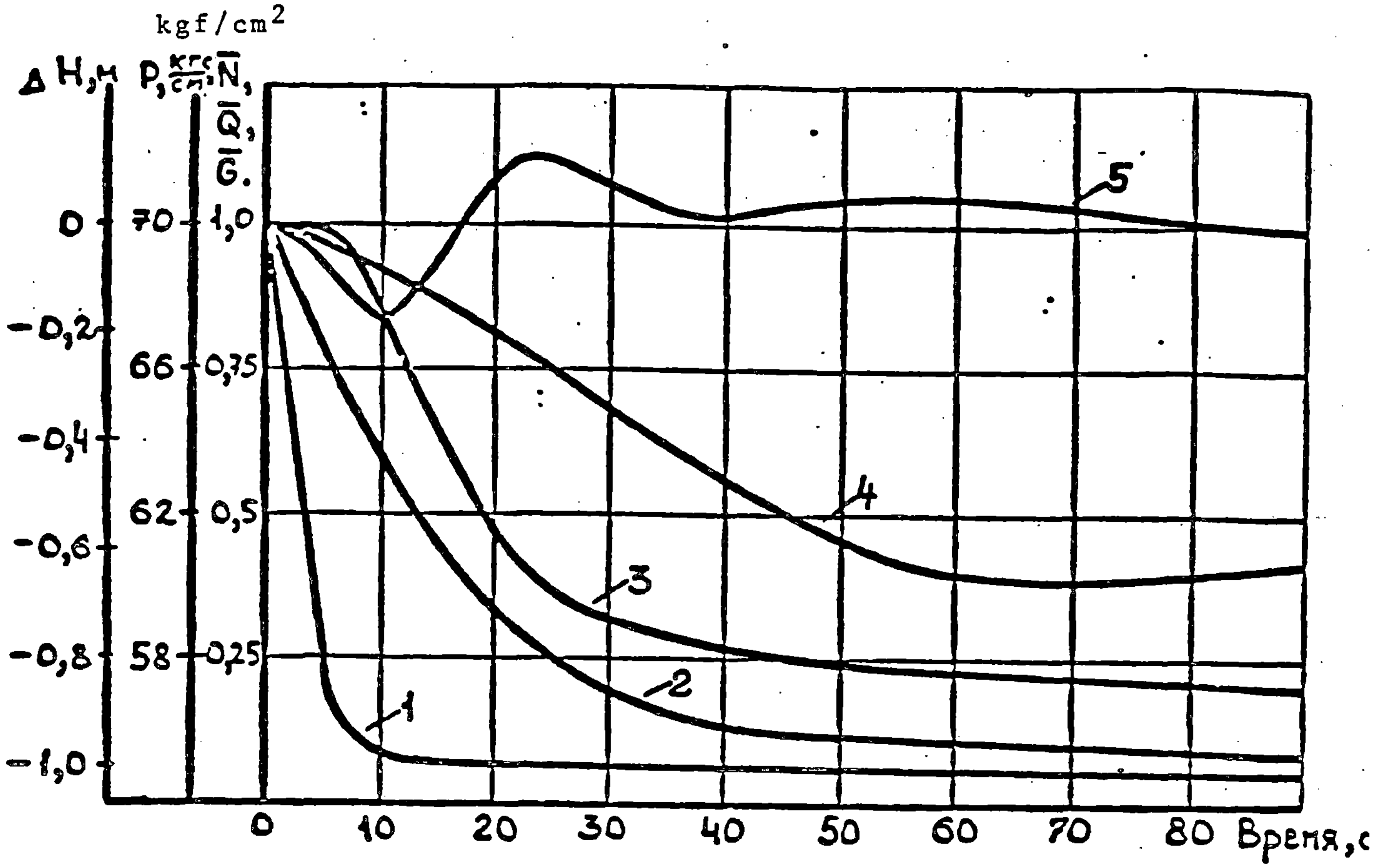


Рис. 2.6.5. Переходный процесс при полном ~~мгновенном~~ прекращении расхода питательной воды.

Обозначения приведены на рис.62

Fig. 2.6.5 Transition process in full instantaneous interruption of the flow of feedwater. The legend is shown in Fig. 2.6.2

List of Abbreviations

AZ	- Protection
AZM	- AZ for working power range
AZMM	- AZ for low power range
ASCP	- AZ for speed in startup range
AZSR	- AZ for speed in working range
AR	- Automated regulator
ARM	- AR for low-power range
AES	- Atomic power plant
BA-86	- Servodrive control automation module
BVRK	- Servodrive control contact relay module
BKS	- AR servodrive control power module
BP	- Power supply module
BP.119	- LAR-LAZ sensor power supply module
BP.38	- Ionization chamber power supply module
BP.39	- Ionization chamber power supply module
BP.30M	- Fission chamber power supply module
BPP	- Program module
BŚ	- Drum separator
BSP	- Synchronous movement module for LAR-LAZ
BT	- LAR-LAZ trigger module
BT-37	- AR trigger module
BUSP	- LAR servodrive control module
BUT	- Thyristor rectifier control module
BShchU-N	- Control panel module, inoperative portion

BShchU-O	- Control panel module, operative portion
VK	- Upper limit switch
VU	- BUSP input circuits

GUI	- BUSP control impulse oscillator
GTsN	- Main circulation pump
DP.1	- Rod position sensor
DRK	- Throttling-actuator valve
ZRTA	- Reactimeter
IK	- Ionization chamber
IM	- Actuator
ISS.ZM	- Count rate meter
K	- Coupling power supply switch
KV.2	- Output stage
KV.3M	- Output stage
KD	- Fission chamber
KtV.17	- Triaxial chamber for LAR-LAZ
KMPTs	- Controlled circulation loop
KNK-56	- Startup range IK
KNK-53M	- Working range IK
KNT-31	- Fission chamber
KOM	- Coupling disconnect key
KT	- Current corrector for AR-AZ
KSVP	- Recording potentiometer
KUT	- Thyristor control key
LAZ	- Local AZ
LAR	- Local AR
NVK	- Lower waterlines
NK	- Lower limit switch

OV	- Excitation winding
PIK	- Startup ionization chamber
PK	- Overcompensation
PK-AZ	- PK in AZ modes

PO	- Operator's panel
PPB	- Pressure-tight box
PS	- Warning signal
RIK	- Working ionization chamber
RPU	- Backup control panel
RR	- Manual control rod
S	- Synchronizer
SIUS	- Individual rod control scheme
SSS AR	- AR rod synchronization system
SUZ	- Control/safety system
STsK SKALA	- Central monitoring system
SChS	- Counter
TA	- Process automation
TVS	- Fuel assembly
TEZ	- Typical replacement element
UZM	- Power level protection amplifier
UZC	- Speed protection amplifier
USM.12	- Summing amplifier
USO	- LAR-LAZ deviation signal amplifier
USO.10	- USO for AR
USP	- Truncated absorber rod
TsZ	- Central hall
ShchO	- Operator's console
ShchEP	- Electronic instrumentation console

EMT	- Electromagnetic brake coupling
Nnom	- Rated power level
ρ	- Reactor reactivity
β	- Percentage of delayed neutrons

Appendix 3

ELIMINATING THE CONSEQUENCES OF THE ACCIDENT AND SHUT-DOWN

3. Eliminating the consequences of the accident and shut-down.

3.1 The progress and prospects for shutting-down the first, second, and third units and their reentry into operation.

Contamination of the surfaces of the equipment and spaces of the AES basically took place through the ventilation system, which continued to operate in the fourth unit for some time after the accident, and due to dispersion of the radioactive dust from the territory of the plant. The individual horizontal sectors of surfaces of the machine hall had the highest level (up to 10^{6} β - particles/cm^{2}/min), since it was contaminated for a long period of time through the collapsed roof.

The rate of the dose of γ- radiation in the contaminated spaces of the first and second units on 20 May 1986 was 10-100 mr/hr and that in the machine hall - 20-600 mr/hr.

The composition of the solutions for decontamination was selected with consideration of the material being washed off (plasticized substances, steel, concrete, different coatings) and the nature and level of surface contamination.

Spray decontamination was widely used during the wash down with the use of washing machines and fire hydrants and some of the spaces were washed manually by swabbing with rags impregnated with decontaminating solutions. A steam ejection method and a method for dry decontamination using polymer coatings were also used.

Testing of the processes of decontamination was performed through direct measurement of the gamma background radiation from the washed surfaces and a "smear" technique. As a result of the decontamination, the levels of contamination of the surfaces of the spaces and equipment were reduced to those standards set in Radiation Safety Standards (NRB)-76 and OSP 72/80:

for the serviced spaces - 2000β- particles/(cm^2 /min);

for the semiserviced spaces - 8000β - particles/cm^2 /min).

After decontamination the levels of the γ-radiation for spaces in the first and second units was 2-10mr/hr.

3.2 Progress and prospects for decontaminating the AES site

During the accident radioactive material was ejected into the territory of the plant, and also fell on the roof of the machine hall, the roof of the third unit and on the metallic supports of the pipe.

The territory of the plant, the walls and roofs of the building also had substantial contamination due to settling of the radioactive aerosols and radioactive dust. However, the total gamma-background radiation in the territory created by radiation from the destroyed fourth unit substantially exceeded the level of radiation from the contaminated territories and buildings. It should be noted that the level of contamination of the territory was uneven.

The shoulders of the roads were treated with rapidly polymerizing solutions in order to reinforce the upper layers of the soil and to eliminate dust formation. This was done to reduce the scatter of radioactive contamination in the form of dust in the territory and on the roof of the building of the machine hall.

In order to deactivate the territory, the AES was divided into zones based on a condition of complex operations.

The sequence for performing operations in each zone is determined proceeding from

- the necessity for personnel to work on objects inside the zones;

-the principle "from dirty to clean" and with consideration of the "wind rose;" and

-the necessity of subsequent support of operations to restart the units.

The decontamination in each zone was performed in the following order:

-removal of refuse and contaminated equipment from the territory;

-decontamination of the roads and external surfaces of buildings;

-removal of a 5-10 cm thick layer of soil and its transport to the containers in the burial facility (the solid waste storage facility of the fifth unit);

-when necessary, laying concrete plates or filling in with fresh soil;

-coating of plates and territory not covered by concrete with a film forming material; and

-limiting access to the treated territory.

The daily rate of treated sites was up to 15,000-35,000 m^2. As a result of the completed measures, it was possible to reduce the total background gamma radiation in the region of the first unit to 20-30 mr/hr, where this residual background radiation is basically caused by external sources, which points to the high effectiveness of decontamination of the territory and the buildings. However, a substantial improvement in the radiation situation in the entire territory of the AES, especially in the regions of the third and fourth units, is possible only after closing the destroyed reactor.

3.3 Progress and prospects for decontaminating the 30-kilometer zone and its return to agricultural pursuits.

The formation of radioactive traces after a one-time ejection is completed after approximately a year. In this period there is a substantive redistribution of radionuclides in regional elements in accordance with the features of the relief. The most intensive redistribution of radioactivity (secondary transfer) occurs in the first 3-4 months after ejection, especially during the occurrence of active biological and atmospheric processes (growth, development, and disappearance of plants, rains and winds). The poorly attached part of the radioactive materials settled on the surface of the soil and vegetation is greatly subjected to redistribution. In coniferous forests such redistribution is completed only after 3-4 years (after full renewal of the needles).

In light of the cited reasons, the radioactive situation within the 30-kilometer zone will continue to greatly change for 1-2 years, especially in regions with a high gradient of contamination levels.

Therefore, the measures being conducted to decontaminate the populated points will generally only lead to a temporary improvement of the radiation situation.

All of this makes it possible to conclude that reevacuation of the population may be conducted only after stabilizing the radiation situation in the entire territory of the contaminated zone (after cessation of ejections from the reactor, decontamination of the commercial sites, and consolidation of the radioactivity in the territory with a high level of contamination). The fastest stabilization of the situation will be achieved in regions of the zone with the low differential in the levels of contamination (for instance, in sectors of the northern and southern projections of the track.

Reliable information about the concentrations of long-lived radionuclides (strontium 90 and cesium 137) in the soils and vegetation grown in them must be available for solving the issues of the possibility of soil sampling is presently being completed in all the fields of the collective and state farms of the region. Cartograms of contamination of the agricultural lands by the cited radionuclides will be compiled upon completion of sample analysis.

Other radionuclides which are part of the contamination (zirconium-90, niobium-95, ruthenium-103 and 106, cerium-141 and 144, cesium-134, barium-140, and strontium-89), make up more than 90% of the total activity and cannot in the future be limiting factors due to their short half-life or low accessability for absorption from the soil by plants.

There is a fundamental possibility for returning the contaminated lands to agricultural use. General organizational and technological principles have been developed for farming in such conditions and many recommendations for individual directions have also been developed. Since the agricultural conditions of the forest area are quite specific, while the nature of the radioactive contamination is not yet well studied, evaluations may be given only upon receipt of specific data. Therefore, the following are required for realizing the capability of returning the land to agricultural use:

(a) restructuring of specialized forms in accordance with the levels of contamination of the territory for land use; elimination of the production of foodstuffs which go directly for human consumption, primarily production of foodstuffs, for planting, for technical purposes, and feed for livestock;

(b) implementation of special measures aimed at rigid fixing and consolidation of the radionuclides in a form inaccessible for plants for a period of time with subsequent plowing under by introducing sorbents (a clay suspension, zeolites) into the upper contaminated soil layer; and

(c) implementation of special decontamination measures by removing the contaminated surface of the soil directly be mechanical means from sodded sectors or after consolidation by chemical agents (SKS-65 gp latex emulsion).

The measures to include the lands in the agricultural cycle will be differentiated in time and level of contamination of the territory.

In the evacuation zone and in the rigid control zone, the agricultural harvesting operations are being conducted in the conventional order with consideration of special measures developed jointly with the State Agricultural Industry of the USSR and the Ukrainian SSR, and the Ministry of Health of the USSR.

With respect to the surface contamination of vegetation and soil, the basic special specifications on the organization and technology for performing operations are reduced to the following for 1986:

(a) to reduce to a minimum the mechanical treatment of soils with increased dust formation;

(b) grain and industrial crops will be harvested by direct combining and will be used regardless of the level of actual contamination of production (after curing in warehouses) for food purposes, for feed, for planting, and for industrial processing; and

(c) an obligatory specification is application to the fields after harvesting ~~perinial~~ perennial grasses and winter crops of lime, mineral fertilizer, and sorbents, which increase the fertility of the soil and reduce the entry of radionuclides into agricultural production.

In solving the issue about the fate of contaminated forests, one must proceed from their well known absorptive role in forest-steppe and steppe regions as an accumulator and storer of moisture.

Investigations also showed that in conditions of radioactive contamination, the forest is also an accumulator of radioactive substances, first in the crowns, and then in the bedding. Radionuclides consolidated in the bedding will long be excluded from the radiation chains.

Therefore, today, the opinion of the majority of specialists, the most expedient method for dealing with contaminated forests is to intensify the fire fighting service.

To date, based on the evaluations of the situation with respect to contamination of the soil and vegetation cover of the 30-km zone, special agrotechnical and decontamination measures have been developed and are being implemented which will make it possible to begin returning the contaminated lands to the national economy. This complex of measures includes a change in the traditional systems for working soils in this region and the use of special compounds for dust suppression, changing the harvesting methods and the processing of the harvest, etc.

The level of radioactive contamination of living quarters and buildings in the agricultural region within the 30-km zone fluctuates widely. The typical construction materials are bricks, wood (boards), both painted and unpainted, with different states of the paint, shingles and roofing tin.

Decontamination was performed by spraying the surface at a decontaminating solution rate of 10-15 ℓ/m^2. Automatic spraying stations were used for the treatment.

As a result of the decontamination, the radiation dosage rate from the buildings was reduced to background radiation levels for this particular region, the β- contamination basically did not exceed 1000 β-particles/(cm$^2\cdot$ min).

After washing off the buildings, the radioactive contamination of the soil near the walls was increased by a factor of 2-2.5 and therefore, the soil along the walls was turned over or was removed by bulldozers and transported away.

Decontamination of transport systems was performed by spraying and by steam ejection methods using the above cited solutions.

Appendix 4

EVALUATION OF THE QUANTITY, COMPOSITION and DYNAMICS OF EJECTION OF RADIO-ACTIVE SUBSTANCES FROM THE DAMAGED REACTOR

4. Evaluation of the Quantity, Composition and Dynamics of Ejection of Radioactive Substances from the Damaged Reactor

4.1. The volume of radioactive substances ejected from the reactor.

The results of aerial gamma photography of the region of the Chernobyl AES and the territory of the country, performed by UNKhV from Soviet Air Force helicopters and from helicopters assigned to the State Committee on Hydrometerorology which were flown from 1 May through 25 June 1986 were used as the primary information for evaluating the volume of radioactive substances ejected from the damaged reactor.

In order to determine the reserve of radionuclides, data from the aerial gamma photography were plotted on the terrain map, isodose lines were drawn and areas calculated which were included by these curves. The results of the evaluations of the recording for 26 July 1986 are cited in absolute and relative values in Table 4.1.

It follows from the data in this table that the total radioactivity of the fission products ejected from the damaged reactor and settled on the soil in the 30-km zone is 8-14 MCi. Analysis of the acquired results showed that by the time the intensive ejection of fission products from the reactor had halted on 6 July 1986, the volume of radionuclides in the 30-km zone was approximately 20 MCi. It should be especially noted that more than half of all of this activity is located in a zone with $r > 20$ mr/hr in a surface area which is a total of 17% of the contaminated territory and includes the territory of the Chernobyl AES. According to the results of an analysis of the aerial gamma photographs of the State Committee for Hydrometerorology, beyond the special zone the activity of radionuclides settled to the soil is 10-30 MCi. It follows from an analysis of the data that the total activity of radionuclides ejected from the damaged reactor into the environment does not exceed 50 MCi, i.e., is approximately 3-4% of the total activity of the fission products in the reactor of the fourth unit of the Chernobyl AES on 6 May 1986.

~~(break in text)~~

An independent evaluation of the amount of fission products discharged from the damaged reactor ~~were~~ was performed by specialists from the Khlopin Radium Institute. Based on scanning by a collimating detector from a helicopter which flew over the commercial site at an altitude of 300 m, analysis of the samples in the 30-km zone, and the use of the correlations between the gamma-activity of Ce and the α-activity of Pu, the volume of fuel located in these zones was determined. These values are somewhat higher than those acquired from data an analysis of the isodoses. The relative values of distribution of the fission products on the commercial site and on the roof of the Chernobyl AES are shown in Table 4.1.

Table 4.1. Evaluation of the Volume of radionuclides in the 30-km zone of the Chernobyl AES region on 26 May 1986.

KEY: (1) Order number

(2) Zone with r, in mr/hr

(3) Area km^2

(4) Activity

(5) Absolute, MCi

(6) Relative, %

(7) Total:

2a

1 № п/п	Зона с Р, мР/час 2	Площадь, $км^2$ 3	Активность 4	
			Абсолютная, Мки 5	Относительн., % 6
1	Р > 20	870	5÷8,7	63,0
2	I0 < Р ≤ 20	480	0,8÷1,4	I0,2
3	5 < Р ≤ I0	II00	1÷1,7	I0,8
4	3 < Р ≤ 5	2780	I,3÷2,2	I6,0
	Итого: 7	5230	8÷I4	I00,0

№ 2B

№ 4B

№ 3

0,2	0,4	0,4	0,3	0,3	0,2	0,I	0,05	0,03	0	0
0,3	0,4	0,45	0,5I	0,8	0,5	0,2	0,I	0,05	0	0
0,3	0,37	0,5	0,8	O	0,3	0,I5	0,05	0,03	0	0
0,5	0,6	0,7	0,8	0,65	0,55	0,42	0,35	0,22	0,I6	0,I0
0,5	0,6	0,8	0,85	0,7	0,55	0,4I	0,37	0,25	0,I5	0,09
0,25	0,3	0,4	0,4	0,3	0,25	0,I8	0,I5	0,I	0,08	0,04

№ 10 № 8

50м 50м

№ 9 № 7B

Fig. 4.1. Relative distribution of γ-radiating fission products at the commercial site of the Chernobyl AES

4.2. The composition of the fission products (PD) of uranium and other radionuclides ejected from the damaged reactor.

The primary information for evaluating the composition of radionuclides ejected from the damaged reactor are the radiometric investigations of samples of aerosols and soil samples performed by the Khlopkin Radium Institute and the Kurchatov Institute of Atomic Energy in the period from 6 through 30 May 1986. These data showed that the composition of the fission products (with the exception of gaseous fission products, such as I. Te, and Cs) ejected from the damaged reactor are close to the composition of the fission products in the fuel in the reactor itself. The averaged data from soil and grass cover investigations in the region from 1.5 to 30 km from the reactor especially attests to this. These data are presented in Table 4.2.

An investigation of dozens of soil samples for content of transuranium elements based on alpha-radiation was performed in the 30-km zone. The radioactivity of the samples was 2-2000 Bq/g and was caused by 90% ^{242}Cm. Approximately 10% of the alpha-radioactivity is associated with

Table 4.2

Data from radiometric measurements of soil samples on 17 May 1986 on the northern track of the fallout within the 30-km zone

KEY: (1)Radionuclide

(2) Specific activity

(3) Sample content, %

(4) Content in radiated reactor fuel, %

(5) Bacquerel units/g

(6) Ci/km 2

* - 1 Ci = 3.7'10 10 Becquerel units

Радионуклид (1)	Удельная активность (2)		Содержание в пробе, % (3)	Содержание в облученном топливе реактора, % (4)
	Бк/г	Ки /км² (6)		
^{141}Ce	$3,2.10^3$	$5,1.10^2$	15,8	18,3
^{132}Te	$3,4.10^2$	$5,4.10^1$	1,7	0,22
^{131}I	$3,1.10^3$	$5,1.10^2$	15,8	2,8
^{103}Ru	$3,5.10^3$	$5,6.10^2$	17,3	21,4
^{106}Ru	$9,6.10^2$	$1,5.10^2$	4,6	16,9
^{134}Cs	$1,6.10^3$	$2,5.10^2$	7,7	4,5
^{137}Cs	$1,7.10^3$	$2,7.10^2$	8,3	3,4
^{95}Zr	4.10^3	$6,4.10^2$	19,8	23,0
^{140}Ba	$1,8.10^3$	$2,9.10^2$	9,0	9,6

Table 4.3

Composition of radionuclides in a soil sample from 8 May 1986 at a distance of 1.5 km from the reactor.

KEY: (1) Nuclide

(2) Specific activity, 10^5 Becquerel units/g

(3) α- radioactive nuclide

Note: 1 Becquerel unit = $2.7 \cdot 10^{-11}$ Ci.

Нуклиды (1)	Удельная активность, 10^5 Бк/г (2)
^{95}Zr	36
^{103}Ru	1,7
^{131}I	6,3
^{140}Ba	21
^{141}Ce	28
^{144}Ce	17
^{239}Np	6,4
α-радиоактивные (3) нуклиды	0,13

Примечание: 1 Бк = $2,7 \cdot 10^{-11}$ Ки.

Table 4.4

Radionuclide composition of α-radiators in air and certain reference relations of the nuclides.

KEY: (1) Nuclide

(2) Activity in air, Ci/1

Table 4.5

Radionuclide composition of α-radiators in soil samples and certain reference relations of the nuclides.

KEY: (1) Nuclide

(2) Soil activity, Ci/sample

Таблица 4.4

Радионуклидный состав α-излучателей в воздухе и некоторые реперные отношения нуклидов

Нуклид	Активность воздуха, Ки/л	Нуклид / ^{144}Ce	$\frac{^{239+240}Pu}{^{242}Cm}$	$\frac{^{238}Pu}{^{239+240}Pu}$
^{238}Pu	3.10^{-14}	4.10^{-4}	$7,8.10^{-2}$	0,55
^{239}Pu	$2,2.10^{-14}$	$2,9.10^{-4}$		
^{240}Pu	3.10^{-14}	4.10^{-4}		
^{242}Cm	$6,7.10^{-13}$	$8,7.10^{-3}$		

Таблица 4.5

Радионуклидный состав α-излучателей в пробах почвы и некоторые реперные отношения нуклидов

Нуклид	Активность почвы, Ки / проба	Нуклид / ^{144}Ce	$\frac{^{234+240}Pu}{^{242}Cm}$	$\frac{^{238}Pu}{^{239+240}Pu}$
$^{239+240}Pu$	$2,5.10^{-10}$		$3,5.10^{-2}$	0,72
^{238}Pu	$1,8.10^{-10}$			
^{242}Cm	$7,1.10^{-9}$			

Table 4.6

Radionuclide composition of α-radiators in soil samples based on measurements form the Khlopkin Institute of Radium

KEY: (1) Nuclide

(2) Activity, Bq/hr

Таблица 4.6

Радионуклидный состав α - излучателей в образцах почвы по данным измерений РИАН

Нуклид (1)	Активность (2) Бк/ч		Нуклид/^{144}Ce (1)		$^{239+40}Pu/^{242}C$		$^{238}Pu/^{239+40}Pu$	
	I	II	I	II	I	II	I	II
$^{239+240}Pu$	790	5,2	$6{,}3\cdot10^{-4}$	$5{,}2\cdot10^{-4}$	$8{,}1\cdot10^{-2}$	$7\cdot10^{-2}$	0,44	0,38
^{238}Pu	348	2,0	$2{,}8\cdot10^{-4}$	$2\cdot10^{-4}$				
^{242}Pu	9784	74	$7{,}8\cdot10^{-3}$	$7{,}4\cdot10^{-3}$				
^{144}Ce	$1{,}23\cdot10^{6}$	$1{,}0\cdot10^{4}$	1	1				

Table 4.7

Radionuclide composition of aerosols at an altitude of 200 m, Bq/1.

KEY: (1) Measurement data

(2) Nuclides

Нуклиды (2)	Дата измерений (1)			
	09.05.86	II.05.86	I3.05.86	24.05.86
^{95}Zr	8,9	I0	0,68	0,06
^{95}Nb	5,8	II	I,2	-
^{99}Mo	3,8	-	-	-
^{99}Tc	I6	7,5	0,28	-
^{103}Ru	36	3I	0,94	I,2
^{131}J	58	45	I,0	0,6
^{132}Te	23	8,5	0,I9	-
^{137}Cs	-	2,0	0,37	0,I
^{134}Cs	-	-	-	0,05
^{140}Ba	I0	4,8	-	-
^{140}La	I2	5,6	0,23	-
^{141}Ce	6,0	8,4	0,4I	0,2
^{144}Ce	-	8,0	0,4I	0,4

Table 4.8

Radionuclide composition of aerosols at an altitude of 3 m, in Bq/l.

KEY: (1) Nuclide

(2) Concentration on 12 May 1986 at points.

Таблица 4.8

Радионуклидный состав аэрозолей на высоте 3 м, Бк/л

Нуклиды (1)	Концентрация I2.05.86 в точках (2)			
	3	IO	8	9
^{95}Zr	44	7,5	3,3	I,8
^{95}Nb	-	9,8	3,7	2,0
^{103}Ru	I55	67	I,6	I,3
^{131}J	I95	IOO	I,0	2,9
^{132}Te	42	24	0,25	0,2
^{140}Ba	7,5	2,8	2,4	I,8
^{140}La	7,5	4,4	2,3	I,2
^{141}Ce	-	6,0	3,0	I,3
^{144}Ce	-	6,6	2,8	-

Table 4.9

Concentration of radioactive aerosols on 22 May 1986 at an altitude of 3 m, in Bq/1.

KEY: (1) Nuclide

(2) Point of Measurements

Таблица 4.9

Концентрация радиоактивных аэрозолей 22.05.86 на высоте 3 м, Бк/л

Нуклиды (1)	Точки (2)		
	2В	4В	7В
^{95}Zr	I,7	I,8	4,I
^{95}Nb	2,0	2,I	3,7
^{103}Ru	I,0	I,2	29,2
^{106}Ru	0,6	0,9	I0,0
^{132}Te	-	I,3	I,3
^{137}Cs	-	0,I9	0,9
^{140}Ba	I,I	0,5	I,I
^{140}La	0,6	I,7	I,7
^{141}Ce	I,I	I,6	2,3
^{144}Ce	I,0	I,5	3,4

isotopes with masses of 238, 239, and 240. The radioactivity of ^{238}Pu is approximately 40-70% with respect to the sume of the radioactivities of the ^{239}Pu and ^{240}Pu nuclides.

Soil samples in the direction of the south-north ejection at the end of the sector with the contaminated forest in the 1.5-km radius may be considered as having relatively greater concentration of transuraniums. The results of the analysis of this sample taken on 8 May 1986 from the surface near a road are presented in Table 4.3. The total value of the α-radiation of the sample is $1.3 \cdot 10^4$ Bq/g.

The radionuclide composition of the α-radiators in air samples (filters) and soil samples based on data from measurement by the Kurchatov Institute of Atomic Energy are presented in Tables 4.4 and 4.5 and from data from the Khlopin Radium Institute, in Table 4.6

Investigation of the aerosol composition of air samples (with pumping through a filtering fabric) also attests to the presence of transfer with airborne dust of both volatile and slightly volatile chemical elements without expressed fractionation, with the exception of iodine, ruthenium and tellurium. The aerosol samples were taken at an altitude of 200 m above the damaged reactor and at an altitude of 3 m above the earth at 10 fixed points on the commercial site. Table 4.7 presents the results of measurements of aerosols at an altitude of 200 m, while Table 4.8 presents those from an altitude of 3 m. Points 3 and 10 of Table 4.8 are related to the northern direction from the building of the Chernobyl AES, while points 8 and 9 are related to the southern. All four points are positioned on a line which passes approximately 150 m east of the damaged reactor (see Fig. 4.1).

Data from Table 4.7 attest to the abrupt reduction in the specific activity of the aerosols after 6 May 1986, which points to the dynamics of ejection of fission products from the damaged reactor in time. Until 7 May 1986 the reactor was a source of elevated ejection of radionuclides, while after 6 May 1986, it ceased to be such a determining factor in the formation of aerosol radioactivity above the commercial site.

This formation began to be determined by the processes of dust formation and secondary wind transfer of radionuclides over the site as a whole (Table 4.9).

The concentration of radioactive aerosols at an altitude of 200 m was the same as at an altitude of 3 m (May 9 and May 11 data for the 200 m and May 12 data for the 300 m altitude). After May 12 the concentration of the aerosols at an altitude of 200 m were reduced by approximately a factor of 100, while at an altitude of 3 m over the commercial site, they were altered little thereafter. This is evident from a comparison of data in Table 4.8 and data in Table 4.9. The latter presents the results of an identification of the concentrations of aerosols at an altitude of 3 m above the commercial site performed on 22 May 1986 at points disposed relatively closely to the points indicated in Table 4.8 (see Fig. 4.1).

The presence of "hot" particles, primarily enriched with a single type radionuclide, was revealed in the compositions of the air and fallout samples. Figures 4.2 and 4.3 cite the results of measurements of the composition of radionuclides of such particles. As is apparent, there are particles which contain essentially only Cs or Ce. There is a 10-time increase in the content of ^{140}Ba as compared with the theoretical value.

4.3. Dynamics of ejection of radionuclides from the damaged reactor.

Materials from systematic investigations of the radionuclide composition of samples of aerosols collected over the fourth unit of the

Chernobyl AES from 26 April 1986 were used as the primary information for revealing the dynamics of ejection of radionuclides from the damaged reactor. The results of such investigations are presented in Table 4.10 and Figure 4.2.

Analysis of these data led to the conclusions that the ejection of radionuclides beyond the damaged unit of the Chernobyl AES are

Table 4.10

Relative content of radionuclides in air above the Chernobyl

AES, δ_i, % *

KEY: (1) Nuclides

(2) Relative content δ of radionuclides in fuel on 26 April 1986.

*) $\delta_i=(A_i/\sum_i A_i)100\%$, where A_i is the activity of the i radionuclide; rounded data are cited.

Таблица 4.10

Относительное содержание радионуклидов в воздухе над ЧАЭС δ_i, % ж)

1 Нуклиды	26.04.86	29.04.86	02.05.86	03.05.86	04.05.86	05.05.86	Относительное содержание δ_i радионуклидов в топливе на 26.04.86 2
^{95}Zr	4,4	6,3	9,3	0,6	7,0	20	3,6
^{95}Nb	0,6	0,8	9,0	I,3	8,2	I8	3,8
^{99}Mo	3,7	2,6	2,0	4,4	2,8	3,7	3,9
^{103}Ru	2,I	3,0	4,I	7,2	6,9	I4	3,9
^{106}Ru	0,8	I,2	I,I	3,I	I,3	9,6	2,I
^{131}J	5,6	6,4	5,7	25	8,2	I9	2,3
$^{132}Te + ^{132}J$	40	3I	I7	45	I5	8,6	6,4
^{134}Cs	0,4	0,6	0,6	I,6	0,6	-	0,6
^{136}Cs	0,3	0,4	0,5	0,9	-	-	0,I
^{137}Cs	-	-	I,4	3,7	I,3	2,2	0,4
^{140}Ba	3,2	4,I	8,0	3,3	I3	I2	3,8
^{140}La	II	4,7	I5	2,3	I9	I7	4,0
^{141}Ce	I,4	I,9	7,6	0,9	6,4	I5	3,6
^{144}Ce	I,6	2,4	6,I	-	5,I	II	3,4
^{147}Nd	I,4	I,7	2,5	-	2,I	5,4	I,4
^{239}Np	23	3,0	II	0,6	2,8	6,8	56,7
$\sum_i A_i$, $\frac{Ки}{л}$	$3,6\cdot10^{-7}$	$3,2\cdot10^{-7}$	$5\cdot10^{-8}$	$7\cdot10^{-8}$	$1\cdot10^{-6}$	$7\cdot10^{-9}$	

ж) $\delta_i = (A_i / \sum_i A_i) \cdot 100\%$, где A_i – активность i-го радионуклида; приведены округленные данные.

Table 4.11

The values of Ω *) which characterize the output of nonvolatile fission products relative to ^{131}I in an example of ^{95}Zr and ^{141}Ce for the period from 26 April through 13 May 1986.

KEY: 1) Date

*) EXTRACT IN SQUARE in air

in fuel,

A is the activity of the radionuclide

**) Initial ejection

Дата (1)	^{95}Zr	^{141}Cr
26.04 **)	11	32
29.04	2,0	8,0
02.05	1,2	I8,3
03.05	I05	85,3
04.05	3,8	5,2
05.05	I,8	3,I
08.05	26	5I
11.05	23	37
I3.05	I3	22

*) $$\boxed{\Omega_i = \frac{[A(^{131}J)/A(i)]}{[A(^{131}J)/A(i)]}} \quad \begin{array}{l} \text{в воздухе} \\ \text{в топливе ,} \end{array}$$

$i = {}^{95}Zr;\ {}^{141}Ce;$

A – активность радионуклида .

