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INSAG-7
The Chernobyl Accident: Updating of INSAG-1

A REPORT BY THE
INTERNATIONAL NUCLEAR SAFETY ADVISORY GROUP



INTERNATIONAL ATOMIC ENERGY AGENCY, VIENNA, 1992

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THE CHERNOBYL ACCIDENT: UPDATING OF INSAG-1
INSAG-7

A report by the International Nuclear Safety Advisory Group

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SAFETY SERIES No. 75-INSAG-7

THE CHERNOBYL ACCIDENT: UPDATING OF INSAG-1

INSAG-7

A report by the
International Nuclear Safety Advisory Group

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 1992

The International Nuclear Safety Advisory Group (INSAG) is an advisory group to the Director General of the International Atomic Energy Agency, whose main functions are:

(1) To provide a forum for the exchange of information on generic nuclear safety issues of international significance;
(2) To identify important current nuclear safety issues and to draw conclusions on the basis of the results of nuclear safety activities within the IAEA and of other information;
(3) To give advice on nuclear safety issues in which an exchange of information and/or additional efforts may be required;
(4) To formulate, where possible, commonly shared safety concepts.

THIS SAFETY SERIES IS ALSO PUBLISHED IN FRENCH, RUSSIAN AND SPANISH

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FOREWORD

by the Director General

New information that has come to light since the Post-Accident Review Meeting on the Chernobyl Accident (held in Vienna from 25 to 29 August 1986) bears on general issues of the operational safety of nuclear power plants in the then Union of Soviet Socialist Republics as well as on specific issues relating to the design of the Chernobyl type light water cooled, graphite moderated RBMK reactors.

As regards general issues, the new information demonstrates the lack of feedback of operating experience and the inadequacy of communication between designers, engineers, manufacturers, constructors, operators and regulators. These deficiencies, coupled with a lack of clear lines of responsibility, were critical factors in the events leading up to the Chernobyl accident. These shortcomings have been tackled at the national level and some improvements have been made.

Valuable lessons can be learned from incidents and accidents, as was demonstrated after the accident at Three Mile Island nuclear power plant in the United States of America in 1979, when far reaching follow-up actions were taken to minimize the risk of a recurrence and to improve procedures for accident management. The accident at Chernobyl demonstrated that the lessons from the Three Mile Island accident had not been acted upon in the USSR: in particular, the importance of systematic evaluation of operating experience; the need to strengthen the on-site technical and management capability, including improved operator training; and the importance of the man–machine interface.

Specific issues relating to the design of the RBMK reactors were the subject of intensive studies in the USSR after 1986, and some modifications have since been made to these reactors and to their operating regimes. More recently, discussions have been held at the international level on the safety of nuclear power plants with RBMK reactors. International efforts to assist in the safety assessment of the RBMK reactors have been intensified in recent months.

Efforts to enhance the safety of RBMK reactors will continue; however, international assistance can only achieve a fraction of what has to be done at the national level. In addition, the general issues mentioned earlier require sustained efforts to upgrade the national nuclear safety regime before a safety culture can be inculcated at all levels and in all the organizations concerned.

The emergence of the new information prompted the IAEA's International Nuclear Safety Advisory Group (INSAG) to review its former conclusions about the causes of the Chernobyl accident of April 1986. The present INSAG-7 report updates

the Summary Report on the Post-Accident Review Meeting on the Chernobyl Accident (IAEA Safety Series No. 75-INSAG-1), published in September 1986. The conclusions formulated in the report will broaden the basis for international consultations to enhance the safety of RBMK reactors.

With this report, INSAG has made a valuable contribution to the task that lies ahead.

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1. INTRODUCTION

IAEA Safety Series No. 75-INSAG-1 (INSAG-1) was prepared by the International Nuclear Safety Advisory Group (INSAG) at the request of the Director General of the IAEA, following the Post-Accident Review Meeting on the Chernobyl Accident held in Vienna over the period 25–29 August 1986. At that meeting, leading Soviet scientists and engineers presented the first published account of the accident at the fourth unit of the Chernobyl nuclear power station in the Ukrainian Republic of the Union of Soviet Socialist Republics on 26 April 1986, along with a discussion of the accident's causes. They delivered their account to a large number of experts from IAEA Member States and from international organizations. IAEA Safety Series No. 75-INSAG-1 was based upon this evidence, on additional material presented by the Soviet experts during the meeting and on discussions in Working Group sessions among the Soviet experts, INSAG members and other experts who assisted them.

Prior to the Vienna meeting, there had been worldwide speculation about the causes of the Chernobyl accident. Analytical studies prompted by this speculation were undertaken at numerous locations on the basis of incomplete information on the design characteristics of RBMK reactors published in the open literature. The Soviet experts' explanation to the meeting of the fault sequence leading to the destruction of Chernobyl Unit 4 was drawn from an analytical model 'normalized' to sparse and inaccurate plant data that had been recorded during the tests carried out on the turbogenerator which led to the accident.

The account given to the Vienna Conference stated that it was possible to explain the nature of the accident and the extent of the resulting structural damage in terms of an uncontrolled reactivity driven excursion. It was said that this excursion had been made possible by the signs and magnitudes of the void and power coefficients of reactivity characteristic of the state of the reactor at the time. These had been inadvertently set up by the operators in preparation for the tests and by the effect of pump coastdown which reduced coolant flow as the test progressed. The Soviet presentation of the detailed account was accompanied by statements of violations by operators of procedures and rules. In the absence of evidence it was not possible to examine the appropriateness of these procedures and rules, or the protection system, in relation to the reactor's design characteristics. The assertion was made that the accident arose through a low probability coincidence of a number of violations of rules and procedures by the operating staff and those responsible for authorizing the test.

INSAG members and the experts who were assisting them found this explanation of the cause of the accident plausible and made no attempt to generate alternative scenarios, although it was recognized that there were other possible explanations. All parties were aware that the analysis of the transient was very complex and sensitive to many factors. In INSAG-1 it was stated (p. 2) that "It would be surprising indeed if the report, issued after a short period of preparation and at a time when many ques-

tions remain to be settled by analysis, were found to be correct in every detail. INSAG therefore had to use its best judgement in forming conclusions and recommendations for action."

Since the 1986 Vienna Conference, a considerable amount of additional analytical work has been carried out by expert groups throughout the world on the causes of the Chernobyl accident. Many of the results have been published. Other information has also come to light, some of it contradictory. Most important among the sources of this information are reports by two Soviet committees chaired by N.A. Shteynberg and A.A. Abagyan respectively. These two reports, translated into English by the IAEA, are reproduced as Annexes I and II to this document since they are not generally available. All this has added to the knowledge that was available to the authors of INSAG-1 at the time of the report's preparation. The present publication updates that part of the INSAG-1 report that concentrated on the causes of the accident.

2. FEATURES OF THE REACTOR

There follow brief summaries of certain design[1] features of the RBMK-1000 reactor and the associated plant systems of Unit 4 of the Chernobyl nuclear power plant at the time of the accident on 26 April 1986. These design features had a primary influence on the course of the accident and its consequences.

2.1. VOID COEFFICIENT OF REACTIVITY

A reactor cooled by boiling water contains a certain amount of steam in its core. The steam bubbles are called voids, and that proportion of the coolant volume which consists of voids is called the void fraction. There is a change of reactivity if the void fraction is changed; the ratio of the two changes is termed the void coefficient of reactivity, which can be either positive or negative depending on the reactor design. A change in reactor power can cause the void fraction to change, and can also cause other effects that alter the reactivity. These changes in reactivity must be offset by control rods. The ratio between the total reactivity change produced in this way and the change in power causing it is called the power coefficient of reactivity, and this also can be positive or negative.

The void coefficient of reactivity is the dominant component of the power coefficient of reactivity of RBMK type reactors, reflecting a high degree of dependence of reactivity on the steam content of the core. The void coefficient depends significantly on the lattice pitch of the core and the core composition (number of control rods inserted into the core, number of installed additional absorbers, fuel enrichment and fuel burnup). It has been reported on the basis of studies made after the accident that the calculated void coefficient of reactivity for the RBMK-1000 reactor extended from -1.3×10^{-4} $\%^{-1}$ ($\delta k/k$) void for a fresh fuel load to $+(2.0–2.5) \times 10^{-4}$ $\%^{-1}$ ($\delta k/k$) void for the steady state refuelling regime; and that the effect on reactivity of a total loss of coolant was -2β for a fresh fuel load and $+(4–5)\beta$ for the steady state refuelling regime (where β is the delayed neutron fraction). In the design documentation for the RBMK reactor the void coefficient of reactivity was stated to be negative for initial and steady state conditions (see Annex II, Section II-3). Thus, although the void coefficient of reactivity varied over a wide range from negative to positive values as a function of the composition of the core and the operating regime of the reactor, the fast power coefficient remained negative under normal operating conditions. At the time of the accident, the void and power coefficients of reactivity were both positive.

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[1] The concept of design of a nuclear power plant is used frequently in this report. The meaning here is that established in the IAEA's Nuclear Safety Standards (NUSS) series of publications: The process and the result of developing the concept, detailed plans, supporting calculations and specifications for a nuclear power plant and its parts.

2.2. DESIGN OF CONTROL AND SAFETY RODS

The control rods and the safety rods of an RBMK reactor are inserted into the reactor core from above, except for 24 shortened rods which are inserted upwards and which are used for flattening the power distribution. A graphite rod termed a 'displacer' is attached to each end of the length of absorber of each rod, except for twelve rods that are used in automatic control. The lower displacer prevents coolant water from entering the space vacated as the rod is withdrawn, thus augmenting the reactivity worth of the rod. The graphite displacer of each rod of all RBMK reactors was, at the time of the accident, connected to its rod via a 'telescope', with a water filled space of 1.25 m separating the displacer and the absorbing rod (see Fig. 1). The dimensions of rod and displacer were such that when the rod was fully extracted the displacer sat centrally within the fuelled region of the core with 1.25 m of water at either end. On receipt of a scram signal causing a fully withdrawn rod to fall, the displacement of water from the lower part of the channel as the rod moved downwards from its upper limit stop position caused a local insertion of positive reactivity in the lower part of the core. The magnitude of this 'positive scram' effect depended on the spatial distribution of the power density and the operating regime of the reactor.

2.3. SPEED OF INSERTION OF THE EMERGENCY PROTECTION RODS

The total time required for insertion of the emergency protection rods (i.e. scram rods) into the core, when starting from the upper limit stop switches, was 18 s. This slow rate of insertion was principally the result of the narrow confinement of the rod in its channel, with the result that the cooling water through which the rod had to move acted somewhat like the fluid in a dashpot or motion damper.

2.4. CONTROL OF POWER

The RBMK-1000 reactor was equipped with two systems to provide power control. The first was the physical power density distribution control system (PPDDCS), which had detectors located inside the core. The second system was the reactor control system, which had detectors located inside the core as well as outside the core in the lateral biological shield tank.

In principle, the two systems were designed to supplement each other. The PPDDCS was designed to control the relative and absolute power distributions within the range of 10–120% of their nominal values and to control the total reactor power in the range of 5–120% of its nominal value. The reactor control system contained the local automatic control and local automatic protection systems (LAC-LAP). The LAC-LAP systems received signals from the in-core detectors and

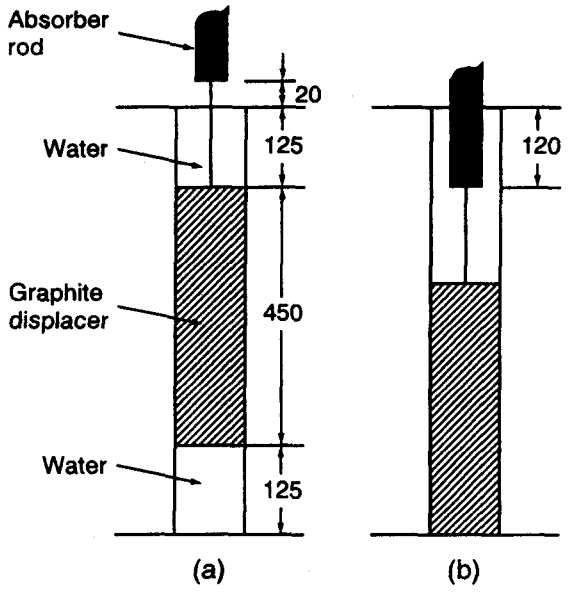


FIG. 1. Uppermost position of a control rod of the RBMK emergency protection system relative to the reactor core (a) before and (b) after the improvements made in response to the Chernobyl accident. Dimensions are given in centimetres.

operated at power levels above 10% of nominal. Control at lower power levels relied solely on the ex-core detectors.

When the reactor was operated at low power with the PPDDCS and the LAC-LAP systems switched off, there was no in-core instrumentation available. The operator had mainly to rely on the ex-core monitors for decisions that affected the total power and its spatial distribution. However, the ex-core detectors could not indicate the distribution of the neutron flux within the core. Moreover, they could not indicate the average axial distribution of flux, since they were all located at the mid-plane of the core.

