JPRS-TND-86-009-L

7 OCTOBER 1986

Worldwide Report

NUCLEAR DEVELOPMENT AND PROLIFERATION

USSR STATE COMMITTEE REPORT

ON CHERNOBYL NUCLEAR POWER PLANT ACCIDENT

FBIS FOREIGN BROADCAST INFORMATION SERVICE

JPRS-TND-86-009-L
7 OCTOBER 1986

WORLDWIDE REPORT

NUCLEAR DEVELOPMENT AND PROLIFERATION

USSR STATE COMMITTEE REPORT ON CHERNOBYL NUCLEAR POWER PLANT ACCIDENT

Vienna THE ACCIDENT AT THE CHERNOBYL NUCLEAR POWER PLANT AND ITS CONSEQUENCES in English 25-29 Aug 86

['Working Document' compiled by USSR State Committee on the Utilization of Atomic Energy for the IAEA Experts' Post Accident Review Meeting of 25-29 Aug 1986 in Vienna, consisting of Part I, General Material and Part II, Annexes.]

[This document and annexes received with no translation attribution. For related material see JPRS-TND-86-004-L of 5 June 1986 and JPRS-UPS-86-038 of 6 August 1986]

CONTENTS

/6091

PART I. GENERAL MATERIAL

PREFACE

This material is taken from the conclusions of the Government Commission on the causes of the accident at the fourth unit of the Chernobyl' nuclear power plant and was prepared by a team of experts appointed by the USSR State Committee on the Utilization of Atomic Energy. The members of this team were:

Abagyan A.A.	Mysenkov A.I.
Asmolov V.G.	Pavlovskij O.A.
Gus'kova A.K.	Petrov V.N.
Demin V.F.	Pikalov V.K.
Il'in L.A.	Protsenko A.N.
Izraehl' Yu.A.	Ryazantsev E.P.
Kalugin A.K.	Sivintsev Yu.V.
Konviz V.S.	Sukhoruchkin V.K.
Kuz'min I.I.	Tokarenko V.F.
Kunzevich A.D.	Khrulev A.A.
Legasov V.A.	Shakh O.Ya.
Malkin S.D.	

During preparation of this document, material was used from the following organizations: the I.V. Kurchatov Institute of Atomic Energy, the Scientific Research and Design Institute for Power Technology, the V.G. Khlopin Radium Institute, the S.Ya. Zhuk 'Hydroproject' Institute, the All-Union Scientific Research Institute for Nuclear Power Plants, the Institute of Biophysics, the Institute of Applied Geophysics, the USSR State Committee on the Utilization of Atomic Energy, the USSR State Committee on Hydrometerology and Environmental Protection, the Ministry of Health, the USSR State Nuclear Power Supervisory Board, the Ministry of Defence, the Main Fire Protection Directorate of the Ministry of Internal Affairs and the USSR Academy of Sciences.

INTRODUCTION

On 26 April 1986 at 1.23 a.m. an accident occurred at the fourth unit of the Chernobyl' Nuclear Power Plant which resulted in the destruction of the reactor core and part of the building in which it was housed.

The accident took place prior to shut-down of the unit for planned maintenance during operating mode tests on one of the turbogenerators. There was a sudden power surge on the reactor leading to the destruction of the reactor and the release into the atmosphere of part of the radioactive products which had accumulated in the core.

During the accident the nuclear reaction in the fourth unit was stopped. The fire which occurred was extinguished and work was begun to limit and eliminate the consequences of the accident.

The population from the areas in the immediate vicinity of the nuclear power plant and from a 30 km-radius zone around the plant was evacuated.

In view of the extraordinary nature of the Chernobyl' accident, an operational team headed by the President of the USSR Council of Ministers, N.I. Ryzhkov, was organized in the Politburo of the CPSU Central Committee to co-ordinate the activities carried out by the ministries and other state departments to eliminate the consequences of the accident and to assist the population. A government commission was set up to study the causes of the accident and to implement the requisite emergency and rehabilitation measures. The necessary scientific, technical and economic resources of the Soviet Union were mobilized.

Representatives of the IAEA were invited to the USSR and given the opportunity to familiarize themselves with the state of affairs at the Chernobyl' power plant and the measures taken to control the accident. They informed the world community of their evaluation.

Governments of a number of countries and many governmental, public and private organizations and individuals from different countries made offers of assistance to various Soviet organizations to help eliminate the consequences of the accident. Some of these offers were accepted.

During the thirty years of its development, nuclear power has occupied an important place in world energy-production and on the whole has demonstrated a very good record of safety for mankind and the environment. It is impossible to envisage the future of the world economy without nuclear power. However, its further development must be accompanied by still greater scientific and technical efforts to guarantee operational reliability and safety.

The Chernobyl' accident resulted from a combination of several unlikely events. The Soviet Union is drawing the appropriate conclusions from the accident.

Abandonment of nuclear energy sources would require a significant increase in the extraction and consumption of organic fuels. This would undoubtedly increase the risk of disease for mankind, and increase the destruction of waters and forests as a result of the constant release of harmful chemical substances into the biosphere.

In addition to its advantages as a source of energy and as a means of conserving natural resources, the world-wide development of nuclear power also has inherent dangers of an international character. These include transboundary transfers of radioactivity, particularly in the event of serious radiation accidents, and the proliferaton of nuclear weapons, the danger of international terrorism and the specific danger represented by nuclear facilities in the event of war. All this emphasizes the crucial need for close international co-operation in the development of nuclear power and in ensuring its safety.

That is the reality of the situation.

The fact that the contemporary world is full of potentially dangerous industrial production processes significantly aggravating the consequences of military actions gives a new perspective to the senselessness and inadmissibility of war in today's world.

In his speech on Soviet television on 14 May, M.S. Gorbachev said: "For us the indisputable lesson of Chernobyl' is that, with the further development of the scientific and technical revolution, questions of the reliability and safety of technology, questions of discipline, order and organization acquire paramount importance. The strictest possible requirements will have to be applied everywhere and to everything.

"Furthermore we consider that co-operation within the International Atomic Energy Agency should be further enhanced."

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

1.1. Design data

The design power output of the Chernobyl' nuclear power plant is 6 GW; as of 1 January 1986 the power of the four operating units of the station was 4 GW. The third and fourth units belong to the second construction stage of the Chernobyl' nuclear power plant and to the second generation of plants of that type.

1.2. Description of the reactor in the fourth unit of the Chernobyl' nuclear power plant

The chief design features of RBMK reactors are the following:

(1) Vertical channels containing the fuel and coolant, enabling local refuelling while the reactor is in operation;

(2) Fuel in the form of bundles of cylindrical fuel elements made of uranium dioxide in zirconium tube-type cladding;

(3) A graphite moderator between the channels;

(4) A boiling light-water coolant in the multiple forced circulation circuit (MFCC), with direct steam feed to the turbine.

These design features, as a group, determine all the main characteristics of the reactor and the nuclear power plant both as regards its merits, which include: the absence of cumbersome pressure vessels which are difficult to manufacture and limit the reactor's unit power and production base; absence of a complex and costly steam generator; the possibility of continuous refuelling and a good neutron balance; a flexible fuel cycle easily adapted to the fluctuations of the fuel market; the possibility of nuclear steam superheating; high thermal reliability and durability of the reactor through channel-by-channel control of coolant flow, channel failure detection, monitoring of the parameters and coolant activity in each channel and on-load replacement of leaking assemblies; and as regards its shortcomings: the possibility that there may be a positive void coefficient of reactivity due to the presence of a phase transition in the coolant, which governs the behaviour of the neutron-flux-determined power during accidents; high sensitivity of the neutron field to reactivity perturbations of different kinds, requiring a complicated control system to stabilize the power density distribution in the core; complex branching of the coolant delivery and removal system for each channel; a large amount of heat energy accumulating in the metal structures, fuel elements and graphite structure; and slightly radioactive steam in the turbine.

The RBMK-1000 reactor with a power output of 3200 MW(th) (Fig. 1) is equipped with two identical cooling loops; to each loop are joined 840 parallel vertical channels containing the fuel assemblies.

The cooling loop has four parallel main circulation pumps (three of which are operational and feed 7000 t/h of water at a pressure of ~1.5 MPa, while one is redundant).

The water in the channels is heated to boiling point and partially evaporates. The steam-water mixture with a mean mass steam quality of 14% is led off through the top of the channel and steam-water communication line to two horizontal gravity-type separators. The dry steam (less than 0.1% moisture content) separated in them is fed at a pressure of 7 MPa from each separator via two steam pipes to two turbines with an output of 500 MW(e) each, (all eight steam pipes of the four separators are joined in a common "ring"), while the water, after mixing with the steam condensate, is fed through 12 downcomers to the section header of the main circulation pumps.

The condensate from the spent steam in the turbines is recycled through the separators by feed pumps to the top of the downcomers, thereby subcooling the water to saturation temperature at the main circulation pump inlet.

As a whole, the reactor consists of a set of vertical fuel and coolant channels inserted into cylindrical openings in the graphite columns, and in the top and bottom shielding plates. A light cylindrical cowling encloses the space occupied by the graphite structure.

This structure consists of graphite blocks assembled in the form of columns, with a square cross-section and cylindrical axial openings. It rests on a bottom plate, which transmits the weight of the reactor to a concrete vault.

About 5% of the reactor power is released in the graphite through the slowing-down of neutrons and absorption of gamma-quanta. To reduce thermal resistance and prevent oxidation of the graphite, the cavity in the stack is filled with a slowly circulating mixture of helium and nitrogen, which serves at the same time to monitor the integrity of the channels on the basis of variations in moisture content and temperature of the gas.

Below the bottom plate and above the top plate there are spaces for laying the coolant pipes along the routes from the drum separators and distributing headers to each channel.

The robot, i.e. the refuelling machine, after removal of the relevant section of floor and lining up with the channel co-ordinates, couples onto the head of the channel, equalizes its own pressure and the channel pressure, unseals the channel, removes the burnt-up fuel assembly and replaces it with a fresh one, reseals the channel, uncouples and transports the spent assembly to the cooling pond. For as long as the refuelling machine is joined to the fuel channel cavity, a small flow of clean water is fed from it, through the thermohydraulic sealing, into the fuel channel, thereby creating a "barrier" to the penetration of hot radioactive water into the refuelling machine from the channel.

The control and protection system (CPS) of the reactor is based on the movement of 211 solid absorber rods in specially separated channels cooled by water from an autonomous circuit. The system ensures: automatic maintenance of a set power level; rapid reduction in power by the automatic control rods and radial controllers on the basis of signals indicating main equipment failure; emergency stoppage of the chain reaction by the scram rods on the basis of signals indicating dangerous deviations of the unit parameters or equipment failure; compensation for reactivity fluctuations when the reactor is heated up and brought up to power; and control of the power density distribution through the core.

RBMK reactors are fitted with a large number of independent regulators, which are inserted into the core at a rate of 0.4 m/s when the emergency protection system is triggered. The comparatively slow motion of the regulators is offset by their large number.

The CPS includes sub-systems for local automatic control and local emergency protection working on the basis of signals from in-core ionization chambers. The local automatic control system automatically stabilizes the principal harmonics of the radial-azimuthal power density distribution, while the local emergency protection system ensures that the reactor is protected in an emergency against the subassemblies exceeding the set power in different regions of the core. To regulate the vertical fields there are shortened absorber rods, inserted into the core from below (24 rods).

Apart from the CPS, the RBMK-1000 reactor has the following main monitoring and control systems:

(1) System for physical monitoring of the radial power density field (more than 100 channels) and the vertical power density field (12 channels), using direct-charge sensors;

(2) System for monitoring startup (reactivity meters, removable startup ionization chambers);

(3) System for monitoring water flow through each channel by means of ball-type flowmeters;

(4) System for fuel failure detection from the short-lived activity of volatile fission products in the steam-water communication lines at the outlet from each channel; the activity is detected successively in each channel over the corresponding optimal energy ranges ("windows") by means of a photomultiplier moved by a special carriage from one steam-water pipe to another;

(5) System for monitoring channel tube integrity from the moisture content and temperature of the gas flushing the channels.

All the data are fed to computers. The information is issued to the operators in the form of deviation signals, readings (when called for) and recorder data.

RBMK-1000 units are used predominantly for base-load operation (at constant power).

In view of the high power of the unit, the reactor is shut down automatically only when the readings for power, pressure and water level in the separator go outside permissible limits; when there is a total loss of current; when two turbogenerators or two main circulation pumps cut out at once; when there is a drop of more than a factor of two in the feedwater flow; or when a rupture occurs over the whole cross-section of the 900 mm diam. main circulation pump pressure header. In other cases where the equipment fails, provision is made only for an automatically controlled drop in power (to a level corresponding to the power of the equipment still operating).

1.3. Principal physical characteristics of the reactor

The RBMK-1000 nuclear power reactor is a heterogeneous channel-type thermal reactor in which uranium dioxide slightly enriched in ^{235}U is used as fuel, graphite is used as moderator and boiling light water is used as coolant. The reactor has the following principal characteristics:

Thermal power	3200 MW
Fuel enrichment	2.0%
Mass of uranium in fuel assembly	114.7 kg
Number/diameter of fuel elements in a fuel subassembly	18/13.6 mm
Fuel burnup	20 MW.d/kg
Coefficient of nonuniformity in radial power density	1.48
Coefficient of nonuniformity in vertical power density	1.4
Maximum design channel power	3250 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 coefficient of reactivity α_φ .... at working point	2.0×10^{-4} vol.% steam
Fast power coefficient of reactivity α_W at working point	-0.5×10^{-4}/MW
Temperature coefficient of fuel α_T............	$-1.2 \times 10^{-5}/^{o}C$
Temperature coefficient of graphite α_C........	$6 \times 10^{-5}/^{o}C$
Minimum "weight" of CPS rods, ΔK	10.5%
Worth of manual control rods, ΔK	7.5%
Effect of replacing spent fuel by fresh fuel (average)	0.02%

An important physical characteristic from the standpoint of reactor control and safety is a quantity known as the operative reactivity margin or excess reactivity. This is defined in terms of a certain number of CPS rods inserted into the core in the region of high differential worth, for fully inserted rods.

The excess reactivity for RBMK-1000 reactors is taken as equalivalent to 30 manual regulating rods. The rate of insertion of negative reactivity, when the emergency protection system is triggered is 1 β/s (β is the fraction of delayed neutrons), which is sufficient to compensate for positive reactivity effects.

The nature of the relationship between the effective multiplication factor and the coolant density in RBMK reactors is largely determined by the different types of absorbers in the core. With the initial loading of the emergency protection system, which comprises about 240 additional absorbers with boron, loss of water draining leads to a negative reactivity effect.

At the same time, a slight increase in steam quality at rated power, with a reactivity margin of 30 rods, results in a reactivity increase ($\rho = 2 \times 10^{-4}$/vol.% steam).

In the case of a boiling water graphite-moderated reactor, the main parameters relevant to operational reliability and thermal safety are the temperature of the fuel elements, the margin to nucleate boiling margin and the graphite temperature.

A series of programs has been devised for RBMK reactors which allow prompt calculations by the plant computers to ensure thermal stability with continuous refuelling and with the valves at the channel inlets in any position. This makes it possible to determine the thermal parameters of the reactor for any channel flow frequency and for any type of control (on the basis of outlet steam quality or the critical power margin) and for any degree of pre-throttling of the core.

To determine the power density fields through the core, the plant relies on physical monitorings based on in-core measurement of the vertical and radial neutron flux. In addition to the physical monitoring system readings, the plant computer also receives data characterizing the core composition, the energy output of each fuel channel, position of the control rods, distribution of water flow through the core channels, and readings from the sensors indicating coolant pressure and temperature. The PRIZMA program calculations carried out by the computer periodically give the operator a digital printout of core configurations, indicating the type of core loading, the position of the control rods, the arrangement of in-core sensors, the power distribution, the critical power margins and margins for the maximum permissible thermal loads on the fuel elements for each fuel channel in the reactor. The plant computer also calculates the overall thermal power of the reactor, the distribution of steam-water mixture flow through the separators, the integral power output, the outlet steam quality from each fuel channel and various other parameters needed to monitor and control the reactor plant.

Experience in operating actual RBMK reactors shows that with the existing means of monitoring and controlling these reactors there is no difficulty in maintaining the temperatures of the fuel and graphite and the critical heat margins at the permissible level.

1.4. Safety systems (Figs 2 and 3)

1.4.1. Protective safety systems

The emergency core cooling system (ECCS) is a protective safety system designed to draw off the residual heat from the core by feeding an appropriate volume of water into the reactor channels in the event of accidents which damage the main core cooling system. Associated with such accidents are ruptures in the large-diameter MFCC pipelines, as well as ruptures in the steam pipes and in the feedwater pipelines.

The system for preventing excess pressure in the main coolant circuit is designed to ensure an acceptable pressure level in the circuit by drawing off steam into a pressure suppression pool where it will condense.

The system for protecting the reactor space is designed to ensure that acceptable pressure within the reactor space is not exceeded in an emergency situation involving the rupture of one fuel channel; it does this by transferring the steam and gas mixture from the reactor space into the steam and gas disposal compartment of the pressure suppression pool and later into the pressure suppression pool itself with simultaneous suppression of the chain reaction by the emergency protection system. The ECCS and the reactor space cooling system can be used to introduce appropriate neutron absorbers (boron salt and ^{3}He).

1.4.2. Localizing (confining) safety systems

The accident localization system as used on the fourth unit of the Chernobyl' nuclear power station is designed to localize and contain radioactive emissions in accidents involving loss of integrity in any of the pipes of the reactor's coolant circuit, with the exception of the pipework of the steam-water communication lines, the upper parts of the fuel channels and that part of the downcomers which is situated in the drum separator compartment, and the pipework for the steam and gas discharges from the reactor space.

The main component of the localization system is a system of leaktight enclosures, including the following compartments within the reactor compartment:

- Reinforced leaktight compartments, distributed symmetrically in relation to the reactor axis and designed for an excess pressure of 0.45 MPa;

- Compartments of the distributing group headers and lower water communication lines; these compartments are designed in line with the strength of the elements used in the reactor construction, not to permit a rise of over 0.08 MPa in the excess pressure level and are calculated to this magnitude.

The compartments containing the reinforced leaktight compartments and steam distribution corridor are connected to the water volume of the bubble condenser by steam discharge channels.

The system of cut-off and sealing devices is designed to ensure leaktightness in the accident localization area by cutting off the pipelines linking the sealed and unsealed compartments.

The bubble condenser is designed to condense the steam formed:

- In the course of an accident involving loss of integrity of the reactor circuit;

- Through operation of the main safety valves;

- By flows through the main safety valves during normal operation.

1.4.3. Safety-assurance systems (service safety systems)

Electricity supply to the plant

The users of electricity at the power plant are divided into three groups, according to the degree of reliability of supply required:

(1) Those unable to tolerate a break in supply lasting from fractions of a second to several seconds under any circumstances, including complete loss of alternating-current voltage from the plant's own working and stand-by transformers, and requiring an assured supply after the reactor's emergency protection system has come into operation;

(2) Those which, under the same conditions, can tolerate a break in supply lasting from tens of seconds to tens of minutes, and which require an assured supply after the reactor's emergency protection system has come into operation;

(3) Those which do not require a supply in the event of loss of voltage from the plant's own working and stand-by transformers, and which, with the unit operating normally, will tolerate a break in supply during the time taken to transfer from the working transformer to the plant's own stand-by transformer.

1.4.4. Control safety systems

The control safety systems are designed to automatically bring into operation the protection, localizing and safety assurance systems and to monitor their functioning.

1.4.5. Radiation control system

The power plant's radiation control system, which forms an integral part (i.e. a subsystem) of its automated control system, is designed to collect, process and display data relating to the radiation situation within the plant premises and in the external environment, the condition of equipment and circuits and staff radiation exposure, in accordance with the standards and legislation, in force.

1.4.6. Power plant control points

There are two levels of control at the plant, namely station level and unit level.

All systems related to power plant safety are controlled at unit level.

1.5. <u>Description of the site of the Chernobyl' nuclear power station and of the surrounding region</u>

1.5.1. Description of the region

The Chernobyl' nuclear power station is situated in the eastern part of a large region, known as the Byelorussian-Ukrainian Woodlands, beside the River Pripyat', which flows into the Dnepr. The region is characterized by a relatively flat landscape with very minor slopes down to the river or its tributaries.

The total length of the Pripyat' before it flows into the Dnepr is 748 km, and its catchment area at the point where it passes the power plant is 106 000 km^2. The river is 200-300 m wide, with an average flow rate of 0.4-0.5 m/sec. The long-time average volume flow is 400 m^3/sec.

The water-bearing horizon used for the above region's drinking water supply lies at a depth of 10-15 m in relation to the present level of the Pripyat' and is separated from the Quaternary deposits by relatively impermeable argillaceous marls.

The Byelorussian-Ukrainian Woodland region is on the whole characterized by a low population density (up to the start of construction work on the Chernobyl' power plant the average population density of the region was approximately 70 inhabitants per km^2).

At the beginning of 1986 the total population within a region of 30 kilometre radius around the power plant was approximately 100 000, 49 000 of whom lived in the town of Pripyat', situated to the west of the plant's three-kilometre safety zone, and 12 500 of whom lived in the town of Chernobyl', the regional centre, situated 15 km to the south east of the plant.

1.5.2. Description of the power plant site and its buildings

The first stage of the Chernobyl' power plant, two units with RBMK-1000 reactors, was constructed between 1970 and 1977. Work on the two power units comprising the second construction stage was completed on the same site in late 1983.

In 1981 work was begun on the construction of two more power units using the same reactors (the third construction stage) at a distance of 1.5 km to the south-east of the existing site.

To the south east of the power plant site and directly within the Pripyat' valley, a 22 km^2 cooling water pond was constructed to provide cooling water for the turbine condensers and the other heat exchangers of the first four units. The normal breast-wall level of the water in the cooling pond is taken to be 3.5 m below the design level of the power plant site.

Under the third construction stage, two powerful water-cooling towers (each with a hydraulic capacity of 100 000 m^3/h) are being built; these will be capable of functioning in parallel with the cooling pond.

The area reserved for the construction base and warehouse facilities is situated to the west and north of the site of the first and second stages.

1.5.3. Information on the number of staff at the power plant site at the time of the accident

On the night of 25-26 April 1986 there were 176 duty operational staff and workers from different departments and maintenance services on the site of the first and second construction stages.

In addition to this there were 268 builders and assemblers working on the night shift on the site of the third construction stage.

1.5.4. Information on equipment situated on-site and previously in operation in the complex containing the damaged reactor, and on equipment used in bringing the accident under control

Construction of the Chernobyl' nuclear power station is being carried out in stages, each comprising two power units with common on-site special water purification systems and auxiliary facilities, among which are:

- A storage facility for liquid and solid radioactive wastes;
- Open distributive systems;
- Gas supply unit;
- Stand-by diesel power plants;
- Hydraulic and other facilities.

The liquid radioactive waste storage facility, built as part of construction stage two, is intended for the receipt and temporary storage of the liquid radioactive wastes arising from the operation of the third and fourth units, and also to receive water from washing operations and to return it for processing. The liquid radioactive wastes are channelled from the main vessel through pipes laid along the lower deck of the pipe bridge, while the solid radioactvie wastes reach the storage facility through the upper corridor of the pipe bridge in electric trolley-cars.

The nitrogen-oxygen station is designed to supply the needs of the plant's third and fourth units.

The gas supply unit comprises a compressor unit, electrolysis unit and helium and argon containers; its purpose is to supply the plant's third and fourth units with compressed air, hydrogen, helium and argon. Receptacles for storing the nitrogen and hydrogen are situated in the open.

The stand-by diesel power plant is an independent emergency source of electricity to supply those systems which are important to the safety of each unit. Each stand-by diesel power plant of the third and fourth units is equipped with three diesel generators having a unit output of 5.5 MW. These plants are served by an intermediate and a base diesel fuel depot, fuel transfer pumps and emergency fuel and oil discharge tanks.

The service water for the third and fourth units is supplied by the cooling water pond.

The water for the circulation pumps, which serves both the third and fourth units, enters the pressure basin and from there flows by gravity to the turbine condensers.

In the case of those users requiring an uninterrupted supply of service water, this is provided for by separate pumping stations for the third and fourth units. A stand-by power supply from the diesel generators is available to these pumping stations.

On 25 April 1986, all four units of the first and second construction stages were in operation, as were all auxiliary systems and on-site facilities associated with their normal operation.

2. CHRONOLOGICAL ACCOUNT OF HOW THE ACCIDENT EVOLVED

The fourth unit of the Chernobyl' nuclear power plant went into operation in December 1983. At the time when the reactor was to be shut down for intermediate maintenance, planned for 25 April 1986, the core contained 1659 fuel assemblies with an average burnup of 10.3 MWd/kg, one additional absorber and one unloaded channel. Most of the fuel assemblies (75%) were first load bundles with a burnup of 12 to 15 MWd/kg.

Before shutdown, tests were to be carried out on turbogenerator No. 8 in a regime whereby the turbine would be supplying plant power requirements during the run down. The purpose of these experiments was to test the possibility of utilizing the mechanical energy of the rotor in a turbogenerator cut off from the steam supply to sustain the unit's own power requirements during a power failure. This regime is in fact used in one sub-system of the reactor's fast-acting emergency core cooling system (ECCS). If carried out in an appropriate way with the requisite additional safety measures, such an experiment would not be forbidden on an operating power plant.

Similar tests had already been carried out at the Chernobyl' plant. At that time it had been found that the voltage on the generator busbars falls off long before the mechanical energy of the rotor is expended during the run-down. In the tests planned for 25 April 1986 the experimenters intended to use a special generator magnetic field regulator to eliminate this problem. However, the "Working Programme for Experiments on Turbogenerator No. 8 of the Chernobyl' Nuclear Power Plant", in accordance with which these tests were to be performed, was not properly prepared and had not received the requisite approval.

The quality of the programme was poor and the section on safety measures was drafted in a purely formal way. (The safety section said merely that all switching operations carried out during the experiments were to have the permission of the plant shift foreman, that in the event of an emergency the staff were to act in accordance with plant instructions and that before the experiments were started the officer in charge - an electrical engineer, incidentally, who was not a specialist in reactor plants - would advise the security officer on duty accordingly.) Apart from the fact that the programme made essentially no provision for additional safety measures, it called for shutting off the reactor's emergency core cooling system. This meant that during the whole test period, i.e. about four hours, the safety of the reactor would be substantially reduced.

Because the question of safety in these experiments had not received the necessary attention, the staff involved were not adequately prepared for the tests and were not aware of the possible dangers. Moreover, as we shall see in what follows, the staff departed from the programme and thereby created the conditions for the emergency situation.

On 25 April at exactly 1:00 hours the staff began to reduce the reactor power (up to then the unit had been operating at rated parameters) and at 13:05 hours turbogenerator No. 7 was switched off with the reactor at 1600 MW(th). The electric power required for the unit's own needs (four main circulation pumps, two electrical feed pumps and other equipment) was switched to the busbars of turbogenerator No. 8.

At 14:00 hours the reactor's emergency core cooling system was disconnected from the multiple forced circulation circuit (MFCC) in accordance with the experimental programme. However, because of control room requirements the removal of the unit from operation was delayed. Thus, the unit then continued to operate with the emergency cooling system switched off, in violation of the operating rules.

At 23:10 hours, the power reduction was resumed. Under the test programme, the rundown of the generator with simultaneous provision of unit power requirements was to be carried out at a reactor power of 700-1000 MW(th). However, when the local automatic regulation system was shut off, which under the operating rules is supposed to be done at low power, the operator was unable to eliminate the resultant unbalance in the measuring part of the automatic regulator quickly enough. As a result of this, the power fell below 30 MW(th). Only at 1:00 on 26 April did the operator succeed in stabilizing it at 200 MW(th). Since the "poisoning" of the reactor was continuing at the same time, a further increase in power was hindered by the small excess reactivity available, which at that moment was substantially below what the regulations called for.

Even so, it was decided to conduct the experiments. At 1:03 and at 1:07 one additional main circulation pump was switched in from either side to join the six pumps already operating, so that when the experiment was finished - during which four main circulation pumps were to be operating through the rundown - four pumps would remain available on the MFCC for safe cooling of the reactor core.

Since the reactor power, and consequently the hydraulic resistance of the core and the MFCC were substantially lower than the planned level and since all eight main circulation pumps were in operation, the total coolant flow rate through the reactor rose to (56 000-58 000 m^3/h, and at some individual pumps to 8000 m^3/h, which meant a violation of the operating rules. An operating regime of this kind is forbidden because of the danger of pump breakdown and the possibility of vibrations arising in the main coolant pipes owing to cavity formation. The switching in of the additional main circulation pumps and the resulting increase in water flow through the reactor brought about a reduction of steam formation, a fall in steam pressure in the drum separators, and changes in other reactor parameters. The operators attempted manually to sustain the main parameters of the system - steam pressure and the water level in the drum separators - but they did not entirely succeed in doing so. At this stage they saw the steam pressure in the drum separators sag by 0.5-0.6 MPa and the water level drop below the emergency mark. In order to avoid shutting down the reactor in such conditions, the staff blocked the emergency protection signals relating to these parameters.

At the same time, the reactivity continued to drop slowly. At 1:22:30, the operator saw from a printout of the fast reactivity evaluation program that the available excess reactivity had reached a level requiring immediate shutdown of the reactor. Nevertheless, the staff were not stopped by this and began with the experiments.

At 1:23:04, the emergency regulating valves of turbogenerator No. 8 shut. The reactor continued to operate at a power of about 200 MW(th). The available emergency protection from the closing of the emergency regulating valves on two turbogenerators (turbogenerator No. 7 had been shut off on 25 April) was blocked so that it would be possible to repeat the experiment if the first attempt proved unsuccessful. This meant a further departure from the experimental programme, which did not call for blocking the reactor's emergency protection with the switching off of two turbogenerators.

Shortly after the beginning of the experiment the reactor power began to rise slowly.

At 1:23:40, the unit shift foreman gave the order to press button AZ-5, which would send all control and scram rods into the core. The rods fell, but after a few seconds a number of shocks were felt and the operator saw that the absorber rods had halted without plunging fully to the lower stops. He then cut off the current to the sleeves of the servo drives so that the rods would fall into the core under their own weight.

According to observers outside unit 4, at about 1:24 there occurred two explosions one after the other; burning lumps of material and sparks shot into the air above the reactor, some of which fell onto the roof of the machine room and started a fire.

3. ANALYSIS OF THE ACCIDENT USING A MATHEMATICAL MODEL

The "Skala" centralized control system of the RBMK-1000 reactor has a program for diagnostic parameter recording (DPRP) under which several hundred analog and discrete parameters are periodically examined and stored in accordance with a specified cycle (minimum cycle time 1 second).

In connection with the experiments, only those parameters were recorded with great frequency which were important for an analysis of the experimental results. Therefore, in trying to reconstruct the course of the accident, we used a mathematical model incorporating not only the DPRP print-out but also instrument readings and the results of questioning of the staff.

To perform a rapid analysis of different variants and versions of the accident situation under consideration, we used an integral mathematical computer model of the RBMK-1000 unit in real time. The dependences of reactivity on steam content and on absorber rod movements were determined from calculations based on three-dimensional neutron physics dispersion models.

In reconstructing the course of the accident, it was particularly important to be sure that the mathematical model correctly described the behaviour of the reactor and other equipment and systems in precisely those conditions which prevailed just before the destruction of the unit. As we have already noted in the previous chapter, the reactor was operating unstably after 1:00 hours on 26 April 1986 and the operators were almost continualy introducing new "perturbations" into the controlled system in order to stabilize its parameters. This has made it possible, for a fairly long time interval involving various influences on the reactor, to compare factual data established fairly reliably by the recording systems with the data obtained through numerical modelling. The results of this comparison have proved to be highly satisfactory, which suggests that the mathematical model satisfactorily reproduced the actual plant.

In order to get as clear an idea as possible of the influence of preceding events on the development of the accident, we analysed data beginning at 1:19:00, i.e. 4 minutes before the beginning of the turbo-generator run down experiment (Fig. 4). This movement is convenient in the sense that the operator was then starting one of the operations involved in the drum-separator make up (the second since 1:00 hours) which produced powerful perturbations in the controlled system. At this moment the DPRP recorded the position of the rods of all three automatic regulators - in other words, the initial conditions of the calculation were very clearly established.

The operator began the drum separator make up in order to prevent a radical drop in the water level of the separators. After 30 seconds he succeeded in maintaining the level by increasing the input flow of make up water by a factor of more than three. It would seem that the operator had decided not only to maintain the water level but to raise it. For that reason the water flow continued to increase and after about a minute was already four times the initial value.

As soon as the colder water from the drum separators reached the core, steam generation was substantially reduced, and this in turn reduced the volumetric steam quality, raising the automatic regulator rods. Within about 30 seconds the rods rose to the upper stops and the operator had to "help" them with the manual control rods, thereby reducing the available excess reactivity (this operation was not recorded in the daily operating log, but without it the operator could not possibly have maintained the power at 200 MW). By moving the manual rods upwards the operator brought about an over-compensation and one of the groups of automatic regulator rods dropped 1.8 metres.

The reduction in steam generation brought about a small drop in pressure. Within about a minute, at 1:19:58, the fast-acting steam dump system was closed off, through which excess steam had been slipping to the condenser. This slowed down the rate of pressure drop a little. Even so, up to the beginning of the experiment the pressure continued to fall off slowly. During this period it changed by more than 0.5 MPa. At 1:22:30 the "Skala" centralized control system provided a print-out of the actual power density fields and of all regulatory rod positions. It was for this instant in time that we attempted to correlate the calculated and recorded neutron fields.

The overall neutron field characteristic at this moment can be described as follows: in the radial-azimuthal direction it showed for all practical purposes a smooth convex shape, but in the vertical direction the curve was double-humped, on average, with a greater release of energy in the upper part of the core. A neutron field distribution of this kind would be completely natural for the state prevailing in the reactor at that moment: a burnt-out core, practically all regulating rods up, volumetric steam quality in the upper part of the core much more than lower down, and greater ^{135}Xe poisoning in the central parts of the reactor than on the periphery.

At 1:22:30 the excess reactivity was only 6-8 rods, in other words not more than half of the minimum permissible value laid down in the operating regulations. The reactor was in an unusual and impermissible state, and to assess the subsequent course of events it was extremely important to determine the differential rod worths of the control rods and the scram rods for real neutron fields and real core multiplication characteristics. Numerical analysis showed that the error in determining the control rod worths was extremely sensitive to the error in re-establishing the vertical power density field. Add to this the fact that at such low power levels (approximately 6-7%) the relative error in measuring the field is much greater than in rated power conditions, then it becomes clear that a vast number of calculational variants will have to be analysed before one can be confident of the rightness or wrongness of any given version.

At 1:23 hours the reactor parameters were closer to stable than at any other time in the interval we are considering, and the experiments began. A minute before this the operator had abruptly reduced the flow of make-up water, and this increased the water temperature at the reactor inlet within a time equivalent to that required for the coolant to pass from the drum separators to the reactor. At 1:23:04 the operator closed the emergency regulating valves of turbogenerator No. 8 and the turbogenerator rundown began. Because of the reduced flow of steam from the drum separators, the steam pressure began to rise slowly (on average at a rate of 6 kPa/s). The total flow of water through the reactor began to fall off owing to the fact that four of the eight main circulation pumps were working off the "running down" turbogenerator.

The increase in steam pressure on the one hand and the reduced flow of water through the reactor together with the reduced input of make-up water to the drum separators, on the other hand, are competing factors in determining the volumetric steam quality and hence the power of the reactor. A point that deserves particular stress is that in the state the reactor had now reached a small change in power would mean that the volumetric steam quality - which has a direct effect on reactivity - would increase much more than at nominal power. The competition between these factors led in the final analysis to a power rise, and this was the circumstance which triggered the pressing of button AZ-5.

Button AZ-5 was pushed at 1:23:40 and the insertion of the scram rods began. At this time the automatic regulator rods, partially compensating for the previous power rise, were already in the lower part of the core, but the fact that the staff were operating with an impermissibly small excess reactivity meant that virtually all other absorber rods were in the upper part of the core.

In the conditions that had now arisen, the violations committed by the staff had seriously reduced the effectiveness of the emergency protection system. The overall positive reactivity appearing in the core began to increase. Within three seconds the power rose above 530 MW, and the total period of the excursion was much less than 20 seconds. The positive void coefficient of reactivity worsened the situation. The only thing that partially compensated for the reactivity inserted at this time was the Doppler effect.

The continuing reduction of water flow through the fuel channels as the power rose led to an intensive steam formation and then to nucleate boiling, over-heating of the fuel, destruction of the fuel, a rapid surge of coolant boiling with particles of destroyed fuel entering the coolant, a rapid and abrupt increase of pressure in the fuel channels, destruction of the fuel channels, and finally an explosion which destroyed the reactor and part of the building and released radioactive fission products to the environment.

In the mathematical model, the destruction of the fuel was simulated by an abrupt increase in effective heat exchange surface when the power density in the fuel exceeded 300 cal/g. It was precisely at that moment that the pressure in the core had risen to the point where an abrupt reduction of water supply from the main circulation pumps occurred (the check valves were closed). This is quite plain from the results obtained with the mathematical model and from the results and measurements recorded by the DPRP. Only the rupture of the fuel channels partially restored the flow from the main circulation pumps; however, the water from the pumps was at this stage no longer directed into intact channels but into the reactor space.

The steam formation and rapid rise of temperature in the core created appropriate conditions for a steam-zirconium reaction and other exothermal chemical reactions. Witnesses observed these reactions in the form of a fireworks display of glowing particles and fragments, escaping from the units.

As a result of these reactions, a mixture of gases was formed containing hydrogen and carbon monoxide, which then led to a thermal explosion upon mixing with the oxygen of the air. This mixing became possible after the reactor space had been vented and destroyed.

4. CAUSES OF THE ACCIDENT

As shown by the analysis presented above, the accident in the fourth unit of the Chernobyl' nuclear power plant belongs to the category of accidents associated with the introduction of excess reactivity. The design of the reactor facility provided for protection against this type of accident with allowance for the physical characteristics of the reactor, including a positive steam void coefficient of reactivity.

The technical means of protection include systems for controlling the reactor and protecting it against a power overshoot, for reducing the examination period, and for self-shielding and protection against malfunctioning in switching operations involving the equipment and systems of the power-generating unit and the emergency core cooling system.

Apart from the technical means of protection, there are also strict rules and instructions for carrying out technological processer at a nuclear power plant, specified in regulations for the operation of each power-generating unit. Among the most important regulations are stipulations referring to the inadmissibility of reducing the operational excess reactivity (reactivity margin) to fewer than 30 rods.

In the process of preparing for and conducting the turbogenerator tests, in which the turbine was to supply the unit's requirements during the run-down, the staff switched off a number of important protection systems and violated the most important provisions of the operating regulations for safe management of technological process.

The table below lists the most dangerous violations of the operating rules committed by the staff of the fourth unit of the Chernobyl' nuclear power plant.

No. 1	Violation 2	Motivation 3	Consequences 4
1.	Reducing the operational reactivity margin substantially below the permissible value	Attempt to emerge from "iodine well"	Emergency protection system of reactor was ineffective
2.	Power dip well below the level provided for by the test programme	Operator error in switching off local automatic control	Reactor proved to be in a condition difficult to control
3.	Connecting of all the main circulation pumps to the reactor, with individual pump discharges exceeding the levels specified in the regulations	Meeting the requirements of the tests	Coolant temperature in the multiple forced circulation circuit approached saturation temperature

4.	Blocking of reactor protection system relying on shutdown signal from two turbogenerators	Intention, if necessary, of repeating the experiment with turbogenerators switched off	Loss of possibility of automatic shutdown of the reactor
5.	Blocking of protection systems relying on water level and steam pressure in the drum-separator	Effort to perform tests despite unstable reactor operation	Reactor protection system based on heat parameters was completely cut off
6.	Switching off of the protection system for the design-basis accident (switching off of the emergency core cooling system)	Wish to avoid spurious triggering of the emergency core cooling system while the experiment was going on	Loss of the possibility of reducing the scale of the accident

The chief motive of the staff was to complete the tests as expeditiously as possible. The failure to adhere to instructions in preparing for and carrying out the tests, the non-compliance with the testing programme itself and the carelessness in handling the reactor facility are evidence that the staff was insufficiently familiar with the special features of the technological processes in a nuclear reactor and also that they had lost any feeling for the hazards involved.

The designers of the reactor facility did not provide for protective safety systems capble of preventing an accident in the combination of circumstances prevailing in unit 4, namely the deliberate switching off of technical protection systems coupled with violations of the operating regulations, since they considered such a conjunction of events to be impossible.

Thus, the prime cause of the accident was an extremely improbable combination of violations of instructions and operating rules committed by the staff of the unit.

The accident assumed catastrophic proportions because the rector was taken by the staff into a non-regulation state in which the positive void coefficient of reactivity was able substantially to enhance the power excursion.

5. PRIORITY MEASURES FOR IMPROVING THE SAFETY OF NUCLEAR POWER PLANTS WITH RBMK REACTORS

It has been decided, for existing nuclear power plants with RBMK reactors, to alter the limit stop switches of the control rods in such a way that, in the extreme position, all the rods are inserted in the core to a depth of 1.2 m. This measure will increase the speed of effective protection and eliminate the possibility of a continuing increase in the multiplying characteristics of the core in its lower part as the rods move down from the upper stops. At the same time, the number of absorber-type control rods constantly present in the core will be increased to 70-80, thereby reducing the void coefficient to a permissible value. This is a temporary measure which will be replaced later on by a conversion of RBMK reactors to fuel with an initial enrichment of 2.4% and by the insertion of additional absorbers in the core to ensure that a positive overshoot of reactivity does not exceed 1 β for any change in coolant density.

A number of additional indicators of the cavitation margin of the main circulating pumps are being installed, and also a system for automatic calculation of reactivity with an emergency shutdown signal when the excess reactivity falls below a specified level. These measures will have a somewhat adverse effect on the economic parameters of nuclear power plants with RBMK reactors but they will guarantee safe operation.

In addition to the technical measures, organizational steps are being taken to reinforce technological discipline and to improve the quality of operations.

6. CONTAINMENT OF THE ACCIDENT AND ALLEVIATION OF ITS CONSEQUENCES

6.1. Fighting the fire at the nuclear power station

The most important task after the accident at the reactor was to fight the fire. Fire had broken out in over 30 places as a result of the explosions in the reactor which had ejected fragments of its core, heated to high temperatures, onto the roofs above several areas housing the reactor section, the de-aeration stages and the machine hall. Because of damage to some oil pipes, electric cable short circuits and the intense heat radiation from the reactor, focuses of fire formed in the machine hall over turbogenerator number 7, in the reactor hall and in the adjoining, partially destroyed buildings. At 01.30, the firemen on duty with the subsection of the fire division responsible for the power station set out from the towns of Pripyat' and Chernobyl' to the scene of the accident. In view of the immediate threat that the fire would spread along the top of the machine hall to the adjoining third unit, and as it was rapidly increasing in strength, the first set of measures taken was directed towards putting out the fire in this critical area. It was therefore decided that the fires inside the buildings should be put out with fire extinguishers and the fire hydrants installed inside. The main focuses had been overcome by 02.10 for the machine hall roof and by 02.30 for the roof of the reactor section. The fire was out by 05.00.

6.2. Evaluation of the state of the fuel after the accident

The accident partially destroyed the reactor core and completely destroyed its cooling system. This being the case, conditions in the reactor vault were determined from:

- Residual heat released by the fuel as a result of the decay of fission products;

- Heat production from various chemical reactions in the reactor vault (hydrogen burning, oxidation of graphite and zirconium and so on);

- Heat removal from the reactor vault through cooling by atmospheric air through openings in the previously hermetic compartments around the core.

In order to prevent the accident from spreading and limit its after-effects, significant efforts were directed during the very first hours after the accident towards evaluating the condition of the fuel and any possible change in that condition with the passage of time. To this end, it was necessary to carry out investigations as follows:

- To evaluate the possible scale of melting (as a result of residual heat production) of the fuel in the reactor vault;

- To study the interaction between the melted fuel and the reactor structural materials and vault (metals, concrete and so on);

- To evaluate the possibility that the reactor structural materials and vault might melt because of the heat released by the fuel.

In the first place, calculations were carried out to evaluate the state of the fuel in the reactor vault, taking into account the leakage of fission products as a function of the time elapsed since the accident.

An analysis of the dynamics of fission product leakage from the reactor in the first few days after the accident indicated that the change in fuel temperature with the passage of time was not monotonic. It could be assumed that there had been several stages in the fuel temperature regime. At the moment of the explosion, the fuel had heated up. An estimate of temperature based on the relative leakage of the iodine nuclides (that fraction of the total isotope content of the fuel which was escaping at any one moment) indicated that the effective temperature of the fuel remaining in the reactor building was 1600-1800 K after the explosion. During the next few tens of minutes, the fuel temperature decreased through heat transfer to the graphite structure and structural materials. There was therefore a corresponding reduction in the leakage of volatile fission products from the fuel.

It was considered that the quantity of fission products ejected from the reactor vault was fundamentally determined during this period by graphite burning and the related migration processes of the finely dispersed fuel and fission products embedded in the graphite as a result of the explosion in the reactor. Subsequently, the fuel temperature began to increase because of residual heat production. As a result, the leakage of volatile radionuclides from the fuel increased (inert gases, iodine, tellurium and caesium). When the fuel temperature had increased further, other, non-volatile radionuclides began to escape. By 4-5 May the effective temperature of the fuel still in the reactor had stabilized and then began to decrease.

The results from the numerical calculation of the condition of the fuel are shown in Fig. 5. The figure shows the residual radionuclide content of the fuel, and also the variation in the fuel temperature taking into account the leakage of fission products as a function of time elapsed since the accident.

The calculations show:

- That the maximum fuel temperature could not have reached the melting point of the fuel;

- That the fission products were coming to the fuel surface in batches, which could lead to only local overheating at the fuel-cladding interface.

The fission products leaving the fuel settled on the structural and other materials surrounding the reactor in the unit, according to their temperatures of condensation and precipitation. Virtually all the krypton and xenon radionuclides left the unit, some of the volatile fission products (iodine, caesium) did so, and practically all the rest stayed within the reactor building. The energy from the fission products was thus dispersed throughout the reactor unit.

These factors indicate that melting of the materials surrounding the fuel and movement of the fuel were unlikely.

6.3. Limiting the consequences of the accident in the core

The potential danger that some melted fuel would concentrate, creating conditions in which a critical mass might be reached and a spontaneous chain reaction occur, made it necessary to take appropriate precautions. In addition, the damaged reactor was releasing significant amounts of radioactivity into the environment.

Immediately after the accident, an attempt was made to reduce the temperature in the reactor vault and to prevent the graphite structure igniting by using the emergency auxiliary feed pumps to supply water to the core space. This attempt proved ineffective.

One of two decisions had to be taken immediately:

- To contain the accident at source by covering the reactor shaft with heat absorbent and filtering materials;

- To allow the combustion processes in the reactor shaft to come to an end of their own accord.

The first line of action was chosen, as the second carried within itself the danger that a significant area would suffer radioactive contamination and the health of the inhabitants of major cities might be threatened.

A group of specialists began to cover the damaged reactor by dropping compounds of boron, dolomite, sand, clay and lead from military helicopters. About 5000 t in all were dropped between 27 April and 10 May, mostly between 28 April and 2 May. As a result, the reactor was covered with a friable layer of material which strongly absorbed aerosol particles. By 6 May, the release of radioactivity had ceased to be a major factor, having decreased to a few hundred Ci, and fell to a few tens of Ci per day by the end of the month.

The problem of reducing the fuel temperature was solved at the same time. To bring down the temperature and reduce oxygen concentration, nitrogen was pumped under pressure from the compressor station into the space beneath the reactor vault. By 6 May, the temperature increase in the reactor vault had ceased, and had begun to reverse itself with the formation of a stable convective flow of air through the core into the open atmosphere.

As a form of double insurance against the extremely low risk (although it was a possibility in the first few days after the accident) of the lower levels of the structure being destroyed, the decision was taken to construct, as a matter of urgency, an artificial heat-removal horizon beneath the foundations of the building. This took the form of a flat heat-exchanger on a concrete slab. This had been done by the end of June.

Experience has shown that the decisions which were taken were basically correct.

A significant degree of stabilization has taken place since the end of May. The damaged parts of the reactor building are stable, and the radiation situation is improving now that the short-lived isotopes have decayed. The exposure dose rate is in the single röntgens per hour range in the areas adjoining the reactor, the machine hall and control and protection areas. Any uptake of radioactivity from the unit into the atmosphere is basically caused by wind removing aerosols. The activity of the releases does not exceed some tens of curies per day. The temperature regime in the reactor vault is stable. The maximum temperatures of the various reactor parts are a few hundreds of degrees centigrade and they have a steady tendency to fall at about 0.5°C per day. The slab at the base of the reactor vault is intact, and the fuel is mostly (~ 96%) localized within the reactor vault, and the compartments of the steam-water and lower water lines.

6.4. <u>Measures taken at units 1, 2 and 3</u>

After the accident in the fourth unit, the following measures were taken at units 1, 2 and 3:

- Units 1 and 2 were shut down at 01.13 and 02.13 respectively on 27 April;
- Unit 3, which is closely linked technically with the damaged fourth unit, but which suffered practically no damage from the explosion, was shut down at 05.00 on 26 April;
- Units 1 to 3 were prepared for a lengthy cold shutdown;
- After the accident, the power station equipment was placed in the cold reserve state.

Units 1 to 3 and the power station equipment are checked by the staff on duty. Significant radioactive contamination of the equipment and buildings of units 1 to 3 of the power station was caused by radioactive materials coming through the ventilation system, which continued to operate for some time after the accident. There was a significant degree of radiation in some parts of the machine hall, which was contaminated through the damaged roof of the third unit.

The Government Commission ordered that decontamination and other work should be carried out on the first, second and third units with the aim of preparing them eventually for startup and operation again.

Special solutions were used for decontamination. Their composition depended on the material being cleaned (plastic, steel, concrete, various coverings) and the type and level of surface contamination. After decontamination, gamma radiation levels dropped by a factor of 10 to 15. Dose rates within the first and second units were between 2 and 10 mR/h in June. Final decontamination and stabilization of the radiation situation at the first, second and third units can only take place when decontamination work has been completed for the rest of the power station area and when the damaged unit has been entombed.

6.5. <u>Monitoring and diagnosis of the state of the damaged unit</u>

The organization of diagnostic measurements envisaged the resolution of the following basic problems:

- Establishment of reliable monitoring of fuel displacement;
- Determination of the scale of contamination in the area adjacent to the nuclear power plant;
- Evaluation of damage and dosimetric survey inside the unit, and determination of possibilities of work in the surviving premises;
- Determination of the distribution of fuel, fission products etc. in order to work out the basic data for designing structures for entombment.

Apart from evaluation of the radiation situation in and around the plant, the priority measurements included monitoring of the condition of the reactor from the air. Helicopters were used to carry out radiation measurements, an infrared survey of the damaged reactor building and its components with a view to measuring the temperature field distribution, analysis of the chemical composition of gases emitted from the reactor vault and a number of other measurements. After it had been determined that the premises and equipment had survived in the lower part of the reactor building, it became possible to conduct initial measurements and to install emergency monitoring instruments. First of all, instruments for measuring neutron flux, gamma dose rate, temperature and heat flow were installed in the evacuated pressure suppression pool. Redundancy was provided for the thermometric instruments. Evaluation of the situation in the pressure suppression pool showed that there was no imminent danger of melting of structural parts. This afforded the assurance that work on construction of a protective slab beneath the unit could be carried out under safe conditions.

The general measurement strategy was cormulated along the following lines:

- Dosimetric and visual survey inside the damaged unit;
- Radiometric and visual survey from helicopters;

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

The main measurement efforts at the initial stage were concentrated on monitoring any downward displacement of the fuel that might occur.

Solution of the diagnostic problem was complicated by the following factors:

- The regular measurement system had broken down completely;

- The outputs of any detectors which might have survived were inaccessible to the personnel;

- Information on the condition of compartments and rooms, and on the radiation situation in them, was limited.

At the next stage it was necessary to determine the location in the building of the fuel ejected from the reactor vault and to evaluate its temperature and heat removal.

Conventional or dosimetric survey methods were used to deal with this problem; in addition, some surviving process pipelines were found through which the measurement probes could be inserted. As a result of these investigations, the distribution of the fuel inside the building was largely determined.

The temperature in the compartments under the reactor did not exceed 45°C as from June, indicating good heat removal.

The monitoring and diagnostic methods were refined on the basis of the data obtained.

6.6. <u>Decontamination of the site</u>

At the time of the accident radioactive materials were scattered over the site and fell on the roof of the turbine hall, the roof of the third unit and on metal pipe supports.

The site as a whole as well as the walls and tops of buildings likewise had substantial contamination as a result of deposition of radioactive aerosols and radioactive dust. The contamination of the site was not uniform.

With a view to reducing the spread of radioactive dust from the site, the roof of the turbine hall and the road shoulders were treated with various polymerizing solutions in order to immobilize the upper layers of soil and prevent dust from rising.

The plant area was divided into separate zones with a view to compehensive decontamination operations. Decontamination is being carried out in the following sequence:

- Removal of refuse and contaminated equipment from the site;

- Decontamination of roofs and outer surfaces of buildings;

- Removal of a 5-10 cm layer of soil and its transfer in containers to the solid waste repository of the fifth unit;

- Laying, if necessary, of concrete slabs on the soil or filling with clean earth;

- Coating of the slabs and of the non-concrete area with film-forming compounds.

As a result of the above measures, it has been possible to reduce the total gamma background in the area of the first unit to 20-30 mR/h. This residual background is due mainly to external sources (damaged unit), indicating that the decontamination of the site and buildings has been sufficiently effective.

6.7. <u>Long-term entombment of the fourth unit</u>

Entombment of the fourth unit should ensure a normal radiation situation in the surrounding area and in the atmosphere and preclude escape of radioactivity into the environment.

For purposes of entombment of the unit it is intended to build the following engineering structures (Figs 6-8):

- Outer protective walls along the perimeter;

- Inner concrete partition walls in the turbine hall between the third and fourth units, in the "B" block and in the de-aerator room along the turbine hall and on the side of the debris by the tank room of the emergency core cooling system;

- A metal partition wall in the turbine hall between the second and third units;

- A protective roof over the turbine hall.

Furthermore, it is planned to seal off the central hall and other reactor rooms and to pour concrete over the debris by the tank room of the emergency core cooling system and over the rooms of the northern main circulation pumps to isolate the debris and to provide protection against radioactive radiation from the reactor sector.

The thickness of the protective concrete walls will be 1 m or more, depending on the design solutions adopted and the radiation situation.

Two variants are considered in the ventilation design:

- An open system with purification of air by aerosol filters and release into the atmosphere through the existing stack of the central ventilation plant;

- A closed system with removal of heat into the heat exchanger located in the upper part of the space to be ventilated and maintenance of negative pressure inside the building to be ensured by pumping of air from the upper part of the space and its discharge through filters and the stack into the atmosphere.

The above activities are to be carried out in the following sequence:

1. The surface layer of soil in the area adjacent to the unit is to be removed to local sites by means of special technology;

2. The area will be covered with concrete and the surface levelled to ensure the movement of self-propelled cranes and other machinery;

3. The roofs and walls of buildings are to be decontaminated.

   At locations where radioactivity is high, special polymer adhesive pastes of various compositions will be used;

4. After the site has been cleaned and covered with concrete, the metal frames for the protective walls will be assembled and then concrete will be provided;

5. As the walls are built, work will proceed with the construction of the main civil engineering structures which are to ensure complete emtombment of the fourth unit.

6.8. Decontamination of the 30 km zone and its rehabilitation for economic use

Significant radioactive contamination of the areas adjacent to the power plant made it imperative to take a number of extreme decisions involving the establishment of surveillance zones, evacuation of the population, bans or restrictions on economic use of land and so on.

It was decided to establish three surveillance zones: a special zone, a 10 km zone and a 30 km zone. In these zones, strict dosimetric monitoring of all transport has been organized and decontamination points have been established. At the zone boundaries there are arrangements for transferring working personnel from one vehicle to another in order to reduce transmission of radioactive substances.

The radiation situation within the 30 km zone will continue to change, especially in areas with a high gradient of contamination levels. There will occur a substantial redistribution of radionuclides over the different parts of the landscape, depending on the characteristics of the topography. The question of re-establishing of the population can be raised only after the radiation situation over the whole contaminated area has been stabilized, by entombment of the fourth unit, decontamination of the plant site and immobilization of radioactivity in locations with a high contamination level.

In June, construction of a complex of hydraulic engineering structures began with a view to protecting from contamination the ground water and surface water in the Chernobyl' nuclear power station area. These include:

- A filtration-proof wall in the soil along part of the perimeter of the industrial site of the power plant and wells for lowering the water table;

- A drainage barrier for the cooling pond;

- A drainage cut-off barrier on the right bank of the river Pripyat';

- A drainage interception barrier in the south-western sector of the power plant;

- Drainage water purification facilities.

On the basis of an assessment of the soil and plant contamination in the 30 km zone, special agro-engineering and decontamination measures have been worked out and are now being implemented. Work aimed at restoration of the contaminated land to economic use has yet started, thanks to these measures, which include: changes in the conventional systems of soil treatment in this region, use of special compositions for dust suppression, modification of harvesting and crop processing methods, and so on.

7. MONITORING OF ENVIRONMENTAL RADIOACTIVE CONTAMINATION AND HEALTH OF THE POPULATION

7.1. Assessment of the quantity, composition and dynamics of the release of fission products from the damaged reactor

The assessment was based on the results of the following:

- Systematic analyses of the radioisotopic composition of aerosol samples collected at points above the damaged unit from 26 April 1986 on;

- Airborne gamma survey of the plant area;

- Analysis of fallout samples;

- Systematic measurement data from the country's meteorological stations.

The release of radioisotopes from the damaged unit took place over an extended period of time which can be divided into several stages.

In the first stage there was a release of dispersed fuel from the damaged reactor. The radioisotopic composition at this point corresponded roughly to that of the irradiated fuel, but enriched by volatile isotopes of iodine, tellurium, caesium and inert gases.

In the second stage - from 26 April to 2 May 1986 - the rate of release from the unit decreased as a result of the measures taken to stop the graphite burning and to filter the releases. During this period the composition of the radioisotopes being released was again similar to that in the fuel. During this stage finely dispersed fuel was being carried out of the reactor by a flow of hot air and the graphite combustion products.

The third stage was marked by a sharp increase in the rate of release of fission products from the unit. In the initial phase of this stage, the release was composed mainly of volatile components, especially iodine, but then the radioisotopic composition once more became similar to that of the irradiated fuel (on 6 May 1986). The reason for this was the heating of the fuel in the core to a temperature exceeding 1700^{o}C as a result of the reactor after-heat. The temperature caused the migration of fission products and the chemical transformation of uranium oxide, which in turn led to an escape of fission products from the fuel matrix and their relsease in aerosol form on the graphite combustion products.

The fourth and last stage, which began after 6 May, was characterized by a rapid drop in releases (Table 1). This was a consequence of the special measures taken, the formation of more infusible fission product compounds as a result of their interaction with the materials introduced, and the stabilization and subsequent lowering of the fuel temperature.

The radioisotopic composition of the releases is shown in Table 2.

The fission products in the air and fallout samples were in the form of individual radioisotopes (mainly volatile ones) and were part of the composition of the fuel particles. Particles (associates) with an elevated content of individual radioisotopes (Cs, Ru etc.) were identified, these having formed as a result of the migration of fission products in the fuel and the filling and structural materials and of sorption on surfaces.

The total release of fission products (excluding radioactive inert gases) was approximately 50 MCi, or about 3.5% of the total inventory of radioisotopes in the reactor at the time of the accident. These figures were calculated on 6 May 1986 and take into account radioactive decay. The release of radioactive materials virtually ceased on that day.

The composition of the radioisotopes released during the accident corresponded approximately to that of the fuel of the damaged reactor, the difference being that the former had a higher content of volatile iodine, tellurium, caesium and inert gases.

7.2. Monitoring system

When the accident occurred, the official meteorological, radiation and public health monitoring system began to operate on an emergency footing. As soon as the scale of the accident became evident, the monitoring system was widened to bring in additional groups of experts and technicians. In the first days after the accident, efforts were concentrated on the most urgent radiation, public health and biomedical monitoring tasks.

During ths period the monitoring system began to be extended to cover long-term problems also. Among the organizations involved in the establishment of the system were the State Committee on Hydrometeorology and Environmental Protection, the Ministries of Health of the USSR and of the Union Republics, the Academies of Science of the USSR, the Ukrainian SSR and the Russian SSR, the State Committee on the Utilization of Atomic Energy and the State Agro-industrial Committee.

The help of specialized medical institutions in Moscow and Kiev was enlisted to treat those exposed to radiation.

In addition to setting up a monitoring system, a programme of radioecological, biomedical and other scientific studies to evaluate and predict the effects of the ionizing radiation on man, the flora and fauna was drawn up and began to be implemented.

The priority objective of the monitoring programme were as follows:

- Assessment of the possible internal and external exposures of plant personnel and the population of Pripyat' and the 30 km zone;

- Assessment of the possible exposure of the population of a number of areas outside the 30 km zone, the level of radioactive contamination in which could have exceeded the permissible limits;

- Preparation of recommendations on measures to protect the population and staff from exposures in excess of the established limits.

These recommendations included:

- Evacuation of the population;

- Restrictions or a ban on the use of food products containing increased amounts of radioactive substances;

- Recommendations on what action people at home and in open places should take.

In order to solve these priority problems, systematic monitoring was introduced in respect of the following:

- The level of gamma radiation in contaminated areas;

- The concentration of biologically significant radioisotopes in the air and water of water bodies, particularly those supplying drinking water;

- The degree of radioactive contamination of the soil and vegetation and its radioisotopic composition;

- The amount of radioactive substances in food products, especially ^{131}I in milk;

- Radioactive contamination of working and non-working clothes, footwear, means of transport etc.;

- Build up of radioisotopes in internal organs of people etc.

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

The determining factors in the radioactive contamination of the environment as a result of the Chernobyl' accident were the dynamics of the radioactive releases and the meteorological conditions.

The radioactively contaminated plume moved first to the west and north; during the 2-3 days after the accident - to the north; and, for a few days from 29 April - to the south. The contaminated air masses then dispersed for great distances over the Byelorussian, Ukrainian and Russian Soviet Socialist Republics. The height of the plume on 27 April exceeded 1200 m, while the radiation levels in it were 1000 mR/h at a distance of 5-10 km from the accident site. The plume and the radioactive track which was forming were regularly surveyed by the airplanes of the State Committee on Hydrometeorology and Environmental Protection, which were equipped with sampling and gamma spectrometry equipment and Roentgen meters, and by the network of meteorological stations.

Fission and induced activity products (^{239}Np and ^{134}Cs) were identified in the air samples.

The main zones of ground contamination after the accident were to the west, north-west and north-east of the Chernobyl' plant, and subsequently - and to a lesser extent - to the south. Radiation levels near the plant exceeded 100 mR/h; on the western track the maximum radiation levels 15 days after the accident were 5 mR/h at a distance of 50-60 km from the accident zone (maximum distances), and the same to the north at a distance of 35-40 km. In Kiev the radiation levels at the beginning of May reached 0.5-0.8 mR/h.

In the zone of the radioactive track near the plant, in addition to the isotopes listed above, plutonium isotopes were identified (their distribution on the ground was insignificant). In this zone isotope fractionation was insignificant, but on the remote radioactive track the radioactive products were considerably enriched by tellurium, iodine and caesium isotopes.

By integrating the contaminated areas it was possible to determine the total activity of the radioactive fallout outside the plant site. In the nearby and remote fallout areas on the European territory of the Soviet Union it amounted to about 3.5% (see subsection 7.1) of the total activity of the fission and activation products accumulated in the reactor (about 1.5-2% on the nearby trail).

Summing the activity (which was determined by taking ground samples) of the radioisotopic fallout on the nearby track yielded an approximate value of between 0.8% and 1.9%.

The plutonium isotope contamination levels in the zones mentioned above were not the determining as regards decontamination work and decisions of an economic nature.

Information on the radioactive contamination of rivers and water bodies was obtained by regularly analysing water samples from the Rivers Pripyat', Irpen', Teterev, Desna and the Dnieper water intake. From 26 April 1986 on water samples were collected from the whole water area of the Kiev reservoir. The highest ^{131}I concentrations were observed in the Kiev reservoir on 3 May 1986, the figure being 3.10^{-8} Ci/L. It should be noted that the spatial distribution of radioisotopes in the aquatic environment was very uneven.

From the first days of the accident, monitoring of the radioisotopic content of sediments at the bottom of water bodies both inside and outside the 30 km zone was organized. The radioisotope concentration in sediments on the bottom of individual parts of the Kiev reservoir adjacent to the accident region was 10^{-7}-10^{-8} Ci/kg and 10^{-10} Ci/L in water during the period 10-20 June 1986.

The radiation dose to which the aquatic organisms in the Kiev reservoir were exposed will not have any serious effect on the population level. Significant radiation effects on the aquatic ecosystem may be observed only in the Chernobyl' plant cooling pond.

The hydrobionts populating the cooling pond were subjected to the highest radiation burdens. For some species of water plant, the internal dose received was as much as 10 rad/h, while near the bottom of the cooling pond the average external exposure was 4 rad/h (at the end of May 1986).

According to expert evaluations, exposure levels of up to 10^{-2} rad/d produce no noticeable effect on terrestrial ecosystems. Inside the 30 km zone around the Chernobyl' plant, higher radiation levels were observed at individual parts of the area contaminated by fallout: this may result in significant changes in the state of radiosensitive plant species at these points.

Radiation levels outside the 30 km zone cannot produce a noticeable effect on the species of which the plant and animal associations are composed.

The results obtained are preliminary in nature. Studies of the effects of the Chernobyl' accident on living organisms and ecosystems are continuing.

7.4. Population exposures in the 30 km zone around the Chernobyl' plant

On the basis of an analysis of the radioactive contamination of the environment in this zone, assessments were made of the actual and future radiation doses received by the population of towns, villages, settlements and other inhabited places. Following of these assessments, decisions were taken to evacuate the population of Pripyat' and a number of other inhabited places: 135 000 people were evacuated.

As a result of these and other measures, it proved possible to keep population exposures within the established limits.

The radiological effects on the population in the next few decades were evaluated. The effects will be insignificant against the natural background of cancerous and genetic diseases.

7.5. Data on the exposure of plant and emergency service personnel. Medical treatment

As a result of their participation in measures to combat the accident in the first few hours after its occurrence, a number of plant and emergency service personnel received high radiation doses (more than 100 rem) and also suffered burns during their efforts to extinguish the fire. All those affected were given immediate medical attention. By 6:00 hours on 26 April 1986, 108 people had been hospitalized and in the course of the day a further 24 persons out of those examined were admitted to hospital. One person died from severe burns at 6:00 hours on 26 April and one of those working at the damaged unit was not found. It is possible that he was working in the area where structures had collapsed and there was high activity.

As a result of the early diagnosis procedures used in the Soviet Union, within 36 hours persons in whom the development of acute radiation syndrome was diagnosed as extremely likely had been identified for immediate hospitalization. The hospitals selected were the clinical institutes in Kiev closest to the site of the accident and a specialized unit in Moscow, the aim being to provide the maximum amount of assistance and expert analysis of the results of examinations.

One hundred and twenty-nine patients were sent to Moscow in the first two days. Of these, in the first three days 84 were identified as suffering from degrees II-IV of acute radiation syndrome and 27 as degree I of acute radiation syndrome. In Kiev there were 17 patients suffering from degrees II-IV and 55 from degree I.

Details of the methods and results of the treatment of these patients is given in the Annex.

The total number of fatalities caused by burns and acute radiation syndrome among personnel stood at 28 at the beginning of July. None of the population received high doses which would have resulted in acute radiation syndrome.

Table 1. Daily release, q, of radioactive substances to the atmosphere from the damaged unit (excluding radioactive inert gases)*.

Date	Days after the accident	q, MCi **
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}

*Release evaluation error ±50%. It is composed of the error of the dosimetric instruments, of the radiometric measurements of the radioisotopic composition of air and soil samples and of the error due to averaging the fallout over the area.

**The values of q were calculated on 6 May 1986 taking into account radioactive decay. (At the time of the release on 26 April 1986, the activity was 20–22 MCi.) For the composition of the release, see Table 2.

Table 2. Assessment of the radioisotopic composition of the release from the damaged unit*.

Isotope**	Activity of release, MCi		Amount of activity released from the reactor by 06.05.86, %
	26.04.86	06.05.86***)	
^{133}Xe	5	45	Possibly up to 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

*Evaluation error ±50%. For explanation, see footnote on Table 1.

**Data on the activity of the main radioisotopes measured in the radiometric analysis.

***Total release by 6 May 1986.

8. RECOMMENDATIONS FOR IMPROVING NUCLEAR POWER SAFETY

8.1. Scientific and technical aspects

In 1985 the Consultative Council for Co-ordination of Scientific Research on Nuclear Safety approved a "List of Priority Tasks", which constitutes the basis for planning experimental and theoretical studies on nuclear safety in the USSR aimed at providing a more detailed justification for safety specifications, evaluating the actual level of nuclear safety and enabling nuclear power plants commissioned before 1985 to be brought up to that level in accordance with the specifications laid down.

After the accident at the Chernobyl' nuclear power plant, the status of theoretical and experimental research on nuclear safety has been reviewed and evaluated, and measures for extending, improving and intensifying it have been developed.

Computer programs for analysing the safe behaviour of nuclear power plants in all possible transient and accident regimes - including conditions not anticipated at the design stage - are being improved, and modelling systems and complexes are being developed.

Research on the possibility of building reactors with passive safety systems - so called "intrinsically safe" reactors, the cores of which cannot be destroyed in any type of accident - is being expanded.

There will be an expansion of research on quantitative probabilistic analysis of safety, on the analysis of risks from nuclear power and on the development of a conceptual and methodological basis for optimizing radiation safety and for comparing radiation hazards with other industrial hazards.

8.2. Organizational and technical measures

The system of monitoring and technical standards in force in the USSR covers all the basic questions of nuclear safety and is continually improving. In 1985 a Summary List and Development Plan for USSR nuclear power regulations and standards was drawn up under the auspices of the State Nuclear Power Supervisory Board (Gosatomehnergonadzor); this co-ordinates and directs the activities of all official bodies involved in the development and co-ordination of the corresponding scientific and technical documents.

A comparison between the existing Soviet document relating to nuclear power station design and operation with similar foreign documents does not reveal any major differences. In general, the nuclear safety standards in force do not require revision. However, more careful verification of their implementation in practice is necessary. The quality of training and retraining of staff needs to be improved and design and construction staff must verify more carefully the quality of plant components during manufacture, assembly and adjustment during commissioning; their responsibility for the subsequent effectiveness and safety of operating nuclear power plants must be increased.

Since the Chernobyl' accident, organizational measures have been taken to improve the safety of nuclear power plants. These can be divided into two stages.

The first stage, which was carried out before a detailed scientific and technical analysis of the course of the accident had been made and in the light of preliminary information from the site, relates to operating nuclear power plants with reactors of the RBMK type and involves operational measures at those plants. The main purpose of these measures is to prevent any recurrence of operating conditions such as those which immediately preceding the accident.

The second stage relates to measures arising from the scientific and technical analysis of the accident and includes steps aimed at improving the safety of nuclear power plants of all types.

The measures that are planned should be adequate to ensure the safe operation of nuclear power plants with reactors of the RBMK type.

For power plants with other types of reactor, the intention is to carry out safety enhancement measures foreseen earlier, which relate mainly to the latest advances in science and technology, operating experience, the possibility of diagnosing the condition of metal in piping and other plant components, automatic process control systems, and so on.

With a view to raising the level of leadership and responsibility for the development of nuclear power and to improving the operation of nuclear power plants, an All-Union Ministry of Nuclear Power has been established.

A whole range of measures to improve State monitoring of nuclear safety is also to be carried out.

8.3. International measures

In the light of the Chernobyl' accident, the Soviet Union, paying due regard to the international nuclear safety work currently being done and desiring to strengthen international security further, has put forward some proposals about the establishment of an international regime of safe nuclear power development and the expansion of international co-operation in this sphere. These proposals are contained in statements by the General Secretary of the Central Committee of the Communist Party of the Soviet Union, M.S. Gorbachev, on 14 May and 9 June 1986.

An international regime of safe nuclear power development would take the form of a system of international legal instruments, of international organizations and structures and also of organizational measures and activities to preserve the health of the public and protect the environment in the context of world-wide nuclear power activities. The establishment of such a regime could be achieved by drawing up international agreements, signing the corresponding international conventions and supplementary agreements, carrying out joint co-ordinated research programmes on nuclear safety problems, exchanging scientific and technical information, setting up international data banks and banks of material resources needed for safety purposes, and so on.

Funds could be set up, with the direct participation of international organizations, for providing emergency assistance, including that required with the urgent provision of the necessary special medical supplies and dosimetric and diagnostic equipment and instruments and with the supply of food, fodder and other material assistance. It is also necessary to set up a system of early notification and provision of information in the event of accidents at nuclear power plants - in particular, those with transboundary consequences. Attention needs to be paid, moreover, to the question of the material, moral and psychological damage associated with such accidents.