**) Первоначальный выброс.

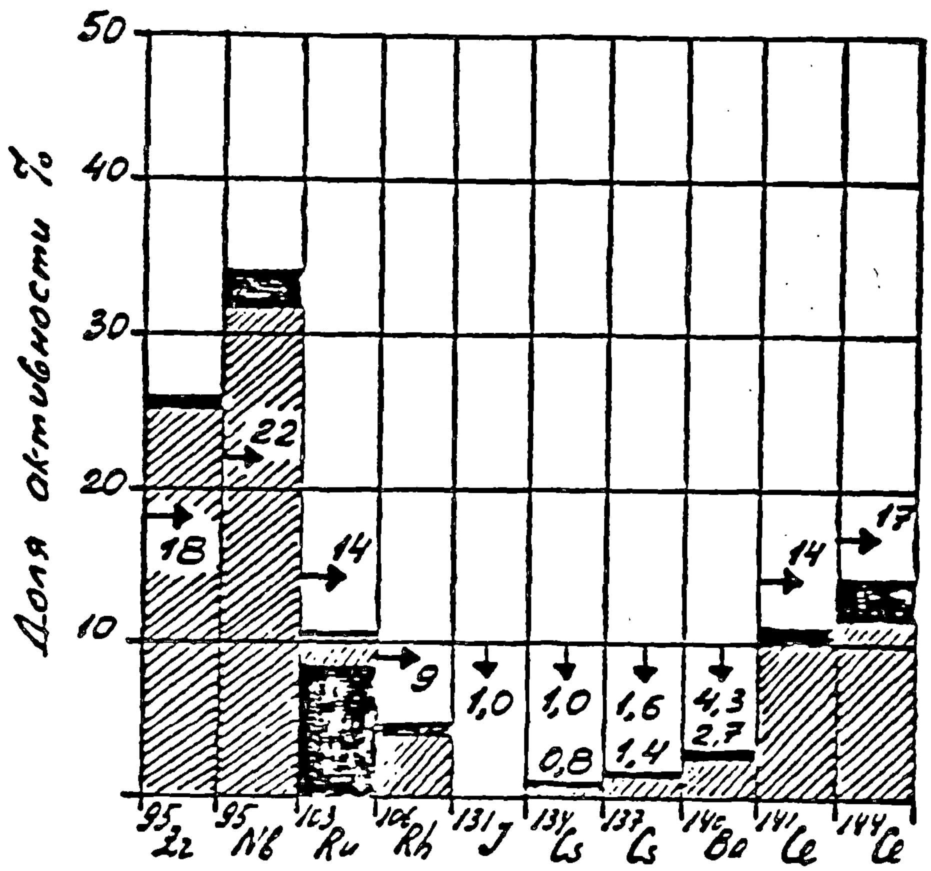


Fig. 4.2. Percentage of activity of radionuclides in the initial sample (▬) and sediment (▨).

The arrows indicate the calculated values of the percentage of activity. The total gamma-activity in the initial sample is 1.0×10^{-5} Ci/g.

KEY: Percentage of activity %.

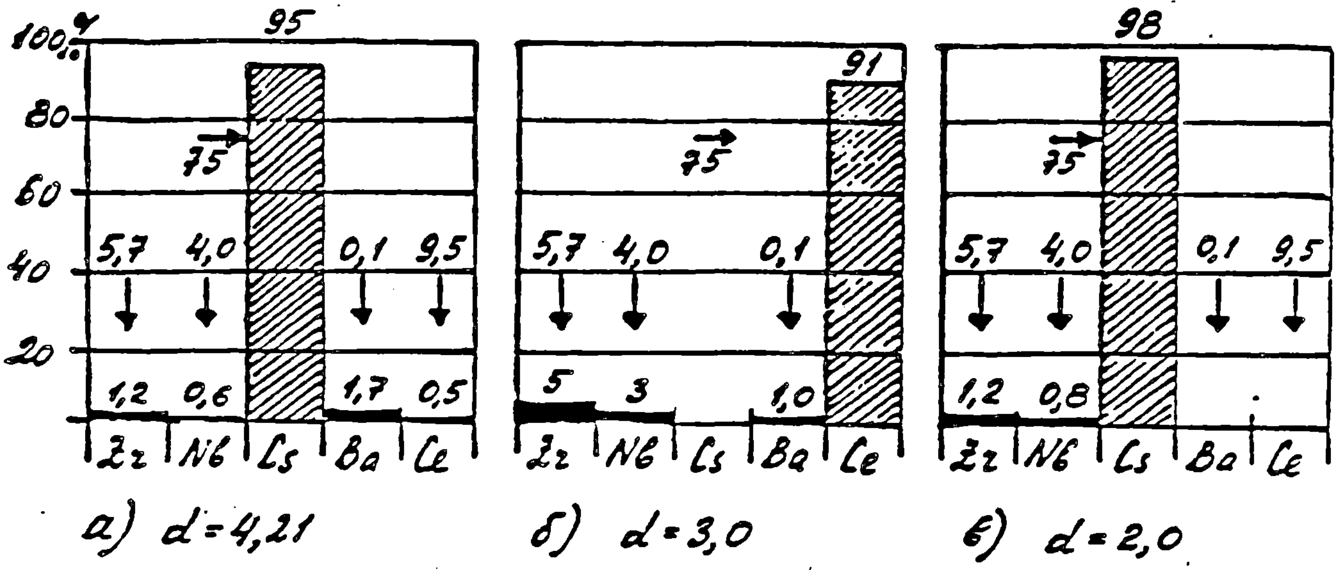


Fig. 4.3. Percentage of radionuclides in the "hot" particles fractionated in Klerich solutions with densities of 4.2; 3.0; 2.0; and 2.0 g/cm from sand and dust samples (# 16-LPD) (Tr. Note: "L" is unknown; PD is fission products.)

Content of cesium-134 and cesium-137(total) in (a) 5.0×10^{-9} g/g and (c) 1.4×10^{-9} g/g.

extended during the process which consists of several stages. The dynamics of the process of ejection are very convexly characterized by the data in Table 4.11, in which the dimensionless values of the iodine-131-normed ejection of nonvolatile fission products in time are presented.

Mechanical ejection of dispersed fuel occured in the first stage as a result of the explosion in the reactor. The composition of the radionuclides of this stage of ejection approximately corresponds to the composition of the fission products in irradiated fuel, but is enriched with volatile nuclides of iodine, tellurium and cesium.

In the second stage from 26 April through 2 May 1986 the power of the ejection beyond the damaged unit was reduced due to measures undertaken to put out the graphite fire and to filter the substances coming from the active zone. In a first approximation, the reduction in the ejection power in this period may be represented by

$$Q(\tau) = Q_0 e^{-0.5\tau}, \qquad (4.1)$$

where Q_0 is the ejection power immediately after the explosion Curies/24-hour period) and

τ is the time after the start of the accident (24-hour periods).

In this period the composition of the radionuclides in the ejection is also similar to their composition in the fuel. In this stage there was a transfer of the finely dispersed fuel directly from the reactor by a steam of hot air and the graphite combustion products.

The third stage of the discharge is characterized by a rapid rise in the radionuclide discharge dose beyond the reactor unit. In the initial part of this stage, transfer of the volatile components, especially iodine, dominates, and then the composition of the radionuclides again approaches their composition in the irradiated fuel (on 6 May 1986). The discharge power of the fission products in the third stage in a first approximation may be described by the expression

Таблица 4.12

Результаты измерений мощности выброса из реактора и оценки запаса в облаке по анализам проб воздуха

Дата отбора 1	Время отбора 2	Удельная активность воздуха, Ки/литр 3	Мощность выброса, Ки/сутки 4	Оценка запаса в облаке, Ки/км^3 5
09.05	18.30	$54\cdot10^{-10}$	12600	
11.05	13.15	$38\cdot10^{-10}$	8700	
13.05	13.15	$2\cdot10^{-10}$	420ж)	
16.05	13.15	$8\cdot10^{-10}$	1680жж)	
19.05	13.15	$0{,}05\cdot10^{-10}$	50	
22.05	09.30	$0{,}05\cdot10^{-10}$	50	
23.05	09.30	$0{,}02\cdot10^{-10}$	20	
24.05	09.30.	$1\cdot10^{-10}$		100
25.05	09.30	$3\cdot10^{-10}$		300
26.05	09.30	$0{,}2\cdot10^{-10}$		20
27.05	09.30	$0{,}2\cdot10^{-10}$		20
28.05	09.30	$0{,}2\cdot10^{-10}$		20
29.05	09.30	$0{,}1\cdot10^{-10}$		10
30.05	13.00	$2\cdot10^{-10}$		200жжж)
01.06	09.30	$0{,}2\cdot10^{-10}$		20
02.06	17.00	$0{,}25\cdot10^{-10}$		25
03.06	14.30	$0{,}12\cdot10^{-10}$		12
04.06	09.30	$0{,}08\cdot10^{-10}$		8
05.06	09.30	$0{,}12\cdot10^{-10}$		12
06.06	09.30			100жжжж)

ж) Отбор производился вне шлейфа, что привело к занижению концентрации нуклидов.

жж) Данные сверены с результатами измерений Института экспериментальной метеорологии.

жжж) Отбор пробы проведен в дневное время. Это обусловило увеличение концентрации нуклидов из-за повышенной запыленности в связи с работами на площадке АЭС.

жжжж) После дождя.

Results of measurements of the discharge dose from the reactor and evaluation of the excess in the cloud from air sample analyses.

KEY: (1) Sample date

(2) Sample time

(3) Specific activity of air, Ci/1

(4) Discharge dose

(5) Evaluation of the excess in the cloud Ci/km

*) Sampling was performed without a loop, which led to a reduction in the concentration of nuclides.

**) Data are collated with the results of measurements by the Experimental Meteorology Institute.

***) The sampling is performed during the day. This caused an increase in the concentration of nuclides due to the higher dust content associated with operations at the AES site.

****) After the rain.

$$Q(\tau) = \text{const}\, e^{\alpha\tau},$$

where α = (6-8) $\times 10^{-2}$ 1/hr.

Such a discharge character is apparently caused by heating of the fuel in the active zone to a temperature above 2000°C due to residual heat liberation. Here, there was a leak of the fission products from the uranium dioxide and their transfer either in aerosol form or in the graphite combustion products (graphite particles) as a result of the temperature-dependent migration of the fission products and the possible carbidization of the uranium dioxide.

The last - the fourth stage, began after May 6 and is characterized by a rapid reduction in the output of the fission products from the fuel and an essential halt in discharge (table 4.13), which was the result of the special measures taken and the formation of harder-to-melt compounds of the fission products as a result of their interaction with the introduced materials.

Basic conclusions:

1. The total discharge of radioactive substances (without radioactive noble gases) was approximately 50MCi, which corresponds to 3.5% of the total volume of radionuclides in the reactor at the moment of the accident. These data were calculated on 6 May 1986 with consideration of radioactive decay.

2. The composition of the radionuclides in the emergency ejection approximately corresponds to their composition in the fuel of the damaged reactor, differing from it only in an increased content of volatile iodine and tellurium.

3. Correlated quantitative information about the change in the discharge dose in time and the composition of the radionuclides discharged from the damaged reactor is cited in Tables 4.13 and 4.14 and in Fig. 4.4.

Table 4.13

Daily discharge q of radioactive substances into the atmosphere from the damaged unit (without the radioactive noble gases)

Дата [1]	Время после аварии, сут [2]	q МКи[жж)] [3]
26.04	0[1)]	I2
27.04	I	4,0
28.04	2	3,4
29.04	3	2,6
30.04	4	2,0
0I.05	5	2,0
02.05	6	4,0
03.05	7	5,0
04.05	8	7,0
05.05	9	8,0
06.05	I0	0,I
09.05	I4	≃0,0I
23.05	28	$20\cdot I0^{-6}$

(1) Date

(2) Time after the accident, days

(3) q MCi

1) Первоначальный выброс.
Приведено значение с учетом распада на 06.05.86.
В момент выброса активность составляла 20–22 МКи.
Состав выброса см. в табл. 4.I4.

ж) Погрешность оценки выброса – ± 50%. Она определяется погрешностью дозиметрических приборов, радиометрических измерений радионуклидного состава проб воздуха и почвы, а также погрешностью, обусловленной усреднением выпадений по площади.

жж) Значения q пересчитаны на 06.05.86 с учетом радиоактивного распада (выброс 26.04.86 составил ≃ 20МКи на этот момент времени). Состав выброса см. в табл. 4.I4.

(1) Initial discharge

The value is cited with consideration of the decay on 6 May 1986. The activity was 20-22 MCi at the moment of discharge. For the composition of the discharge see Table 4.14.

*) The error of evaluation is 50%. It is determined by the error rates of the dosimetric instruments, the radiometric measurements of the radionuclides composition of the air and soil samples and by the error rate caused by averaging fallouts by site.

**) The values of q are calculated on 6 May 1986 with consideration of radioactive decay (the discharge of 26 April 1986 was $\approx$ 20 MCi at that moment in time). For the composition of the discharge products, see Table 4.14.

Table 4.14

Radionuclide composition of the discharge of the damaged unit of the Chernobyl AES *)

(1) Нуклид жжж)	(2) Активность выброса, МКи		(3) Доля активности, выброшенной из реактора на 06.05.86, %
	26.04.86	06.05.86 жж)	
I33Xe	5	45	(4) Возможно до I00
85	0,I5	-	-"-
85	-	0,5	-"-
I3II	4,5	7,3	20
I32	4	I,3	I5
I34	0,I5	0,5	I0
I37	0,3	I	I3
99	0,45	3	2,3
95	0,45	3,8	3,2
I03	0,6	3,2	2,9
I06	0,2	I,6	2,9
I40	0,5	4,3	5,6
I4I	0,4	2,8	2,3
I44	0,45	2,4	2,8
89	0,25	2,2	4,0
90	0,0I5	0,22	4,0
239	2,7	I,2	3,2
238	$0,I.I0^{-3}$	$0,8.I0^{-3}$	3%
239	$0,I.I0^{-3}$	$0,7.I0^{-3}$	-"-
240	$0,2.I0^{-3}$	$I.I0^{-3}$	-"-
24I	0,02	0,I4	-"-
242	$0,3.I0^{-6}$	$2.I0^{-6}$	-"-
242	$3.I0^{-3}$	$2,I.I0^{-2}$	-"-

ж) Погрешность оценки - ± 50%, объяснение в примечании к табл. 4.I3.

жж) Суммарный выброс к 06.05.86.

жжж) Приведены данные по активности основных радионуклидов, измеряемых при радиометрических анализах.

KEY: (1) Nuclide

(2) Discharge activity, MCi

(3) Percentage of activity discharged from the reactor on 6 May 1986 %

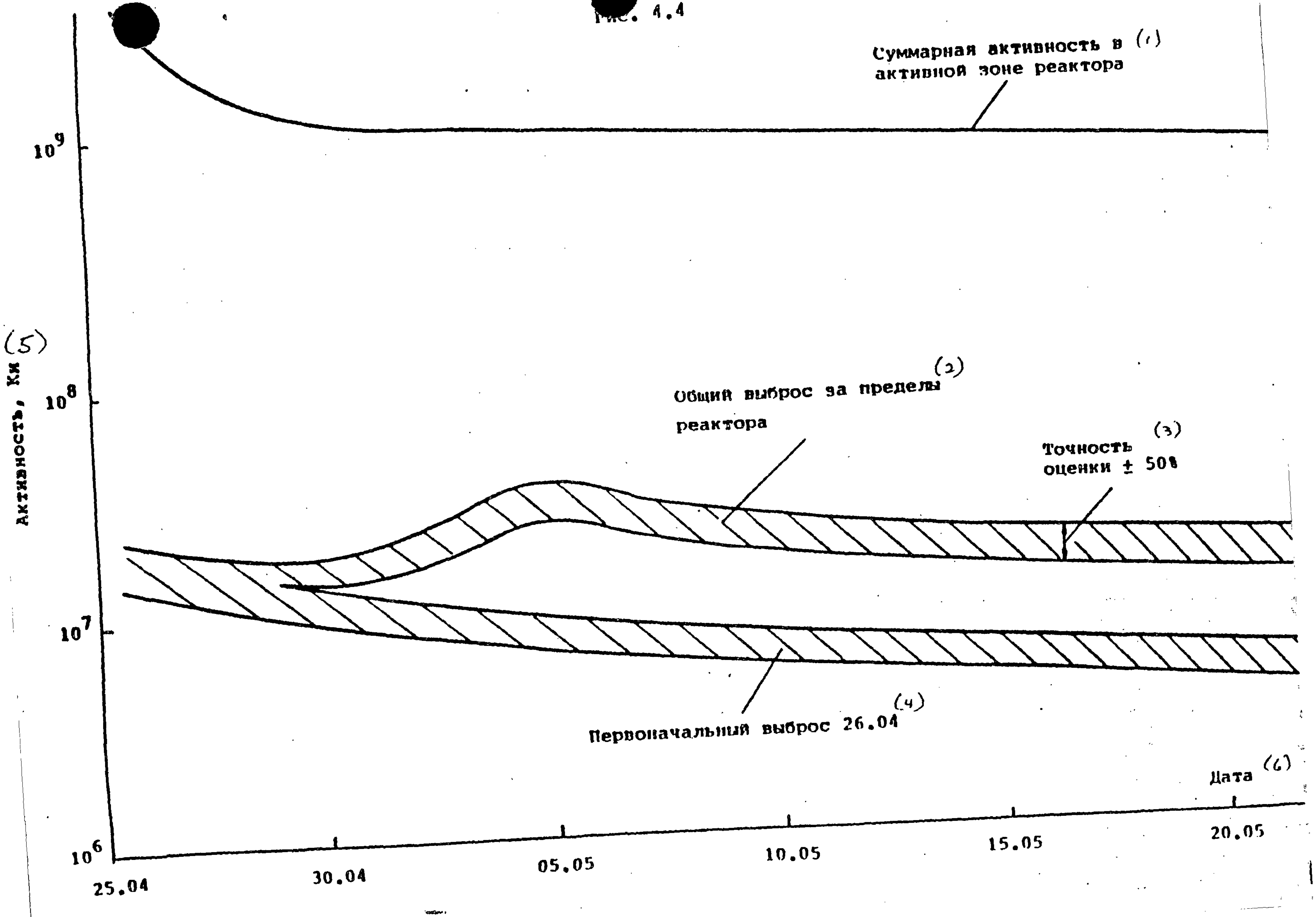
(4) Possibly to 100

*) Evaluation error is ±50%, explanation in the note to Table 4.13.

**) Total discharge by 6 May 1986.

***) Data are cited based on the activity of the basic radionuclides measured during radiometric analyses.

Рис. 4.4



Суммарная активность в (1)
активной зоне реактора
Общий выброс за пределы (2)
реактора
Точность (3)
оценки ± 50%
Первоначальный выброс 26.04 (4)
Активность, Ки (5)
Дата (6)
10^9
10^8
10^7
10^6
25.04
30.04
05.05
10.05
15.05
20.05

KEY: (1) Total activity in the active zone of the reactor.

(2) Total discharge beyond the reactor

(3) Evaluation precision ± 50%

(4) Initial discharge, 26 May 1986

(5) Activity, Ci

(6) Date

Appendix 5

ATMOSPHERIC TRANSFER AND RADIOACTIVE CONTAMINATION

OF THE ATMOSPHERE AND THE TERRAIN

Appendix 5

5. ATMOSPHERIC TRANSFER AND RADIOACTIVE

CONTAMINATION OF THE ATMOSPHERE AND TERRAIN

5.1 Formation of the basic source of contamination - the cloud and gas stream as a result of the accident at the Chernobyl AES.

As a result of the accident, a significant volume of the radionuclides accumulated in the reactor during its operations was ejected beyond the plant.

At the time of the accident a cloud was formed, which subsequently formed a radioactive track in the terrain in the western and northern directions in accordance with the meteorological conditions for transfer of air masses. Then, for an extended period a jet of gaseous, volatile, and aerosol products continued to leak from the accident zone. The strongest jet was observed for the first 2-3 days after the accident in a northerly direction, where the radiation levels on 27 April reached 1000 mr/hr and on 28 April, 500 mr/hr at a distance of 5-10 km from the accident site (at an altitude of 200 m). The altitude of the jet on 27 April, based on aircraft data, exceeded 1200 m in a northwesterly direction at a distance of approximately 30 km from the accident site, where the radiation levels were approximately 1 mr/hr. On subsequent days the altitude of the jet did not exceed 2-0-400 m.

Portions of the contaminated air masses (clouds and portions of the jet of radioactive products) were propagated according to the wind directions to great distances in the territory of the USSR.

Based on the results of gamma-spectral analysis the following fission products accumulated in the reactor were identified in the air samples: Zr^{95}, Nb^{95}, Mo^{99}, Ce^{141}, Ce^{144}, I^{131}, Te^{132}, I^{132}, Ru^{103}, Ru^{106}, Ba^{140}, La^{140}, Cs^{137}, and Nd^{147}, as well as isotopes of directed activity: Np^{239} and Cs^{134}.

A characteristic feature of the identified radioactive products in the atmosphere is their enrichment with iodine and cesium radionuclides. Table 5.1 cites the calculated coefficients of fractionation (enrichment) of the radionuclides relative to zirconium-95 based on the results of an analysis of the samples of atmospheric aerosol in the first days after the accident in the near zone.

5.2 Meteorological conditions for propagation of the radioactive products from the AES

On 26 April 1986 the region of the AES was in a low gradient baric field with a slight wind of varying directions. At an altitude of 700-800 m and 1.5 km the area of the AES was on the southwestern periphery of a high pressure region with transfer of air masses in this layer to the northwest at a rate of 5-10 m/sec. This is confirmed by data from measurements of the levels of radiation and radioactive fallout along the trajectories of propagation of the air particles at an altitude of 0.7 and 1.5 km (Figures 5.1 to 5.3).

Further propagation of the air particles in the 0.7-1.5 km layer which departed the AES region on 26 April 1986 occurred in a northwesterly direction with a subsequent turn to the North.

In the near-earth layer of air on 26 April, the further transfer of the air masses occurred to the west and northwest with outlet of the air particle propagation trajectory on 26 and 27 April into regions on the Polish border, which is confirmed by data from measurements of radioactive fallout. On subsequent days, from the 27 through the 29 of April, based on data from aircraft measurements, the transfer of the radioactive products in the near-earth layer of air at an altitude of 200 m took place in northerly and northeasterly directions from the AES.

The meteorological conditions from 26 April through 29 April 1986 for the propagation of air masses in the region of the AES essentially determined the basic zone of the formed near-by radioactive fallout in the northwestern and northeastern directions from the AES. This is confirmed by aircraft measurements of the distributions of radiation levels in the terrain in the near zone, which were taken in subsequent days.

Subsequently, the slight discharge of radioactive products from the AES zone and their transfer continued primarily in a southerly direction up until 7-8 May 1986, causing radioactive fallout in a southerly direction.

5.3 Radioactive contamination in the region of the AES and evaluation of the total volume of radioactive products which settled in the near zone

A zone of near-by radioactive fallout was formed as a result of the meteorological conditions for propagation of radioactive products in the atmosphere and their deposition on the surface form 26 April through 30 April 1986. Aerial gamma photography of the distribution of the radiation levels in the terrain has been regularly performed from 29 April 1986 through the present. The chart of distribution of the levels of radiation on the terrain on 29 May 1986 is shown in Figure 5.4.

Based on data about the distribution of the gamma-radiation dose rate in the terrain at different moments in time, the total volume of radioactive products settled in the near zone of the radioactive track was evaluated.

Table 5.2 cites the results of integration of the sites limited by different isolevels of dose rate (in units of r/hr m^2), as well as the total volume of radioactive products in the near track (beyond the commercial site of the AES and up to a distance of 80 km), expressed in units of energy liberation (MeV/sec) and in units of activity (Ci).

The following radionuclides were identified in the near zone of the formed track of fallout in the period from 10 to 30 days after the accident: Mo^{99}, Zr-95, Ce^{141}, Ce^{144}, I^{131}, $Te^{132}+I^{132}$, Ru^{103}, Ru^{106}, Nb^{95}, $Ba^{140}+La^{140}$, Cs^{134}, Cs^{137}, Sr^{89}, Sr-90, and Y^{91}. Plutonium isotopes were detected on the earth's surface.

Based on data about the densities of terrain contamination in different zones of the track, the volumes of individual radionuclides which settled in the near zone of the track were evaluated in a comparison with the dosage rate on "D" + 15 (Table 5.3).

5.4. Radioactive contamination of the atmosphere and terrain of Soviet territory

A zone of radioactive fallout was formed from 26 April through 5 May 1986 in the territory of the USSR and beyond its confines according to the meterological transfer conditions.

The volume of radioactive products which settled in the European part of the USSR was approximately $1.2 \cdot 10^{8}$ (r/hr) m^{2} or 4.0×10^{17} MeV/sec. Thus, the total volume of fallout in the near and far zones is approximately 7.0×10^{17} MeV/sec on 5 May 1986 or approximately 3% of the total energy liberation of the radioactive products in the rector for this time.

The radioactive products entered the near-earth layer of the atmosphere at different times for different points depending on the meteorological transfer conditions.

Table 5.4 shows that change in the concentrations of certain radionuclides at a permanently manned observation point of the grid run by the State Committee on Hydrometeorology (Goskomgidromet) (Baryshevka, approximately 140 km to the southeast from the AES).

Figures 5.5 and 5.6 cite the results of identification of individual radionuclides in the near-earth layer of air at the Berezina preserve at a station for comprehensive background radiation monitoring located 120 km northeast of the city of Minsk.

Fallout of radioactive products from the atmosphere were tested in the Goskomgidromet grid using plotting boards exposed for several days.

Table 5.5 presents observation data from Kiev and Kalininingrad.

5.5 Radioactive contamination of rivers and bodies of water.

Information about radioactive contamination of rivers and bodies of water is acquired by performing isotopic analysis of one-liter water samples withdrawn from the surface level on a regular basis (every 1-3 days) at the mouths of the Pripyat, Teterev, Irpen, and Desna Rivers and at the Dneprovsk water intake (Vyshorod). Water samples were taken throughout the Kiev reservoir and during special ship-borne inspections beginning on 26 April 1986.

It is established as a result of the investigations that the radioactive contamination of the waters of rivers and bodies of water occurred primarily as a result of fallout of aerosols to the surface of the bodies of water and then due to run-off from contaminated spillways (there was essentially no rain in this region in May). The initial results of identification of the content of radioactive substances in the inspected bodies of water are cited in Table 5.6.

Table 5.7 cites the maximal concentrations for the observation period, beginning from 1 May 1986, in different water basins.

The highest concentrations of Iodine-131 in the Kiev reservoir in the region of the Dneprovsk water intake were observed on 3 May. They are explained by fallout of radioactive aerosols onto the surface of the Kiev reservoir and by the contaminated waters from the Pripyat River reaching this section.

The total beta-activity of the water in the Dnepr River in the region of the "Kiev" weather station was in a range of $(1-5)\times10^{-9}$ Ci/1 for the period 13-22 May 1986.

5.6 Plutonium contamination of the atmosphere and terrain.

Investigations of plutonium isotope contamination were performed using a special vehicle for taking air, soil, and grass samples on the ring-road around the Chernobyl AES beyond the limits of the 30-km zone (May 1986).

The results of the analysis are cited in Table 5.8. As is evident from the table, the concentration of plutonium in the air is below the allowable (the maximal permissible concentration of ^{239}Pu is 3×10^{-17} Ci/1) at all sampling points.

Table 5.9 cites data about the density of plutonium contamination of the soil and grass at different points.

Таблица 5.I

Радионуклид	^{144}Ce	^{141}Ce	^{140}Ba	^{131}I	^{103}Ru	^{106}Ru	^{137}Cs
Коэффициент фракциониро-вания	I,23	I,03	0,84	5,22	2,2I	I,7	5,64

KEY: (1) Radionuclide

(2) Fractionation coefficient

Таблица 5.2

Дата съемки следа, I986г.	Интеграл, (Р/час).м^2	Количество радиоактивных продуктов на следе		Доля от суммарного энерговыделения радиоактивных продуктов в зоне реактора на данное время, %
		МэВ/с	Ки	
II.05	7,9.I0^7	3,3.I0^{I7}	I,2.I0^7	I,6

KEY: (1) Track survey date, 1986

(2) Integral (r/hr) m

(3) Volume of radioactive products in the track

(4) Percentage of the total energy liberation of the radioactive products in the reactor zone at that particular time, %

(5) MeV/sec

(6) Ci

Таблица 5.3

Радио-нуклид 1	^{95}Zr	^{141}Ce	^{144}Ce	^{131}I	^{132}Te	^{103}Ru	^{106}Ru	^{140}Ba
Колич-во 2 на следе, Ки	I,8.I0^{5}	I,7.I0^{6}	I,3.I0^{6}	I,3.I0^{6}	2,5.I0^{5}	I,5.I0^{6}	5,7.I0^{5}	9,I.I0^{5}
Доля от 3 количества в зоне реактора,%	I,5	I,7	I,0	5,I	5,0	I,4	0,8	I,4

	^{134}Cs	^{137}Cs	^{89}Sr	^{90}Sr	^{91}Y	Суммарное количество 4
	I,3.I0^{5}	2,8.I0^{5}	6,2.I0^{5}	8,5.I0^{4}	6,4.I0^{5}	I,I2.I0^{7}
	0,6	I,9	I,2	0,85	0,85	I,4

KEY: (1) Radionuclide

(2) Volume in the track, Ci

(3) Percentage of the volume in the reactor zone, %

(4) Total volume

Таблица 5.4

Ки/м3 (1)

Дата отбора проб (2)	^{131}I	^{95}Zr	^{140}Ba	^{137}Cs
26-27.04	$3,2.10^{-12}$			
27-28.04	$5,7.10^{-15}$			
28-29.04	$2,4.10^{-13}$	$2,7.10^{-14}$	$2,4.10^{-14}$	$5,4.10^{-14}$
29-30.04	$2,2.10^{-11}$	$1,6.10^{-12}$	$2,1.10^{-12}$	$8,4.10^{-12}$
30.04-01.05	$8,3.10^{-9}$	$2,2.10^{-9}$	$5,7.10^{-9}$	2.10^{-9}
1-2.05	$1,1.10^{-9}$	$1,9.10^{-10}$	$8,7.10^{-10}$	$5,5.10^{-10}$
2-3.05	$2,5.10^{-11}$	$4,1.10^{-11}$	$4,5.10^{-11}$	$1,1.10^{-11}$
3-4.05	$3,1.10^{-11}$	$1,1.10^{-11}$	$1,2.10^{-11}$	$5,1.10^{-12}$
4-5.05	$1,6.10^{-11}$	9.10^{-12}	$8,7.10^{-12}$	$1,4.10^{-12}$

KEY: (1) Ci/m

(2) Sample date

Таблица 5.5

Дата отбора проб (1)	Ки/км^2 сутки г.Киев (2)				Ки/км^2 сутки г.Калининград (3)			
1986г.	^{131}I	^{140}Ba	^{95}Zr	^{137}Cs	^{131}I	^{140}Ba	^{95}Zr	^{137}Cs
26-27.04					$2,0.10^{-2}$	$3,8.10^{-3}$	$3,8.10^{-3}$	$2,3.10^{-3}$
27-28.04					4,6	$6,3.10^{-2}$	$2,7.10^{-2}$	$1,0.10^{-1}$
28-29.04	$1,9.10^{-2}$	$1,9.10^{-3}$	$1,3.10^{-3}$	$1,1.10^{-3}$	4,9	$5,1.10^{-2}$	$3,8.10^{-2}$	$8,4.10^{-2}$
29-30.04	$6,6.10^{-2}$	$6,4.10^{-3}$	$2,3.10^{-3}$	$7,8.10^{-3}$	10,5	$1,3.10^{-2}$	$1,6.10^{-2}$	$1,2.10^{-2}$
30.04-01.05	2,5	$7,1.10^{-1}$	$8,3.10^{-1}$	$9,6.10^{-2}$	2.10^{-1}	$1,1.10^{-2}$	$4,3.10^{-3}$	$4,8.10^{-3}$
1-2.05	10,3	3,2	$6,0.10^{-1}$	$3,2.10^{-1}$	$1,6.10^{-1}$	-	2.10^{-2}	$3,2.10^{-3}$
2-3.05	2,6	$8,6.10^{-1}$	$5,7.10^{-1}$	$9,2.10^{-2}$	4.10^{-2}	$1,3.10^{-2}$	$4,3.10^{-3}$	$9,2.10^{-4}$
3-4.05	$3,3.10^{-1}$	$2,2.10^{-1}$	$2,5.10^{-1}$	$1,1.10^{-2}$	$2,4.10^{-2}$	$7,3.10^{-3}$	$7,5.10^{-3}$	1.10^{-3}
4-5.05	$7,8.10^{-1}$	$1,0.10^{-1}$	$2,2.10^{-1}$	$2,3.10^{-2}$	$1,6.10^{-2}$	$3,8.10^{-3}$	$1,0.10^{-2}$	$7,3.10^{+4}$

KEY: (1) Sample date

(2) Ci/km days Kiev

(3) Ci/km days Kaliningrad

Таблица 5.6

Водоем и дата отбора проб (1)	Радионуклиды (10^{-9} Ки/м^3) (2)					
	^{131}I	^{140}Ba	^{95}Zr	^{89}Sr	^{91}Y	^{3}H
р.Припять, (3) I.05.86	80	25	I8	5,6	I,6	0,6

KEY: (1) Body of water and sample date

(2) Radionuclides (10^{-9} Ci/m^3)

(3) Pripyat River

Таблица 5.7

Водоем (1)	Дата наблюдения (2) максимальной концентрации	Концентрация, 10^{-9} Ки /л (3)		
		иод-I3I (8)	барий-I40 (9)	цирконий-95 (10)
р. Припять (4)	2 мая (5)	I20	60	42
Киевское водохранилище (6)	3 мая (7)	28	I7	20

KEY: (1) Body of water

(2) Maximal concentration observation date

(3)Concentration, 10 Ci/1

(4) Pripyat River

(5) 5 May

(6) Kiev Reservoir

(7) 3 May

(8) iodine-131

(9) barium-140

(10) zirconium-95

Concentration (C) of the total of plutonium isotopes in air (Ci/m) at an altitude h= 1.5 m