Thus, when controlling the reactor at lower power levels, a reactor operator had to rely mainly on experience and intuition rather than on the readings of the instrumentation of the control system. Under these circumstances an operator could be required to perform manipulations at a rate of up to 1000 per hour.

Yet control of the RBMK-1000 at startup, when the reactor was free of the neutron absorber or neutron poison xenon-135, was different from and much simpler than control of the power density distribution of the non-uniformly poisoned reactor at low power. In this latter situation, which existed to a gross degree during the test that ended in the destruction of the Chernobyl Unit 4 reactor, large field non-uniformities and high disturbances of both axial and radial power density distributions can occur. The operators had little or no experience of control under these circumstances.

2.5. INSTRUMENTATION INDICATING THE REACTIVITY MARGIN

The computer and instrumentation used to determine the reactivity margin for the RBMK-1000 reactor were located approximately 50 m from the control console. The data acquisition system received information from about 4000 data input points. The system was used to calculate periodically the operating reactivity margin (ORM), which is the extra reactivity that would arise if all control and safety rods were withdrawn, expressed as a multiple of the total reactivity controlled by a standard rod. This data system required about 10–15 min to cycle through all measurements and to calculate the ORM. The system was designed to provide guidance to the operator on steady state control of the power density distribution, and was used for this purpose in conjunction with the system for monitoring the spatial power distribution.

2.6. SIZE OF THE REACTOR CORE

Owing to the largeness of the reactor core (height 7 m, diameter 11.8 m) of the RBMK-1000 reactor, the chain reaction in one part of the core is only very loosely coupled with that in other, distant, regions. This leads to a requirement to control the spatial power distribution almost as if there were several independent reactors within the core volume. This situation in extreme conditions can be highly unstable, because small spatial redistributions of reactivity can cause large spatial redistributions of the power. One manifestation of this decoupling of the core is that just prior to the accident the chain reactions in the upper and lower halves of the reactor were proceeding almost independently, a situation that was exacerbated by heavy xenon poisoning in the intervening central region. When control and safety rods were inserted from fully withdrawn positions under these circumstances, the positive scram effect discussed earlier could cause the lower part of the core to become supercritical and the neutron distribution to shift quickly downwards irrespective of the distribution just prior to rod insertion. Under the conditions of the accident, the shift in power distribution resulting from the positive scram could be substantial.

2.7. CAPABILITY TO ALTER SAFETY SYSTEMS, PLANT TRIPS AND ALARMS

At Chernobyl Unit 4, there was a ready capability for the operators manually to disable certain safety systems, bypass automatic scram trips, and reset or suppress various alarm signals. This could be done ordinarily by connecting jumper wires to accessible terminals. The operating procedures permitted such disabling under some circumstances.

2.8. SUBCOOLING OF THE INLET WATER

The RBMK reactors are boiling water reactors. The coolant enters the reactor core from below as water, subcooled below the boiling temperature, and boiling begins at some distance along the flow path through the core. Analysis and experiment have shown that the amount of subcooling of the entry coolant of a boiling water reactor is important for the stability of the reactor. If the subcooling falls to near zero, boiling begins almost at the core inlet and, because of the void coefficient of reactivity, reactivity effects become very sensitive to the inlet coolant temperature.

Furthermore, since there is not much change in fluid temperature between the coolant pumps and the core inlet, the temperature of the water in the pumps and at their intakes is near boiling if the subcooling is very small. Pump behaviour under these circumstances can become erratic, and pumping action can be reduced substantially or can even stop completely under some conditions (a process termed cavitation). This is discussed further in Section 2.9.

2.9. PRIMARY COOLANT SYSTEM

The RBMK-1000 reactor includes two independent primary coolant loops, each of which cools half of the reactor. Each loop has four primary coolant pumps, three of which are used in normal operation; the fourth is on standby as a backup to be used if one of the three in use must be shut down. Each pump has a capacity of 5500 to 12000 m^3/h. The discharge line from each pump also has a flow control valve and a check valve to prevent backflow should the pump fail. Each pump has shutoff valves to isolate it if necessary.

The coolant discharged from each of the three pumps in a coolant circuit is sent to a common header and then to 22 distributor headers in each half of the reactor. From these headers the flow is distributed to the individual pressure tubes containing the nuclear fuel. Each channel has a flow control valve used for optimizing the radial distribution of cooling across the core. Boiling occurs as the coolant passes through the pressure tubes where they transit the reactor core. The steam–water mixture from the various fuelled pressure tubes is carried by individual pipes to two parallel horizontal drum type separators in each loop. Steam exits from the top of each separator into two steam headers, from which it passes to the turbines. The condensate stream from the turbine in each loop forms a feedwater stream which joins the recirculation flow of water from the steam generators, to become the inlet coolant stream at the pump intakes. This completes the coolant circuit flow.

Under normal conditions, the flow rate of each pump is 8000 m^3/h. The normal core inlet temperature is 270°C and the core outlet temperature is 284°C at a pressure of 7 MPa (approximately 70 atm). The temperature of water flowing into the suction header of the main circulating pump depends on the rate of steam produc-

tion of the reactor, since the steam is condensed after passage through the turbine and becomes the cooler feedwater component of the coolant entering the pumps and the core. When the flow of this feedwater component of the coolant is decreased through reduction of the reactor power, the temperature of the coolant at the pump intake and the core inlet increases accordingly. During normal startup or shutdown operations, the flow rate through the primary circuit is controlled by using the throttle type control valves to reduce the flow from the normal rate of 8000 m^3/h per pump to the range 6000–7000 m^3/h. Fewer pumps are used during the low power phases of startup and shutdown. These measures ensure that the temperature at the main circulating pump inlet is low enough to prevent cavitation in the pumps and to maintain an appropriate axial distribution of steam production in the fuel channels.

Just prior to and at the start of the accident at Chernobyl, all eight pumps were running. Four were powered by the turbine remaining on line and four received power from the external power source. The use of all eight pumps increased the coolant flow rate above that under nominal conditions at full power, reducing the already low steam content of the core. This low steam fraction reduced the friction factor for coolant flow. In addition, owing to the lower reactor power level at that time, the core inlet coolant was only slightly subcooled and, depending on the exact settings for feedwater flow and recirculation flow and the pressure distribution in the system piping, may not have been subcooled at all.

These conditions led to the onset of boiling at or near the bottom of the core. Under the prevailing operating conditions, the void coefficient of reactivity was strongly positive, and the core was in a condition of enhanced susceptibility to divergent positive feedback of void reactivity if an increase in power were to occur. Moreover, at the higher flow rate, the margin to pump cavitation was also reduced.

When the turbine was tripped, the four pumps it was powering began to slow down as the turbine speed was reduced and the associated generator voltage fell. This reduced rate of core flow caused the void content of the core to rise and caused an initial positive feedback of reactivity which was at least in part the cause of the accident. There remain questions of whether pumping capability deteriorated further during this period, with pumps circulating a mixed steam–water mixture, or whether pumps even cavitated and ceased to circulate the coolant. The report by the Commission of the USSR State Committee for the Supervision of Safety in Industry and Nuclear Power (SCSSINP) (Annex I, Section I-4.5) refers to studies which concluded that the pumps did not cavitate. At the very least, the positive void coefficient of the RBMK reactor causes the design to be grossly sensitive to pumping disturbances or failure under the circumstances of the accident.

2.10. CONTAINMENT

RBMK reactors have 'localized' containments. That is, separate parts of the reactor and coolant circuit are enclosed in individual containment spaces, each of

which is meant to protect against rupture of the primary circuit boundary only within that space. In particular, the reactor core rests within an enclosure whose side walls serve also as shielding. The bottom of the enclosure is the heavy plate on which the core is stacked, and the top is a 2000 tonne cover plate. The extensions of the fuel channels penetrate the lower plate and the cover plate and are welded to each. The separate enclosure spaces of the containment are vented through pipes to an array of 'bubbler pools' below, which serve as pressure suppression pools, a feature somewhat resembling an aspect of most boiling water reactors in western countries.

As was the case at other RBMK reactors having such a confinement space about the reactor core, that space was capable of withstanding a pressure buildup resulting nominally from steam released by two simultaneous channel ruptures. This limitation on capacity was due to the sizes of the pressure relief pipes discharging to the bubbler pools. Simultaneous failure of a greater number of fuel channels would generate a pressure high enough to fail the containment function by lifting the cover plate, in the process severing the remainder of the fuel channels.

3. THE ACCIDENT

The sequence of events described in INSAG-1 was derived from the information presented by Soviet scientists at the 1986 Post-Accident Review Meeting on the Chernobyl Accident and in discussions between the Soviet scientists and the IAEA during the following week. Table I of INSAG-1 and the accompanying text presented the sequence of events as then understood, based upon a combination of plant data and computer modelling. Since the Vienna meeting there has been considerable further analysis of the events which has led to new insights into the physical characteristics of the RBMK reactor (described in Section 2) and also into some details of the progression of the accident on 26 April 1986. These insights have led to a need to revise some of the details of the scenario presented in INSAG-1, and to alter some important conclusions.

Detailed accounts of the sequence of events as are now believed to have occurred are presented in the Soviet reports by the Commission of the SCSSINP, chaired by N.A. Shteynberg, and the Working Group of USSR Experts, chaired by A.A. Abagyan (Annexes I and II). Furthermore, some of the information towards the end of Table I of INSAG-1 relies heavily on the results of computer modelling presented in 1986, which has been superseded by more sophisticated analysis. Section 3 contains no discussion of the significance of modelling differences. The times, events and significance referred to in the following are as quoted in Table I of INSAG-1.

(1) Isolation of the emergency core cooling system (14:00:00, 25 April)

It was stated in INSAG-1 that blocking of the emergency core cooling system (ECCS) was a violation of procedures. However, recent Soviet information confirms that blocking of the ECCS was in fact permissible at Chernobyl if authorized by the Chief Engineer, and that this authorization was given for the tests leading up to the accident and was even an approved step in the test procedure. INSAG believes that this point did not affect the initiation and development of the accident. However, it must be recognized that the plant was being operated at half power for the period of approximately 11 hours leading up to the accident, with the ECCS blocked out. This could be viewed as no violation only if the 11 hour period of half power operation were part of the planned test, which it clearly was not. Blocking the ECCS over this period and permitting operation for a prolonged period with a vital safety system unavailable are indicative of an absence of safety culture.

(2) Minimum allowable operating power of the reactor (23:10:00, 25 April)

The statement made in INSAG-1 (p. 15) that "continuous operation below 700 MW(th) is forbidden by normal safety procedures owing to problems of thermal-hydraulic instability" was based on oral statements made by Soviet experts during

the week following the Vienna meeting. In fact, sustained operation of the reactor at a power level below 700 MW(th) was not proscribed, either in design, in regulatory limitations or in operating instructions. The emphasis placed on this statement in INSAG-1 was not warranted. After the fact, it is clear that such a proscription should have applied.

(3) Transfer from local to global power control (00:28:00, 26 April)

The INSAG-1 report describes the precipitous fall in power to 30 MW(th) as being due to an operator error. Recent reports suggest that there was no operator error as such; the SCSSINP Commission report (Annex I, Sections I-4.6, I-4.7) refers to an unknown cause and inability to control the power, and A.S. Dyatlov, former Deputy Chief Engineer for Operations at the Chernobyl plant, in a private communication refers to the system not working properly.

(4) Turbogenerator trip signal blocked (01:23:04, 26 April)

Both the time and significance of the blocking of the turbogenerator trip have changed in the light of new information. The event occurred at 00:43:27 rather than at 01:23:04 as stated in INSAG-1. The time at which the second turbogenerator was shut off remains unchanged.

This trip was blocked in accordance with operational procedures and test procedures, and the SCSSINP Commission (Annex I, Section I-4.7.4) does not support the apportionment of any blame to operating personnel. In the light of new information regarding positive scram, the statement made under the significance column of Table I in INSAG-1 that "This trip would have saved the reactor" seems not to be valid.

(5) Required operating reactivity margin violated (01:00:00, 26 April)

The recent reports confirm that the minimum ORM was indeed violated by 01:00:00 on 26 April, and in fact claim that this minimum ORM was also violated for several hours on 25 April. According to the record, the computer SKALA, which was used to calculate the ORM, became unreliable in the period in which the test took place. In the view of INSAG, it is likely that the operator did not know the value of the ORM during the critical part of the test. Probably he was aware that continued operation under conditions of increasing xenon content of the reactor was reducing the ORM. The operators had been accustomed to regarding the lower limit on ORM as necessary for control of the reactor's spatial power distribution, but were not aware that it had safety significance by virtue of the increase in positive void coefficient as the ORM was reduced. Nor were they sensitized to the need to retain a suitable number of control or safety rods in a partially inserted position, for fast reactivity decrease if necessary. In fact, the safety significance of the reduction of

the ORM is much greater than was indicated in the INSAG-1 report. This whole issue is discussed in detail in Section 4 of the present report.