There is yet another aspect of nuclear safety, that of the prevention of nuclear terrorism. A task of overriding importance in this connection is the development of a reliable system of measures to prevent nuclear terrorism in any form.

An important role in the establishment of an international regime of safe nuclear power development must be played by the IAEA.

It is gratifying to note that initial steps have already been taken to carry out the proposals in respect of the establishment of an international regime of safe nuclear power development. Intensive work has begun on preparations for the conclusion of two international conventions, relating to early notification of nuclear accidents and to assistance in the event of nuclear accidents and radiological emergencies. Certain aspects of the expansion of international co-operation, in particular, the IAEA's research programmes on nuclear safety, are being actively discussed.

The proposals for establishment of an international regime of safe nuclear power development are inextricably linked with problems of military détente and nuclear disarmament. The Chernobyl' accident has demonstrated once again the danger of nuclear energy getting out of control and has made people aware of the devastating consequences which would ensue from its military application or from damage to peaceful nuclear installations in the course of military action. It is absurd, at the same time as discussing and solving problems of the safe utilization of nuclear energy, to develop ways and means of applying it in the most dangerous and inhuman way possible.

9. THE DEVELOPMENT OF NUCLEAR POWER IN THE USSR

Owing to the extremely rapid development of nuclear power, a reduction in the consumption of organic fuel by thermal power plants in the European part of the country is planned in the Soviet Union's energy programme. The contribution of oil to electric power production is to be cut by more than half. Nuclear power should then cover most of the economy's increased electricity requirements. There are plans for the maximum possible use of nuclear fuel for centralized heating and industrial heat supply, and for the creation of nuclear industrial complexes.

The Soviet Union is a pioneer in the peaceful uses of atomic energy. The world's first nuclear power plant, with a uranium-graphite channel-type reactor, has been operating for thirty-two years. The subsequent programme for the establishment in the USSR of so called demonstration power reactors for nuclear power plants with relatively low power capacities made it possible to select the most promising of these for further development and improvement.

The three types of reactor adopted in the USSR for the needs of the country's growing nuclear power programme allow great flexibility and reliability of energy supply, more efficient use of nuclear fuel resources than would otherwise be possible, and are also well adapted to the special requirements of a developing power engineering infrastructure.

The nuclear power plants being built in the USSR are based on the WWER, RBMK and fast breeder reactors. The first two are thermal reactors with light-water coolant. The fast breeder reactors use liquid sodium as coolant and are being built at present with a view to full-scale industrial testing of the technical solutions which have been adopted and the gradual future development of a closed fuel cycle based on plutonium.

At present, nuclear power plants with WWER and RBMK reactors provide the basic nuclear power production in the USSR. The country's installed capacity has reached nearly 30 million kilowatts. The Soviet nuclear power plants are characterized by high operational availability. The utilization factor of the installed capacity at nuclear power plants during recent years has been relatively high.

In accordance with the "Main Lines of Economic and Social Development of the USSR for 1986-1990 and up until the year 2000", it is expected that nuclear power will be developed extremely rapidly in the European part of the country and in the Urals. In 1985, power generation at nuclear plants reached nearly 170 000 million kWh and by the year 2000 it will increase by 5-7 times.

This development means that it will be primarily nuclear plants that provide the additional capacity needed for the energy systems of the European part, relieving us of the need to build new thermal plants burning organic fuel for base load operation.

The Soviet Union is also developing nuclear sources of heat supply based on high-temperature gas-cooled reactors. The construction of safe plants with such reactors will make it possible to produce high-temperature heat for a number of industrial technological processes.

The Soviet Union is actively participating in international collaboration in the nuclear power field, co-operating effectively in the competent bodies and commissions of the United Nations, in the IAEA, the World Energy Conference and others.

The development of nuclear energy in the USSR is being carried out in close co-operation with CMEA countries.

List of principal installations of the main block of the plant

No.	Installation or item	Measurement unit	Unit weight in tons	No. per NPP unit
	Reactor section			
1	Graphite stack	Assembly	1850	1
2	System "S" metal structures	"	126	1
3	System "OR" metal structures	"	280	1
4	System "E" metal structures	"	450	1
5	System "KZh" metal structures	"	79	1
6	System "A" metal structures	"	592	1
7	System "D" metal structures	"	236	1
8	Drum-type steam separator	pc	278	4
9	Main circulation pump TsVN-8	"	67	8
10	Main circulation pump electric motor	"	33	8
11	Main isolating gate valve (diameter 800)	"	5.7	8
12	Intake header	"	41	2
13	Pressure header	"	46.0	2
14	Distributing group header	"	1.3	44
15	Lower water communication lines	Set	400	1
16	Steam-water communication lines	"	450	1
17	Downcomers (diameter 300)	"	16	1
17a	Primary coolant circuit pipes (diameter 800)	"	350	1
18	Refuelling machine	"	450	1
19	Overhead crane of central hall Q50/10ts	pc	121	1
20	Overhead crane of main circulation pump room Q50/10ts	"	176	2
21	Supply fan, type VDN at level + 43.0	"	3.5	30
22	Exhaust ventilator at level + 35.0	"	3.5	50
23	Controlled leakage tank	"	1.4	2
24	Controlled leakage heat exchanger	"	0.2	2
25	Scheduled preventive maintenance tank	"	25	4
26	Metal structures and pipes of the accident confinement zone	Set	270	1
27	Check valves of the lower water communication line room	"	2.5	11
28	Accident confinement system release valve	pc	2	8
29	Accident confinement system condensers	"	3.7	36
30	Carriage container	"	146	1
31	Crane in gas activity reduction system room Q30/5ts	"	45	1
	Carbon steel pipes	Set	1170	1
	Stainless steel pipes	"	760	1
	Machine room			
32	Turbo-unit K-500-65/3000	pc	3500	2
33	Moisture separator/reheater SPP-500	"	15	8
34	Low-pressure heater	"	37.5	4
35	First stage condensate pump units	"	2.5	6
36	Overhead crane of machine room Q 125 ts	"	211	1
	Carbon steel pipes	Set	3825	1
	Stainless steel pipes	"	1300	1
37	De-aerator	pc	4.5	2

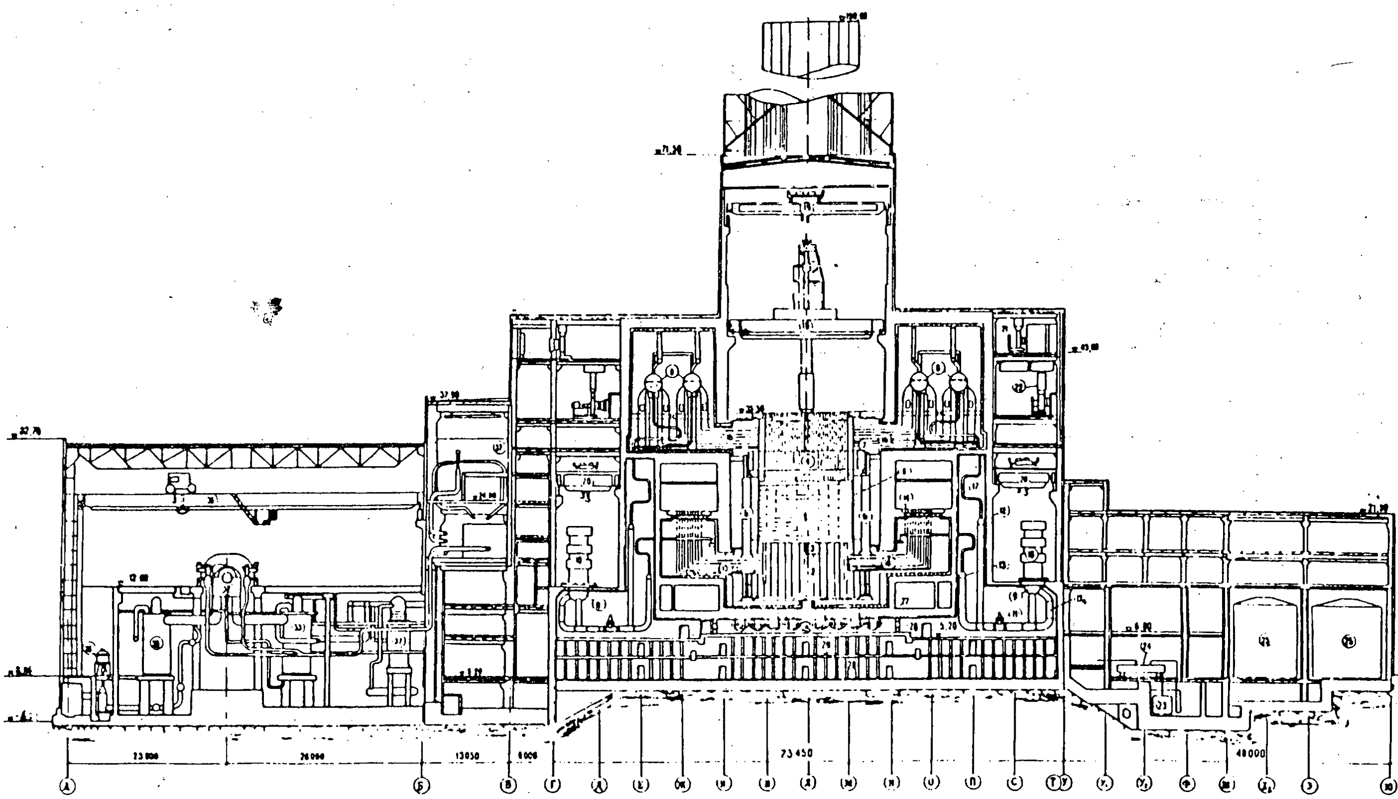


Fig. 1. Cross-section through the main structures of a power plant with an RBMK-1000 reactor (with localization zone)

FOR OFFICIAL USE ONLY

Declassified and Approved For Release 2013/07/15 : CIA-RDP09-00997R000100260001-0

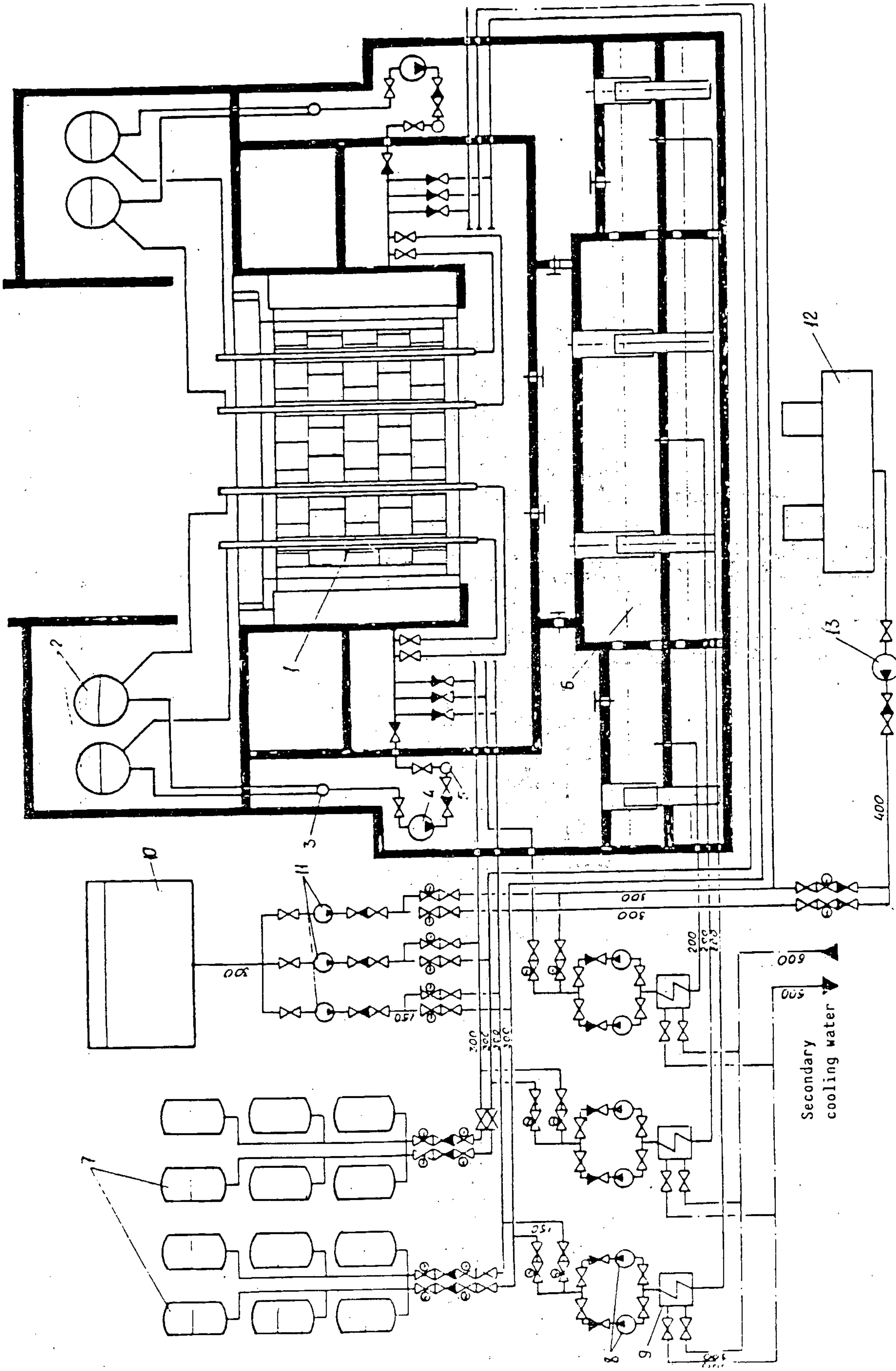


Fig. 2. Schematic diagram of the emergency core cooling system (ECCS):
1. Reactor; 2. Steam separator; 3. Suction header; 4. Main circulation pump; 5. High pressure header; 6. Pressure suppression pond; 7. ECCS vessels; 8. ECCS pumps for cooling the damaged half of the reactor; 9. Heat exchanger; 10. Clean condensate tank; 11. ECCS pumps for cooling the undamaged half of the reactor; 12. Deaerator; 13. Feed pump.

Declassified and Approved For Release 2013/07/15 : CIA-RDP09-00997R000100260001-0

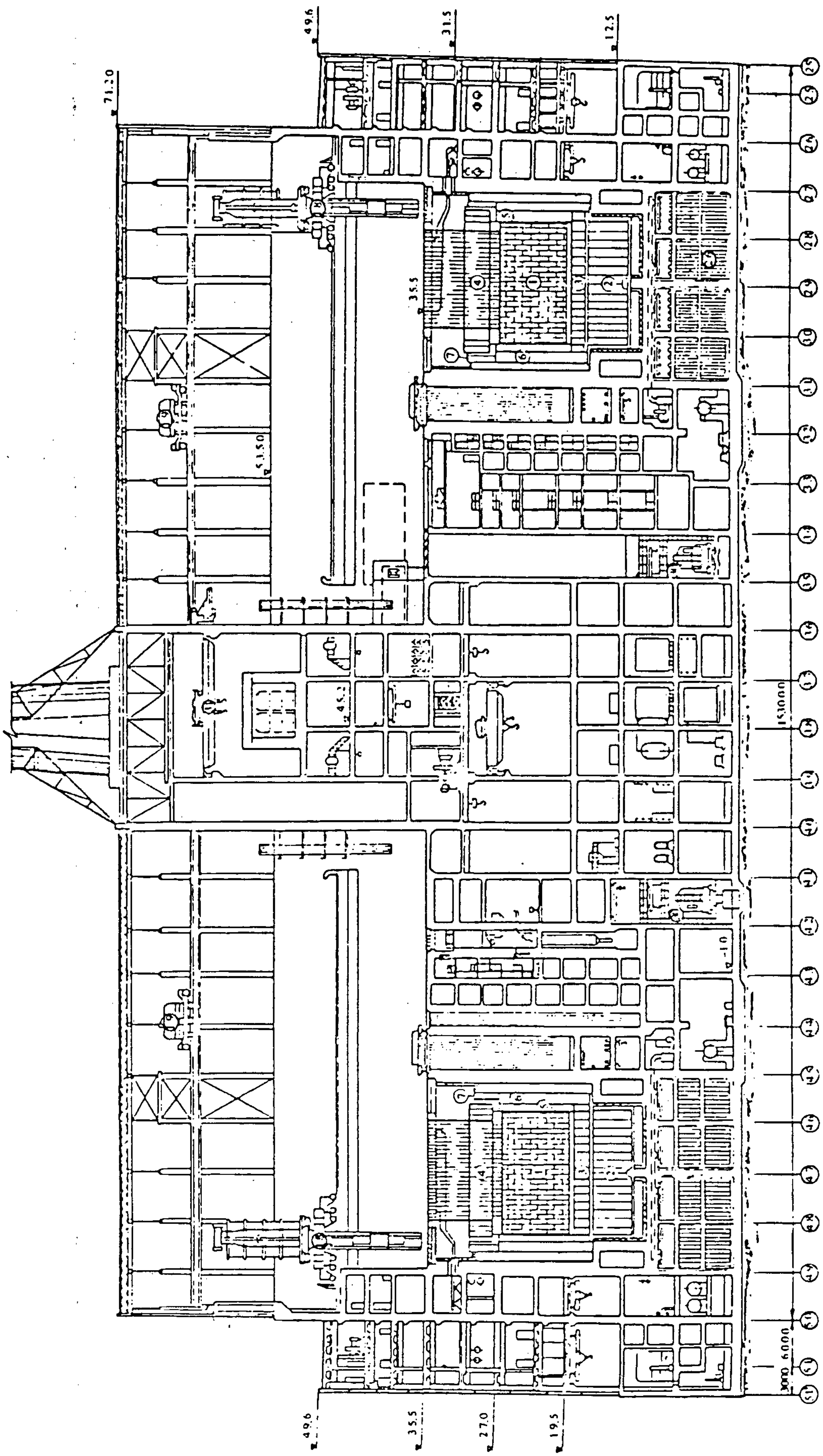
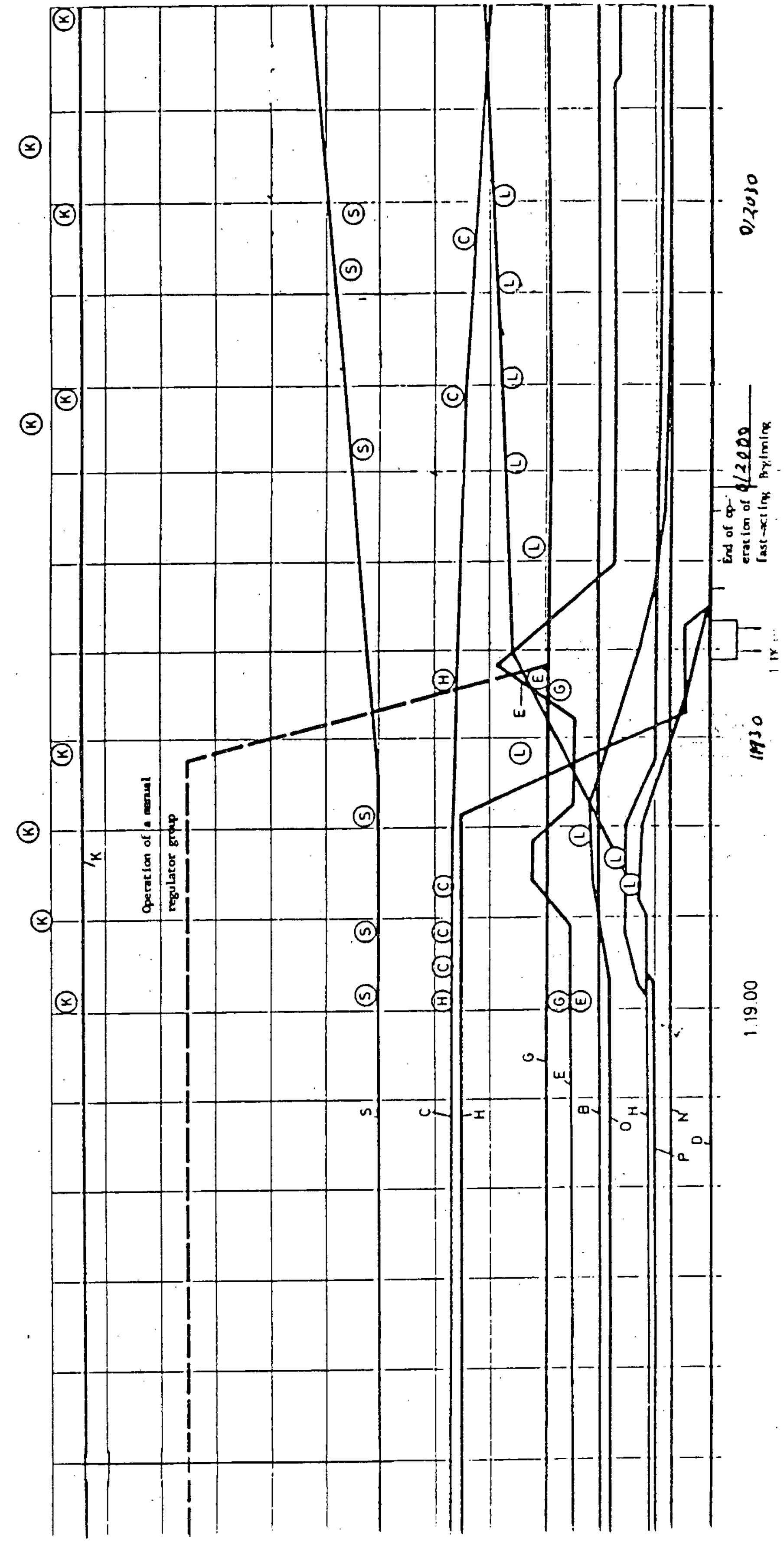


Fig. 3. Cross-section through the reactor sector of a nuclear power plant with an RBMK-1000 reactor (with localization zone)

Fig. 4



Note: Legend from left side of printout missing because of illegibility of original.

Fig. 4

2.

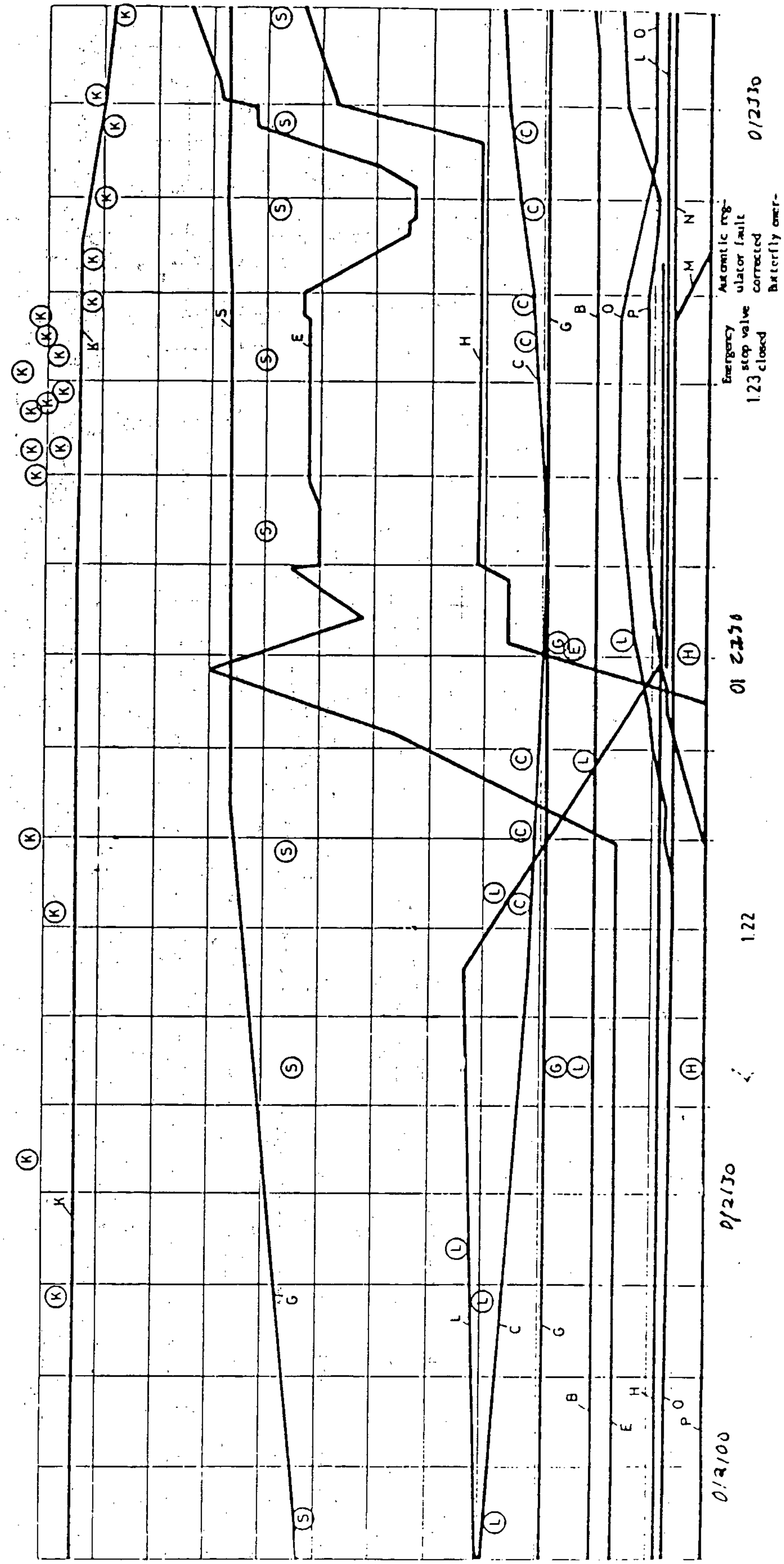
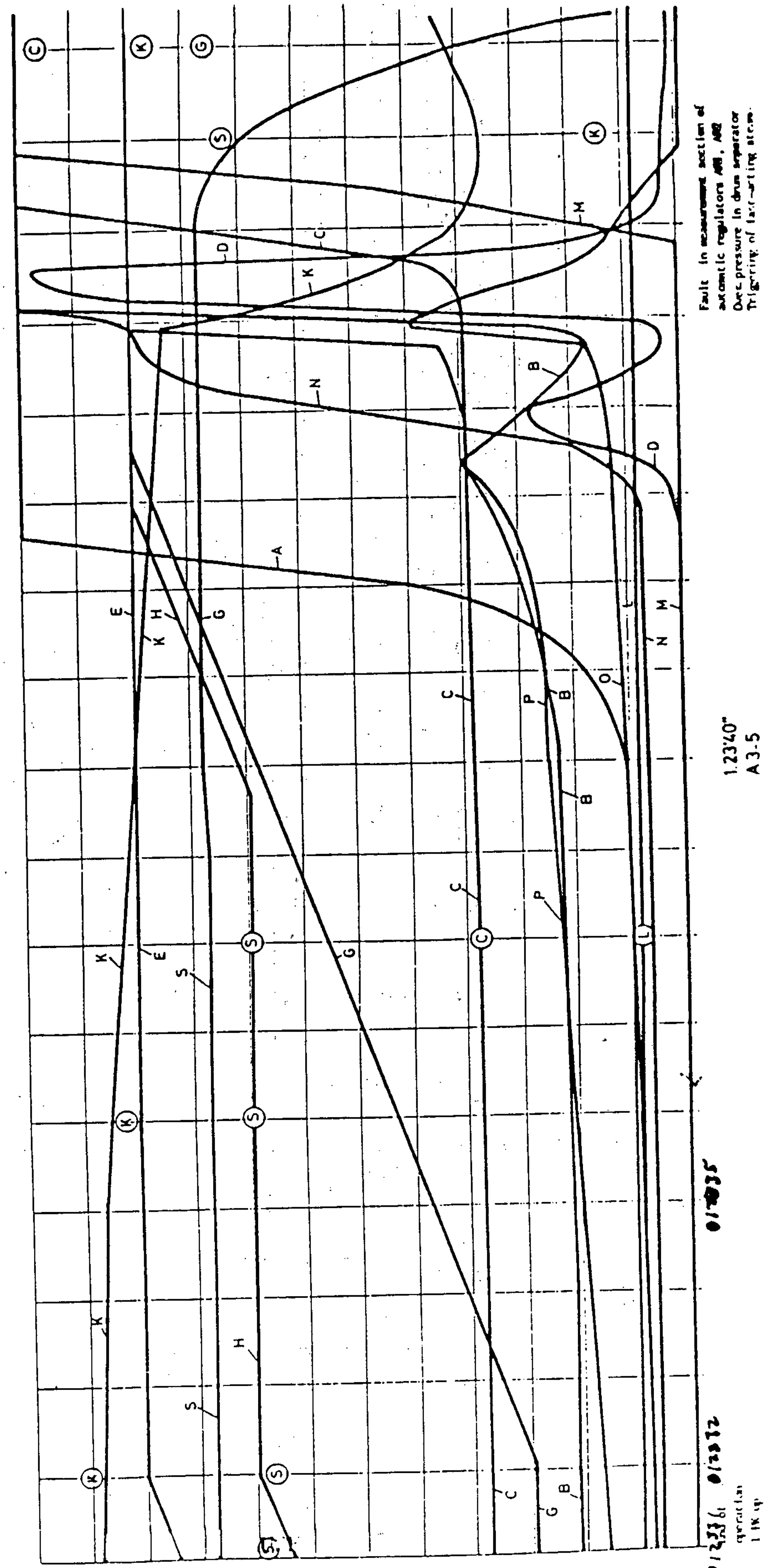

0/2330
01 2230
1.22
0/2130
0/2100
Emergency stop valve closed
1.23
Automatic regulator fault corrected

Fig. 4
3



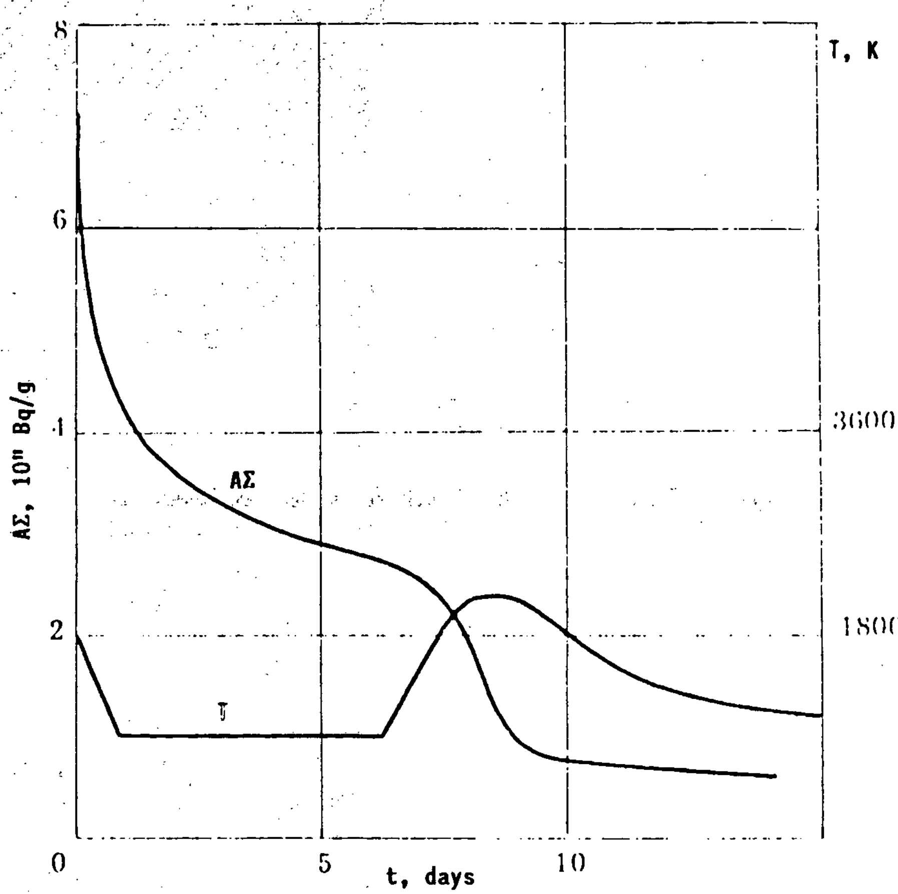


Fig. 5. Variation of activity and temperature of the fuel with time

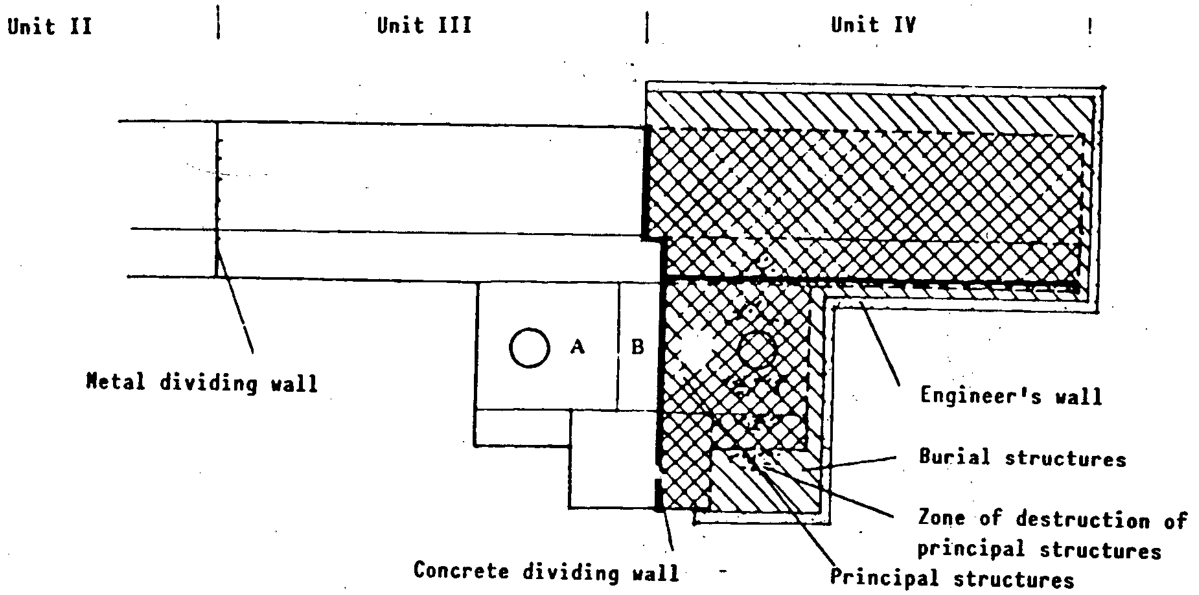


Fig. 6. Diagram showing one scheme for the isolation and encasement of unit 4 (horizontal cross-section)

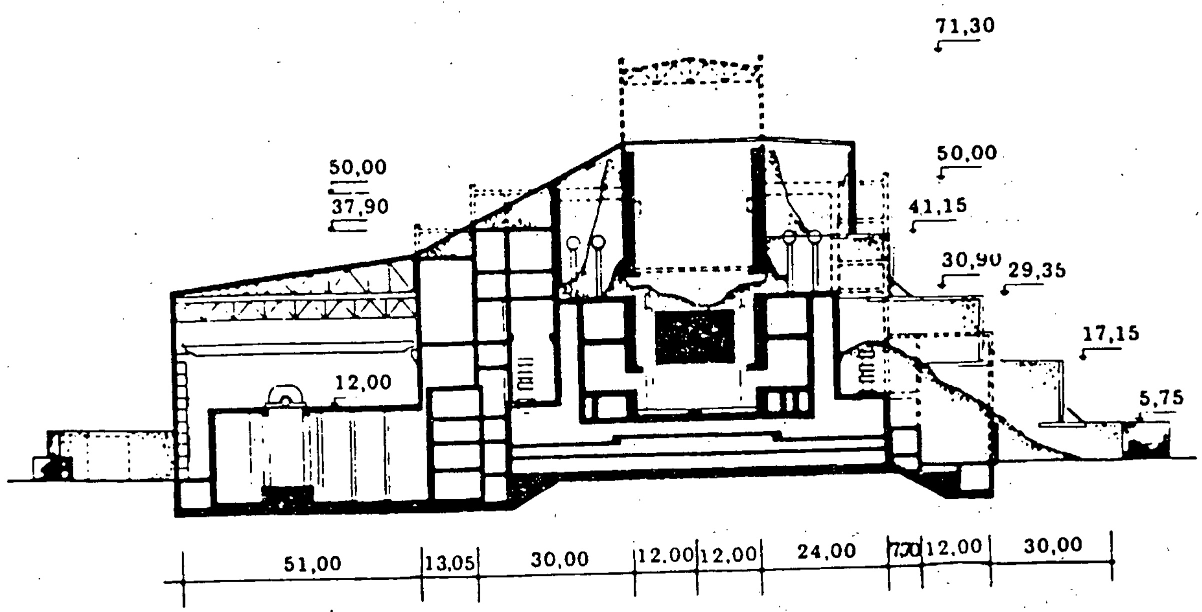


Fig. 7. Diagram showing one scheme for the isolation and encasement of unit 4 (vertical cross-section)

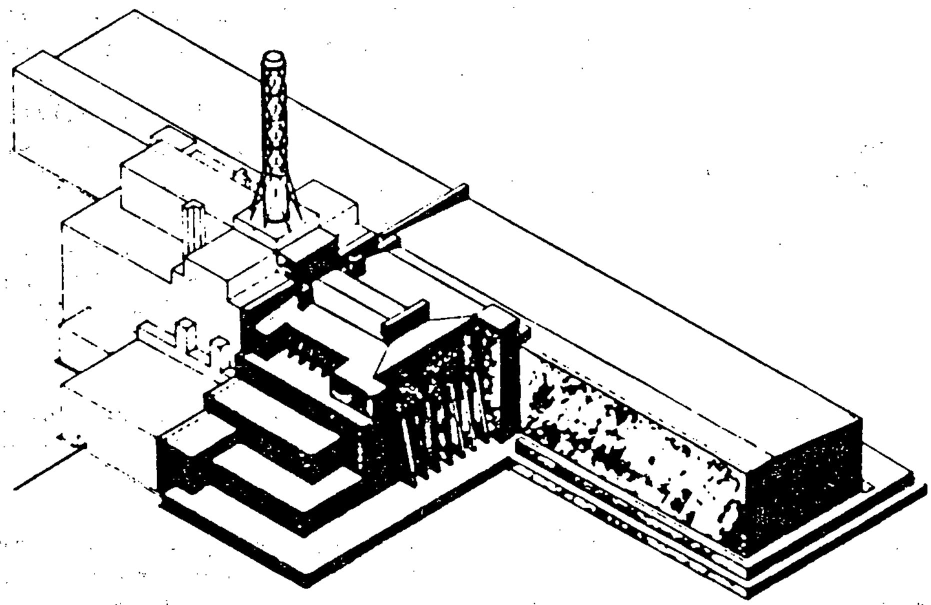


Fig. 8. General view of one scheme for the isolation and encasement of unit 4

PART II. ANNEXES

ANNEX 1

WATER-GRAPHITE CHANNEL REACTORS AND OPERATING EXPERIENCE WITH RBMK REACTORS

1. Water-graphite channel reactors and operating experience with RBMK reactors.

1.1. Water-graphite channel reactors use ordinary water as a coolant and graphite as a moderator. The distinctive features of channel reactors are: the absence of a pressure vessel, the relative simplicity of the design, the extensive scope for channel-by-channel inspection and control, the possibility of refuelling while the reactor is operating, the flexibility of the fuel cycle, and the practically unlimited potential for increasing the power by means of standard structural elements.

The first power reactor in the USSR was a channel-type reactor - the water-graphite reactor of the First Atomic Power Station, which had an electrical capacity of 5 MW and was started up in June 1954 at Obninsk, near Moscow.

The experience accumulated in the construction and operation of the First Atomic Power Station was utilized in planning the Beloyarsk Nuclear Power Plant (NPP) (1964, 300 MW).

The further development of the water-graphite reactor concept in the USSR led to the construction of the high power channel-type boiling-water reactor RBMK-1000 with an electrical capacity of 1000 MW, which, along side the WWER-1000 reactor, became the basic reactor for large-scale nuclear power production in the USSR.

The commissioning of the first RBMK-1000 reactor at the Leningrad NPP in 1973 marked the inception of a series of reactors of this type.

The broad programme of construction of RBMK-1000 reactors carried out in the 1970s led to the commissioning, over the period 1973 to 1985, of 14 reactors (4 reactors each at the Leningrad, Kursk, and Chernobyl' NPPs and 2 at the Smolensk NPP) with a total installed electrical capacity of 14 GW. In each new generation of reactors, improvements aimed at increasing their reliability and safety were introduced.

The knowledge and experience gained from operating generating units with RBMK-1000 reactors revealed their intrinsic safety margins and made it possible to design, on the basis of this type of reactor, the even more powerful RBMK-1500 reactor, with an electrical capacity of 1500 MW, which was brought into service in 1983 at the Ignalinsk NPP and by early 1985 reached its design output.

Channel-type reactors have advantages which facilitate the solution of safety problems. These advantages include:

- their convenience for carrying out individual checks on the state of the fuel elements and fuel assemblies and on channel integrity;

- the possibility of operationally exchanging failed fuel assemblies without shutting down the reactor;

- the reduced hazard from primary circuit pipe breaks due to the increased number of coolant loops and the corresponding reduction in tube diameters;

- the design option of increasing the unit power of the reactor without making the emergency core cooling system more complicated.

On the other hand, certain specific features of channel-type graphite reactors cooled with boiling water call for fundamentally new solutions to be found in developing the safety systems. These features include, in particular:

- the large steam volume in the coolant circuit, which considerably slows the rate of coolant pressure reduction after an accidental pipe break;

- the potential occurrence of a positive void reactivity effect, which to a large extent governs the behaviour of the neutron-flux-determined power of the reactor in emergencies due to interruptions of the coolant circulation through the core;

- the large amount of thermal energy accumulated in the metal structures and graphite stack of the reactor, which affects the decline in thermal power after a response of the safety system (scram).

1.2. As of now, the cumulative service life of RBMK reactors within the nuclear power system is approaching a total of 100 reactor-years.

On the basis of analysis and generalization of this operating experience, the various reactor components and plant systems and operating regimes are continually being updated and improved. As a result, a large number of measures have been developed and introduced with a view to increasing the reliability and operating safety of nuclear power plants. The most important of these measures are the following:

- modernization of the design of the isolating and regulating valves, the ball detectors of the flowmeters and the fuel channel shut-off plugs;

- optimization of the routing of the steam/water piping and the steam discharge tubes of the drum separators;

- improvement of the drum separator internals;

- improvement of the main circulation pumps and their auxiliary systems;

- introduction of prognostic programmes for operational calculations, of programmes for recording the state of equipment during emergencies and of diagnostic programmes for monitoring the process systems;

- development and introduction of local automatic control systems and local emergency protection systems based on in-core sensors;

- justification and experimental operation at one of the generating units of fuel assemblies with an initial enrichment of 2.4%;

- development of afterheat removal systems allowing extended continuous repairs of reactor equipment and components.

These measures and others carried out at the generating units have ensured the reliable and safe operation of nuclear power plants with RBMK reactors, some statistics on which are given in Table 1.2 showing the production of electrical energy by nuclear power plants over the operating period 1981-1985 and the installed capacity factor for 1985.

The maximum values obtained for the installed capacity factor in 1985 were 91% at unit 4 of the Leningrad NPP, 90% at unit 2 of the Chernobyl' NPP and 87% at unit 1 of the Leningrad NPP.

Generalization from the operating experience accumulated and from the scientific research and development work carried out has indicated some ways of increasing the effectiveness of RBMK generating units, including:

increasing the power of existing generating units, improving and optimizing the operating regimes of the units, introducing automatic reactor and equipment protection systems, improving the conditions under which repairs are carried out on the reactor plant and increasing the maintainability of its individual components.

As is well known, the limiting parameters for the power of RBMK reactors are the fuel temperature, the temperature of the graphite stack and metal structures, and the burnout ratio of the fuel channels. At existing reactors, these parameters are lower than the maximum permissible. For example, at the nominal thermal power of the reactor the maximum power of a fuel channel is about 2600 kW, where 3000 kW would be permissible, and the maximum temperature of the graphite stack is 923 K (650°C), where 1023 K (750°C) would be acceptable; the maximum temperature of the metal structures is 573 K (300°C), where 623 K (350°C) would be allowed, and the burnout ratio is not less than 1.35. The main items of equipment in the turbine room of RBMK generating units (turbogenerators, unit transformers, de-aerators, condensate and feed pumps) also have a safety margin of ~10% with respect to power.

These demonstrated safety margins have made it possible to justify operating the generating units at a higher power level, while the full-scale tests carried out in several generating units at powers of up to 107% of nominal have confirmed this possibility.

Table 1.1. The main technical characteristics of nuclear power plants (NPPs) with RBMK-type reactors are given in Table 1.1.

Characteristics	RBMK-1000	RBMK-1500
Electrical power (MW)	1000	1500
Thermal power (MW)	3200	4800
Steam output (t/h)	5800	8800
Steam parameters before turbines:		
Pressure (kgf/cm^2)	65	65
Temperature (°C)	280	280

Table 1.2. Performance data for the operation of NPPs with RBMK-1000 reactors.

Performance data	Leningrad NPP	Kursk NPP	Chernobyl' NPP	Smolansk NPP
Installed capacity as of 1/1/86 (MW)	4000	4000	4000	2000
Electricity production for the period 1981-1985 (10^9 kW.h)	140.4	82.4	106.6	23.4
Installed capacity factor for 1985 (%)	84	79	83	76

ANNEX 2

2. DESIGN OF THE REACTOR PLANT

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

2.1. Reactor

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

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

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

---

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

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

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

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

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

<u>Main characteristics of the reactor</u>

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

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

2.1.1. Design of the fuel assembly and fuel element

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

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

The fuel assembly is 1015 mm long.

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

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

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

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

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

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

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

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

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

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

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

The design of both fuel sub-assemblies is identical.

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

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

2.1.2. Fuel channel (Fig. 2.3)

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

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

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

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

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

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

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

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

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

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

2.1.3. Control channels (Fig. 2.4)

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

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

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

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

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

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

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

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

2.1.5. Graphite stack (Fig. 2.1)

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

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

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

2.1.6. Biological shielding

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

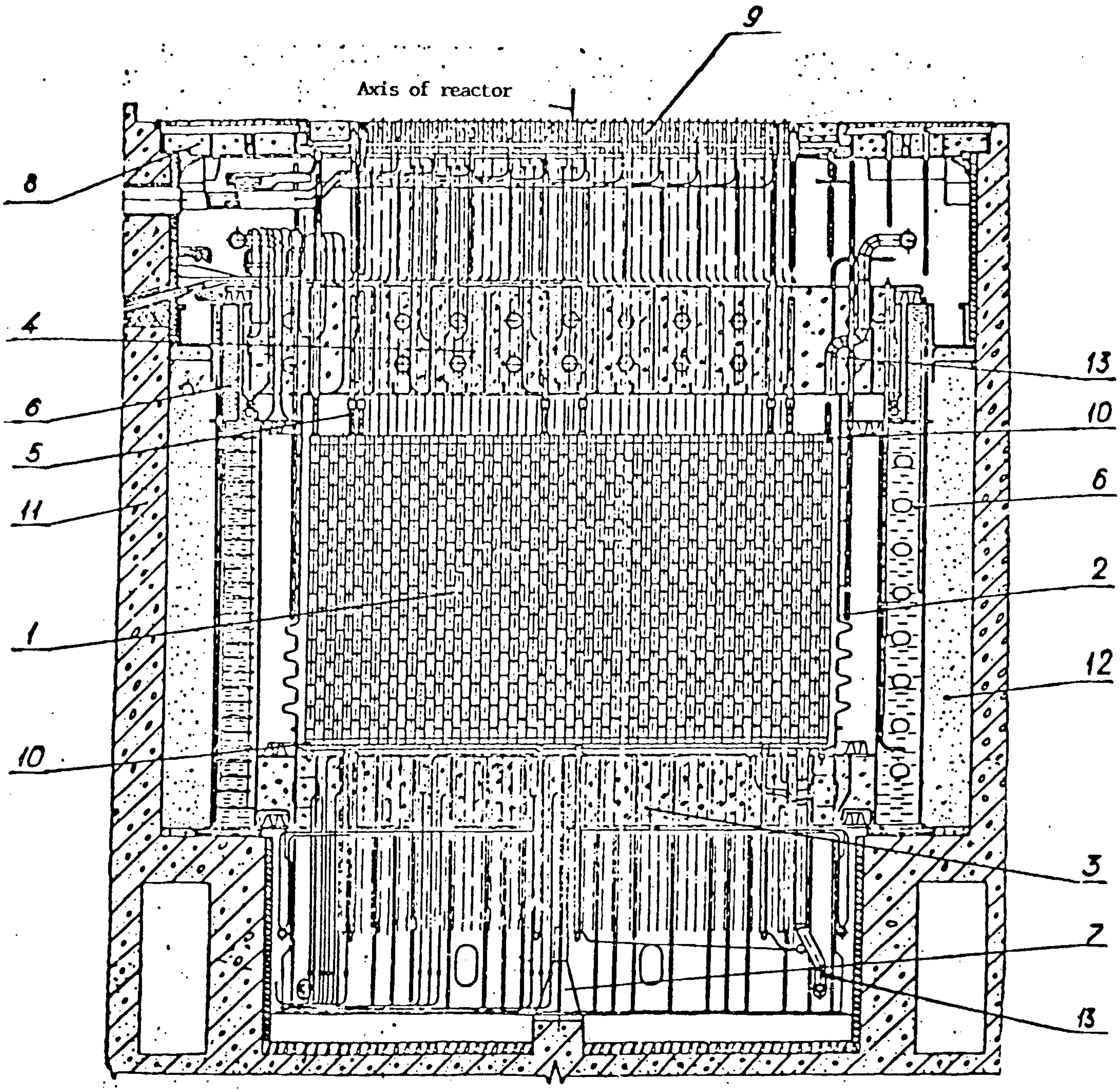


Fig. 2.1

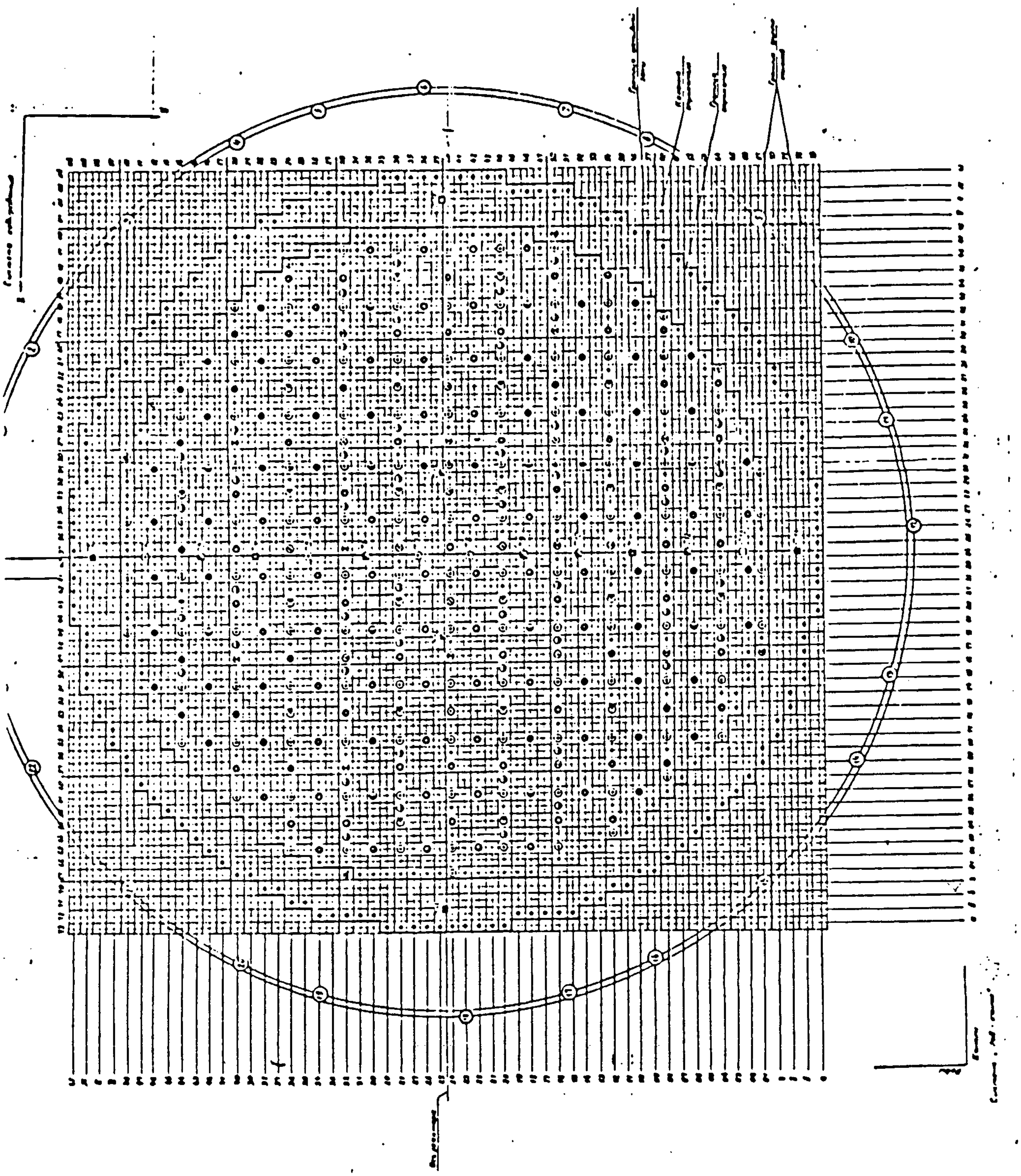


Fig. 2.Ia

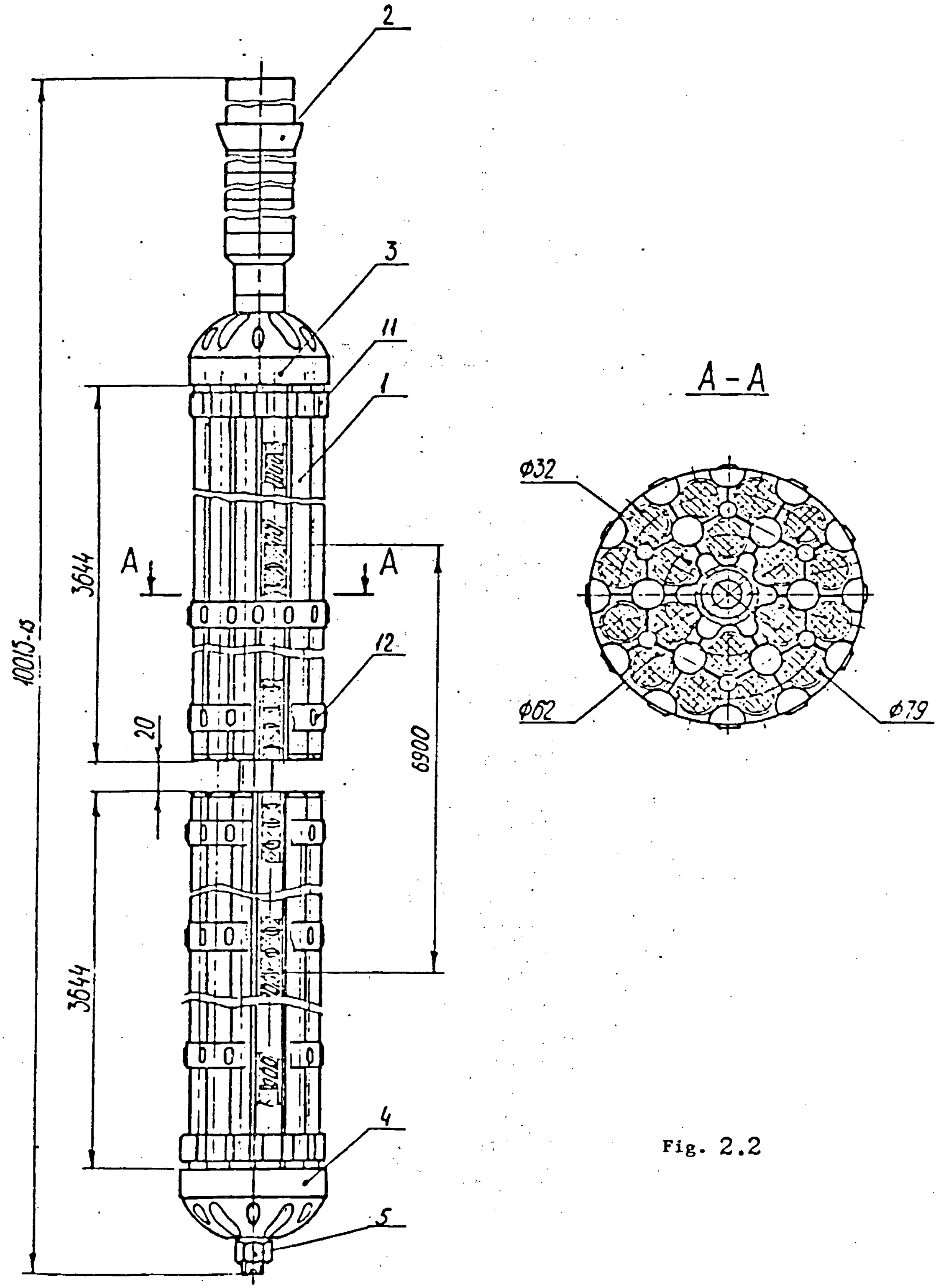


Fig. 2.2

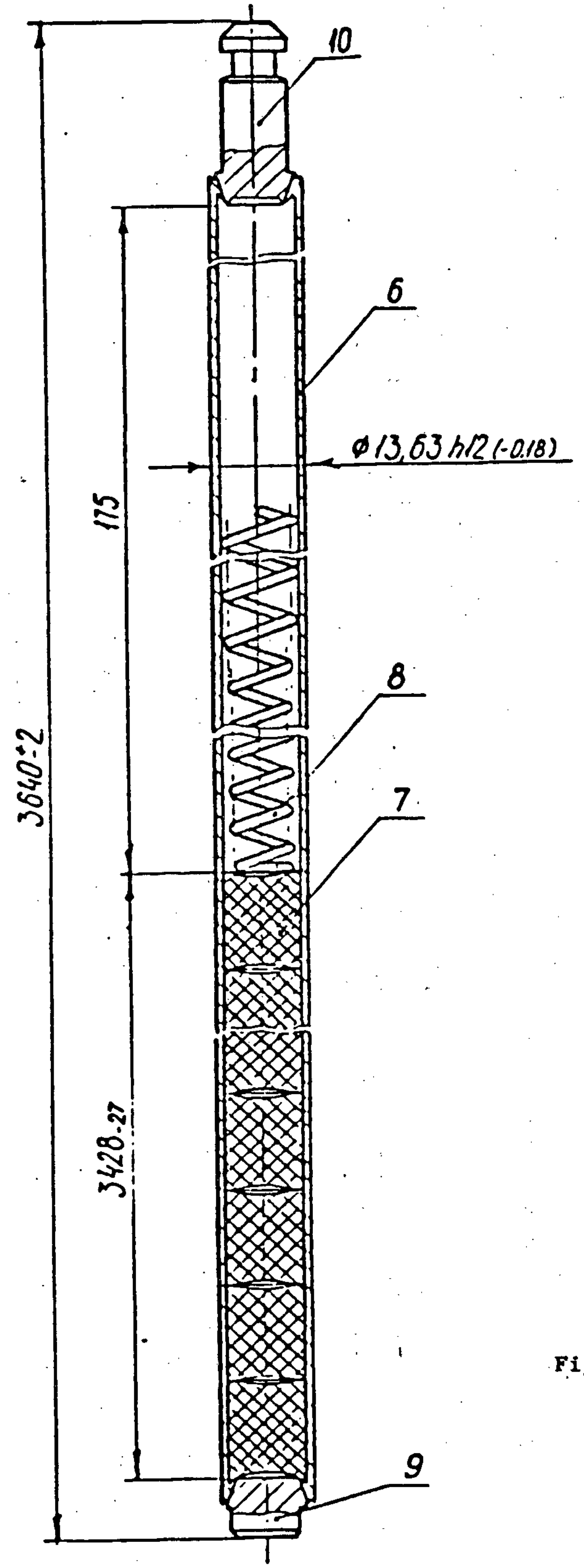
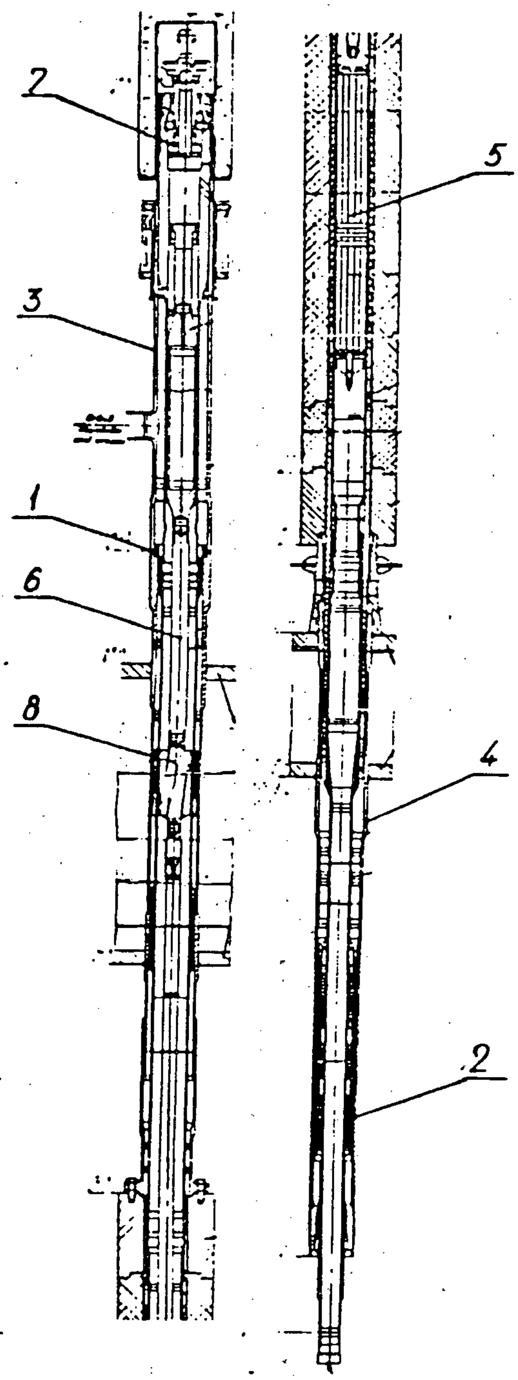


Fig. 2.2a

RBMK fuel channel



Water feed

Fig. 2.3

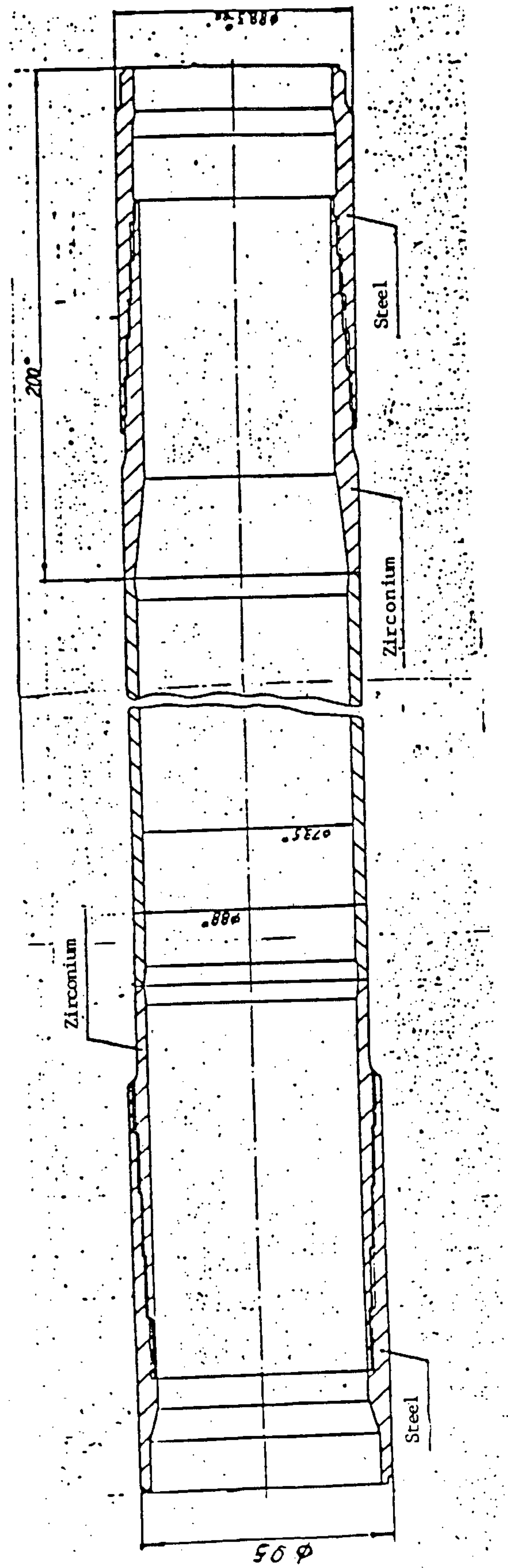


Fig. 2.3a

Declassified and Approved For Release 2013/07/15 : CIA-RDP09-00997R000100260001-0

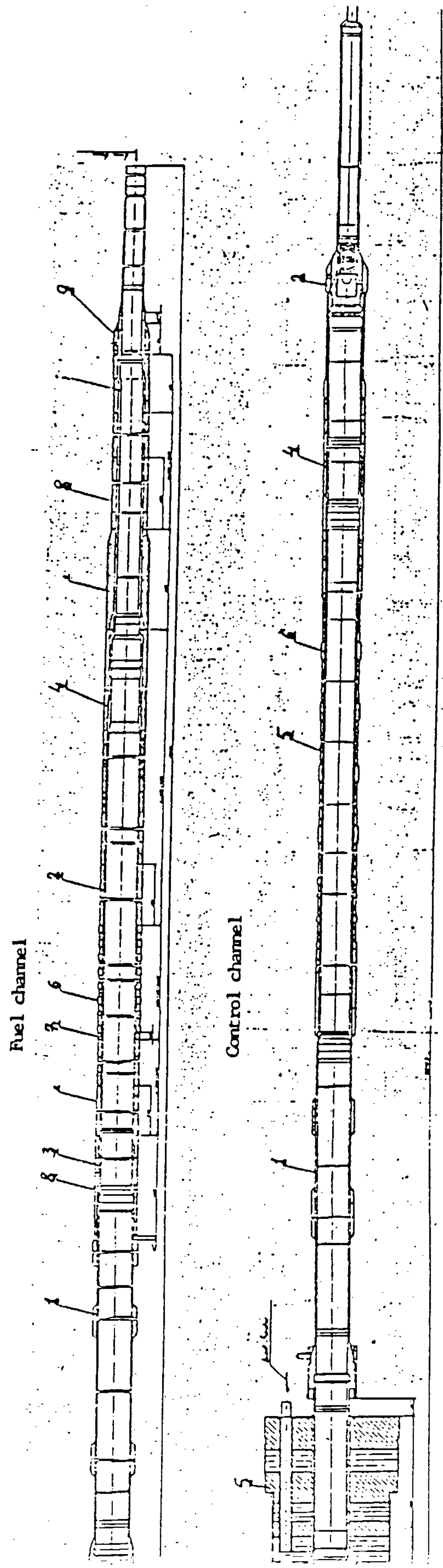


Fig. 2.4

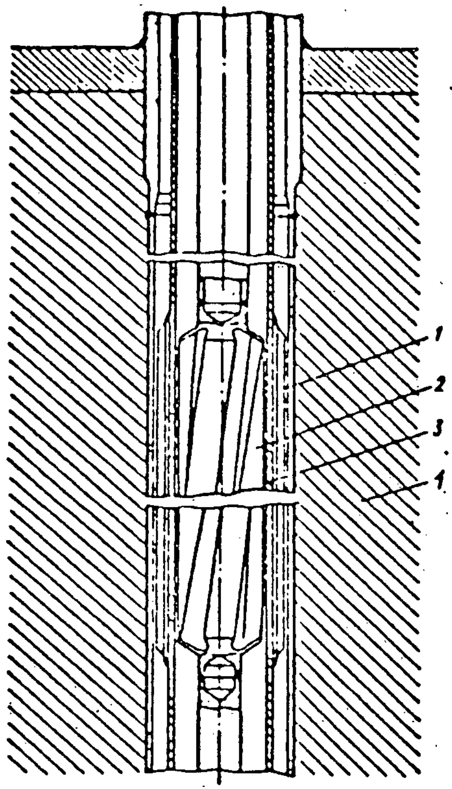


Fig. 2.5. Position of shielding plug in fuel channel: (1) steel sleeves; (2) helical steel plug; (3) channel tube; (4) serpentinite filling

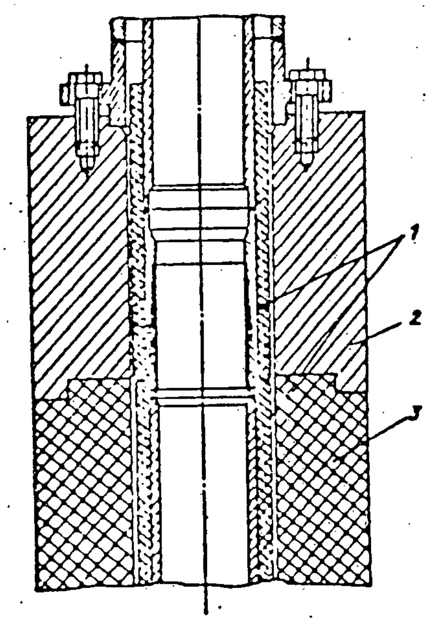


Fig. 2.6. Arrangement of shielding sleeves in upper reflector area: (1) graphite sleeves; (2) steel shielding block; (3) graphite reflector

2.2. Primary circuit (multiple forced circulation circuit)

(Fig. 2.6) [?]

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

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

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

2.3. Special control channel cooling circuit

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

2.4. The gas circuit

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

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

2.5. Basic physics data

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

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

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

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

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

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

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

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

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

The main parameters determining the limiting thermal load of the fuel channel and element are the critical channel power N_c^{cr}, at which departure-from-nucleate boiling occurs on the surface of fuel elements, causing overheating of the fuel cladding, and secondly, the maximum permissible linear load to the fuel element q_1^{cr}, above which the dioxide fuel melts.