Таблица 5.8

Концентрация суммы изотопов плутония (С) в воздухе (Ки /м³) на h = I,5 м

Дата отбора (1)	Расстояние от источника (2)	C, Ки /м³ (3)
I9.05.86	I00 км ЮЮВ (4)	$0{,}48.10^{-14}$
20.05.86	72 км ВСВ (5)	$0{,}35.10^{-14}$
-"-	I05 км ССВ (6)	$0{,}75.10^{-15}$
2I.05.86	48 км ССВ (6)	$0{,}65.10^{-14}$
-"-	60 км ССВ (6)	$0{,}39.10^{-14}$
22.05.86	55 км З (7)	$0{,}21.10^{-14}$
-"-	45 км ЗЮЗ (8)	$0{,}85.10^{-14}$
-"-	35 км ЗЮЗ (8)	$0{,}17.10^{-14}$
-"-	45 км ЮЗ (9)	$0{,}70.10^{-15}$

KEY (1) Sample date

(2) Distance from source

(3) C, Ci/m

(4) km SSW

(5) km ENE

(6) km NNE

(7) km W

(8) km WSW

(9) km SW

Density of the surface contamination of the surface layer and individual grasses by the total of the plutonium isotopes

Таблица 5.9

Плотность поверхностного загрязнения поверхностного слоя и отдельно травы суммой изотопов плутония

Дата отбора (1)	Расстояние от источника (2)	$б_т$, Ки/м^2 (3)	$б_п$, Ки/м^2 (4)
20.05.86	I05 км ССВ (5)	$I,3.I0^{-I0}$	$3,6.I0^{-I0}$
2I.05.86	48 км ССВ	$2,3.I0^{-9}$	$5,I.I0^{-9}$
22.05.86	55 км З	$I,6.I0^{-9}$	$3,3.I0^{-9}$

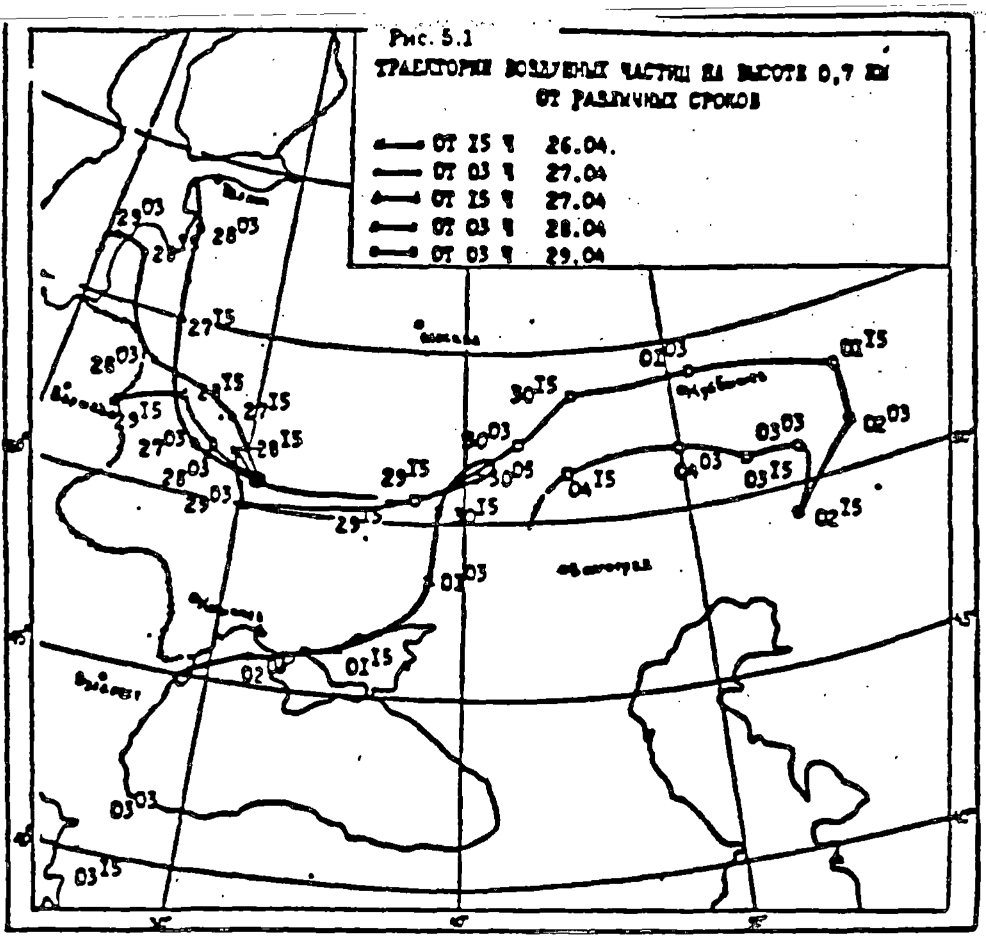
KEY: (1) Sample date

(2) Distance from source

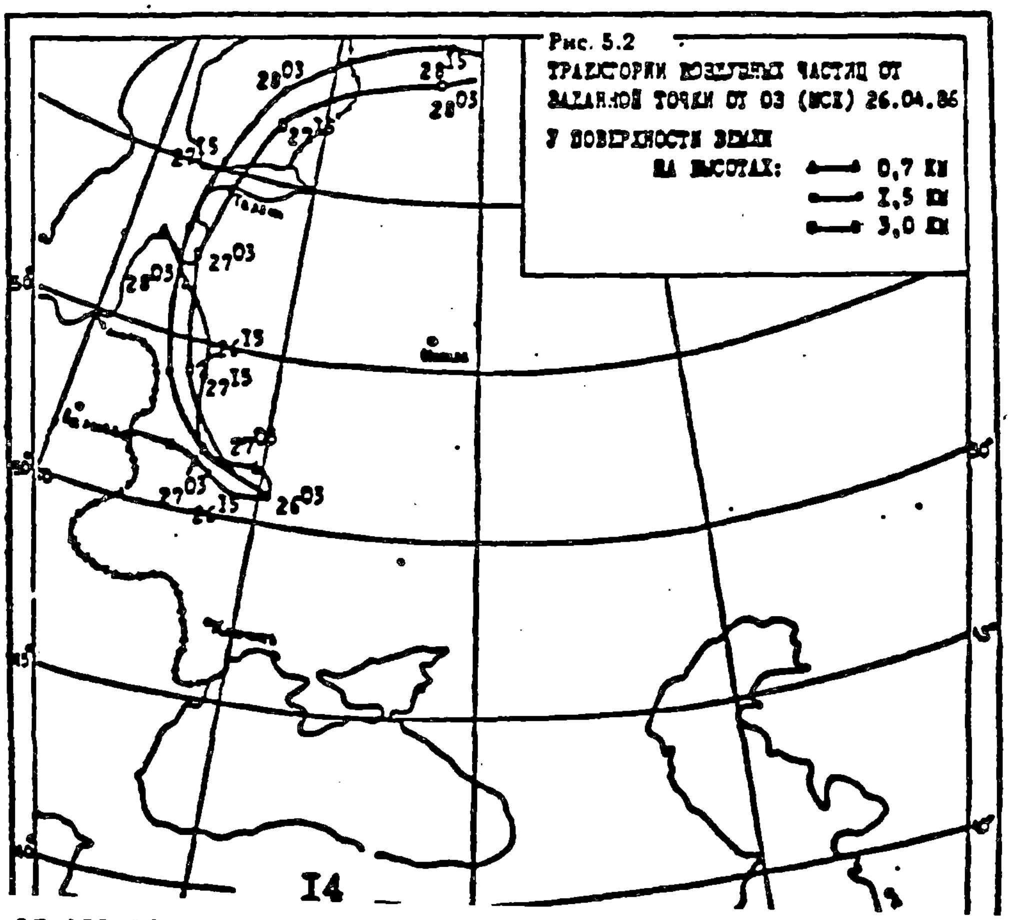
(3) $б_т$, Ci/m^2 (Tr. note: is density of grass contamination)

(4) $б_п$, Ci/m^2 (Tr. note: is density of surface contamination)

(5) km NNE



TRAJECTORIES OF AIR PARTICLES AT AN ALTITUDE OF 0.7 KM FROM DIFFERENT TIMES



TRAJECTORIES OF AIR PARTICLES FROM AN ASSIGNED POINT TO OZ (MOSCOW TIME) 26 APRIL 1986 NEAR THE SURFACE OF THE EARTH AT ALTITUDES:

time) 30 APRIL 1986.

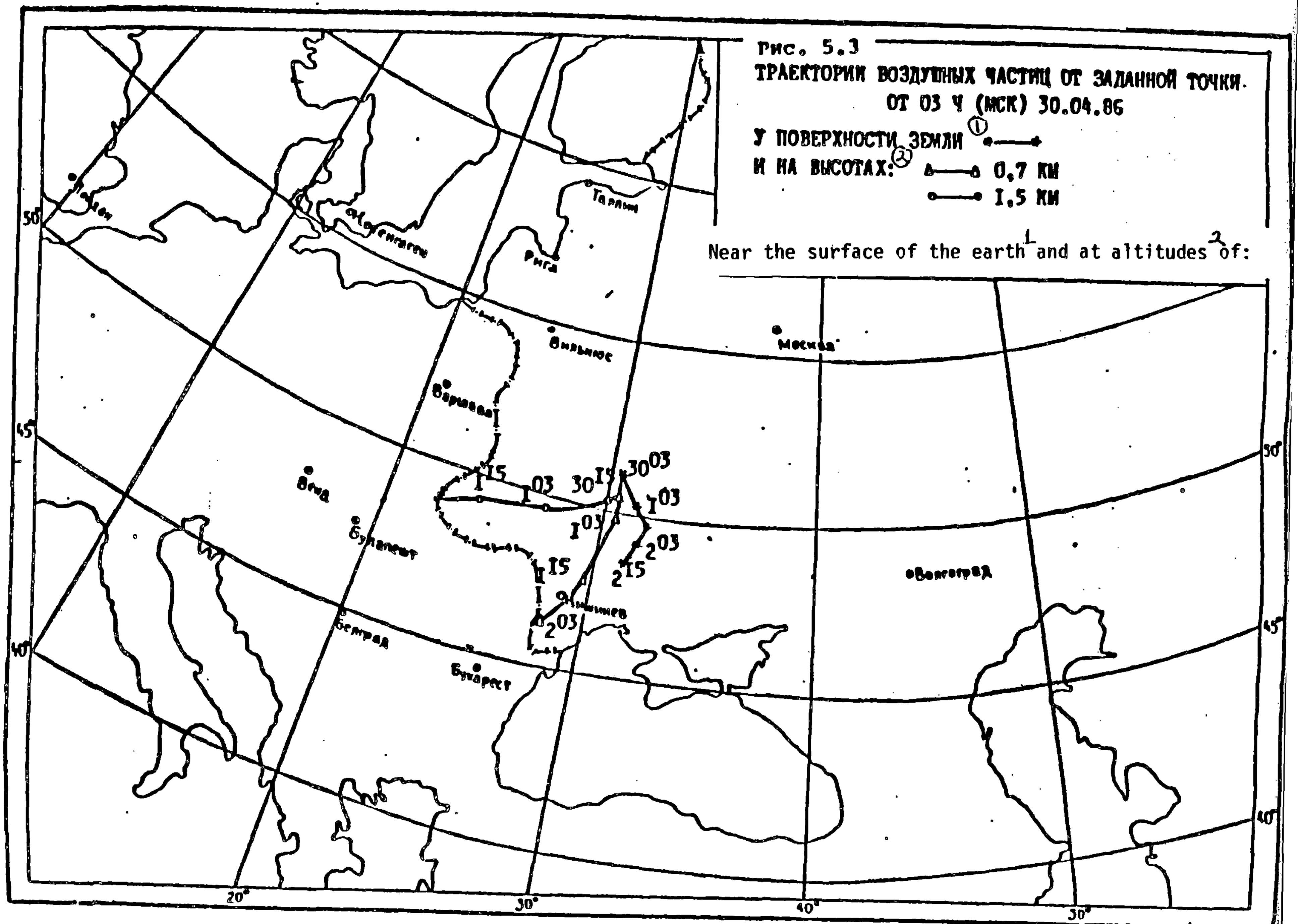


Рис. 5.3
ТРАЕКТОРИИ ВОЗДУШНЫХ ЧАСТИЦ ОТ ЗАДАННОЙ ТОЧКИ.
ОТ 03 Ч (МСК) 30.04.86
У ПОВЕРХНОСТИ ЗЕМЛИ (1)
И НА ВЫСОТАХ: (2) 0,7 КМ
I,5 КМ

Near the surface of the earth 1 and at altitudes 2 of:

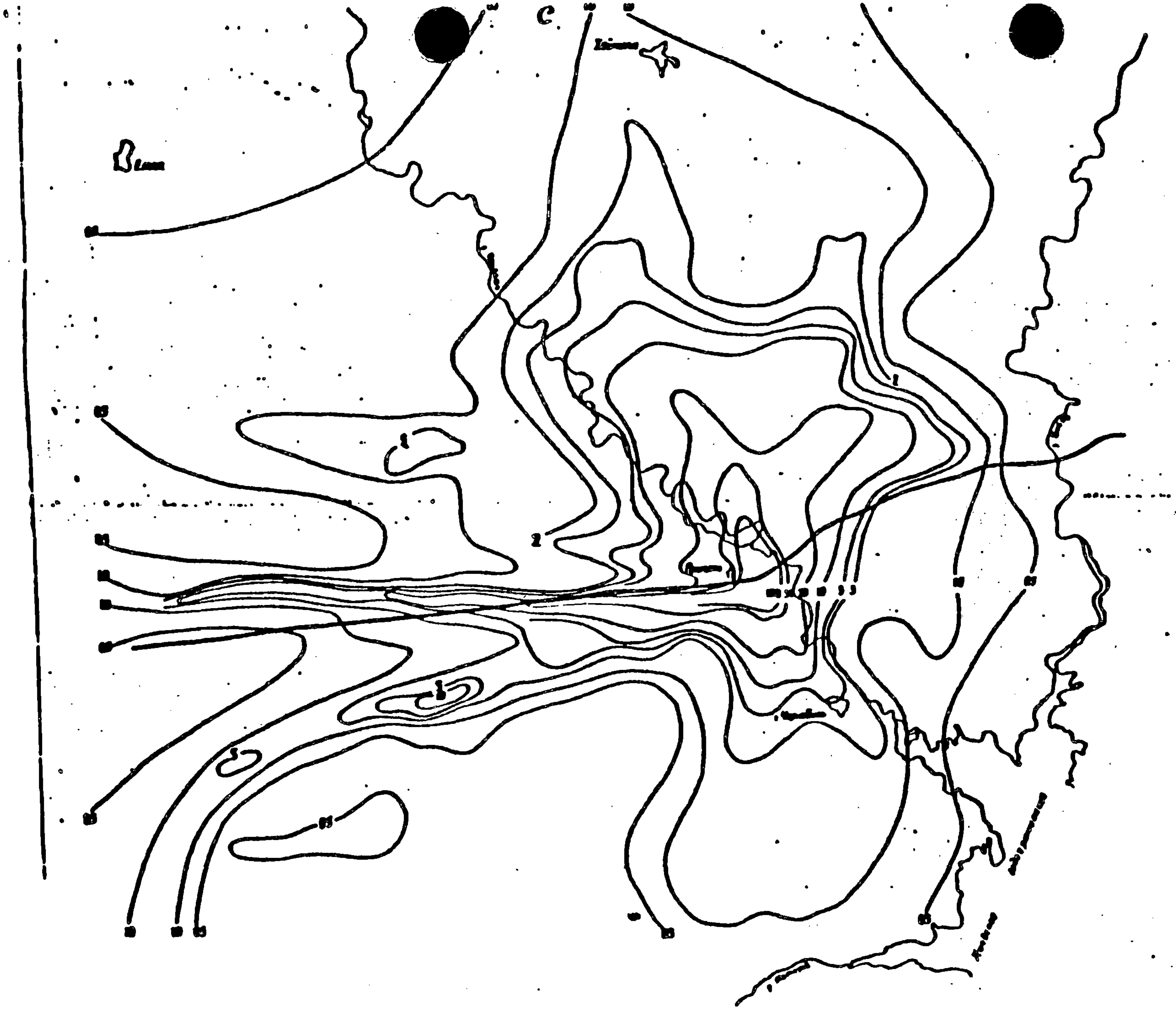


Рис. 5.4. Распределение уровней радиации на местности, мр/ч 29.05.86 г.

Distribution of the radiation levels in the terrain, mr/hr, 29 May 1986.

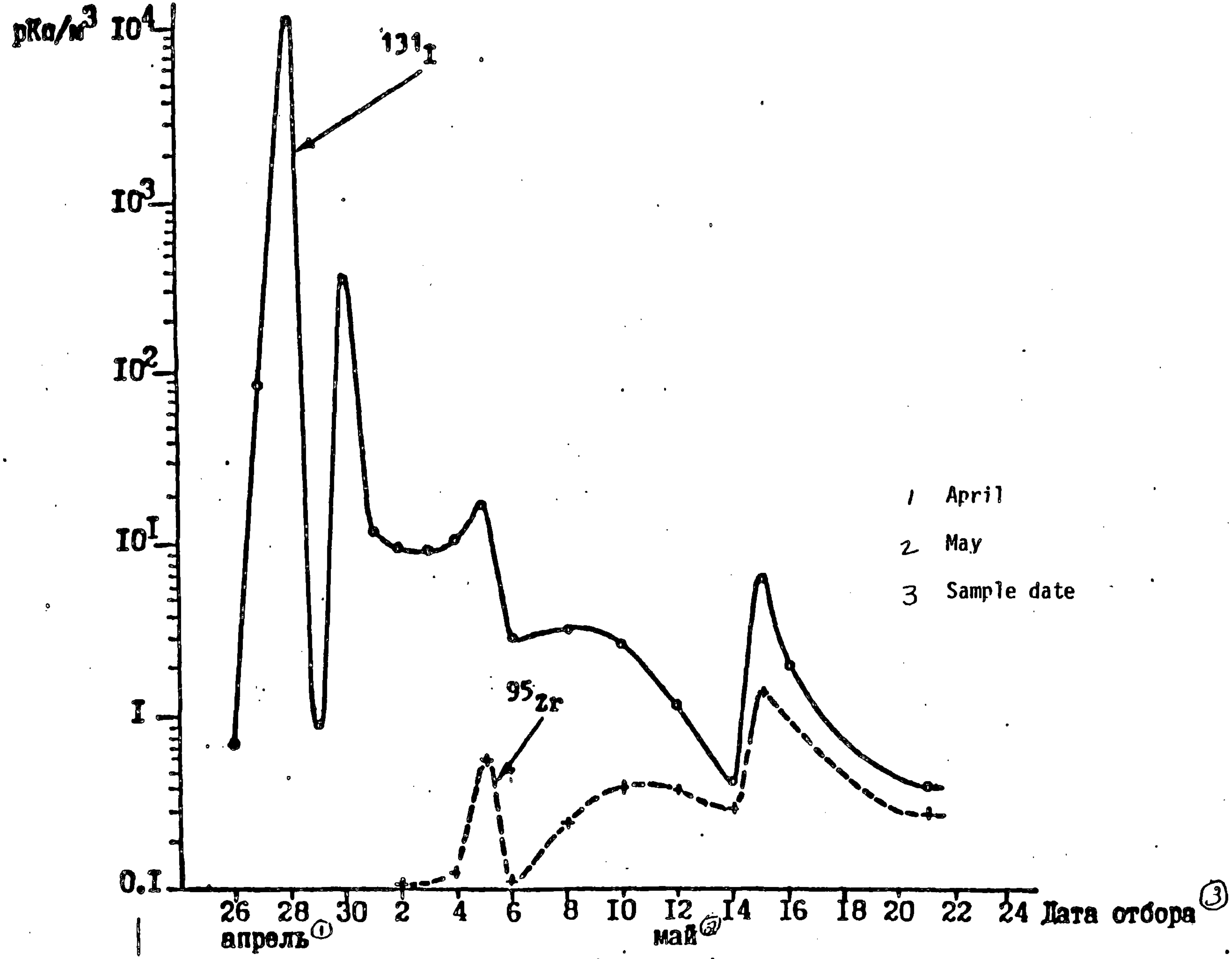


Рис. 5.5. Удельная активность (pKu/м^3) радионуклидов ^{131}I и ^{95}Zr в атмосферном воздухе на станции фонового мониторинга в Березинском БЗ.

Specific activity (*Ci/m^3) of ^{131}I ^{95}Zr radionuclides in the air at the background radiation monitoring station at the Berezina Preserve. * pico-

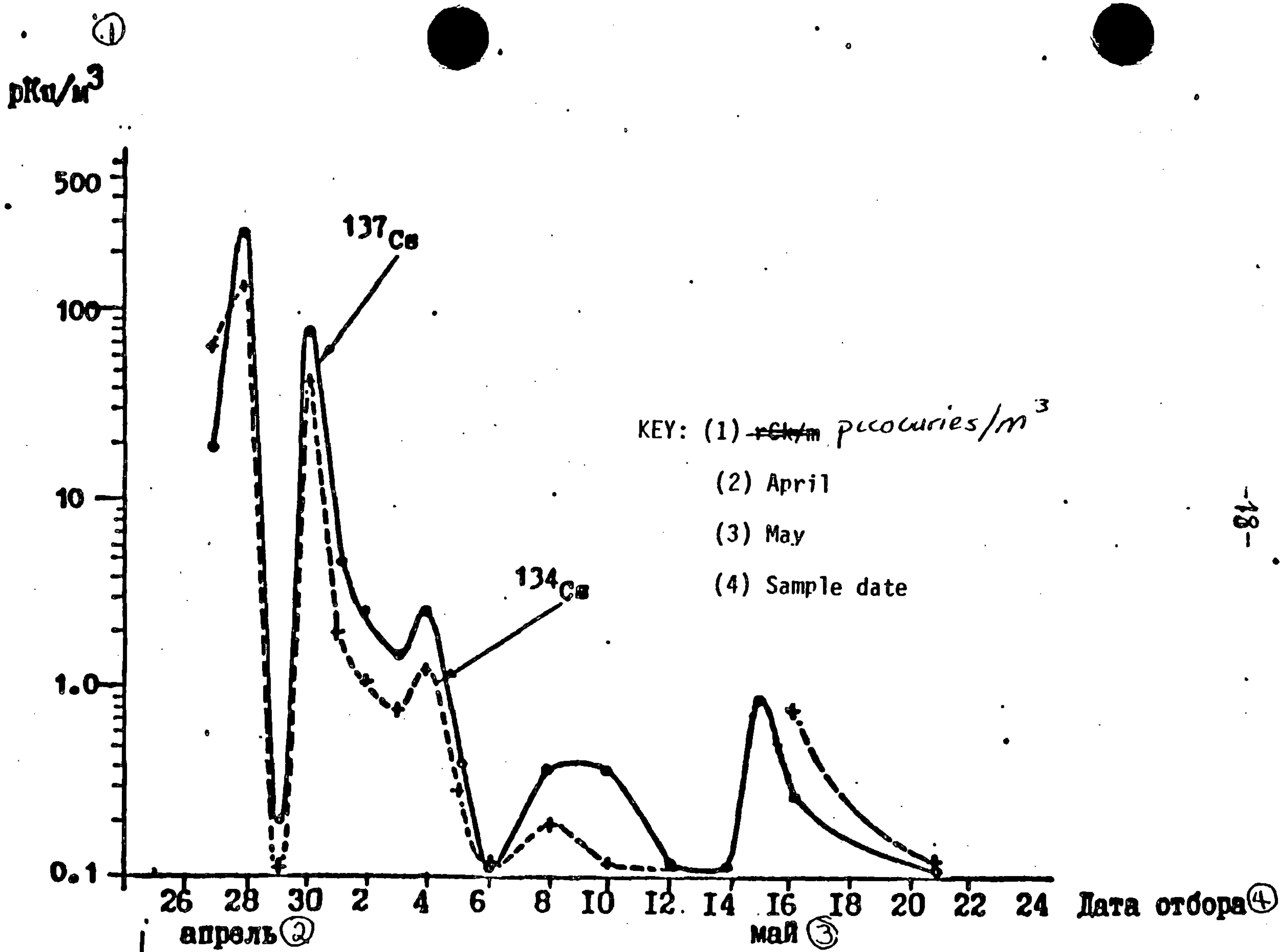


Рис. 5.6. Удельная активность (pKu/м³) радионуклидов ^{137}Cs и ^{134}Cs в атмосферном воздухе на станции фонового мониторинга в Березинском БЗ.

Specific activity (pCi/m³) of ^{137}Cs and ^{134}Cs radionuclides in air at the background radiation monitoring station at the Berezina Preserve.

Appendix 6

EXPERT EVALUATION AND PREDICTION OF THE RADIOECOLOGICAL CONDITIONS OF THE NATURAL ENVIRONMENT IN THE REGION OF THE RADIATION TRACK FROM THE CHERNOBYL AES (THE WATER ECOSYSTEMS)

Appendix 6

6. Expert evaluation and prediction of the radiological conditions of the natural environment in the region of the radiation track of the Chernobyl AES (water ecosystems).

For a number of reasons, the water medium plays a special role in determining the scales and potential consequences of radioactive contamination. Radioactive substances at the site of a water intake as a result of run off enter bodies of water, where redistribution and accumulation of radionuclides occur in such components as the bottom sediments, water vegetation, and in fish. This leads to additional irradiation of both water organisms, and man, who is linked by the food chain with the hydrosphere.

From the very first days of the accident testing was set up of the content of radionuclides in the water and bottom sediments, both directly inside the 30-km zone adjacent to the Chernobyl AES, and beyond it. As a result of the processes of sedimentation, the basic part of the radioactivity entering the water medium rather quickly entered the bottom sediments, whose radionuclide concentration is higher than the activity of the water by a factor of 2-4 (Table 6.1). According to experimental data acquired by the Nuclear Research Institute of the Ukrainian SSR Academy of Sciences (IYaI AN UkrSSR), the Vernadskii Institute of Geochemistry and Analytical Chemistry of the USSR Academy of Sciences (GeoKhI AN SSSR), and the All Union Scientific Research Institute for Atomic Power Plants (VNIIAES), the spatial distribution of radionuclides in the water medium is characterized by substantial nonuniformity. The maximal concentrations of the radionuclides are observed directly in the cooling pond of the Chernobyl AES, where the total activity of the water and the bottom sediments reaches values on the order of 10 Ci/ℓ and 10 Ci/kg, respectively. The concentration of artificial radionuclides in the Kiev reservoir and in the rivers which flow into it is much lower (by a factor of 10^{2}-10^{4}).

The time dynamics of the radionuclides may be broken down into three

Table 6.1

Specific activity (Ci/kg) of bottom sediment samples in June 1986 (10-20 June 1986) (based on data from IYaI AN UkrSSR GeoKhI AN SSSR and VNIIAES)

Место пробоотбора [1]	Zr-95	Nb-95	Ce-141	Ce-144	Ba-140	La-140	Ru-103	Cs-134	Cs-137	I-131
Пруд охладитель ЧАЭС [2]	10^{-3}	$2 \cdot 10^{-3}$	$3 \cdot 10^{-4}$	$3 \cdot 10^{-4}$	10^{-4}	$2 \cdot 10^{-4}$	$6 \cdot 10^{-4}$	$4 \cdot 10^{-5}$	$7 \cdot 10^{-5}$	$1{,}0 \cdot 10^{-}$[illegible]
р.Припять [3]	$8 \cdot 10^{-6}$	$1{,}1 \cdot 10^{-5}$	$2{,}7 \cdot 10^{-6}$	$4 \cdot 10^{-6}$	$5{,}4 \cdot 10^{-7}$	$1{,}3 \cdot 10^{-6}$	$1{,}6 \cdot 10^{-6}$	$2{,}7 \cdot 10^{-7}$	$4{,}7 \cdot 10^{-7}$	$9{,}4 \cdot 10^{-}$[illegible]
р.Днепр [4]	$5{,}3 \cdot 10^{-8}$	$7{,}1 \cdot 10^{-8}$	$2{,}8 \cdot 10^{-8}$	$4{,}1 \cdot 10^{-8}$	$2{,}6 \cdot 10^{-8}$	$4{,}3 \cdot 10^{-8}$	$3{,}6 \cdot 10^{-8}$	$2{,}4 \cdot 10^{-9}$	$5{,}1 \cdot 10^{-9}$	$6{,}3 \cdot 10^{-}$[illegible]

KEY: (1) Sampling point

(2) Cooling pond of the Chernobyl AES

(3) Pripyat River

(4) Dnepr River

characteristic stages. In the first stage (through May, 1986) the level of radioactive contamination was basically determined by short-lived radionuclides, primarily iodine-131. In mid-May the concentration of iodine-131 in the drinking water was $n \cdot 10^{-9}$ Ci/ℓ which was somewhat higher than the safe permissible concentration in accordance with NRB-76 (NRB is Radiation Safety Standard) (2-10 times that in the Pripyat River). However, by early June the iodine-131 content in the river water was reduced by more than or order of magnitude (approximately $n \cdot 10^{-10}$ Ci/ℓ).

In the second stage, as the iodine-131 decays, the comparitively short-lived radionuclides, such as strontium-89, zirconium-95, niobium-95, cesium-141, ruthenium-103, barium-140, and lanthanum-140, begin to noticeably contribute to the formation of artificial radioactivity. The concentration of these radionuclides in the second ten days of June was $\sim 10^{-10}$ Ci/ℓ in the water and 10^{-7}-10^{-8} Ci/kg in the bottom sediments of the Kiev Reservoir and 10^{-9} Ci/ℓ in the water and 10^{-7} -10^{-5} Ci/kg in the bottom sediments of the Pripyat River.

In the third stage after decay of the iodine-131 and other relatively short-lived radionuclides, the basic significance in the formation of artificial radioactivity will be taken over by the long-lived radionuclides, such as cesium-137, cesium-134, and strontium-90. In the second ten days of June the concentration of cesium-137 in the water of the Dnepr River was 10^{-10} Ci/ℓ and was 10^{-9} Ci/kg in the bottom sediments. The concentration of cesium-137 in the cooling pond of the Chernobyl AES and the Pripyat River was noticeably higher (by a factor of $10^{4}-10^{3}$). The concentrations of strontium-90 in the river waters were basically changed in a range of $10^{-11}-10^{-10}$ Ci/ℓ. Considering the long half life (approximately 30 years) further reduction in the activity of cesium-137 and strontium-90 in the bodies of water will occur quite slowly.

Based on experimental data about the distribution of radionuclides in the components of the water ecosystem evaluations were performed of the doses of radiation of water organisms with consideration of the geometrical characteristics of the hydrobionts.

Calculations of the radiation doses from gamma radiators were performed with consideration of the disseminated radiation accumulation factor. The following dose components were evaluated: external radiation from the water, bottom sediments and organisms which have accumulated radionuclides and internal radiation from incorporated radionuclides. The results of calculations based on data for June 1986 and a prediction of the rate of the absorbed dose for June 1987 are presented in Figure 6.1 and Table 6.2.

Analysis of the evaluations of additional doses of irradiation of hydrobionts attests to the following:

- Hydrobiata living directly in the cooling pond will endure the greatest doses. The natural level of external radiation is 4.3 rad/hr on the average near the bottom and the level of internal radiation for water plant reaches 10 rad/hr. Bottom dwellers, roe and the young of phytophylic species of fish which multiply and feed in the thicket of water vegetation will suffer the greatest doses. The effects of long-term radioactive radiation, both direct and indirect, may be expected for these organisms.

-Doses of radiation for organisms which populate the rivers of the contaminated zone (the Pripyat River and others) are much lower(by approximately a factor of ~~2~~ 10^2) as compared with cooling pond inhabitants.

- The dose stress for hydrobiota living in the Dnepr River in terms of order of magnitude are similar to the natural background radiation, exceeding it for individual hydrobionts by a factor of 5-10.

- Analysis of the contributions of radionuclides to the total radiation dose shows that the basic dose forming nuclides at the present time are comparatively short-lived elements: zirconium-95, niobium-95, cesium-141, barium-140, lanthanum-140, strontium-89, ruthenium-103 and so on, whose contribution to the total radiation dose for the majority of the components of the water ecosystem exceeds 70-80%. At the present time, the contribution of long-lived cesium isotopes (cesium-137 and cesium-134) to the total dose does not exceed 4-5%. This means that as the short-lived radionuclides decay, the radiation dose of the water organisms will gradually be reduced and a reduction in the dose stresses for the majority of hydrobionts by a power of magnitude may be expected by the 1987 growing season for essentially the entire contaminated territory, including the cooling pond. After this, a relative stabilization of the additional dose stress on water organisms may be expected, since by 1987 it will be basically determined by long-lived radionuclides, such as cesium-137 and strontium-90.

When speaking about potential biological effects of ionizing radiation, it should be remembered that individual groups of living organisms display enormous differences in resistance to the effect of radioactive radiation.

Of the hydrobionts, the most vulnerable link are the fish, the commercial varieties of which are the final link in the accumulation of radionuclides in the food chain from the water ecosystem to man. It is common knowledge, that the radiation dose rate for fresh-water fish in natural conditions varies in a range of 0.007-0.023 mrad/hr.

Experimental investigations to evaluate the effect of low doses on fish showed that a dose rate below 0.4 mrad/hr (4 rad/year) causes no negative consequences for the vital activity of the fish.

In a range of up to 40 mrad/hr (365 rad/year) there are a variety of disruptions to the function of the organs, but on the whole, the radioecological resistance is preserved at the population and organism level. A further increase in the threshold of irradiation to above 140 mrad/hr (3.5 rad/day) may have a negative impact on the population level and cause a decline in individual more radioactive-sensitive species.

As initial predictive evaluation show, the radiation doses of a majority of water organisms in the Kiev Reservoir (0.1-1.0 mrad/hr)do not go beyond the limit of doses at which radiation damage to the populations occurs. The radiation doses of fish in the Pripyat River (buoy #204) are approximately 50 mrad/hr, i.e., a negative effect of radiation on the hemogenic, immune, and reproductive systems is possible. Of the listed effects, the most significant will be the genetic effects - the negative effect on the reproductive cells.

The coolant pond of the Chernobyl AES had the highest doses of radiation of the hydrobionta (up to 5 rad/hr in a number of sectors) which will lead to a noticeable radiation effect on the water ecosystem, primarily on the fish society.