(6) Steam level protection disabled (01:19:00, 26 April)

The recent information suggests that the steam drum protection was changed as early as 00:36 on April 26 and not at 01:19:00 as stated in INSAG-1. However, according to Annex I (Section I-4.7.4), "Contrary to what is stated in official documents, the [SCSSINP] Commission does not consider that personnel should be held to blame for having blocked the steam pressure protection system of the steam separators."

This change of view hinges on the fact that two levels of protection are provided for with respect to low steam drum level, one at 600 mm and another normally at 1100 mm, depending on the power level. The operators did not reset this level and were technically in violation of Item 9, Procedures for Reswitching Keys and Straps of the Engineered Protection and Blocking Systems (according to the SCSSINP Commission (Annex I, Sections I-4.7.4, I-4.7.8)). However, protection at the lower steam drum level remained effective throughout the event.

4. MORE RECENT ANALYSES OF THE FAULT SCENARIO

4.1. THE SCENARIO

The analytical work which followed in late 1986 had the benefit of the Soviet data made available in Vienna. Critical data had been provided there on the control rod configuration, the power level and the spatial distribution of power just before the accident, as well as information on the thermal-hydraulic conditions that prevailed. The information that a double humped spatial distribution of power had been created appeared at the outset to bias the magnitude of the positive void coefficient of reactivity towards the smaller positive values associated with lower fuel irradiation at the upper and lower core boundaries. Some analysts found that with the diminished void coefficients it was difficult to match the time history of the power excursion as it had been published by the Soviet scientists at the Vienna meeting. A search therefore began for an additional mechanism that might have come into play. It was in this connection that the positive scram effect of safety rod insertion came to be publicly postulated, apparently first in some western analyses.

Detailed analysis indicated that the reactivity injected by the positive scram, when added to that provided by voids from increased boiling, was sufficient to generate a severe reactivity driven transient, comparable with that described at the Vienna meeting.

The existence of the positive scram effect was first acknowledged by Soviet experts at the Conference on Nuclear Power Performance and Safety in Vienna in 1987[2]. The SCSSINP Commission report states that this phenomenon had been known of at the time of the accident and that it had first been identified at the Ignalina RBMK plant in the Lithuanian Republic in 1983 (Annex I, Section I-3.8). Although the Chief Design Engineer for RBMK reactors promulgated this information to other RBMK plants, and stated that design changes would be made to correct the problem, he made no such changes, and the procedural measures he recommended for inclusion in plant operating instructions were not adopted. Apparently, there was a widespread view that the conditions under which the positive scram effect would be important would never occur. However, they did appear in almost every detail in the course of the actions leading to the accident.

Most analyses now associate the severity of the accident with the defects in the design of control and safety rods in conjunction with the physics design characteristics, which permitted the inadvertent setting up of large positive void coefficients. The scram just before the sharp rise in power that destroyed the reactor may well have been the decisive contributory factor.

On the other hand, the features of the RBMK reactor had also set other pitfalls for the operating staff. Any of these could just as well have caused the initiating event for this or an almost identical accident. They included:

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[2] Nuclear Power Performance and Safety (Proc. Conf. Vienna, 1987) 6 vols, IAEA, Vienna (1988). See Ref. [2] to Annex I, p. 134.

— Pump failure, disturbance of the function of coolant pumping or pump cavitation, combined with the effect of the positive void coefficient. Any of these causes could have led to sudden augmentation of the effect of the positive void coefficient.

— Failure of zirconium alloy fuel channels or of the welds between these and the stainless steel piping, most probably near the core inlet at the bottom of the reactor. Failure of a fuel channel would have been a cause of a sudden local increase in void fraction as the coolant flashed to steam; this would have led to a local reactivity increase which could have triggered a propagating reactivity effect.

Thus the question arises: Which weakness ultimately caused the accident?

There is a second question: Does it really matter which shortcoming was the actual cause, if any of them could potentially have been the determining factor?

4.2. OPERATING REACTIVITY MARGIN

The ORM is expressed in terms of the number of 'equivalent' control rods of nominal worth remaining within the core. The definition is not precise, and the importance of the quantity for the safety of the plant seems to have been poorly understood by the operators. It was widely believed that the importance of the ORM centred on the need for a number of control elements in the core adequate for manoeuvring to keep the power distribution balanced throughout, especially in the light of the tendency for xenon instability in such a large and loosely coupled core. Yet the magnitude of the ORM was not conveniently available to the operator, nor was it incorporated into the reactor's protection system. In the discussion of the scenario, the operators seemed not to be aware of the other reason for the importance of the ORM, which was the extreme effect it could have on the void and power coefficients.

One design approach to prevent unacceptably large void coefficients is to increase the fuel enrichment and to balance excess reactivity by the introduction of absorbers. In the initial core loading of RBMK reactors, these absorbers were present, fixed in fuel channels and separate from the control and protection system. As the fuel burnt out, the designers allowed these absorbers to be removed and the fuel irradiation to increase. This shifted the void coefficients significantly in the positive direction and also made them crucially sensitive to the extent of insertion of the control and protection rods. Under the circumstances of the accident, the void coefficient increased to such an extent that it overwhelmed the other components of the power coefficient, and the power coefficient itself became positive.

There is one other aspect of the safety importance of the ORM that has generally received too little emphasis. The staff of the Chernobyl Unit 4 reactor apparently believed that as long as the lower limit on ORM (15 equivalent rods) was

satisfied, no matter what the actual rod configuration was, the demands of safety were met. This is far from true. An array of safety rods can only contribute to safety against a power transient if, in the first entry of the rod into the core after a scram signal, it already begins to reduce the reactivity by a significant amount. This capability can only be ensured if the absorbing tips of the rods are kept in a region where a small insertion causes a relatively large reduction in reactivity. Such a region is not found near the boundary of the reactor core. No policy for proper rod positioning seemed to have applied during the test that led to the destruction of the reactor.

The SCSSINP Commission (Annex I, Section I-3.8) reports that, after discovery of the positive scram effect at Ignalina in 1983, the chief engineering organization informed other organizations and all nuclear power plants with RBMK reactors that it intended to impose restrictions on the complete withdrawal of control and safety rods from the core. Such restrictions were never imposed and apparently the matter was forgotten.

5. VIEWS OF INSAG

The previous sections recount and elaborate on information that has been received since the 1986 Post-Accident Review Meeting. It is the purpose of Section 5 to comment on the issues raised, in terms of any revisions necessary to INSAG-1 and the importance of the new information in the context of the accident. The three interrelated aspects considered are the design features, the actions of the operators and the overall framework of control in safety matters. It should be noted that the new information is as firmly based as it can be at this time. However, changes in that information in the future, as well as in the perception of its significance, cannot be ruled out.

5.1. DESIGN

A number of possibilities have been cited for the final accident initiator, all of which are dependent on specific design features. Rather than entering the debate by adopting a firm view, which could shed little if any new light, INSAG prefers to consider those design issues that raise fundamental questions.

INSAG-1 repeated the view stated in the Soviet presentation, that a major reactivity induced transient made possible by the positive power coefficient had been the primary cause of the accident. A general comment in INSAG-1 was that automatic safety systems must act as soon as the safety of the plant is seriously threatened (p. 77). The prevention of an accident due to a fast acting positive power coefficient depended on quick action by the operating staff; this was in unacceptable conflict with this fundamental design tenet.

The feature of the plant design which led to extensive comment, and which was not pointed out in the initial Soviet account, was the deficient system for emergency shutdown, which laid the basis for the positive scram. The most likely final event now seems to have been the insertion of safety rods at a vital moment in the test, which worsened to a destructive level the conditions already prevailing because of the positive power coefficient. The accident would then have been a result of questionable procedures and practices that manifested and compounded the two crucial design defects of rod design and positive reactivity feedback. A particular configuration of control and safety rods was necessary for the positive scram to occur, and a double humped power distribution points to the fact that decoupling had occurred between the upper and lower halves of the reactor. All of these conditions prevailed at the same time.

Whether this version of the accident corresponds to reality may never be known for certain. Yet it does not really matter whether a positive scram was the final step that caused the reactor's destruction. It only matters that such a deficiency existed and that it could have been the cause of the accident. It is reprehensible that such a deficiency had been known of for so long without its having been eliminated.

Certainly the account in INSAG-1 would have been different had the features of the control and safety rods been revealed at the Post-Accident Review Meeting in 1986.

Implicit in the design, and well recognized at the time of preparation of INSAG-1, was the requirement to maintain a control rod configuration conferring at least the minimum ORM. If, as has been stated since, there was no effective facility in the control room for informing the operators of this parameter, then again they were ill served by the plant's design, and a judgement different from that made in INSAG-1 is necessary. In recent deliberations, INSAG has in fact questioned the ORM concept, because its definition (see Section 4.2) is inappropriate for complete assurance that the control rod configuration secures fully adequate reactor protection.

With today's knowledge, a general impression gained at the time of preparation of INSAG-1 might have been given greater prominence. The design introduces conflicting requirements on the control/shutdown system. From the operator's standpoint, as a routine matter that system provides means for controlling reactor power and trimming the power distribution. The system also affects the value of the void coefficient, and it is needed to produce shutdown of the plant in emergencies. There is no reason why in normal circumstances these requirements cannot all be met. However, the operator's actions in raising almost all the rods to extreme positions led to conflict with the simultaneous requirements to maintain shutdown capability and appropriate values of the power coefficient (though this latter point was not appreciated at the time by the operators). The possibility of conflict among these objectives is an undesirable design feature which made the plant unduly reliant on sound operator action. INSAG-1 contains the general point as its first lesson and recommendation that "Nuclear plant designs must be as far as possible invulnerable to operator error and to deliberate violation of safety procedure" (p. 31).

Regarding the specific characteristics of the shutdown system, INSAG at the time judged that it was not sufficiently fast acting, and there is no reason to change this view, despite the new views about the possible causes of the accident. Another general impression formed at the time is now firmer. The procedures by which the plant was controlled were insufficiently well founded in analysis of its safety features. This recognition was in fact raised in the second of the Lessons and Recommendations of INSAG-1: "Procedures relating to the operation of the plant must be carefully prepared with the safety significance of what is intended continuously in view" (p. 31).

5.2. OPERATOR ACTIONS

5.2.1. Violations of procedures

In INSAG-1, views presented by the Soviet experts on the actions of operators were given prominence, and it is appropriate here to take advantage of more recent

information. Specific violations of procedures were identified in 1986 as major causes of the accident. In particular:

— The statement was made that there was a proscription on continuous operation of the reactor at power levels below 700 MW(th). This statement was based on incorrect information. There should have been such a proscription, but there was none at the time.

— Eight main coolant pumps were operating at full flow, and it appears that flow rates in several exceeded prescribed values. INSAG took the view that such operation was unsatisfactory. The SCSSINP Commission (Annex I, Section I-4.7.7) reports that operation of all eight pumps at once was not forbidden by any document, including the test procedures, though the excessive flow rates, where they occurred, violated procedures. This point is related to the question of subcooling, discussed in Section 5.2.3.

— In INSAG-1 it was stated that operation with too low an ORM was a violation of requirements. INSAG now repeats that the violation took place, but that it was important for different reasons from those previously understood. It led to increased void coefficients, and it led to the positioning of control and safety rods such that they were not only ineffective but also destructive.

— It was stated in INSAG-1 that, at the time of the test, three components of reactor protection had been disabled at Chernobyl. The information now available suggests the following, contrary to what was expressed in INSAG-1:

  • Disabling of the ECCS was not prohibited in principle under normal procedures at Chernobyl. INSAG understands that it was a requirement of the test schedule, and, in accordance with regulations, special approval for this disabling had been obtained from the Chief Engineer. In any case, it was not necessary to disable the ECCS for such a long period of time. INSAG believes that this did not affect the accident, but it did manifest a poor level of safety culture.

  • Disabling of the trip on steam drum water level would have been allowable; however, it did not occur, INSAG considers that this would not have affected the accident, and in any case another line of protection existed.

  • Disabling of the 'two turbine' trip was allowed, and indeed was required by normal procedures at low power levels, such as the power level for the revised test. In any event, the occurrence of this trip might well have caused the destruction of the reactor at the time of turbine trip rather than shortly afterwards.

INSAG wishes to comment further that, while all this may be so, disabling of reactor protection seems to have been regarded rather lightly both in the operating procedures and by the operators: witness the length of time for which the ECCS was out of service while the reactor was operated at half power.

5.2.2. Departure from test procedures

It is not disputed that the test was initiated at a power level (200 MW(th)), well below that prescribed in the test procedures. Some of the recent comments addressed to INSAG boil down to an argument that this was acceptable because nothing in normal procedures forbade it. However, the facts are that:

— the test procedure was altered on an ad hoc basis;

— the reason for this was the operators' inability to achieve the prescribed test power level;

— this was the case because of reactor conditions that arose owing to the previous operation at half power and the subsequent reduction to very low power levels;

— as a result, when the test was initiated the disposition of the control rods, the power distribution in the core and the thermal-hydraulic conditions were such as to render the reactor highly unstable.