In order to estimate anticipated N_c and q_1 values in the reactor, a probabilistic method for determining possible deviations was used which takes into account different factors influencing the limiting values of N_c and q_1, including the accuracy of measurement and maintenance of reactor power as a whole and its distribution over the core (the coefficients of inhomogeneity over the core radius C_r and height C_z and also over fuel elements in an assembly C_{ass} determining the maximum theoretical channel power M_c^{max} and linear load to the fuel element q_1^{max}). A Gaussian distribution was assumed for random deviations in maximum power from the most probable value of N_c^{max}. The limiting channel power is determined from the relationship:

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

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

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

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

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

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

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

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

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

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

These include the following:

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

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

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

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

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

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

2.5.1. Basic thermal physics data

[See end of this section for list of symbols]

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

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

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

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

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

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

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

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

2.5.1.2. Thermal physics characteristics in steady-state operating regimes

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

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

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

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

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

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

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

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

for the actual volume steam content in the channel:

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

and for the coefficient of phase transition:

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

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

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

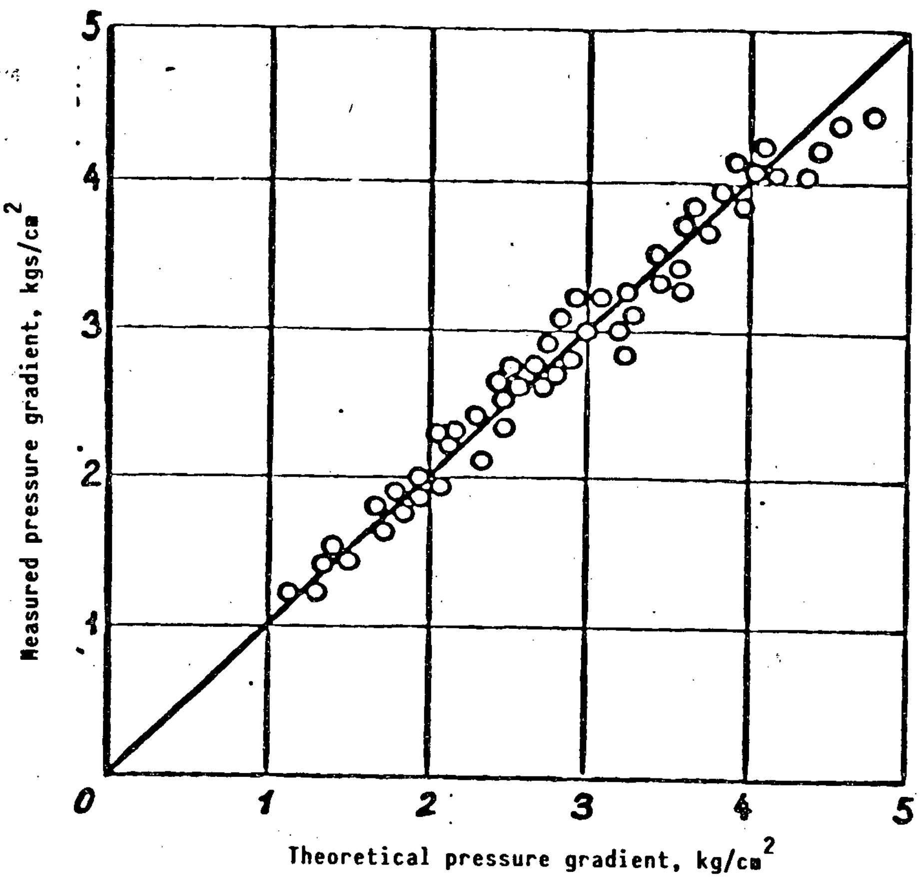


Fig. 2.8. Comparison of experimental and theoretical values for hydraulic resistance of a full-scale test rig.

o – experimental values

___ – theoretical values

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

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

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

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

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

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

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

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

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

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

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

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

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

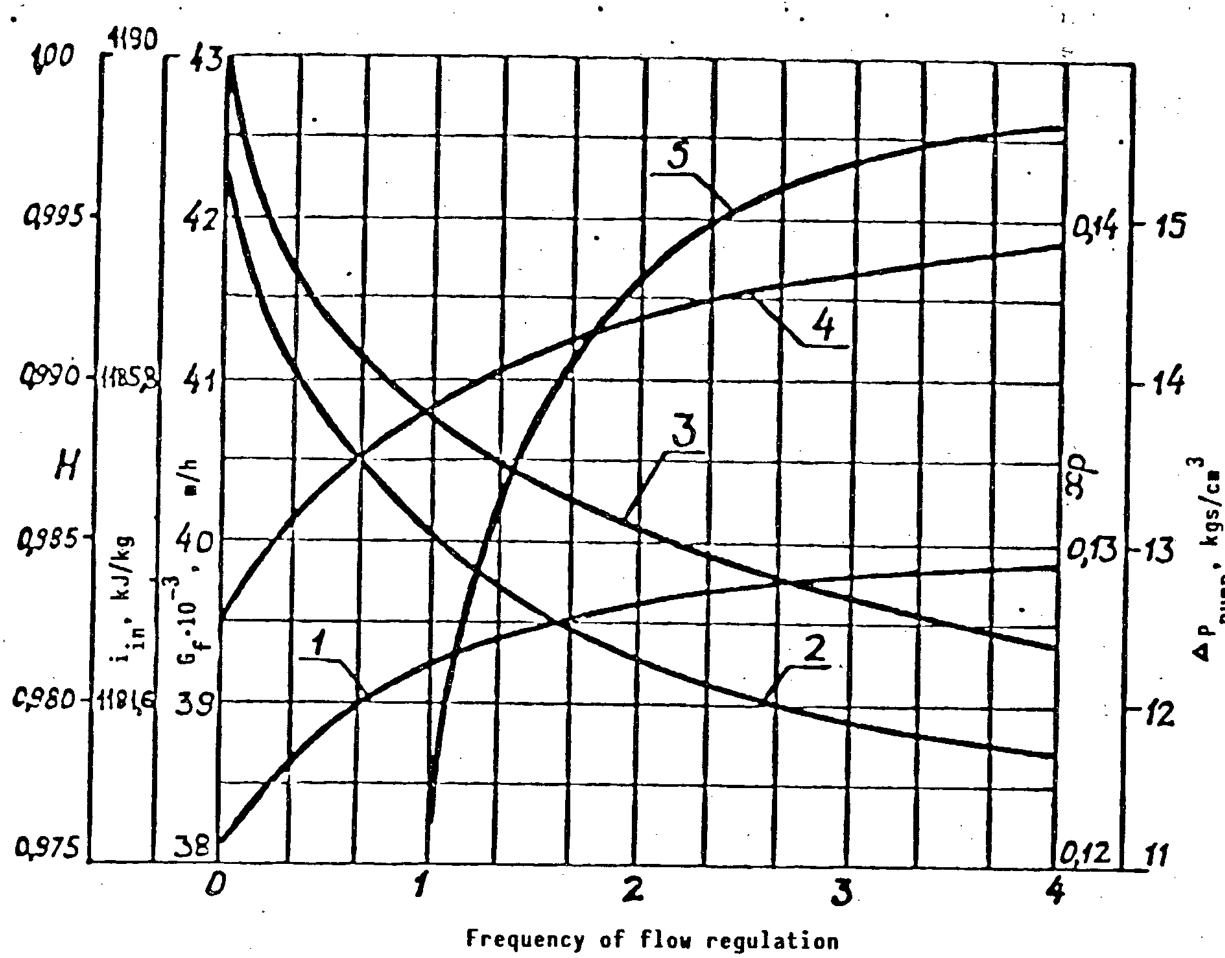


Fig. 2.9. Reactor parameters as a function of frequency of individual channel flow regulation:

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

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

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

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

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

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

Symbols used in Section 2.5.1 (Basic thermal physics data)

G – flow rate, kg/s;

S – cross-sectional area, m^2;

d – diameter, m;

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

x – mass steam content;

ρ – density, kg/m^3;

P – pressure, kgs/cm^3

q – thermal flux density, kW/m^2;

w – velocity, m/s.

Superscripts and subscripts

he – heated;

g – hydraulic;

cr – critical;

/ – water on saturation line;

// – steam on saturation line.

2.6. TECHNOLOGICAL LAYOUT OF THE UNIT

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

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

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

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

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

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

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

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

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

- Emergency core cooling system;

Chemical composition of primary coolant circuit materials

Material	C	Mn	Si	S	P	Ni	Cr	Cu	Nb	Mo
Steel "Krezelso" 330E	≤0,23	0,9-I,2	0,2-0,4	≤0,025	≤0,025	<0,3	≤0,4	≤0,3	–	–
Steel 1CL473Nb	<0,05	≤0,2	≤0,75	≤0,02	≤0,035	8,5-IO,5	I8-20	8x% C-0,65	–	0,6

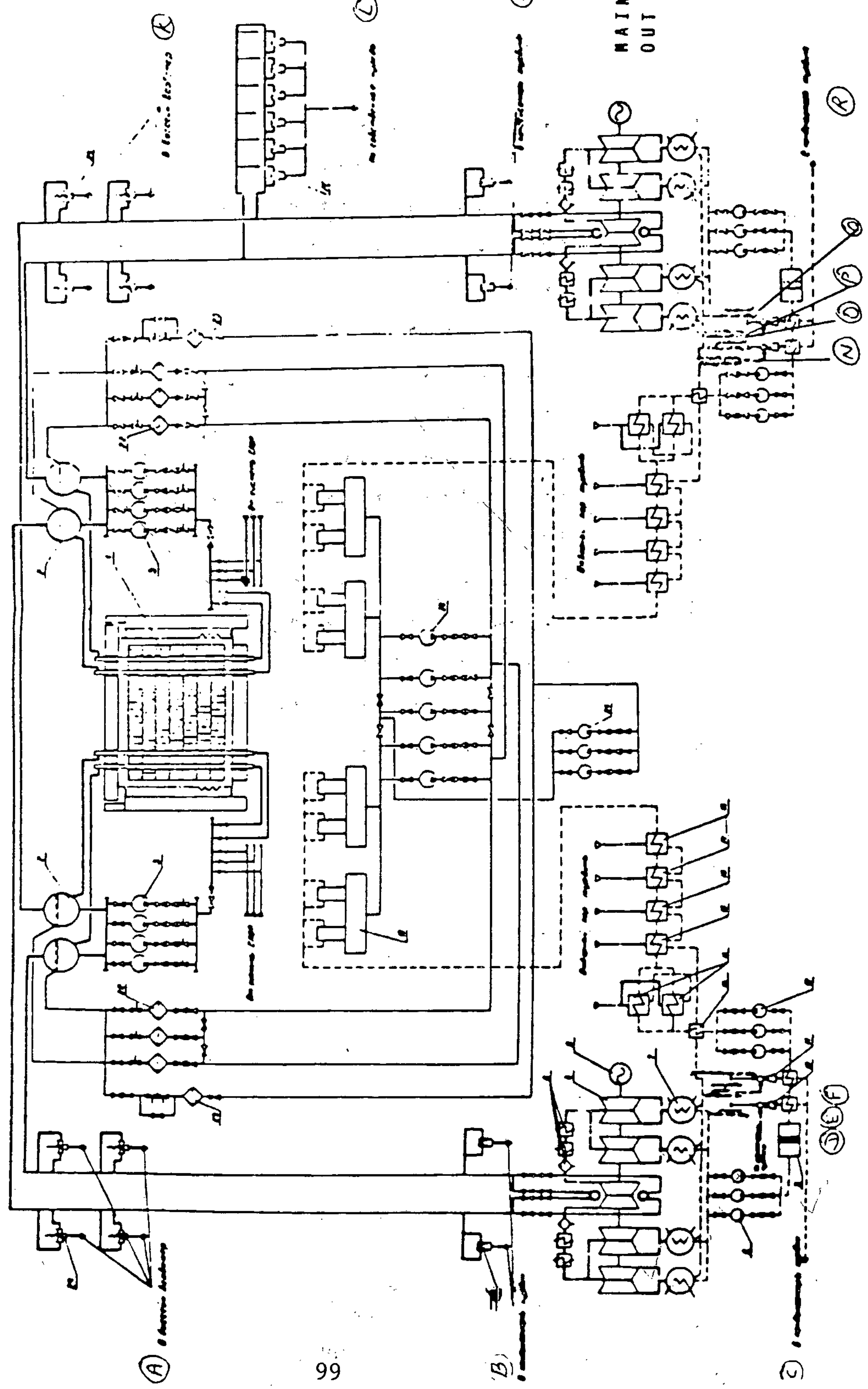
Physical and mechanical properties of the materials

Material	At temperature, °C	$\alpha \cdot 10^6$ 1/°C	E, N/mm²	σ_B kgf/mm²	$\sigma_{0,2}$ kgf/mm²	δ_5 %	ψ, %	a_{HI} kgf/cm²	a_{IV}, kgf/cm²
Steel "Krezelso" 330E	20	II,I	205200	44-60	≥22	≥20	≥48	≥7	≥4
	350	I5,0	I88000	≥36	≥I9	≥I8	≥43	–	≥3
Steel 1CL473Nb	20	I6,5	I93000	≥50	≥20	≥38	≥50	–	–
	350	I7,5	I79000	≥36	≥I5	≥24	≥40	–	–
08Cr18Ni10Ti	20	I6,4	205000	52	22	35	55	–	–
	350	I7,6	I75000	42	I7	26	5I	–	–

MAIN THERMAL LAYOUT OF THE UNIT

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

SECTION 2.6.



MAIN THERMAL LAY-OUT OF THE UNIT

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

Primary coolant circuit

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

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

- Reactor fuel channels;

- Steam-water communication line pipes.

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

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

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

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

Primary coolant circuit flushing and shutdown cooling system

The purpose of this system is:

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

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

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

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

2.6.1. Gas circuit. Condenser and filter station

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

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

The mixture then enters the condenser and filter station.

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

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

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

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

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

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

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

2.6.2. Channel tube failure detection system

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

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

- Identify damaged reactor channels;

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

- Dry the reactor graphite stack.

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

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

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

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

2.6.3. Helium purification plant

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

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

The main technical characteristics of the helium purification plant are:

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

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

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

2.6.5. Cooling system for biological shield tanks

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

2.6.6. Cooling system for control and protection system channel

Function and design bases

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

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

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

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

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

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

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

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

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

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

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

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

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

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

2.6.7. Intermediate circuit system of the reactor section

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

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

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

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

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

2.6.8. Water regime

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

- Reduce the amount of contamination getting into the reactor core;
- Prevent the deposition on core components of impurities contained in the water.

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

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

- pH value: 6.5-8.0;
- Specific conductance: not greater than 1.0 μohm/cm;
- Hardness: not more than 10 μg-equiv./kg;
- Silicon acid: not more than 100 μg/kg;
- Chloride-ions + fluoride-ions: not more than 100 μg/kg;

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

The quality of feedwater must meet the following requirements:

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

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

The main active water treatment plants include the following:

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

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

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

- Transfer of resin to the solid and liquid waste storage tank;
- Preparation of decontaminating solutions;
- Reuse of decontaminating solutions;
- Equipment decontamination.

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

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

2.6.9. Bypass purification plant for MFCC flushing water

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

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

The system components are:

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

2.7. Main equipment of the unit

Reactor

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

Turbine

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

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

Steam separator

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

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

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

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

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

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

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

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

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

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

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

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

The technical data on the steam separator are:

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

- Basic dimensions of the separator:

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

De-aerator

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

Main circulation pump

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

The main characteristics of the pump are:

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

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

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

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

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

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

Feed pump

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

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

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

The main characteristics of the unit are:

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

Condensate pumps

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

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

The basic characteristics of this unit are:

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

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

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

The basic characteristics of the unit are:

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

Pipelines

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

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

Refuelling machine

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

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

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

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

- Mechanized control of some accident situations.

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

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

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

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

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

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

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

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

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

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

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

2. A simulator rig intended for:

   - Adjustment and checking of the machine's mechanisms;

   - Filling of the pressure equalization chamber with condensate;

   - Simulation of regular refuelling;

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

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

   - Replacement of the inflatable collars of the connecting sleeve.

   The simulator rig has the appropriate equipment for these operations.

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

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

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

2.8. Control and protection system

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

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

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

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

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

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

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

- The CPS electrical power supply system.

2.8.1. Location of the main CPS equipment items

The hangers for the neutron flux detectors are located:

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

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

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

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

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

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

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

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

- The servo drive temperature monitoring rack.

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

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

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

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

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

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

2.8.2. Neutron flux monitoring

The monitoring ranges are shown in Fig. 2.12.

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

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

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

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

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

2.8.3. Automatic regulating of the reactor power

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

The equipment in all three automatic regulator sets is identical.

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

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

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

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

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

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

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

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

- 0.0075% N_{nom} per second in the range 0.5-1% N_{nom};
- 0.0125% N_{nom} per second in the range 1-6% N_{nom};
- 0.15% N_{nom} per second in the range 5-20% N_{nom};
- 0.25% N_{nom} per second in the range 20-100% N_{nom}.

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

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

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

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

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

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

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

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

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

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

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

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

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

2.8.4. Emergency protection of the reactor

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

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

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

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

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

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

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

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

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

- A power overshoot of 10% N_{nom};
- A reduction in the period to 10 s;
- A drop or excess in the level in the drum separators of either half;
- A drop in the feedwater throughput;
- A pressure excess in the drum separators of either half;
- A pressure excess in the leaktight compartments, drum separators or lower water lines;
- A pressure excess in the reactor cavity;
- A fall in the level in the CPS coolant tank;
- A reduction in water throughput through the CPS channels;
- Trip of two turbogenerators, or of the only operating turbogenerator;
- Trip of three of the four operating main circulation pumps in any pump room;
- Voltage loss in the plant auxiliary power supply system, or indication of one of the protection level regimes (3, 2 or 1) without its being carried out, or order from the command units (protection level 5 button, declutching key) at the control desk and at a number of other locations in the plant.

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

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

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

- Outage of one of two turbogenerators;

- Emergency load rejection of one of two turbogenerators.

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

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

- When the water throughput in the primary circuit falls;

- When the feedwater throughput falls;

- When the water level in the drum separators falls;

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

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

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

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

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

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

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

- By the startup-range speed protection system from 4×10^{-7} to 5×10^{-2} N_{nom};
- By the working-range speed protection system from 10^{-5} to 1.2 N_{nom}.

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

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

- By the low-power power protection system in the range from 0.005 to 0.1 N_{nom}, when a power level given by the ZM power transducer is exceeded by 0.005 N_{nom} (by 0.5% N_{nom}), as detected by not less than two out of four of the low-power protection system channels;
- By the working-range power protection system in the range from 0.06 to 1.2 N_{nom}, when the set power level is exceeded by 0.1 N_{nom} (10% N_{nom}), as detected by two out of the eight working-range power protection system channels; in this case there must be an emergency protection signal in at least one channel of each of the two groups of four working range channels.

Each working-range power protection system channel consists of:

- A UZM.11 amplifier (power protection system);

- An ionization chamber: KNK-56 in the low power range and KNK-53M in the working range;

- A BP.39 ionization chamber power supply block;

- A KrT.5 chamber current corrector.

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

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

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

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

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

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

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

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

Figure 2.11:

1. Shortened absorber rods

2. Manual regulating and emergency protection rods

3. Local automatic regulating system rods

4. Automatic regulator rods

5. Servo drives

6. Servo drive control relay contact block - shortened rods

7. Servo drive relay control contact block - manual rods

8. Servo drive control block - local automatic regulating system

9. Power control block - automatic regulator system

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

11. Individual rod control circuit

12. Shortened rods manual rods local automatic regulating system

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

14. Automatic process equipment

15. Signalling system

16. Protection level 5

17. Local emergency protection system

18. Local automatic regulating system on and in working order

19. Power transducer setting control

20. Low-power protection system

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

22. KNK-53M

23. KNK-56

24. KNT-31

25. KNK-53M

26. KV.2

27. KV.5M

28. BP.38

29. BP.38
ZRTA reactimeter

30. Power recorder (N)

31. Reactivity recorder (ρ)

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

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

34. Current indicator   Period indicator

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

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

37. Period indicator

38. Current indicator (g)

39. Power recorder (N) on logarithmic scale

40. Period indicator

41. Speed indicator

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

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

44. BP.39

45. Power recorder (N) at standby control desk

46. Working-range speed protection system

47. Startup-range speed protection system

48. Automatic regulator 1 on and in working order

49. Power protection system

50. Automatic regulator 2 on and in working order

51. Indicators at control desk and board

52. CPS logic

53. Indicators at control desk and board

54. Power transducer setting

55.)
56.) Unbalance, power protection amplifier

57. Unbalance, deviation signal amplifier

58. Local automatic regulating system on

59. BT_{Σ}

60. Synchronized drive block

61. Trigger block

62. BT_{Σ}

63. Power transducer

64. Synchronized drive block

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

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

67. Power protection amplifier

68. Current corrector

69. KtV.17

70. Deviation signal amplifier

71. Control key

72. Power supply block

73. Unbalance, power protection amplifier

74. Unbalance, deviation signal amplifier

75. Power transducer setting

76. BT.37

77. ZM.9

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

79. USM.12

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

81. KNK.56, KNK.53M

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

83. USM.12

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

85. Power transducer setting

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

87. BT.37

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

89. KrU.4

90. ZM.9

91. BSP.36

92. KrU.4

93. ZM.9

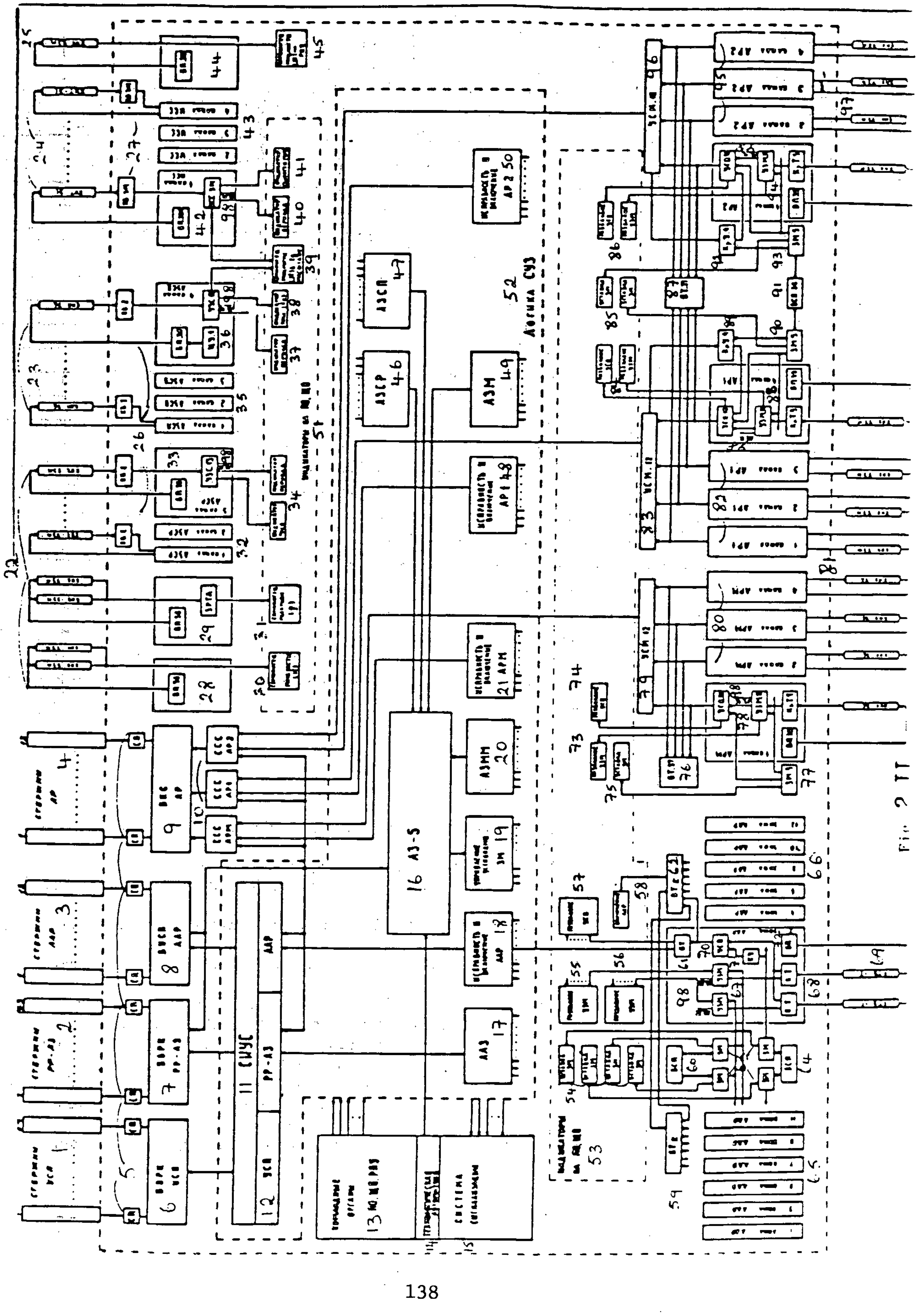
94. Automatic regulator 2, channel 1
USO.10
UZM.11
KrT.5
BP.39

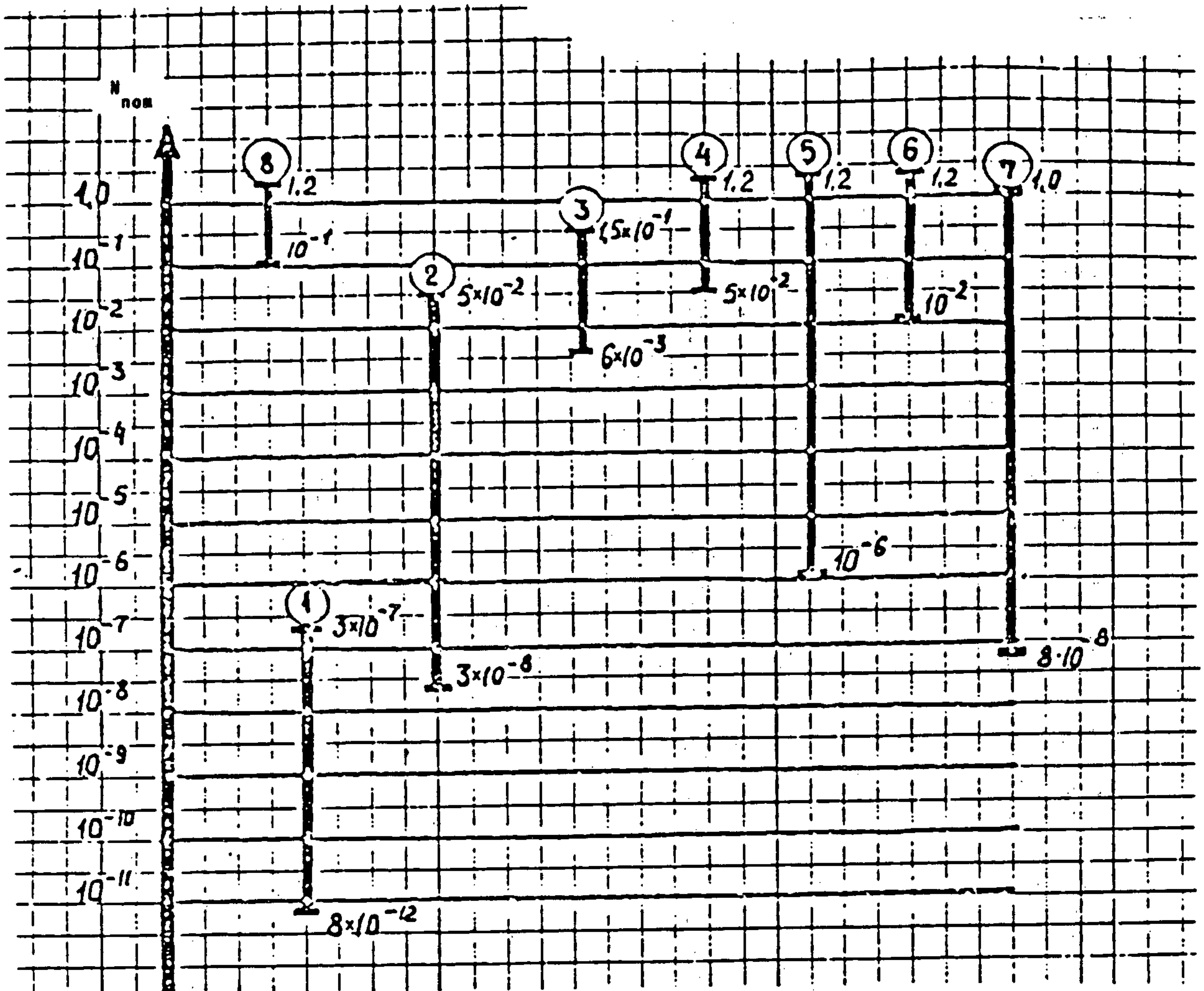
95. Automatic regulator 2, channel 2, channel 3, channel 4

96. USM.12

97. KNK.53M

98. Alarm and emergency protection systems





Neutron flux monitoring ranges.

(1) Range of power monitoring on logarithmic scale by ISS.3M units wit KNT-31 chambers.

(2) Range of power monitoring on logarithmic scale by UZS.13 (spee protection in startup range) units with KNK-56 chambers (with lea screen).

(3) Range of power monitoring by low-power automatic regulator using KNK-5 chambers.

(4) Range of power monitoring by automatic regulator 1 (2) using KNK-53 chambers.

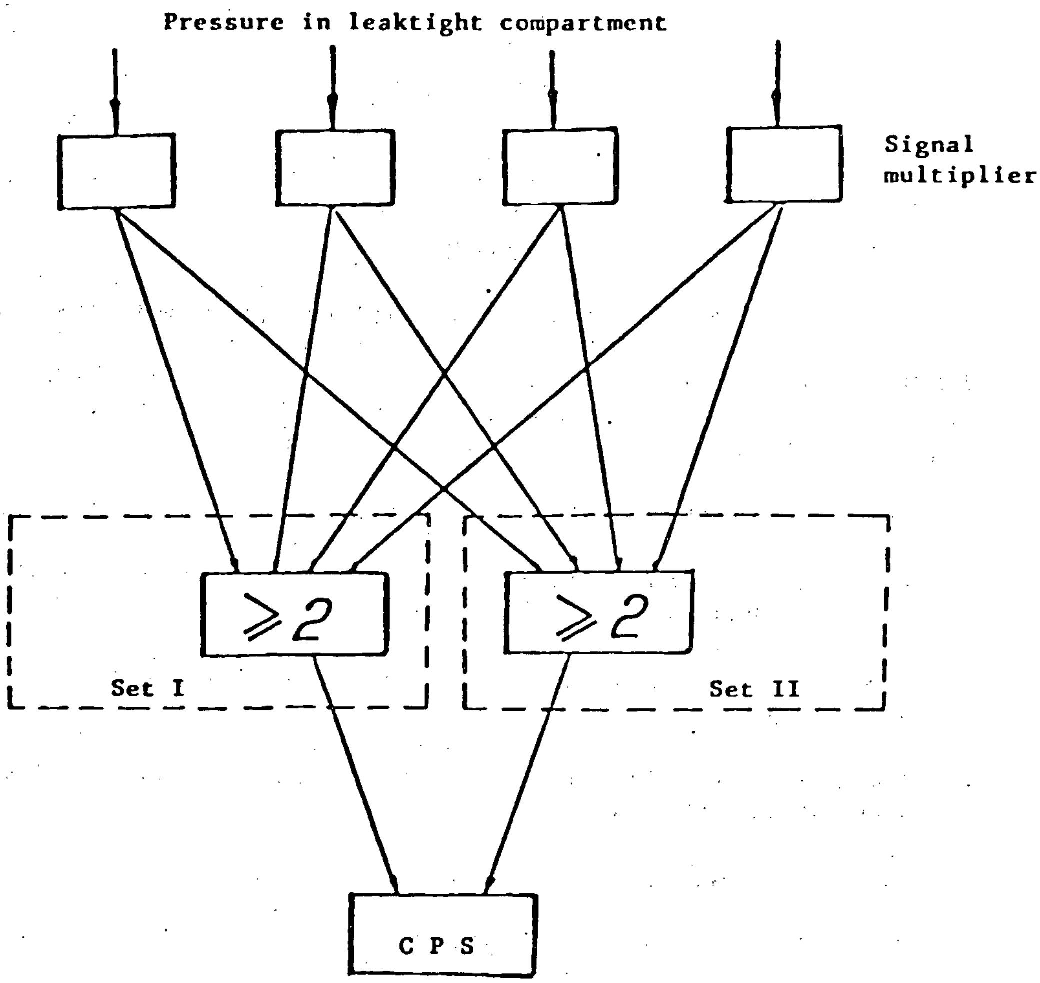
(5) Range of power monitoring on logarithmic scale by UZS.13 (spee protection in working range) units with KNK-53M chambers.

(6) Range of power monitoring on linear scale by automatic recorder a standby control desk using KNK-53M chamber.

(7) Range of power monitoring on linear scale by automatic recordin potentiometer at control and instrumentation board using KNK-53 chambers.

(8) Range of power monitoring by local automatic regulator channels usin KtV.17 chambers.

Fig. 2.12.



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

Fig. 2.I3 Structural diagram of protection system for process parameters.

2.9. Reactor process monitoring system

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

The following systems relate to the process monitoring system:

- Channel-by-channel coolant flow rate monitoring in the process and control channels;

- Temperature monitoring of the graphite stack and the metal structure;

- Channel integrity monitoring from the temperature and humidity of the surrounding gas;

- Physical power density monitoring system;

- Fuel cladding failure detection;

- "Skala" central monitoring system.

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

The reactor has the following monitoring points:

- Fuel channel flow rate measurement: 1661 points;

- Control channel flow rate measurement : 227 points;

- Temperature measurement of the metal structure and biological shielding: 381 points;

- Measurement of the graphite stack and plates: 46 points

- Radial and vertical power measurement: 214 points;

- Gas temperature measurement: 2044 points;

- Measurement of coolant activity: 1661 points.

2.9.1. Coolant flow rate monitoring

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

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

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

Section 2.9.2.

Temperature Monitoring

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

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

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

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

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

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

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

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

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

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

2.9.3. Process channel integrity monitoring

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

Detection of non-hermetic reactor channels;

Containment of the spread of humidity from the damaged channel;

Ventilation of the reactor space.

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

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

2.9.4. Fuel element cladding integrity monitoring

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

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

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

2.9.5. Monitoring of the multiple forced circulation circuit

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

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

2.9.6. The "Skala" central monitoring system

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

(1) Number of monitoring signals:

Analogue signals	7200
Discrete signals	6500

Signals are accepted from:

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

Platinum and copper resistance thermometers;

Tachymetric ball flowmeters;

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

(2) Monitoring periods:

Mass parameters: 1-5 min.

Calculated parameters: 30 min.

(3) Functions:

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

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

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

(4) Operation time to failure:

- Monitoring functions: 1×10^4 hours;

- Calculating functions: 2×10^3 hours.

(5) Electric power required: 95 kW.

2.9.7. System for physical monitoring of the power density distribution

Purpose and structure of the system

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

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

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

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

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

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

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

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

measuring devices of the power density monitoring equipment;

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

scanning instruments of the power monitoring equipment.

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

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

measuring device of the power density monitoring equipment;

scanning instruments of the monitoring equipment.

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

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

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

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

The monitor verification system includes:

calibration detectors of the radial power monitor type;

tri-axial calibration fission chambers;

annular ionization chamber;

measuring equipment of the monitor verification system.

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

Radial power density monitors

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

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

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

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

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

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

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

$$W_i = K_{2pi} \cdot \xi_D(I_i) \cdot \xi_{TD}(E_i) \cdot J_i , \qquad (1)$$

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

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

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

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

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

- The emitter breaks off and as a result there is no current at the monitor connecting joint;
- The monitor readings in the "Skala" computer system are rejected when making operative calculations with the "Prizma" program;
- There is a drop in the monitor's sensitivity, allowing for burn-up, of more than 15% between two calibrations;
- There are rapid fluctuations in the monitor's signal that are not confirmed by the neighbouring monitor readings;
- The resistance of the monitor's insulation drops below 100 kohm.

Vertical power density monitors

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

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

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

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

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

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

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

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

$$(nv_o)_{ij} = K_{zp\,ij} \cdot \xi_D(I_{ij}) \cdot J_{ij} \,, \qquad (2)$$

where: n is the neutron density;

v_0 = 2200 m/s;

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

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

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

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

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

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

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

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

The resistance of the section insulation drops below 100 kohm.

Power density monitoring equipment

Purpose and composition of equipment

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

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

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

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

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

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

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

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

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

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

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

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

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

Principal specifications of the equipment

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

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

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

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

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

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

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

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

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

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

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

$$U_{ав i} = \frac{1}{K_{\phi i}} \cdot \frac{\delta_{ав}}{100} \cdot U_{100}, \qquad (5)$$

and a "warning" signal threshold:

$$U_{пр i} = \frac{1}{K_{\phi i}} \cdot \delta_{пр} \cdot \overline{U}, \qquad (6)$$

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

$$U_{зон i} = \frac{1}{K_{\phi i}} \delta_{зан} \cdot \overline{U}, \qquad (7)$$

where: $K_{\phi i}$ is the coefficient of transmission for the comparator unit amplifier for the i-th monitoring line and is fixed smoothly within the limits 0.6-2.5;

δ_{ab} is the coefficient of transmission for the "emergency level %" divisor and is established discretely in percentage of the nominal power between 0 and 100%;

v_{100} is the voltage corresponding to nominal power level and is regulated smoothly from 1 and 5 V;

δ_{np} is the relative "warning" signal level and is established smoothly between 0.65 and 1.25;

δ_{3aH} is the relative "undershoot" signal level is established smoothly between 0.65 and 1.25;

$\bar{v}$ is the arithmetic mean of the radial monitor output signals.

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

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

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

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

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

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

"Emergency" - $$U_{a\ell\, ij} = \frac{1}{K_{\phi ij}} \cdot \delta_{a\ell} \overline{U}_j, \tag{10}$$

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

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

where: $K_{\phi ij}$ is the transmission coefficient for the comparator unit amplifier in the i-th monitoring line of the j-th vertical monitor and is fixed smoothly between 0.6 and 2.5;

δ_{ab}, δ_{np}, δ_{3aH} are the corresponding relative signal levels and are set smoothly between 0.75 and 1.85;

$\bar{v}$ is the arithmetic mean of the output signals from the sections of the j-th vertical monitor.

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

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

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

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

1. Functions to be carried out and structure of calculations

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

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

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

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

The input data for the "Prizma" program are:

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

2.9.8.1. Periodicity and accuracy of calculations

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

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

2.9.8.2. Basic theoretical relationships

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

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

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

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

The power density distribution is calculated from the following relationships.

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

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

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

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

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

where: N_D is the number of properly working monitors;

K_{Tpj} is the calibration coefficient for the j-th monitor;

ξ_{Dj}, ξ_{TDj} are coefficients taking into acount burnup of the monitor emitter and the fuel in the monitor channel.

For assemblies with monitors one calculates the ratios:

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

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

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

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

For assemblies with monitors the values:

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

are calculated.

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

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

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

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

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

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

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

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

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

Calculation of the power margin coefficient

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

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

where: W_i is the power of the fuel assembly;

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

$$c_{1i} = 1 - \varkappa^2 (D_i + D_{Tn2}); \qquad c_{2i} = W_{kpi}^2 (1 - \varkappa^2 D_{Tn1}) - \varkappa^2 D_{Tn3}; \tag{23}$$

where: χ is the normal distribution quantile corresponding to the set probability that the assembly will experience critical heat flow;
D_i is the dispersion of the relative error determining the assembly power;
D_{Tn1} is the dispersion of the relative error in determining the critical power of the assembly (methodological);
D_{Tn2} is the dispersion of the error in determining the reactor power;
D_{Tn3} is the dispersion of the determination of W_{kri} through errors in measuring the flow, pressure and temperature.

Calculation of the graphite temperature

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

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

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

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

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

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

The thermocouple readings which stand out strongly are processed.

Calculation of the in-core monitor settings

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

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

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

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

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

Calculation of the operative reactivity reserve

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

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

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

2.9.9. Presentation of information on the calculation results

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

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

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

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

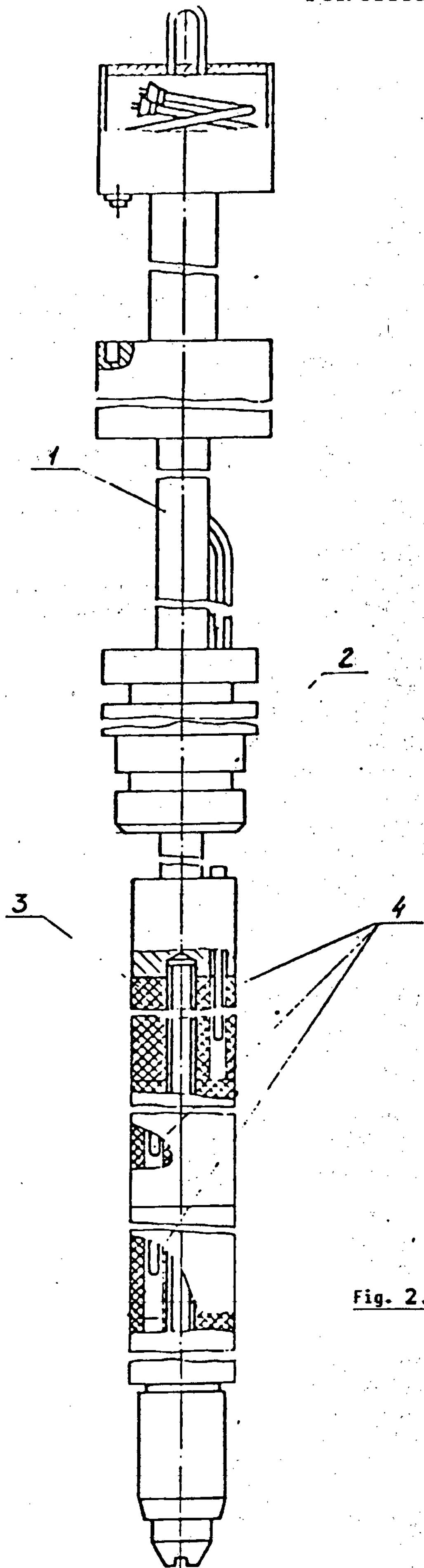


Fig. 2.36. Thermocouple unit and assembly:

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

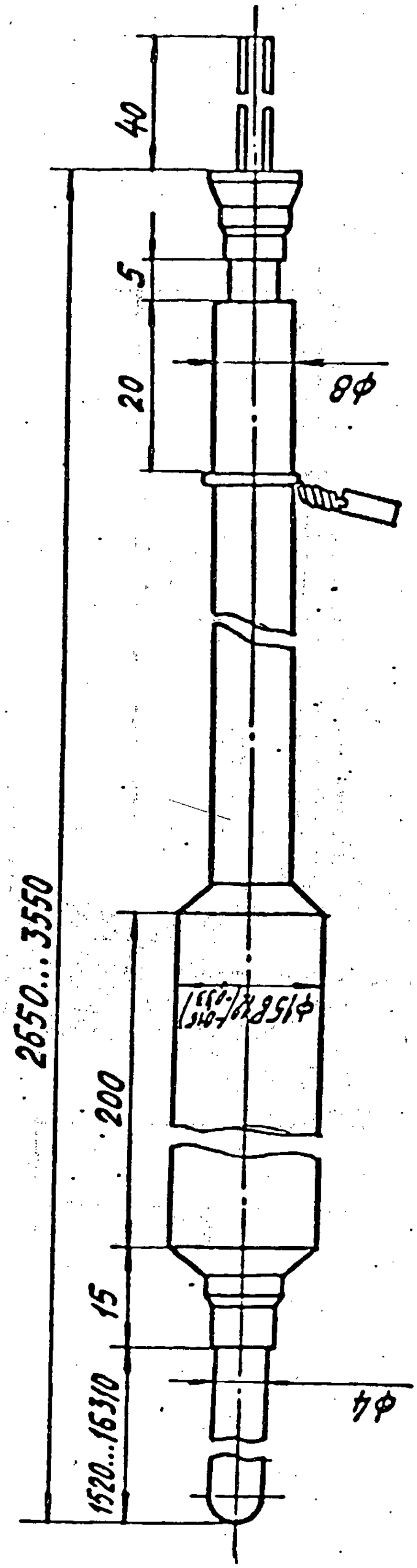


Fig. 2.37. Outline drawing of Chromel-Alumel cable-type thermocouple.

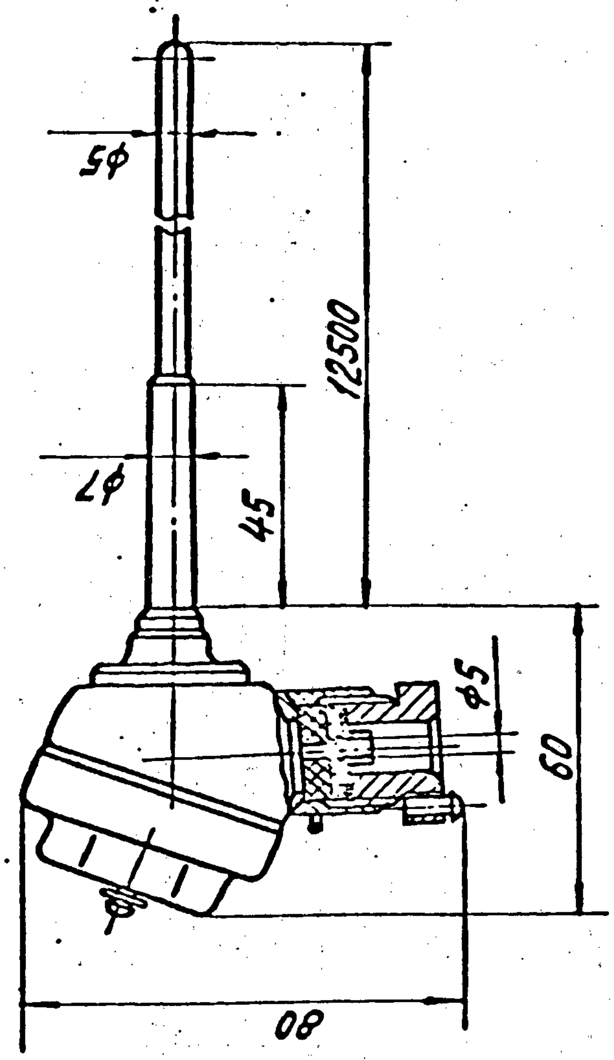


Fig. 2.38. Outline drawing of headed Chromel-Alumel cable-type thermocouple.

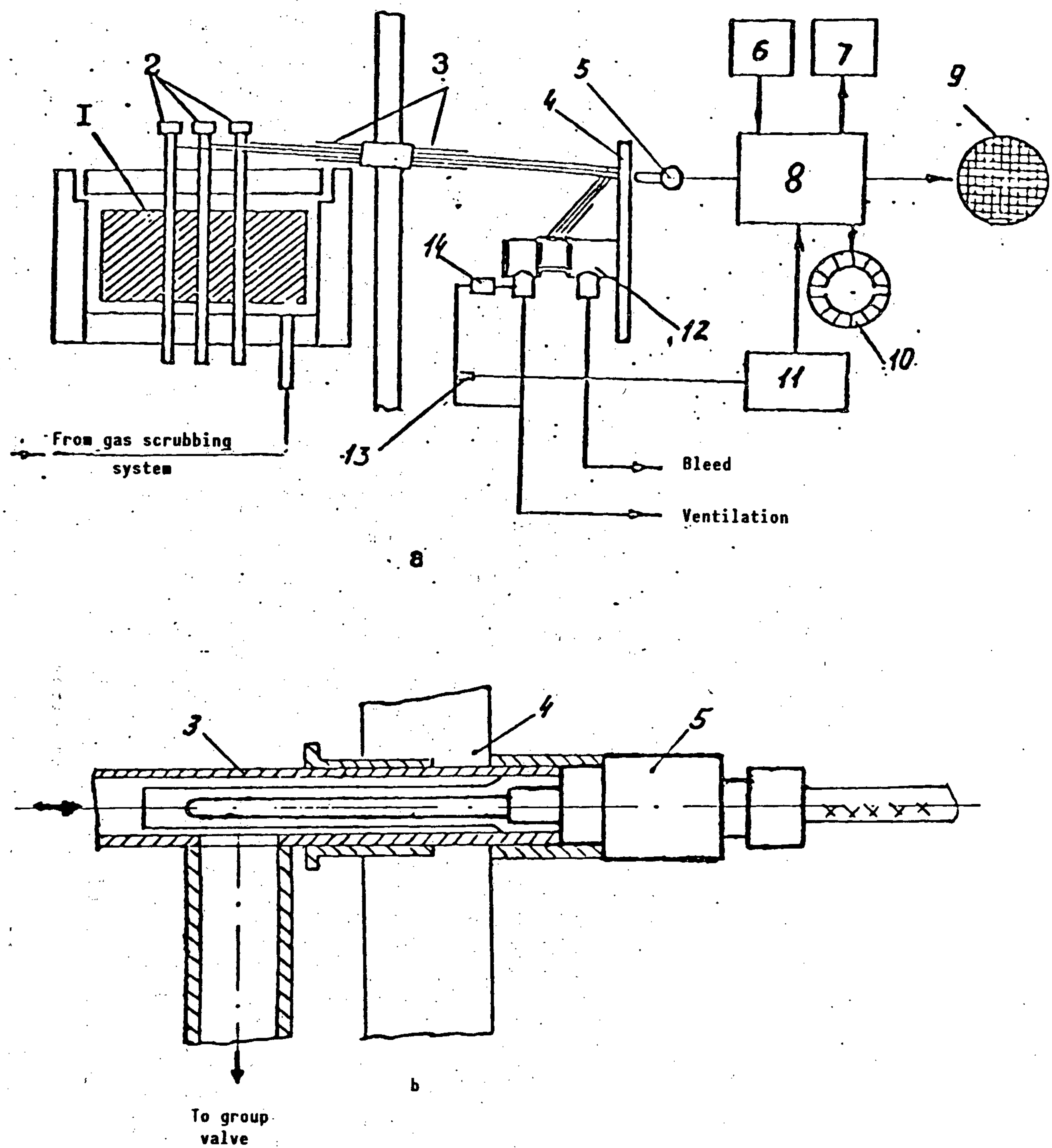


Fig. 2.39. (a) Process channel integrity monitoring system;
(b) Chromel-Copel temperature sensor unit.

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

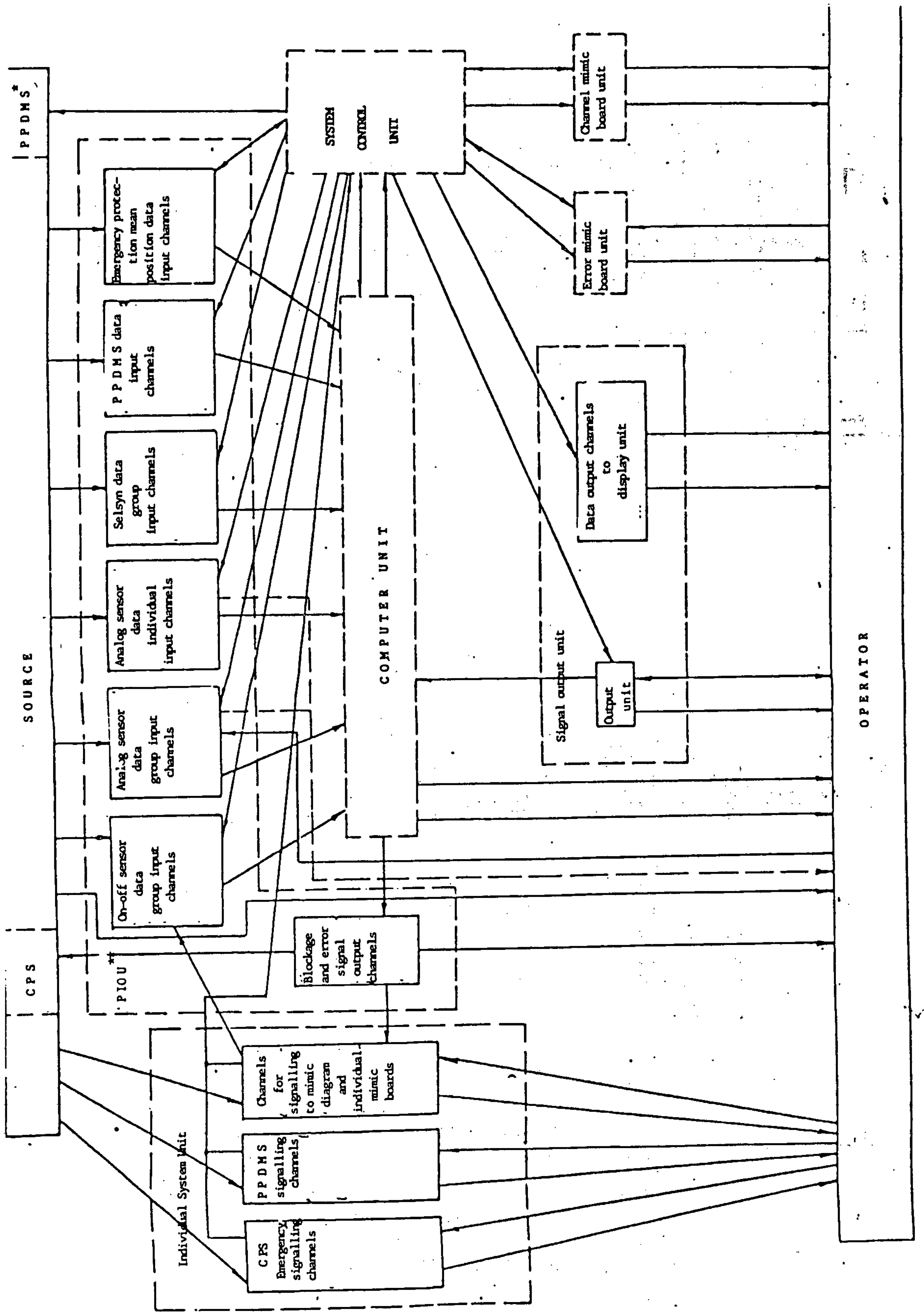


Fig. 2.40 STRUCTURAL DIAGRAM OF THE "SKALA" SYSTEM

*PPDMS = Physical power distribution monitoring system

**PIOU = Parameter input-output unit

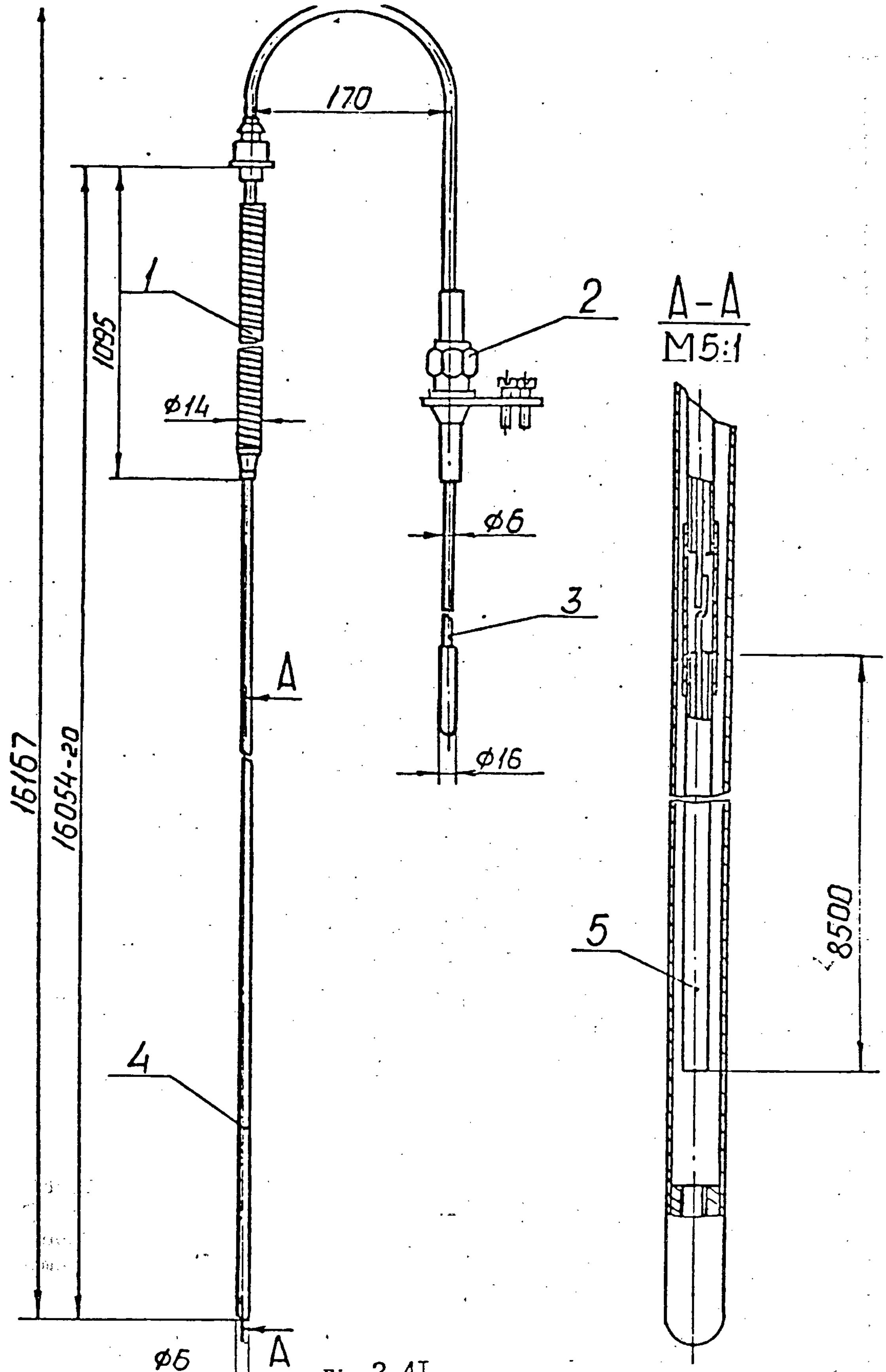


Fig.2.4I
Design of the radial power density monitor

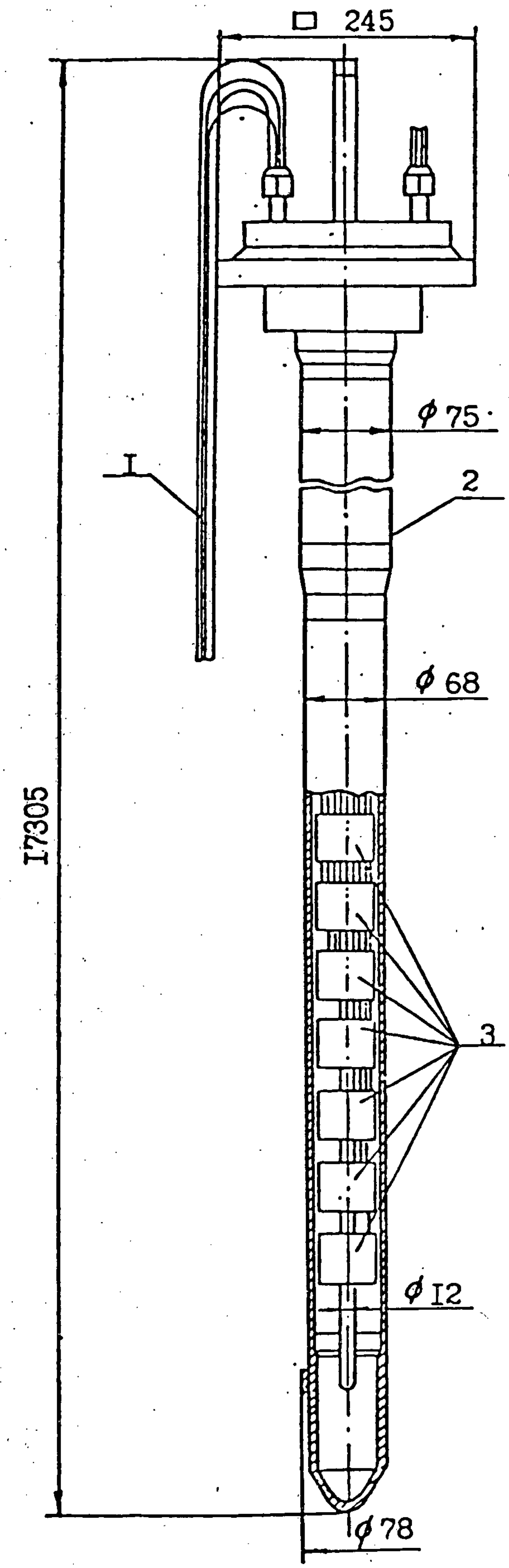


Fig. 2.42 Design of the vertical power density monitor: (1) cable; (2) leaktight tube; (3) sensitive elements

2.10. Safety systems

2.10.1. Protective safety systems

2.10.1.1. Reactor emergency cooling system

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

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

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

The ECCS was designed with the following requirements in mind:

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

The system satisfies the following main requirements:

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

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

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

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

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

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

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

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

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

The steam discharge systems include:

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

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

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

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

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

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

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

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

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

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

2.10.1.3. System to protect the reactor space from excess pressure

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

The system satisfies the following main requirements:

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

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

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

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

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

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

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

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

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

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

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

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

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

The protection system includes:

- Pressure control (underpressure) in the reactor space;
- Control of the level of steam-gas discharges in the compartment;
- Reliable drainage of the steam discharge pipes.

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

2.10.2. Safety based on confinement systems

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

2.10.2.1. System of sealed locations

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

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

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

- Location of the steam distribution corridor (5);

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

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

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

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

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

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

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

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

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

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

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

2.10.2.2. Penetrations and doors

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

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

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

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

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

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

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

2.10.2.3. Cut-off and sealing valves

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

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

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

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

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

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

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

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

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

The valves are closed automatically by DBA signals.

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

2.10.2.4. Condensation-type pressure suppression system

The purpose of this system is to condense steam formed:

- during an accident involving reactor circuit failure;

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

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

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

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

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

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

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

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

The sprinkler cooling system carries out the following functions:

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

- Cooling of the water volume in the condensation system.

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

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

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

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

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

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

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

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

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

2.10.2.5. Hydrogen removal system

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

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

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

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

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

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

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

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

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

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

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

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

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

2.10.3. Power-supply safety systems

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

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

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

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

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

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

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

The emergency power sources consist of the following:

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

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

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

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

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

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

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

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

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

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

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

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

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

2.10.3.3. Emergency power supply system

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

- Startup of diesel generator;
- Switching off of both section switches linking the working own-requirement 6 kV section with the emergency power supply section;
- Switching-off of the load on the 6 kV emergency power supply section ("clearing of the section");
- Blocking, by automatic stand-by startup of mechanisms connected to the particular emergency power supply section.

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

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

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

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

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

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

2.10.4. Controlling safety systems

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

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

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

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

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

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

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

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

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

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

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

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

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

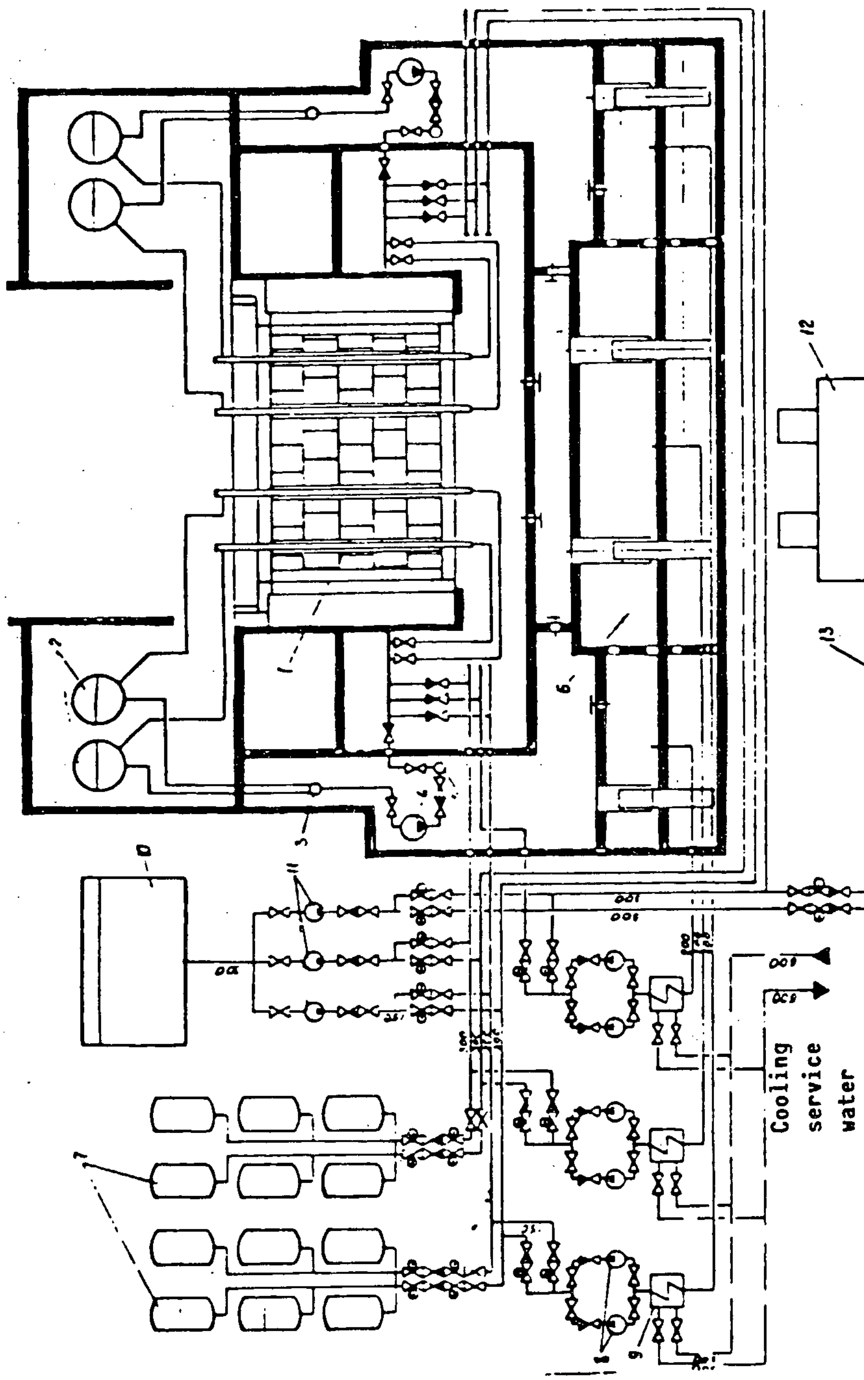
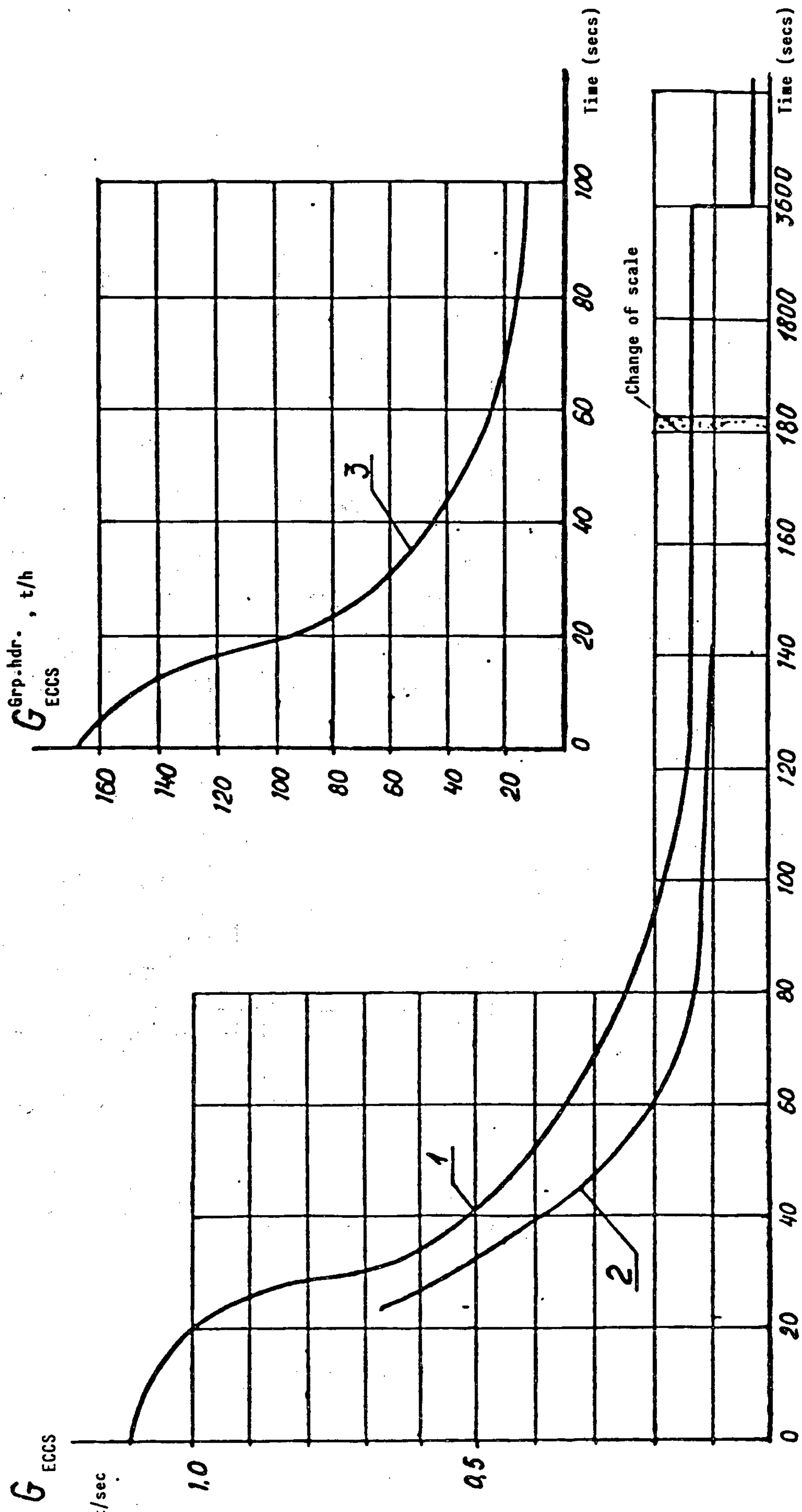


Fig. 243. Schematic drawing of the reactor's emergency cooling system.

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

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



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

Fig. 2.44

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

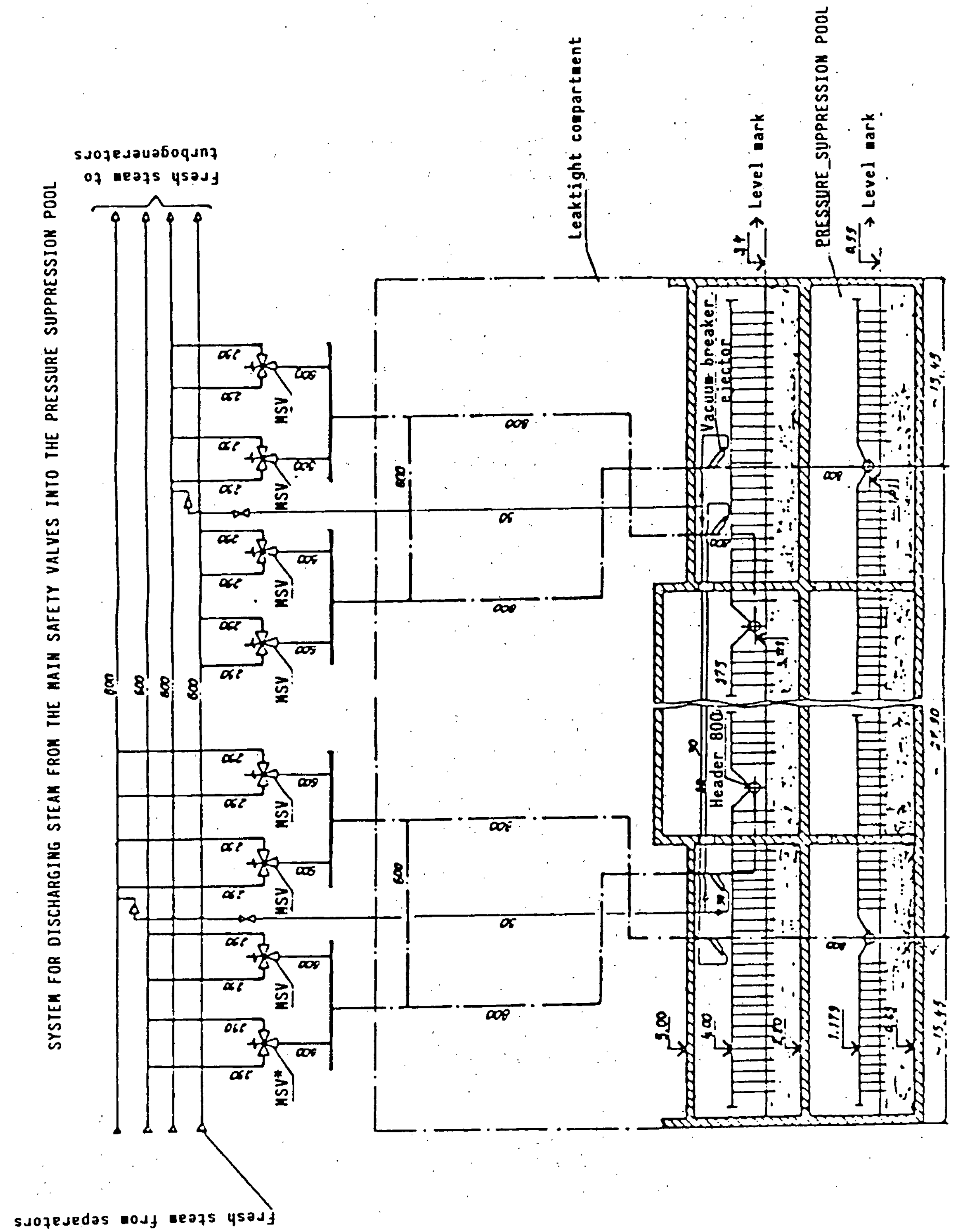


Fig. 2.45.

*/ MSV = main safety valve.