The proposed evaluation of the radioecological conditions of the bodies of water is preliminary and must be further refined with consideration of the following information:

- the dynamics of entry of the radionuclides into the bodies of water from the surface, especially in the periods of the autumn rains and the spring floods;

-the time dynamics and spatial distribution of radionuclides in hydrobionta, especially in commercial species of fish;

- refinement of the species make-up, the nature of migrations, the spectra of feeding, and the ecological and physiological parameters of the water organisms;

Table 6.2

Calculated evaluations and prediction of doses of irradiation of water organisms (mrad/hr) in the region of the radiation track of the Chernobyl AES (10-20 June 1986)

Водоем 1	Внешнее облучение 2		Внутреннее облучение 3		
	от воды 4	от донных отложений 5	водные растения 6	планктон 7	рыба 8
Пруд-охладитель ЧАЭС 9	I0 (2) *	4300 (300)	I0000 (I000)	I000 (I00)	500 (5C)
Река Припять (буй 204) 10	0,I (0,009)	40 (3,3)	II0 (I5)	I2 (2)	6 (0,8)
Река Днепр (Киев, ИГБ АН УССР) 11	0,002 (0,0002)	0,3 (0,025)	I,0 (0,2)	0,I (0,0I5)	0,04 (0,0I)
Естественный радиационный фон 12	0,000I-0,006	0,002-0,02	0,08-0,2	0,002-0,0I6	0,003-0,005

* В скобках - прогноз на июнь 1987 года

KEY: (1) Body of water

(2) External irradiation

(3) Internal irradiation

(4) from the water

(5) from bottom sediments

(6) water plants

(7) plankton

(8) fish

(9) Coolant pond of the Chernobyl AES

(10) Pripyat River (buoy #204)

(11) Dnepr River (Kiev, Institute of Hydrobiology (Tr. note: possible expansion of the acronym "IGB") of the Ukrainian SSR Academy of Sciences)

(12) Natural background radiation

*In parenthesis - prediction for June, 1987.

- physico-chemical forms of existence of radionuclides in components of the water ecosystem; and

- refinement of the basic hydrological parameters and ways to transfer radionuclides in water ecosystems (the coolant pond - Pripyat River - Kiev Reservoir - and the Dnepr River).

In a scientific and practical sense, the following problems are of noticeable radioecological interest:

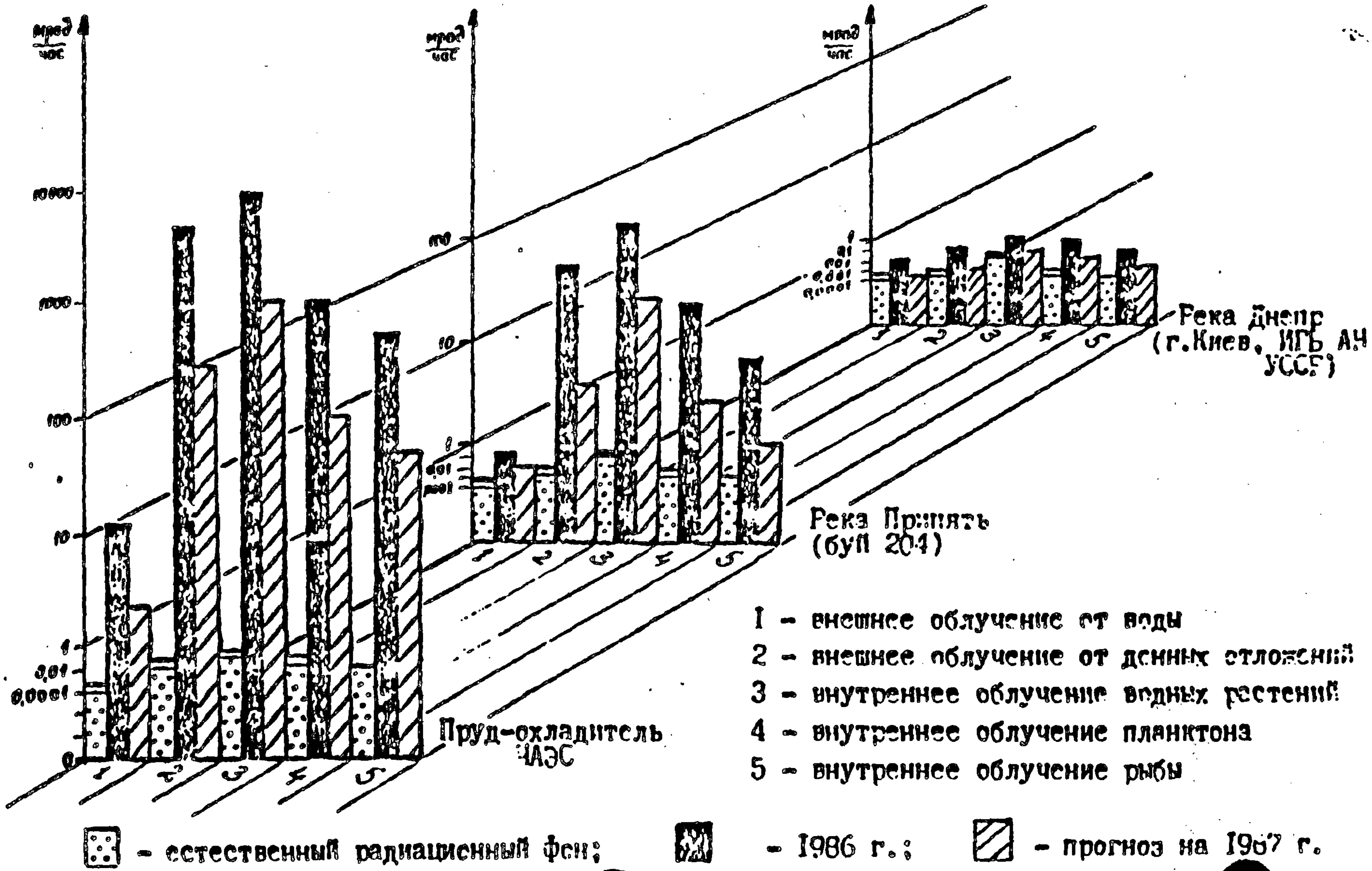
- evaluation of the mechanisms of potential radioecological effects with long-term (chronic) effects of small doses of radiation by artificial radionuclides and

- evaluation of the mechanisms of secondary ecological effects which form as a result of the nonidentical effect of the radiation factor on the ecological characteristics of species. A nonidentical weakening (or strengthening) of the species with radioactive contamination will lead to changes in the interaction between species and, as a result of this, to changes in the structure of the ecosystem. Among examples of secondary effects of radioactive contamination are the increase in the population of harmful organisms in the active sit in the contaminated sectors of the near-water ecosystems, change in the self-cleaning capability of bodies of water, and others.

Organization and performance of long-term comprehensive radioecological investigations both within the 30-km zone and beyond it are required to solve the listed issues.

Figure 6.1

Calculated evaluations and predictions of irradiation doses of water organisms in the radiation track of the Chernobyl AES (10-20 June 1986).



KEY (1) mrad/hr

(2) Dnepr River (Kiev IGM AN Ukr SSR)

(3) Pripyat River (buoy 204)

(4) Coolant pond Chernobyl AES

(5) natural background radiation

(6) prediction for 1987

1. external irradiation from water

2. external irradiation from bottom sediment

3. internal irradiation of water plants

4. internal irradiation of plankton

5. internal irradiation of fish

Appendix 7. Medical and Biological Problems

Table of Contents

7.1 Data on Operating and Emergency Personnel of the AES Subjected to Radiation: Magnitudes of Doses Received and Health Consequences. Treatment Practice.

7.1.1 First Information on the Accident and Actions of Medical Personnel

The medical and sanitation unit servicing the AES, received information concerning the accident at the plant 15 minutes after its occurrence (02 hours 26 April 1986).

Assistance to the first 29 victims, independently evacuating the accident site, was ordered in the first 30-40 minutes by the intermediate medical personnel on duty at the public health center. The victims discarded contaminated clothing and shoes before entering the public health center. In connection with the expressed primary reaction, they were quickly sent to the hospital when sanitation processing and the first medical examination were conducted.

Over the next 4 hours first aid teams administered assistance to the victims: they were taken from the area of the plant site, primary sanitary processing was conducted at the decontamination center and victims with a primary reaction (nausea, vomiting) were transported to the hospital. Persons in satisfactory condition were sent home, and on the next morning 26 April 1986 were actively called for examination. At 06 hours on the morning of 26 April, 108 persons were hospitalized, and over the course of the day--another 24 of those examined.

One victim at 06 hours on the morning of 26 April 1986 died from severe burns and one person from the number of personnel on duty at the time was not discovered. It is possible that his work station was located in the area of the collapse and high activity.

Twelve hours after the accident, a specialized team arrived and was sent to work, comprising physicists, radiology therapeutists and hematology lab workers. Thirty-six hours after the accident, they examined at the MSCh base in the hospital and ambulatorily nearly 350 persons and performed approximately 1000 blood analyses (at least 1-3 for each victim over the first 36-48 hours). Ambulatory written charts were completed with indication of the clinical manifestations after the accident, complaints of the victims, number of leukocytes and leukocytaric formula.

Based on the criteria of early diagnosis adopted in the USSR: times and expression of primary general and local (skin) reaction, expression of lymphopenia and neutrophils leukocytosis over the first 36 hours, persons sent for emergency hospitalization where those for which the development of acute radiation sickness (OLB) with the greatest probability had been prognosticated. To provide maximum assistance and competent subsequent analysis of the results of examinations, the clinical facilities in Kiev and a specialized hospital in Moscow were chosen as hospitalization.

The specialized hospital in Moscow received 129 patients over the first two days, of which over the first three days, 84 were defined as OLB patients of stages II-IV of severity and 27--stage I OLB which indicates the adequacy of primary classification.

The diagnosis of stage II-IV OLB was established over the first three days. To refine the diagnosis to stage I OLB, a longer observation period was usually necessary (to 1-1.5 months).

In all, there were 203 persons recognized as OLB patients.

No victims of OLB were discovered among the population.

7.1.2 Principles of Diagnostics and Prognostication in the Specialized Hospital

The primary diagnostic and prognostic criteria for determining the conditions for admitting patients and choosing methods of treatment, including indications for bone marrow transplants, decontamination, etc., were determined over the first 3 days of their stay in the hospital.

The criteria for classifying the patients over the first days were clinical and clinical-laboratory based on actual experience and recommendations of other international centers on radiology.

The time and severity of the primary general (vomiting) and local (hyperemia and edema of the skin and mucosa) reactions were demonstrative over the first hours to the first three days. The expression of lymphopenia was evaluated quantitatively according to days of observation, and based on it the average dose of overall uniform radiation was tentatively estimated. The possible radiation dose of bone marrow was determined according to a direct method of counting aberrations in the bone marrow cells.

Over the first 10-14 days, in addition to this, the times of manifestation and degree of thrombocytopenia, dates of revelation and expression of leukopenia and granulocytopenia were established as severity criteria. Quantitative assessment of the dose on the bone marrow was performed with regard to the number of dicentrics in the cultures of peripheral blood lymphocytes stimulated with FGA.

The dynamics of skin change over the period from the first days to two weeks were semiquantitatively evaluated according to accepted clinical parameters.

The totality of these criteria developed by Soviet scientists made it possible to evaluate the prognosis:

- general clinical course of the disease;
- dynamics of the blood chart;
- possible degree of affectation of isolated sections of the skin and mucosa.

The average dose of uniform radiation of bone marrow with gamma radiation or its equivalent could be approximately evaluated to a known degree from individual biological parameters.

The course of the sickness and its possible outcome, being defined on the initial days from indicated prognostic criteria, later satisfactorily coincided with this prognosis in its manifestation.

Four stages according to criteria adopted in USSR were isolated with regard to severity of OLB bone marrow and intestinal syndrome.

Cases of sickness with a brief latent period (to 6-8 s) expressed by an early (in the first 1/2 hour) primary reaction (vomiting, headache, increased body temperature) were designated as extremely severe (Stage IV). The number of lymphocytes for the first 3-6 days was less than 100/microliter. On the 7-9th days, enteritic phenomena were manifest. The number of granulocytes in 7-9 s $\leq$500/microliters, thrombocytes--$\leq$40,000/microliters--from 8-10 s. Profound overall poisoning, fever, affectation of the mouth and salivary glands were noted. The sickness included such affectations for 20 persons of the number of those treated as the specialized hospital.

A dose of more than 6 GR (to 12-16 GR) of general uniform radiation equivalent in biological effect in hemogenesis was determined for 18 victims.

For 17 patients the accident was lethal on the days +10 up to days +50. These persons were burned over 40-90% of the body surface, and for the majority they were extremely severe, practically fatal, even without regard to other clinical OLB clinical syndromes.

For two patients of this group, the radionuclides level in the organism was the highest (see Section 7.1.8). Two more patients with stage IV severity died on days +4 and +10 at the hospital in Kiev from combined thermoradiation damage.

A total of 23 persons was diagnosed as victims of stage III OLB. The approximate dose of overall gamma radiation was 4.2-6.3 GR. Criteria for determining OLB of this degree of severity were the times of development of an expressed reaction of 30 min.--1 hour (vomiting, headache, subfebrile body temperature converting to skin hyperemia). Lymphopenia at 3-6 s, 200-100 cells/microliter. The latent period lasted 8-17 days. Epilation was typical. The number of thrombocytes dropped to $\leq$ 40000/microliter on the 10-16th days, neutrophils $\leq$ 10000/microliter (8-20th days). Fever, infectious complications, bleeding were expressed at the height of sickness. This stage of severity was diagnosed at the specialized hospital in 21 persons, at the hospital in Kiev, in 2 persons. Seven persons died over the first two days to seven weeks. Of them, the number of persons with severe skin damages significantly aggravating their condition and largely predetermining a lethal outcome, was six.

Criteria for diagnosing stage II OLB were: development of a primary reaction over 1-2 hours; lymphopenia at times of 3-6 s of the order of 500-300 cells per microliter. The length of the latent period is up to 15-25 days. Neutrophils decreased in 20-30 s to 1000 cells per microliter, thrombocytes in 17-24 s--to 40000 per microliter. At the height, there were concrete infectious complications and weakly expressed indications of bleeding. Moderate acceleration of SOE--to 25-40 mm/hr.

At the specialized hospital and at the hospitals in Kiev, affectations of this degree of gravity were determined in 53 persons (a level of 2-4 GR equivalent in biologic effect. There were practically no persons with burns significantly aggravating their condition.

The dosage level in patients with stage I acute radiation sickness according to cardiologic data was from 0.8 to 2.1 GR. There were no persons with skin damages considerably aggravating the clinical pattern of the disease. Criteria for diagnosing Stage I OLB were: presence of primary overall reactions at periods after 2 hours from the moment of exposure, absence of a general skin response, latent period length of 30 days, reduced number of lymphocytes on the first days to 600-1000 cells per microliter, leukocytes in 8-9 s--to 4000-3000 per microliter, and at the height of sickness--to 3,500-1,500, thrombocytes to 60,000 to 40,000 per microliter (25-28) days, moderate SOE acceleration. These criteria evaluate the degree of severity of bone marrow syndrome. Data of a systematic clinical and laboratory examination over the course of 1-1/2 months (with regard to the length of the latent period and the presence of data on the frequency of chromosomal aberrations in the blood and bone marrow lymphocytes) were quite considerable for this group of patients.

7.1.3 Extent of Biophysical Investigations and Assessment of the Primary Damaging Factors and Dosage Levels

Dosimetric monitoring was conducted for all entering the reception room by means of accepted Soviet devices recording external emission from the body

(RUP, SRP-68-01, AKTINIYa, TISS and others). This made it possible to assess dosage rate distribution about the body (region of the thyroid gland, chest, back, hands, feet, legs, etc.) and to determine readings for repeated sanitary treatment and decontamination of the skin. The dosage rate recordable by the instrument, was due to incorporation of radionuclides and partially, residual contamination of the persons' skin. The use of smears and washings using moistened tampons and aluminum filters (screens) during measurements make it possible to roughly estimate the contribution of radionuclides incorporated in the body and applied to the skin to emission from the body. The first determinations, examination and questioning of the patients in receiving confirmed that the majority of them, in addition to the effect of extraneous gamma-beta-radiation, had had direct contact with beta-gamma-active nuclides, and in some cases, these nuclides had entered the organism.

Though for the majority there was a combination of two or three of the indicated factors, for the victims, the foremost was external gamma-beta-radiation on the entire body and additionally greater irradiation of the skin relative to weakly penetrating radiation. The curves plotted of the decay of radioactive substances contained in te patients' urine samples already for the first two days after the accident, certified the presence of radionuclides in the victims' organism with half lives of 185-190 hours or about 8 days, most frequently iodine isotopes.

The patients were carefully and repeatedly washed and placed in wards. Potassium iodide was continuously administered already from the first days (0.25 two times a day). Burns and oropharyngeal syndrome observed over these periods was initiated and continued for a major portion of the patients. Special diagnostic procedures of both general clinical and biophysical nature were developed to refine the possible dosage level and nature of exposure.

Instrumentation and methodical provision for biophysical investigations was provided by a complex of procedures and devices.

1. To measure iodine-131 level in the thyroid gland, a scintillation counter (64x64 mm) placed in a lead collimator was used. Emission collimation was done in such a way so as to intercept photo emission from the person's body, and to isolate only that emission resulting from the thyroid gland. Measurements was taken in a narrow hole (about 364 keV--iodine-131 peak). Evaluation of the addition to the reading from iodine-131 incorporated in the blood circulating through the area of the neck was done by measuring the iodine-131 level in the patient's forearm. Calibration was done by means of a spot iodine-131 source situated in the patient's neck phantom.

2. To measure the level of radioactive substances in urine samples, the biosubstrate analyzers BIO-1 and SICh 2.1 were used. The first facility has a high volume scintillation counter, and the second device employes a detector based on a drift-type semiconductive detector. These facilities were used to measure gamma radiation from samples of sectional material. The facilities were calibrated by means of certified gamma radiation sources in geometry approximating the real as closely as possible.

3. To measure the activity incorporated in the organism of persons, the SICh 2.2 devices and a semiconductive detector with local shielding were utilized. The first device employs a high volume scintillation detector, and the second--semiconductive detector based on pure germanium. The facilities were calibrated by means of human body phantoms,

made from standard containers filled with calibrated solutions of various radioactive substances.

4. Multichannel amplitude analyzers (memory size to 8000 channels) were used to assemble and analyze the resulting gamma spectra. The spectrometric circuit of the devices was assembled from high-resolution spectrometers produced by the "Nokiya" (Finland) and R T (USA) firms.

Spectrometric data acquired in all these facilities were recorded on magnetic carriers.

To establish the level of total alpha-activity of transplutonium elements in the victims' exreta, urine samples were studied for 10 of the patients under different conditions of exposure. In all cases, no plutonium was discovered in the urine (sensitivity of the method is 0.2 decays/min. of a 500 ml sample).

For three victims, for which the level of alpha-active radionuclides in the urine upon entering the hospital (28 April 1986) was 2.0, 0.67, 0.1 nCurie per 0.1 ml urine, diagnostic testing was conducted using pentatsin accelerating the release of plutonium from the organism. No effect from triple administration of the drug was detected in any instance.

In studying the organs of the one fatality (beta and alpha-active radionuclide content upon entering the clinic on 27 April 1986, was 1.5 nCurie per 0.1 ml urine), the total alpha-activity of transplutonium elements was detected only in the lungs in the amount less than or equal to

3.4 nCurie/organ and trace quantities--in the urinary bladder. The analysis of bone tissue is not completed.

Alpha-spectrometry of lung samples detected approximately 90% curium-242 and approximately 10% plutonium and americium. At such plutonium content level (combined with transplutonium elements) and low release constant of it from the organism, the level of nuclide release with the urine is below the sensitivity of the determination procedure being used.

Simultaneously with primary dosimetric monitoring by means of gamma-radiometers, to assess levels of radioactive contamination, immediately after admission, blood and urine samples were taken for biophysical studies (measurements of the total activity and gamma-spectrometric readings). The studies of comprehensive activity were conducted in a biophysical laboratory by means of precision radiometers, and the isotopic composition in the samples in question was determined by means of a gamma-spectrometer based on a pure germanium semiconductive detector.

One to two days after the victims were admitted to the hospital, the radio iodine level was determined for them in the thyroid using the "Gamma" gamma radiometer (Hungarian manufacture). On subsequent days, these studies were repeated several times (from 4 to 6) to obtain data on the half-lives of radio iodine from the thyroid gland. Collective results of reading of the levels of radio iodine content for the period from 29 April to 6 May 1986 (day +3 to day +10)indicate that for the major portion of the victims (94%), the content of radioactive iodine in the thyroid gland on 29 April 1986 was less than 50 microCurie, and for 6% of those studied, these levels were 2-4 times higher (Fig. 7.1.1 and 7.1.2).

Several days after hospitalization, when the levels of residual superficial contamination were close to background, the major portion of the victims (with the exception of persons in extremely critical condition), was examined with SICh stationary instruments (human emission spectrometers). Peaks of more than 20 different radionuclides were noted in the gamma emission spectra originating from their organism, but the primary radionuclides determining the dosage of victims' internal radiation were 131-I, 132-I, 134-Cs, 137-Cs, 95-Nb, 144-Ce, 103-Ru and 106-Ru. Typical emission spectra are given in Fig. 7.1.3 and 7.1.4.

For all the fatalities, during autopsy, samples were taken of various organs and tissues for subsequent determination of the radionuclide level in them (up to 35 samples per person, including 17 samples for various lung sections, Fig. 7.1.5). Preliminary results of determining the content of individual radionuclides for five victims which died on the 17th to 19th days from acute radiation sickness were obtained. A typical chart of sample analysis in relative units per 1 g tissue is given in Fig. 7.1.6.

The emission spectra resulting from the human body, varied in individual observations.

Analysis of data and assessment of individual doses of radiation of individual organs and tissues continue. Only in one instance with maximum RV level in the organism is the contribution of internal irradiation considered in early clinical manifestation of respiratory and intestinal tract organ damage.

Results of Biosubstrate Investigation for Sodium-24.

The first samples of biosubstrates (urine and blood) were collected 27 April 1986 at 15 hours. They were spectrometered for emitted photon radiation. The resulting photon spectrum was quite complex. However, the line 1274 keV (sodium-22, half-life of 2.6 years) having 99.95% yield and the line 1368 keV and 2754 keV (sodium-24, half-life of 15 hours), having 99.87 and 99.99% quantum yield, respectively, were not detected.

Figure 7.1.1. Results of measuri adio iodine activity on 29 April 1986

Рис.7.I.I. Результаты измерения вности радиоиода 29 апреля 1986 года (щитовидная железа) (Thyroid Gland)

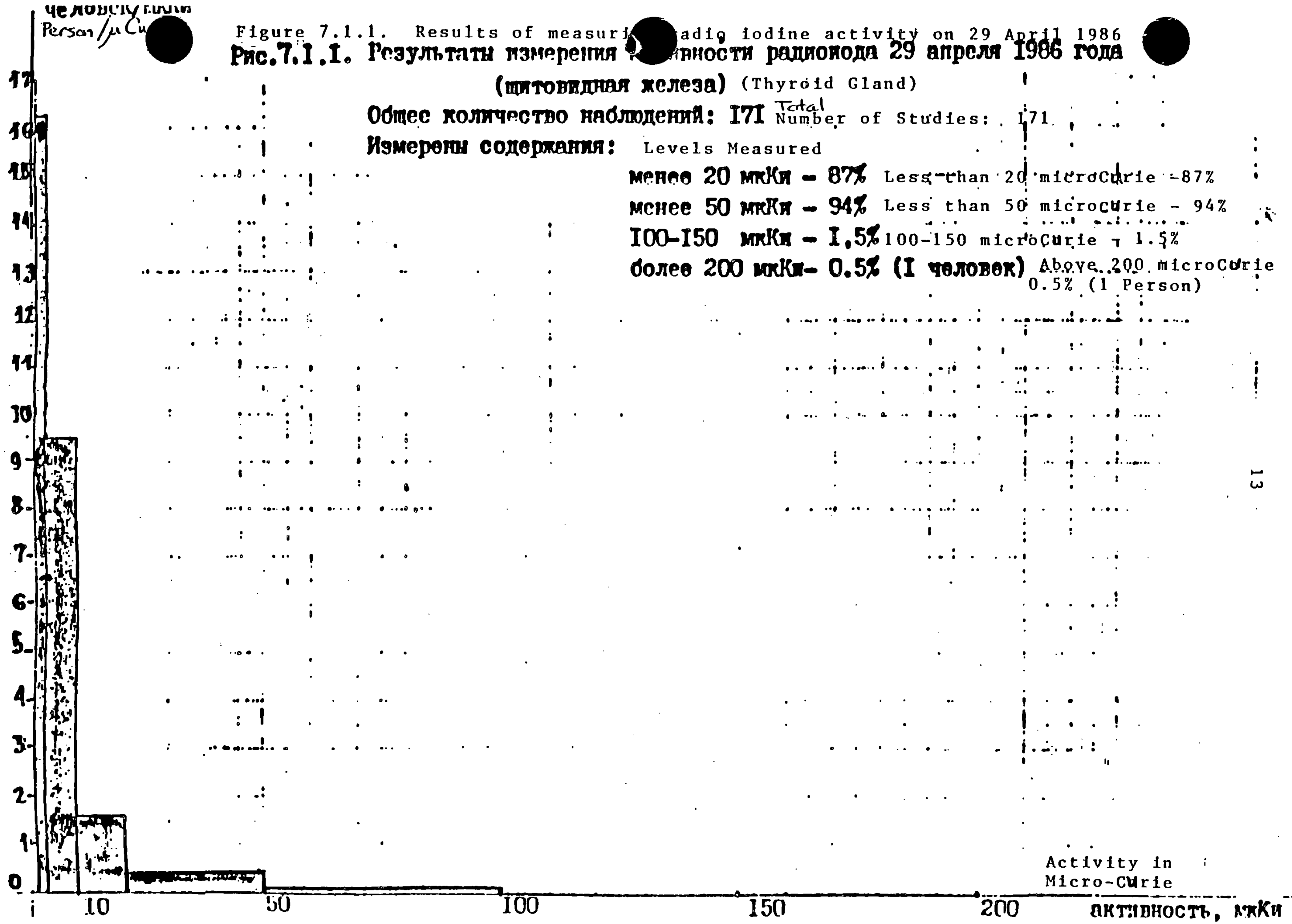


Figure 7.1.2. Results of Measuring radio iodine activity on 6 May 1986

Рис. 7.1.2. Результаты измерения активности радиойода 6 мая 1986 года

(щитовидная железа) (Thyroid Gland)

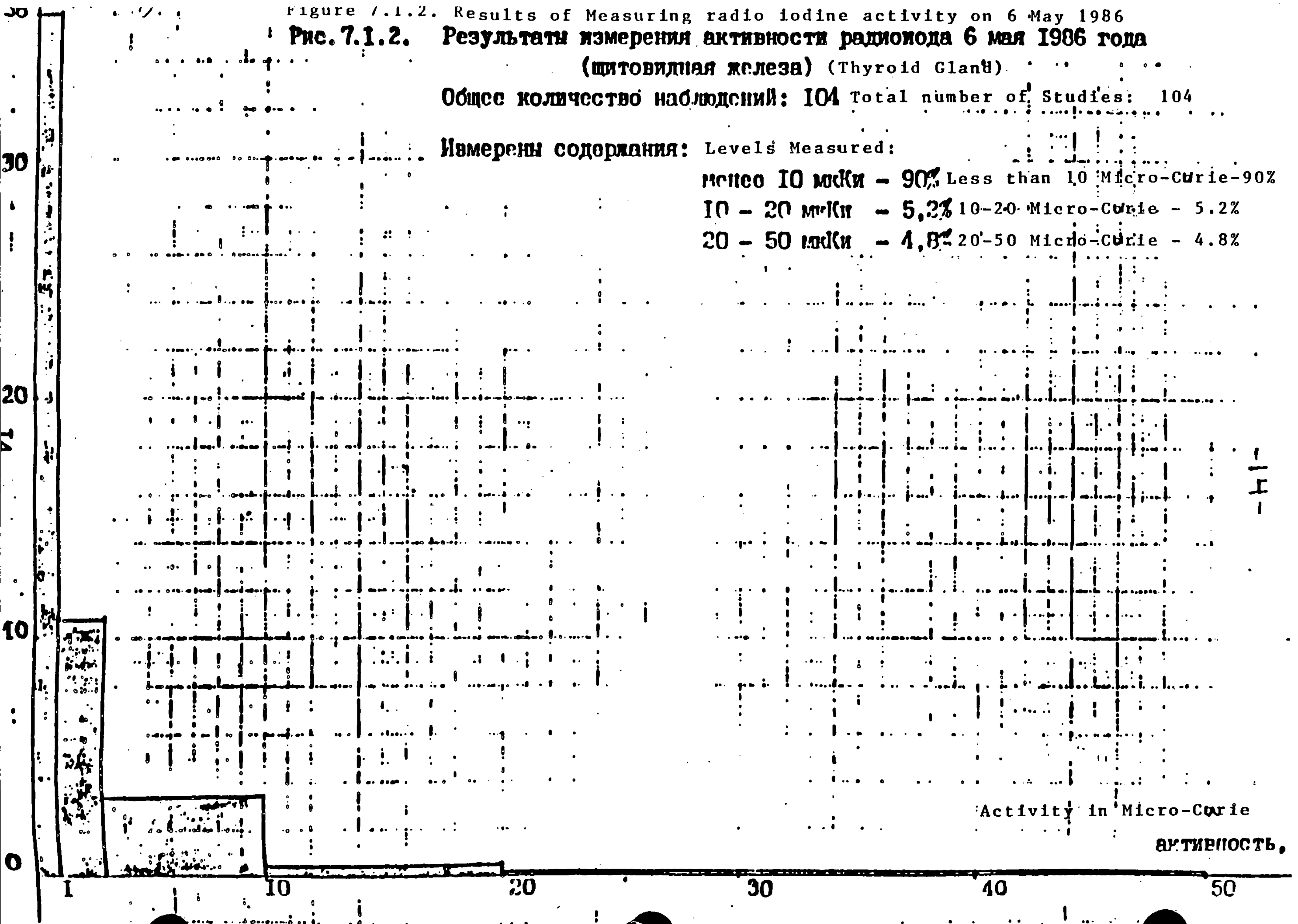
Общее количество наблюдений: 104 Total number of Studies: 104

Измерены содержания: Levels Measured:

менее 10 мкКи – 90% Less than 10 Micro-Curie-90%

10 – 20 мкКи – 5,2% 10-20 Micro-Curie - 5.2%

20 – 50 мкКи – 4,8% 20-50 Micro-Curie - 4.8%



PLOT

Augmented four-fold along Y

1/2 увеличено в 4 раза по Y

Рис. 7.1.3. Спектр фотонного излучения инкорпорированной смеси радионуклидов. Полупроводниковый блок детектирования на основе чистого германия чувствительным объемом 60 см^3. Диапазон энергий более 100 кэВ

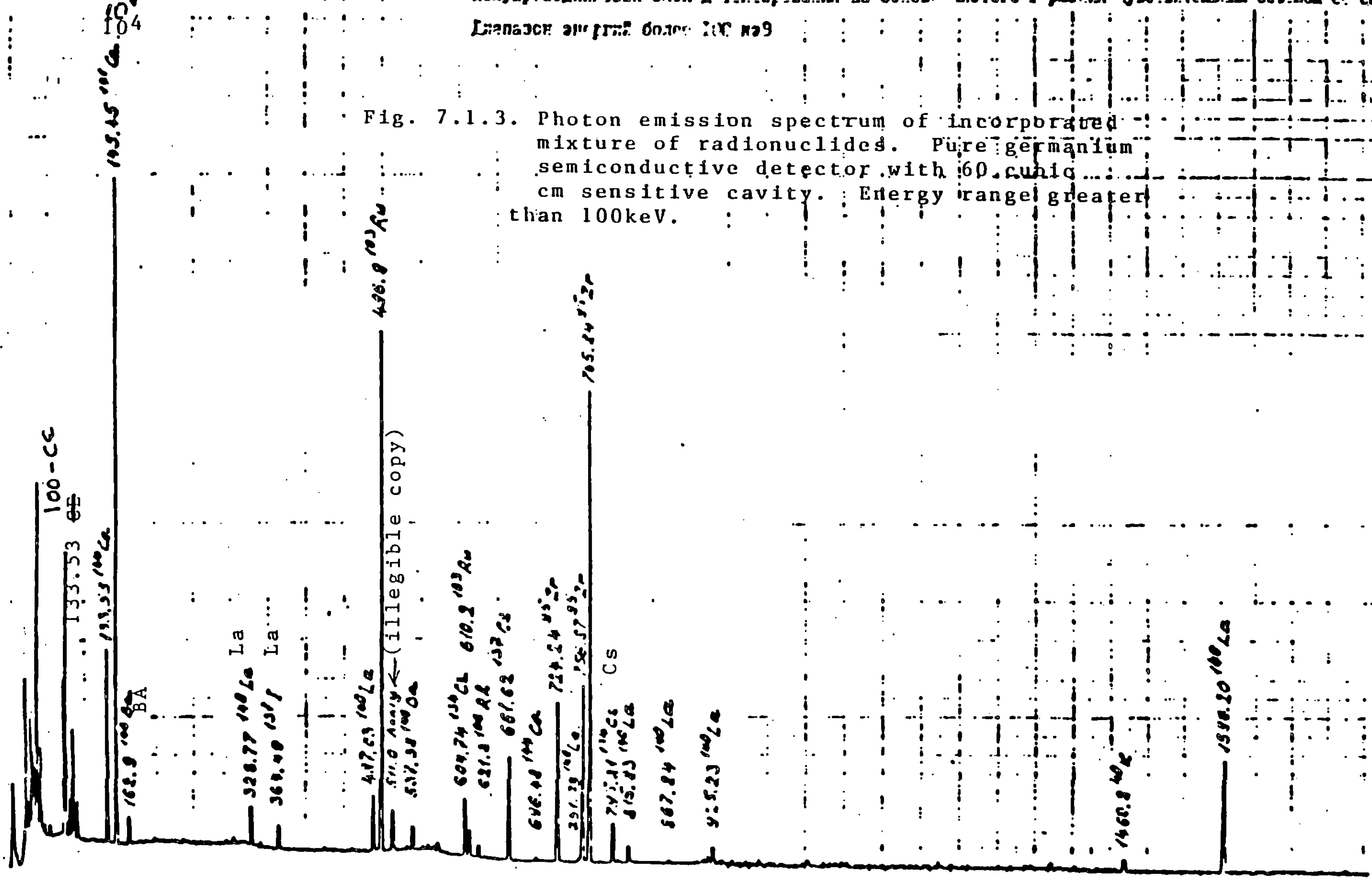


Fig. 7.1.3. Photon emission spectrum of incorporated mixture of radionuclides. Pure germanium semiconductive detector with 60 cubic cm sensitive cavity. Energy range greater than 100keV.

Everywhere in spectrum augmen two-fold along Y

Весь спектр завышен в 2 раза по Y Рис.7.1.1. Спектр фотонного излучения инкорпорированной смеси радионуклидов.

Полупроводниковый блок детектирования на основе чистого германия с чувствительной площадью 260 см^2 Область энергий менее 100 кэВ

Fig. 7.1.1. Photon emission spectrum of incorporated mixture of radionuclides. Pure germanium semiconductive detector with 260 square cm sensitive surface. Energy range less than 100 keV.