When the reactor power could not be restored to the intended level of 700 MW(th), the operating staff did not stop and think, but on the spot they modified the test conditions to match their view at that moment of the prevailing conditions.

Well planned procedures are very important when tests are to take place at a nuclear plant. These procedures should be strictly followed. Where in the process it is found that the initial procedures are defective or they will not work as planned, tests should cease while a carefully preplanned process is followed to evaluate any changes contemplated.

5.2.3. Other deficiencies in safety culture

The foregoing discussion is in many ways an indication of lack of safety culture. Criticism of lack of safety culture was a major component of INSAG-1, and the present review does not diminish that charge. Two examples already mentioned are worthy of emphasis, since they bear on the particular instincts required in reactor operation.

The reactor was operated with boiling of the coolant water in the core and at the same time with little or no subcooling at the pump intakes and at the core inlet. Such a mode of operation in itself could have led to a destructive accident of the kind that did ultimately occur, in view of the characteristics of positive reactivity feedback of the RBMK reactor. Failure to recognize the need to avoid such a situation points to the flaws in operating a nuclear power plant without a thorough and searching safety analysis, and with a staff untutored in the findings of such a safety analysis and not steeped in safety culture.

This last remark is especially pertinent to the second point, which concerns operation of the reactor with almost all control and safety rods withdrawn to positions where they would be ineffective in achieving a quick reduction in reactivity if

shutdown were suddenly needed. Awareness of the necessity of avoiding such a situation should be second nature to any responsible operating staff and to any designers responsible for the elaboration of operating instructions for the plant.

5.3. SAFETY FRAMEWORK

INSAG-1 concentrated on the immediate issues of the Chernobyl accident and made little reference to the regulatory and general safety framework within which the plant was operated. A number of matters have since come to light and assertions have been made that make it right at this time to present broader views.

The SCSSINP Commission (Annex I, Section I-3) has compared the design of Chernobyl Unit 4 with the declared safety requirements at the time of design, stating that the design fell well short of the standards set. INSAG notes that a number of the issues raised in the report of the SCSSINP Commission mirror its own concerns.

This point is further discussed in the following sections.

5.4. IMPLICATIONS OF IGNORING DEFICIENCIES

Annexes I and II indicate that important problems now recognized in the Chernobyl plant design had in fact been recognized before the accident. INSAG notes the observations made at the Ignalina plant in 1983, when the possibility of positive reactivity insertion on shutdown became evident, and the event at the Leningrad nuclear power plant in 1975 which, in retrospect, indicated that events excited by local reactivity feedback could cause damage to the reactor. These two events pointed to the existence of design problems. Although the events had the semblance of potential precursors to an accident, apparently no thorough analysis was performed. It is a matter of great concern that this important information was not adequately reviewed and, where it was disseminated to designers, operators and regulators, its significance was not fully understood and it was essentially ignored.

5.5. IMPORTANCE OF COMPETENT SAFETY ANALYSIS

Independent technical review and safety analysis are a cornerstone of a satisfactory safety regime, and in this connection INSAG judges that the design and operation of Chernobyl Unit 4 as well as of other RBMK reactors should have received a great deal more such attention. The design deficiencies would surely have come to light in the course of such a review. The improved understanding derived from the review, coupled with a regime requiring independent and formal approval for changes to safety related aspects of design and operating procedures, would have gone a long way towards averting the accident altogether. Even apart from its

obvious intrinsic value, competent safety analysis helps to create an environment of attention to safety as a primary objective. This point underlies the importance of effective transfer to operators of the knowledge gained through safety analysis.

5.6. DEFICIENCIES IN THE REGULATORY REGIME

5.6.1. General deficiencies

The assurance of safety in the face of the inevitable pressures to meet production goals requires a dedicated operating organization and a strong and independent regulatory regime, properly resourced, backed at Government level and with all necessary enforcement powers. This sort of regime did not exist in the USSR at the time of the accident.

INSAG is informed that the regulatory regime was ineffective in many important areas, such as analysing the safety of the design and operation of plants, in requirements for training and for the introduction and promotion of safety culture, and in the enforcement of regulations. It did not function as an independent component in ensuring safety.

5.6.2. The SCSSINP Commission report

The SCSSINP Commission report (Annex I) contains extensive information emphasizing this lack of an effective nuclear regulatory regime over the years prior to the accident.

The basic design of the RBMK reactors was approved despite the lack of conformity to many of the USSR's design requirements for nuclear power plants.

5.7. GENERAL REMARKS ON THE LACK OF SAFETY CULTURE

In its report on the Chernobyl accident, INSAG coined the term 'safety culture' to refer to the safety regime that should prevail at a nuclear plant. In its later report, INSAG-4, Safety Culture[3], which expounded the concept, INSAG traced the development of a safety culture to its origin in the national regime of law relating to nuclear safety. This establishes the proper chain of responsibility and authority for the required level of safety. In both operating and regulatory regimes, safety culture must be instilled in organizations through proper attitudes and practices of management. It has been pointed out several times in the preceding discussion that safety culture was lacking in the operating regime at Chernobyl. In conformity with its

[3] INTERNATIONAL NUCLEAR SAFETY ADVISORY GROUP, Safety Culture, Safety Series No. 75-INSAG-4, IAEA, Vienna (1991).

views as expressed in INSAG-4, INSAG now confirms the view that safety culture had not been properly instilled in nuclear power plants in the USSR prior to the Chernobyl accident. Many of its requirements seem to have existed in regulations, but these were not enforced. Many other necessary features did not exist at all. Local practices at nuclear plants, of which it may be assumed that practices at Chernobyl were typical, did not reflect a safety culture.

5.8. SUMMARY ASSESSMENT

In reviewing information made available since the Post-Accident Review Meeting, INSAG judges that factors leading to the accident are to be found in the safety features of the design, the actions of the operators, and the general safety and regulatory framework. There is a need to shift the balance of perception so as to emphasize more the deficiencies in the safety features of the design which were touched on in INSAG-1, and to recognize the problems conferred by the framework within which plant operation was carried out. However, INSAG remains of the view that in many respects the actions of the operators were unsatisfactory.

6. CONCLUSIONS ON FACTORS CONTRIBUTORY TO THE ACCIDENT

(1) Information coming to light since 1986 on the accident at Chernobyl Unit 4 has been reviewed. It has been approached with caution in recognition that further information yet to come might change the picture again. However, it appears now that the main outlines of the problems are becoming clear.

(2) In 1986, INSAG issued its report INSAG-1, which discussed the Chernobyl accident and its causes on the basis of information presented to the Post-Accident Review Meeting in Vienna in August 1986 by Soviet authorities. The new information now come to light has affected the views presented in INSAG-1 in such a way as to shift the emphasis to the contributions of particular design features, including the design of the control rods and safety systems, and arrangements for presenting important safety information to the operators. The accident is now seen to have been the result of the concurrence of the following major factors: specific physical characteristics of the reactor; specific design features of the reactor control elements; and the fact that the reactor was brought to a state not specified by procedures or investigated by an independent safety body. Most importantly, the physical characteristics of the reactor made possible its unstable behaviour.

(3) Two earlier accidents at RBMK reactors, one at Leningrad (Unit 1 in 1975) and a fuel failure at Chernobyl (Unit 1 in 1982), had already indicated major weaknesses in the characteristics and operation of RBMK units. The accident at Leningrad Unit 1 is even considered by some to have been a precursor to the Chernobyl accident. However, lessons learned from these accidents prompted at most only very limited design modifications or improvements in operating practices. Because of lack of communication and lack of exchange of information between the different operating organizations, the operating staff at Chernobyl were not aware of the nature and causes of the accident at Leningrad Unit 1.

(4) It is not known for certain what started the power excursion that destroyed the Chernobyl reactor. Some positive reactivity is likely to have been generated from the growth in voids as the coolant flow rate fell. Addition of further positive reactivity by insertion of the control and safety rods that had been fully withdrawn during the test was probably a decisive contributory factor. This latter effect was a result of faulty design of the rods, the nature of which had been discovered at the Ignalina nuclear power plant in 1983. However, no correction was made following this discovery at Ignalina, no compensatory measures were taken and any dissemination of information to operating organizations was not followed up.

(5) The accident can be said to have flowed from deficient safety culture, not only at the Chernobyl plant, but throughout the Soviet design, operating and regula-

tory organizations for nuclear power that existed at the time. Safety culture, fully discussed in INSAG-4 (see footnote 3), requires total dedication, which at nuclear power plants is primarily generated by the attitudes of managers of organizations involved in their development and operation. An assessment of the Chernobyl accident in this respect demonstrates that a deficit in safety culture was inherent not only to the stage of operation, but also and to no lesser extent to activities at other stages in the lifetime of nuclear power plants (including design, engineering, construction, manufacture and regulation).

(6) The weight given in INSAG-1 in 1986 to the Soviet view presented at the Vienna meeting, which laid blame almost entirely on actions of the operating staff, is thereby lessened. Certain actions by operators that were identified in INSAG-1 as violations of rules were in fact not violations. Yet INSAG remains of the opinion that critical actions of the operators were most ill judged. As pointed out in INSAG-1, the human factor has still to be considered as a major element in causing the accident. The poor quality of operating procedures and instructions, and their conflicting character, put a heavy burden on the operating crew, including the Chief Engineer. It has also to be noted that the type and amount of instrumentation as well as the control room layout made it difficult to detect unsafe reactor conditions. However, operating rules were violated, and control and safety rods were placed in a configuration that would have compromised the emergency protection of the reactor even had the rod design not been faulty on the ground of the positive scram effect mentioned earlier. Most reprehensibly, unapproved changes in the test procedure were deliberately made on the spot, although the plant was known to be in a condition very different from that intended for the test.

(7) INSAG, with the present report, does not retract INSAG-1, nor does it alter the conclusions of that report except as clearly indicated here. While the balance of INSAG's judgement of the factors contributing to the accident has shifted, the many other conclusions of INSAG-1 are unaffected.

(8) To summarize, the new information has highlighted a number of broader problems contributing to the accident. These include:

- A plant which fell well short of the safety standards in effect when it was designed, and even incorporated unsafe features;
- Inadequate safety analysis;
- Insufficient attention to independent safety review;
- Operating procedures not founded satisfactorily in safety analysis;
- Inadequate and ineffective exchange of important safety information both between operators and between operators and designers;
- Inadequate understanding by operators of the safety aspects of their plant;
- Insufficient respect on the part of the operators for the formal requirements of operational and test procedures;

— An insufficiently effective regulatory regime that was unable to counter pressures for production;
— A general lack of safety culture in nuclear matters, at the national level as well as locally.

Appendix

MEASURES TO IMPROVE THE SAFETY OF RBMK PLANTS

It is reported that organizational and technical measures to improve the safety of operating RBMK plants were developed immediately following the Chernobyl accident. These included placing restrictions on the remaining RBMK plants, the implementation of changes that had earlier been seen as necessary, and other changes that were clearly beneficial in terms of safety.

Firstly, INSAG has been told that measures had been developed and implemented that were aimed at:

- Reducing the positive steam (void) coefficient of reactivity and the effect on reactivity of complete voiding of the core;
- Improving the speed of the scram system;
- Introducing new computational codes for the ORM, with numeric indication of the ORM in the control room;
- Precluding the possibility of bypassing the emergency protection system while the reactor is at power, through an operating limit requirement and the introduction of a two key system for the bypass action;
- Avoiding modes of operation leading to reduction of the departure from nuclear boiling (DNB) margin for the coolant at reactor inlet (this addresses the question of adequate subcooling at the core inlet).

INSAG has also been told that a reduction of the void coefficient of reactivity has been provided by the installation of additional fixed absorbers (up to 90 pieces) into the core, and through introduction of the use of fuel with 2.4% ^{235}U enrichment in all RBMK reactors. All 1000 MW(e) reactors have been provided with the amount of more highly enriched fuel that is necessary to compensate for the additional fixed absorbers, and the transition to use of only the higher enrichment fuel is to be completed. Parenthetically, INSAG notes that the benefit of the increase in fuel enrichment will only be retained if there is no extension of fuel burnup beyond that customary in the past. If the added fuel enrichment were to be used to extend fuel life, the fuel at end of cycle would contain less ^{235}U and more ^{239}Pu, and this would act to increase the positive void fraction.

It is claimed that the ORM has been increased in this way to between 43 and 48 control rods, depending on the reactor.

INSAG has been told that existing control and safety rods have been replaced by new ones of an improved design which would not leave water columns towards the bottom and which have longer absorbing sections.

INSAG has been informed that the speed of insertion of control and safety rods has been increased, with the time required for full insertion into the core being reduced from 18 s to 12 s.

