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USSR STATE COMMITTEE ON THE UTILIZATION OF ATOMIC ENERGY

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

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PART I. GENERAL MATERIAL

DRAFT

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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:

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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^{-6} vol.% steam
Fast power coefficient of reactivity α_W at working point	-0.5×10^{-6}/MW
Temperature coefficient of fuel α_T............	$-1.2 \times 10^{-5}/^oC$
Temperature coefficient of graphite α_C........	$6 \times 10^{-5}/^oC$
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 rundown. 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