16

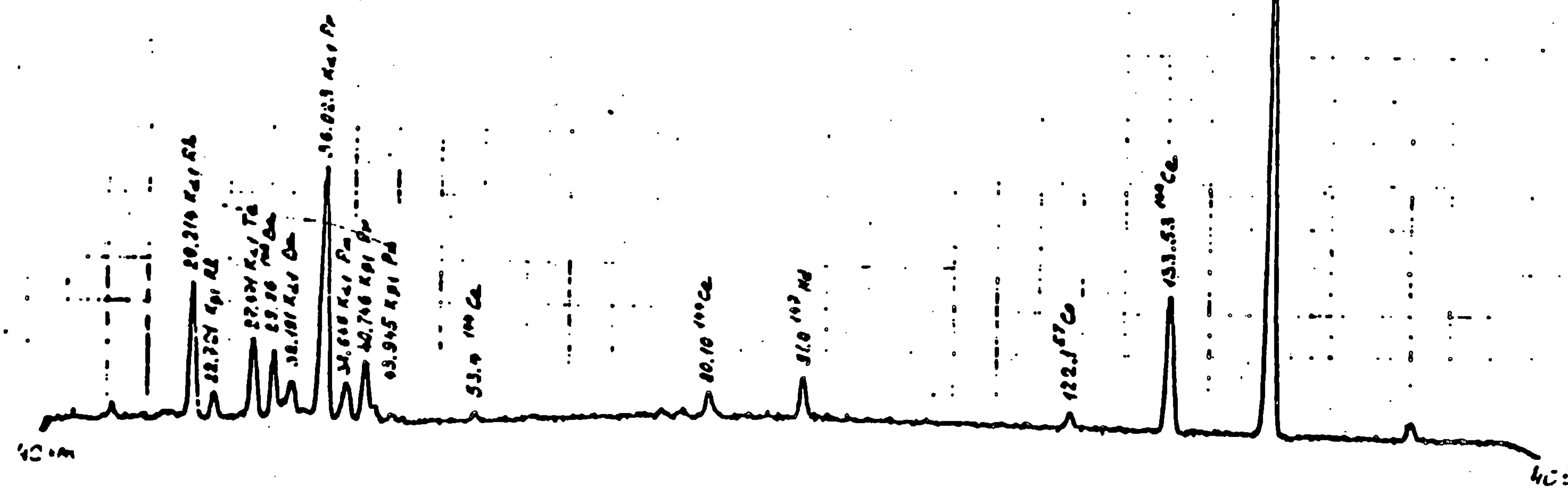


Рис.7.1.5. **СХЕМА ОТБОРА ПРОБ СЕКЦИОННОГО МАТЕРИАЛА ИЗ ЛЕГКИХ**

Fig. 7.1.5. Sampling diagram of sectional lung tissue.

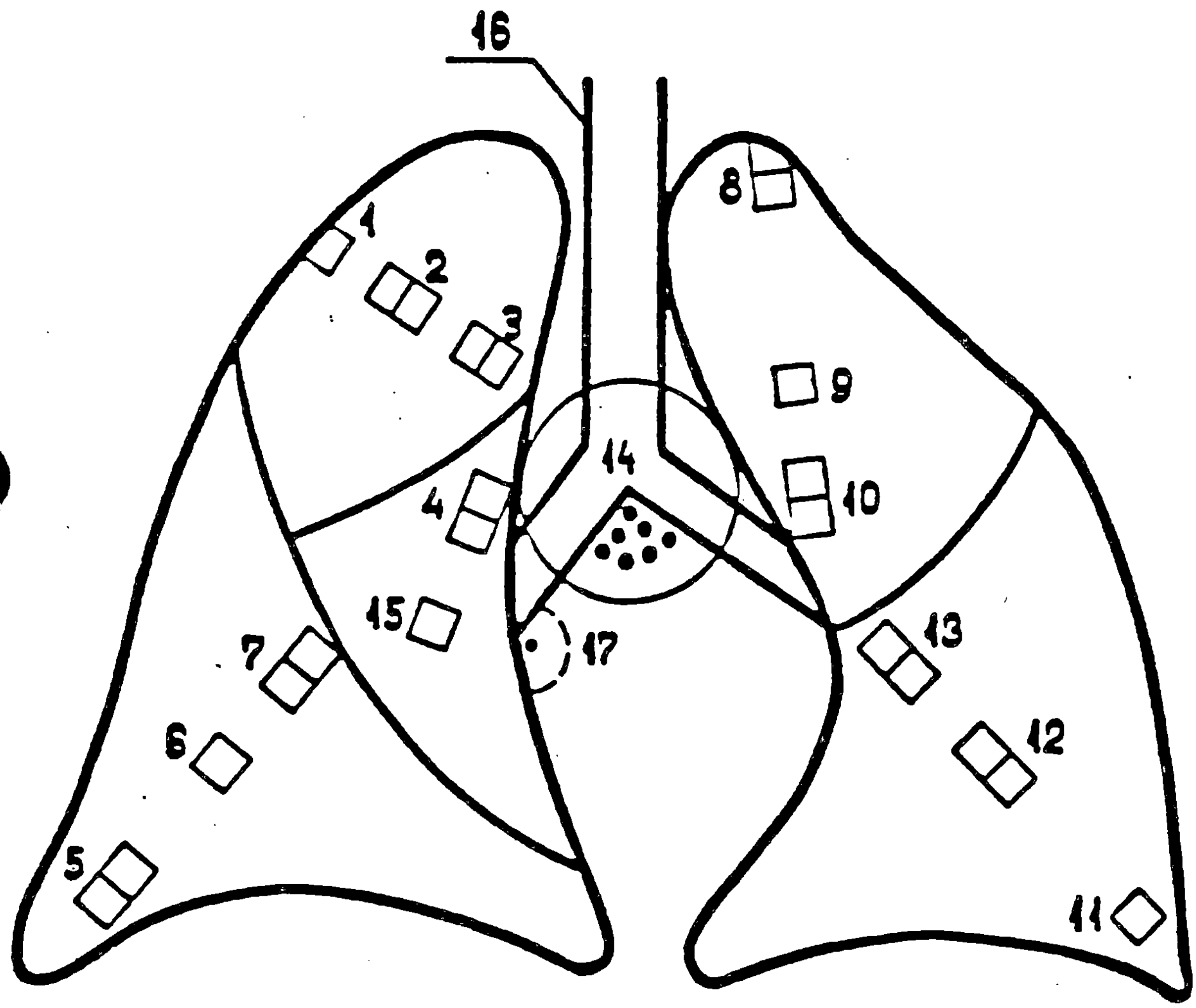
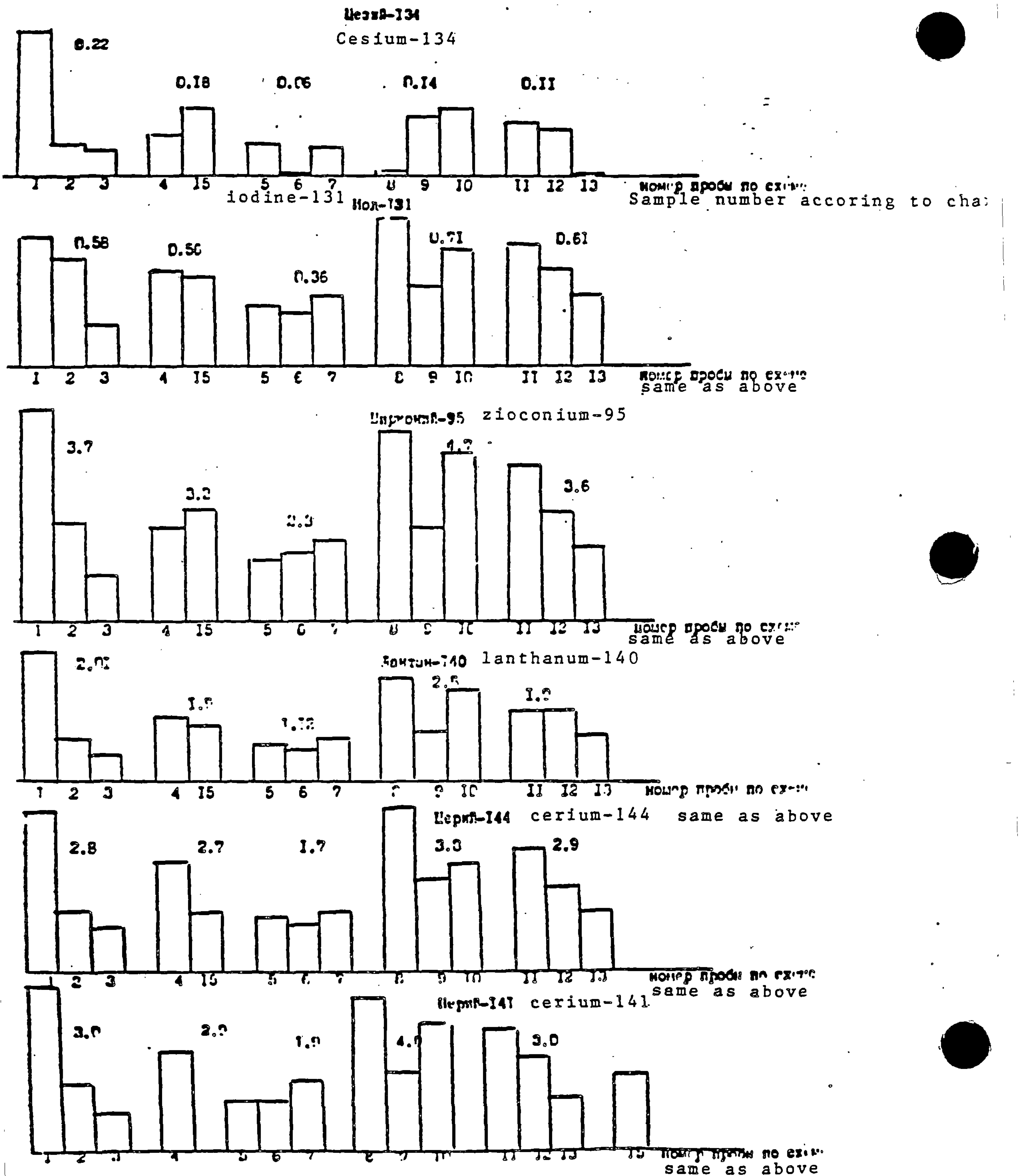


Fig. 7.1.6. Distribution of radionucldes about the victims' lungs (ref. lung sampling diagram) (in relative unites). Mean valves are indicated.

Рис.7.1.6. Распределение радионуклидов по легким пострадавшего (см. схему отбора проб легких) (относительные единицы), указаны средние значения



Measurements were made by means of a semiconductive detector with 60 cubic cm sensitivity cavity. Estimated activity of iodine-131 in these samples was 0.5 microCurie/ml, and activity of the cesium isotopes (cesium-137 and cesium-134) was 0.1 microCurie/ml.

The line 511.0 keV present in the spectrum has insignificant area and corresponds to annihilation emission due to the line 1597 keV (lanthanum-140).

Thus, approximately 35 hours after the accident, there were no significant data certifying neutron irradiation of the victims.

Estimates of the Total Activity of Iodine-131 and Cesium-134, 137 Isotopes Entering the Victims' Organism (Two Victims Having the Highest Radionuclide Level in the Organism)

According to primary results of urine and blood sample analysis, samples were collected certifying that those indicated persons have the highest internal radioactive contamination.

The activity of the urine samples was 0.5 microCurie/ml (iodine-131) and 0.1 microCurie/ml (cesium-134, 137) for one victim and 0.2 microCurie/ml (iodine-131) and 0.07 microCurie/ml (cesium-134, 137) for another victim. According to spectrometric data for urine and blood samples, it may be concluded that these isotopes yield nearly 90% of the absorbed dose of internal irradiation.

The total activity according to preliminary evaluations was nearly 30 mCurie iodine-131 and 10 mCurie cesium isotopes for one victim and 12 mCurie iodine-131 and 4 mCurie cesium isotopes for the second victim.

Preliminary estimates of the dosage loads of internal irradiation of the entire body made for iodine-131 and cesium-134, 137, were nearly 4 Sv (400 REM)--for the first victim and nearly 1.5 Sv (150 REM)--for the second.

Spectrometry of the blood and urine samples, and also direct spectrometry of the entire body and thyroid gland certify that entrance internally of radionuclides for the remaining victims is at a much lower level (tens and hundreds of times less).

These data are preliminary. Spectrometric data is recorded on magnetic carriers and is being processed.

7.1.4 Hematologic and Cardiologic Investigatory Procedures in Assessing the Prognosis of Sickness and the Dosage Level of Overall External Irradiation

The hematology lab examined all persons exposed to ionizing radiation in fault conditions and hospitalized at the specialized clinic. Investigations of the morphologic composition of peripheral blood at the specialized hospital were made daily for 1-1/2 to 2 months (quantity of erithrocytes, leukocytes, reticulocytes, leukocytaric formula, quantity of thrombocytes, hemoglobin content, SOE).

For some, cellular composition of the bone marrow was analyzed once every 7-14 days (or more frequently according to special instructions).

Based on the resulting data, the course of bone marrow syndrome was predicted, which later was well confirmed by actual manifestations of acute radiation sickness for victims, including satisfactory coincidence with preliminary classification by degree of severity and ranges of exposure doses.

Graphs were compiled for each patient, which presented the dynamics of bone marrow syndrome with regard to the change in the number of neutrophils, thrombocytes and lymphocytes.

Cytogenetic analysis was performed for 154 persons. Peripheral blood and bone marrow taken from the victims at various time intervals after exposure (from 1.5 days to 6 weeks) served as studied material. Lymphocytes of peripheral blood and bone marrow were cultivated at 37 degrees C in medium 199 containing antibiotics, FGA and 5-bromodesoxyuridine (10-20 micrograms/ml) for 50-67 hours. Cytogenetic analysis was conducted in 50 first mitosis cells (preliminary results). To identify first mitosis cells, differential dyeing of sister chromatids was used.

The dosage was estimated from the number of dicentrics based on 100 cells. Each tricentric, quadricentric and pentacentric was considered as 2, 3 and 4 dicentrics, respectively. To calculate the dose, the dose-effect curve was used for dicentrics (most accurately the accountable aberrations) derived while studying patients with acute leukosis in remission which sustained therapeutic relatively uniform total gamma radiation in doses of 1.5-5 GR: Y=(10.79+200)XD+(5.16+0.51)XD (subscript 2) where Y--dicentric frequency (per 100 cells) and D--dose (GR). It was indicated that the radiosensitivity of peripheral blood lymphocytes for patients with acute leukosis in remission and of healthy donors after gamma radiation in vitro at a 4 GR is approximately identical.

Irradiation uniformity was evaluated by comparing the observed distribution of dicentrics by cells with the theoretical Poisson distribution. It is common knowledge, that in case of relatively uniform exposure, cell dicentric distribution is subordinate to Poisson's law, and in case of irregular irradiation--deviates considerably from Poisson-type.

Cytogenetic analysis has allowed evaluating the absorbed dose for victims.

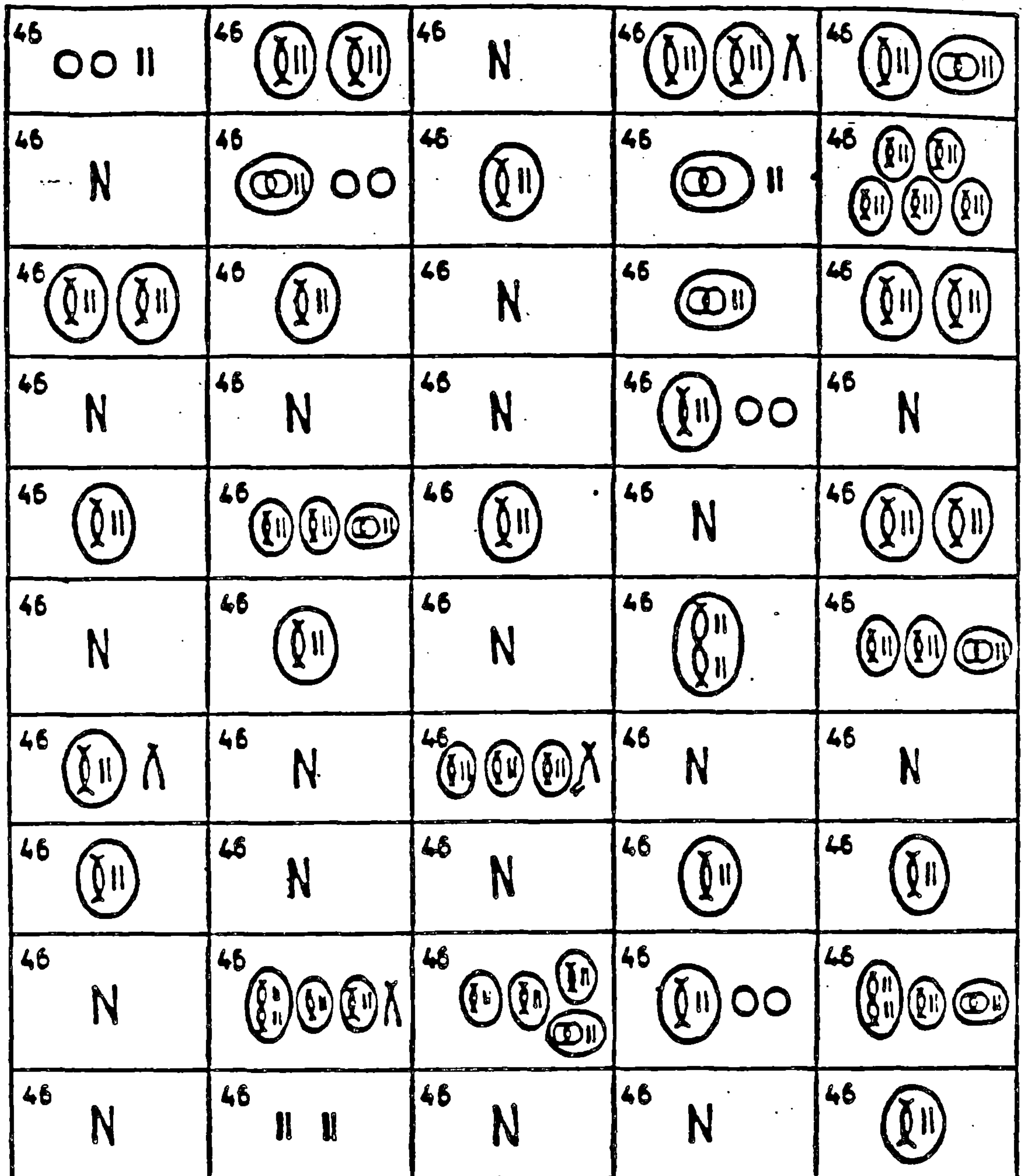
For almost all victims, exposure was relatively uniform: the cell dicentric distribution was subordinate to Poisson's law or deviated little from the theoretical distribution. The severity of bone marrow syndrome was prognosticated from the most informative hematologic index--the number of peripheral blood neutrophils in dynamics (at various time intervals after irradiation). For this, the expected neutrophil curve was plotted for the dose calculated from the number of dicentrics, and it was compared with the real curve observed for each specific individual. Preliminary analysis showed that for uniform exposure, the neutrophil curves for the majority of victims in the cell number reduction phase coincide well with the prognostic curve. Upon irregular irradiation, neutropenia was less profound than at the same level of chromosomal aberrations in the instance of uniform exposure. Figs. 7.1.7 and 7.1.8 present exemplary results of cytogenetic study and the peripheral blood neutrophilic curve for patient D. The prognostic curve is designated as a dotted line, and the real curve--solid line. According to Fig. 7.1.8, both curves quite satisfactorily coincide with regard to time of onset of neutropenia and the degree of its expression.

Fig. 7.1.7 Patient D.

РИС. 7.1.7. Д.

No. analyzed cells - 50 — ЧИСЛО ПРОАНАЛИЗИРОВАННЫХ КЛЕТОК-50

No. aberrant cells - 31 (62%) — ЧИСЛО АБЕРРАНТНЫХ КЛЕТОК-31 (62%)



No. dicentrics - 47 (94 per 100 cells)

ЧИСЛО ДИЦЕНТРИКОВ- 47 (94 НА 100 КЛЕТОК)

ДОЗА ПО ДИЦЕНТРИКАМ - 3,3 Гр

Dose according to dicentrics - 3.3 GR

Т.К. РАСПРЕДЕЛЕНИЕ ДИЦЕНТРИКОВ ПО КЛЕТКАМ СООТВЕТСТВУЕТ ПУАССОНОВСКОМУ, ТО ОБЛУЧЕНИЕ НОСИТ ОТНОСИТЕЛЬНО РАВНОМЕРНЫЙ ХАРАКТЕР.

Since all dicentric distribution corresponds to Poisson-type, exposure has a relatively uniform character

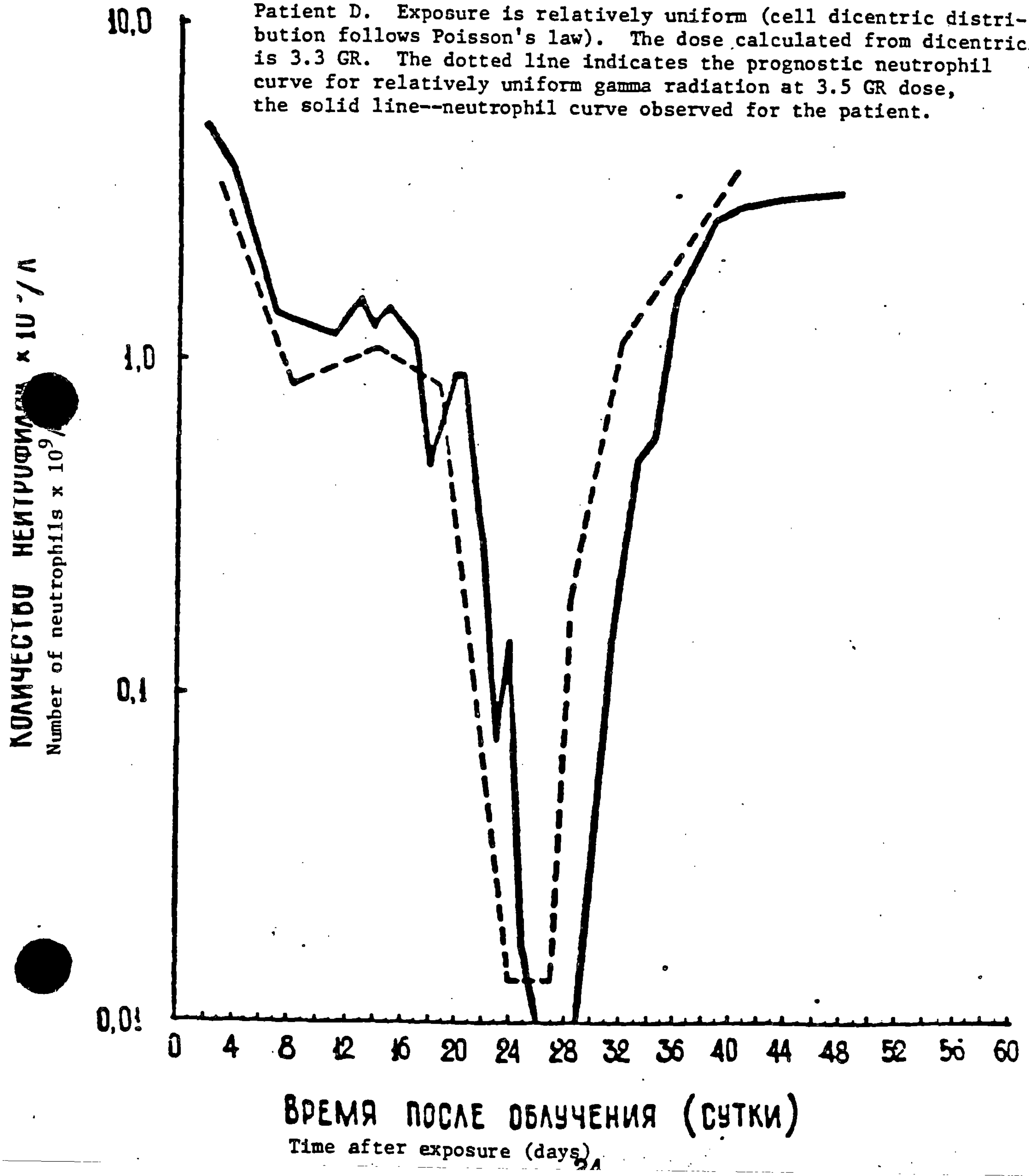
Рис. 7.1.8.

Прогнозирование тяжести костномозгового синдрома по результатам цитогенетического исследования стимулированных ФГА культур лимфоцитов.

Больной Д. Облучение относительно равномерное (распределение дицентриков по клеткам подчиняется закону Пуассона). Доза, рассчитанная по дицентрикам, составляет 3,3 Гр. Пунктиром обозначена прогностическая кривая нейтрофилов для относительно равномерного гамма-облучения в дозе 3,5 Гр, сплошной линией - кривая нейтрофилов, наблюдаемая у больного.

Fig. 7.1.8. Prediction of the severity of bone marrow syndrome from the results of cytogenetic investigation of lymphocytaric cultures stimulated with FGA

Patient D. Exposure is relatively uniform (cell dicentric distribution follows Poisson's law). The dose calculated from dicentrics is 3.3 GR. The dotted line indicates the prognostic neutrophil curve for relatively uniform gamma radiation at 3.5 GR dose, the solid line--neutrophil curve observed for the patient.



Results of cytogenetic investigation were used to select persons needing allogenic bone marrow or embryonal liver cell transplants. For patients on which transplantation was performed, cyotgenetic control of the survival effectiveness of the transplant was carried out. For this purpose, bone narrow punctates and peripheral blood and bone narrow lymphocytaric cultures stimulated with FGA were studied periodically. When cells were transplanted from a donor of the opposite sex, the sex chromosomes were used as markers, and when cells were transplanted from donors of the same sex -- radiation induced marker chromosomes were used (symmetrical interchromosomal exchanges, non-dicentric inversions).

7.1.5. Preliminary Assessment of Certain Biochemical and Immunologic Tests in Accidental Exposure

The list of tests for biochemical observation corresponded to that adopted in the Soviet Union for clinical labs and included nearly 35 parameters characterizing the primary metabolic processes, and 16 tests of the status of the blood coagulation system (Table 7.1.1.).

Results of the determinations were compared with the control and normal values, and also with the dynamics of the index for a given patient (see exemplary record of the dynamics of indices for Patient S. in Table 7.1.2). Immunologic tests and the directions of investigations are given in the composite Table 7.1.3. In addition to the main labs of the specialized hospital, various other institutions around the country were involved. Specialists from the U.S. participated in efforts at typing the antigenic structure of lymphocytes and release of the graft -- haploidentical bone marrow from T-lymphocytes.

Table 7.1.1 Таблица 7.1.1

Перечень и количество тестов, использованных
List and Number of Tests for Patient Studies
при обследовании пострадавших

I. Общий белок - I47I
Overall protein
2. Альбумин - I49I
Albumin
3. Белковые фракции - 266
Protein fractions
4. Мочевина - 750
urea
5. Креатинин - I505
Creatinin
6. Мочевая кислота - 38
uric acid
7. Холестерин общий - I0I4
Overall cholesterol
8. Билирубин общий - 982
Overall bilirubin
9. Билирубин связанный - 982
Bonded bilirubin
I0. Билирубин свободный - 889
Free bilirubin
II. Калий - 889
Potassium
I2. Натрий - 889
Sodium
I3. Кальций - 256
Potassium
I4. Железо - I02
Iron
I5. Фосфор - 5
phosphorus
I6. Алат - I335
Alate
I7. Асат - I335
Asate
I8. а-Амилаза - 673
a-Amylase
I9. Фруктозо-I-монофосфатальдолаза - 5
Fructoso-1-monophosphate aldose
20. КК - 896
KK
2I. Изо-фермент МВ КК - I4
MV KK Iso-enzyme
22. Гамма-ГТП - I22
Gamma-GTP
23. ЛДГ - 96I
LDG
24. Изо-фермент ЛДГ$_{I,2}$ - 68
Iso-enzyme LDG$_{1,2}$
25. Фосфатаза щелочная - 977
Alkaline phosphatase
26. Глюкоза - I850
glucose
27. Проба Реберга - 2
Reberg test
28. Время свертывания крови - I0
Blood coagulation time
29. Время кровотечения - I0
Circulation Time
30. Время рекальцификации - 395
Recalcification time
3I. Аутокоагуляционный тест - 290
Autocoagulation test
32. Протромбиновый индекс - 5I6
Prothrombin index
33. Индекс ретракции - 234
Retraction index
34. Фибриноген - 430
Fibrinogen
35. Фибринолитическая активность - 357
Fibrinolytic activity
36. Паракоагуляционные тесты (этаноловый) - 676
Paracoagulation tests (ethanol)
37. Тромбиновое время - 240
Thrombin time
38. Рептилазное время - 80
Reptylase time
39. Продукты деградации фибриногена - 68
Fibrinogen degredation products
40. Активированное время рекальцификации - I42
Activated recalcification time
4I. Парциальное тромбопластиновое время - 2I0
Partial thrombotastin time
42. Антитромбин-3 - I52
Antithrombin-3

Table 7.1.2

Таблица 7.1.2

Biochemical Index Dynamics Chart

Карта динамики биохимических показателей

~~[illegible]~~

S. V. I.

Фамилия, и., О. С.В.И.

Last name, first, patronymic

Дата поступления 26 апреля I986 г.

Date of admission 26 April 1986

Диагноз: Острая лучевая болезнь

Diagnosis: Acute radiation sickness

Дата исследований Date of Study / Наименование теста, Test	Норма Norm	27.04	28.04	29.04	2.05	7.05	II.05	I3.05	I4.05	I5.05	I6.05	I7.05	I8.05	I9.05	20.05
I		2	3	4	5	6	7	8	9	I0	II	I2	I3	I4	15
Общий белок / Overall protein	62–82 г/л g/l	7I		56	70	79	6I	6I	59	53	56	65	47	58	65
Альбумин / albumin	35–52 г/л g/l			33			3I	26	24	24		27	32	32	28
Алат / Alate	до 40 МЕ to	52	26	20	20	27	22	23	27	I8	I3	I8		II	14
Асат / Asate	до 40 МЕ	57	28	9	I5	26	I6	27	23	I5	I8	32		43	54
КФК / KFK	до I70 мЕ	I600	I43		I70	380	89	89		26				28	
ЛДГ / LDG	до 460 мЕ		282		233	326	266	I60	65					352	
Изо-фермент ЛДГ$_{1,2}$ / Iso-enzyme LDG$_{1,2}$	до 280 мЕ		I33		89	I50	I62							2II	
Изо-фермент ЛДГ$_{3-5}$ / Iso-enzyme LDG$_{3-5}$	70–I40	20400		I60											
Мочевина / Urea	2,5–8,3 ммоль/л mmole/l	4,9					9,8	I0,0	I0,7	I4,5	22,7	23,4	27,7	8,3	27,[illegible]

Continuation Table 7.1.2
Продолжение табл. 7.1.2

	I	2	3	4	5	6	7	8	9	I0	II	I2	I3	I4	I5	[illegible]
Креатинин Creatinin	34–I34 мкмоль/л micromole/l		94	74	I04	I24	I80	I42	I54	200	265	285	348		270	[illegible]
Холестерин общий overall Cholesterol	3,5–6,5 ммоль/л mmole/l		3,7	2,6	4,0	3,6	2,3	I,5	2,2	2,0						
Кальций Calcium	2,0–3,0 ммоль/л mmole/l					I,9		I,7	I,85			I,9				
Глюкоза Glucose	3,2–5,6 ммоль/л	3,9	2,0	2,6	3,85	4,2	4,I	4,6	3,8	4,4	2,8	2,0		2,0	1,8	[illegible]
Билирубин общий overall Bilirubin	4,6–I5,I мкмоль/л micromole/l	23,9		9,6	I0,3			I8,3	I7,I	3I,3	29,8	33,2	4I,0	38,8	50,8	[illegible]
Билирубин связанный Associated Bilirubin	до 5,0 мкмоль/л micromole/l	6,8						I0,8	I3,2	25,8	26,7	3I,3	4,0	30,I	34,5	[illegible]
Калий Potassium	3,5–5,5 ммоль/л mmole/l						5,0	4,0	4,I	3,8		3,2		3,2	2,7	[illegible]
Натрий Sodium	I39–I53 ммоль/л mmole/l						I34	I36	I40	I42		I50		I52	160	[illegible]

Таблица 7.1.3

Основные направления и тесты иммунологического исследования

Группа больных (1)	Группа крови и резус-фактор (2)	Типирование по антигенам сист. H A (3)	Подбор доноров из числа родственников по группам крови и АГ H A (4)	Типирование по эритроцитарным антигенам (5)	Освобождение КМ от Т-Л при гаплоидентичных пересадках (6)	Определение Субпопуляций лимфоцитов (7)	Определение концентрации иммуноглобулинов А,М,G (8)	Активация иммунитета Т-активином (9)	Исследование изосенсибилизации (10)	Контроль за приживлением КМ по эритроцитарным химерам и по H A (11)
Тяжелые больные нуждаются в пересадке костного мозга (12)	Всем без исключения (16)	Всем в первые дни поступления в стационар (17)	Всем обязательно (18)	В процессе решения вопроса о пересадке (20)	При отсутствии совместимого донора по H A (21)	В период восстановления кроветворения I раз в неделю (22)	Предположительно I раз в неделю (24)	Выбор курса лечения по показаниям (25)	У всех больных с переливаниями компонентов крови по мере развития трансфузионных реакций (26)	При пересадке АВО несовм. костного мозга и гаплоидентичного костного мозга (27)
(13) Тяжелые больные, не нуждающиеся в пересадке костного мозга	-"-		По спец. показаниям (19)			По показаниям I раз в неделю (23)				
(14) Больные средней степени тяжести	-"-									
(15) Больные с легкой степенью поражения	-"-									

Key: (1) Patient group; (2) Blood group and rhesus factor; (3) antigen typing according to HLA system; (4) donor selection form relatives according to blood groups and HLA antigens; (5) erithrocytaric antigen typing; (6) KM release from T-L in haploidentical grafts; (7) determination of lymphocyte subpopulations; (8) determination of A, M, C immunoglobulin levels; (9) T-Activin immunity activation; (10) isosensitization study; (11) Monitoring KM adaptation according to erithrocytaric chimera after HA; (12) critical patients needing bone marrow transplant; (13) critical patients not requiring bone marrow transplant; (14) patients of an average degree of severity; (15) patients with slight degree of damage; (16) all without exceptions; (17) all during first days of hospitalization; (18) all compulsorily; (19) according to special indices; (20) when examining the question of a transplant; (21) in the absence of a compatible donor according to HLA; (22) during recovery of hemogenesis once a week; (23) according to indications once a week; (24) presumably once a week; (25) course of treatment chosen according to indications; (26) for all patients with blood component transfusions according to the degree of transfusion reaction development; (27) in case of ABO noncompatible bone marrow and haploidentical bone marrow.