INSAG has been told that a Fast Acting Emergency Protection (FAEP) system has been installed at all operating reactors. This system includes 24 additional safety rods. The FAEP is meant to ensure that on demand, a negative reactivity of more than 2β (where β is the delayed neutron fraction) will be inserted in less than 2.5 s. The value of 2β was calculated conservatively to cover any reactivity addition associated with complete loss of coolant from the reactor. INSAG is told that all RBMK reactors are now equipped with the FAEP system.

Measures to reduce the void coefficient and to increase the rate of reactivity reduction on scram could also be of benefit in connection with uncontrolled power increase in the event of a loss of coolant water.

INSAG has been informed that operating instructions have been updated to take into account lessons learned from the Chernobyl accident and the measures taken to improve RBMK plant safety. Among the new provisions is one which now sets a lower limit of 700 MW(th) for steady operation of an RBMK reactor.

It has been reported that other measures have also been taken to improve the capability for mitigation of the consequences of an accident. These are covered in the report by the Working Group of USSR Experts (Annex II).

Annex I

REPORT BY A COMMISSION TO THE USSR STATE COMMITTEE FOR THE SUPERVISION OF SAFETY IN INDUSTRY AND NUCLEAR POWER[1]

Causes and Circumstances of the Accident at Unit 4 of the Chernobyl Nuclear Power Plant on 26 April 1986 (Moscow, 1991)

The report, submitted by a Commission at the request of the State Committee for the Supervision of Safety in Industry and Nuclear Power (SCSSINP), concludes that the Chernobyl accident, which was initiated by erroneous actions on the part of the operating personnel, had disproportionately disastrous consequences because of deficiencies in the design of the reactor. This conclusion is based on an analysis of the findings of Soviet and foreign studies, the design information and the documentation on technical standards.

The Commission believes that the analysis of the causes and circumstances of the accident should be continued and that appropriate measures should be taken to improve the safety of nuclear power plants.

The Commission was set up by SCSSINP Order No. 11 of 27 February 1990.

Bagdasarov, Yu.E.
Kuznetsov, A.G.
Miroshnichenko, M.I.
Petrov, V.A. *(Deputy Chairman)*
Shteynberg, N.A. *(Chairman)*
Zhuravlev, A.D.

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I-1. INTRODUCTION

A nuclear accident with disastrous consequences occurred at Chernobyl Unit 4 on 26 April 1986 during experiments to test the design operation of the independent power supply in the event of the loss of external power sources.

[1] This report was translated into English by the IAEA.

The task of dealing with the consequences of the accident has relegated the problem of clarifying the causes and circumstances of the accident and learning lessons from it to a position of secondary importance as far as the public is concerned. However, among experts these aspects are not yet regarded as settled. This fact is demonstrated by the continuing calculations and studies, as well as by the discussions on these issues at various national and international forums, including seminars and scientific and technical councils.

Unfortunately, so far no scientific organization in the USSR has published a thoroughly substantiated comprehensive account explaining how the accident originated and developed. As a result, the public continues to be very suspicious about reactors of the Chernobyl type, and all assurances that such an accident could not possibly happen again merely sound like the assurances in the not so distant past about the high safety levels of the RBMK-1000 reactor.

The Commission, set up by the USSR State Committee for the Supervision of Safety in Industry and Nuclear Power (SCSSINP), has attempted to analyse and make generalizations from the documentation and reports that are available on the accident. The official account of the accident, which was approved by the Government Commission and formed the basis of the report (information) presented by the USSR at the IAEA meeting of experts from 25 to 29 August 1986 in Vienna [1], suggests that the primary cause of the accident was an extremely improbable combination of violations by the unit personnel of operating instructions and procedures. A year later a Soviet report entitled The Accident at the Chernobyl Nuclear Power Plant: One Year After [2] presented at the IAEA International Conference on Nuclear Power Performance and Safety (Vienna, 28 September–2 October 1987) also confirmed the official account. However, an analysis and comparison of those reports raises a number of questions casting doubts on the credibility of this account.

A report by the I.V. Kurchatov Institute of Atomic Energy [3], approved after presentation of the report [1] to the IAEA, states that: "the primary cause of the accident was an extremely improbable combination of violations by the unit personnel of operating instructions and procedures, *in the course of which deficiencies in the design of the reactor and the control rods of the control and protection system emerged*" (no words are italicized in the official version). Moreover, as is stated in the same report [3]: "*...it is fairly obvious that the only version that does not contradict the available data is the version that deals with the effect of the control rod displacers.*" These contradictions are sufficient grounds for continuing the analysis of the causes and circumstances of the accident in order to establish the truth and to develop measures based on facts to prevent similar accidents from occurring in future.

In accordance with the responsibilities of the supervisory-regulatory body (SCSSINP), the members of the Commission focused their attention on examining and evaluating whether the reactor design and actions of the operating personnel complied with the requirements of the technical standards and operating documentation in force at the time of the design and operation of Chernobyl Unit 4. They tried

to determine why the actions taken by the personnel caused the disaster, particularly since they were taken not by one person alone but by a whole shift.

The results of an analysis of the personality and sociopsychological characteristics of the Chernobyl plant personnel before and after the accident are presented in publications by the Prognoz Psychological Research Laboratory of the USSR Ministry of Nuclear Power and Industry [4–6]. The analysis showed that the differences between the data on the personalities of the operating personnel at the Chernobyl plant and those of personnel at other plants were not such as to have been a direct cause of the accident. As a whole, the Chernobyl personnel in 1986 were characterized as a fairly typical, mature and stable group of specialists with qualifications regarded in the USSR as satisfactory. They were no better, but no worse, than the personnel at other nuclear plants.

This suggests that the personnel at the Chernobyl plant did not have any extraordinary characteristics which could account for the violations and errors that occurred. It is therefore essential to continue to analyse the causes and circumstances of the accident in order to establish whether the accident really occurred as a result of "a highly improbable combination of violations of operating instructions and procedures" or whether the causes of the accident were a combination of deficiencies in the design of the reactor for which the designers were responsible and erroneous actions on the part of the personnel.

In accordance with established international and national practice, the design of nuclear facilities and their components has to comply strictly with special safety standards and regulations. In the aforementioned reports, no indication is given of whether an expert review of the design of RBMK reactors in general, and of Chernobyl Unit 4 in particular, was conducted to see whether they comply with the safety standards and regulations. In sifting through the available documentation, the Commission found that some of the engineering defects in the RBMK-1000 reactor design and violations of the safety standards and regulations had already been realized by the end of May or beginning of June 1986. This information is contained in various reports presented to the Government Commission. However, the defects identified in the reactor design and its unsatisfactory physical parameters have not been widely publicized among the scientific community and general public in the Soviet Union. They were also not included in the papers presented to the IAEA. Much earlier, on 28 December 1984, the Interdepartmental Science and Technology Council on Nuclear Power approved proposals made by expert commissions Nos 4 and 5, which were set up by the Council to develop measures to bring existing RBMK-1000 units partially into line with the safety standards and regulations. Regrettably, however, the expert commissions ignored certain characteristics of the RBMK-1000 reactor which, as it turned out, played an important role in the initiation and subsequent development of the accident on 26 April 1986.

This report considers only those design features which may have caused the accident, played a role in its development or affected its consequences. The Commission found it necessary to draw attention to the way in which design defects affected

the quality of the operating documentation used by the personnel in running the unit. Some consideration is given to the measures and requirements which were introduced at all plants with RBMK-1000 reactors immediately after the accident and subsequently as more sophisticated improvements become available. These changes can be seen as evidence of earlier deficiencies in reactor design. The Commission draws attention to the discrepancy between the scope and substance of these measures and the official version which attributes the causes of the accident solely to personnel errors.

I-2. BRIEF DESCRIPTION OF THE DESIGN OF CHERNOBYL UNIT 4

By a decree of 29 September 1966, the USSR Council of Ministers approved a plan for the introduction of a generating capacity of 11.9 GW at nuclear power plants for the period 1966–1977, including 8 GW at plants with RBMK-1000 reactors. The decree approved the construction of the Leningrad plant, the first in the series of plants with RBMK-1000 reactors, proposed by the USSR State Planning Committee, the USSR Ministry for Intermediate Sized Machinery and the USSR Ministry of Power. The Ministry for Intermediate Sized Machinery was to be responsible for the plant's construction and the Ministry of Power for its operation. Under this decree, the Ministry for Intermediate Sized Machinery was given responsibilities for the scientific management of the design of the reactor units and research and development work; for providing the manufacturers with working drawings; for the designs adopted; for the scientific management of the startup of the reactors and bringing their parameters in line with the design values; for the fabrication and supply of fuel to the nuclear power plants; and for reprocessing of the spent fuel. The Ministry of Power was made responsible for the overall design, construction and operation of the plants.

Siting of the Chernobyl plant was based on a technical and economic report entitled Siting of the Central Ukrainian Nuclear Power Plant, prepared by the Kiev Department of the Teploehlektroproekt Institute and the Kiev Department of the Ehnergoset'proekt Institute. Two sites were proposed in the report: the village of Ladyzhino, in the Vinnitsa province, and the village of Kopachi, in the Kiev province.

On 4 March 1966 the Council of Ministers of the Ukrainian SSR decided to site a fossil fuel power plant at Ladyzhino. On 15 March 1966 the USSR Ministry of Power approved the siting of the Central Ukrainian nuclear power plant at Kopachi. On 18 January 1967 the Ukrainian State Planning Committee agreed to the siting of the plant near the village of Kopachi, in the Kiev province, and called it the Chernobyl nuclear power plant. On 2 February 1967 a joint decree passed by the Central Committee of the Communist Party of the Soviet Union and the USSR Council of Ministers confirmed the decision taken by the Ukrainian State Planning Committee.

The Urals Department of Teploehlektroproekt was given the task of developing the design specifications for the construction of the 2000 MW Chernobyl plant. The design specifications were approved by the USSR Ministry of Power on 29 September 1967. Three alternative design specifications were prepared:

— for an RBMK-1000 reactor;
— for a gas cooled RK-1000 reactor;
— for a WWER-1000 reactor.

The design specifications indicated that the engineering and economic factors for the first option were the worst, but that the state of development and the availability of equipment for that option were more satisfactory than for the other options.

On 21 September 1968, the Ministry of Power and the Ministry for Intermediate Sized Machinery jointly approved the design specifications for the gas cooled graphite reactor. However, subsequently, owing to the better equipment availability, the Ministries changed their decision and on 19 June 1969 decided in favour of a plant with an RBMK-1000 reactor. The reworked design specifications were approved by the USSR Council of Ministers on 14 December 1970. An order by the USSR Ministry of Power on 30 March 1970 entrusted the Gidroproekt Institute with all further work on the design of the Chernobyl plant. In accordance with the USSR Council of Ministers' decree of 29 June 1966, the All-Union Research and Design Institute for Integrated Power Technology (of the Ministry for Intermediate Sized Machinery), as a subcontractor of the General Designer (the Gidroproekt Institute), developed the reactor design for the first stage of the Chernobyl plant.

A feasibility study to expand the Chernobyl plant to 4000 MW was approved by the Science and Technology Council of the Ministry of Power on 30 March 1972. A joint decision by the Ministry of Power and the Ministry of Intermediate Sized Machinery regarding the design and construction of the Smolensk plant and the second stage of the Chernobyl plant was taken on 4 January 1974. Under that decision, Gidroproekt and the All-Union Research and Design Institute for Integrated Power Technology were made jointly responsible for the design. The technical design of the second stage of the Chernobyl plant was developed by Gidroproekt and examined by the USSR State Committee on Construction and the USSR State Planning Committee, which sent it to the USSR Council of Ministers on 30 September 1975 with a joint letter. The Council of Ministers then approved the technical design by its decree No. 2638 R of 1 December 1975.

The technical design of the RBMK-1000 reactor for the Leningrad plant, which was the first in the series, was developed by the Scientific Research and Design Institute for Power Technology following instructions from the Ministry for Intermediate Sized Machinery and was approved by the Science and Technology Council of this Ministry in October 1967 [7]. The technical design of the RBMK reactor was not redeveloped or revised for any of the subsequent units.

I-3. VIOLATIONS OF SAFETY STANDARDS AND REGULATIONS IN THE DESIGN OF CHERNOBYL UNIT 4

This part of the report describes some of the ways in which Chernobyl Unit 4 violated the Nuclear Safety Regulations for Nuclear Power Plants (NSR) [8] and the General Safety Provisions for the Design, Construction and Operation of Nuclear Power Plants (GSP) [9] that were in force at the time of the plant's design and construction. Article 1.1.4 of GSP-73 states that: "the scope of the requirements of the General Safety Provisions which shall apply to newly designed nuclear power plants with RBMK-1000 and WWER-440 reactors shall be determined for each specific power plant or a group of plants by a special resolution of the organization which approved the General Safety Provisions." The resolution of 2 July 1975 on safety in the design of the second stages of the Kursk and Chernobyl plants states that designers must observe the safety standards in force at the time, i.e. GSP-73 and NSR-04-74. The Commission cites only those violations of the aforementioned documents which played a role in the initiation and development of the accident on 26 April 1986. In order to facilitate the presentation of the findings, the wording of the relevant articles of the regulations is given, then reference is made to the violations in question and finally the substance of the violations is described.