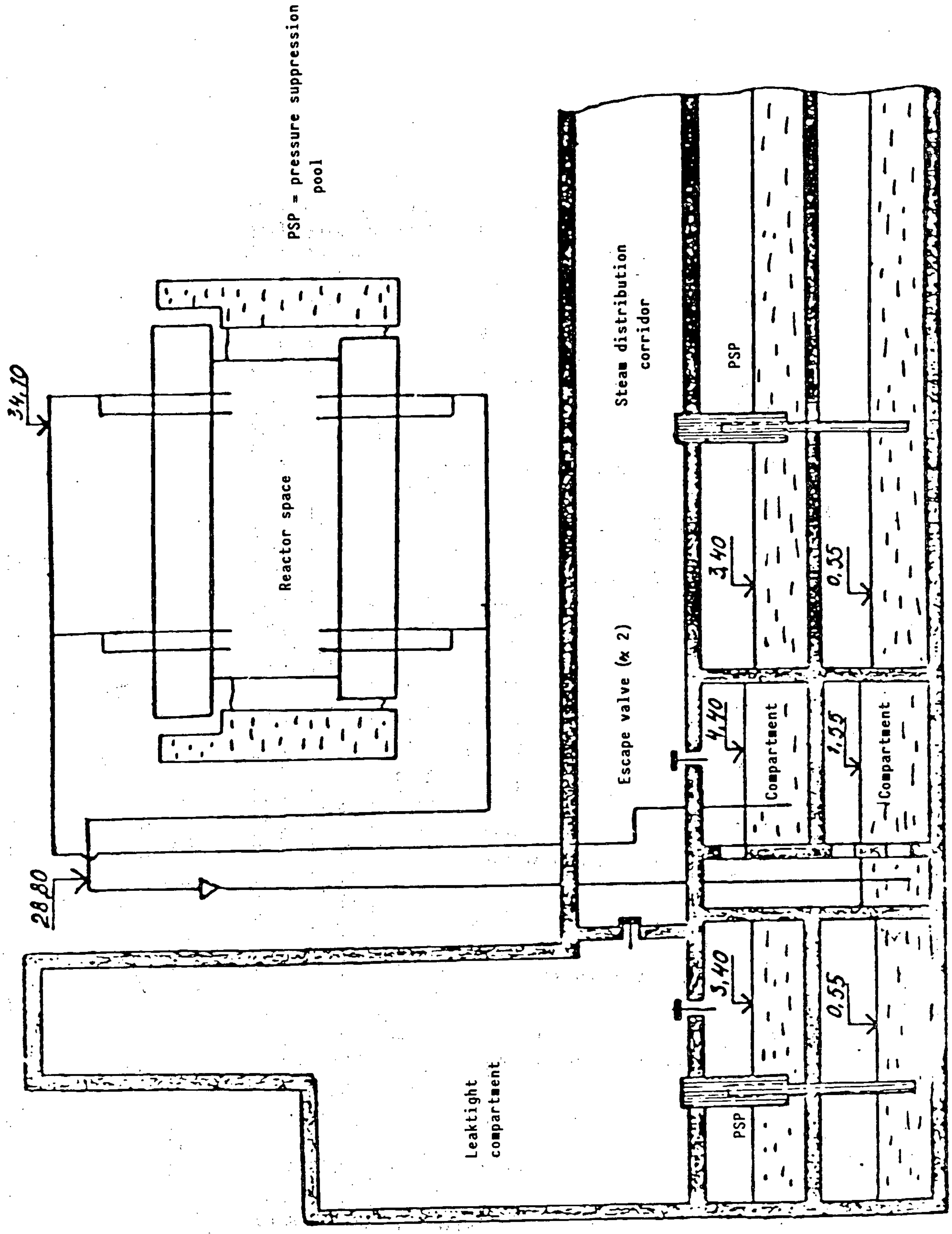


Fig. 2.46 : System to protect the reactor space from excess pressure

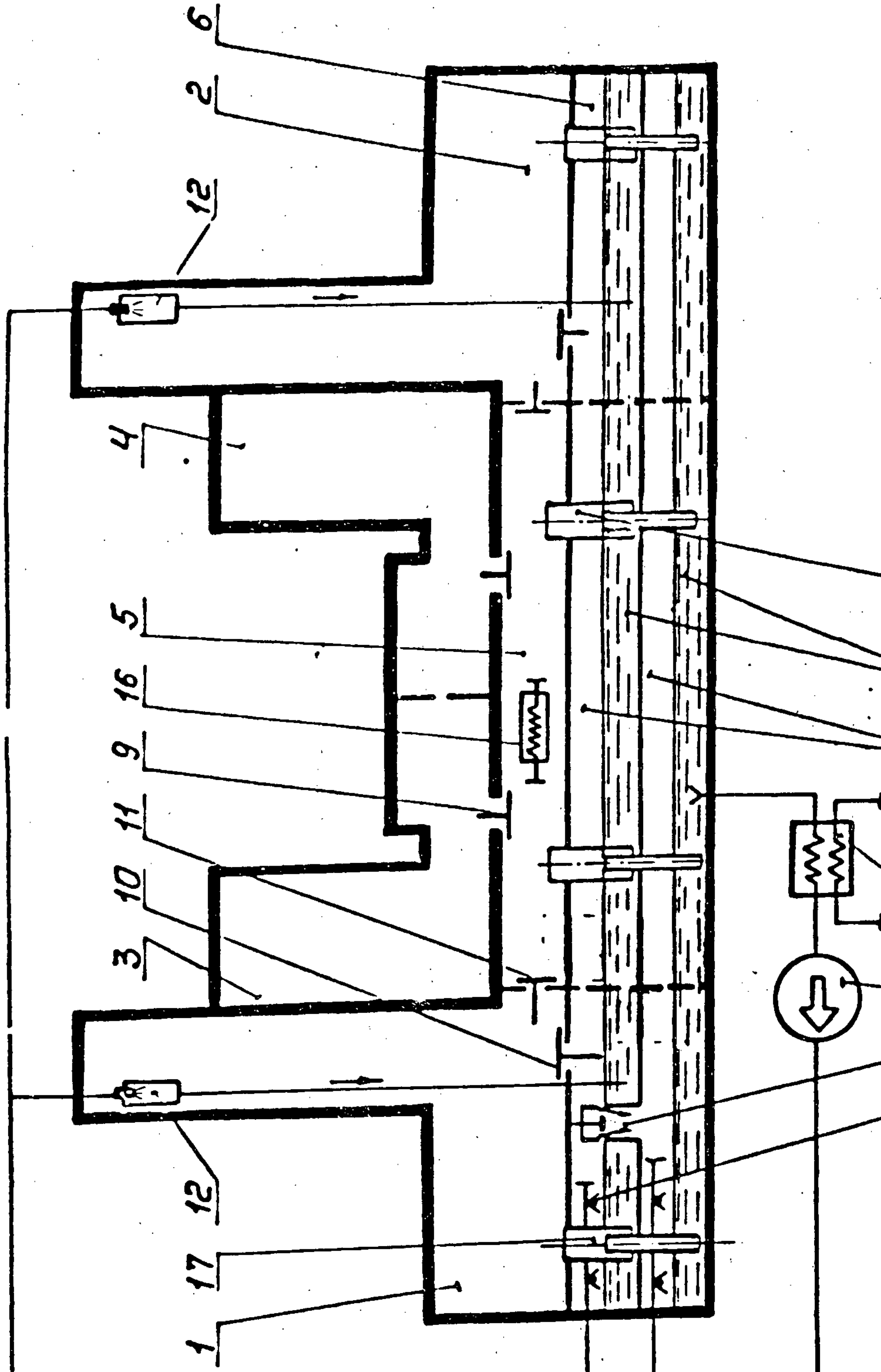
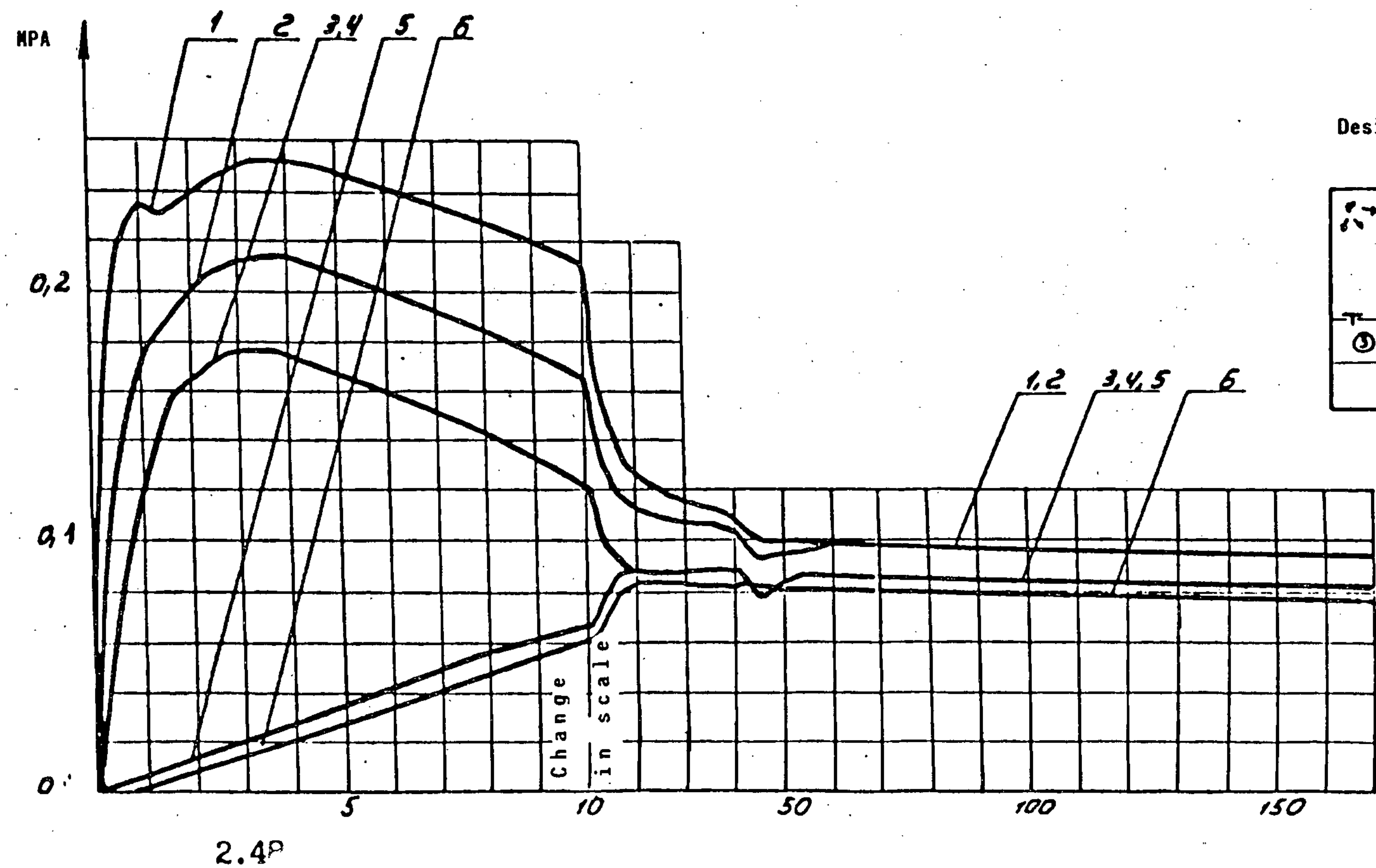
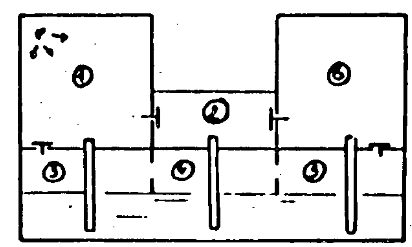


Fig 2.47 Schematic diagram of the confinement system

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



Design diagram of buildings



Design diagram of buildings

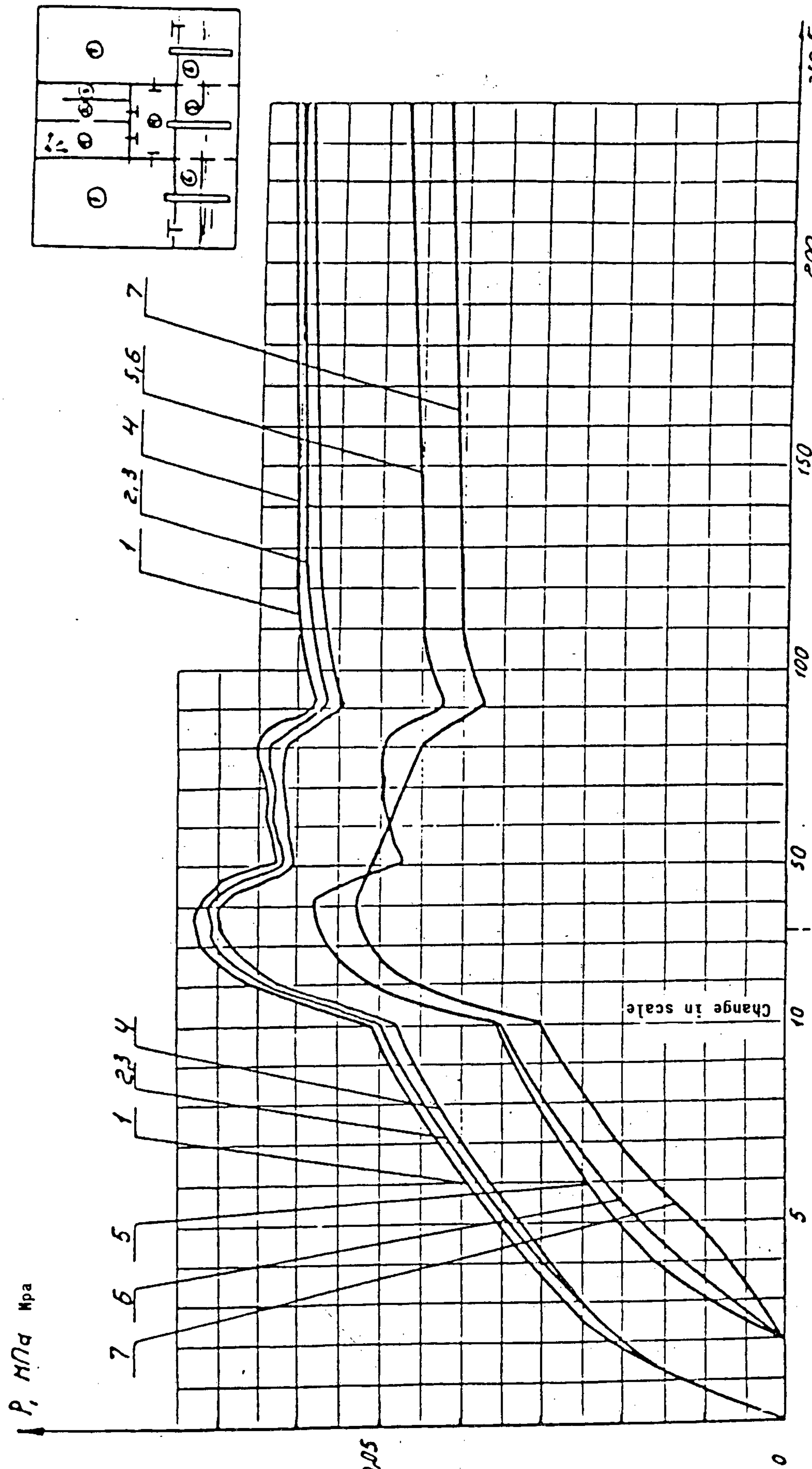


Fig 2.49 Pressure changes in the buildings of the second unit of the Smolensk nuclear power station and the fourth units of the Kursk and Chernobyl' nuclear power stations during failure of the disturbing group header

Entry of H_2 into the accident confinement area

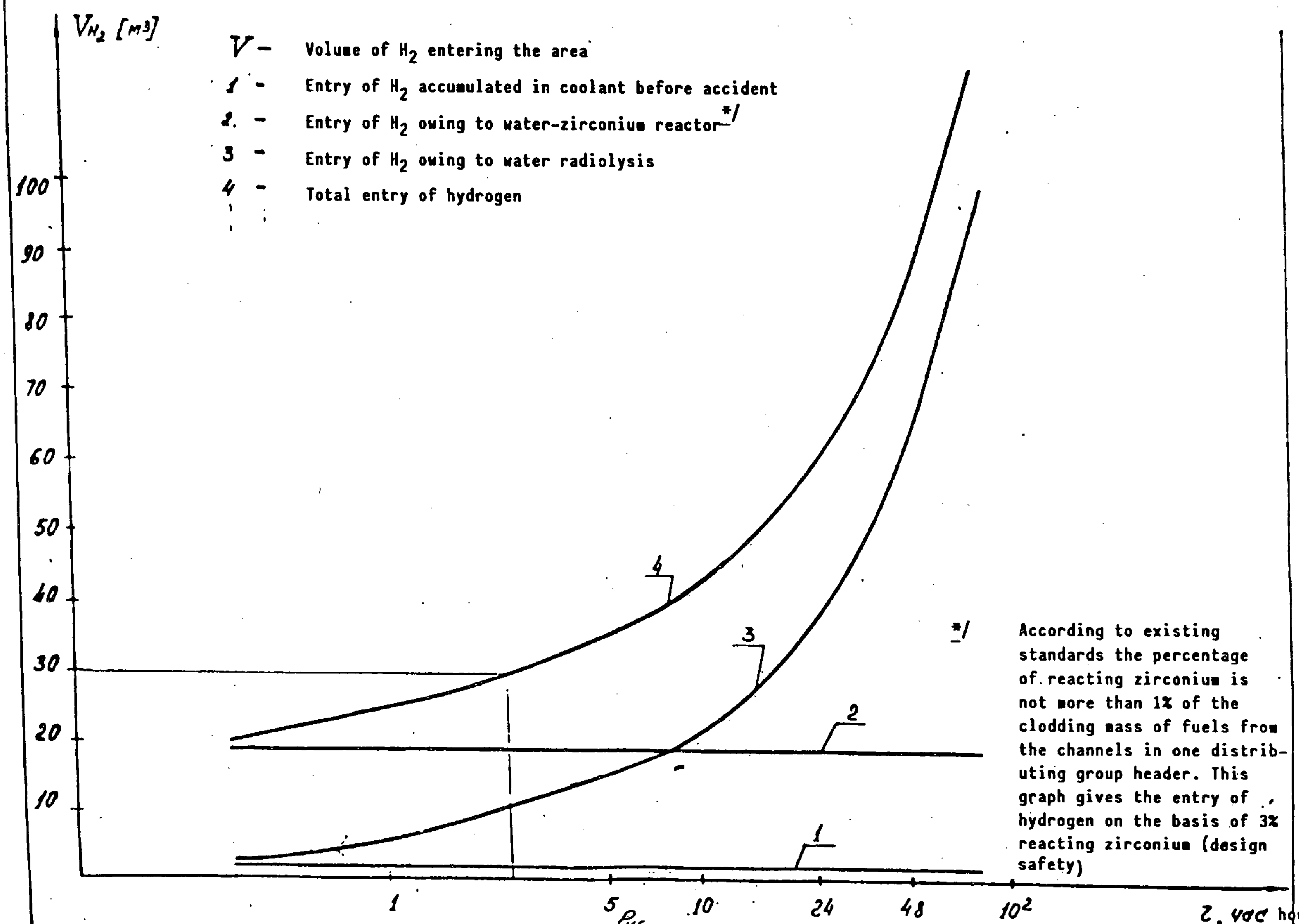


Fig 2.50

LEGEND TO FIG 2.51

HYDROGEN REMOVAL SYSTEM

1. Nitrogen supply
2. Discharge to vent flue via V-1, A
3. Discharge into GARS or to vent flue without GARS
4. To trap
5. Water outlet - water inlet
6. Air suction
7. Air suction
8. Sub-System No. 1
9. Sub-System No. 2
10. Sub-System No. 3
11. Steam-gas discharge component
12. Downcomer shaft
13. Pressure suppression pool
14. Downcomer shaft
15. Discharge for gas analysers
16. Measurement of hydrogen concentration

Hydrogen removal system

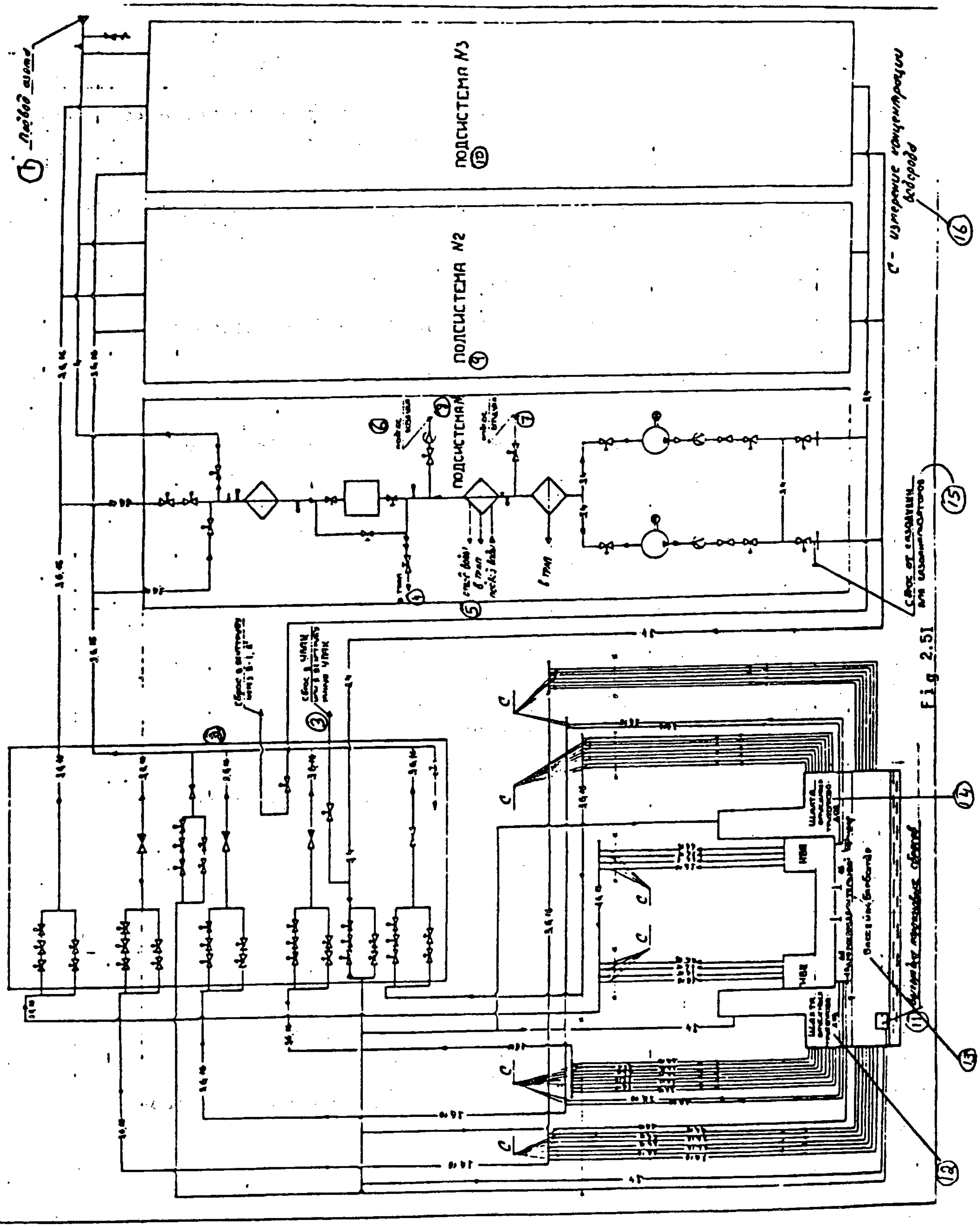


Fig. 2.51

Key to Fig. 2.52 [This figure has been translated to the extent that the quality of the original permits]

Figure caption: Own-requirement circuit diagram for the fourth unit at the Chernobyl' NPP

1. 63 MVA 350/6.3-6.3 kV reserve transformer
2. To 330 kV outside switching station
3. 63 MVA 20/6.3-6.3 kV working own-requirement transformer
4. Unit transformer
4a. To 750 kV outside switching station
5. 20 kV
6. 6 kV reserve busbar
7. 6 kV sections of the third emergency power supply group
8. 7RB
9. 7RA
10. 8RB
11. 8RA
12. Main circulation pumps
13. First-stage condensate pump
14. Second-stage condensate pump
15. Feed pump
16. Other mechanisms
17. Working own-requirement transformers
18. Other own-requirement transformers
19. To 6 kV section of the third unit
20. 6/0.4 kV reserve transformer
20a. 0.4 kV reserve busbar
21. 6 kV sections of the third emergency power supply group
22. Diesel generator 2
22a. Diesel generator 1
23. Diesel generator 3
24. 7RNB
25. 7RNA
26. 8RNA
27. Process water pump
28. Non-accident-half ECCS pump
29. Accident-half ECCS pump
30. Emergency protection system channel cooling circuit pump
31. Sprinkler-cooling system pump
32. Emergency supply transformer 82 TNP
33. 0.4 kV sections of the second emergency supply system group
34. 2NNBC
35. 72NNB
36. 71NNB
37. 92NNB
37a. 91(?)NNB
38. 3NNBC

39. 0.4 kV users of the second safety system
40. 0.4 kV users of the first safety system
41. Electric motors of ventilation systems
42. Whole-unit users
43. Whole-unit emergency supply system facilities:
44. Emergency supply system facility for second safety system (SD II)
45. Emergency supply system for first safety system (SB I)
46. First
47. Second
48. Emergency supply system facility for third safety system (SB III)
49. Monitoring and control instruments, automatic control systems, regulators
50. Emergency protection system and "Skala" central monitoring system users
51. Electric drives of gate valves
52. 21NNA
53. 22NNA
54. NNB
55. 11NNA
56. 12NNA
57. NNB
58. 11NNA
59. 12NNA
60. 13NNA
61. 14NNA
62. 21NNA
63. 22NNA
64. 23NNA
65. 24NNA
66. 0.4 kV sections of first emergency power supply group
67. 2NNAS
68. 1NNAS
69. 3NNAS-1
70. 3NNAS-2
71. Electric motors of isolating mechanism and ECCS
72. Automatic control, protection and regulation devices
73. 2 MPS
    PTS-63
74. 2 VUS
75. 1 MPS
    PTS-63
76. 1 VUS
77. 1 VU
78. 11 MP
79. 12 MP
80. 13 MP
81. 14 MP
82. 1 VUP
83. 91 NNB

84. 21 MP
85. 22 MP
86. 23 MP
87. 24 MP
88. 2 VU
89. 2 VUP
90. 91 NNV
91. 3 MPS-1
    PTS-63
92. 3 MPS-2
    PTS-63
93. 3 VUS
94. Direct current distribution panel 3 S (3ShchPTS)
95. Direct current distribution panels
96. = 232 V
97. Direct current distribution panel 2 S (2 ShchPTS)
98. = 232 V
99. Direct current distribution panel 1 S (1ShchPTS)
100. Direct current distribution panel 1 (1ShchPT) = 253 V
101. = 232 V
102. Direct current distribution panel 2 (2ShchPT) = 253 V
103. = 232 V
104. = 232 V
105. Direct current users
106. 3ABC
     SK-16
     108 elec.
107. 1ABC
     SK-16
     108 elec.
108. 1AB
     SK-52
     118 elec.
109. From 108 elec.
110. Emergency power supply transformer 2 S (2TNPS)
111. Emergency power supply transformer 1 S (1TNPS)
112. Emergency power supply transformer 3 S (3TNPS)
113. Emergency power supply transformer 72
114. Emergency power supply transformer 73
115. Emergency power supply transformer 71
116. Emergency power supply transformer 93
117. Emergency power supply transformer 92
118. Emergency power supply transformer 91
119. Emergency power supply transformer 81 [?]
120. Transformer 225 T

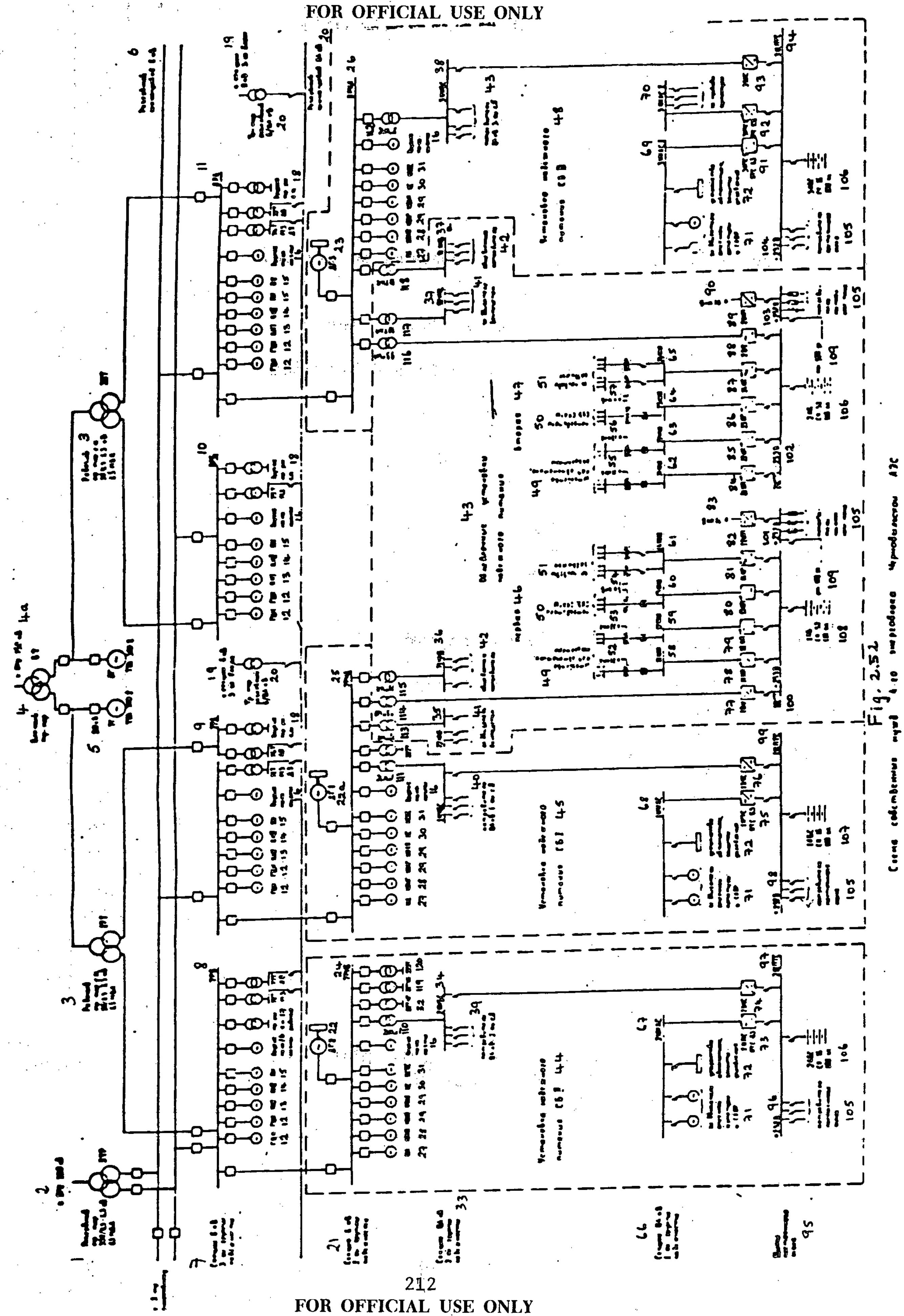


Fig. 2.52

Key to Fig. 2.53

Horizontal captions (see column numbers on figure)

1.	No.
2.	Name of emergency power supply user
3.-13.	Starting data
3.	Number attached to one safety sub-system
5.-6.	Power (kW)
5.	Rated
6.	Consumption
7.	Nominal current, I_n (A)
8.	Startup current, I_{st} (A)
13.	rpm
14.-17.	Theoretical values
14.	Theoretical power (kW)
15.	Startup power (kW)
16.	Slippage (S_{nom})
17.	Cosϕ startup
18.-32.	Design-basis accident regime
19.-29.	Switching stages
18.	Number of motors started up
19.	15 s I
20.	20 s II
21.	25 s III
22.	30 s IV
23.	35 s V
24.	40 s VI
25.	45 s VII
26.	50 s VIII
27.	55 s IX
28.	60 s X
29.	65 s XI
30.-32.	$\frac{P_{rated}\ (kW)}{I_{rated}\ (A)}$
30.	Per stage
31.	Up to 10 min
32.	Up to 30 min and beyond
33.-47.	Loss of current by own-requirement users
33.	Number of motors started up
34.-35.	Switching stages
34.	15 s I
35.	20 s II
36.	25 s III
37.	30 s IV
38.	35 s V

39. 40 s VI
40. 45 s VII
41. 50 s VIII
42. 55 s IX
43. 60 s X
44. 65 s XI
45.-47. $\frac{P_{rated}\ (kW)}{I_{rated}\ (A)}$
45. Per stage
46. Up to 10 min
47. Up to 30 min
48. Notes: [opposite vertical columns 12, 13:] Not started up in stages

Caption at bottom right of figure

This table was compiled for one safety sub-system. The tables for the other two sub-systems are similar.

Vertical captions (see line numbers on figure)

1. Transformers for emergency supply system and plant's own requirements
2. Process water pump
3. Non-accident-half ECCS pump
4. Sprinkler-cooling system pump
5. Accident-half ECCS pump
6. Emergency protection system channel cooling circuit pump
7. Reserve
8. Clean condensate pump
9. Emergency feed pump
10. Fire pump
11. Emergency supply system transformer
12. Circuit cooling pump
13. Tank pump

Исходные данные / Расчетные величины

№ 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 $\varphi_н$	$К_п = \frac{I_п}{I_н}$	$К_м = \frac{М_п}{М_н}$	Скорость вращения (1/мин)	Расчетная мощность (кВт)	Пусковая мощность (кВт)	Скольжение $S_н$ (о.е.)	cos $\varphi_{пуск}$
1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17
1	Трансформаторы надежного питания и с.н. АЭС	4				100	200						800	1600		
2	Насос технической воды [illegible] - НА	1	АВ-13-39-16	1250	1100	160	810	[illegible]	0,8	5,2	0,8	370	1170	1900	0,015	0,22
3	Насос [illegible] - НСБП	1	2АЗМ 800/6000	800	720	88	460	[illegible]	0,91	5,2	1,1	2970	755	1540	0,01	0,3
4	Насос спринклерной системы - НСОС	1	А-113-4М	250	241	28	144	[illegible]	0,89	5,0	1,3	1480	241	405	0,013	0,34
5	Насос аварийной подпитки - НОАП	2	2АЗМ 800/6000	800	720	88	459	[illegible]	0,91	5,2	1,1	2970	755	1540	0,01	0,3
6	Насос расхолаживания [illegible] СУЗ - НС	1	А-113-4М	250	216	28	144	92,0	0,89	5,0	1,3	1480	227	405	0,013	0,34
7	Резерв															
8	Насос чистого конденсата - НЧК	1	А-113-4М	250	200	28	144	92	0,89	5,0	1,3	1480	210	405	0,013	0,34
9	Аварийный питательный насос - АПН	1	2АЗМ 800/6000	800	720	88	460	[illegible]	0,91	5,2	1,1	2970	755	1540	0,01	0,3
10	Пожарный насос - ПЖН	1	А-113-4М	250	204	28	144	92	0,89	5,0	1,3	1480	214	405	0,013	0,34
11	Трансформатор надежного питания - ТНП	1	ТСЗ-1000	800		68	136						560	1120		
12	Насос [illegible]	1	2АЗМ 500/6000	500	204	54	330	[illegible]	[illegible]	6,0	1,2	[illegible]	223	1050	0,01	0,31
13	Насос [illegible] - НППГ	1	А-113-4М	250	200	28	144	95	0,89	5,0	1,1	1480	210	405	0,013	0,34

Режим максимальной проектной аварии

№ п/п	Количество пускаемых двигателей (шт)	Включение по ступеням $\frac{P_п (кВт)}{I_п (А)}$: 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	$\frac{P_{уст} (кВт)}{I_{уст} (А)}$ по каждой ступени	до 10 минут	до 30 минут и более
	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32
1		1600/200											800/100		
2	1		2100/910										1910/260		
3	1			3510/270									2725/348		
4	1				3130/492								2966/373		
5	2					6096/1288							4476/554		
6	1						4851/638						4703/583		
7								—					—		
8	—								—				—		
9	—									—			—		
10	1										5108/2221	-	4911/611,9		
11	1											6032/747	5437/689		
12	—														
13	—														

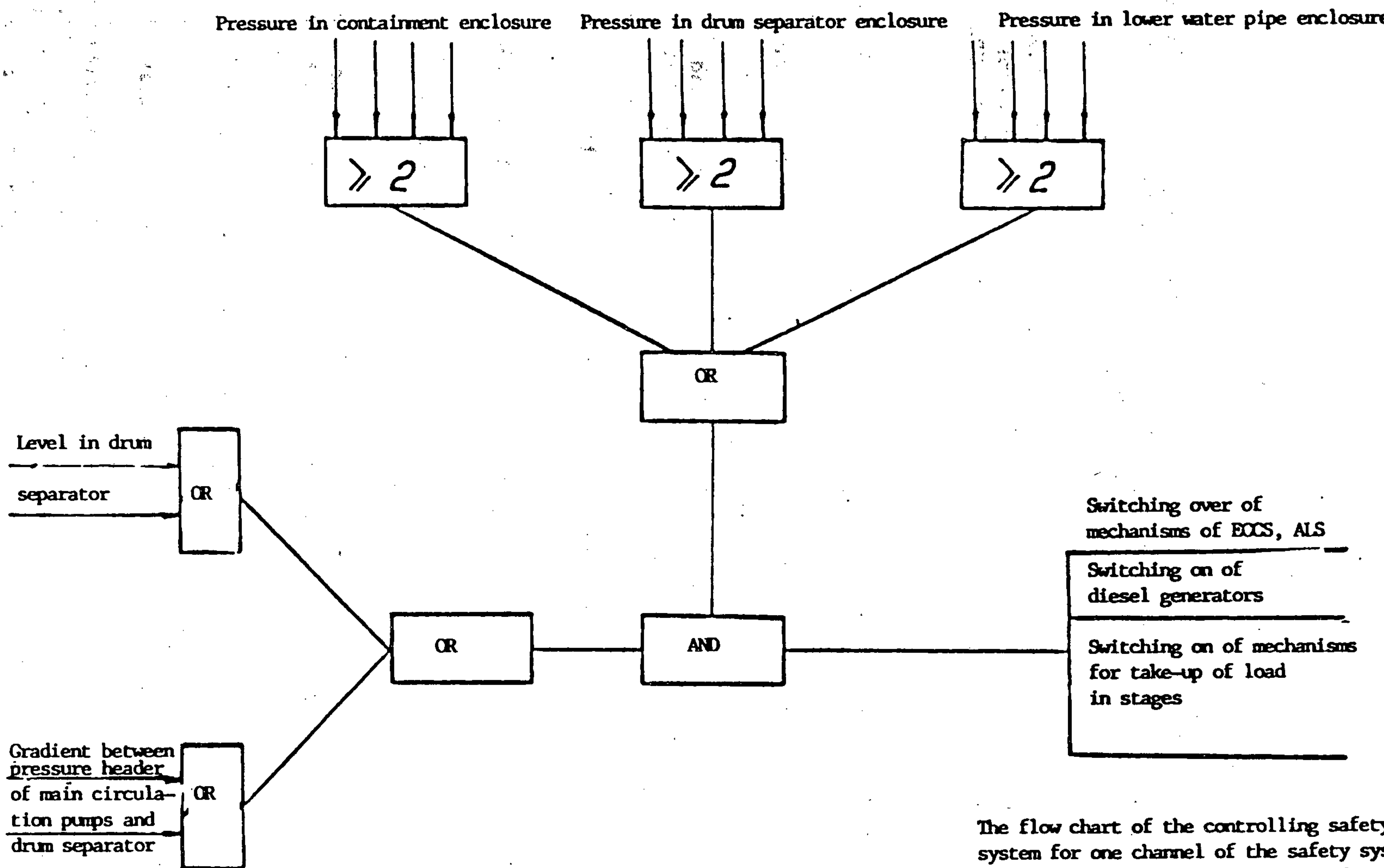
Режим обесточивания собственных нужд

№ п/п	Количество пускаемых двигателей	Включение по ступеням $\frac{P_п (кВт)}{I_п (А)}$: 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	$\frac{P_{уст} (кВт)}{I_{уст} (А)}$ по каждой ступени	до 10 минут	до 30 минут	Примечания
	33	34	35	36	37	38	39	40	41	42	43	44	45	46	47	48
1		1600/200											800/100			
2	1		2100/530										1910/260			
3	—			—									—			
4	1				2375/404								2211/289			
5	—					—							—			
6	1						2616/433						2438/318			
7	—							—					—			
8	1								2843/461				2648/346			
9	1									4188/807			3403/435			
10	1										3808/579		3617/464			
11	1											4037/599	4137/532			
12	—															[illegible] не участвует
13	—															В ступенчатом пуске не участвует

Данная таблица составлена для одной подсистемы безопасности. Для двух других подсистем таблица — аналогична.

Рис. 2. 51

Fig. 2.53



The flow chart of the controlling safety system for one channel of the safety system is shown. The charts for the second and third channels are similar.

2.54

Flow chart of controlling safety system

2.11. Other safety-related systems

2.11.1. Multiple forced circulation circuit (MFCC).

A description of the multiple forced circulation circuit (primary coolant circuit) and its main components is given in sections 2.6 and 2.7.

2.11.2. CPS channel cooling system

The system for cooling the CPS (control and protection system) channels is designed to ensure the requisite temperature conditions for these channels together with the control elements and servodrives of the CPS.

The system performs the following functions:

- It maintains a temperature of 40° at the coolant water inlet to the control channels;
- It removes a thermal output of 28.1 MW from the channels for the CPS control elements and servodrives;
- It ensures cooling of the CPS control element and servodrive channels at a nominal flow rate for ~6 minutes when the pumps are not functioning;
- It maintains a nonexplosive concentration of hydrogen under all working conditions;
- It maintains the requisite amount of water cooling the channels and CPS servodrives;
- It ensures that there is emergency protection of the reactor if the cooling system is disrupted.

These functions are performed with allowance for single failure in the system of an active element or a passive element with moving mechanical parts.

A schematic diagram of the CPS channel cooling system is shown in Fig. 2.55.

The system constitutes a circulation loop operating by gravity.

Water from the top emergency supply tank flows by gravity into the pressure (distributing) header and is distributed through the channels. The channels contain the elements of the control and protection system and tubes containing the fission chambers and power density monitors. Some of the channels are used to control the flow of water cooling the graphite of the lateral reflector.

After flowing through the CPS channels the cooling loop water transfers its heat to the service water in the heat exchangers of the system. Depending on the temperature of the service water and degree of contamination of the heat exchange surface, the required heat removal is ensured by two heat-exchangers, two others being redundant.

After the heat-exchangers the water flows into the lower tanks of the system, in which there is automatically maintained a level ensuring stable functioning by the pumps under steady-state and transient conditions. The total volume of the lower tanks is such that they can hold all the water in the system to be received if the pumps stop.

There are four pumps for feeding the water from the lower tanks to the emergency supply tank. The delivery rate for each pumps is ~700 t/h at a pressure head of 0.9 MPa. Two of these are in operation, while two are redundant.

Provision is made for reducing the probability of all pumps failing for the same reason (the pumps are located in different rooms, have independent power supplies and so forth).

The output of the working pumps exceeds the throughput of the cooling system, hence some of the water is always being discharged from the emergency supply tank into the lower tanks of the system (the level of the water in the emergency tank is kept at the overflow mark).

Radiolysis in the reactor core causes the generation of hydrogen from the water in the CPS cooling system.

To prevent the formation of an explosive concentration of hydrogen there is continuous ventilation of the space above the water in the top and bottom tanks, together with monitoring of the hydrogen content in the CPS cooling water as well as in the space above the water in its tanks.

The emergency water supply tank is connected to the atmosphere by four breather pipes which lead off from the top of it. In addition to this the space above the water in the tank is constantly blown through with compressed air. If there is a failure in the compressed air supply system, the space above the water is ventilated by air ejection, using the excess water continuously drained from the emergency tank into the lower tanks through the overflow. If the system stops, all the water from the emergency tank is drained into the lower tanks.

The lower tanks of the system are continuously blowndown with compressed air and they also receive the air ejected from the emergency tank; furthermore, since the tanks are at reduced pressure, they receive air from the building through a special line with a valve. Reduced pressure in the tanks and removal of air blown through them is ensured by a special tank ventilation system.

If any of the blowdown (ventilation) systems should fail, those still functioning keep the hydrogen concentration at a safe level.

To maintain the requisite quality of the water in the cooling system, it has to be constantly purified. Water is fed for clean-up from the pump pressure header and returned to the lower tanks.

If there are disruptions in the cooling system (reduced level in the emergency supply tank or a fall in the water flow rate), signals are sent for emergency shutdown.

Parameter monitoring and control of the CPS channel cooling system is carried out by the operating staff from the unit control panel. The system was thoroughly checked during the startup and adjustment operations and during operation of the unit.

2.11.3. Blowdown and cooling system

The blowdown and cooling system shown in Fig. 2.56 is intended to cool the blowdown water of the MFCC bled off for clean-up, followed by reheating before it is returned to the MFCC under nominal conditions, and to reduce the temperature of the circulation water to the required level under cooling conditions.

Under nominal conditions the MFCC coolant flowing at 200 t/h (100 t/h from each loop) is pumped by the main circulation pumps to a regenerator where it cools from 285°C to 68°C through heat removal to a cold counterflow, and is then further cooled down to 50°C by the water of the intermediate circuit in the blowdown afterheater, from where it enters the circuit water clean-up system. As it passes through the regenerator in the opposite direction, the cleaned water heats up from 50°C to 269°C and is recycled to the steam separators through mixers in the feed water piping. It should be pointed out that either of the two after-heaters in the blowdown and cooling system may operate in this mode.

When cooling the unit the blowdown and cooling system reduces the temperature of the water in the MFCC, starting at 180°C, down to the temperature required for repairs to the unit. Circulation takes place along the line: steam separators - cooling pumps - larger after-heater - steam separators.

The blowdown and cooling system may also be used to remove residual heat from the reactor when there loss of current for the power unit's own needs. In this mode the operational system is the same as for the cooling mode.

2.11.4. Gas circuit system

The main layout of the system is shown in Fig. 2.57.

Under nominal operating conditions the gas circuit system works in the following way: the nitrogen-helium mixture, when it emerges from the reactor, passes through the fuel channel integrity monitoring system, where a channel-by-channel temperature check is made and the moisture content of the nitrogen-helium mixture is monitored for groups of channels.

Having cleared the fuel channel integrity monitoring system, the mixture passes through a series of condensers, air heaters and filters in which iodine vapours are deposited, and reaches the compressor intake of the helium scrubbing unit, in which apparatus hydrogen, oxygen, methane, carbon dioxide, carbon monoxide and ammonia impurities are removed from the mixture, down to a concentration permitting normal reactor use.

Removal of radioactive argon-41 takes place in cooling tanks [at $-195^{o}C$].

After passing through the scrubbing system, the mixture is returned to the reactor pile. A hydraulic seal is fitted to the pipe which introduces the mixture into the pile to prevent the pressure from rising above permited levels, i.e. higher than 1-3 kPa.

In order to reduce leakage of helium from the reactor pile, nitrogen (99.9999% pure) is introduced into the metal structures of the reactor at a pressure of 2-5 kPa. A hydraulic seal is fitted to the feed pipe.

In the gas circuit system there exists the possibility of flushing the reactor pile with nitrogen. In this event the nitrogen is dumped via the activity reduction system.

In the gas circuit system measurements are made of flow rate, impurity concentration, moisture content, temperature and pressure of the nitrogen-helium mixture, and the circuit is monitored for radiation. All results are displayed on the gas circuit control panel.

The system is controlled from the gas circuit control panel.

2.11.5. Cooling of spent fuel storage ponds

The pond cooling system is designed to stabilize the temperature of the water ponds, which is heated by the decay heat from the spent fuel, in all operating modes including a total power failure affecting in-house requirements. The system maintains the temperature of the water in the cooling ponds:

- Under normal operating conditions - at no more than 50°C, where the maximum decay heat is 1800 kW;

- During simultaneous unloading into the pond of 5% of the fuel assemblies from damaged channels - at no more than 70°C, with maximum decay heat in the pond of no more than 3000 kW;

- When heat removal ceases as a result of any departures from normal operating conditions or loss of electric power to the system (- the water temperature should rise to no more than 80°C in the 20 hours after heat removal has ceased.)

During this time measures must be taken to restore the functioning capacity of the system.

Water quality in the cooling ponds is maintained by a bypass purification system. Arrangements exist to exclude the possibility of accidental drainage of the ponds. The space above the water is ventilated.

The main layout of the cooling pond system is shown in Figure 2.58.

The cooling of the ponds is achieved by means of a closed loop. Water, heated in the ponds by decay heat production, passes from the upper regions of the ponds to heat exchangers, where the heat is transferred to service water. The required heat reduction is achieved with one heat exchanger, while a second stands in reserve. Having passed through the heat exchangers the water is returned to the pond by one of two pumps at a rate of ~ 160 m^3/hr and at a pressure of ~ 20 m water column (the second pump is a reserve pump).

The cooling water outflow and return pipes are so arranged that if they were to rupture the level in the ponds would not fall below the permissible minimum.

To prevent overfilling of the ponds each has an overflow.

To prevent the formation of an explosive concentration of hydrogen in the space above the water in the ponds, constant ventilation is provided by air taken from the central hall. Should the pond ventilation system suffer a malfunction, the flow cross-section of the ducts which connect the ponds with the central hall is such that one may view them as a single compartment with a volume of 40 000 m^3, ventilated by an independent ventilation system.

The water in the ponds is purified by means of a loop which is independent of the cooling system.

The following parameters are monitored:

- Water levels in the ponds;
- Temperature of water in the ponds;
- Flow rate of cooling water etc.

Operating personnel control the system and monitor its process parameters from the unit control room.

The system underwent comprehensive and direct testing against design specifications during commissioning operations and while the power unit was in use.

2.11.6. Ejection cooling system

The ejection cooling system (Fig. 2.59) is designed to remove heat from the leaktight compartments. For each of the two leaktight compartments there are four groups of coolers set at level 5.0 in the housing of the main circulation pump tanks. Each group consists of four coolers, and each cooler has a capacity of 2500 m^3/hr. Each group has an independent air supply. As regards their water supply, the coolers are divided into two independent sub-systems of eight coolers each and connected to different feedback Sylphon pumping systems.

Air at maximum temperature is drawn off from the upper regions of the downcomer shafts through four pipes, fed to each group of coolers where it is cooled by jets of water down to 35°C in summer, and to 18°C in winter, and then passes into the compartment containing the main circulation pump tanks. The cooled air removes heat from the mechanical equipment and unplanned coolant leakages. To prevent escape of dispersed moisture with the air, separators are installed at the outlet from the coolers. Apart from cooling the air and eliminating excess moisture, the ejection coolers also remove aerosols, including radioactive iodine.

The ejection cooling system is compact and contains no active elements which require maintenance or control while it is operating.

2.11.7. Radiation monitoring system

The nuclear power station radiation control system is a component part (sub-system) of the automated station control system and is designed to gather, process and present information on radiation conditions in the station compartments and outside, on conditions in the process media and circuits and on irradiation doses to personnel and individuals from the population in accordance with the norms and statutes in force.

The radaition monitoring system as a whole can be divided into two: the process radiation monitoring and the radiation dosimetry systems. The purpose of the process radiation monitoring system is process optimization, and also to monitor the condition of the protective barriers against the spread of radionuclides. The purpose of the radiation dosimetry system is to monitor the radioecological factors arising from the operation of the facility and, in the final account, to determine the internal and external irradiation doses received by staff and individuals in the population.

The off-site dosimetric monitoring system is distinct from the radiation dosimetry system, and:

- Determines the activities and nuclide compositions of radioactive substances in the atmosphere;
- Monitors gamma radiation dose exposures in the area;
- Monitors radioactive fallout;
- Monitors ground water activity in test bore-holes;
- Determines the content of radioactive substances in soil, vegetation, locally-produced feedstuffs, food products and so on.

A structural diagram of the radiation dosimetry system is shown in Fig. 2.60.

The following are used for radiation monitoring:

(1) The combined AKRB-06 unit, which includes detection units and equipment, informtion processing equipment, surface contamination monitoring units and units and dosemeters for monitoring the station personnel;

(2) Individual, portable and wearable devices;

(3) Laboratory equipment and instruments.

The radiation monitoring structure takes the form of a data and measurment system with a large number of dispersed information sources and capture devices, arranged in a radial-annular manner; under this system, the detection units, the UNO-06r information storage and processing unit and the local units for signalling when established values are exceeded are linked radially. The UNO-06r system's internal links and links with the UNO-01r monitoring and data exchange device are arranged in a ring.

The AKRB-06 thus monitors continuously the readings from the detecting units and devices, transmits information on all channels to the computer, signals failures of its components and controls the shut-off equipment on the sampling lines. The devices for displaying information (display, console, signalling units) are located on the radiation monitoring board.

The detection units feeding into the AKRB-06 measure:

- The gamma exposure rate within the range 10^{-5} to 10^{3} R/h (BDMG-41, BDMG-41-01, UDMG-42, UDMG-41-02);

- The activity concentration of gamma emitters in liquid process media and circuits within the range 5×10^{-11} to 10^{-3} Ci/L (UDZhG-04r, UDZhG-05r, UDZhG-14r1);

- The activity concentration of iodine vapours in air within the range 10^{-11} to 10^{-6} Ci/L (BDAD-06);

- The activity concentrations of aerosols with dispersion phases containing beta emitters within the range 10^{-13} to 10^{-9} Ci/L (BDAB-05);

- The beta activity concentration of inert gases in the air and process media within the range 10^{-9} to 1.4×10^{-4} Ci/L (UGDB-08) and 10^{-5} to 0.3 Ci/L (UDGB-05-01);

- The activity of long-lived beta-emitting aerosols in the gas-aerosol releases to the vent stack within the range 3×10^{-14} to 3×10^{-10} Ci/L;

- The activity of short-lived beta-emitting aerosols in the gas-aerosol releases to the vent stack within the range 1.5×10^{-12} to 1.5×10^{-8} Ci/L;

- The activity of beta-emitting inert gases in gas-aerosol releases to the vent stack within the range 8×10^{-9} to 8×10^{-5} Ci/L;

- The activity of the gamma-emitting vapours in the gaseous phase of the gas-aerosol releases to the vent stack within the range 3×10^{-13} to 3×10^{-10} Ci/L.

Gas-aerosol releases to the vent stack are measured using RKS-03-01 and RKS-2-02 radiometers. The airflow through the vent stack is measured using a partial flowmeter with a metal-polymer sensing element. The measurement data are processed by a computer.

Each power station unit has a total detector complement of 490 units, of which approximately 400 are in production areas with a continuous or restricted staff presence.

The surface contamination monitoring unit notifies staff when contamination exceeds the following established threshold levels:

- For the skin of the hands: beta emitters within the range 10 to 2000 counts/min./cm^2 (RZG-05-01, SZB-03, SZB-04);

- For the skin of the body or basic protective clothing, beta emitters within the range 5 to 2000 counts/min./cm^2 (RZB-04-04);

- For means of transport, on leaving the station in order to detect objects to be investigated in detail using other means, gamma radiation within the range 2.78×10^{-2} to 0.278 μR/s (RZG-05);

- For personnel, on leaving the station, for detection and subsequent detailed investigation, gamma radiation within the range 1.4×10^{-2} to 0.14 μR/s (RZG-04-01).

The personnel irradiation monitoring unit continuously monitors external irradiation. To this end, the basic items of equipment used are:

- Sets of individual dosimetric photomonitors to measure the total exposure to gamma radiation within the range 0.05 to 2 R at energies of 0.1 to 1.25 MeV (IFKU-1);

- Sets of thermoluminescence dosemeters to measure exposures to X-ray and gamma radiation in the energy range 0.06 to 1.25 MeV within limits of 1.0 to 1000 R and 0.1 to 1000 R (KDT-02);

- Gamma exposure rate dosemeter-indicators in the range 0.1 to 9.9 R/h; there is also a range of other dosemeter variants with similar characteristics.

The internal irradiation recording equipment measures the whole-body burden of ^{137}Ce and ^{60}Co nuclides and the thyroid ^{131}I burden (MSG-01). In addition, DGDK-type semiconductor detectors and their analysis and processing equipment are used to identify a range of radionuclides in the human body. There is a wide range of portable and wearable dosemeters and radiometers in the total stock. For example, the following dosemeters are used:

- For measuring exposures to gamma and X-ray radiation in the energy range between 15 keV and 25 MeV with a measurement range between 0.1 μR/s and 11 R/s (DRG, DKS);

- For measuring neutron dose equivalent rates between 0.05 and 5000 μrem/s (KDK-2);

- For express measurements of the specific activities of samples (RKB4-1 beta radiometer) within the range 2×10^{-12} to 10^{-7} Ci/L;

- For measuring nuclide activity concentration in liquids and air for alpha, beta and gamma radiation over various energy ranges (RZhS-05, RGA-01, MKS-01 and others).

Off-site dosimetric monitoring is carried out in the area of the station within a radius of approximately 35 km. It is carried out by the off-site dosimetry service of the plant and is designed to obtain the information required to evaluate the external and internal doses to individuals in the population. The monitoring equipment is located at 38 posts, and includes total gamma dosemeters, vials for collecting atmospheric fallout and seven aspiration sets.

Samples are analysed using semiconductor detectors, spectrometers and analysers with microcomputers. On the basis of the data on releases into the atmosphere through the power station vent stack and by means of automatic measurement of the meterological parameters, a forecast is made using a microcomputer of the radiation situtation in the power station area.

2.11.8. NPP control

The NPP is controlled on two levels:

- As a station;

- By unit. (See Fig. 2.6.1 "Basic structural diagram of station control sytem"). Control over all plant safety systems is carried out at unit level.

Station-level control

At the station level, operational control is effected from the central control board. At the station level, the operating staff are responsible for:

- Control over the electrical equipment in the main electrical connection circuit (750-kV line interrupters, unit transformers, autotransformer and so on, 20 kV generator interrupters, 330 kV autotransformer interrupters and interrupters for the 6 and 330 kV back-up medium voltage transformers);

- Distribution of active and reactive power;

- Co-ordinating the work of the operating staff at the unit control panels and in separate installations on the site.

On the central control board there are remote control keys for the aforementioned interrupters, and also audible and visual signalling of accident and fault conditions, and visual signalling of the condition of the switching apparatus (interrupter in or out) on the mimic diagram.

The protection relay equipment, anti-accident automatics and telemechanics are housed in the protection relay buildings for the corresponding 750 kV and 330 kV distribution equipment. Microchip integrated circuits form the basis of the 750 kV line protection relay equipment; they monitor the function of each separate channel and make it possible to test them. Mass-produced electromechanical relays are also used in the protection relay control devices and anti-accident automatics.

Unit-level control

The process installations and structures of a given unit are controlled at unit level:

- the reactor and its supply facilities (main circulating pumps, feed pumps, emergency feed pumps and so on);

- turbogenerators and auxiliary equipment;

- normal and back-up medium voltage supplies and so on;

- separate installations on the site: diesel power station, process water supply pumps and so on.

The above are controlled from the unit control board, which includes the control board and display. The operator control circuit of the unit control panel is split into control areas:

- reactor control;
- steam generator set control;
- turbine, generator and medium voltage supply control.

On the operator circuit of the unit control board are located the operators' work stations and the control panels for the:

- senior reactor control engineer;
- senior unit control engineer;
- senior turbogenerator control engineer.

These control panels contain the following:

- control apparatus;
- monitoring system instruments;
- "Skala" central monitoring system, call-up devices and display units;
- communications apparatus.

On the unit control panel operator circuit board are the following:

- reactor channel mimic board;
- CPS mimic board;
- mimic diagram of the thermal and electrical parts of the unit;
- individual instruments for the monitoring system signalling equipment.

The "Skala" central monitoring system monitors the main bulk of the parameters. The most important parameters required for correct process operation also have their own individual monitoring instruments. These include instruments showing reactor output, drum separator level and pressure, steam flow rate ex drum separator, feedwater flow rate into drum separator, measurements from the physical power distribution monitoring system and the CPS and so on.

Electricity is supplied to the unit control panel and the "Skala" central monitoring system from the secure power supply so that even when there is a power loss at the medium voltage buses, the operator does not lose information on the conditions of the process parameters. On the unit control panel operating circuit are located the controls for the process protection and monitoring devices. Overall operative control of the unit is carried out from the controller's board, which has telephone apparatus and loudspeaker links.

Included in the operator circuit of the unit control panel there are in addition special safety panels for each of the three safety sub- systems; on these the back-up medium voltage power supply (diesel generators) and the emergency reactor cooling and accident containment systems are controlled and monitored.

A back-up control board is provided for the eventuality that the reactor cannot be shut down and maintained in sub-critical condition from the unit control board. On the operator circuit of the back-up control board are located control panel, operator circuit panels and safety panels. On the control panel are located the AZ-5 emergency protection system button, CPS coupling disconnection switch, signalling board and so on. On the operator circuit panels are the recorders for neutron power, drum separator pressure and so on. The safety panels of the back-up control board are analogous to those on the unit control board.

Local control panels are provided for the range of systems which operate independently of the main processes; these include the gas circuit, active waste treatment, radiation monitoring system, ejector gas sorption scrubbing unit and the turbines.

Local boards are also provided for a range of units involved in the main process (main circulation pump, electric feed pump, emergency electric feed pump and so on), and these are installed with the equipment itself.

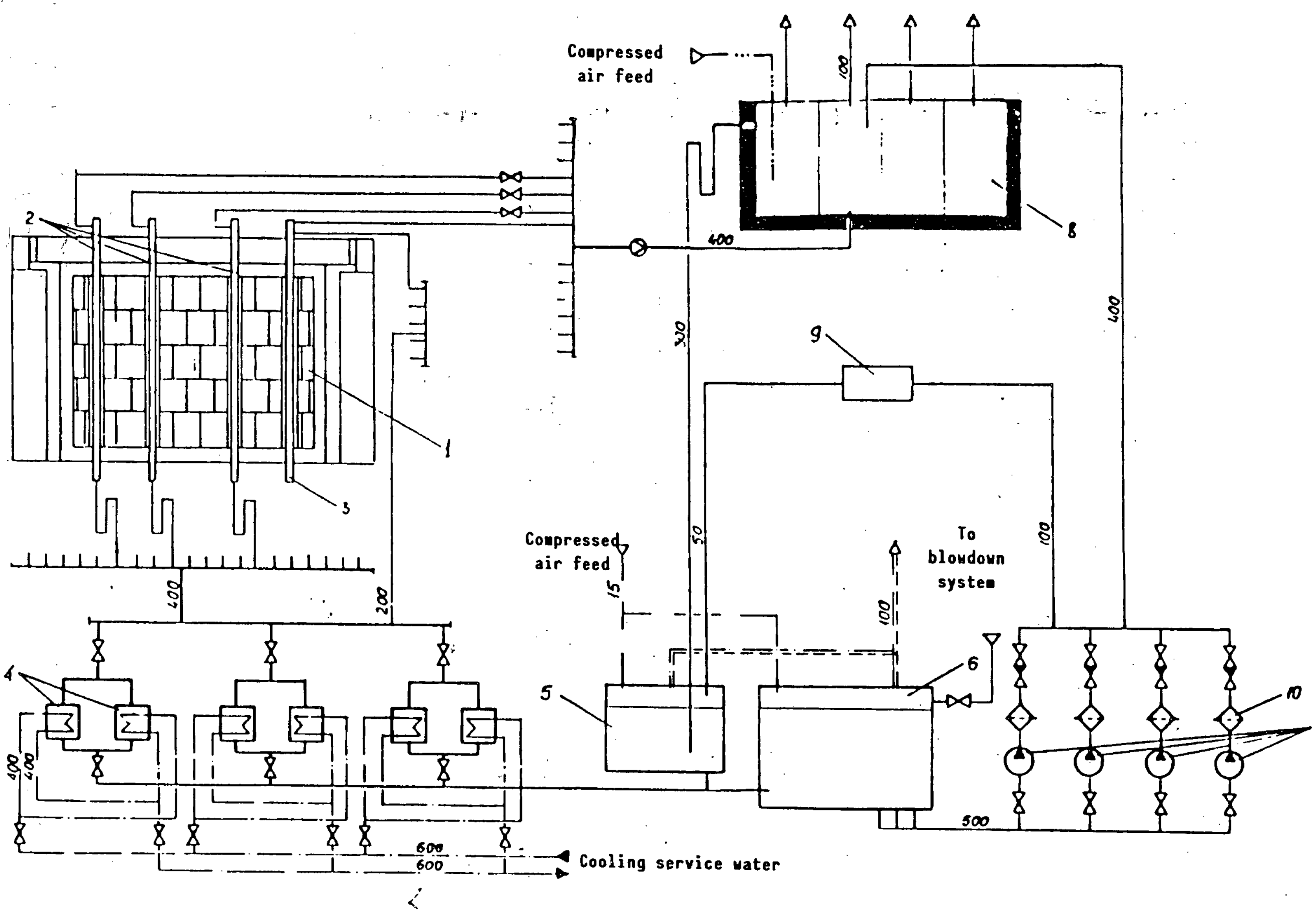
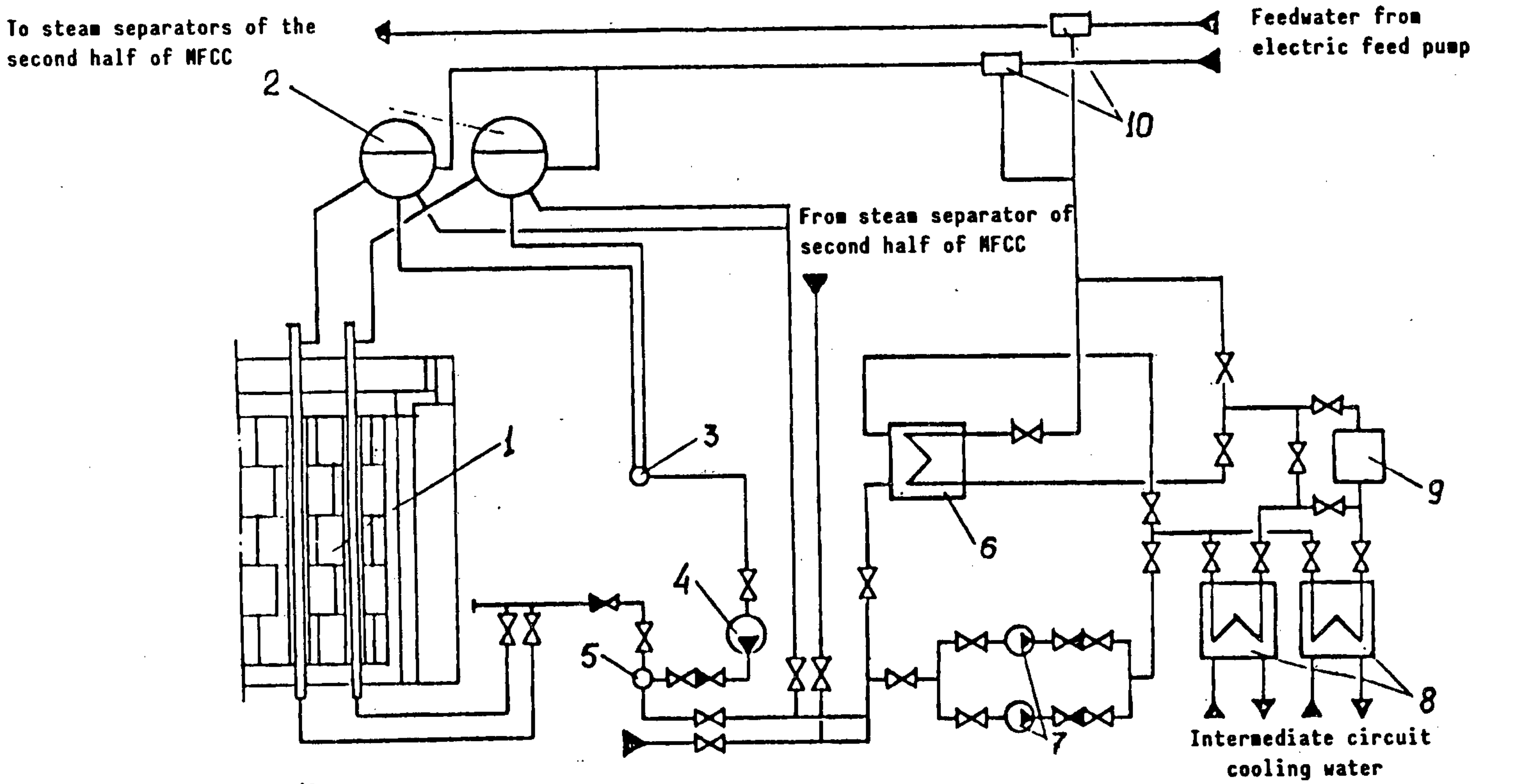


Fig. 2.55 Schematic representation of CPS fission chamber, power density monitor and failure detection channel cooling system: (1) Reactor; (2) CPS, fission chamber and power monitor channels; (3) Failure detection channels; (4) CPS heat-exchangers; (5) Drainage tank; (6) Circulation tank; (7) CPS pumps; (8) CPS emergency tank; (9) Bypass cleaning unit; (10) Filters.



From pressure header of second half of MFCC.

Fig. 2.56. Schematic representation of blowdown and cooling system: (1) Reactor; (2) Steam separators; (3) Intake header; (4) Main circulation pump; (5) Pressure header; (6) Regenerator; (7) Cooling pump; (8) Blowdown afterheater; (9) Plant for cleaning circuit water; (10) Blowdown and feedwater mixers.

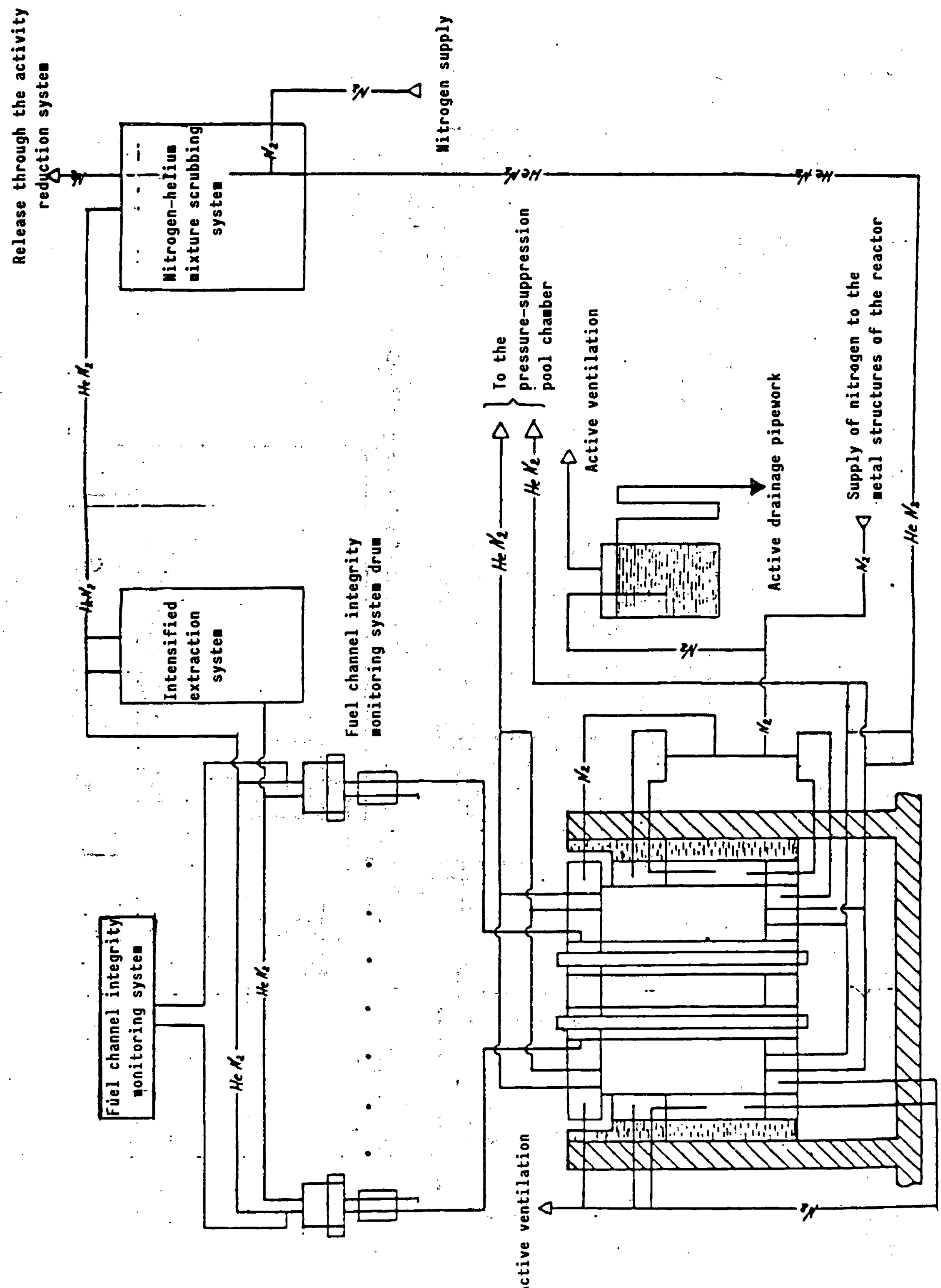


Fig. 2.57 Gas Circuit System.