Biochemical examination of critical patients and patients with average degree of severity was done daily. Stage I and II patients were examined twice a week according to this same scheme.

The number of analyses performed during observation of one critical patient, was nearly 800, and for a stage I patient -- on the order of 200 analyses.

The data are in the process of being analyzed. Brief preliminary information is given below.

Biochemical investigations of the critical victims 36-48 hours after exposure revealed expressed hyperamylasemia and hyperamylasuria. The multiplicity of exceeding the norm corresponded to damage severity and reached 10-100-fold overstatement of the norm in the most affected group.

For patients with extensive radiant burns, high creatinine kinase levels were determined a day after exposure. Enzyme activity in isolated cases exceeded the norm by a factor of 10-20. Parallel with these changes, a moderate increase in aspartate amino transferase was revealed.

During the observation period (especially at the height of sickness) quite expressed changes in the protein spectrum were noted; hypoproteinemia and hypoalbuminemia.

Seven to 10 days after exposure, for many there were considerable shifts in a range of biochemical tests and enzymes certifying the disturbance of renal function; hyperfermenemia (aspartate and alaninaminotransferase, alkaline phosphatase, lactate dehydrogenase), and also hyperbilirubinemia with manifestation of a fraction of direct bilirubin.

Disturbances in kidney function for the critical patients was manifest by a significant increase in creatinin level (3-4 times greater than the norm).

Certain special studies were made, the results of which are being analyzed (determination of hydroperoxides, succinoxidismutase, malonic dialdehyde, ceruloplasmin, alphatocoferrol and erithrocyte peroxide hemolysis).

In studying hemostasis of victims initiated 5 days after the accident, plasma procoagulants activation was observed and maintained during development of profound thrombocytopenia which is confirmed by readings of the autocoagulation test.

On the 10th day, sharply positive paracoagulation tests were noted for the majority of the victims. The fibrinogen and antithrombin III levels dropped. The prothrombin index continued to decrease to the fourth week of sickness; the drop in K-dependent factor of the prothrombin complex was most apparent.

At the beginning of the month all coagulation indices for the overwhelming majority of patients approximated the normal. In spite of clinical assumptions about the presence of DVS syndrome, there was no laboratory confirmation of typical dynamics in any case.

Immunologic investigation was used mostly in typing and selecting bone marrow transplant donors.

Immunologic selection and control of blood group and Rh factor (more than 200 persons) made it possible to more adequately provide transfusion therapy (erithrocyte administration). For a limited group of patients, typing was done according to erithrocytaric antigens (TABLE 7.1.3).

Isosensitization to tissue antigens was evaluated by organizing an indirect Kumbs test, lymphocytotoxic reaction, aggregate hemagglutination. In isolated cases, the lymphocyte subpopulations were determined by various techniques. The overall volume of studies for assessing microdestructive processes in the nervous system was performed by means of neuroimmunologic cellular serum tests.

Bacteriologic investigation of environmental microbial dissemination was used widely under various conditions of patient confinement. Inoculations were made of blood, excrement, urine, from mucosa of the mouth and throat, wound surfaces. The level in the blood of certain antibacterial preparations and antibiotics was quantitatively assessed.

Results are in the process of being analyzed.

7.1.6. Skin Changes and Their Role in the Outcome of Sickness.

Characteristics of the reaction of the skin and mucosa in this situation involved the presence of several variants of affection, sometimes simultaneously present for the same patient.

- superficial propagated, primarily found on the body parts unprotected by clothing;

- limited to areas of primarily direct contact with beta-gamma-emitting sources (wet clothing or shoes contaminated with technological solution application of dust or contact with contaminated objects);

- damages to the skin and mucosa, mouth and throat, intestine from relatively uniform gamma radiation in doses exceed threshold for the indicated tissues.

Radiation damage of the skin (beta-burns) over more than 1% of the body surface were observed in 48 persons.

The contribution of radiation damage of the skin to the overall clinical syndrome of OLB with its considerable aggravation was determined by the extent and depth (stage) of damages. In this case, for certain patients (14 persons), skin damages were practically incompatible with life.

The clinically manifested extent of skin damages for the majority of victims sustained some dynamics over time and was characterized by the manifestation of several, at least two-three erythema "waves" and subsequent skin changes.

Primary skin erythema detectable the first and second days after exposure, was not a reliable enough criterion for predicting the subsequent course in view of its instability and the absence of reliable methods for quantitative evaluation of its expression.

In regard to the extent and expression of the primary wave of erythema on the days from the end of the 1st to the 3rd week, 8 persons are isolated with almost total skin damage (from 60 to 100% of gross area). Cutaneous hyperemia for them was accompanied by edema; bubbles and erosions (erosive ulcerous dermatitis) developed early. All these people died on days 15 to 24. They had critical and extremely critical hemogenetic damage and radiation intestinal syndrome. However, we feel it is expedient to again stress that these patients sustained radiation skin damage incompatible with life.

Damage extending over 30-60% of the body surface was revealed in 12 cases until the end of the third week. For the majority of them (7 persons), the severity of bone marrow syndrome was evaluated as extremely critical; for three, as critical; for one, average severity. The number of deaths in this group was 9.

For six victims, skin damages could be evaluated as incompatible with life (extent greater than 50%, earlier formation of extensive erosive ulcerous surfaces). These six persons died; for one skin damage was the primary cause of death (death on the 48th day) under wholly restored peripheral blood chart. Endogenic intoxication for this patient caused the development of toxic edema and terminal coma.

Skin damage with total area to 30% at day 21 was observed for 21 persons. Of them, for 6 victims serious aggravation of the overall condition can be considered due to both the extent (25-30%) and severity of skin damage, with early development of erosive ulcerous changes. Bone marrow damage in this patient group varied from extremely critical to mild. There were no fatalities in this group due to skin damage.

On days 36-45 (6-8 weeks), i.e., in the period of total or almost total recovery of hemogenesis, the previously altered skin covering had recovered. Simultaneously, unexpectedly late, new changes arose on previously unaffected sections in the form of bright erythema with edema. The gross area of damage rose accordingly: pre-evaluated at 25-30%, it reached 90-100% of the body surface. On the areas of previously altered skin, sometimes edema again intensified, the dimensions of areas of healing ulcers and erosions increased. For certain victims with such "late" skin damage -- there were practically no skin changes on the early days (up to 3 weeks).

On days 36-45, the most typical were damages in the area of the crura and thighs. The patients noted the appearance (or intensification) of pains in the legs -- to the point of being unable to stand. Phenomena of lymphostasis and edema of the more distal "focus" of the skin damage (e.g., edema of the ankles in erythema on the crura) were observed, as well as an overall response in the form of increased body temperature, sleep disturbances and so on.

Recovery of skin damage by the 50-60 day was primarily ended. It proceeded as a type of dry and moist desquamation according to the degree of damage. By this time for many patients, erosions and superficial ulcers were epithelialized.

The absence of active epithalization by this time for large areas (20-25 square cm) was evaluated as an indication for surgery.

7.1.7 Treatment Procedures and Preliminary Assessment of Their Effectiveness

Treatment procedures tested in daily practice were used for individual syndromes of acute radiation sickness.

The primary directions in treatment were: prophylaxis and treatment of infectious complications and blood cell substitute therapy in connection with bone marrow syndrome, detoxification therapy and total parenteral feeding; in connection with extensive burns, oropharyngeal and intestinal syndromes, intensive correction of aqua-electrolyte exchange for patients with intestinal syndrome and toxicoseptic status due to burns and agranulocytaric infections.

The dynamics of primary manifestations and treatment methods are illustrated in clinical charts (Fig. 7.1.9 and 7.1.10).

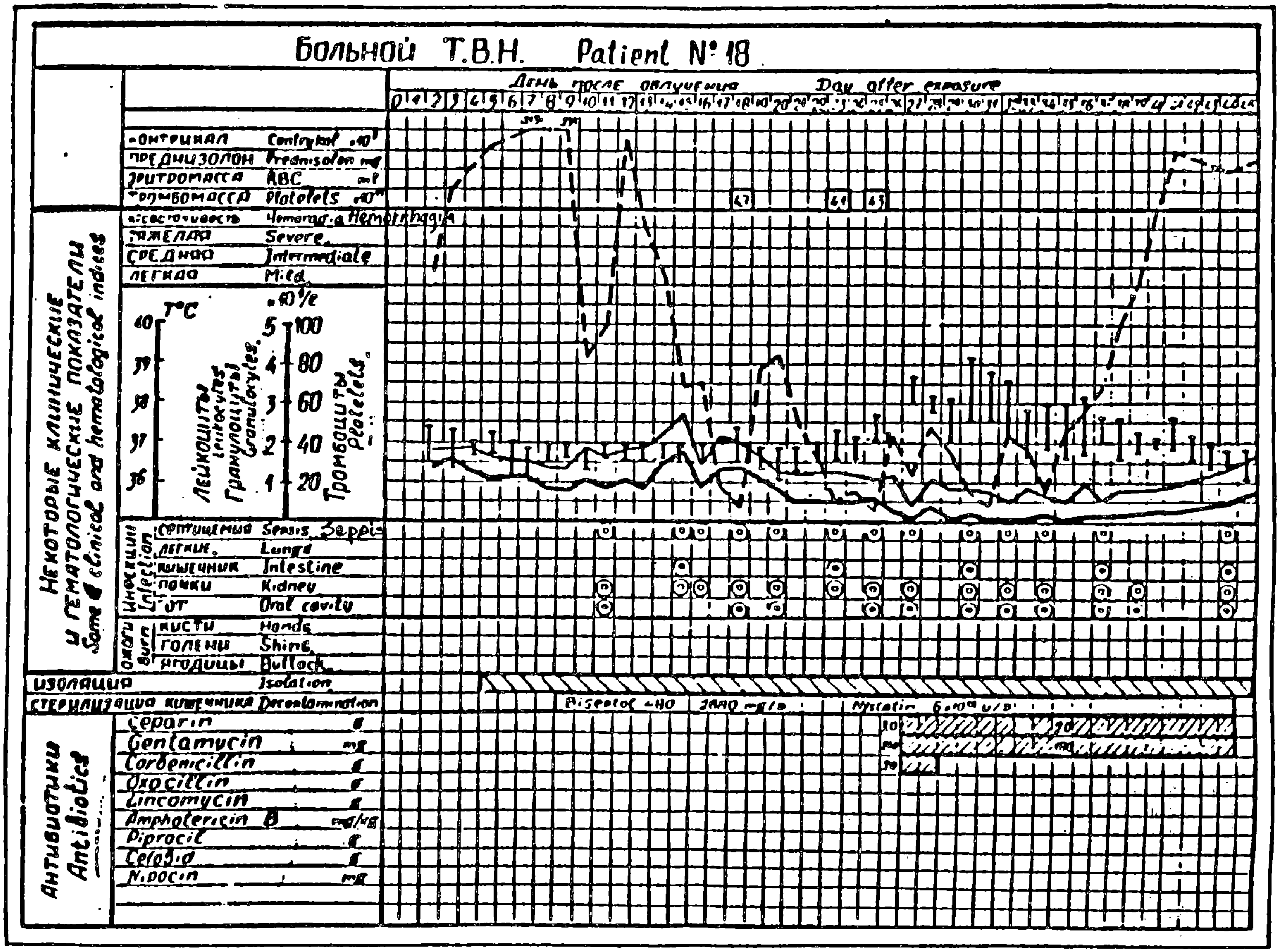
Treatment of Bone Marrow Syndrome: a) Maintenance and substitute therapy. All patients with stage II or greater bone marrow syndrome were located in their own ordinary hospital rooms equipped with provisions for aseptic management of the patients: sterilization of the air with UV lamps; strict observance by personnel of hand washing upon entry and exit from the room, obligatory use of individual gowns found in the room, masks, covering shoes with slippers soaked with antiseptic, changing the patient's linen once a day. The regime of relatively simple aseptic confinement of patients by specially trained personnel was provided at the earliest times for the entire hospital. Bacterial contamination indices were monitored. Paper linen and clothing for the personnel were used. This regime provided low microorganism level in the air -- no more than 500 colonies per cubic meter.

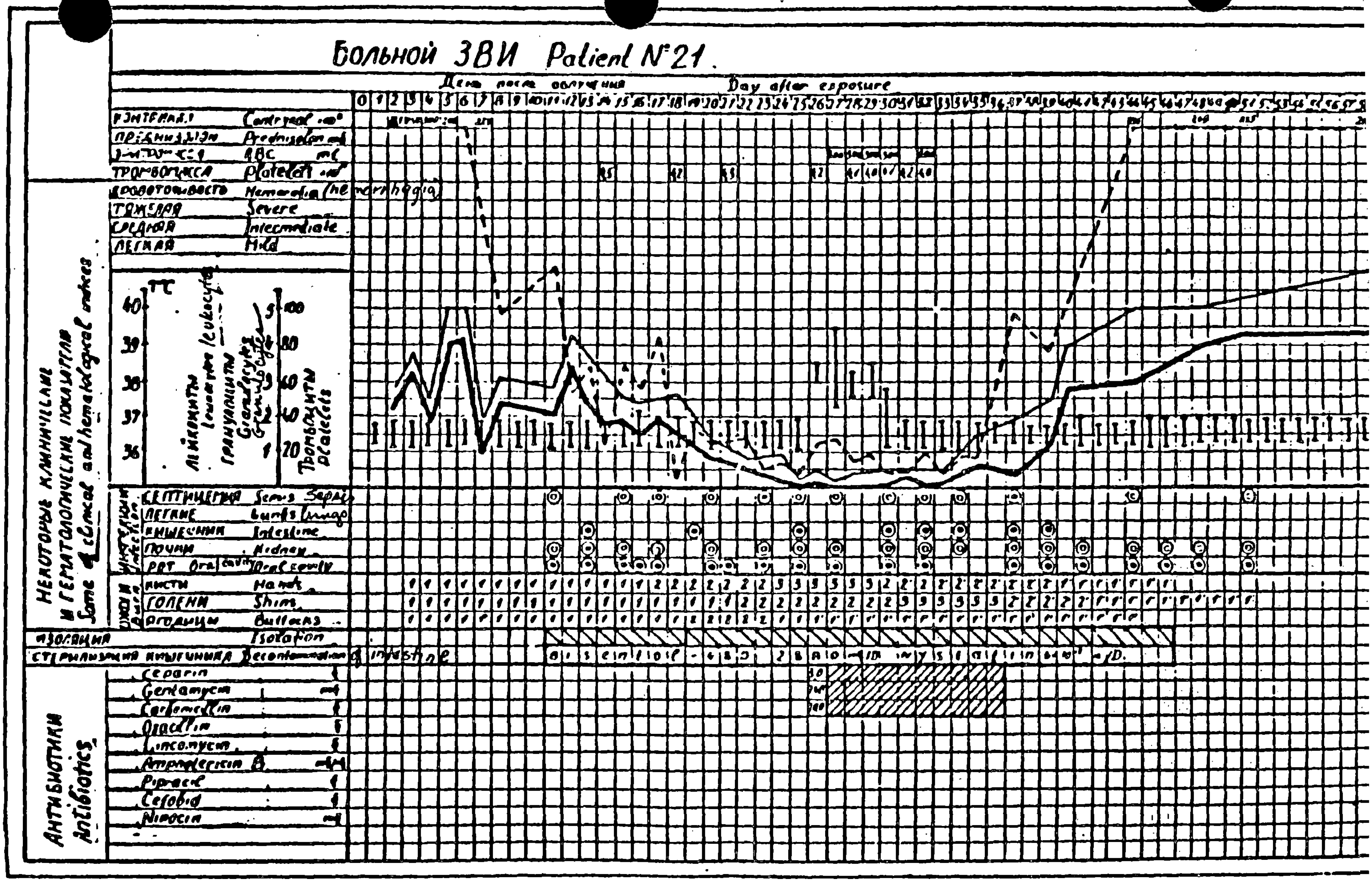
The food was ordinary; raw vegetables and fruits and canned products were excluded.

The effectiveness of such aseptic conditions was clearly demonstrated, as this was conducted earlier by us (A. Ye. Baranov et al., 1978, 1982), by the absence of exogenic bronchopulmonary infections (pheumonias) for patients with acute radiation sickness of stages II-III.

All stage II-IV patients of bone marrow syndrome were provided prophylaxis for endogenic infections with biseptol and nystatin, initiated after 1 or 2-3 weeks before the development of agranulocytaric

Fig.
Рис. 7.1.9.


Больной Т.В.Н. Patient № 18
День после облучения Day after exposure
Centrylal
Prednisolon
ABC
Platelets
Hemorrhagia
Тяжелая Severe
Средняя Intermediate
Легкая Mild
T°C
Лейкоциты Leukocytes
Гранулоциты Granulocytes
Тромбоциты Platelets
Некоторые клинические и гематологические показатели
Some clinical and hematological indices
Инфекции Infection
Септицемия Sepsis
Легкие Lungs
Кишечник Intestine
Почки Kidney
Рот Oral cavity
Ожоги Burn
Кисти Hands
Голени Shins
Ягодицы Buttock
Изоляция Isolation
Стерилизация кишечника Decontamination
Biseptol
Nystatin
Антибиотики Antibiotics
Ceporin
Gentamycin
Carbenicillin
Oxocillin
Lincomycin
Amphotericin B
Diprocil
Cefobid
Nipocin



Больной ЗВИ Patient N°21.
День после облучения
Day after exposure
Centrifugal
Prednisolon
ABC
Platelets
Hemorrhagia (hemorrhagia)
ТЯЖЕЛАЯ Severe
СРЕДНЯЯ Intermediate
ЛЕГКАЯ Mild
T°C
40
39
38
37
36
ЛЕЙКОЦИТЫ Leukocytes
ГРАНУЛОЦИТЫ Granulocytes
ТРОМБОЦИТЫ Platelets
100
80
60
40
20
СЕПТИЦЕМИЯ Sepsis
Lungs
Intestine
ПОЧКИ Kidney
Oral cavity
Hands
ГОЛЕНИ Shins
Buttocks
Isolation
Decontamination of intestine
Ceparin
Gentamycin
Carbenicillin
Oxacillin
Lincomycin
Amphotericin B
Piperacil
Cefobid
Nitrocin
НЕКОТОРЫЕ КЛИНИЧЕСКИЕ И ГЕМАТОЛОГИЧЕСКИЕ ПОКАЗАТЕЛИ
Some of clinical and hematological indices
АНТИБИОТИКИ
Antibiotics

infections. The comparative effectiveness of the two versions for initiating selective intestinal decontamination is being evaluated.

In case of agranulocytaric fever, two or three wide spectrum antibiotics from the group of aminoglucosides (gentamycin, amicasin), cephalosporins (kefsol, sephamesin, sephobid) and semisynthetic penicillins with antipyocyaneus activity (carbenicillin, pipracyl) were administered intravenously. In at lest half of the cases, such antibiotic prescription terminated fever. In the case of an absence of an effect over a 24-48 hour period, intravenous gamma-globulin, supplied by Sandos was widely used to treat this group of victims. Gamma-globulin (sandoglobulin) was administered in 6 g doses every 12 hours 4-5 times.

The practice of "early" experimental prescription of amphotericin B was conducted if agranulocytaric fever was not arrested within the first week by the indicated antibiotics combined with intravenous administration of gamma-globulin.

In this situation, acyclovir used in treating herpes infections (nearly one-third of the patients occasionally has sever herpes simplex of the face, lips and oral mucosa), was widely used for the first time to treat patients with acute radiation sickness with good effect. Acyclovir was not used prophylactically. A salve containing acyclovir yielded a positive effect in treating viral infections of the skin.

This regime of primary experimental antibacterial, antifungal and antiviral treatment was highly effective -- such that there were practically no fatalities due to infection in patients with bone marrow acute radiation sickness, even of the severe and extremely severe stage (without burns). In addition, upon autopsy of patients who died from affections other than to the bone marrow, there were definite indications of bacterial or mycotic septicemias.

In life and posthumously, epidermal staphylococcus was inoculated from the blood most frequently for the dead patients. Its role as a pathogen of terminal septicemias, is being studied in regard to this antibacterial regime.

Several patients with stage IV bone marrow syndrome had acute diffuse interstitial pneumonia accompanied by rapid development of hypoxemia incompatible with life. The bacterial and mycotic nature of the pneumonia was not confirmed at the autopsies. Most frequently, acute radiation pulmonitis had occurred, with possible activation of cytomegaloviruses.

Definite success in treating this patient group with severe acute radiation sickness was realized by wide utilization of fresh donor platelets. The platelets were obtained by 4-fold thrombocytapheresis from individual donors. Indication for platelet infusion was initiated hemorrhaging or a drop in platelet level below 20,000/microliter. Platelets obtained from one donor (an average of nearly 300 x 10exp9 platelets) were usually used for one infusion. Platelet infusion was repeated every 1-3 days. The platelets, like other blood components, before infusion were exposed to 1500 rads in an ordinary gamma-therapeutic device. This allowed prophylaxis of secondary illness.

The high effectiveness of platelet transfusions performed according to these principles is confirmed not only by the absence of life-threatening hemorrhagia even for patients with long-term (more than 2-4 weeks) and severe thrombocytopenia (less than 5-10 thousand/microliter), but also the absence in general of any symptoms of hemorrhaging in the majority of patients.

Organizing the acquisition of the necessary amount of platelets at the prescribed time in the period of greatest thrombocytopenia at once for tens of patients, required major efforts of the blood service. There was no deficit of platelets. In addition, sometimes there was even extra due to the impossibility of accurately planning the demand a day or two in advance. In connection with this, a technique for freezing both allogenic and autologic thrombocytes which were then used at the necessary time with high effectiveness, was widely employed for the first time.

There were no cases of the development of nonsusceptibility to platelet transfusions in connection with alloimmunization.

On the average, 3-5 infusions of platelets were required when treating one patient with stage III severity of bone marrow syndrome.

Leukocytes for treating agranulocytaric infectious complications were not used.

The demand for erithrocytes was much higher than expected, even for patients with stage II-III acute radiation sickness uncomplicated by radiation burns in connection with the development of early and expressed anemia.

b) Bone marrow transplant. The first group of patients with irreversible myelopoetic damage in which spontaneous recovery was practically not expected, was selected over the first three days after exposure. Diagnosis of irreversible myelodepression was made according to guidelines developed earlier based on such criteria, as time of onset of vomiting number of peripheral blood lymphocytes and estimate of absorbed dose from the number of aberrations in bone marrow cells taken 36 hours after exposure.

On subsequent days (from the 4th to the 9th), according to these criteria and also from the dosage estimate according to chromosomal aberrations in the lymphocyte culture, peripheral blood, the degree of myelopoetic damage was refined and a group of persons was conclusively formed, who, in regard to the extremely severe (possibly irreversible) affection of myelopoesis (dose of 6 GR or more), were indicated for bone marrow transplant. At these times, the American specialists headed by Professor Robert Gale, actively participated in the work.

Transplantation was done only from close relatives (natal brothers and sisters or parents) identical (6 cases) and haploidentical according to HLA (4 cases) or haploidentical plug one common antigen in the second haplotype (3 cases). Typing, in view of the urgency of transplantation, was done only according to A, B and C loci. In the instance of transplanting haploidentical bone marrow, removal of T-lymphocytes -- T-depletion -- was done for secondary disease prophylaxis.

Special difficulty in selecting HLA appropriate donors involved the need to determine HLA fenetin for most of the patients, namely over the first 1-3 days after exposure, so long as the number of lymphocytes did not drop to very low numbers.

Major organization difficulties occurred in connection with the need for quickly summoning and studying a large number of potential related donors. In this situation, 113 donors were checked.

In the final calculation, only 13 allogenic bone marrow transplants were performed from the 4th to the 16th day after exposure. In six cases with extremely severe damage to the skin and intestine and extremely unfavorable prognosis was human embryonal liver cell transplantation performed.

In general, it may be said that bone marrow transplantation was not a decisive method of treatment in this emergency situation. All seven patients, where in view of the specially intense exposure, stable adaptation of the donor bone marrow could have occurred, died earlier (on the 9-19th day after TKM) from radiation damage to the skin and intestine. At the same time, for the remaining six patients which had no skin and intestinal damage incompatible with life, only temporary or partial adaptation of the donor bone marrow occurred, apparently, in view of the fact that transplantation immunity of these patients was not fully suppressed by exposure. In spite of the poor (inadequate or incomplete or lost) myeloid function of the transplant, for all these patients, some or other disturbances were observed, which may be due to the reaction of the host vs. transplant or transplant vs. host reaction. In two cases, these reactions could have facilitated the lethal outcomes -- from a renal or pulmonary insufficiency syndrome not identified according to genesis and from pyocyanic septicemia, unexpectedly developed on a background of a normal number of neutrophils.

Retrospective analysis of the initial section of the neutrophilic curve for all six patients allows doubt in the irreversibility of myelopoetic damage for them. Adaptation, even temporary, of the grafts and the development of associated immunologic conflicts, evidently indicates the negative influence both on recovery of myelopoesis, and on the severity of the course of the disease as a whole (two died).

Thus, experience of allogenic bone marrow transplant in this radiation accident makes it possible to draw two important conclusions for future treatment in regard to this procedure:

- in radiation accidents, there is always a very small fraction of persons, for which allogenic bone marrow transplant is absolutely indicated and may clearly be useful;

- in reversible damage of myelopoesis caused by doses of gross gamma radiation of the order 6-8 GR, adaptation of the transplant is possible; however, such adaptation will always be unfavorable in a therapeutic regard and even dangerous to life, in view of the high risk of secondary disease.

The latter conclusion is principally new, since it was previously felt that allogenic bone narrow transplant yields no negative effects in case of inadequate radiation conditioning of the recipient in the zone of limited exposure doses.

Radiation Skin Damage Treatment

Considering the major, and, in a number of cases, the determining role of local radiation damage in the gross clinical syndrome (intoxication, pain), their treatment played a large part in therapeutic measures.

Basic modern methods of detoxification, as well as disaggregant, anti-infective and symptomatic therapy were employed; hemo- and plasmosorption, plasmapheresis were performed, direct anticoagulants, drugs improving microcirculation (rheopolyglucine, neohemodes, troxevasin, trental, solkoseryl) were used.

Local treatment procedures corresponded to the stage and severity of damage: early on bactericidal and analgesic antiinflammatory aerosols were used, with the Soviet-manufactured preparation lioxanol offering immediate relief. Moistening bandages based on tanning solutions with bactericidal properties (baliz) were used. Smear bandages with derivatives of hydrocortisone based on propolis and wax with direct action antiseptics and antibiotics, were used later.

Treatment of Oropharyngeal Syndrome

The primary methods of treating severe radiation mycocytes involved mechanical removal of a huge quantity of rubbery mucus accumulated in the nasopharynx, washing off this mucus and irrigating erosive surfaces with solutions of mycolytics with antiseptics.

Experience shows that mycolytic preparations and employment techniques for rapid and reliable elimination of mucus from the cavity of the mouth, entrance to the mouth and nasopharynx require special refinement.

Treatment of Intestinal Syndrome

The primary procedure for treating intestinal syndrome was total paraenteral feeding with intensive adjustment of the volume of nutritional liquid and electrolytes, which in this case, proved highly effective. Experience showed that a major supply of ready mixtures for parenteral feeding must be on hand if indication of specialized aid to patients with severe radiation damage to the intestine, mouth and throat is proposed.

In conclusion, it may be noted that each patient with stage III-IV severity of bone marrow syndrome usually accompanied by radiation burns, required individual round-the-clock nursing by highly qualified and specialized intensive care nurses to accomplish the indicated therapy. At the cost of efforts, which naturally are possibly only in peacetime, such specialized nursing stations were organized for each victim with stage III-IV OLB.

On the whole, the effectiveness of treatment may be assessed as wholly satisfactory: in the group of persons with stage II OLB (dose of 2-4 GR), there was not one death. Fatalities among those with stage III and IV severity of OLB in 19 cases could be due only to severe damages over 50-90% of the body surface incompatible with life, which, in turn were of stage II-IV degree of severity.

Among the operative and emergency personnel participating in operations on 26 April 1986 at Chernobyl AES, 203 patients with acute radiation sickness (OLB) were established. Its clinical manifestations (syndromes), degree of severity and outcomes were diverse. All patients were recognized opportunely and treated at qualified institutions around the country.

Of the 22 patients with extremely severe OLB, 19 died. In 14 cases, the fatal outcome was defined by severe radiation and thermoradiation damage to the skin on a background of profound hemogenetic suppression, and for a portion of the patients, damage to the alimentary canal.

Of the 23 patients with severe bone marrow syndrome, 7 died, in 6 cases, the outcome of the disease was also determined by the presence of disseminated severe radiation burns of the skin with expressed gross intoxication accompanying them.

Among the patients of stage I and II severity, not one died.

Clinical recovery for the majority occurred toward the close of 2 months from the day of the accident. Thirty persons are now hospitalized and recovering.

The primary damaging factor for all victims was the relatively uniform influence of gamma-beta-radiation at a dose exceeding 1 GR according to biologic criteria, and for 35 of them, more than 4 GR (to 12-16 GR).

For 50 persons, major portions of the skin's surface and part of the face and mucus nasopharynx and gastrointestinal tract sustained additional beta radiation. Radiation damage to extensive portions of the skin (up to 50-90%) were the leading among the causes for severity of the general state of patients and defined the mechanism of primary fatal complications (cerebral edema, toxic encephalomyelopathy, renal and liver failure, myocardial damage).

For individual patients in the range of maximum external gamma radiation (approximately 8 GR), the terminal period was characterized by the development of pulmonitis and expressed respiratory failure.

For practically all victims, without apparent association with the presence and severity of OLB, entrance into the body of a complex mixture of nuclides, primary iodine, cesium, zirconium, niobium and ruthenium isotopes, was discovered. However, their quantity and the dosage level for all, with the exception of one patient, were below the clinically significant for direct effects. The level of iodine isotopes in the thyroid gland for 94% of the victims did not exceed 50 microCurie over the first 10 days from the moment of the accident.

Experience gained previously allowed timely and relatively complete assessment of the prognosis for the disease for the overwhelming majority, and proper determination in the first day of indications for urgent hospitalization, over the first three days -- a rational volume of medical assistance, and later -- achieving definite therapeutic results.

In the process of intensive clinical observation and active therapeutic measures, a great volume of information was accumulated which is now being processed.

Preliminary results are reduced to several primary statements.

1. The fundamental principles of diagnosis and prognostic criteria of the course of the disease previously developed with regard to bone marrow syndrome were justified.

2. Disseminated beta damage of the skin for a major portion of the patients defining the severity and outcome of the disease, played a determining aggravating role in this emergency situation.

3. Among the therapeutic measures, in accordance with the structure and syndromology of damage, of primary importance were measures on prophylaxis and treatment of complications associated with the advanced, but reversible hemogenetic depression, decontamination therapy and local treatment of skin damage.

4. The nature of (relatively shallow, but very wide-spread beta-dermatites) primarily demanded conservative therapy and only in rare cases (at this time, for 5 persons), surgery.

5. Bone marrow transplant was indicated (dose of more than 6 GR) and possible (absence or weak expression of other factors for an unfavorable outcome), only for a quite limited contingent (13 persons). In view of those factors and due to retention of capabilities, though gradual and partial reparation of natural resources, the effectiveness of transplantation was low.

6. Dynamic observation has been organized for all patients, which will later define the completeness of their rehabilitation and the need for therapeutic and prophylactic measures.

7.2 DATA ON THE RADIATION DOSE MAGNITUDES OF THE POPULATION WITHIN A 30-KM RADIUS AROUND THE AES AND IN VARIOUS REGIONS OF THE EUROPEAN PART OF USSR: RADIATION CONSEQUENCES OF THE ACCIDENT

7.2.I Introduction

Measures were taken immediately after the accident to implement continous operative control of the radiation situation parameters, both on the territory of the Chernobyl AES, as well as in populated sites in the vicinity. Special attention was given to the city of Pripyat', with a population of about 45,000, consisting primarily of AES personnel and their dependents. Based on the data of the developing radiation situation, the extent and volume of dosimetric control significantly rose over time. To implement this, over 7,000 subdivisions of radiation laboratories, health stations, as well as many groups of radiation safety specialists, a large number of scientists and practical institutions and organizations were activated.

The primary and most important tasks of radiation control were:

-- to evaluate the possible level of external and internal radiation of the Chernobyl AES personnel, the residents of the city of Pripyat', and people that were then evacuated from the 30-km zone, with the purpose of finding individuals in need of medical assistance;

-- make an estimate of the possible levels of radiation of the population in the areas of increased radioactive contamination within the 30-km zone, in order to make a decision concerning the need for a complete or partial evacuation, or the development of appropriate temporary recommendations concerning food and daily activity in the given region;

-- eliminate the distribution of radiactive matter from the contaminated regions via contact, as well as the use of good products containing radionuclides above the recommended magnitude.

In order to solve the above problems, a systematic control was implemented for the following:

-- the levels of gamma radiation over the entire area of European USSR through use of aerial and ground radiation checks;

-- the concentration and radionuclide composition of radioactive substances in the air at various points of the 30-km zone, primarily at sites where efforts were being conducted to eliminate the consequences of the accident and removal of the personnel, as well as outside the 30-km zone in populated areas where increased radiation levels were noted;

-- density of radioactive contamination of the soil and vegetation and the radionuclide composition of this contamination;

-- content of radionuclides in water supply reservors, as well as in water used by the commercial food products network;

-- content of iodine radionucleides that make up the principal dosage of internal radiation during the initial period after the accident, collecting in the thyroid gland of the population that had been evacuated from the 30-km zones and living in the areas having an increased level of gamma-background;

-- levels of radioactive contamination of safety clothing or personal clothing and shoes, external and internal surfaces of transport conveyances within the boundary of the affected zones (determined on the basis of types of efforts and the radiation conditions in existences), at airports, railroad and bus terminals.

7.2.2. LEVELS OF EXTERNAL RADIATION OF THE POPULATION OF THE CITY OF PRIPYAT' FROM THE POINT OF THE ACCIDENT TO THE POINT OF EVACUATION

From the start of the accident at the 4th unit and during the fire that followed it, the wind carried radiactive products, bypassing the city of Pripyat'. Subsequently, when the height of discharged products from the damaged reactor had been substantially reduced, due to the fluctuation of the wind direction in the above-ground layer of air, the radioactive flare covered the territory of the city during certain time intervals, and slowly contaminated it. Until 21:00 26 April 1986, the magnitude of exposure to gamma radiation on individual streets of the city, measured at a height of 1 m above the earth's surface, was within 14-140/mR per hour.