I-3.1.

> Article 3.1.6 of NSR-04-74 states that: *"In the technical design of a nuclear power plant, the design documents relating to nuclear safety must be included as a separate part in the technical safety report on the construction and operation of the nuclear power plant.* **Note:** *All violations of the regulations are to be included in this part. Any violations must be authorized by the USSR State Committee for the Supervision of Nuclear Power Safety at the technical design stage."*

The technical design documentation for the second stage of the Chernobyl plant (Units 3 and 4), developed by the General Designer (Gidroproekt) in 1974 [10], contained the Technical Safety Report for the Chernobyl plant, approved by the Scientific Manager (I.V. Kurchatov Institute of Atomic Energy) and Chief Design Engineer (Scientific Research and Design Institute for Power Technology). The Nuclear Power Plant Technical Safety Report [11] was prepared with account taken of both the Technical Safety Report for the RBMK Four Reactor Facility [12], developed by the Scientific Research and Design Institute for Power Technology, and the Technical Report of Glavatomehnergo, USSR Ministry of Power [13].

No list of violations of the regulations in the design of the plant or of the reactor facility for the second stage of the Chernobyl plant was given in any of the aforementioned design documents. Furthermore, no analysis was made of the acceptability of these violations and they were not authorized by the State Committee for the Supervision of Nuclear Power Safety: "There were at least two weak points in the RBMK

design: the positive void coefficient of reactivity and the emergency protection system which, when the operating instructions were violated, did not shut down the reactor fast enough, and in some cases could even briefly increase its power'' [14]. Both of these 'weak points' were the result of violations of the safety standards and regulations and they are considered in the following. Since officially there were no violations, no technical and organizational measures were developed to compensate for them.

The plant Technical Safety Report [11] was approved by the Department for Nuclear Power Supervision of the USSR State Committee for the Supervision of Safety in the Mining Industries (Letter of 5 March 1975, No. 24-11/73), by the USSR State Committee for the Supervision of Nuclear Power Safety (Letter of 18 May 1975, No. H18) and by the USSR State Committee for Public Health Supervision (Letter of 20 January 1975, No. 32–57).

> **Note:** Until 1984 the State Committee for the Supervision of Nuclear Power Safety was a division of the USSR Ministry for Intermediate Sized Machinery.

The Commission concludes that the requirements of Article 3.1.6 of NSR-04-74 were not met by the designers of the Chernobyl plant and reactor, and notes that since the Technical Safety Report did not contain a list of the violations of the safety standards and regulations or measures to compensate for those violations, the operating documentation used by the personnel to run the plant could not possibly correspond to the actual parameters of the reactor.

I-3.2.

> Article 3.2.2 of NSR-04-74 (as well as Article 2.2.3 of GSP-73) states that: *''When designing the reactor it is desirable to ensure that the total power coefficient of reactivity is not positive under any operating conditions. If the total power coefficient of reactivity becomes positive under any operating conditions, the nuclear safety of the reactor during steady state, transient and emergency operating conditions must be guaranteed in the design and explicitly proved.''*

The void coefficient of reactivity α_φ is the dominant component of the total power coefficient of reactivity of RBMK reactors and numerically reflects changes in reactor reactivity as a result of changes in the steam quality in the core. It was originally foreseen in the design of the RBMK-1000 that with the uranium–graphite ratio that was selected, when the fuel burnup fraction corresponded to steady state refuelling conditions, the void coefficient of reactivity would have a high positive value. These high positive values for the void coefficient of reactivity were the result of a desire to attain large fuel burnups (i.e. high economic efficiency). It was intended that the region of reactor stability would be guaranteed in the range of void coefficients of reactivity of -3.2×10^{-4} δk/k to $+ 9.6 \times 10^{-4}$ δk/k. This coefficient depends largely on the lattice pitch of the core and the core composition (number

of control rods submerged in the core, number of additional absorbers installed, fuel enrichment and fuel burnup).

The void coefficient of reactivity α_φ and the total power coefficient α_N of reactivity were determined experimentally using the applicable techniques, beginning with the startup of Leningrad Unit 1, i.e. from 1973.

For reactors with fuel enriched to 1.8% in uranium-235, the experimental data showed that the void coefficient of reactivity increased as a function of fuel burnup and removal of additional absorbers (AA):

(a) from $-0.22\beta_{eff}$ (211 AA) to $+ 5.1\beta_{eff}$ (32 AA) at Leningrad Unit 1 [15];
(b) from $-0.16\beta_{eff}$ (215 AA) to $+ 4.9\beta_{eff}$ (39 AA) at Chernobyl Unit 1 [16];
(c) from $-0.38\beta_{eff}$ (179 AA) to $+ 5.3\beta_{eff}$ (40 AA) at Chernobyl Unit 2 [16].

It was confirmed experimentally that as α_φ increased, there was a decrease in the time taken for the development of the first azimuthal harmonics, which is a very important parameter characterizing the stability of the reactor power density field and the ability of operating personnel to control the reactor effectively. When α_φ was about $+5\beta_{eff}$, this time decreased to 3 min, making the reactor unstable and difficult to control.

In order to improve reactor stability, it was decided in 1976 to convert the RBMK reactor to fuel enriched to 2% in uranium-235 enriched fuel and to install a local automatic control system. Second generation plants with the RBMK-1000 reactor (Leningrad Units 3 and 4, Kola Units 3 and 4, Chernobyl Units 3 and 4, Smolensk Units 1 and 2) were loaded from the beginning with fuel enriched to 2% in uranium-235. However, even with that fuel enrichment, as fuel burnup increased to 1100–1200 MW·d/t per fuel assembly and with an authorized operating reactivity margin (ORM) corresponding to 26–30 manual control rods, the void coefficient of reactivity approached $+5\beta_{eff}$. There were similar fuel burnups at Chernobyl Unit 4 before the accident.

The Commission notes that all the aforementioned data relate to reactor power levels which are higher than 50% of the nominal level. There were no calculations or experimental data on the value of α_φ for power levels of less than 50% or for various transient and accident conditions.

Measurements of the fast power coefficient of reactivity, characterizing the change in reactor reactivity in response to a change in power, showed that when α_φ increased from $-(0.2\text{–}0.4)\beta_{eff}$ to $+5\beta_{eff}$, α_N changed from $-4 \times 10^{-4}\ \beta_{eff}$/MW(th) to $+0.6 \times 10^{-4}\ \beta_{eff}$/MW(th). However, these data were valid only for power levels of more than 50% N_{nom} [16].

In view of the lack of calculations on coefficients of reactivity at power levels of less than 50%, the Commission is bound to conclude that the designers of the reactor evidently did not expect any dangerous characteristics in the behaviour of the reactor at low power levels and notes that they did not impose any restrictions on reactor operation at low power levels before the accident on 26 April 1986.

The Scientific Manager and Chief Design Engineer of the RBMK-1000 reactor determined the dependence of reactor reactivity on coolant density in the core using calculation codes in order to analyse the development of the design basis accident (DBA). The DBA considered in the design was a rupture in the pressure header of the multipass forced circulation circuit (MFCC) resulting in the loss of the water and steam phases of the core coolant. According to the calculated dependence, during coolant vaporization in the core (reduction of coolant density) the positive reactivity initially increases to $+2\beta_{eff}$, and then the reactivity decreases as the coolant density approaches zero (full steaming of the channels or coolant vaporization in the core) and becomes negative. This leads to the reactor shutting itself down even if the reactor control and protection system does not affect the reactivity. This was why problems of shutting the reactor down in the event of coolant leaks were not considered [12]. In fact, according to calculations made in 1980, 1985 and 1987, when the water in the core is replaced by steam, there is an increase in positive reactivity to $+5\beta_{eff}$ [17], which leads not to the reactor shutting itself down, but to a large increase in positive reactivity and reactor runaway.

There is no safety analysis of the void coefficient of reactivity in the RBMK-1000 design documentation. All plants with RBMK-1000 reactors therefore operated with values of this coefficient obtained in practice, rather than the values established in the design. As has already been noted, α_{φ} depends largely on the composition of the reactor core, which in turn depends on the specific calculation methods and refuelling schemes for each plant. These methods were also not analysed in the design.

Neither the designers, nor the plant operators, nor the regulatory body attached proper importance to the large positive coefficients of reactivity which became apparent from experiments, and they did not attempt to find acceptable theoretical explanations. The obvious discrepancy between the actual core characteristics and the projected design values was not adequately analysed and consequently it was not known how the RBMK reactor would behave in accident situations.

There are a number of explanations for the poor quality of the calculational analysis of the safety of the design. These include the fact that, until recently, Soviet computer techniques were chronically outdated and the standard of computer codes was very low. Three dimensional non-stationary neutron-thermal-hydraulic models are required in order to calculate the physical parameters of an RBMK reactor under different operating conditions. Such models first became available only shortly before the Chernobyl accident and were not really developed until after the accident.

The Commission concludes that the design of the RBMK-1000 reactor and the nuclear physics and thermal-hydraulic characteristics of the core predetermined the positive void and power coefficients of reactivity under steady state refuelling conditions. The nuclear safety of the reactor with such coefficients was not "guaranteed and explicitly proved" either for operation at rated power or for intermediate power levels ranging from the minimum controllable power level to the rated level. This was also not done for transient and accident regimes. As a result of the misguided

selection of the core's physical and design parameters by the designers, the RBMK-1000 reactor was a dynamically unstable system with regard to power and steam quality perturbations. The steam quality, in its turn, was dependent on many parameters characterizing the reactor state.

The Commission concludes that the design and characteristics of the core of the RBMK-1000 reactor violated the requirements of Article 3.2.2 of NSR-04-74 and Article 2.2.3 of GSP-73.

I-3.3.

> Article 3.1.8 of NSR-04-74 states that: *"The reactor alarm system must produce the following signals: Emergency signals (optical and audible signals, including an emergency siren), when the parameters reach the point at which the emergency protection system (EPS) is triggered and when accident deviations in the operating conditions occur; warning signals (optical and auditory signals), when the parameters approach the EPS trip point, when the radiation level exceeds the established limits, and when normal operation of equipment is impaired."*

It is well known that both the information [1] and report [2] presented by the USSR State Committee on the Utilization of Atomic Energy to the IAEA identify the main personnel error as operation of the reactor with an ORM below the established limit. However, the design documentation and research work conducted to analyse the design made no provisions for the ORM to be treated as a parameter which would trigger the alarm system or the EPS when it reached the limit value. Only after the accident, in the Summary of Measures to Improve the Safety and Reliability of Nuclear Power Plants with RBMK Reactors in Operation or under Construction [18] was mention made of the development of an ORM recorder with a display on the control board, and an instrument to produce an emergency signal to shut down the reactor if the ORM reached the emergency trip point.

Similarly, the design made no provisions for an alarm system or a protection system for what were referred to as violations of permissible limits for a number of other parameters. In some cases, as a result of incorrect design decisions, the protection systems did not operate over the whole range of possible reactor operating conditions (see section 4.7.4 of this report) [*Section I-4.7.4*].

The Commission concludes that for a number of the most important parameters, violations of which on 26 April 1986 were considered by the reactor designers to have played a critical role in the initiation and development of the accident, no emergency or warning signals were provided in the design. This is a clear violation of Article 3.1.8. of NSR-04-74.

I-3.4.

Article 3.3.1 of NSR-04-74 states that: *"The reactor control and protection system must ensure that the power (chain reaction rate) is reliably controlled, that the chain reaction is controlled and can be swiftly terminated and that the reactor is maintained in a subcritical state."*

The RBMK emergency protection system (EPS) was designed to compensate for the following reactivity effects [19]:

— coolant vaporization in the fuel channels in a cold reactor state;
— steam collapse in the core when the fuel elements cool to 265°C;
— the possible sticking of some of the emergency protection rods.

The RBMK-1000 designers considered that it was sufficient to take only these reactivity effects into account when designing the EPS. However, these effects do not cover a wide range of different effects which were already known at the early stages of development of the reactor. In particular, the designers ignored the fact that the power and void coefficients of reactivity vary over a wide range from negative to positive values as a function of the core composition and the reactor operating conditions. They also ignored the fact that the design of the control rods was such as to predetermine an initial increase in positive reactivity as they start to move into the core from the upper limit position. The slow speed of the emergency protection system (the time for total insertion into the core from the upper limit position is 18 s) and defects in the design of the rods (i.e. the positive reactivity excursion) resulted in a situation where, for a number of reactor operating modes, the EPS not only did not function, but itself initiated a reactor runaway.