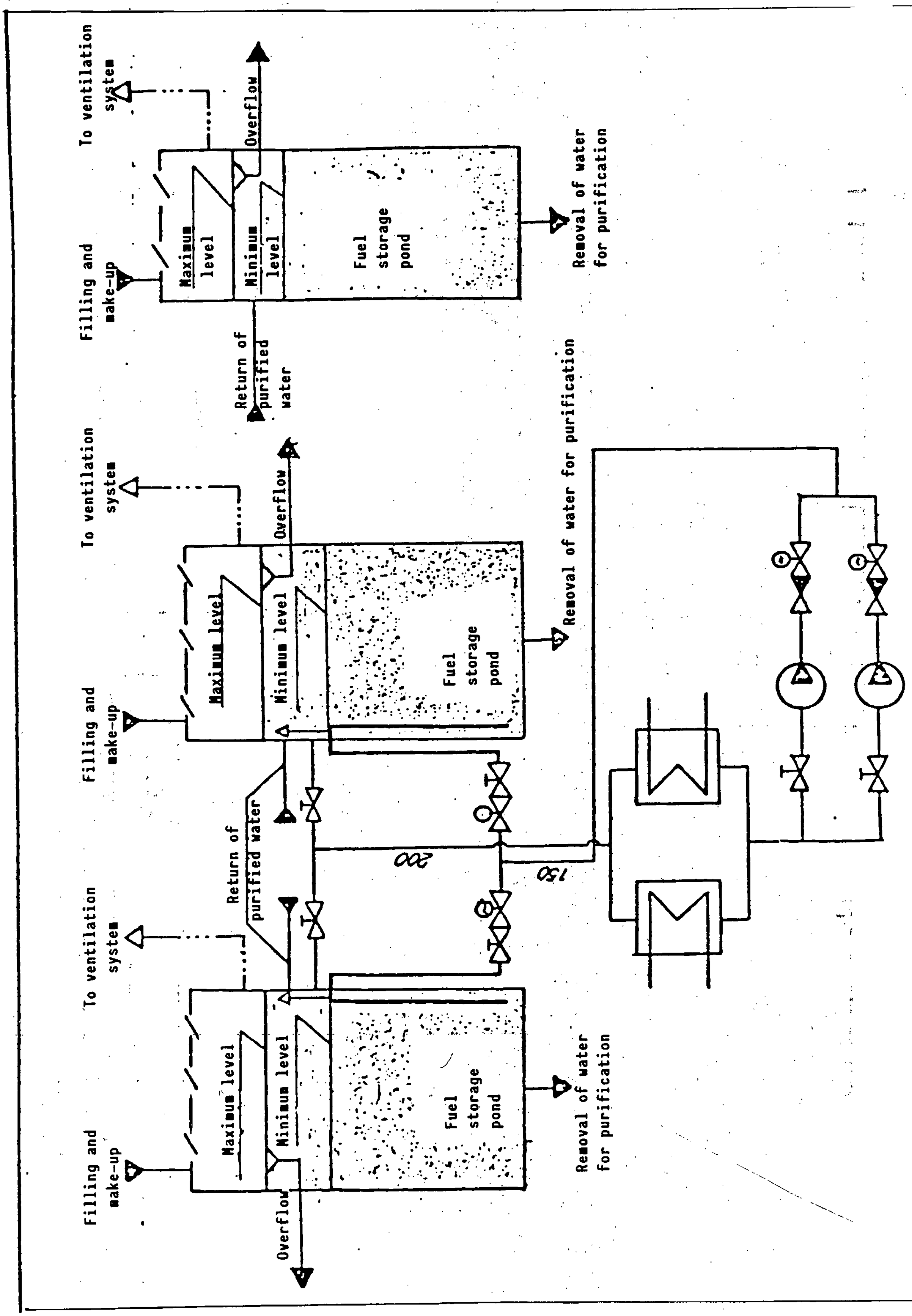


Fig. 2.58 System for Cooling Spent Fuel Storage Ponds

Declassified and Approved For Release 2013/07/15 : CIA-RDP09-00997R000100260001-0

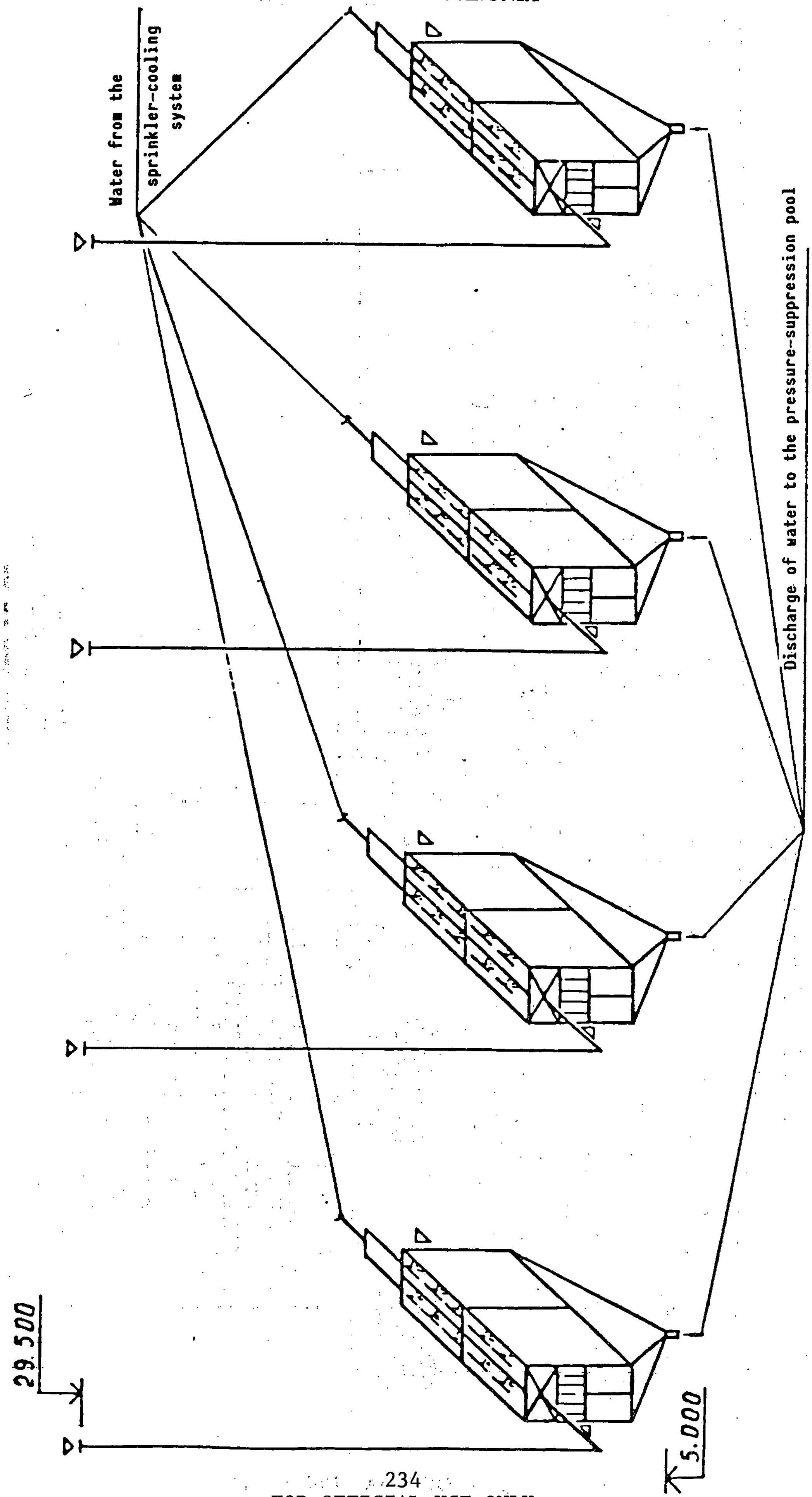
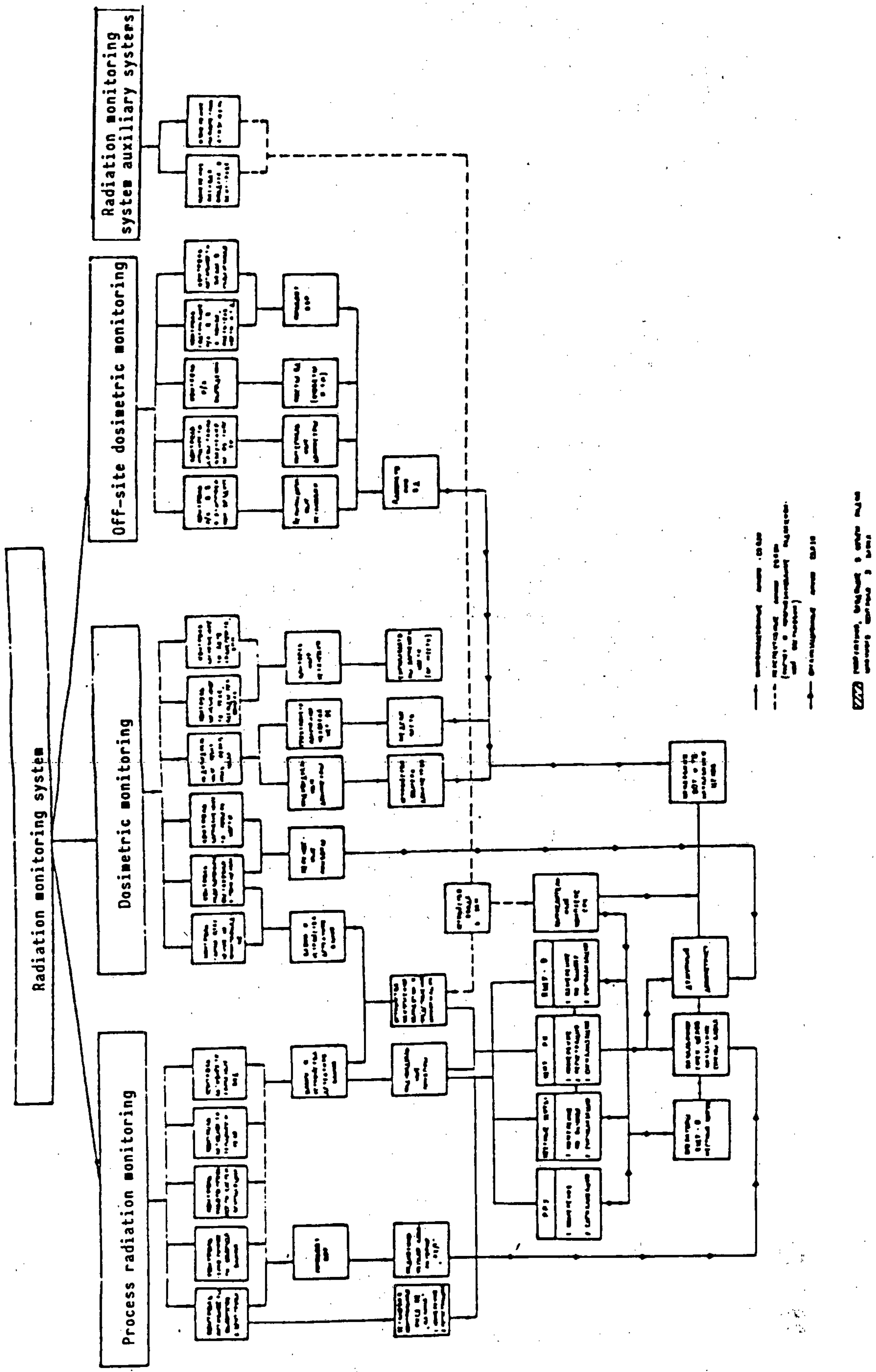


Fig. 2.59 Layout of the Ejection Cooling System.


Radiation monitoring system
Radiation monitoring system auxiliary systems
Off-site dosimetric monitoring
Dosimetric monitoring
Process radiation monitoring

[Other text illegible]

Fig. 2.60

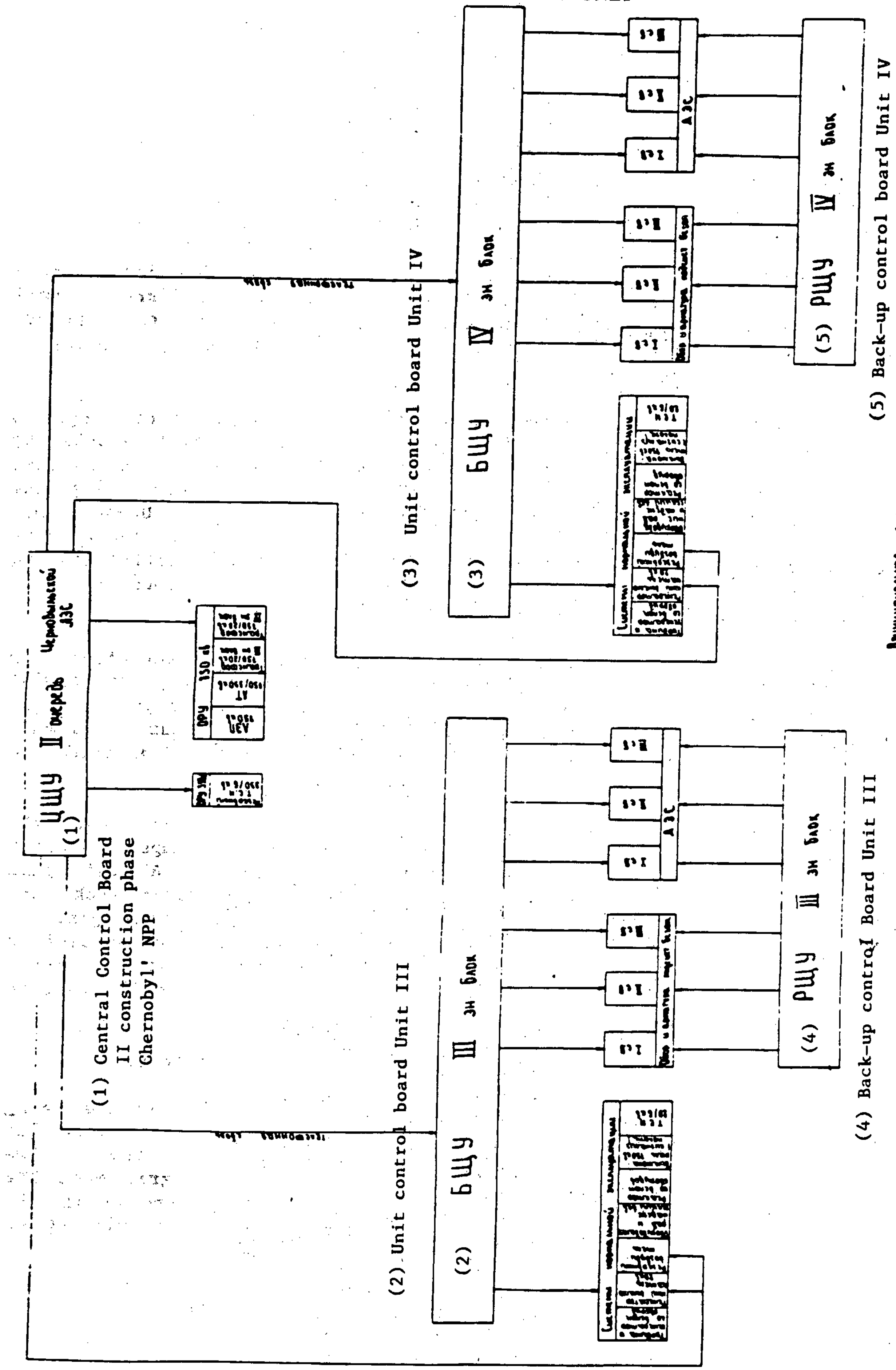


Fig. Рис. 2.61

Принципиальная структурная схема управления станции

Basic structural diagram of station control system

[Other text illegible]

2.12. Reactor and unit operating modes

2.12.1. Normal operating modes

The reactor and unit operating modes can be divided into normal operating modes and transient modes associated with equipment failure. Normal operating modes consist of unit startup and shutdown, unit operation at power, reactor cooling modes during equipment maintenance (maintenance repair modes).
Reactor startup and shutdown

RBMK power units are started up with the main circulation pumps in operation, at a "sliding" pressure and at a separator water level selected by the operator within a given range. The required cavitation margin at the main circulation pump intake is ensured by reducing pump delivery using the throttle-regulating valves installed at the pump discharge. Under these conditions the cooling water flow rate in all the fuel channels of the core is continuously monitored and reactor safety is thereby ensured. Initial heating of the unit is carried out at a "sliding" pressure in the separators, i.e. the pressure is not constant but increases as the temperature rises.

During the startup and initial heating of the unit the circulation loop is fed by the emergency feed pumps. Reactor power during startup and initial heating is maintained at an average level of 2-3% of nominal capacity. The thermal power of individual fuel channels during this process can be as much as 6% of nominal because of the non-uniformity of power density distribution in the core.

Reactor power ascension and initial heating of the circuit can take place with one, two or three of the main circulation pumps (capacity 6000-7000 m^3/h each) operating on each side of the reactor. At this pump capacity it is possible to monitor the water flow rate through each fuel channel and at the same time to ensure an adequate pump cavitation margin. At a reactor power of 2-3% of nominal, the circuit installations are heated to a temperature of about 200^oC. The circuit is heated at a rate of about 10^oC per hour, the limiting factor being the thermal stresses in the reactor metal structures.

At a pressure of 2-4 kgf/cm^2, the de-aerators begin to heat up.

A vacuum begins to build up in the condensers of the turbine being started at a separator pressure of about 15 kgf/cm^2. Once the vacuum has been created, the turbine starts up and begins to build up speed. The turbogenerator is normally synchronized and connected to the grid when the pressure in the separators is about 50 kgf/cm^2. Further increase in the parameters up to rated values takes place in parallel with the build-up of electric load.

Figure 2.6.2 gives an example of the evolution of the main reactor parameters from the time the reactor reaches the minimum power level that can be monitored until the turbogenerator is synchronized and connected to the grid.

The main circulation pumps remain in operation during scheduled shutdown and cooling of RBMK units. Before the onset of shutdown cooling, the reactor power is run down to the after-heat level and the unit turbogenerators are disconnected from the grid and shut off. When reactor power is reduced to the 20% level, the capacity of the main circulation pumps in service should be cut to 6000–7000 m^3/h. The circuit is cooled down to a temperature of 120–130°C by gradually lowering circuit pressure by discharging steam in a controlled manner from the separators to the turbine condensers or to the process condenser. To achieve a greater degree of cooling, a special shutdown cooling system is employed which consists of pumps and heat exchangers.

The factor limiting the cooling rate, and also the heating rate, is the thermal stresses in the reactor metal structures. Since during shutdown cooling the rate of temperature reduction in the circuit is determined principally by the rate of controlled steam discharge from the separators, it is not difficult to keep the cooling rate at the prescribed level under these conditions.

Unit operation at power

During power operation of the unit, reactor safety is ensured by keeping its critical parameters within the permissible range.

Up to the 500 MW(t) power level, the coolant is circulated through the reactor by the main circulation pumps operating at 6000–7000 m^3/h. At a power of 500 MW(t), the throttle-regulating valve is opened and the main circulation pump capacity increases to 8000 m^3/h. At power levels above 500 MW(t) up to the rated level, the unit operates at a constant main circulation pump capacity. When the power level exceeds 60% of rated, no fewer than three main circulation pumps should be operating on each side of the reactor. The hydraulic distribution of an RBMK reactor core is such that, when rated capacity is reached, the throttle-regulating valves are fully open and the total flow through the reactor is 48 000 m^3/h.

Maintenance/repair modes

The main requirement when inspecting or servicing any item of reactor equipment is that the core must be safely cooled throughout this period. Also, the reactor design and the organization of maintenance work should be such as to ensure that all the circuit equipment can be serviced.

From the standpoint of maintenance work, the primary coolant circuit is split into four sections: the discharge (delivery-side) section, which extends from the discharge valves of the main circulation pumps to the channel isolating and regulating valves; the fuel channel ducts from the isolating and regulating valves to the separators; the separators and downcomers to the intake valves of the main circulation pumps; and the section between the intake and discharge valves which includes the circulation pumps and the associated fittings.

The maintenance of equipment and pipes located in the section between the intake and discharge valves of the main circulation pumps does not pose any difficulties, and in theory can be carried out while the reactor is operating.

To do this, it is necessary to close the isolating and intake gate valves on the pipes of the main circulation pump in question; once the coolant has been drained, the pump itself and the discharge and intake pipe sections adjacent to it as far as the gate valves are accessible for servicing. In this instance the coolant is circulated through the reactor by the other main circulation pumps of the relevant loop.

To repair structural elements of fuel channels, the fuel assembly is withdrawn from the channel under repair, the isolating and regulating valve at the channel inlet is closed and the water level in the separators is lowered to below the level at which the steam-water communication pipe of this channel is connected to the separator casing. The remaining channels in the core are cooled either by forced or natural coolant circulation.

To repair the separators, downcomers and intake valves of the main circulation pumps, the discharge valves of these pumps are closed and the level in the fuel channels is lowered. To ensure safe cooling of the core under these conditions, a special maintenance tank is connected to the main circulation pump pressure header; the channels are fed from this tank and the steam which forms in them is evacuated to the separators. To allow inspection and repair of the separators, a system has been installed which draws off steam from the separators to the process condenser.

During repair work on the equipment of the discharge section, this section is cut off from the core by closing the isolating and regulating valves, and the residual heat is removed by water fed into the channels from the separators. This mode of fuel channel cooling (the bubbling mode of cooling) was studied on special test units during the reactor design stage. It was established experimentally that, when the isolating and regulating valves are closed, safe cooling of the fuel channels in the bubbling mode is ensured when the following conditions are met:

- The water level in the primary circuit is higher than the levels at which the steam-water communication pipes connect with the separator;

- The pressure in the separator is atmospheric;
- The after-heat in the fuel assembly is not greater than 25 kW;
- The water temperature in the separator is not more than 80–90°C in order to prevent water hammers in the steam-water communication pipes.

The most complicated repair operation relates to the channel flow meters and the isolating and regulating valves. To do this work, a technique is used whereby the water is frozen in the inlet pipes of the fuel channels. When this technique is employed, the fuel assemblies are cooled in the same manner as when repairs are being conducted with the isolating and regulating valves at the fuel channel inlet closed.

The water is frozen in the vertical sections of the inlet pipes by means of group and single refrigerating chambers attached to these pipes. The refrigerant is air at a temperature of -100°C which is supplied from the nitrogen-oxygen station. While the freezing operation is taking place the isolating and regulating valves remain closed and the fuel assemblies are cooled by the bubbling mode. Once ice plugs about 0.5 m high have formed in the pipes, the isolating and regulating valves and the flow measurement detectors are accessible for repair. This freezing method has repeatedly been used successfully at the Leningrad and Chernobyl' nuclear power plants.

2.12.2. Transient modes resulting from equipment failures

Because of the large unit power of RBMK boiling-water, graphite-moderated reactors and their extreme importance in energy systems, the control and protection system (CPS) of such reactors provides for rapid controlled power reduction at a prescribed rate to safe levels in the event of the failure of certain types of equipment. When a signal is transmitted indicating a fault in the process installations, emergency protection systems of three kinds (AZ1, AZ2, AZ5) are triggered.

The following algorithm for the operation of emergency protection systems has been developed for the CPS of existing RBMK-1000 reactors:

- AZ1 is triggered when one of the six main circulation pumps shuts off, the feedwater flow rate decreases and the level in the separators is reduced. At the AZ1 signal, reactor power is reduced to the 60% level;
- AZ2 is activated in the case of emergency load shedding or the failure of one of the two operating turbogenerators. At this signal, the reactor power drops to the 50% level;

In other accident situations caused by equipment failure the AZ5 emergency protection system is activated and triggers an uncontrolled power reduction to complete shutdown;

In order to study emergency conditions at RBMK units, a mathematical model of the plant was developed at the design stage which contains kinetic, hydrodynamic and heat-exchange equations and a description of the algorithms for the operation of the equipment and systems which automatically regulate NPP parameters. A subsequent comparison of the theoretical results with data on individual dynamic regimes actually experienced at operating nuclear power plants indicated that the mathematical model developed provides a satisfactory description of unit dynamics. Transient regimes mainly associated with transition to natural circulation of the coolant have been studied on special mock-up test stands.

Operating experience from units in service has shown that the measures and systems foreseen guarantee the safety of RBMK reactors in all modes resulting from equipment failures.

A great deal of research has been done to demonstrate the safety of reactor operation in the power reduction mode when the AZ5 emergency protection system is activated since this mode is accompanied by major changes in the process parameters and, in particular, by a reduction in the water level in the separators.

The behaviour of the main reactor parameters under transient conditions due to the activation of the AZ5 protection system is shown in Figure 2.6.3.

A loss of power plant internal load is one of the most severe accident situations that can occur at the unit. When internal load is lost, the coolant is circulated through the core at the start of the accident by the running down main circulation pumps and thereafter by natural circulation. The transient mode resulting from the loss of the internal load of the unit is shown in Figure 2.6.4.

This figure shows that, in the initial phase of the process, the decrease in the water flow rate is somewhat higher than the rate at which the reactor thermal power decreases; this results in a brief increase in steam content and reduction in the departure from nucleate boiling (DNB) ratios. More detailed studies have shown that under such conditions the reduction in DNB ratios - even in those channels which are under greatest thermal stress - is insignificant and poses no danger to the reactor, since in the initial phase of an accident the reactor is safely cooled by the running down main circulation pumps.

The running down pumps have a significant effect on coolant circulation through the reactor only for the first 30-35 seconds of the transient regime. Thereafter the core is cooled by natural circulation. The reliability and degree of natural circulation depends to a large extent on a number of factors such as the primary coolant circuit design, pressure behaviour in the circuit, the change in feedwater flow rate and temperature and so on.

Experimental research on natural circulation regimes has been conducted both on heat engineering mock-up rigs of the reactor primary coolant circuit and directly on operating reactors at the Leningrad and Kursk nuclear power plants. The experiments at the test rigs established, and those at reactors confirmed, the safety of cooling the core by natural circulation both in steady state and in dynamic regimes, given a constant pressure in the circuit. At the operating reactors, the tests under steady state conditions were conducted at a power level of 5 and 10% of nominal, while under dynamic conditions the main circulation pumps were switched off at a power of 25% and 50% of nominal. When the pressure drops as a result, for example, of safety valves opening and then not closing tightly, the coolant boils, the level in the separators increases and, as a result, the steam-water mixture is removed from the circuit. It was established at the test rig that, in the case of pressure reduction to a certain level, partial removal of the steam-water mixture and of the water from the circuit does not reduce the "levelling" head or stop coolant circulation. Overheating of the experimental channel fuel elements was observed only when the pressure in the separators dropped below 35 kgf/cm^2.

To ensure reactor safety follwoing a loss of internal load of the unit and a sharp drop in pressure, the emergency core cooling system is activated and feeds water to the fuel channels.

The safety of natural circulation regimes at RBMK units has been confirmed by accident situations which have occurred under real operating conditions at nuclear power plants. For example, at one unit of the Kursk NPP in January 1980 a total loss of station internal load occurred. During the transient conditions, readings from the thermocouples of the fuel assemblies and from the flowmeter at the inlet to one of the reactor fuel channels were recorded. During the entire transient regimes, no increase in the temperature of the fuel element cans was registered and the flow through the channel recorded under natural circulation conditions was not less than 20% of the flow rate at nominal capacity. The normal system for monitoring fuel can integrity showed no increase in coolant activity when the reactor power was subsequently increased. Experimental data on natural circulation regimes was correlated and compared with the results of calculations from the theoretical programs developed. In view of the good agreement between the results, theoretical predictions were made which showed that reliable and safe operation of RBMK-1000 units under natural circulation conditions is possible at power levels up to 35-40%.

The reactor loss of feedwater and the separator level protection system ensures the safe operation of these units at all power levels. When the AZ-5 loss of feedwater emergency protection system is triggered, not only are the emergency systems activated but the main circulation pumps are disconnected after a certain time lag. This is done to stop the level in the separator dropping too much and to prevent cavitation disruption of the main circulation pumps, i.e. to ensure optimal conditions for effective natural circulation. As indicated above, the safety of disconnecting the main circulation pumps and of reactor shutdown cooling by natural circulation has been confirmed by numerous experiments and by operating experience from nuclear power plants.

Figure 2.6.5 shows the theoretical transient regime following total instantaneous cut-off of feedwater flow.

The safety of the reactor following accidents in the feedwater supply system has also been confirmed by operating experience at RBMK units.

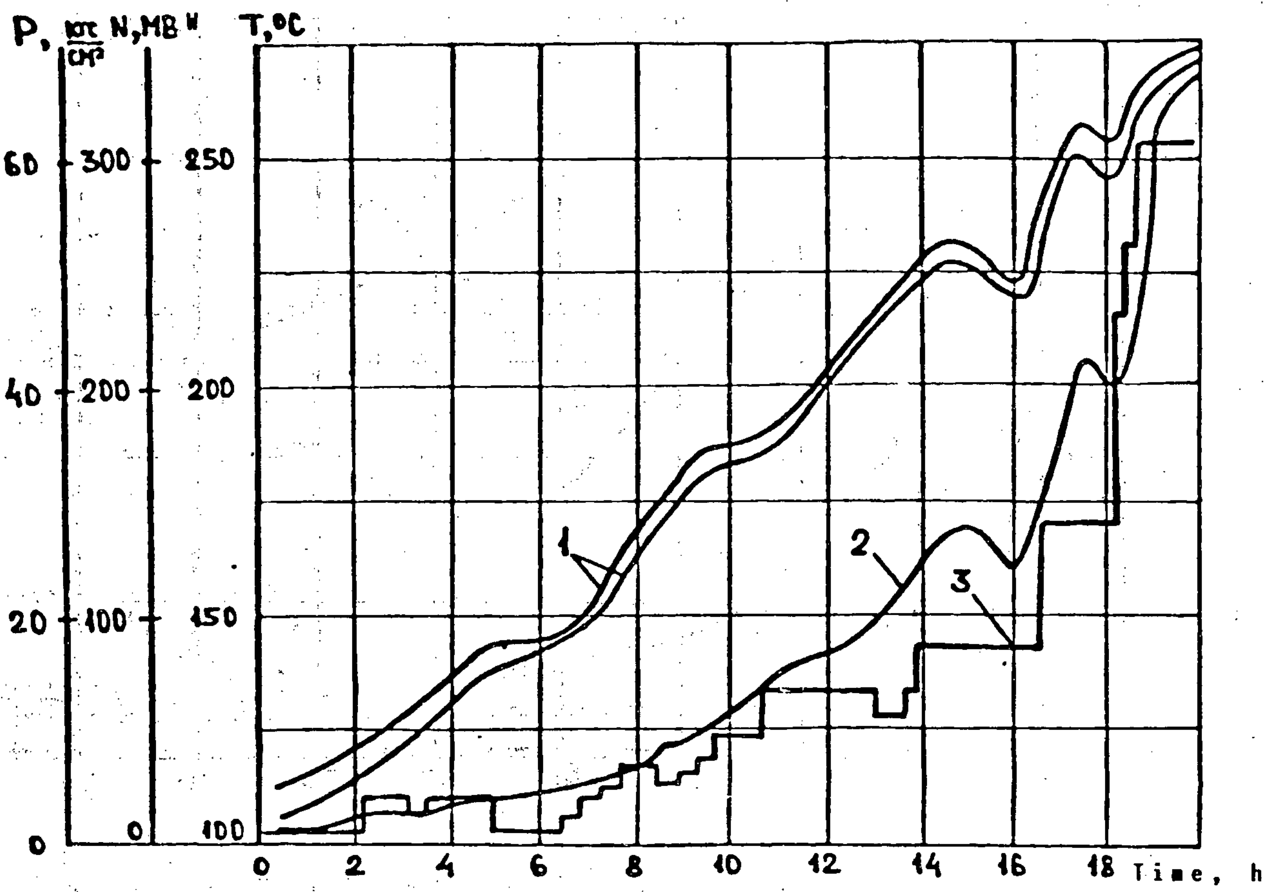


Fig 2.6.2. Evolution of reactor parameters during startup

I - Water temperature (T) in reactor circulation loops

2 - Pressure (P) in separators

3 - Thermal power (N) of reactor

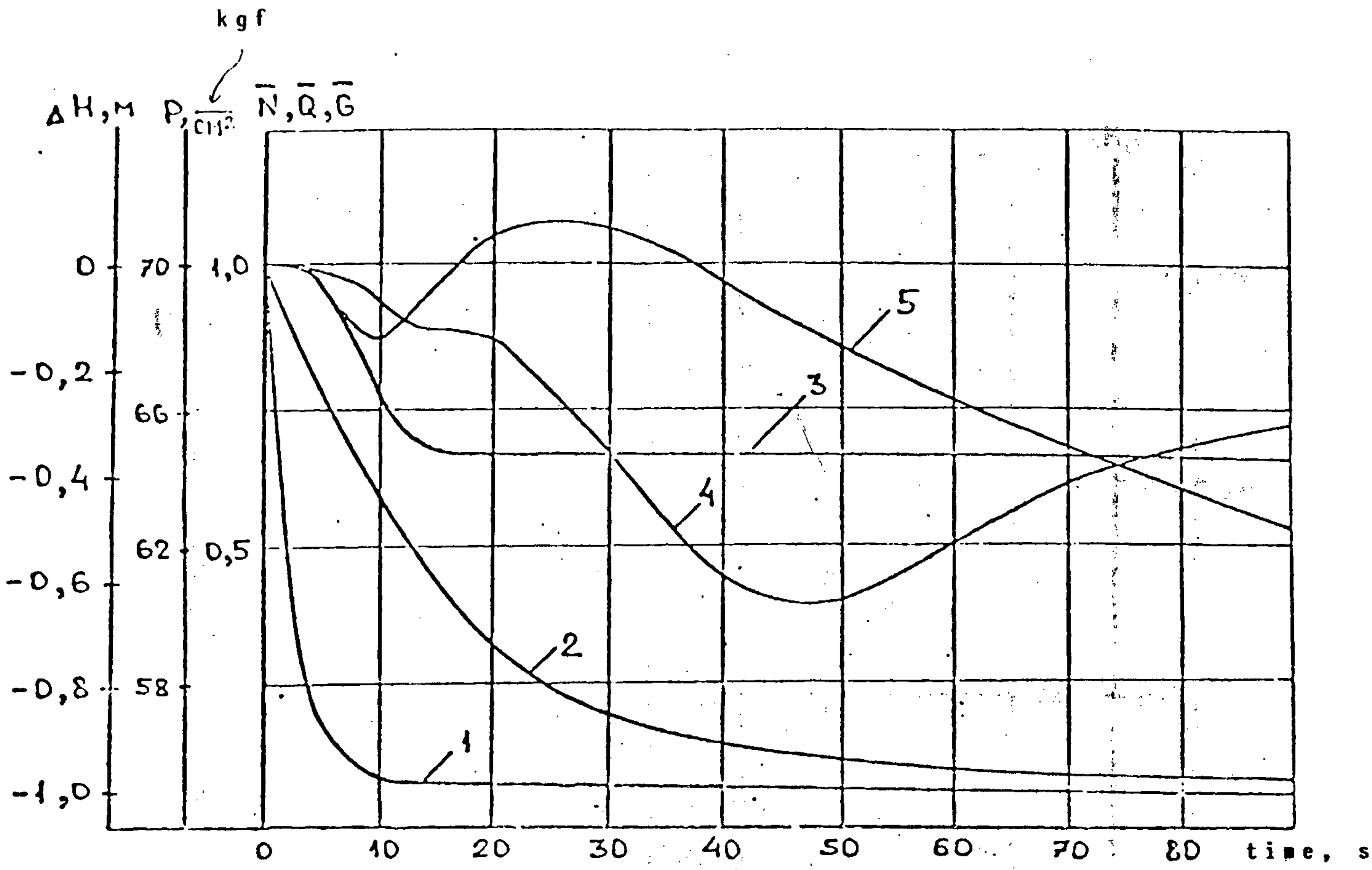


Fig 2.6 3. Behaviour of reactor parameters following activation of emergency protection system AZ-5

I - Nuclear power ($\bar{N}$)

2 - Thermal power ($\bar{Q}$)

3 - Circulating water flow rate ($\bar{G}$)

4 - Change in the level in the separators (ΔH)

5 - Pressure in separators (P)

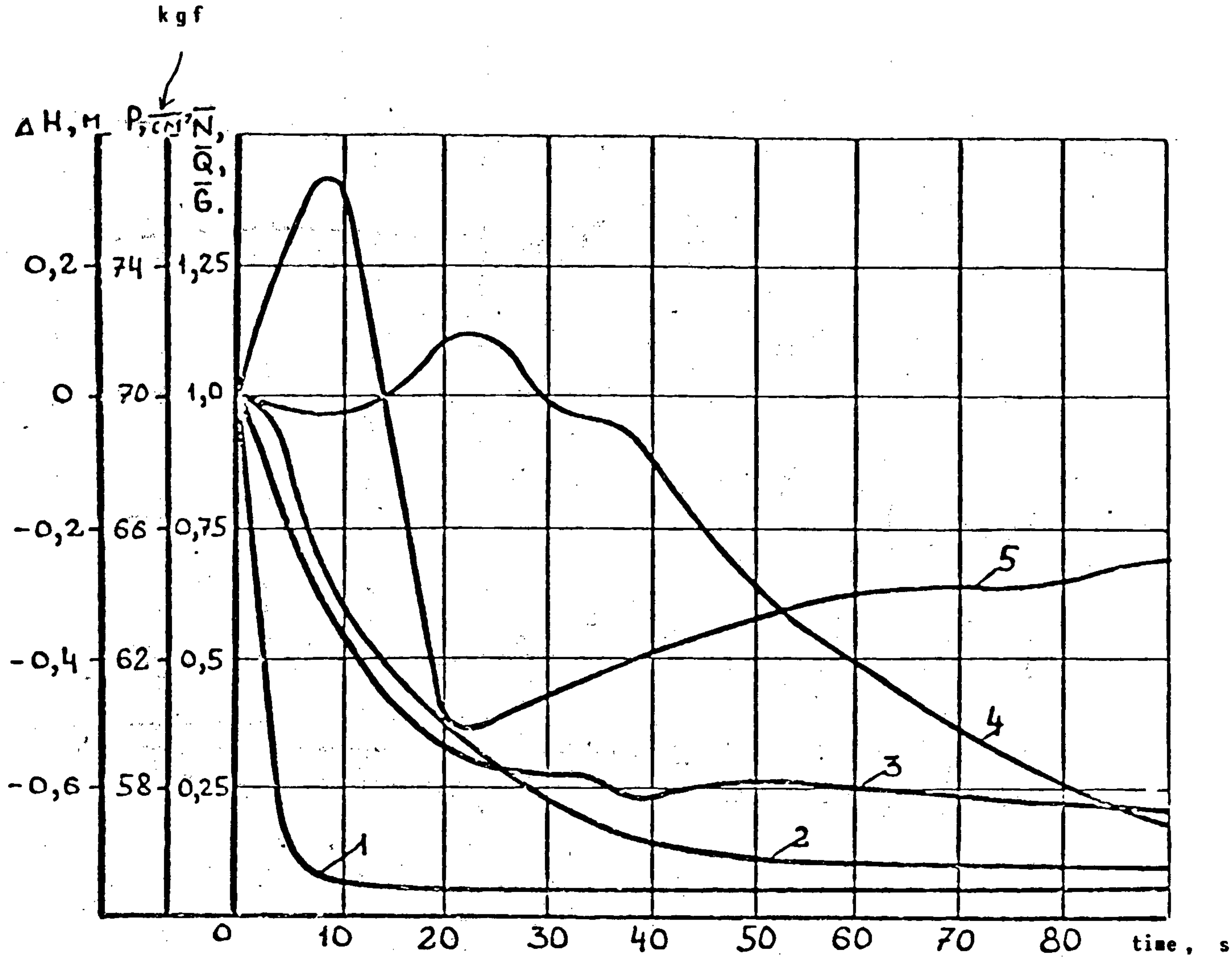


Fig 2.6.4. Behaviour of reactor parameters following loss of unit internal load. For legend, see Fig 2.6.3.

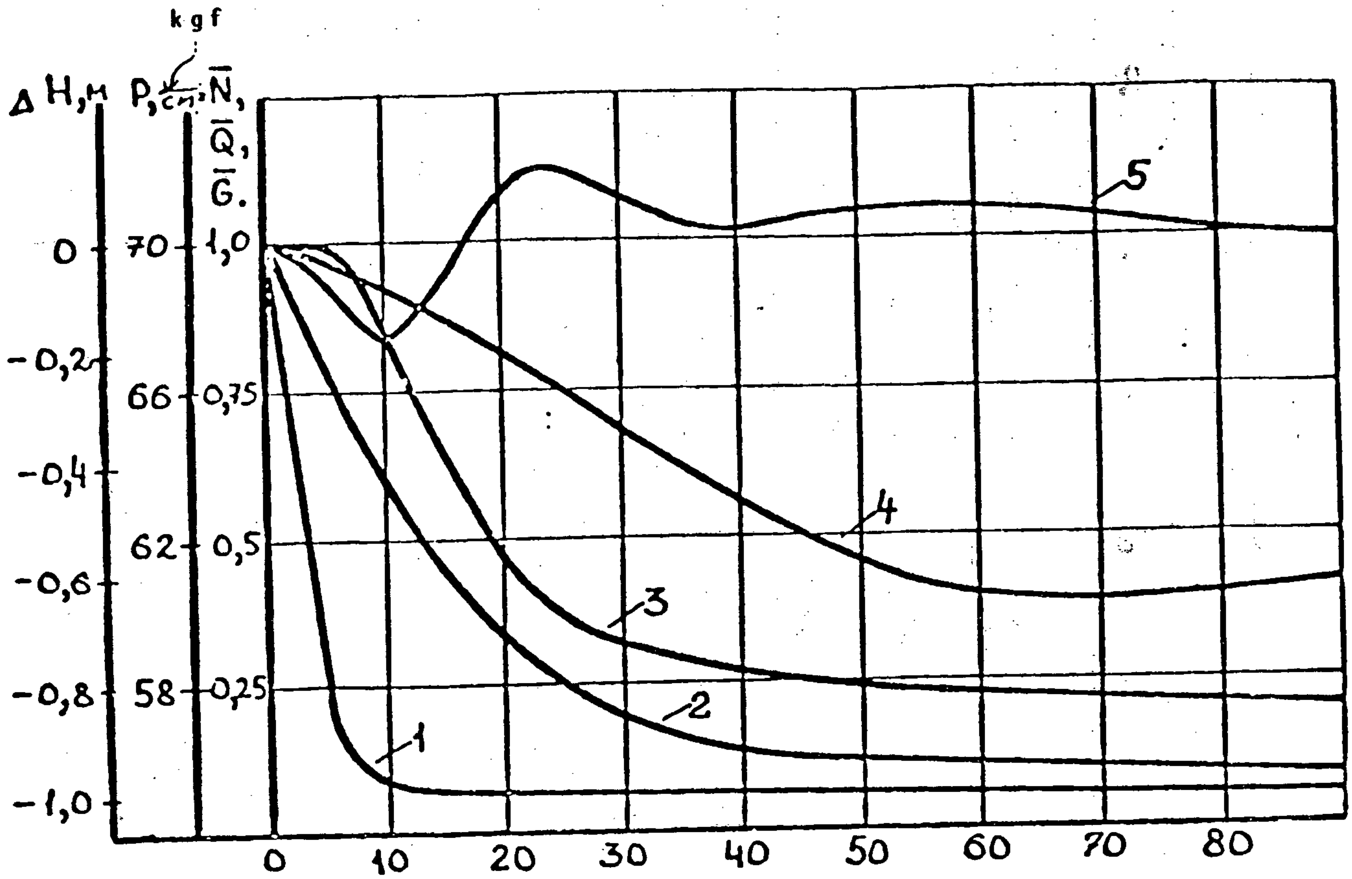


Fig. 2.6.5. Transient process following a total instantaneous cut-off of feedwater. For legend, see Fig. 2.6.3.

ANNEX 3

ELIMINATION OF THE CONSEQUENCES OF THE ACCIDENT AND DECONTAMINATION

3. ELIMINATION OF THE CONSEQUENCES OF THE ACCIDENT AND DECONTAMINATION

3.1. Progress and prospects for decontamination and startup of the first, second and third units

The surfaces of the equipment and compartments of the nuclear power plant were contaminated mainly through the ventilation system which continued to operate for some time after the accident at the fourth unit and also as a result of the spread of radioactive dust from the site of the plant. The highest levels were recorded for separate horizontal sections of the surfaces in the turbine building (up to 10^6 β-part/(cm^2/min), since it was contaminated for a prolonged period through the damaged roof.

The γ-radiation dose rate in the contaminated compartments of the first and second units on 20 May 1986 was 10-100 mR/h and in the turbine building 20-600 mR/h.

The washable nature of the materials (plastic, steel, concrete and various coverings) and the nature and levels of the surface contamination were taken into account when choosing the composition of the decontaminating solutions.

The spraying decontamination method was widely applied in the washing process making use of washing machines and fire hydrants. Some of the compartments were washed manually by wiping with a rag soaked in decontaminating solutions. The steam ejection method was also employed as well as dry decontamination methods using polymer covers.

The decontamination processes were monitored by direct measurement of the gamma background from the washable surfaces and by the smear method. As a result of decontamination, the contamination levels of the surfaces of the compartments and equipment were on the whole reduced to the norms established by the Radiation Safety Standard No. 76 and the Basic Health Regulations No. 72/80:

For service compartments - 2000 β-part/(cm^{2}min);

For semi-service compartments - 8000 β-part/(cm^{2}min).

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

3.2. Progress and prospects for decontamination of the power plant site

During the accident, radioactive material was scattered over the site of the plant and also fell on the roof of the turbine building, the roof of the third unit and on metal support pipes.

The plant site, the walls and the roofs of the buildings also had considerable contamination as a result of the fallout of radioactive aerosols and radioactive dust. However, the overall gamma background at the site consisting of radiation from the destroyed fourth unit greatly exceeded the radiation levels from the contaminated site and buildings. It should be noted that the contamination of the site was uneven.

In order to reduce the spread of radioactive contamination in the form of dust, the site, the roof of the turbine building and the sides of the roads were treated with rapid polymerizing solutions to reinforce the upper layers of the soil and to prevent the formation of dust.

In view of the complex nature of the work the nuclear power plant site was divided into zones for the purposes of decontamination.

The sequence of work carried out for each zone was based on the following criteria:

- The need for staff to work at facilities inside the zones;
- The principle "from dirty to clean" and taking account of wind roses;
- The need for subsequent work involved in startup of the units.

Decontamination in each zone was carried out in the following order:

- Removal of debris and contaminated equipment from the site;
- Decontamination of roofs and external surfaces of the building;
- Removal of a layer of soil, 5-10 cm thick, and transportation of it in containers to repositories (the solid waste storage vault of the fifth unit);
- Laying, where necessary, of concrete slabs or filling in with clean soil;
- Covering of slabs and non-concreted parts of the site with film-forming material;
- Restriction of access to the treated site.

The total number of sites treated ranged from 15 000-35 000 m^2 per 24 hours. As a result of these measures the overall gamma background in the area around the first unit was reduced to 20-30 mR/h. The fact that the residual background was caused mainly by external sources demonstrates that the decontamination of the site and buildings was fairly effective. However, significant improvement in the radiation conditions over the whole site of the nuclear power plant and particularly in the areas around the third and fourth units will be possible only after the destroyed reactor has been enclosed.

3.3. Progress and prospects for decontamination of the 30-km zone and its return to economic activity

The formation of the radioactive trail following a single release of effluent ends after about a year. After this period there is a significant redistribution of the radionuclides among the elements of the landscape in accordance with the characteristics of the relief. The most intensive redistribution of radioactivity (secondary transfer) occurs during the first 3-4 months after the release, particularly during the course of active biological and atmospheric processes (growth, development and drying off of plants, rains and winds). The loosely attached part of the radioactive substances which have settled on the surface of the soil and vegetation are subject to considerable redistribution. In coniferous forests such redistribution ends only after 3-4 years (after complete renewal of the needles).

For these reasons the radiation conditions within the 30-kilometre zone will continue to change significantly for 1-2 years particularly in regions with a high contamination level gradient.

Therefore the measures taken to decontaminate populated areas generally result only in a temporary improvement in the radiation conditions.

All this leads one to conclude that the evacuated population can only return to the area after the radiation conditions over the whole territory of the contaminated zone have stabilized (when releases from the reactor have ceased, the industrial site has been decontaminated and the radioactivity has been fixed over the territory where there is an increased contamination level). Conditions will stabilize most rapidly in regions of the zone with a low contamination level gradient (for example in the northern and southern projections of the radioactive path).

In order to decide whether agricultural production can be resumed and the evacuated population can return, it is necessary to have reliable information about the concentrations of long-lived radionuclides (strontium-90 caesium-137) in the soils and the crops cultivated on those soils. Soil samples have now been taken from all the fields at the collective and state farms in the region. When these samples have been analysed, cartograms will be drawn up showing the contamination of the agricultural lands and indicating the radionuclides.

Other radionuclides making up the contamination (zirconium-90, niobium-95, ruthenium-103 and 106, cerium-141 and 144, caesium-134, barium-140, strontium-89) and accounting for more than 90% of the total activity will not be limiting factors in the future either because they have short half-lives or because they are not easily absorbed from the soil by plants.

In principle the contaminated lands could be reused for agricultural purposes. General organizational and technical principles for agricultural management under such conditions have been worked out and numerous recommendations established for specific aspects. Since the agricultural conditions of Poless'ye are very specific and the nature of the radioactive contamination has not yet been studied in detail, specific evaluations can be made only when the specific data have been obtained. Resumption of agricultural activity in these areas requires:

(a) The re-organization of agricultural specialization in accordance with the contamination levels of the lands used; exclusion of production and produce from directly entering human food; primary seed production, industrial production and animal fodder production;

(b) The implementation of special measures aimed at the durable fixation and consolidation of radionuclides in a form which is inaccessible to plants for a prolonged period with subsequent cultivation by applying sorbents (clayey suspension, zeolites) to the upper contaminated layer of soil;

(c) The implementation of special decontamination measures involving the removal of the contaminated surface layer of turf directly by mechanical means or after consolidation using chemical agents (latex emulsion SKS-65 gp).

The measures taken to enable the land to be reused for agricultural purposes will be differentiated according to the time and level of contamination of the territory.

In the evacuation zone and in the strict control zone, agricultural harvesting work is being carried out as normal in accordance with the special measures worked out together with the State Agricultural Programme of the USSR and Ukranian SSR and the USSR Ministry of Health.

With regard to the surface contamination of vegetation and soils in 1986, the basic special requirements for the organization and the technological aspects of the work can be summarized as follows:

(a) To reduce to a minimum the mechanical cultivation of soils with increased dust formation;

(b) Grain and industrial crops are being harvested by direct combine harvesting and depending on the actual contamination levels of the produce are being used (after being stored) for food purposes, fodder, seed and industrial reprocessing;

(c) A compulsory requirement after the harvesting of perennial grasses and winter crops is the introduction of lime, mineral fertilizers and sorbents to increase soil fertility and reduce the entry of radionuclides into agricultural production.

When considering the fate of the contaminated forests one has to bear in mind their well-known role as absorbents and accumulators and conservers of moisture in forest-steppe and steppe regions.

Research has also shown that in conditions of radioactive contamination, forests also act as accumulators of radioactive substances, first in the crown then in the forest litter. The radionuclides fixed in the litter will for a long time be excluded from the radiation chains.

Therefore at present, the majority of experts believe that the best way of dealing with the contaminated forests is to increase the fire-prevention service.

At present, special agrotechnical and decontamination measures, which are designed to enable the contaminated lands to be reused for economic purposes, have been developed and are being implemented based on the evaluations of the contamination conditions of the soil and vegetation cover in the 30-km zone. These measures include changing the traditional system of soil cultivation in this region, the use of special dust-suppression compounds, changing the harvesting and crop processing methods and so on.

The level of radioactive contamination of houses and buildings in the countryside in the 30-kilometre zone fluctuates within significant limits. Typical building materials are bricks, wood (boards) both unpainted and painted, where the condition of the paint varies, slate and roofing iron.

Decontamination was carried out by spraying the surfaces with decontaminating solution at a flow rate of 10-15 L/m^2. Automatic filling machines were used.

As a result of decontamination, the radiation dose rate from the buildings dropped to the background levels for this region generally, the β contamination did not exceed 1000 β-part./(cm^2 min).

After washing the buildings, the radioactive contamination of the earth along the walls increased by 2-2.5 times and therefore this earth was dug over or removed with bulldozers and taken away.

The transport vehicles were decontaminated by the spraying and steam ejection methods using the above solutions.

ANNEX 4

ESTIMATE OF THE AMOUNT, COMPOSITION AND DYNAMICS OF THE DISCHARGE OF RADIOACTIVE SUBSTANCES FROM THE DAMAGED REACTOR

4.1 Amount of radioactive substances discharged from the reactor

The initial information for estimating the amount of these substances discharged from the damaged reactor consisted in the material obtained from aerial gamma-ray photography of the region of the Chernobyl' plant and the national territory carried out by the UNKhV with helicopters of the Air Force and the USSR State Committee on Hydrometeorology and Environmental Protection (Gaskomgidromet) as from 1:05 a.m. on 25 June 1986.

In order to determine the amount of radionuclides, the data from the aerial gamma-ray photography were plotted on a map of the district, isodose lines were drawn and the areas encompassed by these curves were calculated. The results of the estimates of this amount, as at 26 June 1986, are given in Table 4.1 in absolute and relative values.

The data of this table show that the total radioactivity of the fission products discharged from the damaged reactor and which had settled on the ground in a 30-km zone, amounted to 8-14 MCi. An analysis of the findings obtained showed that at the time the intense discharge of fission products from the reactor ceased on 6 June 1986, the amount of radionuclides present in the 30-km zone was approximately 20 MCi. It should be noted in particular that more than half of this activity is found in a zone with $R > 20$ mR/h, in an area consisting all told of 17% contaminated land and including the grounds of the Chernobyl' plant.

According to an analysis of Goskomgidromet's aerial gamma-ray photography, beyond the limits of the special zone, the activity of the radionuclides settling on the ground was 10-30 MCi. It follows from an analysis of the data that the total activity of the radionuclides released from the disabled reactor to the environment does not exceed 50 MCi, i.e. it represents approximately 3-4% of the total activity of the fission products in the reactor of the fourth unit of the Chernobyl' plant on 6 May 1986.

An independent estimate of the amount of fission products discharged from the damaged reactor was made by experts of the V.G. Khlopin Radium Institute. The amount of fuel present in these zones was determined on the basis of scanning with a collimated detector from a helicopter flying over the production area at a height of 300 m, analysis of samples taken in the 30-km zone and use of correlation ratios between the gamma activity of Ce and the alpha activity of Pu. This value is somewhat higher than the one obtained from the analysis of the isodose data. The relative values for the distribution of fission products over the production area and on the roof of the Chernobyl' power plant building are given in Fig. 4.1.

4.2. Composition of fission products of uranium and other radionuclides released from the damaged reactor

As the initial information for estimating the composition of the radionuclides released by the damaged reactor, use was made of radiometric investigations of aerosol samples and soil specimens carried out by the V.G. Khlopin Radium Institute and the I.V. Kurchatov Atomic Energy Institute in the period from 6 to 30 May 1986. These data led to the conclusion that the composition of the fission products (except for gaseous I, Te, Cs) discharged by the damaged reactor is similar to that of the fission products in the fuel in the reactor itself. This is confirmed, in particular, by the averaged data from studies of the soil and grass cover in the zone from 1.5 to 30 km from the reactor. These are shown in Table 4.2.

In the 30-km zone, dozens of soil samples were examined for their content of transuranium elements with respect to alpha radiation. The radioactivity of the samples ranged from 2 to 2000 Bq/g and was dependent on 90% ^{242}Cm. Approximately 10% of the alpha radioactivity was associated with isotopes of masses 238, 239 and 240. The radioactivity of ^{238}Pu is about 40-70% relative to the sum of the radioactivities of the nuclides ^{239}Pu and ^{240}Pu.

The soil sample in a radius of 1.5 km having the relatively largest concentration of transuranic elements may be considered one found in a discharge in a south-western direction at the end of a sector with a contaminated forest. The results of an analysis of this sample, taken on 8 May 1986 on the surface of a road, are shown in Table 4.3. The total alpha radioactivity of the sample is 1.3×10^4 Bq/g.

The radioactive composition of alpha emitters in the air samples (filters) and soil samples, according to the data of the I.V. Kurchatov Atomic Energy Institute are shown in Tables 4.4 and 4.5, and according to the data of the Khlopin Radium Institute in Table 4.6.

Investigations of the aerosol composition of air samples (with pumping through a filtering tissue) also confirm the transport via airborne dust of both volatile and slightly volatile chemical elements without distinct fractionation, with the exception of iodine, ruthenium and tellurium. The aerosol samples were taken at a height of 200 m above the damaged reactor and at a height of 3 m above the earth at 10 fixed points in the production area. Table 4.7 shows the results of the aerosol measurements at a height of 200 m, and Table 4.8 at a height of 3 m. Points 3 and 10 of Table 4.8 refer to the northern direction from the Chernobyl' power plant building and points 8 and 9 to the southern direction. All four points are located on a line running approximately 150 m more to the east of the damaged reactor (see Fig. 4.1).

The data of Table 4.7 indicate a sharp drop in specific aerosol activity after 6 May 1986, which is evidence of the dynamics of fission product discharge from the damaged reactor with time. Up to 7 May 1986, the reactor was a source of elevated release of radionuclides, but after 6 May 1986 it ceased being in any degree the determining factor in the formation of aerosol radioactivity above the production area. This formation was determined initially by processes of dust formation and secondary wind transport of radionuclides over the production area as a whole (Table 4.9).

The concentration of radioactive aerosols at a height of 200 m was the same as that at a height of 3 m (data of 9 and 11 May for 200 m and of 12 May for 3 m). After 12 May, the concentrations of aerosols at a height of 200 m become approximately 100 times lower, but at 3 m above the production area they subsequently showed little change. This can be seen by comparing the data of Table 4.8 with those of Table 4.9. The latter gives the results of a determination of aerosol concentrations at a height of 3 m above the production area, carried out on 22 May 1986 at points located relatively close to the points referred to in Table 4.8 (see Fig. 4.1).

The compositions of the air and fallout samples showed the presence of "hot" particles enriched primarily in radionuclides of one type. Fig. 4.2 and 4.3 show the results of measurements of the radionuclide composition of such particles. As can be seen, there are particles containing practically nothing but Cs or Ce. There is a tenfold increase in the content of ^{140}Ba over the theoretical value.

4.3. Dynamics of radionuclide discharge from the damaged reactor

The material used as initial information for analysing the dynamics of the discharge of radionuclides from the damaged reactor was data from systematic studies of the radionuclide composition of aerosol samples taken above the fourth unit of the Chernobyl' plant on 26 April 1986. The results of the investigations are shown in Table 4.10 and in Fig. 4.2.

Analysis of these data led to the conclusion that the release of radionuclides beyond the limits of the damaged nuclear power plant unit was a process extended over time, consisting of several stages. The dynamics of the discharge process are characterized especially prominently by the data of Table 4.11, whch presents non-dimensional values of fission-product release (normalized in terms of ^{131}I) in time.

In the first stage, the mechanical discharge of dispersed radioactive fuel took place as a result of an explosion in the reactor. The composition of radionuclides at this stage of discharge corresponded approximately to the composition of fission production in spent fuel, but enriched n volatile nuclides of iodine, tellurium and caesium.

In the second stage, from 26 April to 2 May, the capacity for discharge beyond the limits of the damaged unit decreased owing to the measures undertaken to terminate the burning of graphite and the filtration of substances emerging from the core. In a first approximation, the reduction in the power of discharge in this period can be represented in the form

$$Q(\tau) = Q_0 e^{-0.5\tau} \qquad (4.1)$$

where Q_0 is the power of discharge immediately after the explosion (Ci/d);
τ is the time after the beginning of the damage (days).

In this period the composition of radionuclides in the discharge was also similar to their composition in the fuel. At this stage, finely disperse fuel escaped from the reactor directly with a flow of hot air and with products from the burning of graphite.

The third stage of release was characterized by a rapid increase in the power of radionuclide discharge beyond the limits of the reactor unit. In the initial part of this stage it was primarily the escape of volatile components, especially iodine, that was observed; subsequently, the composition of the radionuclides again resembled that of their composition in spent fuel (on 6 May 1986). The power of fission product discharge in the third stage can be described by the expression

$$Q(\tau) = \text{const.}\ e^{\alpha\tau}, \qquad (4.2)$$

where $\alpha = (6\text{-}8) \times 10^{-2}$ 1/h.

The fact that the discharge was of this nature was apparently caused by the heating of the fuel in the core to a temperature above 2000°C due to residual heat release. As a result of temperature-dependent migration of fission product, and also possible carbidization of uranium dioxide, there was a leak of fission products from the dioxide and their escape either in aerosol form or in products of the burning of graphite (graphite particles).

The final - fourth - stage, starting on 6 May, is characterized by a rapid decrease in the escape of fission products from the fuel and virtual termination of discharge (Table 4.13), which was a consequence of the special measures taken, by the formation of more refracting fission products as a result of their interaction with the materials introduced.

Main conclusions:

1. The total release of radioactive substances (not including radioactive noble gases) was about 50 MCi, which represents 3.5% of the total amount of radionuclides in the reactor at the time of the accident. These data were calcualted on 6 May 1986, with allowance for radioactive decay.

2. The composition of radionuclides in the discharge due to the accident was notable for its elevated content of volatile iodine and tellurium.

3. Generalized quantitative information concerning the variation in power of discharge with time and composition of the radionuclides released from the damaged reactor are given in Table 4.13 and 4.14 and in Fig. 4.4.

Table 4.1 Estimate of the amount of radionuclides on the ground in a 30-km zone of the region of the Chernobyl plant as of 26 June 1986

Point No.	Zone with R, mR/h	Area, $км^2$	Activity: Absolute, MCi	Activity: Relative, %
1	R > 20	870	5-8,7	63,0
2	I0 < R ≤ 20	480	0,8-1,4	I0,2
3	5 < R ≤ I0	II00	1÷1,7	I0,8
4	3 < R ≤ 5	2780	I,3-2,2	I6,0
	Total:	5230	8÷I4	I00,0

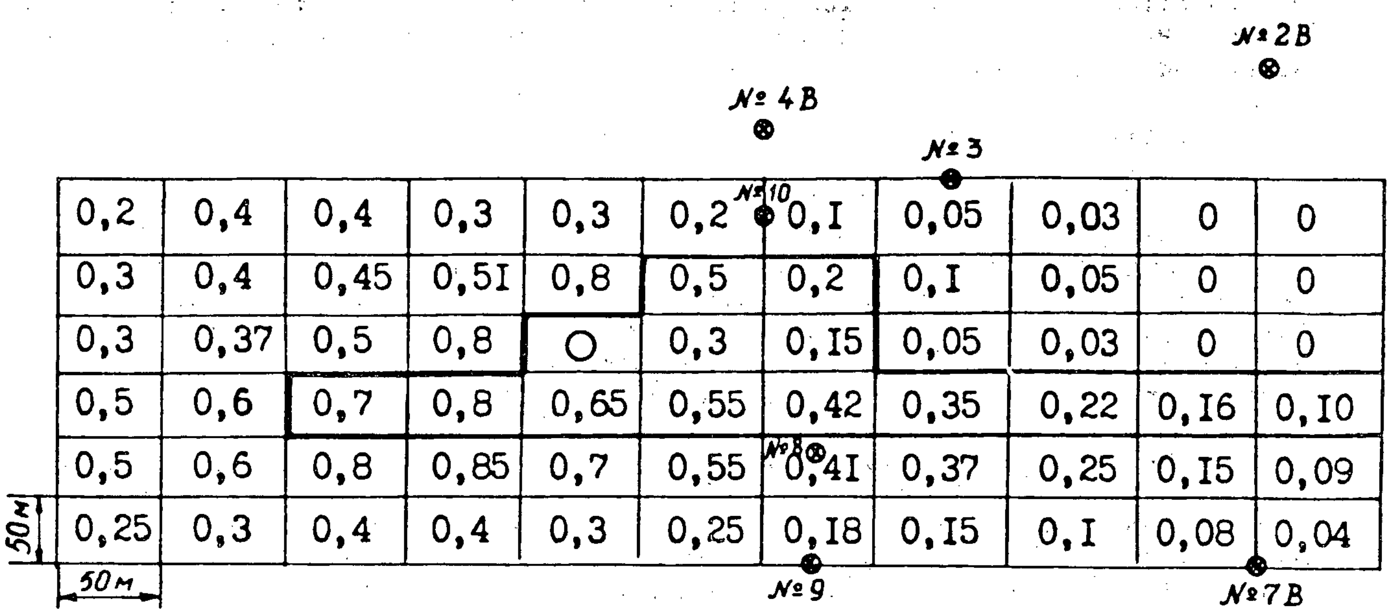


Fig. 4.1 Relative distribution of gamma-emitting fission products over the production area of the Chernobyl nuclear power plant

Table 4.2 Data from radiometric measurments of soil samples on 17 May 1986 on the northern path of fallout within the limits of the 30-km zone

Radionuclide	Specific activity Bq/g	Specific activity Ci/km^2	Content in sample %	Content in irradiated reactor fuel, %
^{141}Ce	$3{,}2 \cdot 10^3$	$5{,}1 \cdot 10^2$	15,8	18,3
^{132}Te	$3{,}4 \cdot 10^2$	$5{,}4 \cdot 10^1$	1,7	0,22
^{131}I	$3{,}1 \cdot 10^3$	$5{,}1 \cdot 10^2$	15,8	2,8
^{103}Ru	$3{,}5 \cdot 10^3$	$5{,}6 \cdot 10^2$	17,3	21,4
^{106}Ru	$9{,}6 \cdot 10^2$	$1{,}5 \cdot 10^2$	4,6	16,9
^{134}Cs	$1{,}6 \cdot 10^3$	$2{,}5 \cdot 10^2$	7,7	4,5
^{137}Cs	$1{,}7 \cdot 10^3$	$2{,}7 \cdot 10^2$	8,3	3,4
^{95}Zr	$4 \cdot 10^3$	$6{,}4 \cdot 10^2$	19,8	23,0
^{140}Ba	$1{,}8 \cdot 10^3$	$2{,}9 \cdot 10^2$	9,0	9,6

$$1 \text{ Ci} = 3{,}7 \cdot 10^{10} \text{ Bq}$$

Table 4.3 Composition of radionuclides in a soil sample of 8 May 1986 at a distance of 1.5 km from the reactor

Nuclides	Specific activity, 10^5 Bq/g
^{95}Zr	36
^{103}Ru	1,7
^{131}I	6,3
^{140}Ba	21
^{141}Ce	28
^{144}Ce	17
^{239}Np	6,4
Alpha-radioactive nuclides	0,13

Note: 1 Bq = $2{,}7 \cdot 10^{-11}$ Ci

Table 4.4 Radionuclide composition of alpha-emitters in the air and some reference nuclide ratios

Nuclide	Atmospheric activity, Ci/L	$\frac{\text{Nuclide}}{^{144}Ce}$	$\frac{^{239+240}Pu}{^{242}Cm}$	$\frac{^{238}Pu}{^{239+240}Pu}$
^{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	$6,7.10^{-13}$	$8,7.10^{-3}$		

Table 4.5 Radionuclide composition of alpha-emitters on soil samples and some reference nuclide ratios

Nuclide	$\frac{\text{Activity of soil, Ci}}{\text{Sample}}$	$\frac{\text{Nuclide}}{^{144}Ce}$	$\frac{^{234+240}Pu}{^{242}Cm}$	$\frac{^{238}Pu}{^{239+240}Pu}$
$^{239+240}Pu$	$2,5.10^{-10}$		$3,5.10^{-2}$	0,72
^{238}Pu	$1,8.10^{-10}$			
^{242}Cm	$7,1.10^{-9}$			

Table 4.6 Radionuclide composition of alpha-emitters in soil samples, according to measurement data of the Radium Institute

Nuclide	Activity, Bq/h		Nuclide / ^{144}Ce		$^{239+40}Pu / ^{242}C$		$^{238}Pu / ^{239+40}Pu$	
	I	II	I	II	I	II	I	II
$^{239+240}Pu$	790	5,2	$6,3 \cdot 10^{-4}$	$5,2 \cdot 10^{-4}$	$8,1 \cdot 10^{-2}$	$7 \cdot 10^{-2}$	0,44	0,38
^{238}Pu	348	2,0	$2,8 \cdot 10^{-4}$	$2 \cdot 10^{-4}$				
^{242}Pu	9784	74	$7,8 \cdot 10^{-3}$	$7,4 \cdot 10^{-3}$				
^{144}Ce	$1,23 \cdot 10^{6}$	$1,0 \cdot 10^{4}$	1	1				

Table 4.7 Radionuclide composition of aerosols at a height of 200 m, Bq/L

Nuclides	Measurement data			
	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 a height of 3 m, Bq/L

Nuclides	Concentration on 12 May 1986 at points			
	3	10	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	I00	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 20 May 1986 at a height of 3 m, Bq/L

Nuclides	Points		
	2C	4C	7C
^{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

Table 4.10 Relative content of radionuclides in the air above the Chernobyl nuclear power plant δ_i, % *)

Nuclides	26.04.86	29.04.86	02.05.86	03.05.86	04.05.86	05.05.86	Rel. content, α_i of radionuclides in fuel on 26 April 1986
^{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
$\sum_i A_i$	$3,6\cdot\mathrm{I0}^{-7}$	$3,2\cdot\mathrm{I0}^{-7}$	$5\cdot\mathrm{I0}^{-8}$	$7\cdot\mathrm{I0}^{-8}$	$\mathrm{I}\cdot\mathrm{I0}^{-6}$	$7\cdot\mathrm{I0}^{-9}$	

*) $\delta_i = (A_i / \sum_i A_i)\cdot \mathrm{I00}\%$, where A_i is the activity of the i-th radionuclide; the data are rounded off

Table 4.11 Values of Ω */, characterizing the emission of non-volatile fission products relative to ^{131}I, taking ^{95}Zr and ^{141}Cr as examples, for the period 26 April to 13 May 1986

Date	^{95}Zr	^{141}Cr
26.04 **)	II	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
II.05	23	37
I3.05	I3	22

*)

$$\Omega_i = \frac{[A(^{131}J)/A(i)] \text{ in air}}{[A(^{131}J)/A(i)] \text{ in fuel}}$$

$$i = {}^{95}Zr;\ {}^{141}Ce;$$

A - activity of the radionuclide

**) Initial discharge

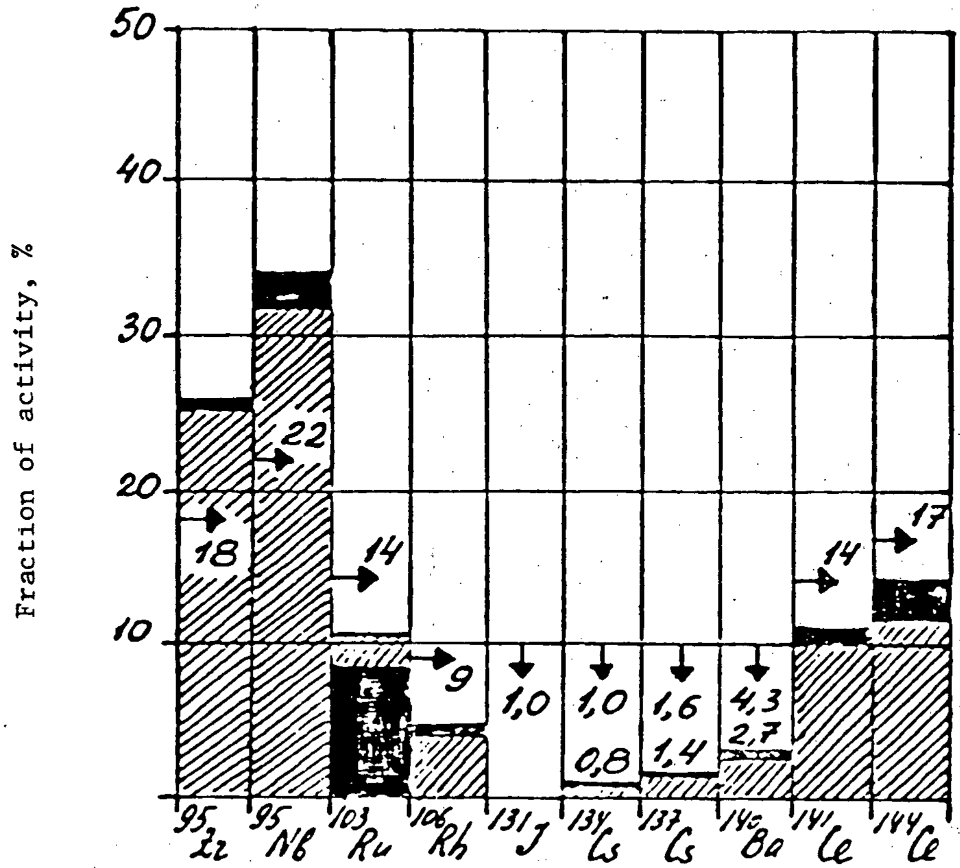


Fig. 4.2. Fraction of radionuclide activity in the initial sample (▀) and the deposit (▨)

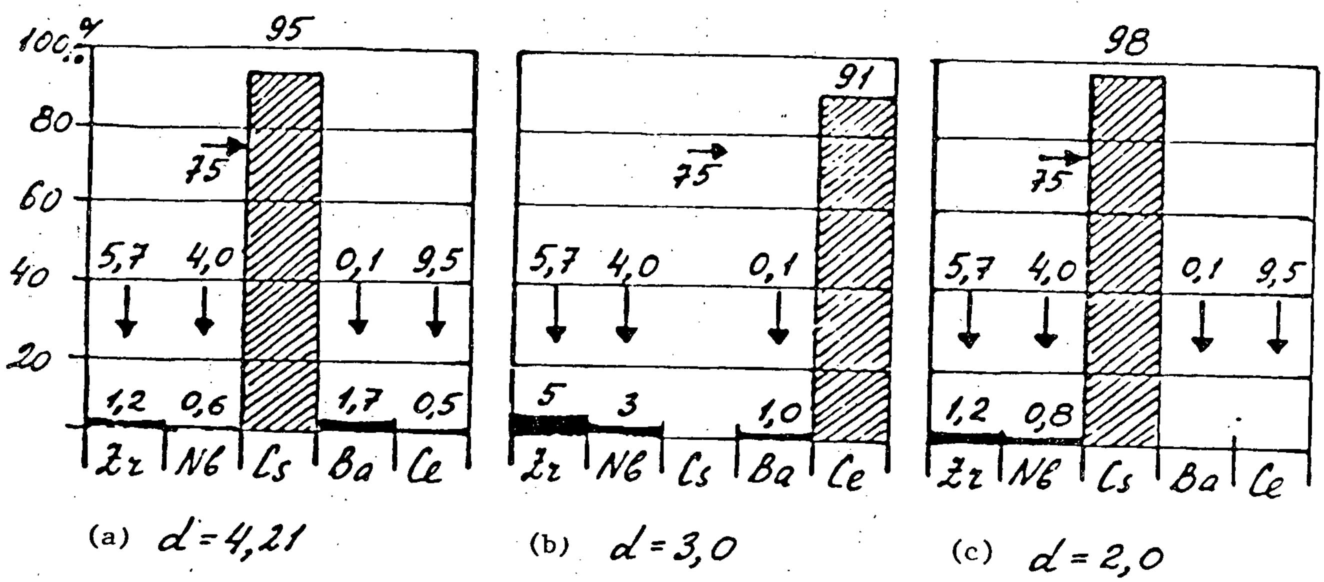


Fig. 4.3 Fraction of radionuclides in "hot" particles, fractionated in clerici solutions with densities of 4.2; 3.0; 2.0 g/cm^3 from samples of sand and dust (No. 16-LPD)

Content of caesium radionuclides (^{134}Cs and ^{139}Cs together) in (a) 5.0×10^{-9} g/g; (c) 1.4×10^{-9} g/g

Fig. 4.4

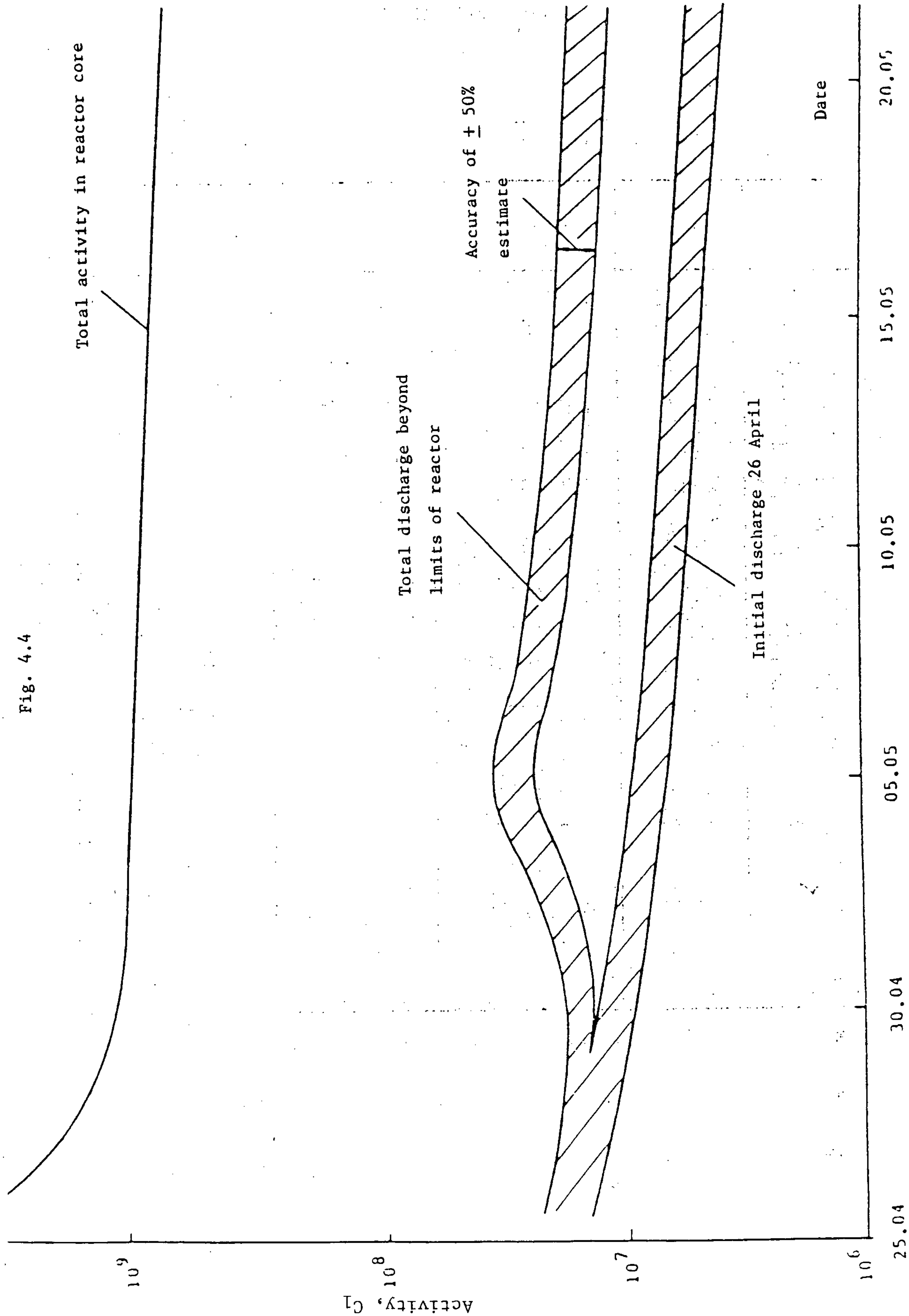

Total activity in reactor core
Accuracy of ± 50% estimate
Total discharge beyond limits of reactor
Initial discharge 26 April
Date
25.04
30.04
05.05
10.05
15.05
20.05
10^9
10^8
10^7
10^6
Activity, Ci

Table 4.12 Results of measurements of power of discharge from the reactor and of an estimated amount in a cloud, on the basis of an analysis of air samples

Date of sampling	Time of sampling	Specific activity of air, Ci/L	Power of discharge, Ci/d	Estimate of amount in cloud, Ci/km^3
09.05	I8.30	$54 \cdot \mathrm{IO}^{-\mathrm{IO}}$	I2600	
II.05	I3.I5	$38 \cdot \mathrm{IO}^{-\mathrm{IO}}$	8700	
I3.05	I3.I5	$2 \cdot \mathrm{IO}^{-\mathrm{IO}}$	420*)	
I6.05	I3.I5	$8 \cdot \mathrm{IO}^{-\mathrm{IO}}$	I680**)	
I9.05	I3.I5	$0{,}05 \cdot \mathrm{IO}^{-\mathrm{IO}}$	50	
22.05	09.30	$0{,}05 \cdot \mathrm{IO}^{-\mathrm{IO}}$	50	
23.05	09.30	$0{,}02 \cdot \mathrm{IO}^{-\mathrm{IO}}$	20	
24.05	09.30.	$\mathrm{I} \cdot \mathrm{IO}^{-\mathrm{IO}}$		I00
25.05	09.30	$3 \cdot \mathrm{IO}^{-\mathrm{IO}}$		300
26.05	09.30	$0{,}2 \cdot \mathrm{IO}^{-\mathrm{IO}}$		20
27.05	09.30	$0{,}2 \cdot \mathrm{IO}^{-\mathrm{IO}}$		20
28.05	09.30	$0{,}2 \cdot \mathrm{IO}^{-\mathrm{IO}}$		20
29.05	09.30	$0{,}\mathrm{I} \cdot \mathrm{IO}^{-\mathrm{IO}}$		IO
30.05	I3.00	$2 \cdot \mathrm{IO}^{-\mathrm{IO}}$		200***)
0I.06	09.30	$0{,}2 \cdot \mathrm{IO}^{-\mathrm{IO}}$		20
02.06	I7.00	$0{,}25 \cdot \mathrm{IO}^{-\mathrm{IO}}$		25
03.06	I4.30	$0{,}\mathrm{I}2 \cdot \mathrm{IO}^{-\mathrm{IO}}$		I2
04.06	09.30	$0{,}08 \cdot \mathrm{IO}^{-\mathrm{IO}}$		8
05.06	09.30	$0{,}\mathrm{I}2 \cdot \mathrm{IO}^{-\mathrm{IO}}$		I2
06.06	09.30			I00****)

*) The sampling was made outside the loop, which led to a reduction in the concentration of nuclides.