Subsequently, the radiation situation in the city began to worsen. By 7:00 on 27.04.86, in the area closest to the AES (Kurchatov Street), the magnitude of exposure to gamma-radiation reached 180-600 mR/h, and on other streets, 180-200 mR/h. The worsening trend of the radiation situation in the city during 27.04.86 continued until 17:00, i.e. until the complete evacuation of the population was carried out, the radiation
360-
was ~~3360-~~540 mR/h, and in the area of Kurchatov Street, 720-1,000 mR/h. Evacuation of the population was begun at 27.04.86, at 14:00.

Figure 7.2.1 shows data for the changing radiation situation in different areas of the city of Pripyat' from the moment of the accident until the evacuation was completed. The exposure dose of gamma-radiation during this time period was 5.9; 7.1; and 20.3 R at points 1, 2, and 3 respectively. By 6 May the levels of radiation in the city of Pripyat' were reduced approximately by a factor of 3. Rough estimates make it possible to assume that the external dose of gamma-

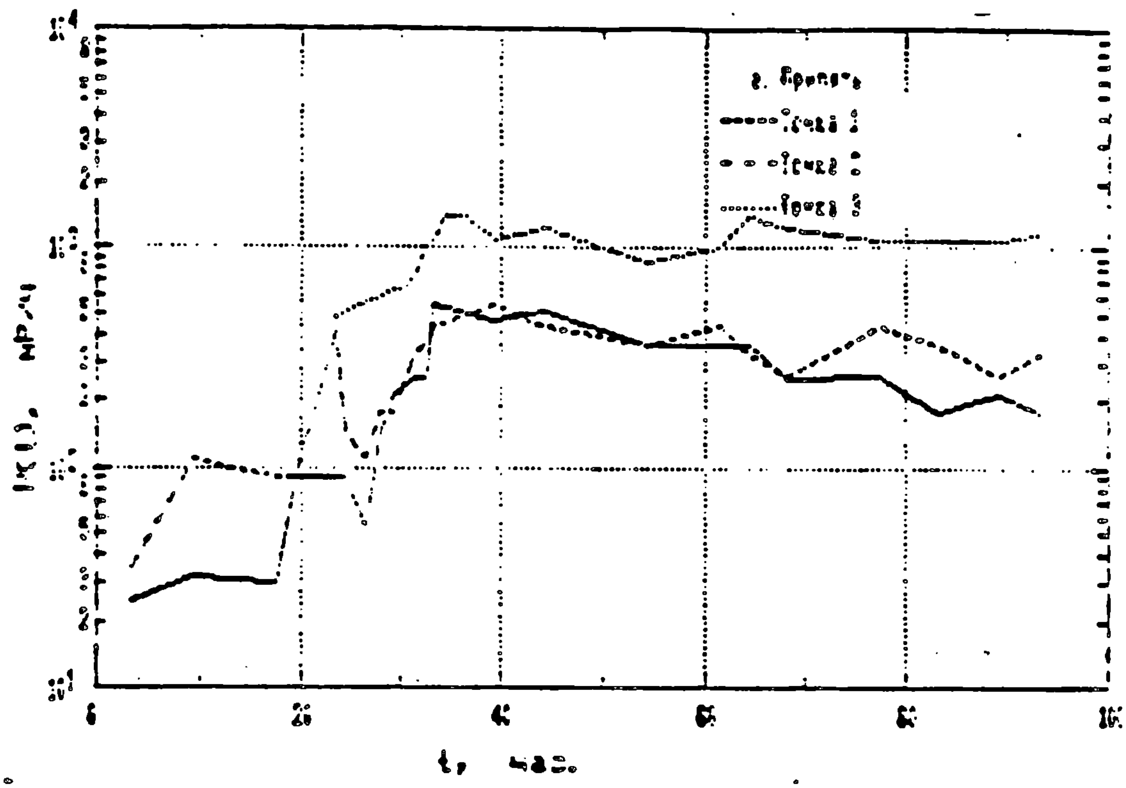


Figure 7.2.1. Dynamics of the change of dosage magnitude in open areas of the city of Pripyat' during the first 4 days after the accident.

gamma-radiation from the passing cloud of discharge matter and during the first hours after the accident was close to 10-15 R.

Evaluation of the levels of radiation received by the city's population was conducted based on the presumed behavior regimen on 26 and 27 April and the data received from individual dosimeters of radiation safety personnel and emergency teams.

Immediately after the start of the accident, the population of the city of Pripyat' received the recommendation to maximally limit the time spent outdoors, and not to open windows. On 26 April any type of open air activity was forbidden at all children's facilities in the city. In addition, iodine was distributed at all the children's facilities. Thus, residents that stayed primarily indoors during the daytime period on 26 and 27 April had received a 2-5 fold lesser effect of gamma-radiation in comparison to the levels taken on the street. Taking into account the above data, there is a basis to assume that the majority of Pripyat's population had received a probable level of radiation with a magnitude of 1.5-5.0 rads of gamma radiation and 10-20 rads of beta-radiation on the skin.

Consequently, the evaluations that were conducted show that the potential levels of external radiation exposure of the residents of Pripyat' were significantly lower than those that may cause any type of direct changes with respect to health. The subsequent medical examinations of the residents of Pripyat confirmed this conclusion.

Measurement of iodine isotopes in the thyroid glands of people evacuated from the city of Pripyat' to the nearby areas of the Polesskiy region rayon showed that 97 percent of the 206 persons examined showed an iodine content in the thyroid that was less than 30 rads. Here a positive role was played by the iodine given for prophylactic reasons, as well as the limits issued for the use of milk from cows kept privately.

The potential level of radiation due to inhaled iodine could also be judged on the basis of measurement data for iodine content in the thyroid glands of 20 Pripyat' residents that were evacuated to the city of Belaya Tserkov', where use of products contaminated by radioactive substances was absolutely non-existent. Based on measurements conducted on 7 May 1986, the majority of examined individuals can show an effect of 1.5-25 rads in the thyroid gland.

7.2.3. RADIATION DOSAGES OF THE POPULATION WITHIN THE 30-KM ZONE AROUND THE CHERNOBYL AES

Based on analysis of the radionucleide composition of radioactive fallout at various points in the 30-km around the Chernobyl

AES, an evaluation was conducted of the dynamics of the decay in time of the magnitude of the dose of external gamma-radiation from the earth's surface. This relationship is shown in Figure 7.2.2 by a solid line, while dots indicate actual measurement values of the dosage magnitude in the area in relative units. A rather good correlation of the estimated curve and the experimentally derived data made it possible to conduct specific extrapolation evaluations both the greater time periods (a year and longer) after the accident, as well as the period during which the discharge cloud passed. Estimates worked on a specially developed computer program, after appropriate corrections for actually observed magnitudes of the parameters under analysis made it possible to get the following correlations between the magnitude of the gamma-radiation dosage on the site on the 15th day after the accident (R gamma, 15 mR/h) and the dosage of external radiation from the radioactive cloud (D cloud, R), the dosage from radioactive fallout at different times after the accident (D fallout, R), as well as the dosage of internal radiation of the thyroid gland in children (D thyroid gland, rad) due to inhalation and use of contaminated cow's milk:

D cloud (10-30 km) = (0.28-0.07) . R gamma, 15

D fallout (7 days) = 0.7 . R gamma, 15

D fallout (1 month) = 1.2 .R gamma, 15

D fallout (1 year) = 2.5 . R gamma, 15

D fallout (50 years) = 8 . R gamma, 15

D thyroid gland (inh.) = 10 . R gamma, 15

D thyroid gland (? per?) = 1000 . R gamma, 15

The last value is for an instance when no limit has been made on the issue of contaminated cow's milk,

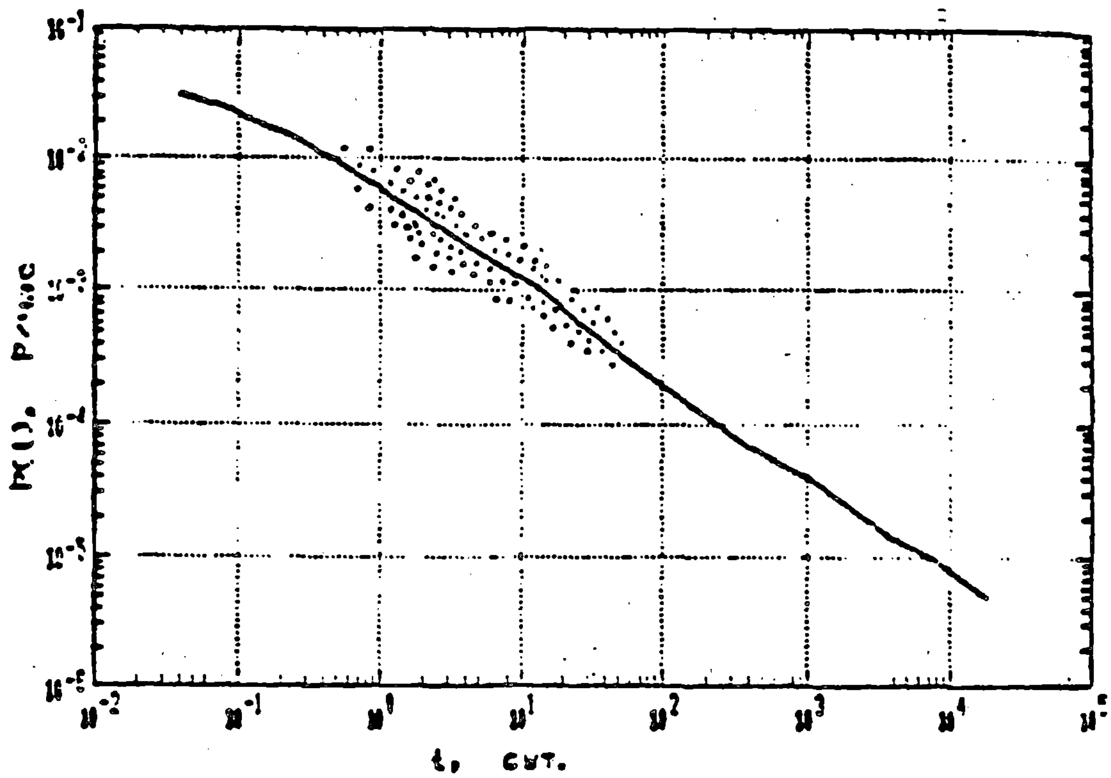


Figure 7.2.2. Changes in magnitude of gamma-radiation doses in open areas in the path of the Chernobyl AES radioactive fallout.

which is naturally possible only in zones with a very low level of vegetation contamination by iodine-131.

In addition to the estimated magnitudes of external radiation dosages of the human body by the fallout cloud and internal radiation of the thyroid gland of children due to inhalation of iodine isotopes, Table 7.2.1. gives a comparison of the dosage magnitudes of external radiation of people in some populated areas of the 30-km zone around the Chernobyl AES. These magnitudes are calculated with the aid of the correlations given above, and the actual values of dosages received, based on the measurements of the gamma-radiation dosage magnitudes on the site. (see Figure 7.2.3.).

Table 7.2.1.

Estimated values of radiation dosages of people in some populated areas in the 30-km zone around the Chernobyl AES.

1) Populated area, KM
2) Distance from Chernobyl AES mR/hr "D" ÷ 15
3) Magnitude of dose per cloud, R
4) Dose from cloud fallout, R
5) Dose from fallout in thyroid gland of children, rads
6) Dose from fallout over 7 days, R
Estimated / [illegible]

1 Населенный пункт	2 Удаление от ЧАЭС км	3 Мощность дозы на "Д"+I5, мР/час	4 Доза от облака выброса, Р	5 Доза на шит.железу детей, рад	6 Доза от выпадений за 7 сут.Р	
					расчёт	измер.
Christogolovka Чистоголовка	5,5	I2	I0	I20	8,4	3,2
Lelev Лелев	9	25	7	250	I7	I0
Chernobyl Чернобыль	I6	8	I,2	80	5,6	3,0
Rud'ki Рудьки	22	8	0,6	80	5,6	2,2
Orevichi Оревичи	29	2,5	0,2	25	I,8	4,4

Analysis of the data in this table indicates that within limits a factor of 2, the estimated and experimental values of the dosages coincide, which made it possible to conduct similar estimates for the entire 30-km zone around the Chernobyl AES during the first days after the accident; this was based on the available data for the radiation situation that had formed. These calculations are shown in Table 7.2.2., which shows

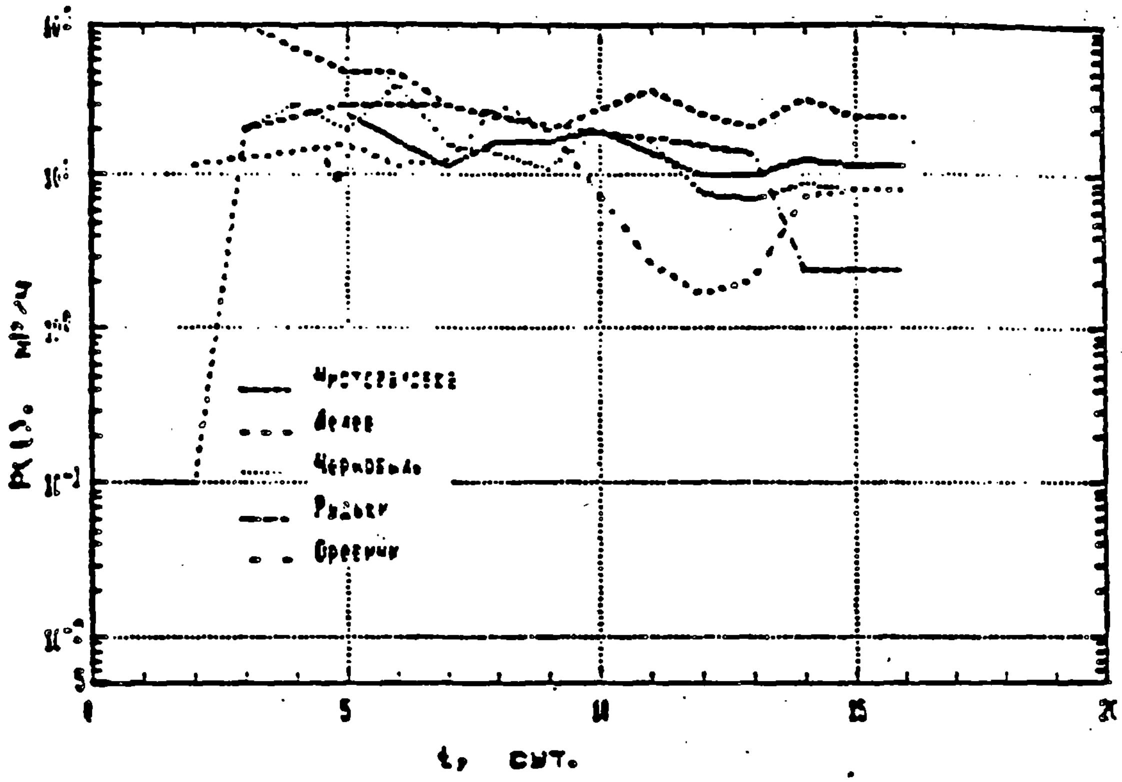


Figure 7.2.3. The dynamics of change of the magnitude of the gamma-radiation dosage in open areas for some populated areas in the 30-km zone.

generalized data for 71 population points in this zone, and indicates the estimated range of dosages of external gamma-radiation in open areas.

A sufficiently broad range (within an order of two) of changes in the dosage magnitude for each zone within the Chernobyl AES is tied to the significant lack of uniformity of the radioactive contamination of the area in the various sections of the resultant pattern of the accident's fallout (see Appendix 5). Based on similar estimates and taking into account the continuing release of gases and aerosols in the accident zone during the first days after the event, a conclusion was formed concerning the wisdom for additional evacuation of residents from the area of the accident. During the first days after the accident, 90,000 people were evacuated from the 30-km zone around the Chernobyl AES. Taking into account 45,000 people moved on 27 April from the city of Pripyat', the total figure of evacuees reached 135,000.

The carrying out of this extreme measure made it possible to assure that the majority of the population received a dose of external gamma-radiation from the discharge cloud and radioactive fallout that did not exceed 25 rem, and only some populated areas that ended up in more heavily contaminated areas of the radioactive pattern (the village Tolstyy Les, Kopachi, and some others), did the radiation fallout dosage in people reach 30-40 rem.However, even with these magnitudes of external radiation, the human body has no danger for acute, direct somatic effects in persons who had received radiation. Maximum estimates of the collective dosage of radiation of the evacuated population (see Table 7.2.3) result in the collective dosage magnitude of radiation in people on the order of

1.6 million of human rem. Taking into account the magnitude of spontaneous morbidity

Table 7.2.2. Estimated dosages of external radiation of farming population(*) within the 30-km zone around Chernobyl AES, rem

Distance from Chernobyl AES, km	Number of populated areas	Dosage of external radiation from fallout for the period of		
		7 days	1 month	1 year
3-7	5	6-80	I0-I30	25-300
7-I0	4	I0-60	I6-I00	35-230
I0-I5	I0	I,2-75	2-I20	4-250
I5-20	I6	0,3-25	0,5-40	I-90
20-25	20	0,4-35	0,6-60	I,3-I20
25-30	I6	0,I-I2	0,2-20	0,4-40

(*) These estimates were derived taking into account the activity regimen of the farming population and the safety coefficients that are provided by village-type structures. For city conditions, these values will be lower approximately by a factor of 2.

Table 7.2.3 Estimated values of collective dosages of external radiation of the evacuated population.

Region around Chernoby AES	Population, thousands of people	Collective dosage, millions of people, rem
Pripyat'	45	0,I5
3 - 7 KM	7,0	0,38
7 - I0 KM	9,0	0,4I
I0 - I5 KM	8,2	0,29
I5 - 20 KM	II,6	0,06
20 - 25 KM	I4,9	0,09
25 - 30 KM	39,2	0,I8
TOTALS	I35	I,6

due to cancern over the 70-year period will result in ~ 14,000 cases for 114,000 evacuated persons; additional morbidity tied to the accident fallout from the Chernobyl AES, will increase natural morbidity from cancern among those receiving radiation by less than 2.0 percent.

More precise data on the radionucleide composition of the contaminated surface of the soil and the nature of decrease in the dosage magnitude of the gamma-radiation on the site (see Figure 7.2.2.) will make it possible to make corrections in the expected values of external radiation dosages in the populations and the determine the possible date for returning the people to their permanent residence site.

Since the problem reactor remained a strong source of radioactive fallout into the atmosphere for a comparatively long time (8-10 days), the picture concerning the contamination of natural features in the environment, both for the level of activity, and the radionucleide composition had a complex nature due to the changingmeteorologic conditions over time, the height and intensity of the discharge. In particular, the formation of abnormally high local contamination of individual sections of the territory were observed. There were also difficults in establishing a certain typical radionucleide compasition of the radioactive emittants in the air and at the contaminated territory. Thus, for example, the content of Iodine-131 varied in air and soil samples from 8 to 40 percent, while cesium-137 ranged from 1 to 20 percent. This complicated forecasts of the possible level of radiation of the population as a result of radioactive products that entered the organism.

Nevertheless, there is a basis for confirming with sufficient assurance that at the given stage, inhaled radionucleides do not have to be considered for those people living along the radioactive trace that had been formed. This is confirmed by data, in accordance to which the air activity within the 30-km zone (Chernobyl, Zorin, Skazochnyy settlement, Pripyat) comprised (TYPIST: See how this is typed on orig. p 64 top) 10^{-12} to 10^{-14} Curie/l for the total beta-activity radionucleides during the period from 3 May to 3 July. Table 7.2.4 shows an example of the relative contribution of gamma-active radionucleides in air samples taken at various populated areas around the Chernobyl AES.

Thus, for the population living in the contaminated area and using locally produced products, the main source of internal radiation are radioactive substances contained in these products. Without question, radioactive substances entering the organism by inhalation during the time the cloud passed should be removed from the calculations. But as will be shown below the resultant internal radiation dosages are substantially lower than those from use of contaminated products.

During the first stage after the accident (~2 months), iodine entering the organism chiefly via milk from milk-producing, pasture-fed cattle was the principal radionucleide for measurable dosages, with the human thyroid gland being the critical organ that gets maximal dosages.

The situation indicated above predetermined the extent and the direction of radiation monitoring and medical examination of the population. Using teams from the specialist institute of the USSR Ministry of Health (Minzdrav), the content of iodine in thyroid glands of people evacuated from the 30-km zone was tested, and was also tested in the residents of a number of populated areas in Ukrainian SSR, Belorussian SSR, and RSFSR, where increased

Table 7.2.4 Relative content of gamma-releasing radionucleides in aerosol air samples (TYPIST: Columns will be listed left to right; I will number cols. 1-4 for ease)

Col 1: Date sample taken Col 2: Place sampled Zorin, Pripyat, Skazochnyy settlement, CHernobyl Col 3: Total gamma activity of sample, Curie/L Col 4: Relative content of radionucleides, %

(TYPIST: Last line is a note) NOTE: Line drawn through indicates that in the given sample the radionucleide was not identified.

Таблица 7.2.4

Относительное содержание гамма-излучающих радионуклидов в аэрозольных пробах воздуха

COL 1 COLUMN 2 COLUMN 3 COLUMN 4

ата отбора проб	Место отбора	Суммарная β активность пробы, Ки/л	Относительное содержание радионуклидов, %											
			I^{131}	Cs^{134}	Cs^{137}	Ba^{140}	La^{140}	Ce^{141}	Ce^{144}	Zr^{95}	Nb^{95}	Ru^{103}	Ru^{106}	Te^{132}
4.05	Зорин	$1{,}5 \cdot 10^{-13}$	20,0	-	4,0	-	5,0	12,0	-	8,0	20,0	30,0	-	1,0
3.06	г.Припять	$4 \cdot 10^{-10}$	-	1,2	0,8	1,1	2,5	11,2	12,1	19,2	26,1	19,1	6,7	-
3.06	п/л Сказочный	$6 \cdot 10^{-13}$	0,46	3,9	7,7	18,8	27,9	3,5	3,2	17,0	2,0	4,9	4,6	-
3.06	Чернобыль	$4{,}0 \cdot 10^{-12}$	5,9	0,8	1,6	3,7	5,0	11,2	9,0	18,6	22,9	11,7	9,6	-

Примечание: Прочерк указывает, что в данной пробе радионуклид не идентифицирован.

levels of radiation were recorded, but the need to take the decision to evacuate was absent. Special attention was given to the children segment of the population, who as a group with an increased radiation risk. This is due to the fact that the accumulated dose of radiation in the thyroid gland, with an equal content of radioiodine is 8-2 times higher in children aged 2-14 years than in an adult. In addition it should be noted that the share of milk products is substantially higher in the diet of children.

Whenever possible, along with tests of iodine content in the thyroid gland, selective determinations were made of the content of other radionucleides in the organism on the basis of active secretions (urine, feces), as well as evaluated the potential entry of other radionucleides into the organism, using data for isotopic composition of contaminated soil and food products. During the time period after the accident, a direct determination of radioiodine was made in a large number of residents, including almost 100,000 children. Practically the entire group of children (up to age 15) was examined, as well as part of the adult population that were evacuated from the 30 km zone, and other populated areas located along the trace of the radioactive cloud and where increased radiation levels were registered. It is necessary to note that the population segment indicated above ate locally produced food, including milk and milk products for 9-10 days until the moment of evacuation (4-5 May); the specific share of these food products is significant in this part of the region.

Measurement results showed that for a majority of people evacuated from the 30 km zone, the dosages in the thyroid gland derived from radioactive substances in locally produced foodstuffs is significantly lower than those that may cause some sort of changes in the state of health.

The sufficiently high dosages that were sometimes observed in thyroid glands of individuals apparently occurred in some instances due to uncontrolled use of milk from privately owned cows, even though the health services put into effect a ban on the use of whole milk having a radioiodine content above $1 \cdot 10^{-7}$ Curie/l. This requirement was strictly carried out within the framework of the centralized milk delivery system. Additional measures were subsequently made for rigid control over the sale and use of milk from cows that were privately owned.

As a preventive measure, the entire group of children from the 30 km evacuation zone was moved under general orders, to summer health facilities in the country. Constant medical supervision was set over children whose estimated radiation dosage in the thyroid could exceed 30 rem prior to complete removal of iodine isotopes.

7.2.4 RADIATION CONSEQUENCES OF THE CHERNOBYL AES ACCIDENT FOR THE POPULATION OF INDIVIDUAL REGIONS OF THE EUROPEAN PART OF USSR

As was indicated in the preceding chapters of the report, the Chernobyl AES accident's radioactive discharge had an effect not only on the radiation situation near the plant, but also at significant distances from it. Figures 7.2.4 and 7.2.5 show changes over time of the magnitude of the gamma-radiation dosage in open areas in certain oblast (county) centers of UkrSSR, BSSR, and RSFSR, that were 100 to 1,000 km away from Chernobyl AES. The figures show that practically in all of the populated areas

Figure 7.2.4 The dynamics of change of the magnitude of the gamma-radiation dosage in open areas for oblast (county) centers in UkrSSR (a), and BSSR (b), located near the Chernobyl AES.

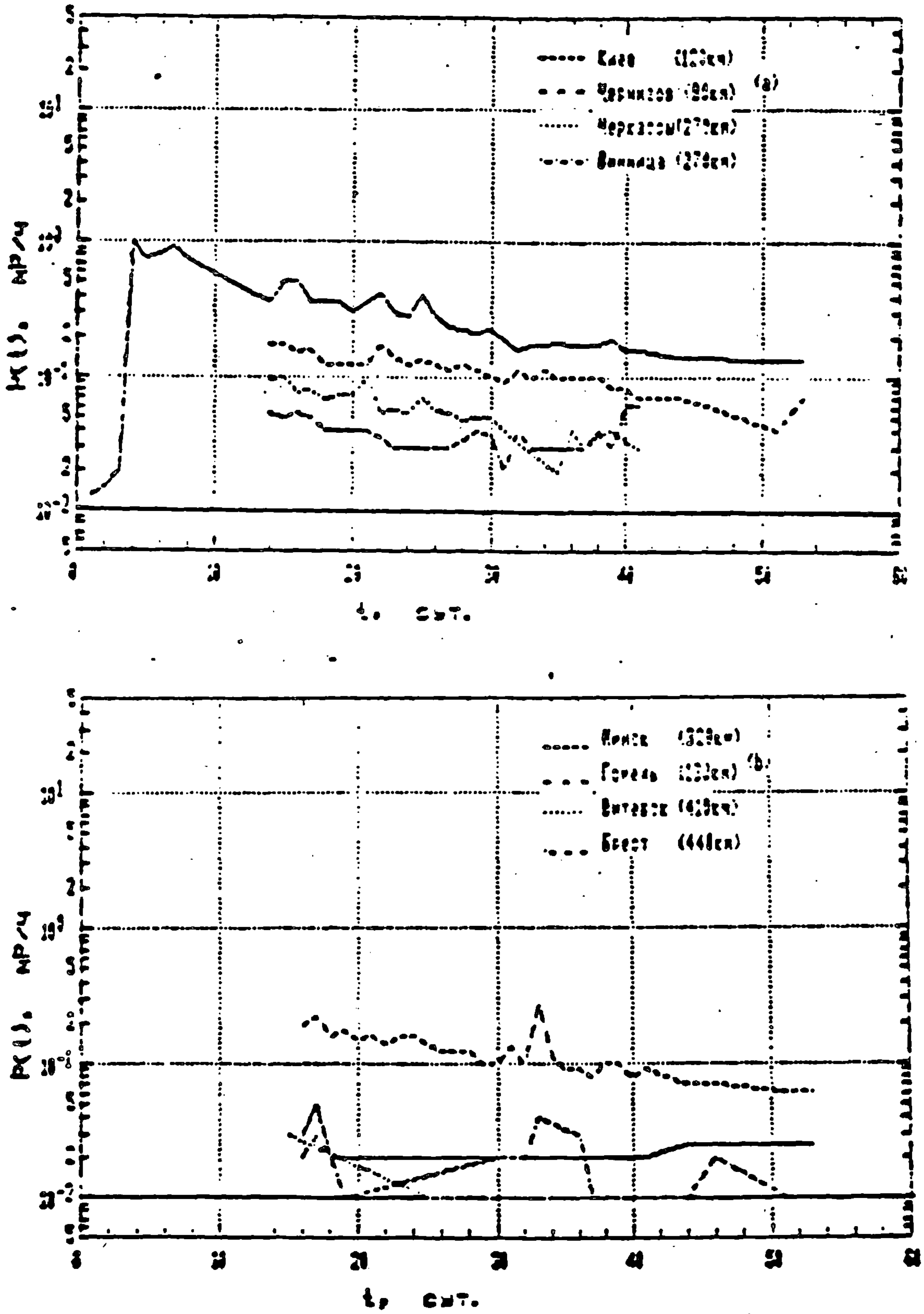


Рис. 7.2.4. Динамика изменения мощности дозы гамма-излучения на открытой местности для областных центров УССР (а) и БССР (в), находящихся вблизи ЧАЭС.

Figure 7.2.4 Dynamics of change in the magnitude of gamma-radiation dosage in open areas for some oblast (county) centers in UkrSSR (a) and RSFSR (b).

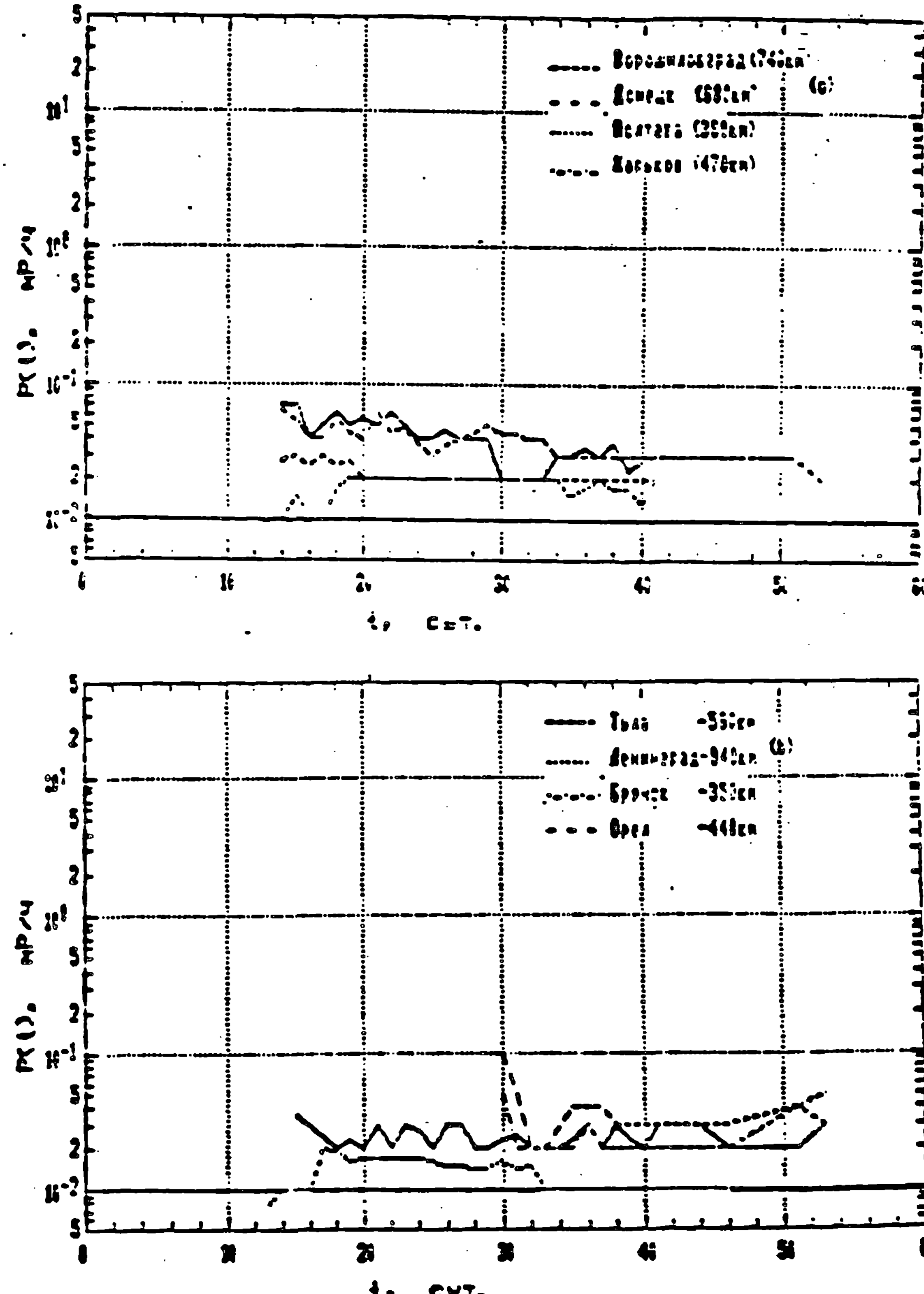


Рис. 22.5. Динамика изменения мощности дозы гамма-излучения на открытой местности для некоторых областных центров УССР (а) и РСФСР (в).

the magnitude dosage of external gamma-radiation exceeded by several times the natural radiation background levels that are typical for this particular zone of European USSR (8-12 mkR/h, thick line in the figures). After averaging numerous measurement results for gamma-radiation dosages in open areas within the administrative boundaries of the oblasts, it was possible to select 10 of them (see table 7.2.5) that had maximum levels of radiation effects of the accidental discharge on the population.

As data in Table 7.2.5 shows, the average values of external radiation of people for the oblasts for 1986 does not exceed (taking into account the activity regimen of the population) 1.5 rem, and 5 rem over 50 years. This confirms the complete lack of danger of the resultant levels of external gamma-radiation from the accidental discharge for the health of the population living outside the 30 km zone around Chernobyl AES. A somewhat more complex situation arises when evaluating internal radiation dosages of people as a result of entry of radionucleides into the human organism together with contaminated, locally produced foodstuffs.

Prior to the Chernobyl accident, in USSR as well as in other countries, only the allowable annual intake of radioactive substances in conjunction with food was rated. An allowable concentration of nucleides in drinking water (NRB-76) was also set. The content of nucleides in individual types of food products was not regulated. In case of an accident, a norm had been set for a critical product (cow's milk), as well as for the important nucleide when an accident occurs -- iodine-131. In the radiation situation that existed prior to the accident, the content levels of strontium-90, cesium-137, and other nucleides in any type of foodstuffs was many times lower than those levels that could reach the established levels of

Table 7.2.5 Radiation levels and predicted values of external radiation dosages in the population in 10 oblasts that have experienced the highest radioactive contamination by the Chernobyl AES discharge products

Oblast (county)	Mean magnitude of radiation dosage for the oblast, "D"+ 15, mr/h	Radiation dosage of the population in 1986, rem		Radiation dosage over 50 years, rem	
		village	city	village	city
Gomel (*)					
Kiev (*)					
Bryansk					
Zhitomir					
Mogilev					
Orlov					
Chernigov					
Tula					
Cherkassk					
Brest					

(*) Outside the 30-km around Chernobyl AES.