There are grounds to think that the reactor designers did not assess the effectiveness of the EPS in the possible operating modes. Research [20] conducted after the accident has shown that the reactivity introduced into the core by the control rods largely depends on the ORM. When the ORM is about 30 effective manual control rods (approximately 100 control rods, each lowered to the 1.4 m level), a strong negative reactivity is introduced. When the ORM is about 15 control rods, during the first six seconds after triggering of EPS-5, less than β_{eff} of negative reactivity is introduced into the core. In the case of a non-regulation ORM of 7 control rods, during the first eight seconds after triggering of EPS-5 the inserted reactivity is positive (i.e. the chain reaction is being accelerated rather than terminated). The designers were evidently not sufficiently aware of this fact before the accident, otherwise it is difficult to believe that they could have expected to ensure safety by organizational measures such as prohibiting reactor operation at low ORMs given the parameters of the EPS just stated.

The problem of ensuring that the power (chain reaction rate) of the RBMK-1000 reactor is reliably controlled needs to be considered. Two systems are designed to ensure power control: the physical power density distribution control

system (PPDDCS), which has sensors located inside the core, and the reactor control and protection system (RCPS), which has sensors located both in the lateral biological shield tank and inside the core. In principle, the systems were designed to complement each other. However, both systems have significant deficiencies which are most apparent at low power levels. This is because the PPDDCS was designed to control the relative and absolute power density distributions in the range of 10–120% of the nominal levels and to control the total reactor power in the range of 5–120% of the nominal level. The local automatic control and local automatic protection system (LAC-LAP), which received signals from the ionization chambers inside the core, controlled the reactor at power levels greater than 10% of the nominal level. It is very difficult to control such a large reactor as the RBMK-1000 (core diameter 11.8 m, height 7.0 m) at low power levels using only the lateral ionization chambers. This is because at low power levels, when the LAC-LAP system is switched off, the lateral ionization chambers do not provide signals for the central parts of the reactor core and, moreover, do not indicate the axial power density distribution, since all the ionization chambers are located at the mid-plane of the core. Thus, at low power levels a reactor operator has to operate 'blindly', relying more on experience and intuition than on the readings of the control instruments. Although 'blind' control of the RBMK-1000 is to some extent acceptable during startup of a poison free reactor when its power density field is controlled in accordance with preliminary calculations, the situation is different when a non-uniformly poisoned reactor is being shut down. In this case, there is a risk of large field distortions and critically high non-uniformities of both the axial and radial power density distributions. These facts were not taken into account before the accident and, unfortunately, no limitations were imposed on reactor operation at low power levels.

The Commission concludes that the reactor control and protection system of the RBMK-1000 did not meet the requirements of Article 3.3.1 of NSR-04-74 in the context of the actual reactor reactivity effects and the design of the RCPS rods.

I-3.5.

> Article 3.3.5 of NSR-04-74 states that: *"At least one of the planned systems for controlling reactivity must be able to bring the reactor to a subcritical state and maintain it in this state under any normal or accident conditions, even if one of the most efficient reactivity control mechanisms does not work."*

The Commission considers that, as has been shown in section 3.4 of this report [*Section I-3.4*], miscalculations by the reactor designers in determining the reactivity effects which needed to be taken into account in the design of the reactor control and protection system meant that the requirements of Article 3.3.5 of NSR-04-74 were inevitably violated.

I-3.6.

Article 3.3.21 of NSR-04-74 states that: *"The reactor control and protection system must include a fast response emergency protection system (class I EPS) which ensures automatic shutdown of the reactor in the event of an accident situation. The signals and trip points for the emergency protection system must be analysed in the design documentation."*

No safety analysis of the fast response emergency protection system is made in the RBMK-1000 design documentation. The time of insertion into the core was the same for all the RCPS rods (18–21 s). This means that their division into functional groups of emergency protection rods and manual control rods was a notional one. During reactor operation, an emergency protection system rod could be converted into a manual control rod and vice versa without any technical or organizational problems. The aforementioned response parameter was not fast enough for a reactor with a large positive reactivity feedback. It can be presumed that no studies were carried out to determine the required speed of response of the EPS rods in order that the time of core insertion be less than 18 s because the reactivity effects were poorly understood and no thorough analysis had been made of emergency operating conditions, including those with low initial power levels.

Note: Authors of the information [1] submitted to the IAEA note that "the 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 mistakenness of this view was highlighted by the very accident on the subject of which the aforementioned information was prepared. After the accident, a fast response emergency protection system with a time of 2.5 s for total insertion of the rods into the core was developed and installed.

The Commission notes that the requirements of Article 3.3.21 of NSR-04-74 were not met in the design.

I-3.7.

Article 3.3.26 of NSR-04-74 states that: *"The reactor's emergency protection system must ensure that the chain reaction is automatically, quickly and reliably terminated in the following cases:*

- *When the trip point of the emergency power protection system is reached;*
- *When the trip point of the emergency power (or reactivity) increase rate protection system is reached;*
- *In the event of failure of the voltage supply to the RCPS bus bars;*
- *In the event of failure or unavailability of any two of the three channels of the power protection system or of the power increase rate protection system;*

— In the event of emergency signals requiring reactor shutdown;
— When the emergency protection buttons are pressed."

As has been shown earlier, the RCPS of the RBMK reactor, including the EPS, was not designed to meet the requirements of this Article of the Regulations. The list of emergency signals was incomplete and did not ensure that the reactor protection system would be triggered when certain parameters reached dangerous levels (for example, the parameters related to the operational reactivity margin, low power level, and so on).

In addition, it should be noted that the insertion of the RCPS rods into the core from the upper position following actuation by an emergency signal or by pressing the button of the emergency protection system could, depending on the core composition, power density field distribution and operating mode, produce precisely the opposite effect, as a result of design deficiencies in the RCPS rods and the physical parameters of the core. In other words, it could introduce positive reactivity, rather than terminate the chain reaction quickly and reliably (see section 4 of this report [*Section I-4*]). Section 4.6.3 [*Section I-4.6.3*] of the report shows that, given the design characteristics of the reactor and RCPS, the increase in reactor power following triggering of EPS-5 might, under certain conditions, be so large that when the trip points for the power emergency protection system and emergency power increase rate protection systems were reached, it would no longer be possible to terminate the nuclear reaction without substantial damage to the fuel elements. In view of the reactor's poor characteristics for dumping steam from the reactor space, this predetermines its possible destruction.

The design did not provide the reactor space with a protection system against multiple ruptures of the fuel channels. A rupture of more than one of the fuel channels could, therefore, cause lifting of the reactor cover plate and upper structure of the reactor (structure E), followed by failure of the whole RCPS rod insertion system and even withdrawal of the RCPS rods from the core. This leads to the introduction of positive reactivity, not to a fast and reliable termination of the chain reaction.

The Commission concludes that the design of the RCPS for the RBMK-1000 reactor did not comply with the requirements of Article 3.3.26 of NSR-04-74.

I-3.8.

Article 3.3.28 of NSR-04-74 states that: *"The number, location, efficiency and insertion rate of the actuating mechanisms of the EPS must be determined and analysed in the reactor design documentation, where it must be shown that, under any emergency operating conditions and even without the operation of one of the most efficient actuating mechanisms, the actuating mechanisms of the EPS are able to guarantee that:*

— *In an emergency, the rate of reactor power reduction is high enough to prevent the possibility that damage to the fuel elements will exceed the permissible limits;*
— *The reactor can be brought to a subcritical state and maintained in this state, taking into account possible reactivity increases, for long enough to enable other slower actuating mechanisms of the RCPS to be introduced;*
— *No local critical masses are formed.''*

The design of the RCPS for the RBMK-1000 reactor seriously violated this Article at the time of the accident in 1986. The number, efficiency and insertion rate of the EPS actuating mechanisms were selected and analysed without account being taken of the theoretically predicted and experimentally confirmed reactivity effects which could be (and were in the Chernobyl accident) disastrous.

The evolution of the design of the RBMK-1000 reactor is of interest as far as the determination and analysis of the number and efficiency of the reactivity control mechanisms are concerned.

In the RBMK draft design documentation [7], developed in 1965, 212 RCPS rods and fuel enriched to 2% in ^{235}U were foreseen, whereas 179 RCPS rods and fuel enriched to 1.8% in ^{235}U were accepted in the technical design. The draft design provided for RCPS rods in the RCPS with a 7 m long absorber and displacer (i.e. completely covering the core), 68 of which were EPS rods. However, in the technical design the length of the absorber was only 6 m for 146 rods, 5 m for 12 rods and 3 m for 21 rods. The number of EPS rods was reduced to 20 with an absorber length of 6 m. In the final working design there were 179 RCPS rods with an absorber length of 5 m (except for the 21 shortened absorber rods with a 3.5 m long absorber). There were 21 EPS rods for the first stage RBMK reactors, and 24 for the second stage. For the second stage reactors, the total number of RCPS rods was increased to 211 without any change in design. Thus, as a result of a long evolutionary process, a design for the RCPS rods was selected in which the reactivity control mechanisms did not prevent the formation of local critical masses, since by virtue of their design they did not cover the entire height of the core (according to data presented in Ref. [21], the critical height of the RBMK-1000 core varied from 0.7 m to 2.0 m for different states of the core).

Since the absorbing capacity of the graphite displacer, which is connected to the rod via a 'telescope', is lower than that of the water being forced out of the lower part of the channel when an RCPS rod moves down from the upper position, local introduction of positive reactivity in the lower part of the core occurs. At a particular core composition and power density field profile, this could cause the formation of a local critical mass.

The Chief Design Engineer [22] and the Scientific Manager knew about this effect before the accident. It was discovered experimentally in November/December 1983 during the physical startup of Ignalina Unit 1 and that of Chernobyl Unit 4, i.e. almost two-and-a-half years before the Chernobyl accident [23]. The Physical

Startup Commissions proposed some measures to eliminate these negative effects, but none of them were implemented before the accident. The proposals included: limitations on the withdrawal of the manual control rods to their upper limit stop switches; the improvement of the design of the RCPS rods to exclude the lower water column; and the introduction of water film cooling in the channels of the RCPS. The Scientific Manager drew attention to the extremely dangerous nature of this effect. In particular, it was noted that "When the reactor power decreases to 50% (for example, when one of the turbines is switched off), the reactivity margin is reduced as a result of poisoning and distortions of the axial field are observed up to [*power peaking factor*] $K_z \approx 1.9$. Triggering of the EPS in this case may lead to the introduction of positive reactivity. It seems likely that a more thorough analysis will reveal other dangerous situations" [24].

The following proposals were then made, which, had they been implemented, might have prevented the Chernobyl accident on 26 April 1986:

- That the design of the RBMK manual control and EPS rods should be improved to exclude the water column under the displacer when the rods are in the upper position;
- That the transient and accident operating conditions of the RBMK should be analysed carefully, taking the actual calibrated characteristics of the RCPS rods into account;
- That until the aforementioned steps have been taken, changes should be made to the RBMK regulations so that the number of rods which may be completely withdrawn from the core is limited.

The Scientific Research and Design Institute for Power Technology had recognized the possibility of a positive reactivity excursion [22] and had suggested some measures to offset this effect. However, the technical measures were not implemented by the Chief Design Engineer (those measures included increasing the number of shortened absorber rods, increasing the length of the 'telescope', and returning to the initial draft design of the RCPS with rods without displacers but with water film cooling in the RCPS channels). The Chief Design Engineer proposed organizational measures to eliminate the dangerous upper position effect of the RCPS and made the following recommendation: "that the number of rods which may be fully withdrawn from the core (up to the upper limit stop switch) should be limited to 150 for the RBMK-1000 reactor and that the other, partially inserted rods must be inserted into the core by at least 0.5 m" [22].

The recommendation permitted a positioning of the RCPS rods whereby the breeding characteristics in the lower part of the core at a height of 1.2 m would increase when the rods moved simultaneously in response to an emergency protection signal. If this recommendation had been implemented it would have been possible to have an ORM of 3–5 manual control rods, which would have contravened the requirements of section 9 of the Operating Procedures, where the minimum permissible ORM is specified as 15 manual control rods.

One of the proposals made by the Chief Design Engineer to offset the positive reactivity excursion caused by insertion of the RCPS rods involved inserting shortened absorber rods into the lower part of the core in response to an emergency protection signal (proposal No. 264 of 22 February 1977). However, at most of the RBMK units, including Chernobyl Unit 4, this proposal was not implemented. A technical proposal made by the Chief Design Engineer (No. 8794) for an experimental RCPS rod (with an absorber extended to 7 m and an extended telescope) was also not implemented.

The Commission considers that the design of the RBMK-1000 reactor did not comply with the requirements of Article 3.3.28 of NSR-04-74.

I-3.9.