**) The data were verified against the results of measurements of the Institute of Experimental Metrology

***) The sampling was made in daytime. This accounted for an increase in the concentration of nuclides due to an elevated dust condition due to work on the area of the power station.

****) After a rainfall.

Table 4.13 Daily discharge, q, of radioactive substances into the atmosphere from the damaged unit (not including noble gases)*

Date	Time after accident, days	q MCi **)
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	$\simeq$ 0,0I
23.05	28	$20 \cdot I0^{-6}$

1) Initial discharge

The value given is with allowance for decay on 6 May 1986.

At the time of discharge, the activity was 20-22 MCi.

The composition of the discharge is shown in Table 4.14.

*) The error in the release evaluation is ± 50%. Contributing to it are the dosimetric equipment error, the error in measuring the isotopic composition of air and soil samples and the error in averaging fallout over a large area.

**) The values of q are adjusted to 6 May 1986 with allowance for radioactive decay (the release on 26 April 1986 amounted to about 20 MCi at this moment in time). For the composition of the release, see Table 4.14.

Table 4.14 Radionuclide composition of discharge from the damaged unit of Chernobyl' nuclear power plant*/

Nuclide ***)	Activity of discharge, MCi		Fraction of activity discharge from reactor on 6 May 1986, %
	26.04.86	06.05.86**)	
I33Xe	5	45	Possibly up to 100
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	$3.I0^{-3}$	$2{,}I.I0^{-2}$	-"-

*) Error of estimate: ± 50%, explanation in Footnotes to Table 4.13

**) Total discharge up to 6 May 1986

***) The data presented relate to the activity of the main radionuclides measured on radiometric analyses

ANNEX 5

5. ATMOSPHERIC TRANSPORT AND RADIOACTIVE CONTAMINATION OF THE ATMOSPHERE AND OF THE GROUND

5.1. Formation of the basic contamination source - the cloud and the gas stream - as a result of the accident at the Chernobyl Atomic Power Station.

As a result of the accident, a significant quantity of the radionuclides which had accumulated in the reactor during operation escaped beyond the confines of the power station.

The cloud which formed at the time of the accident produced a radioactive trail on the ground in a westerly and a northerly direction depending on the meteorological conditions governing the transport of the air masses. Subsequently, for a considerable time, a stream of gaseous, volatile and aerosol products continued to flow from the accident zone. The most intense stream was observed during the first 2-3 days after the accident in the northerly direction, where the radiation levels reached 1000 mR/hour on 27 April and 500 mR/hour on 28 April at a distance of 5-10 km from the reactor site (at an altitude of 200 m). According to aircraft monitoring data, the height of the stream on 27 April exceeded 1200 m in the north-westerly direction at about 30 km from the reactor site, and the radiation levels at that height were about 1 mR/hour. During the following days, the height of the stream did not exceed 200-400 m.

Portions of the contaminated air masses (the cloud and portions of the stream of radioactive products) spread for considerable distances over the territory of the USSR depending on the wind direction.

Gamma-spectrum analyses of air samples revealed fission products which had accumulated in the reactor (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}, Nd^{147}) and also isotopes resulting from induced activity: Np^{239}, Cs^{134}.

A characteristic feature of the radioactive products escaping to the atmosphere is that they were enriched in iodine and caesium radionuclides. Table 5.1 gives calculated fractionation (enrichment) coefficients of various radionuclides relative to zirconium-95 based on analyses of atmospheric aerosol samples collected in the immediate vicinity of the reactor (the "close-in zone") during the early days after the accident.

5.2. Meteorological conditions affecting the dispersion of radioactive products from the Chernobyl Atomic Power Station

On 26 April 1986, the area around the Station was in a low-gradient pressure field with a slight wind varying in direction. At an altitude of 700-800 m to 1.5 km, the area around the Station was at the south-west periphery of a high-pressure zone with the air masses in this layer moving north-westwards at a speed of 5-10 m/sec. This was confirmed by measurements of radiation levels and radioactive fallout along the air particle dispersion trajectory at an altitude of 0.7-1.5 km (see Figs 5.1-5.3).

The further dispersion in the 0.7-1.5 km layer of air particles leaving the area around the Station on 26 April was in a north-westerly direction, subsequently turning to north.

On 26 April, the long-distance transport of the air masses in the ground layer was in the westerly and north-westerly directions, the air particle dispersion trajectory reaching areas on the frontier with Poland on 26 and 27 April; this was confirmed by radioactivity fallout measurements. On the following days, from 27 to 29 April, aircraft monitoring data indicated transport of radioactive products in the ground layer of air at a height of 200 m in a northerly and a north-westerly direction from the Station.

The meteorological conditions from 26 to 29 April which governed the dispersion of air masses in the area around the Station virtually determined the basic zone of close-in radioactive fallout to the north-west and the north-east of the Station. This was confirmed by aircraft measurements - made during the following days - of radiation level distributions on the ground in the close-in zone.

Subsequently, there was an insignificant escape of radioactive products from the area around the Station, with their transport in a southerly direction until 7-8 May, causing radioactive fallout in that direction.

5.3. Radioactive contamination in the area around the Station and assessment of the total quantity of radioactive products deposited as fallout in the close-in zone

As a result of the meteorological conditions governing the dispersion of radioactive products in the atmosphere and their precipitation on the underlying surface during the period 26-30 April, a zone of close-in radioactive fallout was formed. Since 29 April, there have been regular airborne gamma surveys of the distribution of radioation levels on the ground. Fig. 5.4 shows the distribution of radiation levels on the ground on 29 May.

From data on the gamma dose rate distribution on the ground at different times it was possible to estimate the total quantity of the radioactive products precipitated in the close-in zone of the radioactive trail.

Table 5.2 gives results obtained by integrating over the areas bounded by the various dose rate isolines (in (R/hour) x m^2) and also the total amount of radioactive products in the close-in zone of the radioactive trail (from the Station boundary out to a distance of 80 km) expressed in MeV/sec and in Curies.

During the period from 10 to 30 days after the accident, the following radionuclides were identified in the close-in zone of the fallout trail: Mo^{99}, Zr^{95}, Nb^{95}, Ce^{141}, Ce^{144}, I^{131}, Te^{132}+ I^{132}, Ru^{103}, Ru^{106}, Ba^{140}+La^{140}, Cs^{134}, Cs^{137}, Sr^{89}, Sr^{90} and Y^{91}. Plutonium isotopes were found on the ground surface.

On the basis of data on ground contamination densities in different zones of the trail, the amounts of individual radionuclides deposited as fallout in the close-in zone of the trail were estimated by comparing with the dose rate at "D"+15 (see Table 5.3).

5.4. <u>Radioactive contamination of the atmosphere and the ground in the USSR</u>

During the period 26 April - 5 May there was formed a zone of radioactive fallout within and beyond the territory of the USSR as a function of the meteorological transport conditions. The amount of radioactive products deposited in the European part of the USSR was about 1.2 x 10^8 (R/hour) x m^2 or 4.0 x 10^{17} MeV/sec. Thus, the total fallout in the close-in and far-out zones was about 7.0 x 10^{17} MeV/sec by 5 May 1986 - about 3% of the total energy production of the radioactive products in the reactor at that time.

Depending on the meteorological transport conditions, the radioactive products entered the ground layer of the atmosphere at different times for different observation points.

Table 5.4 shows the change in the concentrations of some radionuclides at a permanent observation point belonging to the State hydrometeorological network (Varyshevka, about 140 km south-east of the Station).

Figures 5.5 and 5.6 gives results of determinations of various radionuclides in the ground air layer at the Berezinsky National Park carried out at a complex background monitoring station located 120 km to the north-east of Minsk.

The fallout of radioactive products from the atmosphere was monitored by a network of stations run by the State Committee for Hydrometrology using collection panels ("planchettes") which were exposed for periods of 24 hours.

Table 5.5 gives observation results for Kiev and Kaliningrad.

5.5. Radioactive contamination of rivers and reservoirs

Information about the radioactive contamination of rivers and reservoirs was obtained through the isotopic analysis of one-litre water samples taken from the surface water layer at regular intervals (of 1-3 days) at the mouth of the rivers Pripyat', Teterev, Irpen' and Desna and at the point of water extraction from the river Dnepr (Vyshgorod). Starting on 26 April, surface water samples were taken from all parts of the Kiev reservoir in a major sampling exercise using boats.

The investigations showed that the radioactive contamination of rivers and reservoirs was due initially to aerosol fallout onto the surface of water bodies and subsequently to runoff from contaminated catchment areas (during May there was almost no rainfall in this region). The initial results of determinations of radioactive substances in the water bodies investigated are given in Table 5.6.

Table 5.7 gives, for two water bodies, maximum concentrations found during the observation period (starting 1 May).

The highest iodine-131 concentrations in the Kiev reservoir near the point of water extraction from the river Dnepr were observed on 3 May. This reflected the fallout of radioactive aerosols onto the surface of the Kiev reservoir and the time taken to reach this point by contaminated water from the river Pripyat'.

During the period 13-20 May, the total beta activity of the water in the river Dnepr near the Kiev Hydro Station was in the range (1-5) x 10^{-9} Curie/litre.

5.6. Plutonium contamination of the atmosphere and the ground

Plutonium contamination investigations were carried out with the help of a special vehicle which was in use in obtaining air, soil and grass samples over a ring-shaped expanse of territory outside the 30-km zone (May 1986).

As can be seen from Table 5.8, at all sampling points the concentration of plutonium in the air was below the maximum permissible level (3 x 10^{-17} Curie/litre for Pu^{239}).

Table 5.9 gives values for the density of the plutonium contamination of soil and grass at various points.

Table 5.1

Radionuclide	Ce^{144}	Ce^{141}	Ba^{140}	I^{131}	Ru^{103}	Ru^{106}	Cs^{137}
Fractionation coefficient	1.23	1.03	0.84	5.22	2.21	1.7	5.64

Table 5.2

Date	Integral (R/hour) x m^2	Amount of radioactive products in trail		Fraction of total energy production of radioactive products in reactor zone on that date (%)
		MeV/sec	Curie	
11/5/86	7.9×10^7	3.3×10^{17}	1.2×17^7	1.6

Table 5.3

Radionuclide	Zr^{95}	Ce^{141}	Ce^{144}	I^{131}	Te^{132}	Ru^{103}	Ru^{106}
Amount in trail (Curie)	1.8×10^5	1.7×10^6	1.3×10^6	1.3×10^6	2.5×10^6	1.5×10^6	5.7×10^5
Fraction of amount in reactor zone (%)	1.5	1.7	1.0	5.1	5.0	1.4	0.8

Radionuclide	Ba^{140}	Cs^{134}	Cs^{137}	Sr^{89}	Sr^{90}	Y^{91}	Total amount
Amount in trail (Curie)	9.1×10^5	1.3×10^5	2.8×10^5	6.2×10^5	8.5×10^4	$6,4 \times 10^5$	1.12×10^7
Fraction of amount in ractor zone (%)	1.4	0.6	1.9	1.2	0.85	0.85	1.4

Table 5.4

(Curie/m^3)

Sampling date (1986)	I^{131}	Zr^{95}	Ba^{140}	Cs^{137}
26-27/4	3.2×10^{-12}			
27-28/4	5.7×10^{-15}			
28-29/4	2.4×10^{-13}	2.7×10^{-14}	2.4×10^{-14}	5.4×10^{-14}
29-30/4	2.2×10^{-11}	1.6×10^{-12}	2.1×10^{-12}	8.4×10^{-12}
30/4-1/5	8.3×10^{-9}	2.2×10^{-9}	5.7×10^{-9}	2×10^{-9}
1-2/5	1.1×10^{-9}	1.9×10^{-10}	8.7×10^{-10}	5.5×10^{-10}
2-3/5	2.5×10^{-11}	4.1×10^{-11}	4.5×10^{-11}	1.1×10^{-11}
3-4/5	3.1×10^{-11}	1.1×10^{-11}	1.2×10^{-11}	5.1×10^{-12}
4-5/5	1.6×10^{-11}	9×10^{-12}	8.7×10^{-12}	1.4×10^{-12}

Table 5.5

Sampling date (1986)	Curie/km² x day Kiev				Curie/km² x day Kaliningrad			
	I^{131}	Ba^{140}	Zr^{95}	Cs^{137}	I^{131}	Ba^{140}	Zr^{95}	Cs^{137}
26-27/4					2.0×10^{-2}	3.8×10^{-3}	3.8×10^{-3}	2.3×10^{-3}
27-28/4					4.6	6.3×10^{-2}	2.7×10^{-2}	1.0×10^{-1}
28-29/4	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}
29-30/4	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}
30/4-1/5	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}
1-2/5	10.3	3.2	6.0×10^{-1}	3.2×10^{-1}	1.6×10^{-1}	-	2×10^{-2}	3.2×10^{-3}
2-3/5	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}
3-4/5	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}
4-5/5	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}

Table 5.6

Water body and sampling date	Radionuclides (10^{-9} Curie / m^3)					
	I^{131}	Ba^{140}	Zr^{95}	Sr^{89}	Y^{91}	H^3
Pripyat' river 1/5/86	80	25	18	5.6	1.6	0.6

Table 5.7

Water body	Date of observation of maximum concentration	Concentration, 10^{-9} Curie/litre		
		iodine-131	barium-140	zirconium-95
Pripyat' river	2 May	120	60	42
Kiev reservoir	3 May	28	17	20

Table 5.8

Concentration of all plutonium isotopes (C) in air ($Curie/m^3$) at h = 1.5 m

Sampling date	Distance from source	C, $Curie/m^3$
19/5/86	100 km to SSE	0.48×10^{-14}
20/5/86	72 km to ENE	0.35×10^{-14}
20/5/86	105 km to NNE	0.75×10^{-15}
21/5/86	48 km to NNE	0.65×10^{-14}
21/5/86	60 km to NNE	0.39×10^{-14}
22/5/86	55 km to W	0.21×10^{-14}
22/5/86	45 km to WSW	0.85×10^{-14}
22/5/86	35 km to WSW	0.17×10^{-14}
22/5/86	45 km to SW	0.70×10^{-15}

Table 5.9

Density of surface contamination of surface layer and of grass by all plutonium isotopes

Sampling date	Distance from source	σ_{grass} (Curie / m^2)	$\sigma_{surf.\ layer}$ (Curie / m^2)
20/5/86	105 km to NNE	1.3×10^{-10}	3.6×10^{-10}
21/5/86	48 km to NNE	2.3×10^{-9}	5.1×10^{-9}
22/5/86	55 km to W	1.6×10^{-9}	3.3×10^{-9}

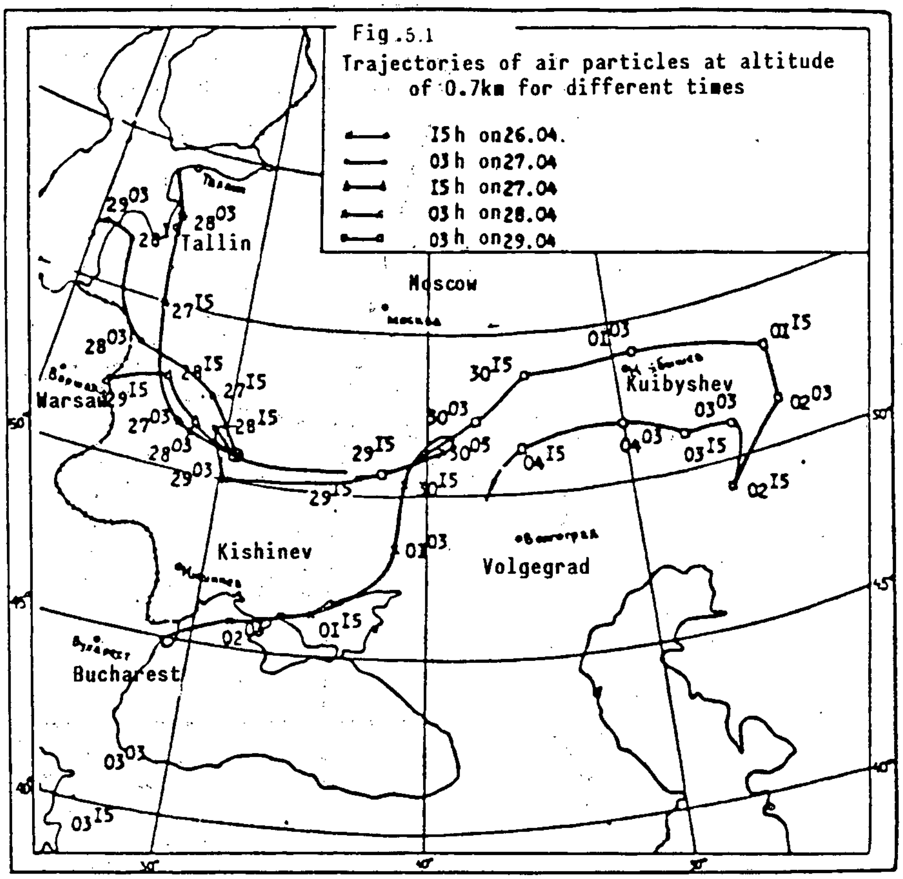


Fig.5.1
Trajectories of air particles at altitude of 0.7km for different times

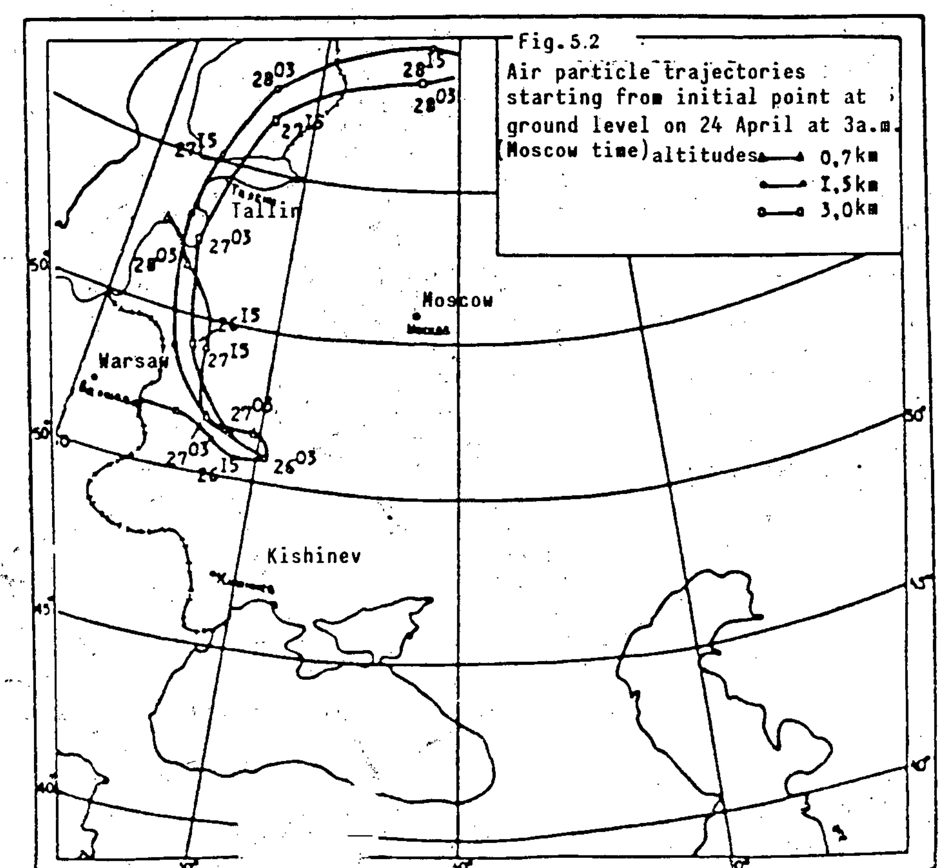


Fig.5.2
Air particle trajectories starting from initial point at ground level on 24 April at 3a.m. (Moscow time)

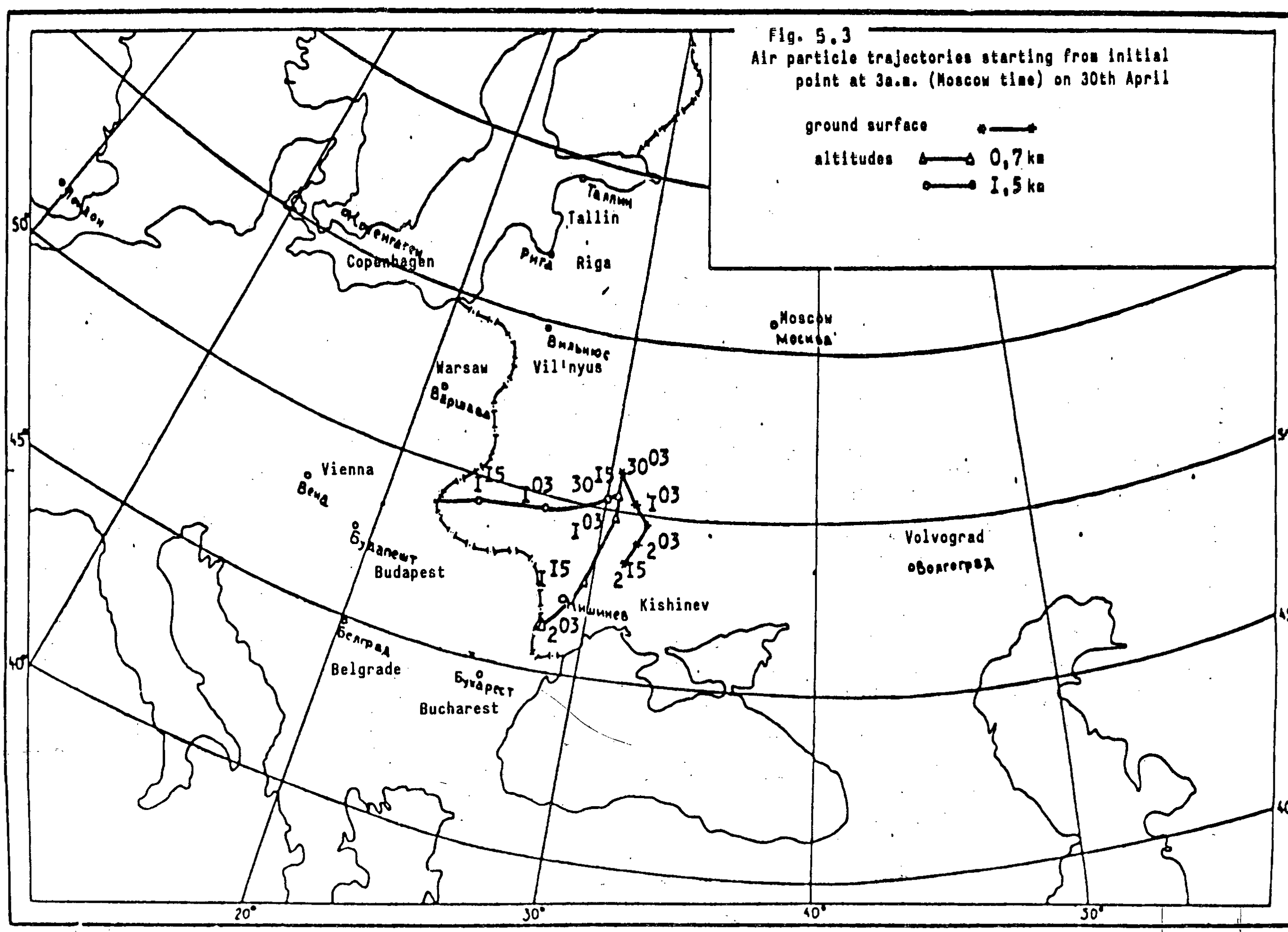
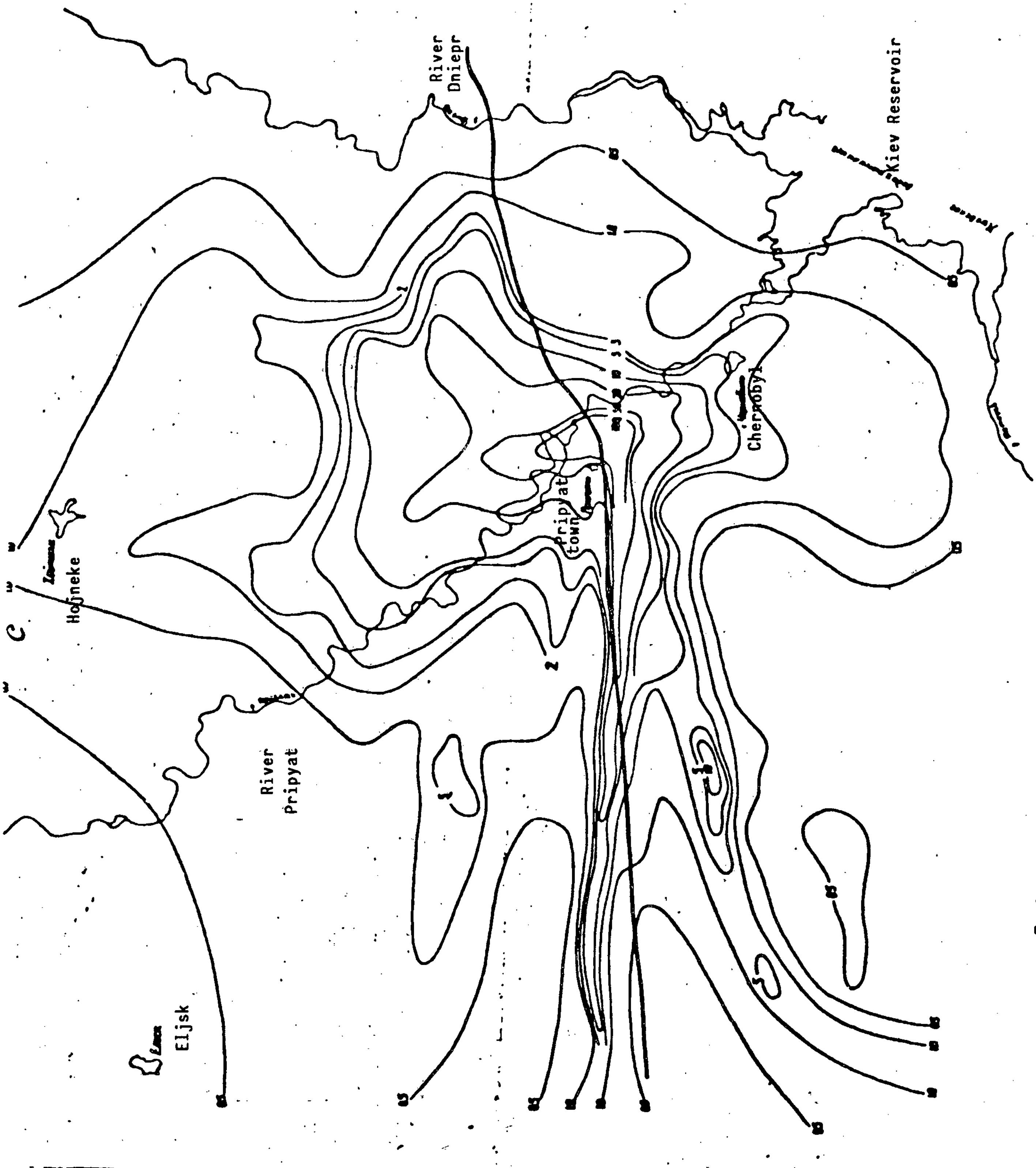


Fig. 5.3
Air particle trajectories starting from initial point at 3a.m. (Moscow time) on 30th April


River Dniepr
Kiev Reservoir
Chernobyl
Pripyat town
Hojneke
River Pripyat
Eljsk

5.4.

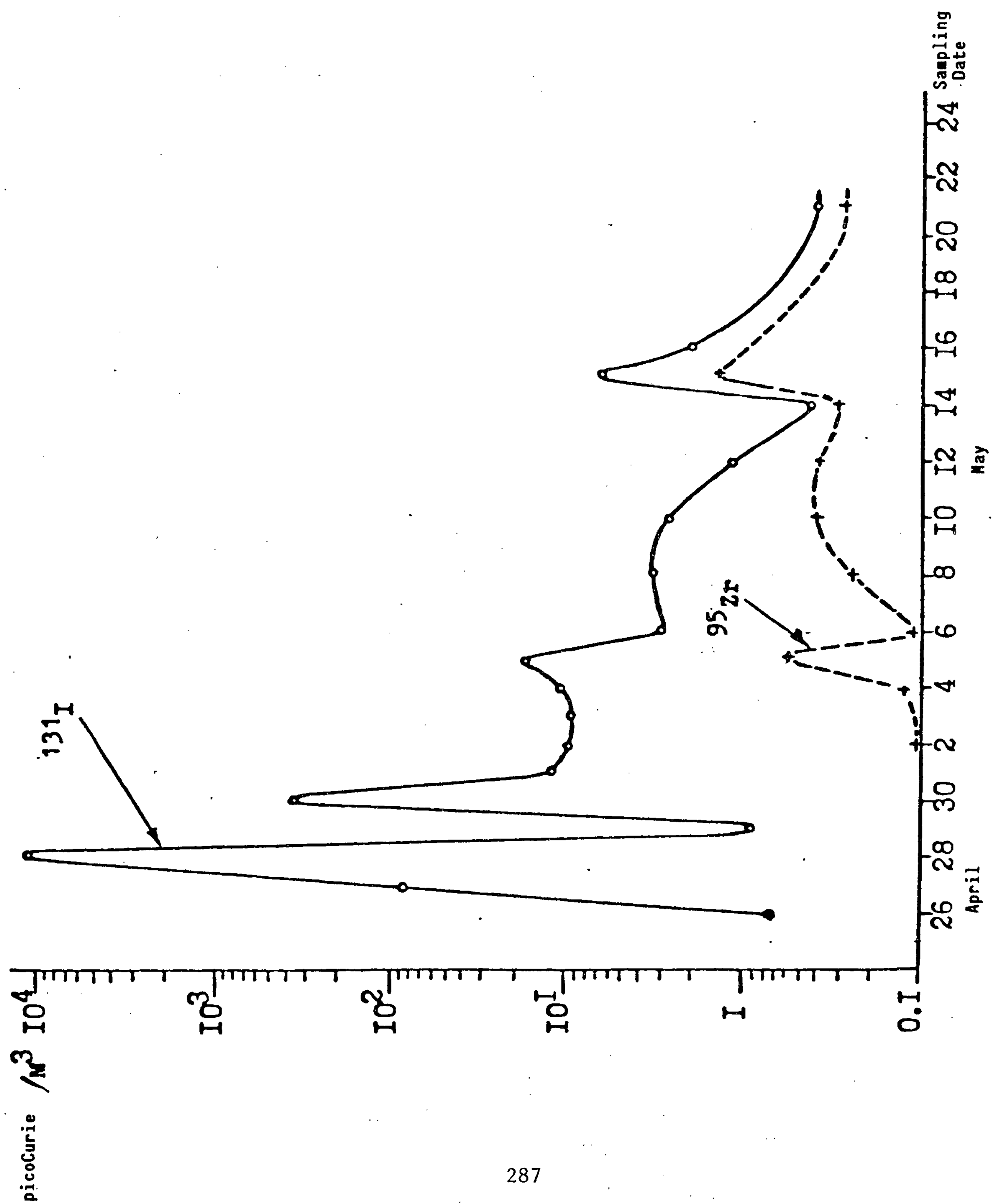


Fig. 5.5. Specific activity (picoCurie/m^3) of ^{131}I and ^{95}Zr in atmospheric air at background monitoring station at Berezinsky National Park

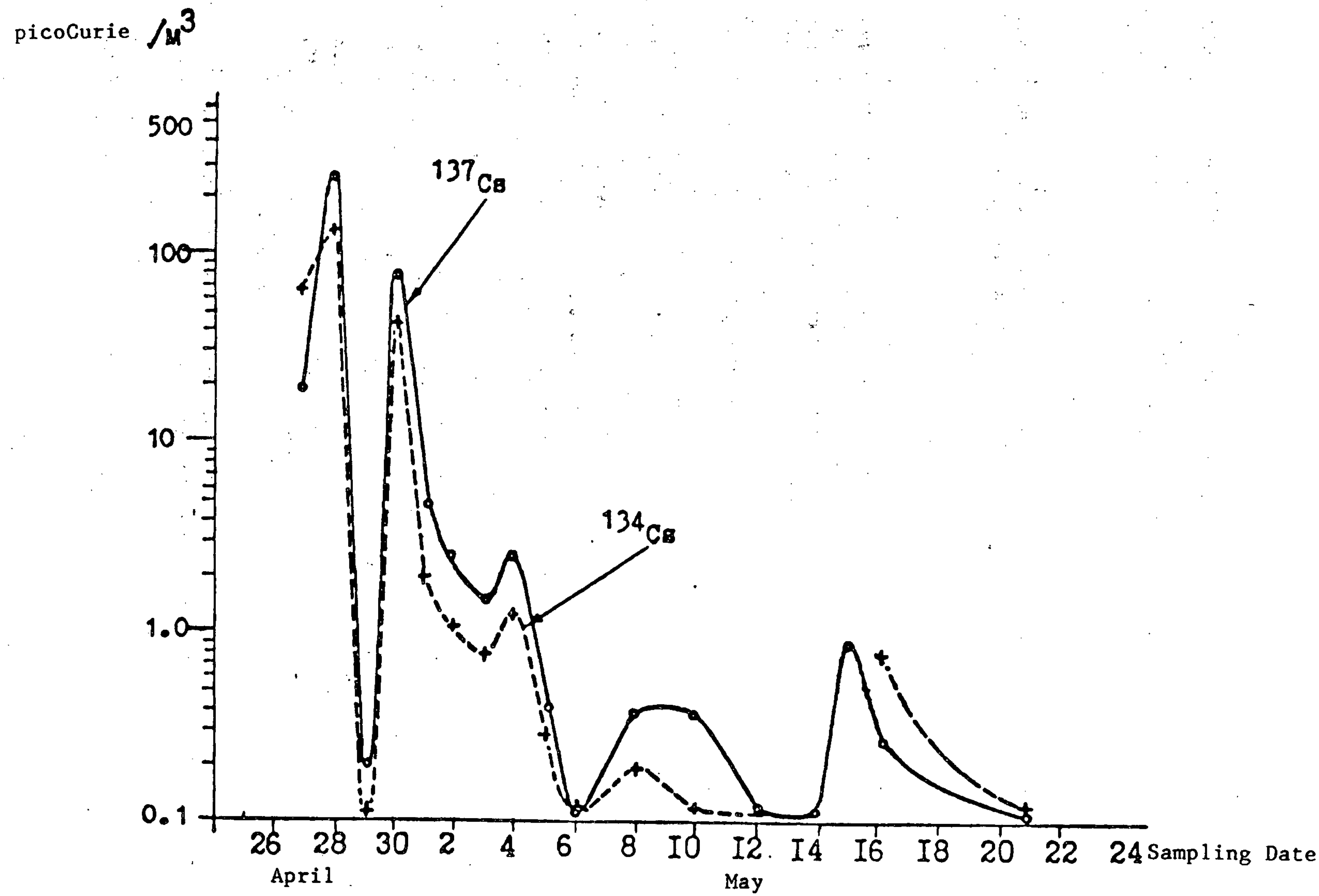


Fig. 5.6. Specific activity (picoCurie/m^3) of ^{137}Cs in atmospheric air at background monitoring station at Berezinsky National Park

ANNEX 6

EXPERT EVALUATION AND PREDICTION OF THE RADIOECOLOGICAL STATE OF THE ENVIRONMENT IN THE AREA OF THE RADIATION PLUME FROM THE CHERNOBYL' NUCLEAR POWER STATION (AQUATIC ECOSYSTEMS)

6. EXPERT EVALUATION AND PREDICTION OF THE RADIOECOLOGICAL STATE OF THE ENVIRONMENT IN THE AREA OF THE RADIATION PLUME FROM THE CHERNOBYL' NUCLEAR POWER STATION (AQUATIC ECOSYSTEM)

For a number of reasons the hydrological environment plays a special role in the determination of the scales and possible consequences of radioactive contamination. As a result of washout, the radioactive substances in the drainage area flow into water reservoirs, where the radionuclides are redistributed and accumulate in components such as bottom sediments, aquatic plants and fish. This leads to additional exposure of both aquatic organisms and man, who is associated with the hydrosphere through the food chain.

From the very first days after the accident arrangements were made for monitoring the content of radionuclides in water and bottom sediments both within and outside the 30 km zone around the plant. Owing to the sedimentation processes, the major part of the radioactivity entering the hydrological environment migrated with comparative rapidity to the bottom sediments, which exhibit a radionuclide concentration higher than the activity of water by two to four orders of magnititude (Table 6.1). According to the experimental data obtained by the Institute of Nuclear Research of the Ukrainian Academy of Sciences, the Institute of Geochemistry and Analytical Chemistry of the USSR Academy of Sciences and the All-Union Research Institute for Nuclear Power Plants, the spatial distribution of radionuclides in the hydrological environment is substantially inhomogeneous. The maximum radionuclide concentration is observed directly in the cooling pond of the power plant, where the total activity of water and bottom sediments attain values of the order of $\sim 10^{-8}$ Ci/L and $\sim 10^{-3}$ Ci/kg, respectively. The concentration of artificial radionuclides in the Kiev reservoir and in the rivers falling into it is substantially lower (by a factor of 10^2–10^4).

In the time dynamics of radionuclides three characteristic stages can be distinguished. In the first stage (up to the end of May 1986) the magnitude of radioactive contamination was determined basically by the short-lived radionuclides, especially ^{131}I. In mid May the ^{131}I concentration in drinking water was n.10^{-9} Ci/L, which was somewhat higher than permissible concentration category B under NRB-76 (Radiation Safety Standards) (by a factor of 2-10 for the river Pripyat'). However, by the beginning of June the ^{131}I content in river water had already fallen by more than an order (< $n.10^{-10}$ Ci/L).

In the second stage, as ^{131}I decays the comparatively short-lived radionuclides such as ^{89}Sr, ^{95}Zr, ^{95}Nb, ^{141}Ce, ^{103}Ru, ^{140}Ba and ^{140}La make an appreciable contribution to the formation of artificial radioactivity. During 10-20 June the concentration of these radionuclides was ~ 10^{-10} Ci/L and 10^{-7}-10^{-8} Ci/kg in the Kiev reservoir water and bottom sediments, respectively, and ~ 10^{-9} Ci/L and 10^{-7}-10^{-5} Ci/kg in the water and bottom sediments of the river Pripyat'.

In the third stage, after the decay of ^{131}I and other relatively short-lived radionuclides, the main role in the formation of artificial radioactivity will be played by the long-lived radionuclides such as ^{137}Cs, ^{134}Cs and ^{90}Sr. During 10-20 June the ^{137}Cs concentration was ~ 10^{-10} Ci/L and ~ 10^{-9} Ci/kg in the water and bottom sediments, respectively, of the river Dnepr. In the cooling pond of the Chernobyl' power station and in the Pryipyat' the ^{137}Cs concentration was appreciably higher (by a factor of 10^4-10^2). The ^{90}Sr concentration in river water varied essentially within 10^{-11}-10^{-10} Ci/L. Considering their long half-life (about 30 years), further decreases in ^{137}Cs and ^{90}SR activity in water bodies will be fairly slow.

On the basis of experimental data on radionuclide distribution in the components of the aquatic ecosystem, calculated exposure dose estimates were made for aquatic organisms with allowance for the geometric characteristics of hydrobionts.

Exposure doses from gamma emitters were calculated with allowance for the buildup factor for scattered radiation. The following dose components were evaluated: external exposure from water, bottom sediments and organisms accumulating radionuclides; internal exposure from incorporated radionuclides. The results of calculations from the data for June 1986 and the predicted absorbed dose rates for June 1987 are given in Fig. 6.1 and Table 6.2.

Analysis of the calculated estimates of the additional exposure doses for hydrobionts indicate the following:

- The highest dose burdens are borne by hydrobionts directly populating the cooling pond. Near the bottom the level of external exposure is 4.3 rad/h on an average; the level of internal exposure for aquatic plants attains 10 rad/h. The highest dose burdens will be borne by the benthic organisms and also the roe and fries of phytophilous species of fish which breed and live in the undergrowths of aquatic vegetation. In the case of these organisms, both the direct and indirect effects of prolonged radioactive exposure may be expected to appear.

- The exposure doses for organisms populating the rivers in the contaminated zone (the river Pripyat' and others) are substantially lower (by a factor of $\sim 10^2$) than for those living in the cooling pond.

- The dose burdens on hydrobionts inhabiting the Dnepr are close in the order of magnititude to the natural background, exceeding the latter in the case of individual hydrobionts by a factor of 5-10.

- Analysis of the contributions of radionuclides to the total exposure dose shows that the principal dose-contributing nuclides at present are the relatively short-lived elements: ^{95}Zr, ^{95}Nb, ^{141}Ce, ^{140}Ba, ^{140}La, ^{89}Sr, ^{103}Ru, etc., whose contribution to the total exposure dose for most components of the aquatic ecosystem exceeds 70-80%. The contribution of the long-lived ceasium isotopes (^{137}Cs and ^{134}Cs) to the total dose does not at present exceed 4-5%. This means that as the short-lived radionuclides decay the exposure dose for aquatic organisms will gradually fall and by the 1987 vegetative season we can already expect a decline in the dose burdens for most hydrobionts by an order of magnitude over virtually the whole contaminated area, including the cooling pond. Thereafter a relative stabilization of the additional dose burden on the aquatic organisms can be expected since by 1987 it will be determined mainly by long-lived radionuclides such as ^{137}Cs and ^{90}Sr.

In speaking about the possible biological effects of ionizing radiation it must be borne in mind that individual groups of living organisms exhibit very wide differences in their resistance to the action of radiation.

Of the hydrobionts the most vulnerable link in the chain is fishes and it is the economic species of these which form the final link in the accumulation of radionuclides in the food chain from the aquatic ecosystem to man. It is known that the exposure dose rate for freshwater fishes under natural conditions varies within 0.007-0.023 mrad/h.

Experimental studies on evaluation of the action of low doses in fish have shown that a dose rate below 0.4 mrad/h (4 rad/year) does not have adverse consequences for the vital activity of fish.

In the range up to 40 mrad/h (365 rad/year) various kinds of disturbances are observed in the functions of organs although on the whole the radioecological resistance is maintained at the population and organism levels. Further rise in the exposure threshold above 140 mrad.h (3.5 rad/day) may produce adverse effects at the population level and cause the disappearance of some of the most radiosensitive species.

As preliminary predictive evaluations show, the exposure doses for most aquatic organisms in the Kiev reservoir (0.1-1.0 mrad/h) do not exceed the range in which populations suffer radiation damage. <u>In the river Pripyat'</u> (buoy 204) the exposure doses for fish are about 50 mrad/h, i.e. there is a likelihood of the adverse action of exposure on the haematopoietic, immune and reproductive systems. Of these, the most significant will be the genetic effects - the advance action on the sex cells.

<u>In the cooling pond of the Chernobyl' power station</u>, there were the highest exposure doses - up to 5 rad/h in a number of locations - for hydrobionts and this will lead to an appreciable radiation effect on the aquatic ecosystem and in particular on the fish community.

The proposed evaluation of the radioecological state of the water bodies is preliminary and should subsequently be refined on the basis of the following data:

- The dynamics of movement of radionuclides from the drainage surface into water bodies, especially during the autumn rains and the spring flood;

- The time dynamics and spatial distribution of radionuclides in hydrobionts, especially in the economic species of fish;

- More accurate information on the species composition, character of migration, feeding spectra and ecophysiological parameters of aquatic organisms;

- Physicochemical forms of the existence of the radionuclides in the components of the aquatic ecosystem;

- Refinement of the basic hydrological parameters of pathways of transfer of radionuclides in the aquatic ecosystems (cooling pond - Pripyat' - Kiev reservoir - Dnepr).

From the scientific and practical standpoints, the following problems are of appreciable radioecological interest:

- Evaluation of the mechanisms of the possible radioecological effects in the case of prolonged (chronic) action of low exposure doses from artificial radionuclides;

- Evaluation of the mechanisms of the secondary ecological effects due to non-uniform action of the radiation factor on the ecological characteristics of species. Non-uniform weakening (or strengthening) of species upon radioactive contamination will lead to changes in the interaction between species and, as a result, to changes in the structure of the ecosystem. As examples of the secondary effects of radioactive contamination one may mention increase in the population of harmful organisms in the active silt and in the contaminated locations of ecosystems near water, changes in the self-cleaning capacities of water bodies and so on.

It will be necessary to organize and carry out long-term comprehensive radioecological studies both within and outside the 30 km zone in order to resolve the above-mentioned problems.

Table 6.I

Specific activity (Ci/kg) of bottom sediment samples in June 1986 (10 - 20 June 1986) (according to the data of the Institute of Nuclear Research, Ukranian Academy of Sciences), Institute of Geochemistry and Analytical Chemistry (USSR Academy of Sciences) and All-Union Research Institute for Nuclear Power Plants)

Sampling location	Zr-95	Nb-95	Ce-I4I	Ce-I44	Ba-I40	La-I40	Ru-I03	Cs-I34	Cs-I37	I-I3I
Chernobyl' Power Station Cooling Pond	$I0^{-3}$	$2 \cdot I0^{-3}$	$3 \cdot I0^{-4}$	$3 \cdot I0^{-4}$	$I0^{-4}$	$2 \cdot I0^{-4}$	$6 \cdot I0^{-4}$	$4 \cdot I0^{-5}$	$7 \cdot I0^{-5}$	$I,0 \cdot I0^{-5}$
River Pripyat'	$8 \cdot I0^{-6}$	$I,I \cdot I0^{-5}$	$2,7 \cdot I0^{-6}$	$4 \cdot I0^{-6}$	$5,4 \cdot I0^{-7}$	$I,3 \cdot I0^{-6}$	$I,6 \cdot I0^{-6}$	$2,7 \cdot I0^{-7}$	$4,7 \cdot I0^{-7}$	$9,4 \cdot I0^{-8}$
River Dnepr	$5,3 \cdot I0^{-8}$	$7,I \cdot I0^{-8}$	$2,8 \cdot I0^{-8}$	$4,I \cdot I0^{-8}$	$2,6 \cdot I0^{-8}$	$4,3 \cdot I0^{-8}$	$3,6 \cdot I0^{-8}$	$2,4 \cdot I0^{-9}$	$5,I \cdot I0^{-9}$	$6,3 \cdot I0^{-9}$

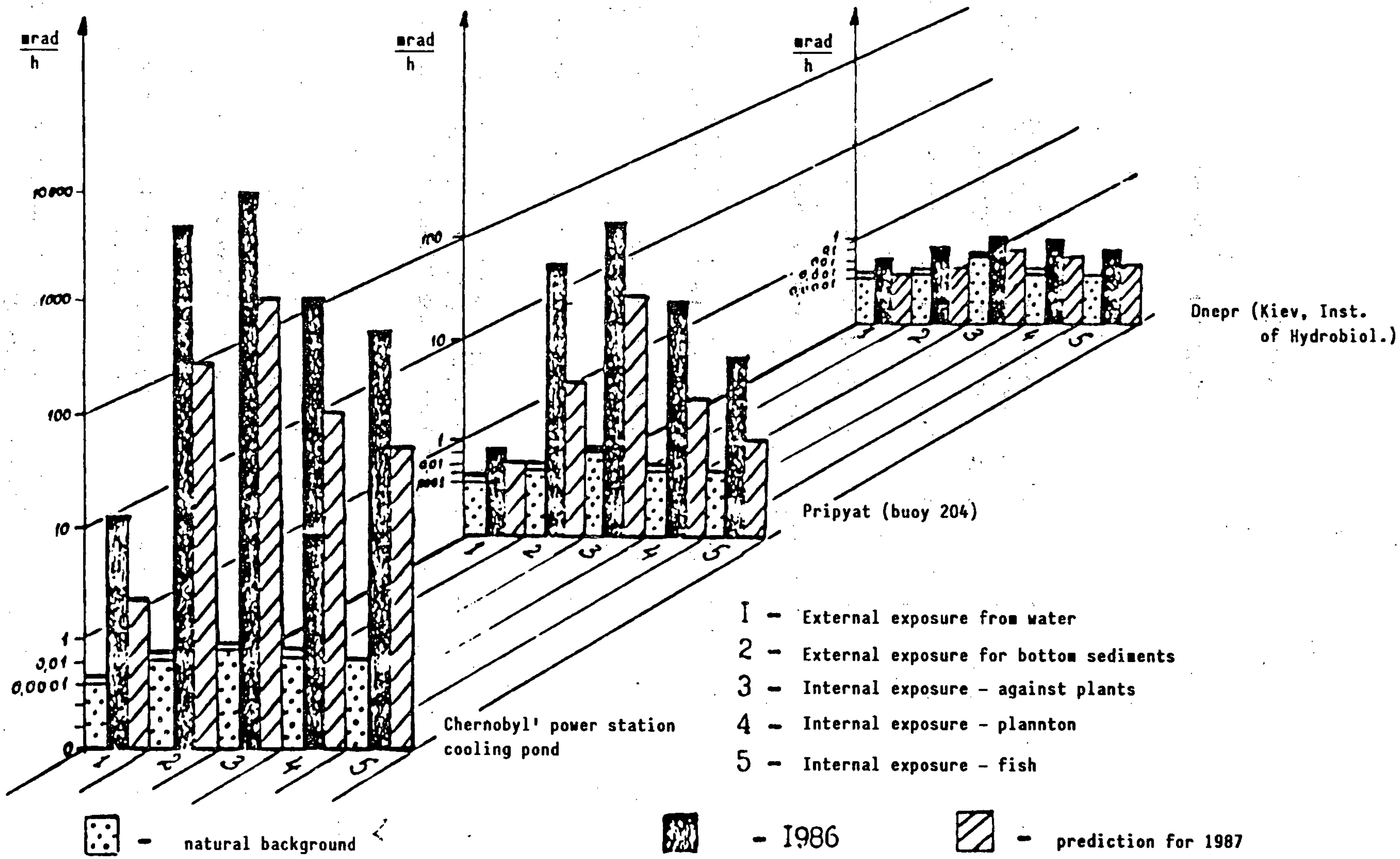
Table 6.2

Calculated estimates and predictions of exposure doses for aquatic organisms (mrad/h) in the area of the radiation plume from the Chernobyl' nuclear power station (10-20 June 1986)

Water body	External exposure		Internal exposure		
	from water	from bottom sediments	aquatic plants	plakton	fish
Chernobyl' Power Station cooling pond	I0 (2) ✱	4300 (300)	I0000 (I000)	I000 (I00)	500 (5C)
River Pripyat' (buoy 204)	0,I (0,009)	40 (3,3)	II0 (I5)	I2 (2)	6 (0,8)
River Dnepr (Kiev, Inst. of Hydrobiology, Ukr. Academy of Sciences)	0,002 (0,0002)	0,3 (0,025)	I,0 (0,2)	0,I (0,0I5)	0,04 (0,0I)
Natural background	0,000I-0,006	0,002-0,02	0,08-0,2	0,002-0,0I6	0,003-0,005

✱ The figures in brackets are predictions for June 1987.

Fig. 6.I. Calculated estimates and predictions of exposure doses for aquatic organisms in the region of the radiation plume from the Chernobyl' nuclear power station (10-20 June 1986)



Declassified and Approved For Release 2013/07/15 : CIA-RDP09-00997R000100260001-0

ANNEX 7

MEDICAL-BIOLOGICAL PROBLEMS

Contents

ANNEX 7

7.1. DATA ON OPERATIONAL AND EMERGENCY STAFF OF THE NUCLEAR POWER PLANT WHO WERE EXPOSED TO RADIATION: SIZE OF THE DOSES RECEIVED AND THE CONSEQUENCES FOR HEALTH. EXPERIENCE OF TREATMENT

7.1.1. First information about the accident and action taken by the medical staff

The medical and health section serving the plant received information at their station about the accident 15 minutes after it happened (2 a.m., 26 April 1986).

Assistance was given to the first 29 victims leaving the site of the accident by themselves within the first 30-40 minutes by the middle-level medical staff on duty at the health station. The victims threw off their contaminated clothing and shoes even before entering the personnel airlock. Owing to the intense primary reaction, they were immediately sent to the hospital where health procedures and the first medical examinations were carried out.

During the next four hours, first aid teams which had immediately arrived on the scene provided assistance to the victims, removing them from the zone of production operations, carrying out preliminary health procedures in the personnel airlock and transporting persons suffering from primary reactions (nausea, vomiting) to the hospital. Persons who felt that their condition was satisfactory were sent home and, subsequently, actively summoned for examination in the morning of 26 April 1986. By 6:00 on 26 April, 108 persons had been hospitalized and, in the course of the day, another 24 persons from the group which had been examined were hospitalized.

One victim died at 6 a.m. on 26 April as a result of severe burns and one member of the staff who had been working at the time was not discovered. It is possible that his place of work was in the high-activity pile-up area.

Twelve hours from the time of the accident a team of specialists arrived and started to work. The group included physicists, radiology therapists, laboratory assistants and haematologists. Within 36 hours from the time of the accident, some 350 persons had been examined at the medical and health centre on a stationary or out-patient basis and approximately 1000 blood analyses had been performed (no fewer than 1 to 3 for each patient during the first 36-48 hours). The outpatient discharge cards contained entries relating to post-accident clinical indications, complaints, leukocyte number and leukocyte formula.

On the basis of the criteria adopted in the USSR for early diagnosis: times and intensity of the primary general and local (skin) reaction, intensity of leukopenia and neutrophilic leukocytosis towards the end of the first 36 hours were taken as a basis for the timely hospitalization of persons in whom the development of acute radiation sickness was diagnosed as a very high probability. Clinical institutions very close to Kiev and a specialized infirmary in Moscow were selected as the places for hospitalization in order to provide maximum assistance and competent follow-up analysis of the results of examinations.

During the first two days, 129 patients were sent to the specialized infirmary in Moscow. Of these, during the first three days, 84 were classified as having acute radiation sickness (second to fourth degree of severity) and 27 (with first degree severity), which is an indication that the primary classification was adequate.

The diagnosis of acute radiation sickness (second to fourth degree) was made in the first three days; specification of the first degree generally required a longer observation period (up to 1-1.5 months).

In all, 203 persons were recognized as having acute radiation sickness. Persons suffering fromn this condition were not found among the population at large.

7.1.2. Principles of diagnosis and prognosis in the specialized infirmary

The main diagnostic and prognostic criteria, of importance for deciding on the conditions for managing the patients and selecting the methods of treatment, including indication for bone marrow transplantation, decontamination etc., were determined during the first three days of the patients' stay in the infirmary.

The criteria for classifying the patients in the first few days were of clinical and clinico-laboratory nature, based on Soviet experience and on the recommendations of international radiology centres.

In the period from the first few hours to three days, the conclusive factors were the time and severity of the initial general (vomiting) and local (hyperaemia as well as cutaneous oedema and myxoedema) reactions. The intensity of lymphopenia was estimated quantitatively for the days of observation and on this basis a rough estimate was made of the mean dose of total uniform irradiation. The possible dose of bone marrow irradation was determined by the direct method, counting the aberrations in the bone marrow cells.

Supplementing this, during the first 10-14 days, severity criteria were applied for determining the extent of thrombocytopenia and the times at which it appeared, and also the severity of leukopenia and granulcytopenia and the times at which they developed. A quantitative estimate of the dose to bone marrow was made on the basis of the number of dicentrics in cultures of peripheral blood lymphocytes stimulated by PHA.

The dynamics of skin changes from the first few days to two weeks were estimated semi-quantitatively in accordance with accepted clinical parameters.

All these criteria developed by Soviet scientists afforded a means of evaluating the prognoses as to:

- overall clinical course of the pathology;

- the dynamics of the blood picture;

- the possible extent of injury to individual parts of skin and mucous membrane.

To the known extent, it was also possible to make a rough estimate of the mean dose of uniform gamma irradiation of the bone marrow or its equivalent according to individual biological parameters.

The course of the sickness and its possible outcome, specified in the initial stages in accordance with the above-mentioned prognostic criteria, showed satisfactory agreement with this prognosis in its subsequent manifestations.

With respect to the severity of the bone marrow and intestinal syndrome of acute radiation sickness, four degrees were distinguished, on the basis of criteria adopted in the USSR.

The classification extremely severe (fourth degree) was applied to cases of the sickness with a short latent period (up to 6-8 d), with a pronounced, early (in the first half hour) primary reaction (vomiting, headache, rise in body temperature). The number of lymphocytes in the first 3-6 days was less than 100/μL. As from the 7th to the 9th day, pronounced symptoms of enteritis. The number of granolocytes at on the 7th to the 9th day was 500/μL; thrombocytes ≤ 40 000/μL from the 8th to the 10th day. Intense general intoxication, fever, lesions of the oral cavity and salivary glands. The condition of 20 of the patients under treatment at the specialized infirmary were assignable to this type of pathology.

A total uniform irradiation dose of more than 6 Gy (up to 12-16 Gy), equivalent to the biological effect in haemopoiesis, was found in 18 patients.

Lethal outcomes in the period of + 10 days to + 50 days occurred in the case of 17 patients. In all these patients, the burns extended to 40-90% of the body surface and in most of them they were extremely severe, virtually fatal, even without taking into account the other clinical syndromes of radiation sickness. In two of the patients of this group, the cont[illegible] of radionuclides was also the highest (see Section 7.1.8). Two other patients with conditions of fourth degree severity died on days + 4 and + 10 at the Kiev hospital as a result of combined thermoradiation injury.

A total of 23 persons was classified as third degree acute radiation sickness patients. The total gamma radiation dose was approximately 4.2-6.3 Gy. The criteria for determining acute radiation sickness of this degree of severity were times of 30 min to 1 h for the development of a pronounced reaction (vomiting, headache, subfebrile body temperature, transient hyperaermia of the skin. Lymphopenia in 3-6 d, 200-100 cells/μL. Duration of latent period: 8-17 d. A characteristic feature is the presence of an epilation effect. Decrease in the number of thrombocytes to ≤ 40 000/μL in 10-16 d, of neutrophils to ≤ 1000/μL in 8-20 d. The climax of the sickness is characterized by pronounced fever, infectiuns complications, hemorrhagic bleeding. This degree of severity was identified on 21 persons at the specialized infirmary and in two at the Kiev hopsital. Seven persons died in periods of from two to seven weeks. Six persons of this group were suffering from severe skin injuries which greatly aggravated their condition and to a large extend predetermined the lethal outcome.

The criteria for the diagnosis of acute radiation sickness of second degree severity was the development of a primary reaction in 1-2 h: lymphopenia within a period up to 3-6 d of the order of 500-300 cells per μL. The length of the latent period was 15-25 days. The decrease in neutrophils in 20-30 d to 1000 cells per μL, thrombocytes in 17-24 d to 40 000 per μL. In the period of climax: genuine infectious complications and slightly pronounced symptoms of hemorrhaging. Moderate acceleration of the E.S.R. up to 25-40 mm/h. In the specialized infirmary and in the Kiev hospital, injuries of this degree of severity were found in 53 persons (level of those equivalent in biological effect to 2-4 Gy). There were practically no persons with burns aggravating their condition.

According to karyological data, the dose level in acute radiation sickness patients of first-degree severity ranged from 0.8 to 2.1 Gy. There were no patients with skin lesions substantially aggravating the clinical picture of their condition. The diagnostic criteria for acute radiation sickness of the first degree were: the presence of a primary general reaction in periods 2 h after the moment of exposure, the absence of a general skin reaction, a latent period longer than 30 d, a decrease in the number of lymphocytes in the first few days to 600-1000 cells per μL, leukocytes in the 8th-9th days to 4000-3000 per μL, and in the climax to 3500-1500, thrombocytes to 60 000-40 000 per μL (25-28 days), moderate acceleration of E.S.R. These criteria are used for estimating the degree of severity of the bone marrow syndrome. A very important factor for this group of patients were the data from systematic clinical and laboratory observations over a period of one and-a-half months (with allowance for the latent period and the availability of data on the frequency of chromosome aberrations in the lymphocytes of the blood and bone marrow).

7.1.3. The scale of biophysical studies and evaluation of the main damage-inducing factors and dose levels

All persons entering the reception room were subjected to dosimetric monitoring by means of the various devices used in the USSR for recording external body radiation (RUP, SRP-68-01, AKTINIYA, TISS and others). This made it possible to estimate the dose rate distribution for the body (region of the thyroid gland, chest, back, wrist, hands, feet, legs etc.) and to obtain indications for repeat treatment and decontamination of integuments.

The dose rate measured by a device was dependent on the incorporation of radionuclides and, in part, on the residual contamination of the integuments of the persons concerned. The use of masks and washing by means of moistened tampons and aluminium metal filters (screens) in the measurements on apparatus permitted a rough estimate to be made of the contribution to the radiation from the body of the radionuclides incorporated in it and applied to the skin. The first determinations, as well as the examination and interrogation of the patients in the reception room, confirmed that most of them, in addition to being exposed to external gamma and beta radiation, had immediate contact with beta- and gamma-active nuclides, and in some cases these nuclides entered the organism.

Although in most cases there was a combination of two or three of the above-mentioned factors, the leading ones exhibited by the patients were external beta, gamma whole-body radiation and, in addition, greater irradiation of the integuments by relatively weakly penetrating radiation. Curves plotted for the decay of radioactive substances contained in urine samples of the patients from as early as two days after the accident, confirmed the presence in the victims' bodies of radionuclides with half-lives of 185–190 hours or approximately 8 days, and mainly isotopes of iodine.

The patients were carefully washed again and accommodated in wards, the potassium iodide treatment which was already started in the first four days was continued (0.25, twice a day). Treatment of the burns was started and continued; the same applied to the oropharyngeal syndrome which was also observed in this period in a considerable number of the patients.

Special diagnostic procedures, of both the general clinical and the biophysical types, were developed for determining possible dose level and nature of irradiation more specifically.

A combination of methods and various types of equipment were available for implementing the biophysical studies.

1. A scintillation detection unit (64 x 64 mm) housed in a lead collimator was used for measuring the content of ^{131}I in the thyroid gland. The radiation was collimated in such a way as to cut off the photon radiation from the human body, and to separate out only the radiation coming from the thyroid gland. The measurements were carried out in a narrow window (approximately 364 keV-^{131}I peak. The estimate of the addition on account of ^{131}I incorporated in the blood flowing through the region of the neck, was made by measuring the content of ^{131}I in the patient's forearm. Calibration was performed by means of point source of ^{131}I located in a phantom of the human neck.

2. The content of radioactive substances in urine samples was measured by devices for biosubstrate analysis: B10-1 and SICH (whole-body counter) 2.1. The former has a large-volume scintillation-type detection unit; the latter uses a semiconductor

detection unit based on a drift-type semiconductor unit. The same devices were used for measuring the gamma radiation from samples of dissected material. They were calibrated by means of certificate controlled gamma radiation sources in a geometry approximating the real one as closely as possible.

3. A SICH 2.2 device and a semiconductor unit with local shielding were used for measuring the activity incorporated in the human body. The former instrument uses a large-volume scintillation-type detector unit, the latter a semiconductor detection unit based on pure germanium. The devices were calibrated by means of phantoms of the human body, prepared from standard receptacles filled with calibration solutions of various radioactive substances.

4. Multichannel amplitude analysers (memory capacity up to 8000 channels) were used for collecting and analysing the gamma spectra obtained; the spectrometric channel of the devices was built from high-resolution spectrometric apparatus produced by the "Nokiya" company (Finland) and RT (USA).

The spectrometric information obtained from all the above-mentioned devices were recorded on magnetic tape.

To determine the total content of gamma activity of transplutonium elements in the excreta who were victims of the Chernobyl' accident, a study was made of urine samples of ten of the patients under examination who had been exposed under various conditions. In no case was plutonium found in the urine (sensitivity of the method: 0.2 disintegrations/min from a 500 mL sample).

Three patients whose levels of alpha-active radionuclides in the urine when they entered the infirmary (on 28 April 1986) were, respectively, 2.0, 0.67 and 0.1 nCi per 1 mL of urine, were given a diagnostic test involving the use of pentacine to accelerate the elimination of plutonium from the organism. In no case was any effect observed from administration of the preparation three times.