Таблица 7.2.5

Уровни радиации и прогнозируемые величины доз внешнего облучения населения в 10 областях, подвергшихся наибольшему радиоактивному загрязнению продуктами аварийного выброса ЧАЭС

Область	Средняя по области мощность дозы на "Д" + 15, мр/час.	Доза облучения населения в 1986г., бэр		Доза облучения за 50 лет, бэр	
		сельское	городское	сельское	городское
Гомельская	0,83 x)	1,39	0,74	4,7	2,5
Киевская	0,44 x)	0,74	0,40	2,5	1,4
Брянская	0,30	0,50	0,27	1,7	0,92
Житомирская	0,20	0,34	0,18	1,2	0,63
Могилевская	0,15	0,25	0,14	0,86	0,46
Орловская	0,14	0,24	0,13	0,81	0,44
Черниговская	0,14	0,28	0,12	0,78	0,42
Тульская	0,12	0,20	0,11	0,67	0,37
Черкасская	0,091	0,15	0,082	0,52	0,28
Брестская	0,081	0,14	0,073	0,46	0,25

x) вне 30-км зоны вокруг ЧАЭС.

of the annual intake of nucleides via food. These norms were based on the fact that individual groups of the population that received the most radiation should not get an annual radiation that exceeded 0.5 rem, while for critical organs in the 2nd group (that included the thyroid in particular) no more than 1.5 rem. In addition it was set that the given dosages of radiation should not be exceeded via any combination of radiation effects, i.e. due to both external and internal radiation (inhalation, intake with food and water). The set norms were calculated for an unconditional and complete prevention of indirect specific radiation consequences (radiation sickness, cataracts, radiation burns, affecting hemogenesis, reduction of immune reaction). In addition, the set norms which had been set substantially below levels that are capable of causing the reactions indicated above are derived from the need to limit the risk of the occurrence of remote radiation effects -- cancer and genetic disorders. For a limited part of the population, Regulation NRB-76 (0.5 rem for the entire body) corresponds to the upper probability limits of the occurrence of cancer at a level of 50-500 additional cases per 1 million population annually. Prior to the accident at Chernobyl AES, the actual radiation levels of the population from food and water were ten times and hundreds of times below the set norms.

After the accident, there was a need for rapid resolution of issues relating to the inspection and prohibition of using specific types of food products. Since at first the principal danger was iodine-131 that came to human beings during the spring-summer period primarily with milk, as well as leafy greens. Regulations were put into effect immediately after the accident concerning the allowable content of iodine-131 in milk and milk products (cottage cheese, sour cream, cheese, butter), as well as in edible leafy greens. Regulations were calculated so that radiation in the thyroid gland of children (critical organ for iodine-131) would not exceed 30 rem. This condition was observed having an allowable content of iodine-131 in milk at a level up to 1 .x10 -7 Curie/l. A similar regulation was introduced in England in 1957 during the Windscale accident. Regulations were also introduced for the allowable content of iodine-131 in meat, fowl, eggs, berries, and medicinal herbs. Data was received in the second half of May which showed that due to the decay of iodine-131, an increasing role in contamination of meat and other types of food was played by cesium-137 and cesium-134; data was also received concerning the presence of rare isotope elements -- cerium-144, ruthenium-106, zirconium-95, barium-140, lanthanum-140, cerium-141, ruthenium-103, niobium-95. To a large part, these along with cesium, were found in edible greens (1 X10 -6 Curie/kg and higher). During the entire month of May high concentrations of iodine remained in milk products in many places. During this period, in order to carry out large scale efforts to control and inspect food products, there was need to set regulations that allowed monitoring with the use of the most uncomplicated equipment, i.e. regulate the total content of beta-activity. These norms were confirmed by USSR Minzdrav on 30 May 1986. They supersede the earlier regulations of 8 and 12 May, and list a broader number of products, reflecting the changes in the radiation situation that were in effect by the end of May. The allowable radiation dosage of the entire body and the internal organs for which the norms have been calculated is 5 rem.

In the first days and weeks after the accident, the principal activity in food products was exemplified by the presence of iodine-131. It appeared in the the milk of cows that were kept in pasture 2-3 days after the accident. In the south of Belorussia its level reached 10 -6 Curie/l. The milk of cows kept in stalls remained considerable more pure. Similar and even higher levels of contamination, up to 10 -5 Curie/l were noted in leafy plants.

As we know, during the migration of radionucleides, from the first link, i.e. the fallout itself and the soil, and until the last link, the human organism, depending on the physico-chemical properties of the nucleide and a number of other factors (soil composition, amount and duration of atmospheric precipitation, the composition of the farm animal diet and so forth), the radionucleides separate, reducing the content of some nucleides and accumulating others. For this reason, a most complete composition of nucleides may be recorded in food products that had been contaminated from the surface, i.e. those that directly absorb nucleides contained in the atmosphere and which settle from the air. Such products include lettuce, dill, coriander, tea and so forth. As an example, table 7.2.6 shows the composition of nucleides discovered in representative plants from areas close to the Chernobyl AES, and contaminated from the surface.

Conversely, in a number of other products, to which radioactive substances arrive through biologic barriers, the composition of radionucleides is substantively reduced. Thus, in the beginning and middle of May, only cesium and iodine-131 isotopes were discovered in meat, while by the end of May and in June, practically only cesium-137 and cesium-134 (at a ratio of 2:1). Also, the content of radioactive cesium in meat ~~(beef)~~ (beef) is sufficiently high, at a level of 10 -8 to 10 -7 Curie/kg.

The exceeded levels in food that were in effect in May regulations are shown by data in Figure 7.2.7.

Table 7.2.6 Content of radionucleides in certain plant products in the vicinity of the Chernobyl AES

Name	Sample taken at	Date taken	Nucleide	Content, Curie/kg
Clover	Chernobyl	26 May	Cerium-144	
			Cerium-141	
			Iodine-131	
			Ruthenium-103	
			Ruthenium-106	
			Barium-140	
			Cesium-134	
			Cesium-137	
			Zirconium-95	
			Niobium-95	
			Lanthanum-140	

Таблица 7.2.6

Содержание радионуклидов в некоторых растительных продуктах вблизи ЧАЭС

Наименование	Место отбора	Дата отбора	Нуклид	Содержание Ки/кг
Клевер	Чернобыль	26 мая	Церий-144	$2 \cdot 10^{-6}$
			Церий-141	$1,4 \cdot 10^{-6}$
			Йод-131	$1,3 \cdot 10^{-6}$
			Рутений-103	$1,2 \cdot 10^{-6}$
			Рутений-106	$7,9 \cdot 10^{-7}$
			Барий-140	$6,7 \cdot 10^{-7}$
			Цезий-134	$3,2 \cdot 10^{-7}$
			Цезий-137	$2,5 \cdot 10^{-7}$
			Цирконий-95	$1,5 \cdot 10^{-6}$
			Ниобий-95	$2,0 \cdot 10^{-6}$
			Лантан-140	$5,3 \cdot 10^{-7}$

Table 7.2.7 Agricultural products in which radioactive contamination was discovered that exceeded allowable levels.

Republic	Oblast	Product and its share (%), in excess of regulations					
		Meat	Milk and milk products	Edible greens	Veget-ables	Berries	Fish
Belorussian SSR	Minsk						
	Gomel						
	Brest						
	Mogilev						
	Grodno						
RSFS	Tula						
	Bryansk						
	Kaluga						
	Kursk						
	Orlov						
Ukrainian SSR	Kiev						

NOTE: Dash "-" indicates absence of data.

Таблица 7.2.7

Для сельскохозяйственной продукции, в которой было обнаружено превышение допустимого радиоактивного загрязнения

Республика	Область	Продукт и его доля (%), несоответствующая нормам					
		мясо	Молоко и молокопродукты	зелень столовая	овощи	ягоды	рыба
Белорусская ССР	Минская	10	5	-	-	-	-
	Гомельская	40	30	15	10	5	90
	Брестская	10	50	5	3	5	-
	Могилевская	20	10.	-	-	-	-
	Гродненская	-	5	-	-	-	-
РСФСР	Тульская	-	15	-	-	-	-
	Брянская	-	30	-	-	-	-
	Калужская	-	20	-	-	-	-
	Курская	-	30	-	-	-	-
	Орловская	-	10	-	-	-	-
Украинская ССР	Киевская	-	10	20	-	20	-

Примечание: "-" - данные отсутствуют.

As was stated above, monitoring of milk contamination by iodine-131 indicated that in many areas of USSR, BSSR and RSFSR in May-June 1986, the concentration of this nucleide in milk exceeded the set standards (0.1 mk Curie/l). Having analyzed the levels of soil, plant, and milk sample contamination and having tied them to the magnitude of the gamma-radiation dosage in the locality, it was possible to evaluate the potential concentrations of iodine-131 in milk in the various regions of the country. Such an example is given in Table 7.2.8., where reported levels of milk sample contamination by iodine-131 are shown for 10 oblasts, based on data for the mean magnitudes of external gamma-radiation dosage for the oblast, in comparison with magnitudes actually noted in May 1986. Information for the other regions of European USSR, with a population of about 75 million (see Figure 7.2.6) were similarly handled. Figure 7.2.6 and the table given below show 11 regions (4 on the territory of UkrSSR, 2 in BSSR, and 5 in RSFSR) that were of interest either from the point of view of sufficiently high levels of radioactive contamination, or from the point of view of the large population living in them.

Evaluation of radioactive consequences for the population of these regions from external gamma-radiation from radionucleides that fell in the area is shown in Table 7.2.9. As the table shows, the expected mean values for the regions of the external radiation dose for humans in 1986 was generally below the annual dosage limit for individuals in category B (limited part of the population), in accordance with NRB-76. The collective radiation dose for this part of the country's population will make up 8.6 million person/rem in 1986, and during the 50 year period after the accident, 29 million person/rems. For comparison, let us note that the annual aggregate

Table 7.2.8 Comparison of estimated and actually recorded levels of milk contamination by iodine-131 in May 1986 in 10 oblasts that experienced the highest amount of radioactive contamination due to the accidental discharge of the Chernobyl AES, mkC/l.

Area	Estimated levels	Actual measurements
Gomel		
Kiev		
Bryansk		
Zhitomir		
Mogilev		
Orlov		
Chernigov		
Tula		
Cherkassk		
Brest		

Таблица 7.2.8

Сравнение расчётных и фактически наблюдаемых уровней загрязнения молока йодом-131 в мае 1986г. в 10 областях, подвергшихся наибольшему радиоактивному загрязнению продуктами аварийного выброса ЧАЭС, мкКи/л

Области	Расчётные уровни	Реальные измерения
Гомельская	0,2-14	0,02-10
Киевская	0,06-7,3	
Брянская	0,04-5,0	0,02-1,3
Житомирская	0,03-3,3	
Могилевская	0,02-2,5	0,02-2,0
Орловская	0,02-2,3	0,01-0,8
Черниговская	0,02-2,3	
Тульская	0,02-2,0	0,06-6;5
Черкасская	0,01-1,5	
Брестская	0,01-1,3	0,2-9,0

Figure 7.2.6 Population numbers in separate regions of European USSR (in millions)

1. City population
2. Village population
3. Ukraine
4. Belorussia
5. Central
6. Western
7. Eastern
8. ~~Western~~ Southern
9. South-eastern
10. ~~South~~ North-western
11. Moldavia
12. Bryansk Oblast
13. Kalinin Oblast
14. Kaluga, Smolensk,Tula Oblasts
15. Orlov, Kursk, Lipetsk Oblasts.

Рис. 7.2.6 Численность населения в отдельных регионах Европейской части СССР

(в млн. чел.)

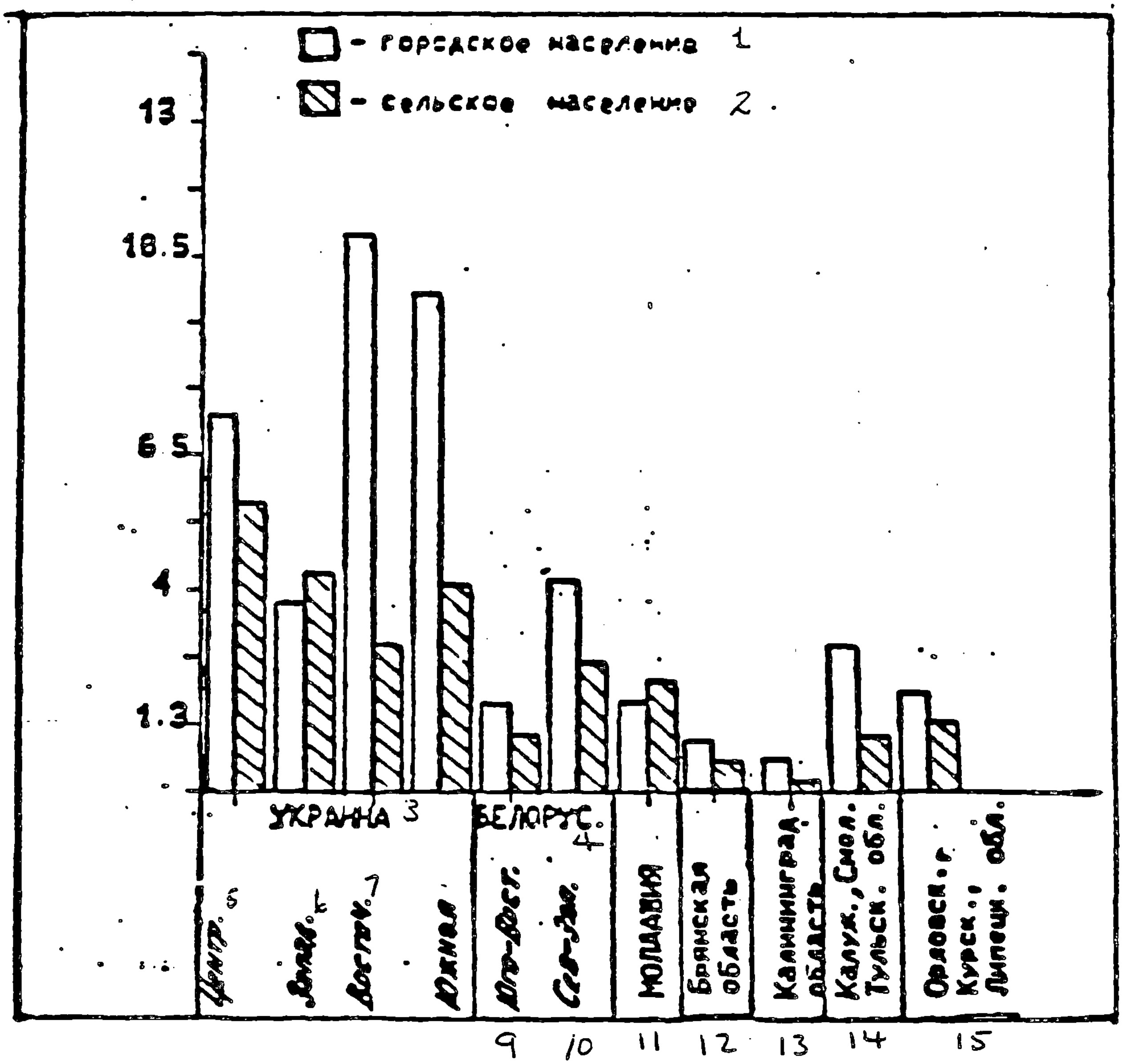


Table 7.2.9 Expected dosages of external radiation for population in individual regions of European USSR

Region	Population, in millions	Dosage for 1986, rem/year		Aggregate dosage, 10 6 person/rems	
		Village	City	for 1986	over 50 yrs
Central UkrSSR					
Western UkrSSR					
Eastern UkrSSR					
South UkrSSR					
S-E BSSR					
N-W BSSR					
Moldavian SSR					
Bryansk Oblast					
Bryansk Oblast					
Kaliningrad Obl.					
Kalinin, Tula, Smolensk Obl.					
Orlov, Kursk, Lipetsk Obl.					
Total:					

Таблица 7.2.9

Ожидаемые дозы внешнего облучения населения отдельных регионов Европейской части СССР

Регион	Численность населен. млн.чел.	Доза за 1986г. бэр/год		Коллективная доза, 10^6 чел.бэр	
		сельское	городское	за 1986 год	за 50 лет
Центр.часть УССР	13,6	0,27	0,15	2,75	9,31
Запад.часть УССР	8,3	0,067	0,036	0,44	1,47
Восточ.часть УССР	14,5	0,077	0,041	0,75	2,52
Южн.часть УССР	14,4	0,045	0,024	0,73	2,47
Юго-Восточ.БССР	2,9	0,98	0,52	2,05	6,94
Сев.-Запад.БССР	7,0	0,094	0,050	0,47	1,58
Молдавская ССР	4,1	0,084	0,045	0,27	0,92
Брянская обл.	1,5	0,50	0,27	0,44	1,49
Калининград.обл.	0,8	0,012	0,003	0,006	0,02
Кал.,Тульск, Смоленск.обл.	4,0	0,12	0,064	0,32	1,08
Орл.,Курск., Липецк.обл.	3,4	0,14	0,075	0,35	1,17
Всего:	74,5	-	-	8,6	29,0

dosage from natural background radiation for the given number of people will comprise 10 million person/rems, i.e. it is comparable to the annual dosage from the accidental discharge in 1986. In 50 years the dosage from natural background will be almost 15 times higher than a corresponding dosage from the Chernobyl AES accident. As a result, the calculated figures, based on the concept of linear limitless relationship "dosage-effect", an increase in the number of instances of additional deaths from cancer will make up less than 0.05 percent in relation to the death rate due to spontaneously arising cancer (about 9.5 million cases over 70 years) among the given population segment.

Risk of death and estimates of morbidity cases involving curable types of cancer and non-malignant growth in the thyroid among people who consumed milk contaminated by iodine-131 was estimated on the basis of the following original data:

-- actual concentration of iodine-131 in cow's milk or estimates of its content, evaluated according to the magnitude of the gamma-radiation dosage on the site;

-- population number, with sex and age breakdown;

-- age relationship of milk consumption, dosage coefficients and death risk coefficients, risk of curable case of thyroid glands on the basis of MKRZ data, NK DAR OON, and materials by Soviet authors.

Generalized data for concentration levels of iodine-131 in cow's milk for the regions under question are shown in Figure 7.2.7. The figure shows that in a number of oblasts in UkrSSR, BSSR and RSFSR, the concentration of iodine-131 in individual samples exceeded the set standards by 20-100 times and even higher (thick line in the figure). Due to the fact that milk sold through the centralized system had a concentration of no higher than 0.1 mkC/l, it was decided that such milk can be used by the entire city populations of these areas and the major part of the village population. For the remaining small segment of village

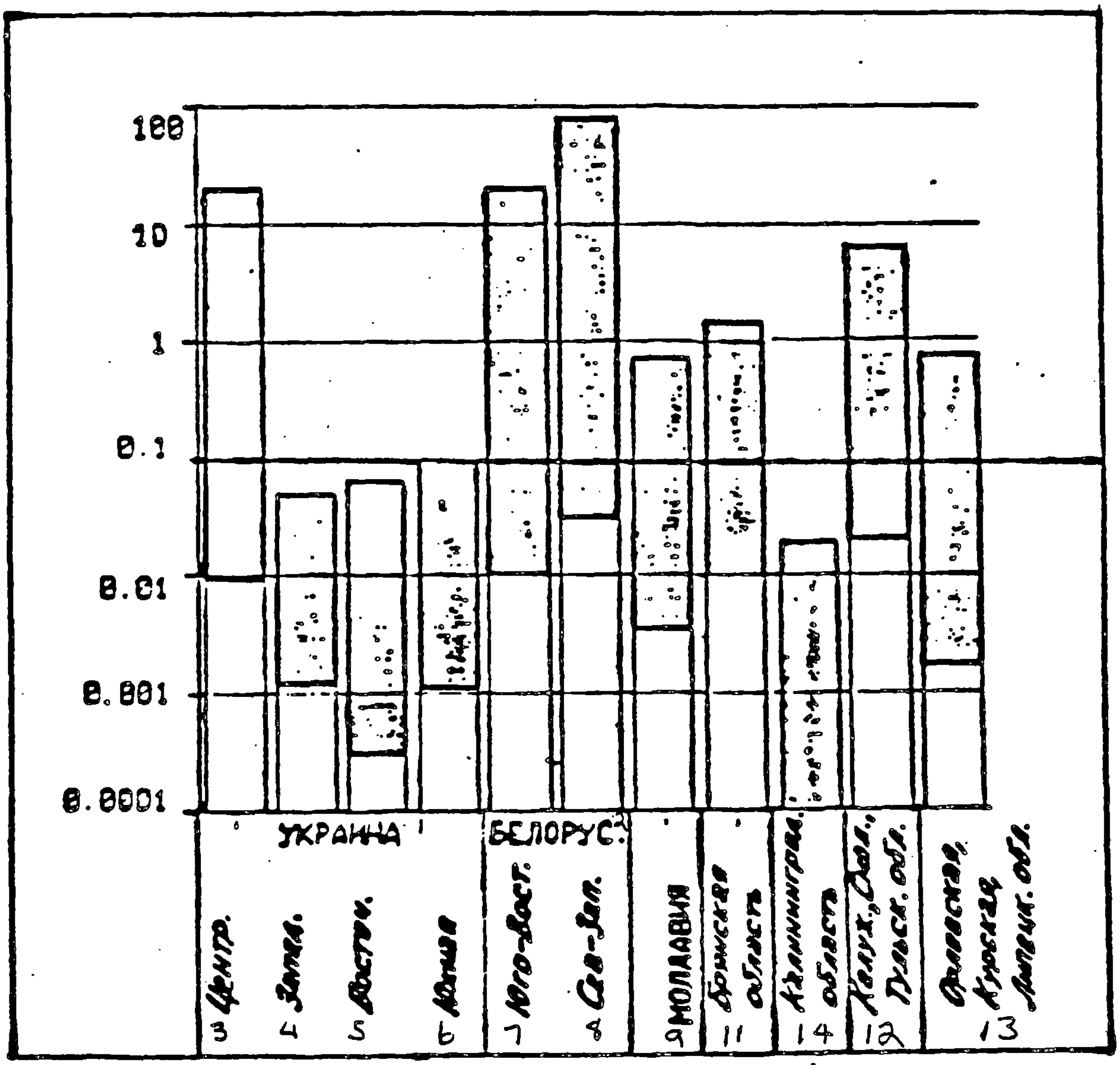
Figure 7.2.7. Concentration of Iodine-131 in cow's milk (in mkC/l).

(TYPIST): See numbers on orig. page).

1. Ukraine
2. Belorussia
3. Central
4. West
5. East
6. South
7. South-east
8. Northwest
9. Moldavia

11. Bryansk Oblast
12. Kaluga, Smolensk,
    Tula Oblasts
13. Orlov, Kursk,
    Lipetsk Oblasts
14 Kaliningrad Oblast

Рис. 7.2.7. Концентрация ^{131}I
в молоке коров
(в мкКи/л.)



village population it was presumed that in some cases, where consumption of milk having an iodine-131 concentration in excess of the allowable levels, it was not possible to implement this. Such an assumption made it possible to conclude that in a number of the areas more highly contaminated by iodine-131, the maximum estimated dosages of internal radiation of the thyroid in humans could reach hundreds of rads.

Comparison of the extent of death risk due to radiation received in the thyroid, which was conducted on the basis of estimates of individual and aggregate human radiation dosages, with the risk occurring over 30 years after entry of iodine-131 into the human organism, and the risk of death due to spontaneously occurring thyroid cancers over the same period of time (about 150,000 cases), this comparison showed that additional deaths from iodine-131 makes up about 1 percent and practically does not increase death rate indicators in the regions under examination.

The given estimates are based on the concept of the limitless linear relationship "dosage-effect", a concept that is accepted by a majority of countries in the world. This concept is based on the theoretical understanding of the carcenogenesis mechanism, on the collective data for the "dosage-effect" concept for instances with higher radiation dosages, as well as on the principle of making decisions for the sake of human beings, i.e. the inalienable assurance of man's safety in the area of small dose radiation. It is no accident that the lead MKRZ publication on the problems of radiation safety (MKRZ publication 26, No 30) indicates:"Use of the linear extrapolation method, based on data for the frequency of effects occurring at high dosages makes it possible to evaluate the maximal risk... However, the more careful the assumption concerning the linearity, the more necessary to take into account the fact that this may result in the overestimation of radiation risks..." Thus, the magnitudes shown in the given section of the report should be viewed as the "upper" estimates for radiation consequences among the population of the European USSR, located within the accidental discharge activity from the Chernobyl AES.

In addition to iodine-131 in the current year, and in particular in subsequent years, it is necessary to make note of other radionucleides that contaminate locally produced food products and water supply sources. Potential levels of food contamination in the near-term and long-term should be viewed separately for the principal nucleides and types of food products.

Ruthenium-106, cerium-144 and other rare nucleides make a noticeable impact only in products that can be surface-contaminated (leafy greens, vegetables, and to a lesser degree, berries, mushrooms and honey), since these nucleides are practically not assimilated from soil to plants, and from plants to humans. The biological significance of this entire group of radionucleides in such products does not exceed 10-20 percent. The subsequent role of rare elements in the contamination of foodstuffs will universally make a substantial drop and will have no practical significance.

Cesium-137 and cesium-134 are the main biologically significant radionucleides which (not counting strontium-90) have been the main contaminators of meat, milk, vegetables and other products since the middle of June. The source for the contamination to-date of plant and animal products has been the air. The grain and potato crops in fall 1986 can be expected to be relatively clean; there will not be a great deal of cesium by air, and soil contamination will not be able to play a role as yet.

Contamination of products by cesium in subsequent years will be substantially different for areas that lie near the Chernobyl AES, where the soil type is different. Since from the Polessie soils (alluvial plain), which are poor in humus, cesium enters plants 10 times and even

100 times higher than from other types of soils, the Polessie regions can expect relatively stable and high levels (almost at the current) of cesium-137 in food products in subsequent years.

Preliminary, purely speculative estimates of the contamination levels of food products by cesium isotopes show the following: having a radioactive fallout density for cesium-137 on the earth's surface equal to 1 Curie per square kilometer and with the consumption of locally produced foodstuffs, the magnitudes of individual radiation dosages for the entire body as a result of peroral ingestion by the organism in the area of Ukrainian and Belorussian Polessie will be (taking into account additional radiation of the organism by cesium-134) 0.70, 0.34, and 3.3 rem for the first and second years, and 70 years respectively. In this instance the aggregate human radiation dosages, taking into account the agricultural products actually produced on 1 square kilometer on the territory of these regions in UkrSSR and BSSR, the dosages for the same periods of time will be 120, 58, and 570 person/rems. For the other territories of the nation, with considerably lower levels of cesium transmission from soil to agricultural products, the corresponding magnitudes of aggregate radiation dosages for the population will be 120, 36, and 170 person/rems. Taking into account that the overall amount of cesium-137 that was discharged into the atmosphere and fell on the earth surface after the Chernobyl AES accident is estimated to be $1.0 \cdot 10^{6}$ Curries (see sections 4 and 5), and taking into account that 10% of the discharged cesium isotopes fell on the Ukrainian and Belorussian Polessie, the aggregate dosage of radiation for the population for a 70 year period after the accident will make up 2.1. 10 person/rems. This may result in an additional death rate from cancer that does not exceed 0.4% of the natural death rate from malignant growths. Further data for the actual coefficients for the transmission of cesium along the food chain, under the specific conditions of the contaminated regions will make it possible to make corrections in the given estimates, and to reduce given figures.

To-date, the data concerning the content of strontium-90 in food is limited, and insufficient for making corrective data estimates. In time, it is possible that this nucleide will be significant, along with cesium-137. The content of strontium-90 for products in diet from the 1987 harvest will apparently be reduced overall. In the area of the Polessie soils, the reduction in the role of strontium-90 in comparison to cesium-137 will be more significant. However, predictions concerning its content in food in subsequent years will be possible only after a complete study of the content of strontium-90 in contaminated areas of RSFSR, UkrSSR, and BSSR.

Thus, it will be possible to make a correct estimate of the dosages per population that consumes locally grown food products contaminated by cesium-137 and strontium-90 only after establishing actual coefficients for radionucleide transfer along the food chains for the given regions. These efforts, which are being activated by various scientific sub-branches of the nation will make it possible to develop recommendations for the most optimal methods, from the point of setting dosage loads per person, to conduct agricultural work in the areas that are contaminated by radioactivity.

7.3 ORGANIZATION FOR CARRYING OUT MEDICAL OBSERVATIONS OF THE POPULATION AROUND THE CHERNOBYL AES REGIONS.

Following the accident, 84,000 people were evacuated from the city of Pripyat' and the Chernobyl area, which included 18,350 children. In addition, in a number of populated areas in the Kiev and Zhitomir oblasts, additional evacuations were carried out.

In order to provide medical assistance to the evacuees during the first days after the accident, 450 brigades that included doctors, nurses, laboratory technicians and dosimetrists were called upon, and who were provided transport vehicles. Overall (taking into account the changes depending on the radiation situation) 1,240 doctors, 920 nurses, 360 doctors-lab technicians and 2,720 lab technicians having a middle-school education, 720 medical school upperclassmen, as well as a large group of associations from NII (National Research Institute) were called to serve.

After (initial) health processing, all evacuees were examined by doctors with a mandatory dosimetric monitoring and laboratory blood test. When necessary, the tests were repeated.

All persons evacuated from the 30 km zones, when it was necessary to evaluate any deviations in the state of health, were hospitalized in special units that were created at central regional hospitals.

In order to provide medical care for workers that were taking part in eliminating the after effects of the accident, a polyclinic with

4 around-the-clock first aid brigades was set up in the city of Chernobyl, at the central regional hospital.

Special attention was given to the examination of children from the 30 km zones, as well as random examination of children living in populated areas that adjoined the 30 km zone (about 100,000 children were examined in all).

7.4 LONG-TERM PROGRAMS FOR MEDICAL AND BIOLOGIC MONITORIC OF THE POPULATION AND PERSONNEL

Long-term programs are being set up to conduct medical and biologic observations of the population and personnel.

The set of measures intended to provide medical care to persons who have experienced radiation effects as a result of the Chernobyl AES includes: -- compiling a list of all persons who have experienced the effects; -- grouping those who have experienced radiation in order to determine the volume of required medical assistance; -- measures for organizing and carrying out medical assistance at the necessary volume.

The purpose of the list was to study the potential after effects of radiation on the entire group of persons affected in order to provide purposeful medical observation, one that meets the expected effects for the given range of dosages.

It is planned to analyze the effects of low dosage overall external radiation using stochastic effect criteria (infectious morbidity, morbidity and death rate due to malignant growths, birth rate, condition of the newborns), and neuro-psychiatric aspects of the reaction to the situation.

Special analyses will be made of the function of the thyroid gland, and in the long-term frequency of adenomas and malignant growths will be studied.

All study efforts will be conducted on the basis of the dynamic characteristics of the background level of the parameters indicated above within the regions where those under observations originate, as well as the evacuation sites. In determining the volume of observations, the intention is to base it on international and domestic recommendations concerning potential biologic effects (NKDAR, MKRZ, etc.).

Serving as criteria for monitoring the general state of health will be data from examinations by therapists and the expanded clinical analysis of blood. All women will examined by a gynecologist, and children by a pediatrician, including data concerning physical development, in accordance with regulations issues in USSR. Within the dosage range that presumes even a minimal risk of thyroid gland disfunction, this examination will be supplemented by special dynamic observation of the thyroid gland by an endocrinologist, and with the use of hormones, thyroxine, triothyroxine, thyreotropic hormones, etc.

Taking into account the risk levels for death rate from cancer as a result of radiation effects, it will be also possible to evaluate the increase in frequency due to radiation (from fractions of a percent to several percent) only by having data for a very large quantitative population sample.

As a result of the above, all persons who currently live, who came temporarily, the organized units who were called to work at the accident site, and subsequently their children and grandchildren, and persons who were evacuated from the contaminated regions are subject to registration.

In order to develop this registration list, plans have been made for registration and dosimetric charts, which will be filled out for each person under observation.

The registration map includes the following information: last name, first name, patronimic, passport data (series and number of passport, or birth certificate, date document was issued); date of birth, place of birth, sex, nationality, place of residence, where located during the exposure period, duration of exposure, anamnesistic data concerning the state of health, pregnancy at the start of the exposure (time pregnant in weeks), pregnancy occurring after the start of the exposure, data for the child at the conclusion of pregnancy, causes of death (adults, children, newborns), measures taken (hospitalization, iodine treatment).

The dosimetric chart will record the healthful character of the region and the degree of radiation effects on man (contamination of clothing, shoes, skin surfaces prior to and after deactivation, in mkR/h).

The chart will include data concerning iodine-131 content in the thyroid gland, which is the dosimetric parameter for clinical examination of persons undergoing a check, as well as data on individual dosimetry (measurement of biosubstrata, using a SICh (not further identified) and other instruments.

The registration and dosimetric charts will be filled out by local health care authorities. The filled out charts will be sent to the Ministries of Health of the republics and USSR Minzdrav. Parallel to filling out the registration charts, all data will be entered in a registration journal which will be permanently kept at the observation locale.

Grouping of persons exposed (or who could be exposed) as a result of the accident at the Chernobyl AES on 26.04.86, and who require the appropriate medical assistance will be based according to dosage level for the entire body and the thyroid gland. All dosages shown are for adults;

for children up to three years of age, as well as for pregnant women, must be reduced by a factor of ten, as compared to adults.

Frequence of examinations will be determined on the basis of initial examination results and evaluation of the dosage level. Preventive and protection measures (iodine treatment, evacuation, limited intake of radioactive substances by the organism via inhalation and ingestion) are being considered.

The extent of these examinations will be supplemental to the dispensary clinic observation done for the entire population of the country.

The program anticipates increasing the necessary number of specialists from different professional branches. The outlay of time, technical equipment and algoriths, software and calculations using the computer will be evaluated in carrying out the set tasks. Clinical data will be interpreted in the light of materials for the dynamics of environmental contamination, the characteristics of the isotopic composition, and the conduct of iodine treatments. It is anticipated that simulation models and research forecasts will be developed for the expected variants of remote effects of a stochastic nature (oncologic effects, genetic effects) for the next 30 years and throughout a lifetime (50 years).

This type of an emergency situation presents an opportunity for the prospect of collecting and accumulating data with respect to oncologic and genetic aspects of morbidity of the population.

The programs under preparation will take into account the experiences of other nations (the Three Mile Island program; the MAGATE conference in Yugoslavia and others).