Examination of the organs of a patient who died (content of beta- and gamma-active radionuclides on entering the clinic on 28 April 1986 was 1.5 nCi/mL urine) showed, just in the lungs, a total alpha activity of transplutonium elements in the amount of $\leq$ 3.4 nCi/organ and trace amounts in the urinary bladder. No analysis was made of bone tissue.

Alpha spectrometry of lung samples showed approximately 90% ^{242}Cm and around 10% plutonium and americium. With a plutonium content at this level (in combination with transplutonium elements) and the low constant of its elimination from the organism, the level of clearance of the nuclide from the urine is below the sensitivity of the method of determination applied.

Immediately after hospitalization, simultaneously with the primary dosimetric monitoring by means of gamma radiometers for the purpose of estimating the level of radioactive contaminations, blood and urine samples were taken for biophysical examinations (measurements of total activity, gamma-spectrometric measurements). The studies of total activity were conducted in a biophysical laboratory and involved the use of precise radiometric equipment; the determination of the isotopic composition of the samples studied was carried out with a gamma spectrometer based on a semiconductor detector with pure germanium.

One to two days after the victims entered the infirmary, they were examined for content of radioiodine in the thyroid gland; this was determined by means of a gamma radiometer of the Gamma Co. (Hungarian People's Republic). In the days that followed, these examinations were repeated a few (4 to 6) times to obtain information concerning the half-clearance of radioiodine from the thyroid gland. The consolidated results of the measurements of radioiodine content in the period from 29 April to 6 May 1986 (day +3 to day +10) show that in the majority of victims (94%) the content of radioiodine in the thyroid gland on 29 April 1986 was less than 50 μCi; in 6% of those examined, these levels were 2-4 times higher (Figs 7.1.1 and 7.1.2).

A few days after entry into the infirmary, when the levels of residual surface contamination were close to the background values, a majority of the victims (except for persons in an extremely serious condition) were examined in stationary whole-body counters. The gamma-ray spectra from their bodies showed peaks of more than 20 different radionuclides, but the main ones determining the internal irradiation dose of the victims were ^{131}I, ^{132}I, ^{134}Cs, ^{137}Cs, ^{95}Nb, ^{144}Ce, ^{103}Ru and ^{106}Ru. The characteristic radiation spectra are shown in Figs 7.1.3 and 7.1.4.

At the autopsies of all the persons who died, specimens were taken of various organs and tissues for subsequent determination of their content of radionuclides (up to 35 samples from one person, including 17 samples from different parts of the lung, Fig. 7.1.5). Tentative results were obtained from the determination of individual radionuclides in five victims who died of acute radiation sickness in periods of 17 to 19 days. A standard sample analysis chart in relative units per 1 g of tissue is shown in Fig. 7.1.4.

The spectra of the radiation emitted by the human body varied in the individual observations.

The analysis of the data and evaluation of the individual irradiation doses of the organs and tissues is being continued. Only in one case, with a maximum content of radioactive substances in the organisms, was there reason to think of a contribution of internal radiation in the early clinical manifestations of injury to the respiratory organs and the digestive tract.

Results of the examination of biosubstrates for sodium-24

The first samples of biosubstrates (urine and blood) were received on 27 April 1986 at 3 p.m. They were subjected to spectrometric study for photon radiation emitted. The photon spectrum obtained was of a very complex

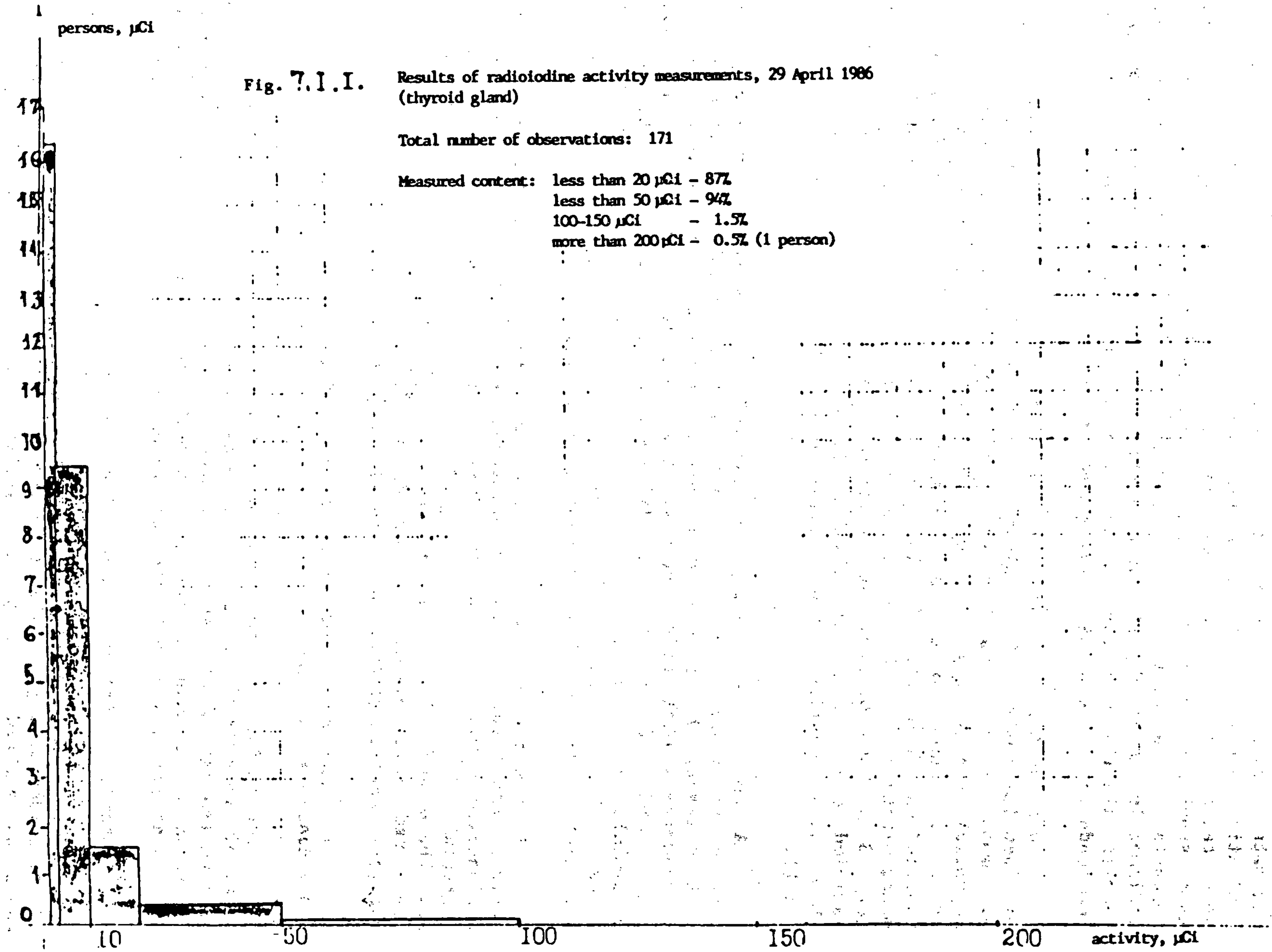


Fig. 7.1.1. Results of radioiodine activity measurements, 29 April 1986 (thyroid gland)

Total number of observations: 171

Measured content: less than 20 μCi - 87%
less than 50 μCi - 94%
100-150 μCi - 1.5%
more than 200 μCi - 0.5% (1 person)

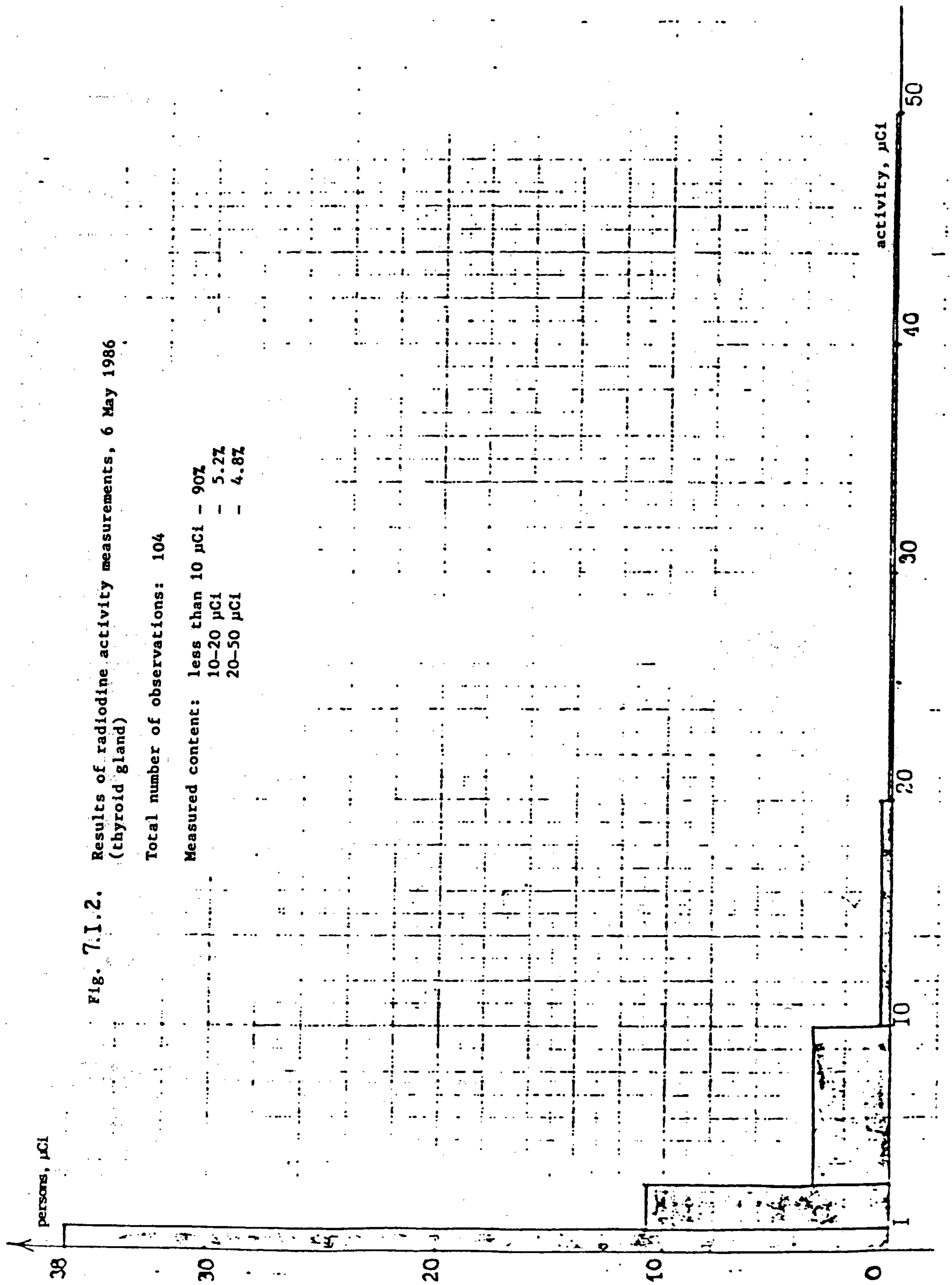


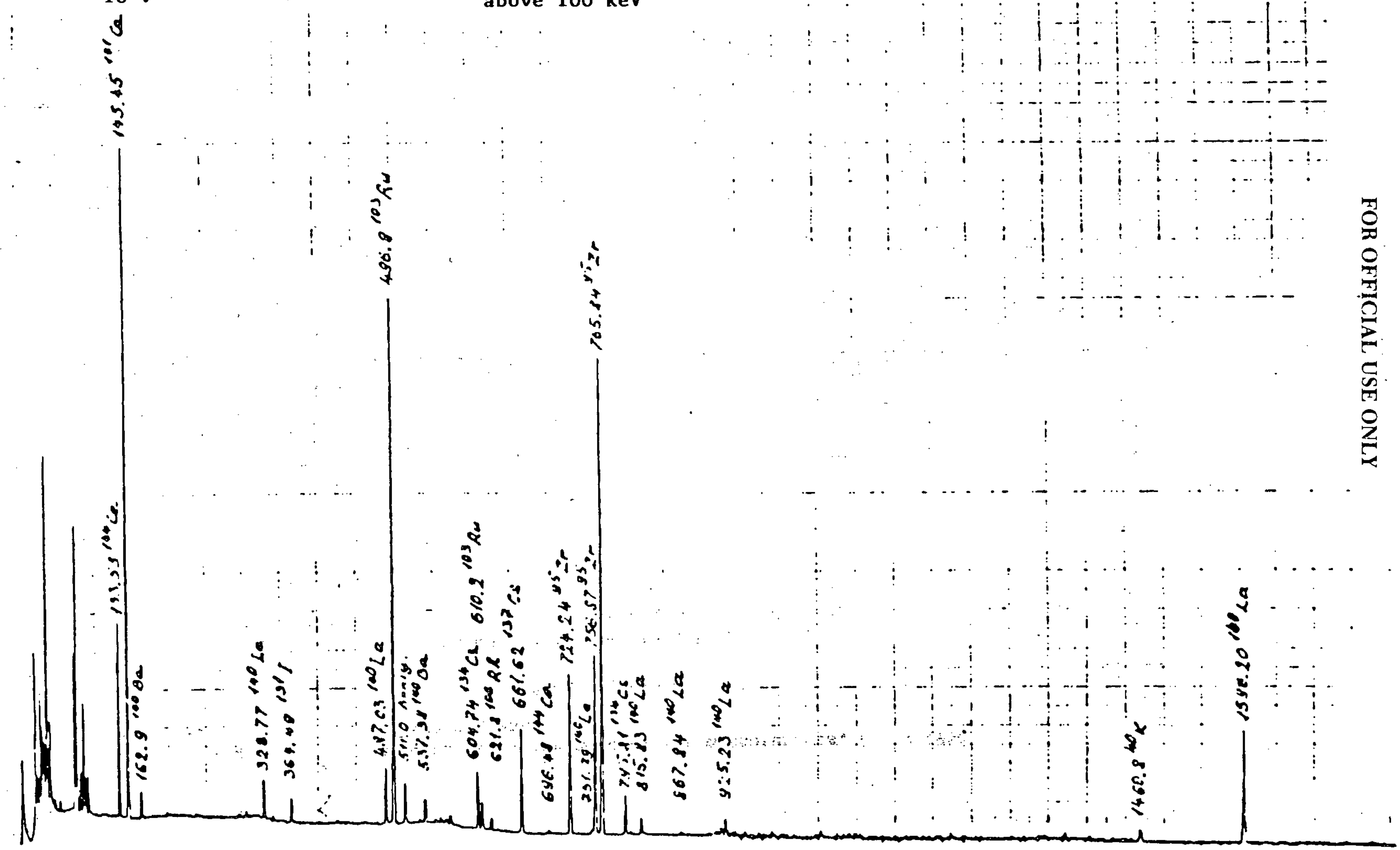
Fig. 7.1.2. Results of radiodine activity measurements, 6 May 1986 (thyroid gland)

Total number of observations: 104

Measured content: less than 10 μCi – 90%
10–20 μCi – 5.2%
20–50 μCi – 4.8%

Spectrum magnified two times with respect to Y

Fig. 7.1.3 Spectrum of photon radiation of an incorporated mixture of radionuclides. Semiconductor detection unit based on pure germanium, with a sensitive volume of 60 m^3. Energy range above 100 keV



Spectrum magnified two times with respect to Y

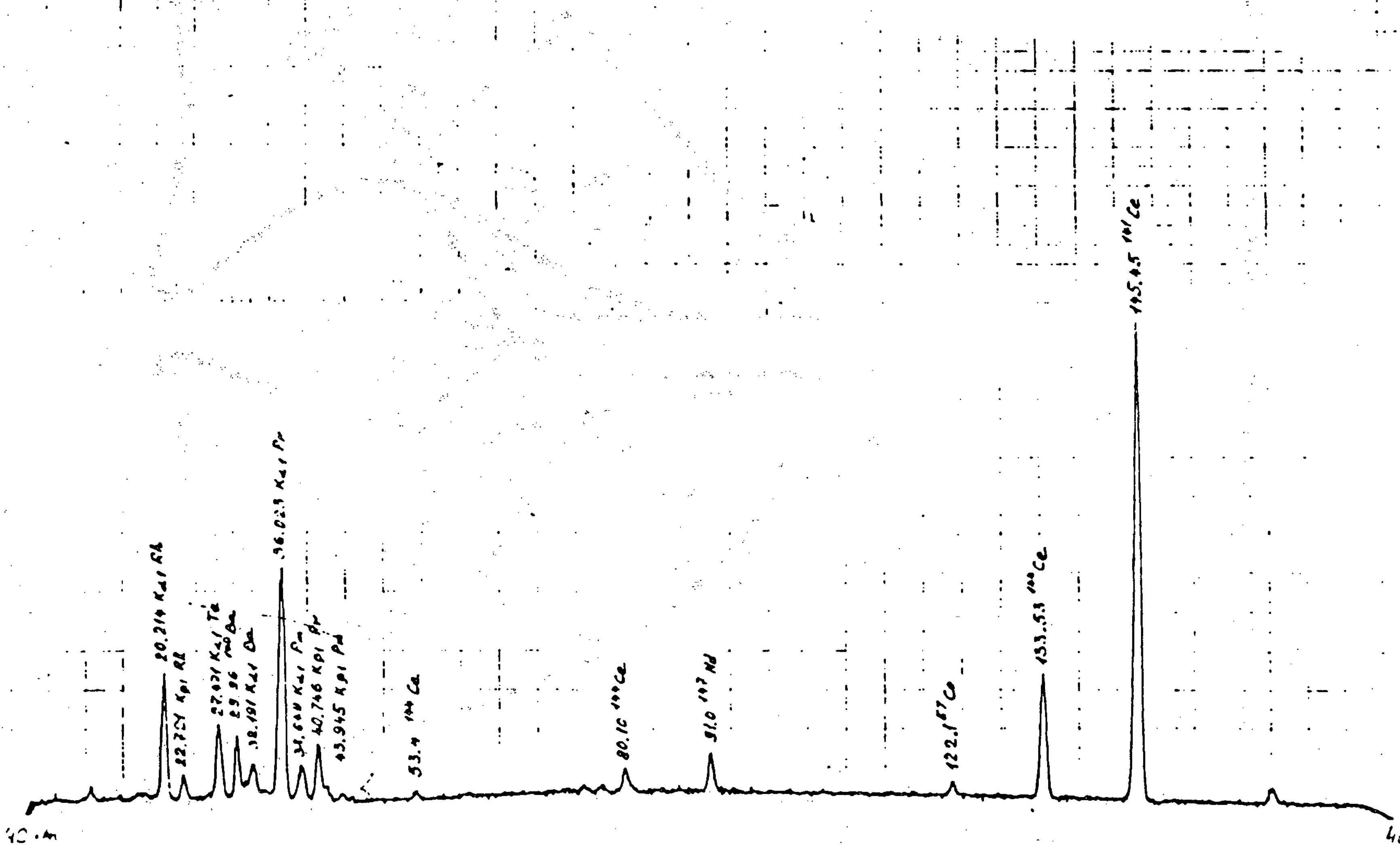


Fig. 7.1.4. Spectrum of photon radiation of an incorporated mixture of radionuclides. Semiconductor detection unit based on pure germanium, with a sensitive area of 260 cm^2. Energy range below 100 keV

Fig. 7.1.5. Scheme for selection of samples of dissected lung material

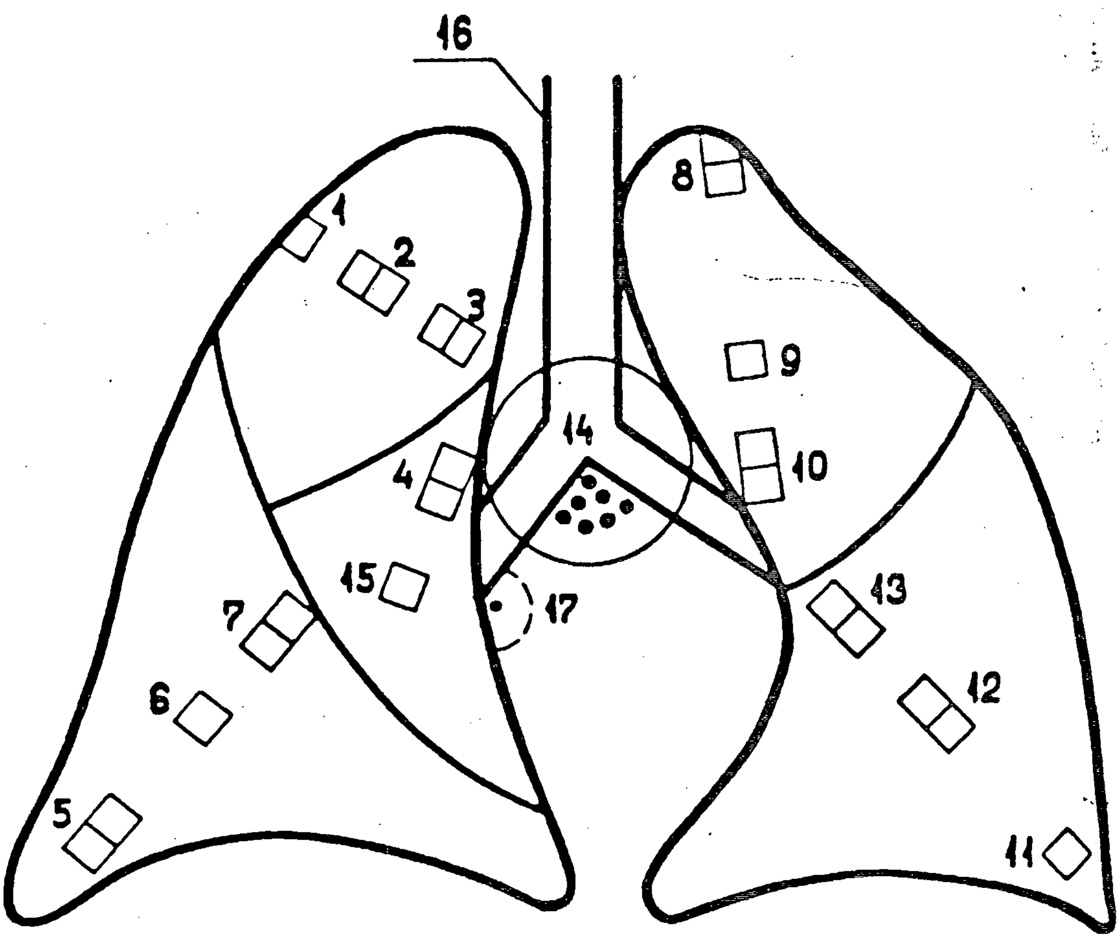
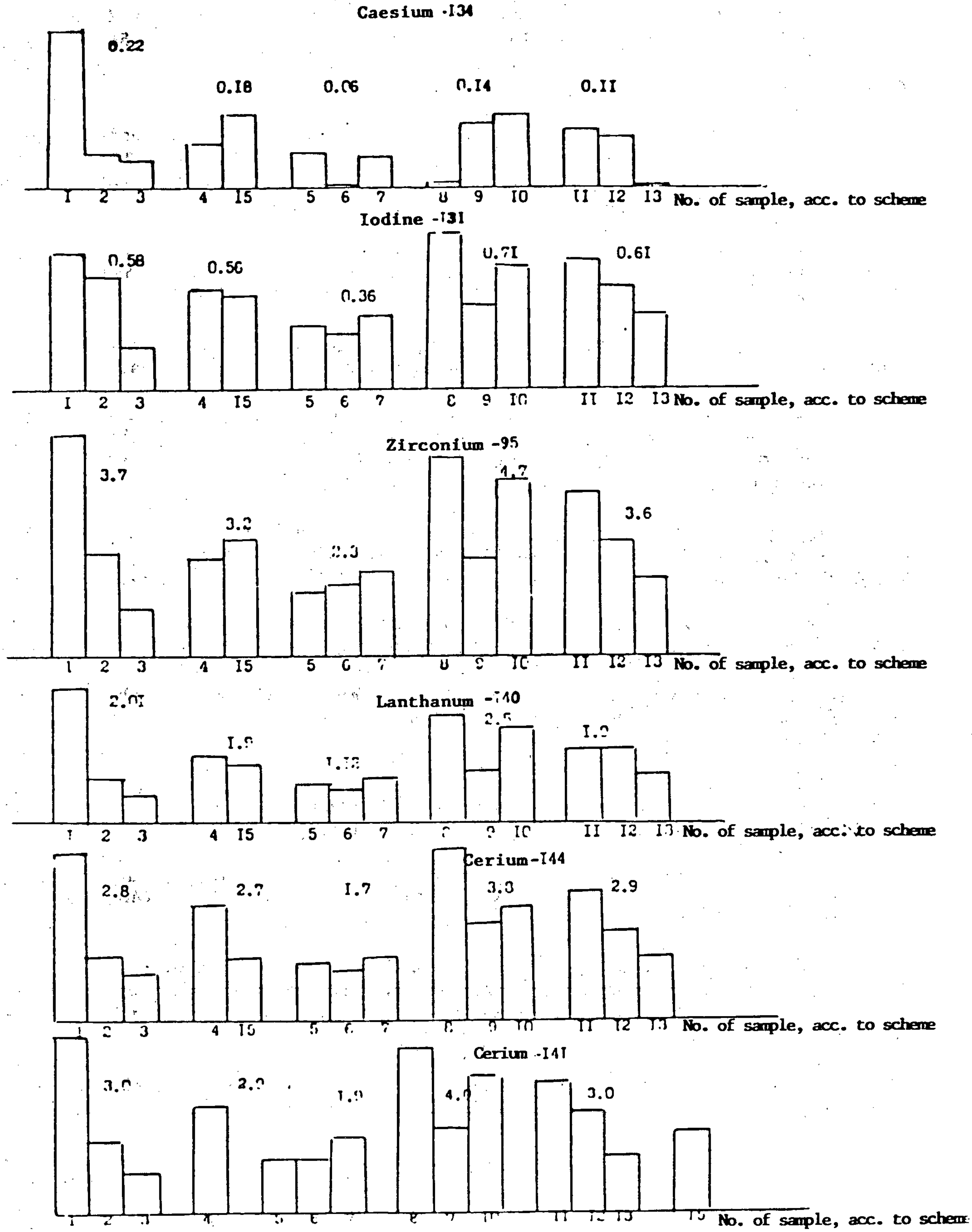


Fig. 7.1.6. Distribution of radionuclides over the lungs of a victim (see scheme for selection of lung samples) (in relative units); the indicated values are mean values



nature. However, it was not found to contain a 1274 keV line (sodium-22, half-life 2.64 y), having a yield of 99.95%, nor 1368 keV and 2754 keV lines (sodium-24, half-life 15 h), having quantum yields of 99.87% and 99.99%, respectively.

The measurements were made with a semiconductor detection unit having a sensitive volume of 60 cm^3. The estimated ^{131}I activity in these samples was 0.5 µCi/mL, and the activity of the caesium isotopes (caesium-137 and caesium-134) was 0.1 µCi/mL.

The 511.0 keV line present in the spectrum is of insignificant cross-section and corresponds to annihilation radiation due to one 1597 keV line (lanthanum-140).

Thus, approximately 35 hours after exposure, no appreciable data had been obtained giving evidence of neutron irradiation of the victims.

Estimates of the total activity of iodine-131 and of the isotopes caesium-134 and caesium-137 which had entered the organisms of the victims (two victims who had the highest content of radionuclides in their organisms)

On the basis of the first results of the analysis of urine and blood samples, examples were taken which provided evidence that the persons in question were suffering from very extensive internal radioactive contamination.

The activity of the urine samples was as follows: 0.5 µCi/mL (^{131}I) and 0.1 µCi/mL ($^{134,137}Cs$) for one patient and 0.2 µCi/mL (^{131}I) and 0.07 µCi/mL ($^{134,137}Cs$) for the other. From the spectrometric data for the urine and blood samples, it may be concluded that the isotopes in question account for about 90% of absorbed dose of internal irradiation.

The total activity, according to tentative estimates, was about 30 µCi ^{131}I and 10 µCi of the caesium isotopes for the one patient and 12 µCi ^{131}I and 4µCi caesium isotopes for the second.

Tentative estimates of the dose loads of whole-body internal irradiation, made for ^{131}I and $^{134,137}Cs$, were about 4 Sv (400 rem) for the first victim and about 1.5 Sv (150 rem) for the second.

Spectrometric examination of the urine and blood samples as well as direct spectrometry of the whole body and the thyroid gland confirmed that the level of internal penetration of the radionuclides in the remaining victims was considerably lower (by factors of tens to hundreds less).

The data presented are tentative in nature. The spectrometric information was recorded on magnetic tape and is being processed.

7.1.4. Haematological and cardiological research methods for evaluating radiation sickness prognosis and the total external dose level

The haematology laboratory conducted a study of all those persons who has been exposed to the action of ionizing radiation during the accident and had been hospitalized in a special clinic. The morphological composition of peripheral blood was studied in a special infirmary every day for one-and-a-half to two months (number of erythrocytes, leulocytes and reticulocytes; leukocyte formula, number of thrombocytes and haemoglobin content, together with the ESR).

In the case of some patients the cellular composition of the bone marrow was analysed once every 7-14 days (or more frequently in special cases).

The data obtained were used as a basis for prognosticating the course of the bone marrow syndrome which was later confirmed satisfactorily by actual way in which the acute radiation sickness progressed in the patients, including satisfactory coincidence with the preliminary grouping based on the severity and range of the exposure.

For each patient graphs were plotted showing the dynamics of the bone marrow syndrome in terms of variation in the number of neutrophils, thrombocytes and lymphocytes.

A cytogenetic analysis was made in 154 cases. The material used for the investigation was peripheral blood and bone marrow taken at various intervals of time following exposure to radiation (between 1.5 days and 5 weeks). The peripheral blood and bone marrow lymphocytes were cultured at 37°C in medium 199 containing antibiotics, PHA and 5-bromodeoxyuridine (10-20 μg/mL) for 50-67 hours. The cytogenetic analysis was conducted in 50 cells of the first mitosis (preliminary results). To identify these cells use was made of the method of differential staining of sister chromatids.

An evaluation of the exposure was made from the number of dicentrics calculated per 100 cells. Each tricentric, quadricentric and pentacentric was regarded as 2, 3, and 4 dicentrics, respectively. To calculate the dose use was made of a dose/effect curve for dicentrics (which take account of aberrations more precisely) obtained from studies during remission of acute leukosis patients who had undergone relatively uniform whole-body irradiation for therapeutic purposes in doses of 1.5-5 Gy: $Y = (10.79 \pm 2.00)CrD + (5.16 \pm 0.51)CrD^2$, where Y is the frequency of dicentrics (per hundred cells) and D is the dose (Gy). It was shown that the radiosensitivity of the lymphocytes in the peripheral blood of persons suffering from acute leukosis during remission, and of healthy donors after in vitro exposure to gamma radiation at a dose of 4 Gy is approximately the same.

The uniformity of the exposure was evaluated by comparing the observed distribution of dicentrics over the cells with a theoretical Poisson distribution. As is known, if there is a relatively uniform exposure the distribution of dicentrics over the cells obeys Poisson's law and that if the exposure is non-uniform there is a considerable deviation from the law.

The cytogenetic analysis made it possible to evaluate the absorbed dose in the patients who were hospitalized.

In the case of almost all patients the exposure was relatively uniform: the cell dicentric distribution obeyed Poisson's law or deviated slightly from the theoretical distribution. The severity of the bone marrow syndrome was prognosticated from the most informative haematological indicator - the number of neutrophils in the peripheral blood under dynamic conditions (at various intervals of time following exposure). With this purpose in mind the anticipated neutrophil curve was plotted for a dose calculated from the number of dicentrics and was compared with the actual curve observed in the case of each specific individual. Preliminary analysis showed that when there is uniform exposure the neutrophil curves in most cases coincide satisfactorily with the prognostic curves during the phase in which there is reduction in the number of cells. When there was no uniform exposure, the neutropenia was more marked than at the same level of chromosome aberrations in the case of uniform exposure. Figures 7.1.7 and 7.1.8 show as an illustration the cytogenetic study results and a neutrophil curve for the peripheral blood from patient D. The prognostic curve is shown by a broken line, while the solid line shows an actual curve. It is clear from this graph that both curves coincide quite satisfactorily in terms of the time when neutropenia occurs and the degree to which it is marked. The results of the cytogenetic study were used to select persons requiring a transplant of allogenic bone marrow or embryonic liver cells. Patients receiving a transplant underwent cytogenetic tests to see how effectively the transplant had taken. For this purpose there was periodic study of bone marrow punctates and PHA-stimulated cultures of peripheral blood and bone marrow lymphocytes. When transplanting cells from a donor of the opposite sex, use was made of sex chromosomes as markers, and when transplanting cells from donors of the same sex, radiation-induced marker chromosomes (symmetric interchromosomal exchanges and pericentric inversions) were employed.

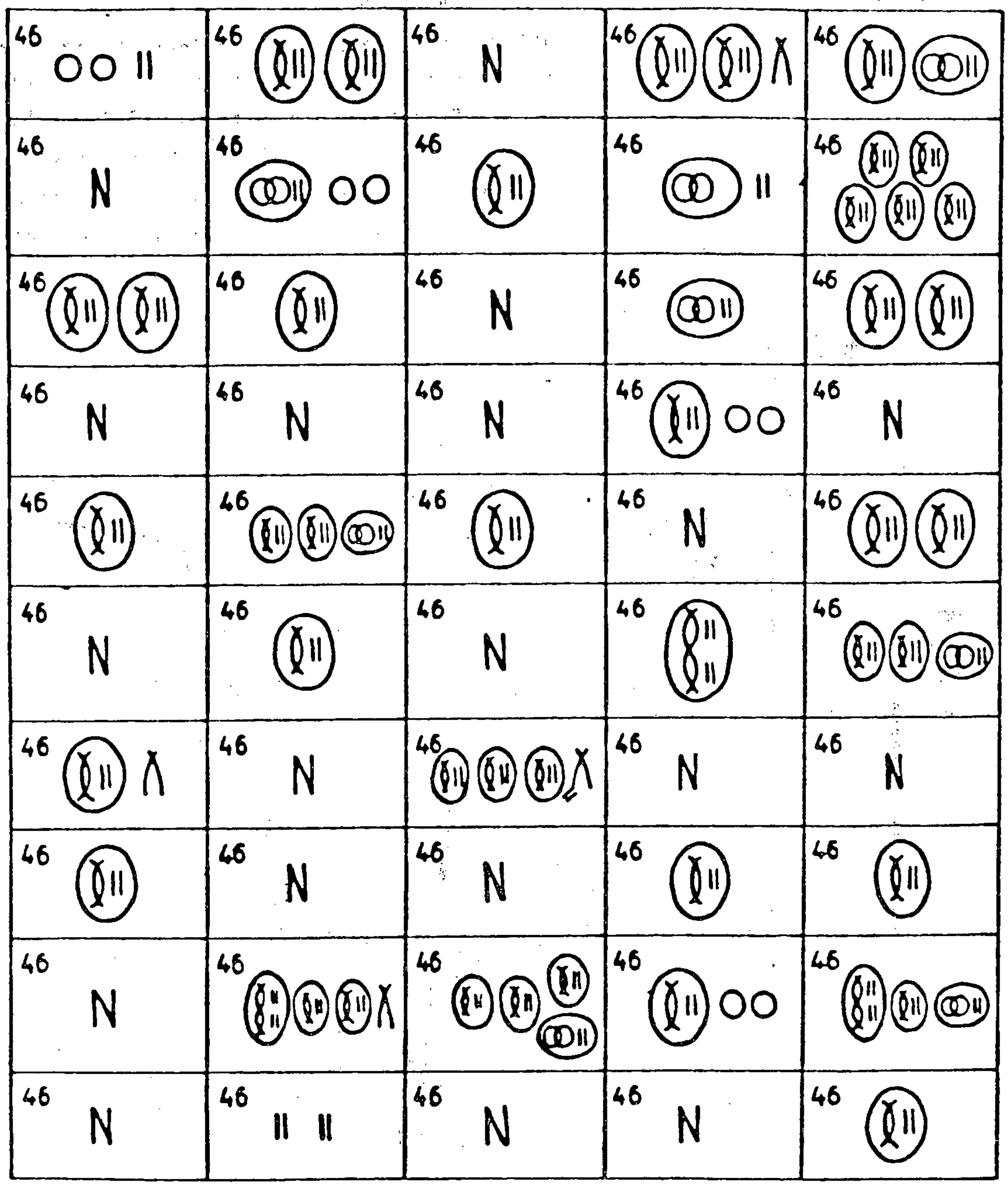
7.1.5. Preliminary evaluation of the use of some biochemical and immunological tests in the event of accidental exposure to radiation

The list of biochemical observation tests corresponded to the one in the USSR for clinical laboratories, and covered about 35 parameters describing the principal metabolic processes, together with 16 tests for the state of the blood clotting system (Table 7.1.1).

The results were compared with control values and norms, as well as with the dynamics of the indicator for the given patient (see the sample recording of the indicator dynamics for patient C in Table 7.1.2). The immunological tests and the trend in the studies are shown in Summary Table 7.1.3. Along with the main laboratories of the specialized clinic a number of other institutions in the country were called upon to assist with

Fig. 7.1.7.

Number of cells analysed - 50 **Number of aberrant cells - 31 (62%)**



Number of dicentrics -
47 (94 per 100 cells)
Dicentric dose - 3.3 Gy

Since the dicentric distribution over cells conforms with the Poisson distribution, exposure to radiation is relatively uniform.

7.1.8. Diagnosis of the severity of the bone marrow syndrome from the results of cytogenic studies of PHA-stimulated lymphocyte cultures.

Patient D. Relatively uniform exposure to radiation (the dicentric cell distribution obeys Poisson's Law). The dose calculated from the dicentrics is 3.3 Gy. The broken line shows the prognostic neutrophil curve for relatively uniform exposure to gamma radiation of 3.5 Gy, while the solid line shows the neutrophil curve observed in the patient.

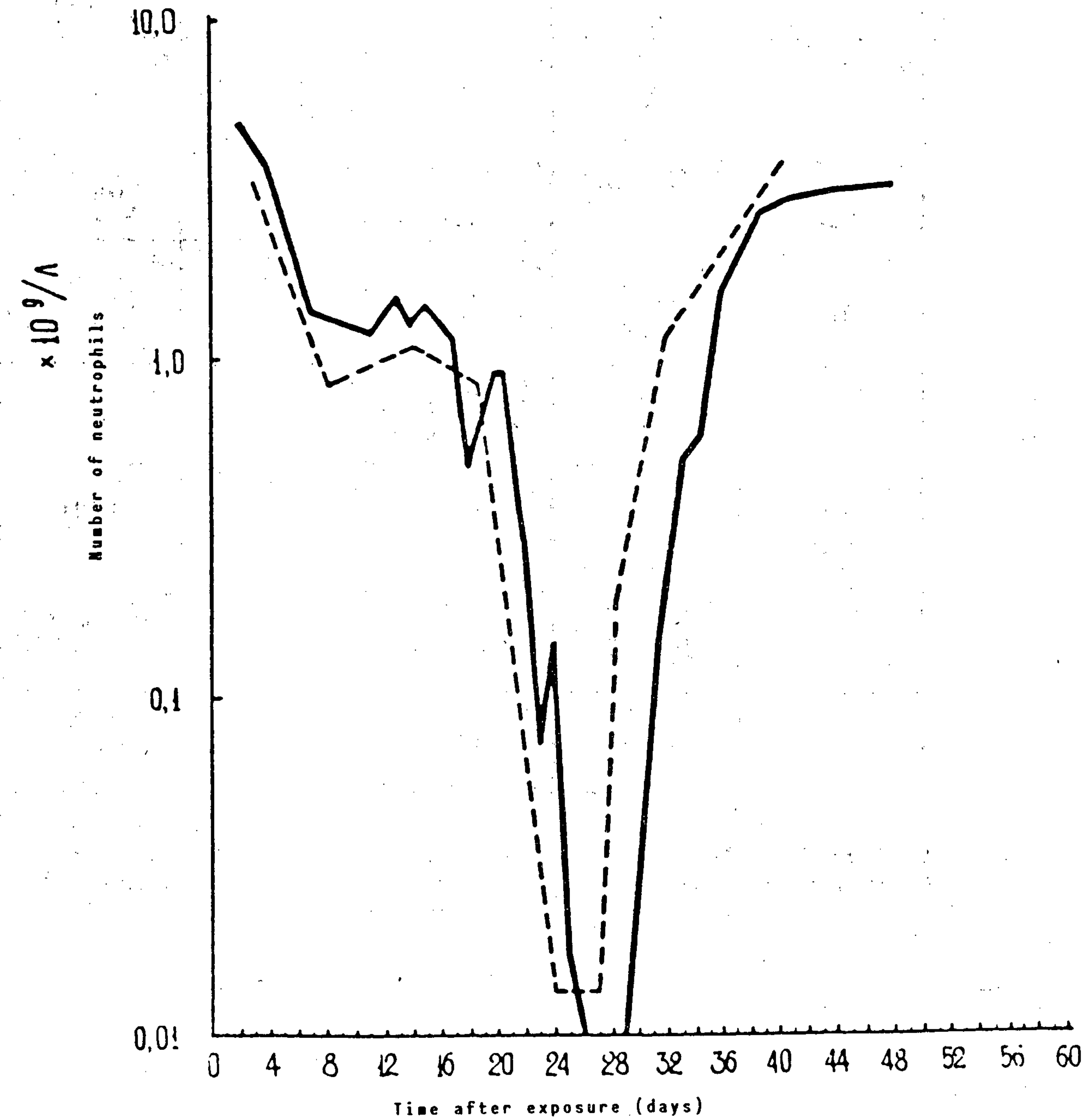


TABLE 7.1.1.

LIST AND NUMBER OF TESTS USED TO STUDY THE PATIENTS

1.	Total protein -	1471
2.	Albumin -	1491
3.	Protein fractions -	266
4.	Urea -	750
5.	Creatinine -	1505
6.	Uric acid -	38
7.	Total cholesterol -	1014
8.	Total bilirubin -	982
9.	Bound bilirubin -	982
10.	Free bilirubin -	889
11.	Potassium -	889
12.	Sodium -	889
13.	Calcium -	256
14.	Iron	102
15.	Phosphorus -	5
16.	Alanine transaminase -	1335
17.	Aspartate transaminase -	1335
18.	a - amylase -	673
19.	Fructose-1-monophosphate-aldolase -	5
20.	Creatine kinase (CK) -	896
21.	Isoenzyme MV/CK -	14
22.	Gamma GTP -	122
23.	Lactate dehydrogenase (LDH) -	961
24.	Isoenzyme $LDH_{1,2}$ -	68
25.	Alkaline phosphate -	977
26.	Glucose -	1850
27.	Rehberg's test -	2
28.	Blood coagulation time -	10
29.	Bleeding time -	10
30.	Recalcification time -	395
31.	Autocoagulation test -	290
32.	Prothrombin index -	516
33.	Retraction index	234
34.	Fibrinogen -	430
35.	Fibrinolytic activity -	357
36.	Paracoagulation tests (ethyl alcohol) -	676
37.	Thrombin time -	240
38.	Reptilase time -	80
39.	Fibrinogen degradation products -	68
40.	Activated recalcification time -	142
41.	Partial thromboplastin time -	2210
42.	Antithrombin-3	152

Table 7.1.2

CHART SHOWING DYNAMICS OF BIOCHEMICAL INDICATORS

Surname and first name. S.V.I.
Date of admission: 26 April 1986
Diagnosis: Acute radiation sickness

Date of examination and name of test	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	I5
Total protein	62-82 r/L	7I		56	70	79	6I	6I	59	53	56	65	47	58	65
Albumin	35-52 r/L			33			3I	26	24	24		27	32	32	28
Al. transam.	up to 40 mU	52	26	20	20	27	22	23	27	I8	I3	I8		II	I4
Aspart. transam.	up to 40 mU	57	28	9	I5	26	I6	27	23	I5	I8	32		43	54
KFK	up to I70 mU	I600	I43		I70	380	89	89		26				28	
LDH	up to 460 mU		282		233	326	266	I60	65					352	
Isoenzyme LDH_1	I,2 up to 280 mU		I33		89	I50	I62							2II	
Isoenzyme LDH_2	3-5 70-I40	20400		I60											
Urea	2,5-8,3 mmole/L	4,9					9,8	I0,0	I0,7	I4,5	22,7	23,4	27,7	8,3	27,2

Continuation of Table 7.1.2

	I		2	3	4	5	6	7	8	9	I0	II	I2	I3	I4	I5
Creatinine	34–I34	μmole/L		94	74	I04	I24	I80	I42	I54	200	265	285	348		270
Total cholesterol	3,5–6,5	mmole/L		3,7	2,6	4,0	3,6	2,3	I,5	2,2	2,0					
Glucose	2,0–3,0	mmole/L					I,9		I,7	I,85			I,9			
Total bilirubin	3,2–5,6	mmole/L	3,9	2,0	2,6	3,85	4,2	4,I	4,6	3,8	4,4	2,8	2,0		2,0	I,8
Bound bilirubin	4,6–I5,I	μmole/L	23,9		9,6	I0,3			I8,3	I7,I	3I,3	29,8	33,2	4I,0	38,8	50,8
Potassium	up to 5,0	μmole/L	6,8						I0,8	I3,2	25,8	26,7	3I,3	4,0	30,I	34,5
Sodium	3,5–5,5	mmole/L						5,0	4,0	4,I	3,8		3,2		3,2	2,7
	I39–I53	mmole/L						I34	I36	I40	I42		I50		I52	I60

Table 7.1.3

BASIC TRENDS IN IMMUNOLOGICAL STUDY AND TESTS APPLIED

Group of patients	Blood group and rh factor	Typing from M and A system antigens	Selection of donors from relatives with blood groups and	Typing from erythrocytic antigens	Removal of T-1 from bone-marrow in haplo-identical transplants	Determination of lymphocyte sub-populations	Determination of A, M and S immunoglobulin concentration	Activation of immunity by T-activin	Study of isosensitization	Verification of bone marrow taken from erythrocyte chimeras and from M and A	Study of autoimmunity due to damage to central marrow system
Severe cases require bone marrow transplant	All without exception	All during the first few days after admission to infirmary	Obligatory for all	When deciding on transplant	When there is no compatible donor for M and A	During recovery of blood production - once a week	Provisionally once a week	Choice of treatment based on indications	In all patients with blood transfusions as transfusion reactions develop	In transplant of AVO incompatible bone marrow and haplo-identical bone marrow	Twice a week
Severe cases not needing bone marrow transplant	"		On the basis of indicators - once a week			On the basis of indicators - once a week					Once a week
Patients with moderate severity	"										
Patients with slight injuries	"										

the studies. Specialists from the United States took part in the work of typifying the antigenic structure of the lymphocytes and removal of T-lymphocytes from the transplant - haplo-identical bone marrow.

Biochemical study of seriously ill patients and those with a moderate degree of radiation sickness was conducted on a daily basis. First- and second-degree patients were examined twice a week, using the same system.

The number of analyses performed during observation in the case of one seriously ill patient amounted to about 800, while in the case of a first-degree patient it was 200.

The materials are in process of being analysed. Brief preliminary results are shown below.

During the biochemical study of acutely affected persons 36-48 hours after exposure to radiation, pronounced hyperamylasaemia and hyperamylosuria were found. The frequency with which the norm was exceeded matched the serious degree of affliction and in the worst affected group attained a factor of 10-100 times the norm.

In the case of patients with extensive radiation burns, high levels of creatine kinase were determined within a day following exposure. The enzyme activity in certain cases exceeded the norm by a factor of 10-20. Concurrently with these changes there was found to be a moderate increase in aspartate transaminase.

During the investigation (especially when the sickness was at its height) there was noted a fairly clear-cut variation in the protein spectrum: hypoproteinaemia and hypoalbuminaemia.

Seven to ten days after exposure to the radiation many patients showed considerable shifts in the case of a number of biochemical samples and enzymes, which testified to malfunction of the kidney: hyperfermentaemia (aspartate and alanine transaminase, alkaline phosphatase and lactate dehydrogenase) as well as hyperbilirubinaemia with occurence of a direct bilirubin fraction.

Disruption of kidney activity in those seriously ill was manifested by a high increase in the creatinine level (three or four times more than normal). There were several other special examinations, the results of which are in process of being analysed (determination of the hydroperoxides, succino-oxydismutase, malonic dialdehyde, ceruloplasmin, alpha tocopherol and peroxide haemolysis of the erythrocytes).

When studying the haemostasis system in exposed persons, with effect from the fifth day following the accident there was observed to be activation of the plasma procoagulants which was still preserved even when there was development of marked thrombocytopenia; this is confirmed by the indicators of the autocoagulation test.

By the tenth day most of the patients showed sharply positive paracoagulation test data. There was a drop in fibrinogen and antithrombin III. The drop in the prothrombin index continued until the fourth week of illness, the most obvious being the decrease in the K-dependent factor of the prothrombin complex.

By the end of the month all the coagulation indicators had approached normal in the overwhelming majority of patients. Despite the clinical assumptions with regard to the presence of a DVS syndrome, in no single case was there laboratory confirmation of the dynamics typical for coagulation tests.

The immunological study was used the most in typifying and selecting donors for bone marrow transplants.

The immunological selection and testing of blood groups and the rh factor (in more than 200 persons) made it possible to provide more adequate transfusion treatment (injection of RBC). In a limited number of patients the typifying was based on erythrocyte antigens (Table 7.1.3).

An assessment was made of the isosensitivization to tissue antigens by applying the Coombs' indirect test and by testing the lymphocytotoxic reaction and aggregate haemagglutination. In a handful of cases different methods were used to determine the lymphocyte sub-population. A large number of examinations were carried out to evaluate microdestructive processes in the nervous system, using neuro-immunological cell serum tests for the purpose.

Wide use was made of a bacteriological study of microbial dissemination in the environment for various conditions under which the patients were maintained. Inoculations were made of blood, faeces, urine, mucosa of the oral cavity and throat, and tranmatized surfaces. The concentration in the blood of certain antimicrobial drugs and antibiotics was quantified.

The results are in process of being analysed.

7.1.6. Alteration to the skin and the part they play in the outcome of the sickness

Characteristic features of the reaction by the skin and mucosa in the given situation were the existence of several variations of the lesion, which were sometimes found simultaneously in the same patient:

- Widespread surface injuries predominantly occurring on open parts of the body unprotected by clothing, on the lips, conjunctiva and at the entrance to the mouth;

- Injuries limited to zones of predominantly direct contact with beta and gamma radiation sources (clothing or footwear that was wet and soiled by a service solution, adhesion of dust or touching of contaminated objects);

4. Injuries to the skin and mucosa, the stomatopharynx and intestine due to relatively uniform exposure to gamma radiation in doses exceeding the threshold values for those tissues.

Radiation injuries to the skin (beta radiation burns) covering more than 1% of the body surface were observed in the case of 48 persons.

The contribution made by radiation damage to the skin to the overall clinical radiation sickness syndrome with considerable aggravation was determined by the extent and depth (degree) of the injuries. Here the damage to the skin was virtually incompatible with survival in the case of some of the patients (14 persons).

The clinically detectable extent of the skin lesions in the cased of most patients underwent certain dynamic changes in time and was characterized by the occurrence of several, at least two or three, "waves" of erythema and ensuing changes in the skin.

The primary skin erythema detected on the first or second day after exposure was not enough of a reliable criterion for prognosis of the sickness by virtue of its instability and the lack of reliable methods for quantifying the intensity of it.

Between the end of the first and the third week eight persons with damage almost to all the skin (from 60-100% of the total surface area) stood out in terms of the extent and intensity of the main erythema wave. The hyperaemia of the skin in these cases was accompanied by edema and there was early formation of blisters and erosion (erosive-ulcerous dermatitis).

All these persons died within a period of 15 to 24 days. They also had severe and extremely severe damage to the haemogenic system and an intestinal radiation syndrome. We consider it advisable, however, to stress that these patients had damage to the skin that was incompatible with survival.

Damage to an area of 30-60% of the total body surface was detected in the case of 12 patients by the end of the third week. In the majority of these (seven of them) the severity of the bone marrow syndrome was assessed as extremely severe, in three cases as severe, and in one case as moderately severe. In all, it proved lethal in this group in nine cases.

In six patients injuries to their skin were assessed as incompatible with survival (covering more than 50%, early formation of extensive erosive-ulcerated surfaces). These six patients died and in the case of one of them the skin damage was the main cause of death (death occurred on the forty-eighth day, although the peripheral blood picture had fully recovered). Effects of endogenous intoxication were reponsible for the development in this patient of toxic edema of the brain and a terminal coma.

Skin damage covering a total area of up to 30% was noticed by the twenty-first day in the case of 21 patients. Six of these can be said to have had serious aggravation of their overall condition through the extent (25-30%) and the severity of the injuries, with early development of erosive ulcerous changes. Damage to the bone marrow in this group of patients differed from extremely severe to slight. There were no deaths in this group due to skin damage.

Over a period of 36-45 days (6-8 weeks), i.e. during the complete or almost complete recovery of blood production, there was also observed recovery of the earlier altered integument. At the same time, at a surprisingly late stage, on the earlier unaffected areas there appeared new changes in the form of bright erythema with cutaneous edema. The overall area of the damage increased accordingly: previously assessed at 25-30%, it attained 90-100% of the body surface. In areas of previously unaltered skin there was sometimes reappearance of more intense edema, and the size of the areas of healing ulcers and erosions increased. Some of the patients with these "late" skin injuries had shown no alteration of the skin at all at an earlier stage (up to 3 weeks).

On the thirty-sixth to forty-fifth day the most typical lesions were those in the area of the shins and hips. The patients noticed the occurrence (or intensification) of aching in the legs, to the extent of not being able to stand up, and the lymphostasis and edema in a more distal direction from the "focal point" of the skin damage (edema of the ankles with erythema on the shins), a general reaction in the form of a rise in temperature, inability to sleep, and so on.

Recovery from the skin injuries usually ended by the fiftieth-sixtieth day. It usually took the form of dry or moist desquamation according to the degree of injury. By this time the erosions and surface ulcers had epithelialized in most of the patients.

The absence of active epithelialization by this time over sizable areas (20-25 cm^2) was interpreted as an indication for surgical intervention.

7.1.7. Methods of treatment and preliminary assessment of their effectiveness

The methods used to treat individual acute radiation sickness syndromes were ones that had been tested and used in everyday practice.

The main emphasis was placed on preventive treatment and treatment of infection complications and substitional therapy using blood cells in the case of the bone marrow syndrome; detoxifying therapy and total parenteral feeding, in the case of widespread burns and oropharyngeal and intestinal syndromes; intensive correction of the water-electrolyte balance in patients with the intestinal syndrome and a toxico-septic status due to burns and agranulocytic infections.

The clinical charts (Figs 7.1.9 and 7.1.10) illustrate the dynamics of the basic manifestations and treatment of the sickness.

Treatment of the bone marrow syndrome:

(a) Supporting and substitional therapy.

All the patients with a bone marrow syndrome of the second or a higher degree were placed separately in normal hospital wards adapted to ensure asepsis while they were being treated: sterilization of the air by ultraviolet lamps, strict observance by the personnel of the habit of washing their hands when entering and leaving the ward, compulsory use of individual gowns and masks in the ward, wiping of footwear on a mat dusted with antiseptic, and a change of the patient's clothing once a day. In the shortest possible time relatively simple aseptic conditions were created throughout the hospital, enabling specially trained personnel to look after the patients. Contamination by microbes was checked and paper linen and clothing for personnel were worn. The conditions described ensured a low micro-organism content in the air - not more than 500 colonies per m^3.

The diet was normal, but raw vegetables and fruit as well as tinned products were excluded.

The effectiveness of this aseptic regime was clearly demonstrated, as shown by us earlier (A.E. Baranov et. al., 1978, 1982), by the absence of exogenous broncho-pulmonary infections (pneumonia) in patients with acute radiation sickness of the second and third degrees.

In the case of all patients with the bone marrow syndrome of the second to the fourth degree, preventive treatment was provided against endogenous infections using biseptol and nistatin, beginning either one or 2-3 weeks before the development of agranulocytic infection. The comparative effectiveness of the two alternatives for the beginning of selective decontamination of the intestine was evaluated.

When agranulocytic fever occurred, two or three antibiotics with a broad antibacterial spectrum from the aminoglucoside group (gentamycin, amicacin), cephalosporins (cefsol, cefamesin, cefobid) and semisynthetic penicillins with antipyocyanic activity (carbenicillin, pipracil) were given intravenously. In at least half the cases the use of antibiotics stopped the fever. If there was no effect within 24-48 hours when treating the given group, extensive use was made of intravenous injection of gamma-globulin provided by the Sandos company. The gamma-globulin (Sandoglobulin) was given in doses of 6 g four or five times every 12 hours.

A policy was pursued of "early" empirical prescription of amphotericin B if the agranulocytic fever was not cured within 1 week by the antibacterial antibiotics mentioned, combined with intravenous gamma-globulin.

In the prevailing situation, "atsiclovir" was used with good effect for the first time in herpetic infections while treating acute radiation sickness (approximately one third of the patients had at times severe herpes simplex of the skin of the face, lips and mucosa of the mouth). "Atsiclovir" was not used for preventive treatment. Good results in the treatment of virus infections of the skin were obtained with ointments containing "atsiclovir".

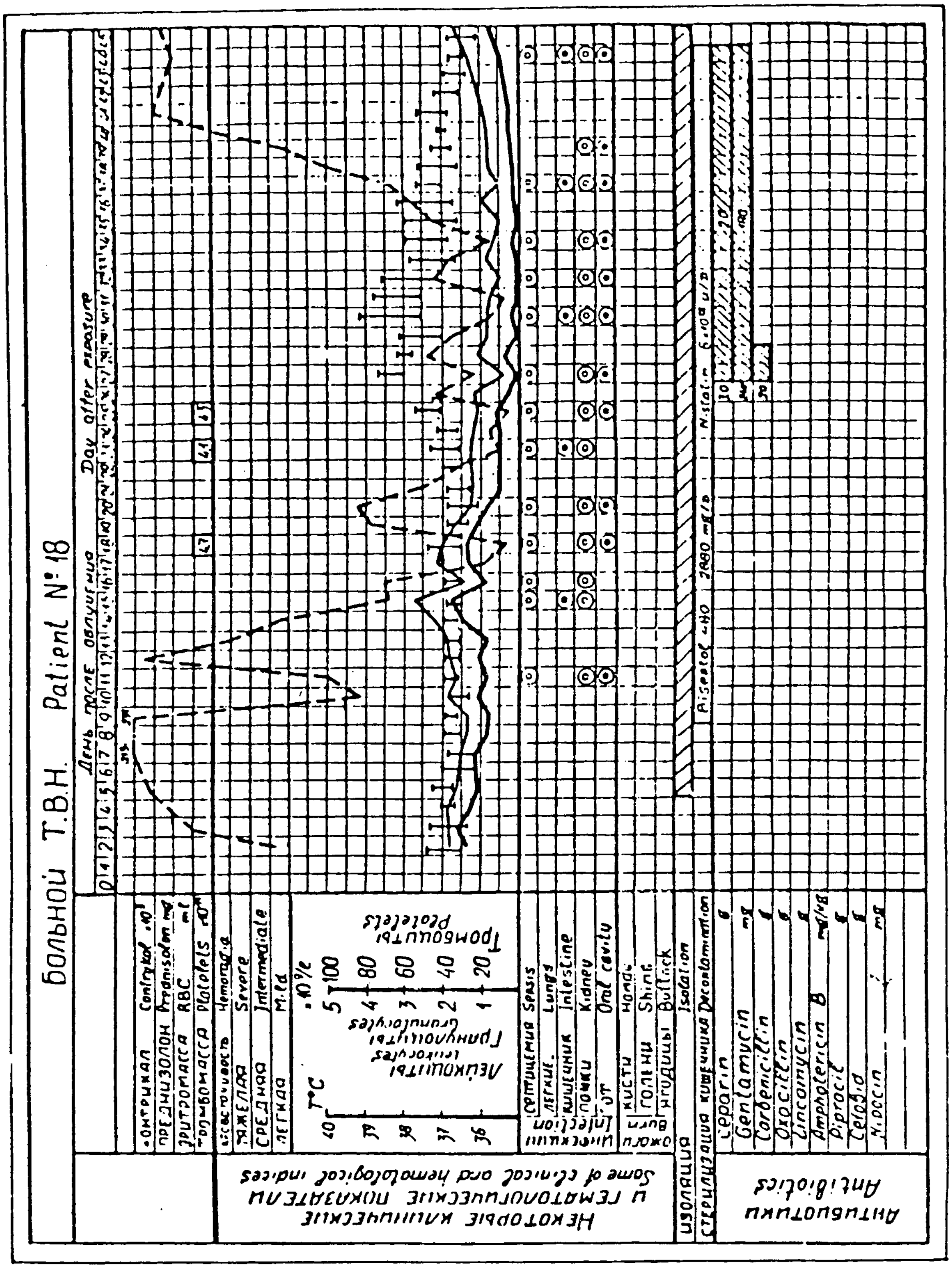
This regime of basically empirical antibacterial, antifungal and antiviral treatment proved highly effective - there was hardly any mortality due to infection in the case of the patients with the bone marrow form of acute radiation sickness, even in severe and extremely severe cases (without burns). Furthermore, the autopsy of patients dying from non-bone- marrow injuries did not show any incontestable signs of bacterial or mycotic septicemias.

Both during their lifetime and posthumously, epidermal staphylococci were isolated from the blood of most of those who died. The part played by them as a pathogen in terminal septicemias despite the antibacterial treatment given is being studied.

Several patients with bone marrow syndrome of the fourth degree were found to have acute diffusional interstitial pneumonia, accompanied by the rapid development of hypoxaemia incompatible with survival. The bacterial and mycotic nature of the pneumonia was not confirmed by the autopsy, but acute radiation pulmonitis, with possible activation of cytomegaloviruses, did tend to be found.

One of the methods used widely - and without a doubt very successfully - for treating these patients with acute radiation disease was that of taking fresh platelets from donors. Platelets were obtained by taking blood four times from a single donor. Indications for a platelet transfusion being required were incipient bleeding or a reduction in the platelet level to

Fig.7.1.9.


Больной Т.В.Н. Patient N° 18
День после облучения Day after exposure
Некоторые клинические и гематологические показатели
Some of clinical and hemotological indices
Лейкоциты Leukocytes
Гранулоциты Granulocytes
Тромбоциты Platelets
Sepsis
Lungs
Intestine
Kidney
Oral cavity
Hands
Shins
Buttock
Изоляция Isolation
Decontamination
Антибиотики Antibiotics
Ceporin
Gentamycin
Carbenicillin
Oxacillin
Lincomycin
Amphotericin B
Nistatin

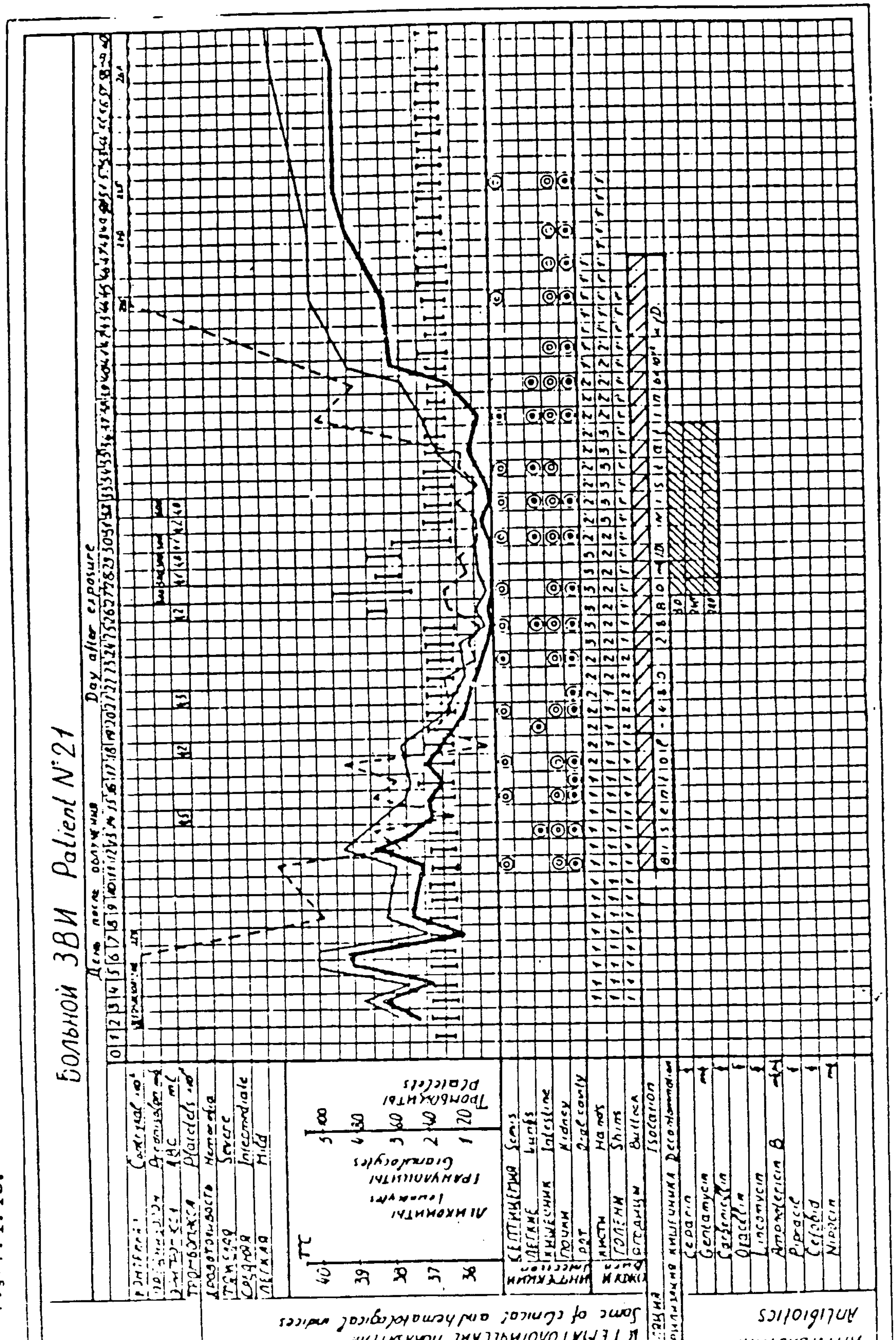


Fig. 7.1.10.

below 20 000 per μL. Usually, blood from a single donor (on average some 300 x 10^9 platelets) was used for each transfusion. The platelet transfusion was repeated after 1-3 days. The platelets and all other blood components were irradiated at a dose of 1500 rad before the transfusion, using a conventional gamma therapy device. This was done in order to prevent secondary disease.

The considerable effectiveness of platelet transfusions carried out in accordance with the principles mentioned is confirmed not only by the absence of dangerous bleeding even in patients with long-lasting (more than 2-4 weeks) and severe thrombocytopaenia (less than 5000-10 000/μL) but also by the absence in most patients of any signs at all of bleeding.

Obtaining the necessary quantity of platelets at the time required during the period of most severe thrombocytopaenia simultaneously in dozens of patients required considerable efforts on the part of the blood donation service. There was no lack of platelets. Moreover, sometimes the supply was "excessive" because of the impossibility of planning requirements precisely one or two days in advance. Because of this, extensive use was made for the first time when treating this group of patients of the method of freeze-drying of both allogenous and autologous platelets, which were then used at the time required with considerable effectiveness.

There were no cases of rejection of blood transfusions as a result of alloimmunization.

On average, 3-5 transfusions of platelets were required to treat one patient with third degree bone marrow syndrome.

Leucocytes were not used for treating agranulocytic infectious complications.

Erythrocyte requirements were found to be much greater than expected, even in patients with acute second- or third-degree radiation sickness with uncomplicated radiation burns because of the development relatively early of severe anaemia.

(b) Bone marrow transplants

During the first three days after irradiation, a first group of patients with irreversible breakdown of myelopoiesis was selected in whom spontaneous restoration of myelopoiesis could therefore hardly be expected.

The diagnosis of the irreversibility of myelodepression was carried out in accordance with rules developed earlier on the basis of criteria such as the onset of vomiting, the number of peripheral blood lymphocytes and an estimate of absorbed dose from the number of aberrations in bone marrow cells taken 36 hours after irradiation.

On subsequent days (from the fourth to the ninth) the same criteria were applied and, in addition, an estimate of dose from chromosomal aberrations in lymphocyte culture and peripheral blood, to gauge the extent of myelopoiesis more exactly and to make a final selection of the group of patients in whom, as a result of extremely severe (and possibly irreversible) myelopoiesis (doses of 6 Gr and more) bone marrow transplants were indicated. At this time an active part in the work was taken by American experts headed by Prof. Robert Gale.

Transplants were taken only from close relatives (natural brothers and sisters or parents) in whom HLA was identical (six cases), haplo-identical (four cases) or haplo-identical plus one common antigen in the second haplotype (three cases). In view of the urgency of transplantation, type testing was carried out only in respect of the A, B and C loci. With transplantation of haplo-identical bone marrow, T-lymphocyte depletion was performed in order to ward off secondary disease.

The main difficulty in matching the HLA of suitable donors was that it was necesary to determine the fenetin HLA in many of the patients in the first three days after irradiation when the number of lymphocytes had not dropped to very low levels.

Considerable organizational difficulties were experienced in connection with the need to find and examine quickly a large number of relatives of patients who constituted potential donors. In this situation 113 donors were examined.

In the end, only 13 transplants of allogenous bone marrow were carried out (between the fourth and sixteenth day after irradiation). In six cases of extremely severe damage to the skin and intestines and extremely unfavourable prognosis, transplantation of human embryo liver cells was performed.

In general, it can be said that bone marrow transplants were not a decisive factor in treatment after this particular accident. All seven patients in whom, in view of their particularly severe irradiation, a donor's bone marrow might have been able to "take" in a stable manner, died earlier (between 9 and 19 days after the bone marrow transplant) from radiation damage to the skin and intestines. At the same time, in the remaining six patients, who did not have skin and intestinal damage incompatible with survival, only a temporary or incomplete "taking" of donor's bone marrow occurred, presumably because the transplantation immunity of these patients was not sufficiently suppressed by irradiation. Despite the unfavourable (insufficient, incomplete or totally arrested) myeloid function of the transplant, the same disorders were observed in all of these patients; they may have been caused by a reaction of the host against the transplant or of the transplant against the host. In two cases these reactions may have contributed to death from a renal and pulmonary insufficiency syndrome of unidentified genesis and from pyocyanic septicaemia which developed unexpectedly against a background of a normal number of neutrophils.

An analysis in retrospect of the initial part of the neutrophil curve in all six patients gives rise to doubt about whether the breakdown in myelopoeisis was irreversible in them. The fact that transplants "took" - even temporarily - and that this gave rise to immunological conflicts would appear to show a negative influence both on the restoration of myelopoeisis and on the course of the disease as a whole (two patients died).

Thus, the experience with transplantation of allogenous bone marrow after the particular radiation accident gives rise to two important conclusions for the future in respect of this method of treatment:

- In radiation accidents the proportion of patients in whom transplantation of allogenous bone marrow is absolutely indicated and for whom this treatment will obviously be beneficial is very small;

- With reversible breakdown of myelopoeisis caused by overall gamma radiation doses of the order of 6-8 Gr, a transplant may "take" but this "taking" will always have a negative effect in therapeutic terms and even endangers life as a result of the high risk of secondary disease developing.

The latter conclusion is essentially a new one since it was assumed earlier that the transplantation of allogenous bone marrow does not give rise to negative effects in the event of insufficient radiation exposure of the recipient in the borderline area zone of radiation doses.

Treatment of radiation damage to skin

In view of the important, and in a number of cases determining, role of local radiation damage in the overall clinical syndrome (intoxication, pain), its treatment occupied an important position in the therapeutic measures taken.

Use was made of basic modern detoxification methods and also of disaggregational, anti-infection and symptomatic therapy; haemosorption, plasmosorption and plasma phoresis were carried out, and direct anticoagulants and means of improving microcirculation (repoliglucin, neogemodes, troxevasin, trental and solkoseril) were applied.

The local treatment methods used were appropriate for the stage and severity of the complaints: initially, spraying was carried out with aerosols with a bactericidal and analgesic/anti-inflammatory effect, including the effective Soviet product "Lioxanol". Use was made of moistening bandages based on tannin solutions with bactericidal properties ("Baliz"). Later, bandages containing ointment with derivatives of hydrocortisone based on propolis and wax with specifically directed antibiotics and antiseptics were applied.

Treatment of oropharyngeal syndrome

The main techniques used for treating severe radiation mucositis were mechanical removal of enormous quantities of rubbery mucus which had accumulated in the nasopharynx, washing this mucus away and bathing the erosive surfaces with solutions of mucolytics with antiseptics.

Experience shows that mucolytic preparations and techniques for using them with a view to rapid and reliable removal of mucus from the buccal cavity, the vestibule of the larynx and the nasopharynx are particularly in need of improvement.

Treatment of intestinal syndrome

The main technique used for the treatment of intestinal syndrome was total parenteral feeding with intensive correction of the volume of nutritive liquid and electrolytes, which in the present case also proved to be highly effective. Experience has shown that it is necessary to have a large reserve of mixtures for parenteral feeding ready if plans are to be laid for providing specialized assistance to patients with acute radiation damage to the intestines, mouth and gullet.

In conclusion, it should be noted that every patient with bone marrow syndrome of the third and fourth degree of severity, which was mostly accompanied by radiation burns, required individual round-the-clock attention from a nursing unit consisting of highly qualified nurses specialized in intensive care in order for the treatment described above to be provided. As a result of considerable efforts - which are, of course, feasible only in peacetime - it was possible to provide such specialized nursing units for each patient with acute radiation disease of the third and fourth degrees.

As a whole, the effectiveness of treatment can be considered entirely satisfactory: none of the patients with second-degree acute radiation disease (doses of 2-4 Gr) died. In 19 cases the deaths of patients with third and fourth degree acute radiation disease occurred only as a result of severe damage, which was incompatible with survival to 50-90% of the surface of their bodies and was itself of between the second and fourth degrees of severity.

CONCLUSION

Of the operative and accident management staff who took part in operations at the Chernobyl' nuclear power station on 26 April 1986, 203 were found to have acute radiation sickness. The clinical manifestations (syndromes), degree of severity and outcome for different patients were varied. All those affected were identified in good time and admitted for treatment to qualified Soviet institutions.

Of the 22 patients suffering from an extremely severe degree of acute radiation sickness, 19 died. In 14 cases, death was caused by severe radiation and thermoradiation damage to the skin, against a background of severe blood formation depression and, in some cases, of damage to the digestive tract.

Of the 23 patients with severe bone marrow syndrome, seven died. In six cases the outcome of the illness was also determined by the presence of widespread and severe radiation damage to the skin, with the attendant pronounced overall intoxication.

There was not a single death among patients with first and second degree illness.

For the majority of patients, clinical recovery occurred toward the end of the second month following the accident. There are at present 30 people undergoing hospital treatment.

The main harmful factor for all victims was the relatively uniform gamma- and beta-radiation effect in a dosage which, according to biological criteria, exceeded 1 Gy, and which, in the case of 35 people, exceeded 4 Gy (up to 12-16 Gy). Fifty people suffered additional beta-irradiation of significant areas of their skin, while in the case of a number of people this extended to the mucous membranes of the naso-pharynx and gastrointestinal tract.

Radiation damage to wide areas of the skin (up to 50-90% of the surface) was one of the main factors contributing to the overall severe condition of the patients, and was a determining factor in the main fatal complications (cerebral oedema, toxic encephalomyelopathy, renal and liver function insufficiency and damage to the myocardium).

In the case of certain patients, in the maximum dose range for external gamma-irradiation (~8 Gy), the terminal period was characterized by the development of pulmonitis and pronounced respiratory insufficiency.

In the case of practically all patients, without there being any apparent link with either the presence or degree of severity of acute radiation sickness, a complex mixture of nuclides was found to have entered the organism, these being mainly isotopes of iodine, caesium, zirconium, niobium and ruthenium. However, their quantities and the dose levels in all except one patient, were lower than those clinically indicated to have direct effects. In the first 10 days following the accident, the content of iodine isotopes in the thyroid gland of 94% of individuals did not exceed 50 μCi.

With the help of previously-acquired experience it has been possible to formulate a timely and sufficiently complete prognosis of the course of the illness in the overwhelming majority of cases. Within the first 24 hours it proved possible to determine correctly which individuals required urgent hospitalization, and, in the first three days, to determine the appropriate level of medical assistance required. It subsequently proved possible to achieve certain therapeutic results.

In the course of intensive clinical observation and active medical attention a great deal of data has been accumulated and is now being processed.

The preliminary results can be reduced to the following main points:

1. The previously formulated main diagnostic principles and prognostic criteria relating to the course of illness through bone marrow syndrome have justified themselves;

2. In the accident situation under consideration, a certain aggravating role was played by widespread beta-damage to the skin. In a considerable number of cases such damage determined the severity and outcome of the illness;

3. At the forefront of therapeutic measures taken, in accordance with the structure and syndromology of the injuries, were those concerned with preventing and treating the complications connected with severe but reversible blood formation depression, detoxification therapy and the local treatment of skin damage;

4. The nature of the damage (relatively superficial but very widespread beta-dermatitis) called, for the most part, for conservative therapy; surgical intervention has only been required in exceptional cases (five people so far);

5. Bone marrow transplantation was indicated (dosage above 6 Gy) and feasible (absence or only minor indication of reasons for unsuccessful outcome) only for a very limited group (13 individuals). On account of those reasons, and given the remaining possibility of self-repair, albeit slow and incomplete, transplantation has only been moderately effective;

6. All patients are now being actively monitored. This will make it possible to determine the complete course of their rehabilitation and the need for preventive treatment.

7.2. Data on the doses from radiation exposure to the population in the thirty-kilometre zone around the nuclear power plant in different regions of the European part of the USSR, radiation consequences of the accident

7.2.1. Introduction

Immediately after the accident, measures were taken to implement continuous effective monitoring of the parameters of the radiation conditions both at the site of the Chernobyl' nuclear power plant and in the neighbouring populated areas. Particular attention was paid to the town of Pripyat' which had about 45 000 inhabitants who were mainly plant personnel and members of their families. As the radiation conditions developed, the scale and volume of dosimetric monitoring increased significantly over the course of time. In the end, more than 7000 subdivisions of radiation laboratories, epidemiological centres and also many groups of radiation safety experts from a large number of scientific and practical establishments and organizations throughout the USSR were mobilized to carry out the monitoring.

The primary most important radiation monitoring tasks were:

- to evaluate the possible external and internal exposure of staff at the Chernobyl' nuclear power plant, of inhabitants of Pripyat' and of the population which was subseqeuntly evacuated from the 30-kilometre zone, in order to identify those persons in need of medical assistance;

- to predict the possible levels of exposure of the population in regions of high radioactive contamination outside the 30-kilometre zone in order to decide whether it was necessary to carry out a further full or partial evacuation or to make appropriate temporary recommendations regarding the diet and activities of those living in the region;

- to prevent the spread of radioactive materials by menas of contact from the contaminated regions and also the consumption of food products with a radionuclide content higher than the regulatory values.

In order to deal with these problems, systematic monitoring was set up to check:

- the gamma radiation levels throughout the Eurpoean part of the USSR using aerial and land radiation monitoring;

- the concentrations and radionuclide composition of radioactive substances in the air at various points in the 30-kilometre zone, primarily in places where work was carried out to eliminate the consequences of the accident and where personnel were stationed, and also outside the 30-kilometre zone in populated areas where high radiation levels were observed;

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

- the radionuclide content of drinking water reservoirs and also of food products reaching shops;

- the iodine radionuclide content which accounted for the main internal exposure during the initial period after the accident in the thyroid gland for the population evacuated from the 30-kilometre zone and that living in regions of high gamma background levels;

- the radioactive contamination levels of overalls or personal clothing and footware, of external and internal surfaces of transport vehicles on the borders of the controlled zones (established on the basis of the nature of the work and the developing radiation conditions) and at airports, railway and bus stations.

7.2.2. Level of external exposure of the population of the town of Pripyat' from the time of the accident until the time of evacuation

From the beginning of the accident at the fourth unit and during the fire which followed, the wind carried radioactive products past the town of Pripyat'. Subsequently, when the height of the products released from the stricken reactor dropped considerably, the radioactive plume gradually covered the area of the town and contaminated it as a result of a change in the wind direction in the near-ground air. From 9.00 p.m. on 26 April 1986 the gamma radiation exposure dose rate measured 1 m from the ground in different streets of the town was within the limits 14-140 mR/h.

Subsequently, the radiation conditions in the town worsened. On 27 April 1986 at 7.00 a.m. in the region closest to the power plant (Kurchatova street), the gamma radiation dose rate reached 180-600 mR/h and at other streets 180-300 mR/h. The tendency for the radiation conditions in the town to get worse during 27 April 1986 continued and at 5.00 p.m., i.e. after complete evacuation of the population it was 360-540 mR/h and in the Kurchatova street area 720-1000 mR/h. Evacuation of the population began on 27 April 1986 at 2.00 p.m.

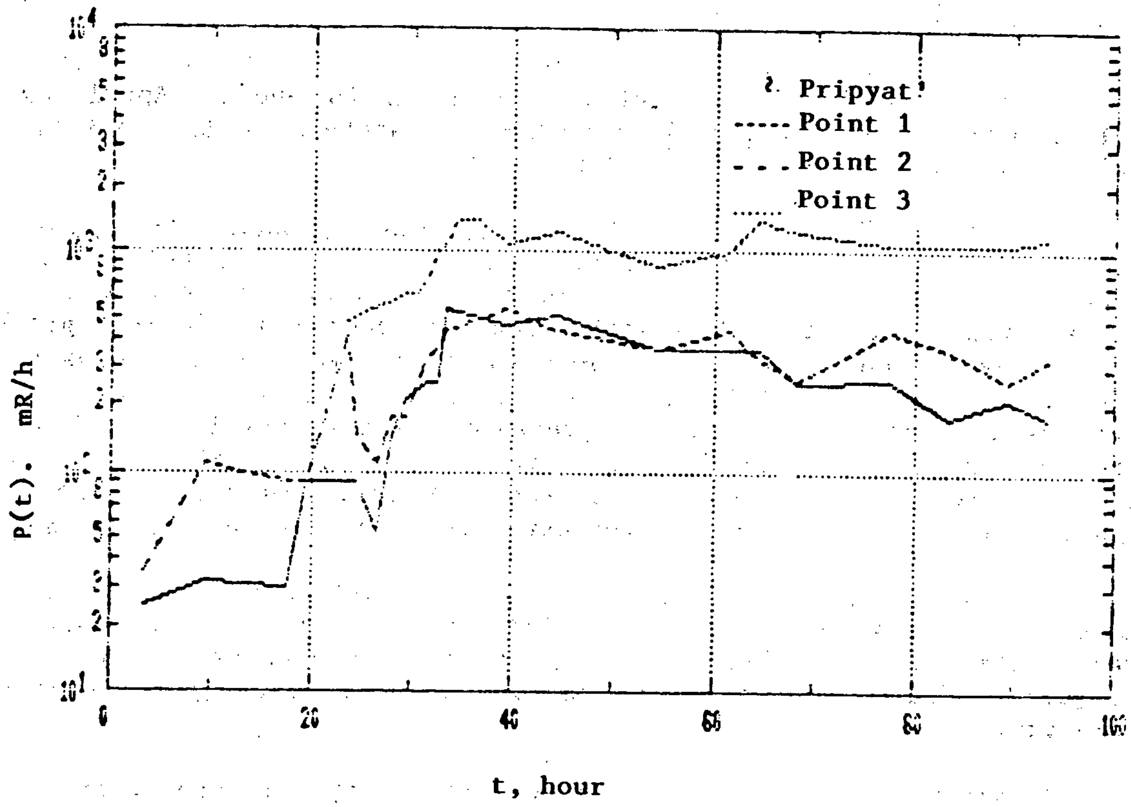


Fig. 7.2.I Dynamics of the change in dose rate in Pripyat' during the first four days after the accident

Figure 7.2.1 shows data on the change in the radiation conditions of various areas of the town of Pripyat' from the beginning of the accident until completion of evacuation. The gamma radiation dose rate during this time was 5.9; 7.1; and 20.3 P at points 1, 2 and 3 respectively. By 6 May the radiation levels in Pripyat' had dropped by a factor of three. Purely rough calculations make it possible to assume that the external gamma radiation dose from the passing cloud of effluent during the first hours after the accident was close to 10-15 R.

The estimate of the exposure levels of the population of the town was based on the possible pattern of behaviour on 26 and 27 April and on the readings obtained from personal dosimeters of workers in the radiation safety services and in the accident brigades.

Immediately after the beginning of the accident, the population in Pripyat' was recommended to minimize the time spent outside and to keep windows closed. On 26 April all open-air activities were banned at all crèches, kindergartens and schools and in addition, iodine prophylactic treatment was given there. Thus the population, most of whom remained inside during the day on 26 and 27 April was exposed to a level of gamma radiation which was 2-5 times less than that measured in the street. In view of this, there is reason to assume that for the vast majority of the population of Pripyat', the probable dose levels were likely to be 1.5-5.0 rad for gamma radiation exposure and 10-20 rad for beta radiation exposure of the skin.

Consequently, these evaluations show that the possible doses from external radiation exposure to the inhabitants of Pripyat' are significantly lower than those which might cause any immediately changes in their health. A subsequent medical examination of the inhabitants of Pripyat' confirmed this conclusion.

Measurements of the content of iodine isotopes in the thyroid glands of people evacuated from Pripyat' to neighbouring populated areas in the Polessk region showed that in 97% of the 206 examined, the iodine content of the thyroid gland indicated a dose of less than 30 rad. Here the iodine prophylactic measures played a positive role as did the restrictions introduced regarding the consumption of milk from cows for personal use.

The measurements of the iodine content of the thyroid glands of the 20 inhabitants of Pripyat' who were evacuated to Belaya Tserkov', where there was a ban on the consumption of products contaminated by radioactive materials, also give an indication of the possible exposure level resulting from inhalation of iodine. Measurements taken (on 7 May 1986) for the majority of people examined indicated that the thyroid gland dose burden may be 1.5-25 rad.

7.2.3. Exposure of the population in the 30-km zone around the Chernobyl' nuclear power plant

On the basis of the analysis of the radionuclide composition of the radioactive fallout at various points in the 30-km zone around the Chernobyl' nuclear power plant, an estimate was made of the decay dynamics for the dose rate from external gamma radiation from the surface of the earth. This relationship is shown in Fig. 7.2.2 by the continuous line and the dots indicate the actual measured values of the dose rate in relative units. The reasonable agreement between the calculated curve and the data obtained experimentally made it possible to make definite extrapolated estimates both for a long time (up to a year or more) after the accident and for the period during the passage of the cloud of effluent. After appropriate corrections for the actually observed values of the parameters analysed calculations made using a specially developed computer program provided the following relationships between the gamma radiation dose rates at districts on the fifteenth day after the accident ($P\gamma$, 15 mR/h) and the dose for external radiation dose from the radioactive cloud (D_{cl} R), the dose from the radioactive fallout at various times after the accident (D_{fall} P) and also the dose for internal radiation exposure of the thyroid gland in children (D_{tg}, rad) as a result of inhalation and consumption of contaminated cows' milk:

$$D_{cl}\ (10\text{–}30\ \text{km}) = (0.28\text{–}0.07).P\gamma\ 15$$

$$D_{fall}\ (7\ \text{days}) = 0.7.P\gamma\ 15$$

$$D_{fall}\ (1\ \text{month}) = 1.2.P\gamma\ 15$$

$$D_{fall}\ (1\ \text{year}) = 2.5.P\gamma\ 15$$

$$D_{fall}\ (50\ \text{years}) = 8.P\gamma\ 15$$

$$D_{tg}\ (\text{inhal.}) = 10.P\gamma\ 15$$

$$D_{tg}\ (\text{peroral}) = 1000.P\gamma\ 15$$

The last value relates to the case where there is no limitation on the consumption of contaminated cows' milk, which is naturally possible only in zones with very low levels of ^{131}I contamination of vegetation.

In addition to the calculated doses from external radiation exposure caused by the cloud of effluent and internal exposure of the thyroid gland in children as a result of inhalation of iodine, Table 7.2.1 gives a comparison of the doses for external radiation exposure to people in some of the

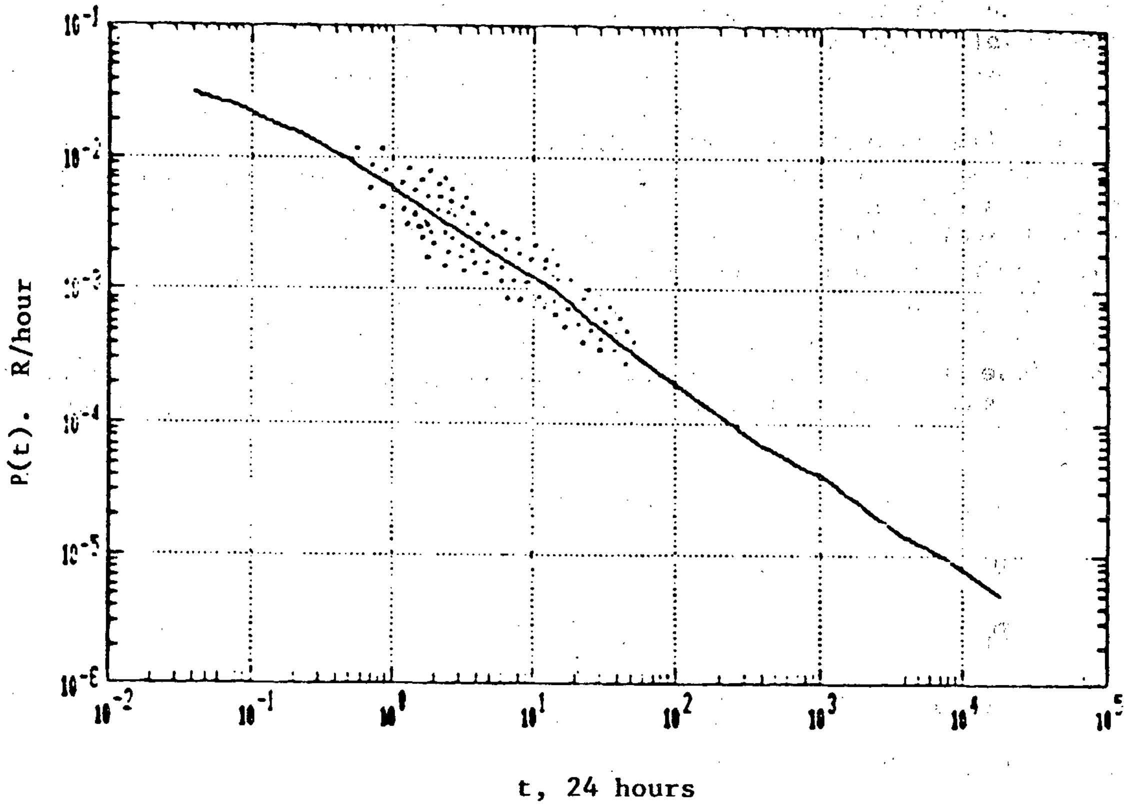


Fig. 7.2.2 Change in gamma dose rate on the path of the radioactive effluent from the Chernobyl, nuclear power plant

Table 7.2.1

Estimated doses to people in some of the populated areas in the 30 km zone around the Chernobyl NPP

POPULATED AREA	DISTANCE FROM CHERNOBYL NPP	DOSE RATE "D" mR/hour	DOSE FROM CLOUD OF EFFLUENT	DOSE TO THYROID GLAND OF CHILDREN, RAD	DOSE FROM FALLOUT OVER A TIME OF 7 DAYS	
					ESTIMATED	MEASURED
CHISTOGOLOVKA	5,5	I2	I0	I20	8,4	3,2
LELEV	9	25	7	250	I7	I0
CHERNOBYL	I6	8	I,2	80	5,6	3,0
RUD'KI	22	8	0,6	80	5,6	2,2
OREVICHI	29	2,5	0,2	25	I,8	4,4

populated areas in the 30-km zone around the Chernobyl' plant calculated using the above relationships and the actual dose values obtained from gamma radiation dose rate measurements (see Fig. 7.2.3).

Analysis of the data in this table shows that the calculated and experimental dose values coincide within a factor of two. This meant that already in the first days after the accident it was possible to make similar estimates for the whole 30-km zone around the Chernobyl' plant based on the available data on the developing radiation conditions. These estimates are given in Table 7.2.2 which contains data for the 71 populated areas in this zone with an indication of the calculated range of external gamma radiation doses in the open.

The fairly wide range (within two orders of magnitude) of changes in the dose for each zone around the Chernobyl' NPP is linked to the considerable unevenness in the radioactive contamination for various sections of the path formed by the effluent released (see Annex 5). On the basis of similar estimates and taking into account the discharge of gases and aerosols from the accident zone which continued during the first few days after the accident, it was decided that it would be wise to proceed to a further evacuation of the population from the area of the accident. In the first few days after the accident, 90 000 people were evacuated from the 30-km zone around the Chernobyl' NPP. Together with the 45 000 people evacuated on 27 April from the town of Pripyat', the total number of those evacuated was 135 000.

This emergency measure made it possible to guarantee that the dose levels of external gamma radiation from the cloud of effluent and radioactive fallout for the vast majority of the population did not exceed 25 rem and only for certain populated areas in the most contaminated parts of the radioactive path (the villages of Tolstyj Les, Kopachi and some others) people may have been exposed to 30-40 rem. However, even for these doses for external radiation exposure there is no danger of acute immediate somatic effects for those exposed. The estimates for the maximum collective dose to the evacuated population (see Table 7.2.3) suggest that the collective exposure dose is 1.6 million man.rems. Taking into account the fact that spontaneous deaths from cancer over a 70 year period for the 114 000 evacuated people may be about 14 000 cases, the natural death rate from cancer among the exposed population will be increased by less than 2% as a result of the the accidental release from the Chernobyl' NPP.

A more exact definition of the radionuclide composition of the contamination of the surface of the earth and the nature of the decay in the gamma-radiation dose rate in the area (see Fig. 7.2.2) will make it possible to introduce subsequent corrections to the expected doses for external radiation exposure to the population and to determine how long it may take before people can return to their place of permanent residence.

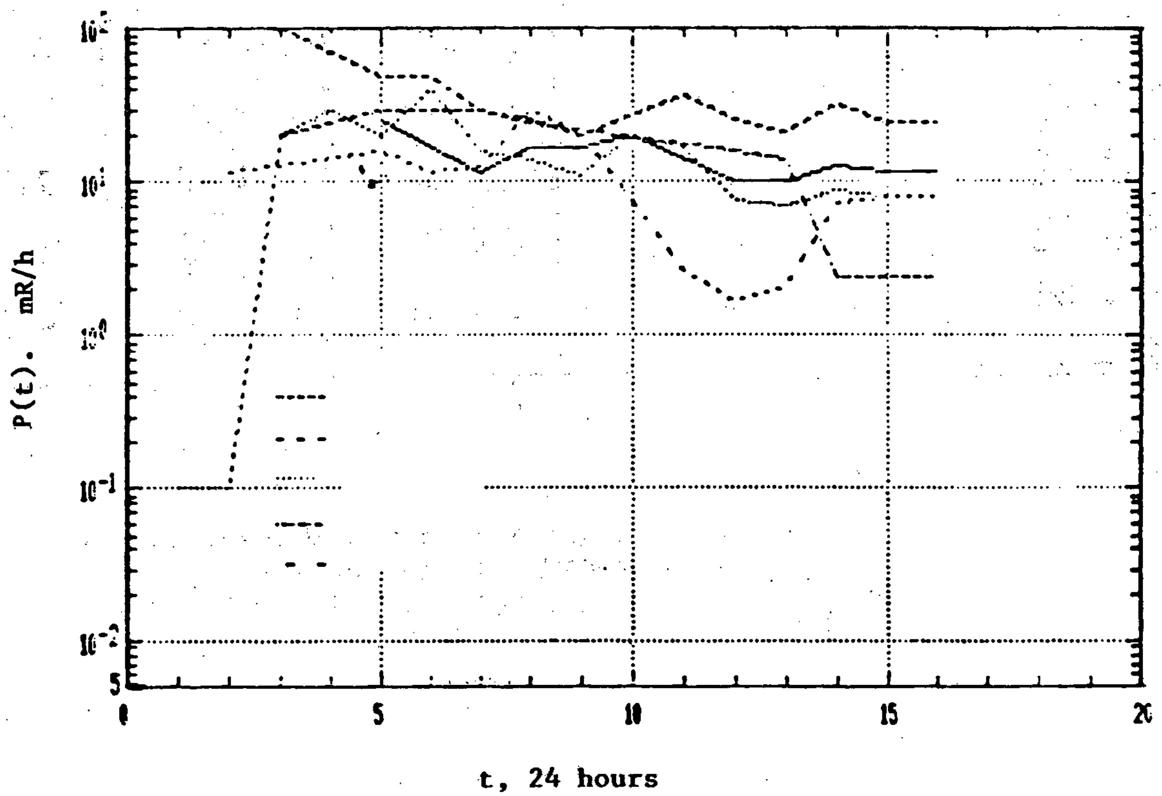


Fig. 7.2.3. Dynamics of the changes in the gamma dose rate for some of the populated areas in the 30-kilometer zone

Table 7.2.2

Calculated doses from external radiation exposure of the rural population* in the 30 km zone around the Chernobyl' nuclear power plant, rem.

Distance from the Chernobyl' nuclear power plant, km	Number of populated areas	Dose from external radiation resulting from fallout for the period:		
		7 days	I month	I 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

x)

*Obtained taking into account the pattern of life of the rural population and the protection coefficients created by rural buildings. For urban conditions these values will be about half the size.

Table 7.2.3

Calculated collective doses from external radiation exposure of the evacuated population.

Area around the Chernobyl' nuclear power plant	Size of population, 1000 people	Collective dose, million man-rem
	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
TOTAL:	I35	I,6

While the stricken reactor continued to release considerable quantities of radioactive products into the atmosphere for a relatively long period (8-10 days), the picture of the contamination of the environment, both in terms of the activity level and in terms of the radionuclide composition, was complicated by the changing meteorological conditions, height and intensity of release. In particular, abnormally high local contamination was observed in different parts of the area. Certain difficulties also arose in establishing a typical radionuclide composition for the radioactive impurities in the air and over the contaminated area. Thus, for example, the iodine-131 content in air and soil samples varied from 8-40% and the caesium from 1-20%. This made it difficult to estimate the possible exposure level to the population resulting from the intake of radioactive products.

Nevertheless, there are fairly good reasons for maintaining that at this stage of evaluation of the dose burdens it is not necessary to take account of the intake of radionuclides by inhalation for people living along the radioactive path. This is confirmed by data which show that the activity of the air in the 30-km zone (Chernobyl', Zorin, Skazochnyj pioneer camp, and the town of Pripyat') from 3 May to 3 June was 10^{-2}-10^{-4} Ci/L for the total beta-activity of the radionuclides. Table 7.2.4 gives an example of the relative contribution of gamma-active radionuclides in the air samples taken from different populated areas around the Chernobyl' NPP.

Thus, for the population living in the contaminated area and consuming locally produced products, the main source of internal exposure consists of the radioactive substances contained in these products. The calculations certainly have to include radioactive substances which were inhaled during passage of the cloud. However, as will be shown below, the doses for internal exposure resulting from this are significantly less than those resulting from consumption of contaminated products.

In the first stage after the accident (about 2 months) the main radionuclide component of the dose was iodine entering the organism mainly through milk from grazing dairy cattle and the critical organ receiving the maximum dose burden was the thyroid gland.

These circumstances predetermined the scale and type of dosimetric and medical examination of the population. The staff and resources of brigades from the specialized institutes of the USSR Ministry of Health were used to examine the iodine content of the thyroid gland for the population evacuated from the 30-km zone, and also for the inhabitants of a number of populated areas in the Ukrainian Soviet Socialist Republic (SSR), the Byelorussian Soviet Socialist Republic (SSR) and the Russian Soviet Federative Socialist Republic (RSFSR) where high radiation levels were recorded but where evacuation was not necessary. Special attention was given to children which belong to the group of high radiation risk, because the accumulated radiation

Table 7.2.4

Relative content of the gamma-ray-emitting radionuclides in aerosol samples taken from the air

Date of sampling	Place of sampling	Total β activity of sample Ci/L	Relative radionuclide content %											
			I^{131}	Cs^{134}	Cs^{137}	Ba^{140}	La^{140}	Ce^{141}	Ce^{144}	Zr^{95}	Nb^{95}	Ru^{103}	Ru^{106}	Te^{132}
I4.05	Zorin	$1{,}5 \cdot 10^{-13}$	20,0	-	4,0	-	5,0	I2,0	-	8,0	20,0	30,0	-	I,0
3.06	Town of Pripyat'	$4 \cdot 10^{-10}$	-	I,2	0,8	I,I	2,5	II,2	I2,I	I9,2	26,I	I9,I	6,7	-
3.06	Skazochnyj pioneer camp	$6 \cdot 10^{-13}$	0,46	3,9	7,7	I8,8	27,9	3,5	3,2	I7,0	2,0	4,9	4,6	-
3.06	Chernobyl'	$4{,}0 \cdot 10^{-12}$	5,9	0,8	I,6	3,7	5,0	II,2	9,0	I8,6	22,9	II,7	9,6	-

Note: A dash indicates that the radionuclide was not detected in that particular sample.

dose in the thyroid gland for the same radioiodine content is 8-2 times greater in children from the age of 2-24 respectively than in adults. In addition, it should be noted that children consume great quantities of milk products.

Where possible, together with the examination of the iodine content in the thyroid gland, there was selective determination of the content of other radionuclides in the organism from excretion activity (urine, faeces) and also the possible intake into the organism of various radionuclides was estimated from data about the isotopic composition of the soil and food contamination. In the period following the accident, the radioiodine content of the thyroid gland was determined directly for a large number of inhabitants including almost 100 000 children. Almost all the children (up to 15 years old) and part of the adult population evacuated from the 30-km zone and populated areas on the path of the radioactive cloud where high radiation levels were recorded were examined. It should be noted that the above population consumed locally produced products prior to the evacuation (4-5 May) for 9-10 days, including milk and milk products, the proportion of which is significant in this region.

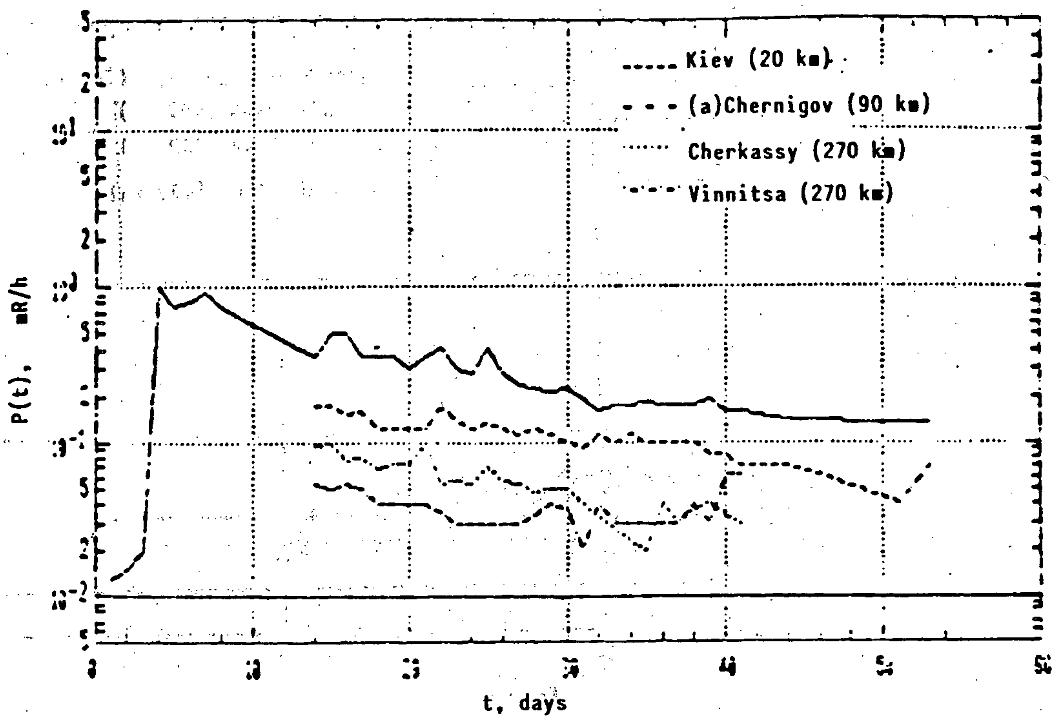
The measurements showed that for the vast majority of people evacuated from the 30-km zone, the thyroid gland dose burdens as a result of intake of radioactive substances by consumption of locally produced products is significantly lower than those which could cause any change in their health.

The relatively high thyroid gland dose burdens observed sometimes is evidently the result of the uncontrolled consumption of milk from cows for personal use despite the ban issued by the health authorities on 1 May 1986 on the consumption of whole milk with a radioiodine concentration higher than 1.10^{-7} Ci/L. This requirement was strictly adhered to within the centralized milk supply. Furthermore, additional measures were taken for strict monitoring of the sale and consumption of milk from cows for personal use.

For prophylactic purposes all the children from the 30-km centralized evacuation zone were sent to summer sanitoriums in the country. There is constant medical observation of children for whom the estimated thyroid gland exposure dose prior to complete clearance of iodine isotopes may exceed 30 rem.

7.2.4. Radiation consequences of the accident at the Chernobyl' nuclear power plant for the population of different regions in the European part of the USSR

As was indicated in the previous sections of the report, the radioactive releases resulting from the accident at the Chernobyl' nuclear power plant affected radiation conditions not only near the plant but also conditions considerable distances from it. Figures 7.2.4 and 7.2.5 show the



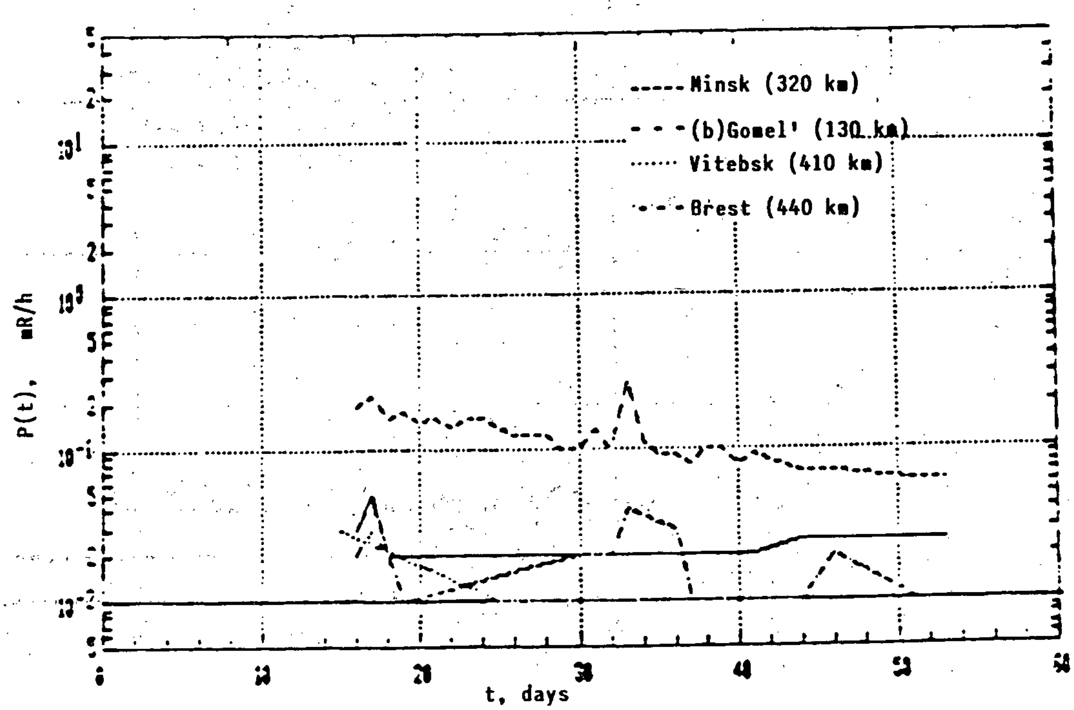
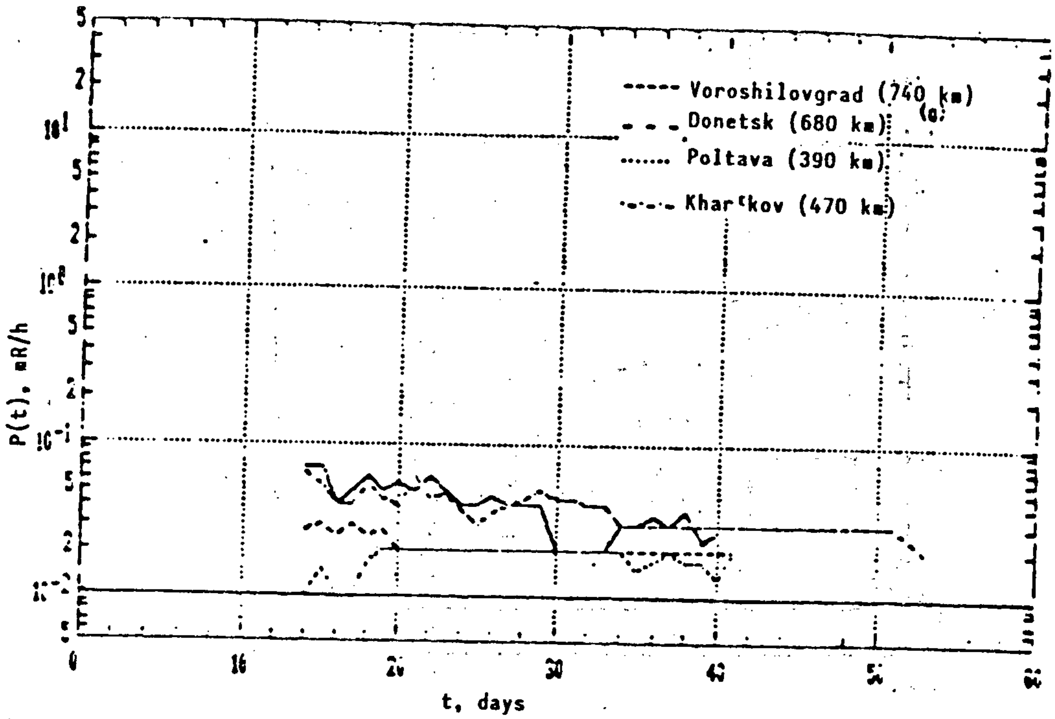


Fig. 7.2.4 Dynamics of the changes in the gamma dose rates in the open for regional centres in the Ukrainian SSR (a) and the Byelorussian SSR, near the Chernobyl' nuclear power plant.



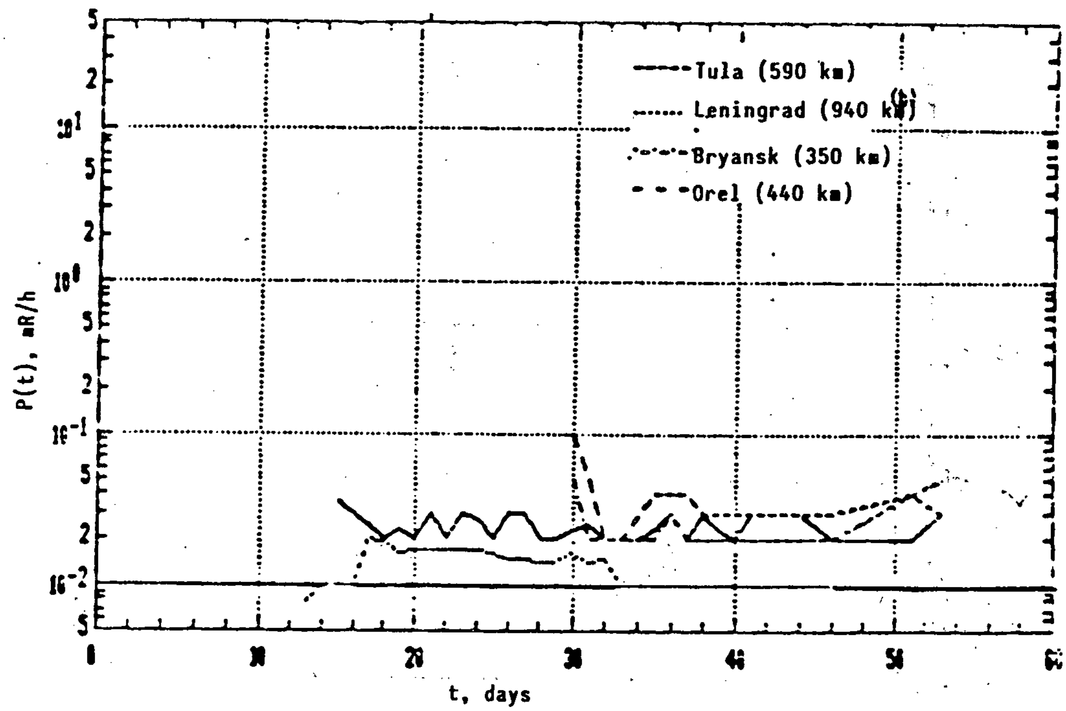


Fig. 72.5. Dynamics of the changes in the gamma dose rates in the open for some regional centres in the Ukrainian SSR (a) and the Russian SFSR (b).

change during the course of time in the gamma radiation dose rate in the open at some regional centres of the Ukrainian SSR , Byelorussian SSR and the RSFSR at distances of 100-1000 km from the Chernobyl' plant. The diagrams show that at virtually all these populated areas, the dose rates for external gamma radiation was several times greater than the natural background radiation characteristic for that area of the European part of the USSR (8–12 μ R/hour – the thick lines on the diagrams). By averaging the numerous results of gamma radiation dose rate measurements for areas within the regional administrative boundaries it was possible to identify the ten (see table 7.2.5) which had the maximum levels of radiation caused by the products released to the population during the accident.

The data in table 7.2.5 show that the regional averages values of external exposure for 1986 do not exceed (taking into account the population's way of life) 1.5 rem and for 50 years - 5 rem. Thus there is no danger to health for the population living outside the 30-kilometre zone around the Chernobyl' plant resulting from the levels of external gamma radiation of the products released during the accident. A more complex situation arises when estimating doses of internal exposure resulting from the intake of radionuclides through consumption of contaminated locally produced food products.

Before the accident at the Chernobyl' plant, in the Soviet Union, as in other countries, regulations were established only for the permissible annual intake of radioactive substances in food products. The permissible concentration of nuclides in drinking water had also been established (Radiation Safety Standard-76). There were no regulations governing the content of nuclides in different forms of food products. In the event of an accident, norms had been established only for the critical product (cows' milk) and the most important nuclide in an accident – iodine-131). In the radiation conditions prior to the accident, the content in all types of food products, strontium-90, caesium-137 and even more in the case of other nuclides, was many times lower than the established annual limits on intake of nuclides through food consumption. These standards were based on the principle that groups of the population receiving the greatest exposure should not receive a dose of more than 0.5 rem per year and for critical organs of the second group (including in particular the thyroid gland) – 1.5 rem. It was established that these exposure doses should not be exceeded for any combination of radiation action - i.e. both external exposure and internal exposure (inhalation, intake through water and food). These standards were worked out on the principle unconditional and complete prevention of the immediate specific consequences of exposure (radiation sickness, cataract, radiation burn, injury to the haemopoietic system, lowering of the immunoreaction). Moreover these norms which are established considerably below the levels capable of causing the reactions mentioned above, are based on the need to limit the risk of late radiation effects occurring – cancer and genetic disturbances. The Radiation Safety Standards-76 for a restricted part

Table 7.2.5

Radiation levels and predicted doses from external radiation exposure to the population in the 10 regions which had the highest radioactive contamination caused by the products released from the Chernobyl' NPP

Regions	Regional average dose rate 15 days after the accident mR/hour	Dose to the population in 1986 rem		Dose over 50 years rem	
		rural	urban	rural	urban
Gomel'skaya	0,83x)	I,39	0,74	4,7	2,5
Kievskaya	0,44x)	0,74	0,40	2,5	I,4
Bryanskaya	0,30	0,50	0,27	I,7	0,92
Zhitomirskaya	0,20	0,34	0,I8	I,2	0,63
Mogilevskaya	0,I5	0,25	0,I4	0,86	0,46
Orlovskaya	0,I4	0,24	0,I3	0,8I	0,44
Chernigovskaya	0,I4	0,28	0,I2	0,78	0,42
Tul'skaya	0,I2	0,20	0,II	0,67	0,37
Cherkasskaya	0,09I	0,I5	0,082	0,52	0,28
Brestskaya	0,08I	0,I4	0,073	0,46	0,25

x) Outside the 30-km zone around the Chernobyl' NPP.

of the population (0.5 rem for the whole body) correspond to the upper boundary of probability for the incidence of cancer which is 50-500 additional cases per million population per year. The actual levels of exposure of the population before the Chernobyl' accident from water and food products were tens and hundreds of time lower than the regulatory levels.

After the accident it became necessary to find effective solutions to problems associated with sorting and prohibiting the consumption of specific types of food products. Since initially the main danger was that of iodine-131 incorporated into the human organism during the spring-summer period primarily through milk and also through leafy green vegetables, standards were introduced immediately after the accident governing the permissible content of iodine-131 in milk and milk products (cottage cheese, sour cream, cheese, butter) and also in leafy green vegetables. The standards were designed to ensure that thyroid gland exposure of children (the critical organ for iodine-131) did not exceed 30 rem. This condition was observed by establishing the permissible content of iodine-131 in milk at 1.10^{-7} Ci/L. A similar standard was introduced in England in 1957 after the Windscale accident. In addition, standards were introduced governing the iodine-131 content in meat, poultry, eggs, berries, and raw materials used for medical purposes. During the second half of May, data were obtained which indicated that as the iodine-131 decayed, caesium-137 and caesium-134 were playing an increasing role in the contamination of meat and a number of other types of foodstuffs and also indicated the presence in food products of rare earth isotopes - cerium-144, ruthenium-106, zirconium-95, barium-140, lanthanum-140, cerium-141, ruthenium-103, niobium-95. The latter were found in significant quantities together with caesium in green vegetables ($1\ 10^{-6}$ Ci/kg or more). At many places during May the concentrations of iodine in milk products remained high. During this period in order to carry out wide-scale monitoring and sorting of food products, it was necessary to establish standards which would make it possible to carry out monitoring using the simplest equipment, in other words, standards governing the total beta-activity content. These standards were issued by the USSR Ministry of Health on 30 May 1986. They were a continuation of the earlier standards of the 8 and 12 May but contained a wider selection of products and reflected the changes in the radiation conditions established at the end of May. The permissible whole body and internal organ exposure dose on which these standards are based is 5 rem.

In the first days and weeks after the accident, the basic activity of food products was accounted for by iodine-131. It appeared in the milk of cows kept out at pasture two-three days after the accident. The iodine levels in the south of Byelorussia were established at 10^{-6} Ci/L. The milk of cows which were kept inside remained much freer. The same and even higher contamination levels - up to 10^{-5} Ci/kg were recorded in green vegetables.

As is well-known, during migration of radionuclides - from the first link in the chain, i.e. the fallout and soil to the last one - the human organism - separation occurs together with a reduction in the content of some nuclides and an accumulation of others as a function of the physical and chemical properties of the nuclide and a number of other factors (soil composition, quantity and time of fall of atmospheric precipitation, the composition of the diet of agricultural animals and so on). Therefore, the fullest composition of nuclides may be recorded in food products contaminated from the surface, i.e. direct sorbing nuclides contained in the atmosphere and settling from the air. These products include lettuce, fennel, kinza, tea, and so on. Table 7.2.6 shows, as an example, the nuclide composition found in surface-contaminated representative vegetation from regions close to the Chernobyl' plant.

On the other hand, in a number of other products where the radioactive substances face biological barriers the radionuclide composition is considerabaly limited. Thus at the beginning and in the middle of May only caesium and iodine-131 isotopes were found in meat and at the end of May and in June almost only caesium-137 and caesium-134 (in a 2:1 ratio) were detected. However, the content of radioactive caesium in meat (beef) was fairly high - 10^{-8} - 10^{-7} Ci/kg.

The data in table 7.2.7 show the amounts by which the levels in food exceeded the standards prevailing in May.

As has already been indicated above, monitoring of the iodine-131 contamination of milk showed that in many regions of the Ukrainian SSR, Byelorussian SSR and the RSFSR in the period May-June 1986, the concentration of this nuclide in milk exceeded the established standards (0.1 μ Ci/L). By analysing the contamination levels of the soil, vegetation and milk samples and linking them with the gamma radiation dose rate in the area, it was possible to estimate the possible concentrations of iodine-131 in milk in various regions of the country. An example of such an estimate is given in Table 7.2.8 where estimated iodine-131 contamination levels of milk samples are given for 10 regions based on the actual or regional average external gamma radiation dose rates alongside the values actually observed in May 1986. Similarly, information was prepared for other regions in the European part of the USSR with a population of about 75 million (see Fig. 7.2.6). In this drawing and in the table given below, 11 regions are selected (4 in the Ukrainian SSR, 2 in Byelorussian SSR and 5 in the RSFSR) which are interesting either because they have fairly high radioactive contamination levels or because they have large populations.

The estimate of the radiation consequences of external gamma radiation for the population of these regions resulting from the radionuclides which fell in the area is given in Table 7.2.9. The Table shows that the expected average dose values for the external exposure for each region in 1986 is

Table 7.2.6

Radionuclide content in vegetation near the Chernobyl' NPP

Type	Place of sampling	Date of sampling	Nuclide	Content Ci/kg
Clover	Chernobyl'	26	Cerium -I44	$2 \cdot IO^{-6}$
			Cerium -I4I	$I,4 \cdot IO^{-6}$
			Iodine-131	$I,3 \cdot IO^{-6}$
			Ruthenium-103	$I,2 \cdot IO^{-6}$
			Ruthenium-106	$7,9 . IO^{-7}$
			Barium-140	$6,7 \cdot IO^{-7}$
			Caesium-134	$3,2 \cdot IO^{-7}$
			Caesium-137	$2,5 \cdot IO^{-7}$
			Zirconium-95	$I,5 \cdot IO^{-6}$
			Mobium-95	$2,0 \cdot IO^{-6}$
			Lanthanum-140	$5,3 \cdot IO^{-7}$

Table 7.2.7

Agricultural products in which a higher than permissible level of radioactive contamination was detected

Republic	Region	Product and the proportion (%) which did not corresponds to the norms					
		Meat	Milk & milk products	Green vegetables	Root vege-tables	Berries	Fish
Byelorussian SSR	Minskaya	I0	5	-	-	-	-
	Gomel'skaya	40	30	I5	I0	5	90
	Brestskaya	I0	50	5	3	5	-
	Mogilevskaya	20	I0	-	-	-	-
	Godnenskaya	-	5	-	-	-	-
RSFSR	Tul'skaya	-	I5	-	-	-	-
	Bryanskaya	-	30	-	-	-	-
	Kaluzhskaya	-	20	-	-	-	-
	Kurskaya	-	30	-	-	-	-
	Orlovskaya	-	I0	-	-	-	-
Ukrainian SSR	Kievskaya	-	I0	20	-	20	-

Note: a dash indicates that data are not available.

generally lower than the annual dose limit for category B workers (restricted group of the population) in accordance with regulation Radiation Safety Standard-76. The collective dose for this population group is 8.6 million man.rem in 1986 and 29 million man.rem for a period of 50 years after the accident. For the purposes of comparison it should be noted that the annual collective dose from natural background radiation for the same number of people is 10 mllion man.rem, i.e. it is comparable with the annual dose resulting from the accidental release for 1986. Over a period of 50 years, the natural background radiation dose will be almost 15 times greater than the corresponding collective dose from the Chernobyl' accident. In this connection, the calculated increase in the mortality rate, based on the non-threshold hypothesis of the dose-effect relationship, is less than 0.05% by comparison with the mortality rate from spontaneous cancer (about 9.5 million cases after 70 years) for the same population.

The risk of mortality and the estimate of the number of cases of curable forms of cancer and benign tumours of the thyroid gland among people who have consumed iodine-131 contaminated milk was calculated on the basis of the following data:

- The actual concentration of iodine-131 in cows' milk or its estimated content based on the external gamma radiation dose rate for the area;

- The size of the population and its composition in terms of sex and age;

- The age dependance for the consumption of milk, the dose coefficients and the risk of mortality coefficients, and the risk of curable diseases of the thyroid gland based on data of the ICRP, UNSCEAR and Soviet publications.

General data on the iodine-131 concentration levels in cows' milk for the regions examined are shown in Fig. 7.2.7. The diagram shows that in a number of regions of Ukrainian SSR, Bylorussian SSR and RSFSR, the iodine-131 concentration in different samples exceed the established standard (the thick line in the diagram) by 20-100 times or even more. Since milk sold centrally had a concentration not greater than 0.1 μCi/L it was assumed that the urban population of these regions and most of the rural population consumed this type of milk. For the remaining small group of the rural population with their own dairy cattle it was assumed that in some cases the 100% ban on the consumption of milk with a concentration of iodine-131 higher than the permissible level was not fully applied. This assumption means that in a number of the regions which were more heavily contaminated with iodine-131, the maximum calculated doses for internal exposure of the thyroid gland could have reached hundreds of rads.

Table 7.2.8

Region	Calculated levels	Actual measurements
Gomel'skaya	0,2-I4	0,02-I0
Kievskaya	0,06-7,3	
Bryanskaya	0,04-5,0	0,02-I,3
Zhitomuskaya	0,03-3,3	
Mogilevskaya	0,02-2,5	0,02-2,0
Orlovskaya	0,02-2,3	0,0I-0,8
Chernigovskaya	0,02-2,3	
Tul'skaya	0,02-2,0	0,06-6;5
Cherkasskaya	0,0I-I,5	
Biestovskaya	0,0I-I,3	0,2-9,0

Comparison of calculated and actually observed iodine-131 contamination levels in milk in May 1986 in the 10 regions which had the highest radioactive contamination caused by products released from the Chernobyl' NPP µCi/L

Fig. 7.2.6. Population in different regions of the European part of the USSR (in millions of people)

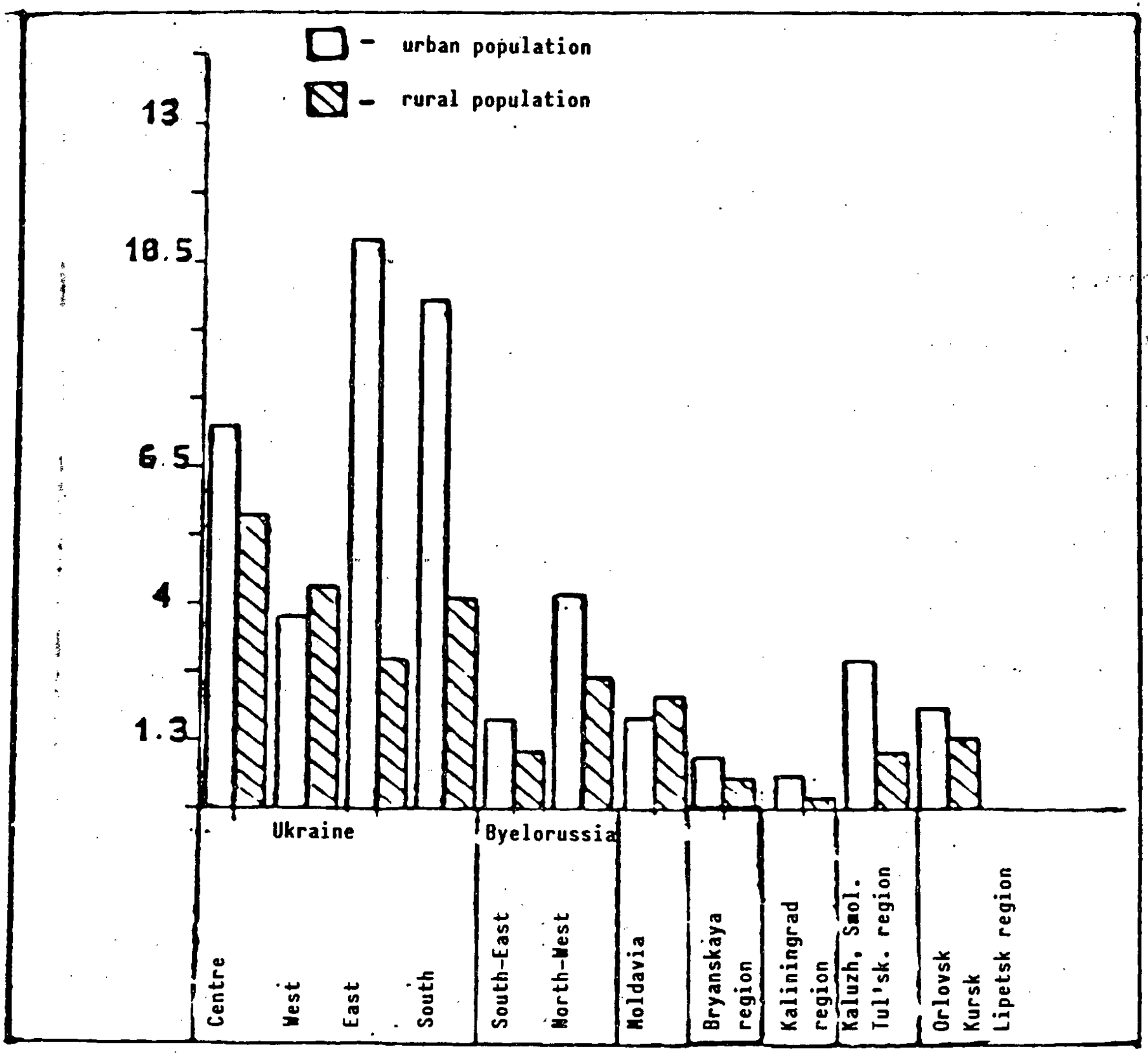


Table 7.2.9

Expected doses from external radiation exposure to the population in different regions of the European part of the USSR

Region	Population, millions of people	Dose for 1986 rem/year		Collective dose 10^6 man-rem	
		rural	urban	for 1986	over 50 years
Central part Ukrainian SSR	I3,6	0,27	0,I5	2,75	9,3I
West. part Ukrainian SSR	8,3	0,067	0,036	0,44	I,47
East. part Ukrainian SSR	I4,5	0,077	0,04I	0,75	2,52
South. part Ukrainian SSR	I4,4	0,045	0,024	0,73	2,47
South-East Byelorussian SSR	2,9	0,98	0,52	2,05	6,94
North-West Byelorussian SSR	7,0	0,094	0,050	0,47	I,58
Moldauskaya SSR	4,I	0,084	0,045	0,27	0,92
Bryanskaya reg.	I,5	0,50	0,27	0,44	I,49
Kaliningrad. reg.	0,8	0,0I2	0,003	0,006	0,02
Kal. Tul'sk, Smotensk reg.	4,0	0,I2	0,064	0,32	I,08
Orl. Kursk. Lipetsk. reg.	3,4	0,I4	0,075	0,35	I,I7
Total:	74,5	-	-	8,6	29,0

The comparison, based on the individual and collective dose estimates, between the risk of mortality as a result of exposure of the thyroid gland during a 30 year period after intake of iodine-131 and the risk of mortality from spontaneous cases of cancer of the thyroid gland during the same period (about 150 000 cases) showed that the increased mortality from iodine-131 is about 1% and hardly increases the mortality indices in the regions examined.

These estimates are based on the hypothesis of a non-threshold linear "dose-effect" relationship which is accepted in most countries. This hypothesis is based on theoretical representations of the carcinogenic mechanism, accumulated data on the dose-effect relationship in a region of high radioation doses and also the principle of "deciding in favour of man" i.e. deliberately ensuring his safety in a region of low radiation doses. It is significant that in the main ICRP publication on the problems of radiation safety (ICRP 26 para. 30) it is stated that: "The use of linear extrapolations from the frequency of effects observed at high doses may suffice to assess an upper limit of risk ... However, the more cautious such an assumption of linearity is, the more important it becomes to recognize that it may lead to an overestimate of the radiation risks ..." Thus the values given in this part of the report should be regarded as "upper" estimates of the radiation consequences for the population of the European part of the USSR as a result of the activity released during the accident at the Chernobyl' plant.

In addition to iodine-131, it will be necessary during this year and particularly in the following years, to pay attention to other radionuclides contaminating locally produced food products and water supplies. The possible contamination levels of foodstuffs in the short-term and in the long-term should be examined separately for the main nuclides and types of food products.

Ruthenium-106, caesium-144 and other rare-earth nuclides make up a significant contribution only to products susceptible to surface contamination (green vegetables, other vegetables, and to a lesser extent - berries, mushrooms and honey), since they are hardly not assimilated from the soil by plants or from plants by animals. The biological significance of this group of radio- nuclides in such products does not exceed 10-20%. Subsequently, the contribution of rare-earth elements to the contamination of foodstuffs will be reduced significantly everywhere and will not have any practical value.

Caesium-137 and caesium-134 are the main biologically significant radio- nuclides which (excluding strontium-90) since the middle of June have been the main contaminants of meat, milk, vegetables and other products. Until now they have been incorporated into vegetable and animal products from the air. Incorporation from the soil has practically no significance. The grain and potato harvest in the autumn of 1986 will probably be relatively clean - not much caesium will already have been incorporated from the air and

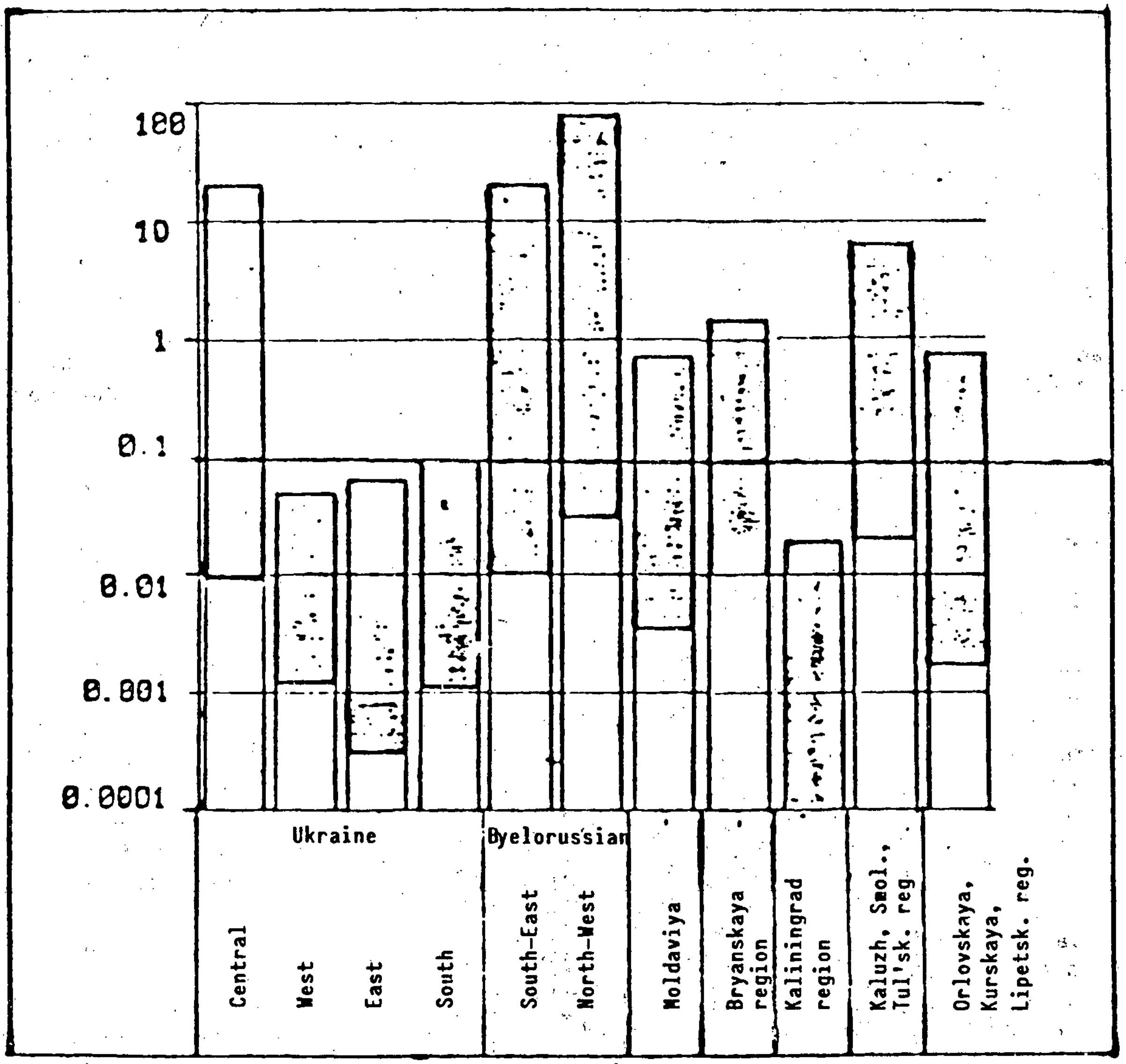


Fig. 7.2.7. Concentration of ^{131}I in cows' milk (in µCi/L)

incorporation from the soil will not yet have played a role. The contamination of products by caesium in the next few years will be significantly different for the air around the Chernobyl' plant where the type of soil is different. Since caesium is incorporated from the Poles'ye soil which is poor humus, into plants at a rate which 10 or even 100 times greater than from other types of soil, the caesium-137 content in food in the Poles'ye region is likely to be relatively persistent and high in the next few years. In other regions, in particular in the north of Byelorussia, in the west of the RSFSR, in the south, north, east and west of the Ukraine, there are reasons to expect a sharp fall, of 10 or more times in the caesium concentration in food products.

The preliminary purely rough estimates of the contamination of food products by caesium isotopes are as follows. For a density of caesium-137 radioactive fallout on the surface of the earth of 1 Ci/km^2 and consumption by the population of locally produced food products, the doses of individual whole body exposure resulting from peroral intake of activity in the Ukrainian and Byelorussian regions of Poles'ye are (taking into account the additional exposure of the organism from Cs-134) 0.7, 0.34 and 3.3 rems for the first, second and 70 years respectively. In this case the collective doses, taking into account agricultural production for 1 km^2 in these regions of the Ukrainian SSR and Byelorussian SSR for the same periods are 120, 58 and 570 man.rems. For the rest of the country, with significantly lower coefficients for caesium transfer from the soil into agricultural products, the corresponding collective doses for the population are 120, 36 and 170 man.rems. If we consider that the total quantity of caesium-137 which was released into the atmosphere and settled on the earth's surface after the Chernobyl' accident is estimated at $1.0 \cdot 10^6$ Ci (see Sections 4 and 5) and taking into account the fact that about 10% of the caesium isotopes released fell over the Ukrainian and Byelorussian Poles'ye area, the collective dose for the population for a period of 70 years after the accident is $2.1 \cdot 10^8$ man.rems. This may lead to an increase in the cancer mortality rate not exceeding 0.4% of the natural mortality rate from malignant neoplasms. When the actual coefficients for the transfer of caesium isotopes to the food chain have been clarified for the specific soil conditions of the contaminated regions, it may be possible to make corrections to these estimates which would to reduce them.

With regard to the strontium-90 content in foodstuffs, the data so far obtained have been only isolated and not sufficient to make any corrections to the estimates. With time, it is possible that this nuclide will be of basic significance together with caesium-137. On the whole, the strontium-90 content in food products from the 1987 harvest will obviously be considerably reduced. In the Poles'ye soil region, the reduction in the role of strontium-90 by comparison with caesium-137 will be more significant.

However, it will be possible to forecast the levels of the strontium-90 content in food in the next few years only after a fuller study of the strontium-90 content in the soils of the contaminated areas of the USFSR, the Ukrainian SSR and Byelorussian SSR.

Thus, an accurate evaluation of the dose burdens for the population resulting from the consumption of locally produced food products contaminated with caesium-137 and strontium-90 will be possible only after the actual transfer coefficients of radionuclides along the food chains have been established for these regions. This work being carried out by the country's various scientific subdivisions will enable recommendations to be made regarding the optimum methods of agricultural management in the zone of radioactive contamination in terms of the formation of dose burdens for the population.

7.3. Organization of medical examinations of the population from the regions around the Chernobyl' plant

Following the accident, 84 000 people including 18 350 children were evacuated from the town of Pripyat' and the Chernobyl' region. In addition there was a further evacuation from populated areas in the Kievskaya and Zhitomirskaya regions.

In order to provide medical care for those evacuated during the first few days after the accident, 450 brigades made up of doctors, nurses, assistants and health physicists were mobilized and provided with ambulances. In all (taking into account rotations related to the radiation conditions) 1240 doctors, 920 nurses, 360 doctors' assistants and 2720 assistants with secondary school education, 720 students from medical institutes of higher education and also a large group of members of Scientific Research Institutes.

After personnel decontamination all those evacuated were examined by doctors and compulsory dosimetric monitoring and laboratory blood tests was carried out. Where necessary, the examination was repeated.

All those evacuated from the 30-kilometre zone who showed any irregularities in their health requiring further examination were hospitalized in special sections set up at central regional hospitals.

In order to provide medical care for workers involved in eliminating the consequences of the accident, a polyclinic with four 24-hour first-aid brigades was set up in Chernobyl' based on the central regional hospital.

Special attention was given to the examination of children from the 30-kilometre zone and also to selected children living in populated areas close to the 30-kilometre zone (a total of about 100 000 children were examined).

7.4. Long-term programmes for the medical and biological monitoring of the population and personnel

Long-term programmes are being established for the medical and biological monitoring of the population and personnel.

The measures taken to provide medical care for those who have een exposed to radiation resulting from the accident at the Chernobyl' plant include: the establishment of a register of all those exposed to radiation; the grouping of those exposed in order to determine the volume of medical care required; measures to organize and provide the necessary volume of medical care.

The purpose of the register is to study the possible consequences of the radiation effects on all those who were exposed and to ensure purposeful medical surveillance appropriate to the expected effect resulting from the range of doses involved.

The effects of low dose total external exposure will be analysed in terms of the stochastic effects (incidence of infectious diseases, incidence of and deaths from malignant tumours, birth-rate, state of health of children who are born), and the neuro-psychological aspects of the reaction to the situation.

A special study will be made of the functioning of the thyroid gland and over extended periods the frequency of development of adinomas and malignant neoplasms.

The whole study will be based on the dynamic characteristics of the background level of the above parameters in the regions where those examined originally lived and also in the places of evacuation. The volume of examinations will be determined on the basis of international and national recommendations regarding the possible biological effects (UNSCEAR, ICRP and others).

The criteria used in examining the general state of health of those involved will be based on the data from therapeutical examinations and the detailed clinical blood analyses. All women will be examined by a gynaecologist and the children by a paediatrist who will provide data on their physical development in accordance with Soviet regulations. In dose ranges presenting even minimal risk of dysfunction of the thyroid gland this examination will be complemented by special dynamic observation of the endocrinology of the thyroid gland using hormones - thyroxine, triiodothyroxine, thyroid stimulating hormone and others.

The increased frequency of the risk of mortality from cancer as a result of the radiation effects (from fractions of a percent to several percent) can be estimated only if data are available for a very large population.

In view of this, the register will include all those who were residing or staying in the area, the organized groups involved in dealing with the accident and its consequences, their children and grandchildren and also those evacuated from the contaminated regions.

The register consists of registration and dosimetric charts for each of those examined.

The registration chart includes the following information: surname, forenames, passport details (passport series and number or birth certificate, date of issue of the document) date of birth, place of birth, sex, nationality, place of residence, whereabouts during the period of the radiation action, duration of the action, anamnestic information about the state of health, pregnancy at the beginning of the action (period in weeks) and occurring after the beginning of the action, data about the child, cause of death (adults, children, babies) measures taken (hospitalization, iodine prophylactic treatment).

The dosimetric chart records the sanitary characteristics of the region and the degree of radiation effect on the individual (contamination of clothing, shoes, skin before and after decontamination in μR/hour).

The chart includes data on the iodine-131 content of the thyroid gland which is a dosimetric parameter for clincial investigation of those being examined and also data on personal dosimetry (measurement of biosubstrates using whole-body counters and other instruments).

Local health authorities are completing the registration and dosimetric charts. The completed charts are sent to the Ministry of Health of the Republic concerned and to the Ministry of Health of the USSR. Apart from the registration charts, all the infomation is entered in a registration journal which is kept at the place of examination.

The grouping of people who were exposed (or who may have been exposed) as a result of the accident at the Chernobyl' plant on 26 April 1986 requiring appropriate medical care is based on the principle of differentiation in terms of the whole-body dose and the dose to the thyroid gland. All the dose gradients are given for adults and these doses should be reduced by a factor 10, for children up to the age of three and for pregnant women.

The frequency of examinations is based on the results of the first examination and the estimate of dose level. The prophylactic treatment and protection measures are taken into account (iodine prophylactic treatment, evacuation, limitation of intake of radioactive substances through the respiratory and the alimentary tracts).

The above examinations are in addition to the medical surveillance provided for the whole Soviet population.

The programme envisages increasing the number of experts in various professional disciplines. An estimate is being made of the time spent, the technical equipment, the algorithm and software support and the computer work involved in carrying out these measures. The clincial data will be interpreted in the light of material on the dynamics of the environmental contamination, characteristics of the isotopic composition and the iodine prophylactic treatment administered. It is planned to set up simulation models and survey prognoses of the expected variants of late stochastic consequences (oncological effects, genetic influences) for the next 30 years and for a lifetime (50 years).

A collection and accumulation of information about the oncological and genetic aspects of morbidity among the population maybe established to be used in the event of a similar accident situation.

The prepared programmes take account of the experience of other countries (the Three Mile Island programme; the IAEA meeting in Yugoslavia and so on).

/6091
CSO: 5100/4501 END