> Article 3.3.29 of NSR-04-74 states that: *"The emergency protection system must be designed in such a way as to ensure that a protection action is normally carried out fully. The design documentation must contain an analysis of any cases where termination of the protection action is permitted in certain instances when the signal triggering the protection system disappears."*

The approach adopted by the Chief Design Engineer to the design of the reactor control and protection system is described in the Technical Design Documentation of the RBMK Reactor Control and Protection System [19], where it is stated that: "the operating conditions of a power plant with an RBMK reactor connected to a power distribution grid where the plant's share is high make *the traditional principle for the design of the RCPS unacceptable when an emergency signal causes rapid insertion of all or part of the RCPS rods to ensure a fast uncontrolled termination of the reaction.* The system developed does not make it possible to reduce the power abruptly, but does permit accelerated controlled reduction of power from the rated value to lower levels and down to those required for the plant's internal load and ensures steady operation of the plant at these levels." In the same document it is also stated that: "Substantially new design solutions have been adopted in the EPS. A complete reactor shutdown by dropping all the RCPS rods is foreseen only for total loss of power to the plant. Under all other accident conditions, the power is reduced in a controlled manner to the specified levels with the required speed."

The Commission did not find in the design documentation any other analyses of the admissibility of terminating a protection action following disappearance of the actuating signal (from the power increase protection system or the power excursion rate reduction protection system).

It can be seen from the foregoing that the reactor designers analysed the operating algorithm of the emergency protection system in terms of the efficiency of the plant's operation in the power supply system, rather than in terms of its ability to ensure nuclear safety, which is the proper function of an emergency protection system.

The Commission concludes that the design of the RBMK-1000 did not comply with the requirements of Article 3.3.29 of NSR-04-74.

I-3.10.

Having considered to what extent the RCPS of the RBMK-1000 reactor at Chernobyl Unit 4 at the time of the accident complied with the safety rules, the Commission considers it necessary to stress that all the design deficiencies of the RCPS were in fact known before the accident. Clear technical proposals to eliminate these deficiencies had been made, such as:

— to lengthen the absorber part of the RCPS rods;
— to lengthen the telescope and displacer of the RCPS rods;
— to install an independent fast response emergency protection system;
— to introduce a number of new engineered protection systems;
— to insert shortened absorber rods into the core in response to an emergency protection signal.

All the above measures were already included in the 'Summary of measures' [18, 25] after the accident. They have been implemented partially and are continuing to be implemented at all RBMK-1000 reactors.

The Commission notes that in addition to the aforementioned violations of Articles 3.1.6, 3.1.8, 3.2.2, 3.3.1, 3.3.5, 3.3.21, 3.3.26, 3.3.28 and 3.3.29 of NSR-04-74, in the design of the RCPS of the RBMK-1000, the design of this system, which is extremely important for ensuring reactor safety, also violated similar requirements of Articles 2.2.5, 2.2.6, 2.2.7, 2.2.8, 2.5.2 and 2.5.8 of GSP-73.

I-3.11.

In addition to the aforementioned violations, the design of nuclear power plants with RBMK reactors contained other safety significant violations of regulations. The Commission believes that it is necessary to focus on one of the issues that has been discussed most widely: namely the lack of containment at Chernobyl Unit 4.

The fourth unit was constructed in accordance with a design developed when the General Safety Provisions of 1973 were in force. In accordance with Article 2.7.1 of GSP-73, a main coolant circuit may be located outside leaktight buildings provided that "should an emergency situation occur, any radioactive substances released are localized in unused leaktight buildings or specially directed if the release is permitted under certain specific conditions." Article 2.7.4 of GSP-73 requires that "if a part of the main coolant circuit or of the auxiliary systems is located outside leaktight buildings, there must be design features to ensure the safety of the public and personnel in the event of a rupture of this part of the circuit."

The design of the second stage of the Chernobyl plant is such that part of the main coolant circuit (70 mm and 300 mm diameter pipelines) is located outside leak-

tight buildings. The buildings which contain this part of the circuit have special deflecting plates which direct the release of the radioactive steam–air mixture into the atmosphere in the event of rupture of the 70 mm and 300 mm diameter pipelines. The radiological consequence of such an accident was estimated to be a dose of 2.1 rem [21 mSv] to a child's thyroid as a result of inhalation of iodine isotopes. These were the grounds given to justify the admissibility of not having a proper accident localization system [11].

The possibility of more serious initial events was disregarded, including accidents in which the reactor space is ruptured and extensive fuel damage is caused by multiple ruptures of fuel channels, resulting in lifting of the upper reactor cover and upper structure of the reactor (structure E), as happened on 26 April 1986.

The Commission thinks that it is necessary to note that there are not sufficient grounds for discussing whether the consequences of the accident on 26 April 1986 would have been markedly reduced if there had been a containment shell, since no serious research has been done in this area. At the same time, it is important to note, firstly, that the lack of a proper accident localization system in the RBMK-1000 reactor indicates that the plant safety philosophy based on the defence in depth principle (the containment shell is the fourth barrier according to this philosophy), which was already widely developed and applied abroad in the 1960s–1970s, was ignored; and, secondly, that the Chernobyl disaster tragically confirmed the cost of violating the multiple containment barrier principle in the design.

I-3.12.

The actual status of the analysis of the nuclear safety of the reactor in 1976, at the time at which a supplement to the Technical Safety Report was published, is reflected in the resolution of 5–6 May 1976 by which a commission of the USSR Ministry of Intermediate Sized Machinery was set up to develop basic initial data for the design of nuclear power plants and refinements of the basic safety provisions for RBMK-1000 reactors. The resolution, in particular, refers to the unresolved difficulties of ensuring the temperature regime of the fuel element cladding and the fuel channels in the event of accidents where the normal cooling water supply is disrupted, taking into account factors such as the effect of an interruption in coolant supply, the variation in neutron power following activation of EPS-5 and release of additional reactivity owing to the void effect. The resolution notes that *an important condition for ensuring safety is the fast termination of the neutron power by means of the emergency protection system, which would compensate for the positive reactivity released during a rapid increase of the steam quality* in the core after rupture and would bring the reaction well below criticality.

The position of the I.V. Kurchatov Institute of Atomic Energy is made clear in the same resolution, where it is stated that *an additional emergency protection system with a faster response time should be developed to compensate for the positive void reactivity effect* in the event of ruptures. The aforementioned commission

recommended that the *Scientific Research and Design Institute for Power Technology, together with the I.V. Kurchatov Institute of Atomic Energy, should study the calculations made by the Kurchatov Institute, perform additional calculations to test the adequacy of the EPS, and formulate appropriate recommendations.* A recommendation was also made that efforts should be made to speed up the calculational and experimental work to analyse the safety system, primarily, with respect to a change in reactivity in the event of a sharp rise in steam quality in the core. Regrettably, the recommendations have still not been implemented, although there is no doubt about the urgency of the proposals made back in 1976.

The aforementioned commission to develop basic initial requirements for the design of nuclear power plants and refinements of the basic safety provisions for RBMK-1000 reactors was set up after an accident at Leningrad Unit 1 on 30 November 1975, which resulted in radioactive releases. The foregoing excerpts show that the members of the commission understood that the accident at the Leningrad plant (the first in the RBMK-1000 reactor series) on 30 November 1975 was a result of the reactor's design characteristics, rather than personnel errors, although the personnel are known to have operated the plant with an ORM well below 15 manual control rods prior to the accident on 30 November 1975. Regrettably, those who were in charge of the operation of plants with RBMK-1000 reactors were not informed of the real causes of the accident.

The officially declared cause of the accident at the Leningrad plant, namely destruction of a fuel channel due to a manufacturing defect, is not very convincing and the foregoing recommendations made by the commission set up by the Ministry of Intermediate Sized Machinery in 1976 demonstrate this fact.

In 1980, the Scientific Research and Design Institute for Power Technology made a study [16], which was subsequently used in the analysis of the safety of the third stage of the Chernobyl plant. The study lists the factors which have a significant effect on nuclear safety and, in particular, it notes the following:

- An increase in the coolant flow rate through a fuel channel impairs the reactor's dynamic properties;
- A decrease in ORM causes a positive shift of all the coefficients of reactivity except the fuel temperature coefficient;
- An increase in fuel burnup causes the coefficient of reactivity to become positive and subsequently to increase;
- An increase in fuel burnup causes an increase in the positive graphite temperature coefficient of reactivity;
- An increase in fuel burnup causes a transition from negative to positive of the total coefficient of reactivity;
- Coolant vaporization in the RCPS cooling circuit gives rise to positive reactivity;
- At low power levels, large irregularities in breeding properties may occur, which can cause large power density distortions with a power peaking factor

of more than 10. This will redistribute the 'weights' of the rods so that the efficiency of the rods in the 'peak' region may be tens of times higher than the efficiencies of those at a distance from it;
- Variations in the axial field profile may cause a change in the weights of the rods which can be partially inserted;
- The coefficients of reactivity for the whole reactor may change as a result of neutron field deformations and consequent redistribution of coolant flow rates along the channels.

The Commission considers that the negative properties of this type of reactor are likely to predetermine the inevitability of emergency situations and that they certainly do not demonstrate that such situations are likely to be extremely rare and would occur only in the event of an extremely improbable combination of reactor operating procedures and conditions adopted by the unit personnel.

Thus, it seems that the reactor designers were well aware of the possible dangerous consequences of the reactor characteristics and understood how the safety of the RBMK-1000 reactor could be improved. This is confirmed by the fact that the main technical measures to enhance the safety of the RBMK-1000 reactor [26] were announced less than a month and a half after the accident. These included:

- Installation of 30 additional absorbers in the reactor core (later the number of additional absorbers was increased to 80);
- Increase in the ORM to 43–48 manual control rods;
- Establishment of the minimum permissible ORM as 30 manual control rods (rather than 15 as was the case before the accident);
- Increase in the number of shortened absorber rods from 21 to 32;
- Insertion of all RCPS rods (except the shortened ones) by 1.2 m into the core (readjustment of the upper limit stop switch);
- Restriction of the movement of the shortened absorber rods to 3.5 m–1.2 m along the deflection angle;
- Recalculation of the ORM every 5 min rather than every 15 min as was the case before the accident;
- A ban on operation of the four main circulating pumps at reactor power levels below 700 MW(th) (this confirms that there was no such ban before the accident).

These measures obviously do not tally with the official version, which blamed the accident entirely on personnel errors.

I-3.13.

The design deficiencies and instability of the physical and thermal-hydraulic characteristics of the RBMK-1000 reactor had been theoretically and experimentally determined prior to the accident on 26 April 1986. However, no adequate remedial

action was taken, firstly, to eliminate the defects and, secondly, to warn the personnel about the consequences of these dangerous characteristics and to provide them with appropriate training in the operation of the reactor, the parameters of which did not comply with the requirements of the technical standards documentation. The designers and authors of the standard operating procedures for the RBMK-1000 reactor did not inform the personnel about the very real danger of a number of reactor characteristics if certain possible personnel actions (including erroneous ones) were taken, because they failed to understand the possible cost of the consequences of personnel actions in operating such a reactor. The safe operating limits and conditions established in the regulations (see section 4 of this report [*Section I-4*]) were not always explicit and intelligible to the personnel, nor were they always analysed. This may have affected the operational safety of the reactor, in the design of which certain protection functions were shifted from engineered features to the personnel. Furthermore, no engineering measures were taken by the designers of the reactor to compensate for the violations of the regulations in the design of the RBMK-1000 reactor. One may assume that, although the designers of the reactor knew about the design deficiencies and physical characteristics of the reactor, they did not make a quantitative evaluation of the possible consequences of those deficiencies and did not understand that they could lead to a disaster.

In general, having reviewed the design documentation, the Commission feels obliged to draw the following conclusions:

— The design of Chernobyl Unit 4 included major violations of the safety standards and regulations in force at the time that the technical design of the second stage of the Chernobyl plant (comprising Units 3 and 4) was approved and authorized;

— The designers did not identify, analyse, verify and approve these violations in the proper way. No technical and organizational measures were developed to compensate for the violations of the safety standards and regulations. Regulations GSP-73 and NSR-04-74 came into force more than 10 years before the accident, during which time Chernobyl Unit 4 was designed, constructed and put into operation. During all that time, neither the Chief Design Engineer, nor the General Designer, nor the Scientific Manager took effective measures to bring the design of the RBMK-1000 reactor into line with the safety standards and regulations. The USSR Ministry of Intermediate Sized Machinery, the USSR Ministry of Power and the Soviet regulatory authorities were just as lax in bringing plants with the RBMK-1000 reactor into line with the safety standards and regulations.

The Commission notes that the design was also not brought into line with the General Safety Provisions (GSP-82) which entered into force in 1982.

I-4. CAUSES AND CIRCUMSTANCES OF THE ACCIDENT

I-4.1. General characteristics of the test programme that initiated the accident at Chernobyl Unit 4

The accident at Chernobyl Unit 4 occurred during testing of the capacity of turbogenerator No. 8 to supply power during its rundown for the unit's internal requirements.

The tests were necessary because one of the most important emergency operating modes had not been properly tested prior to commercial operation of units in this series. The proposal to use the rundown mode of the turbogenerator to supply power for the unit's internal requirements was made by the Chief Design Engineer [27] in order to guarantee forced circulation in the reactor cooling circuit by providing reliable electric power supply to the main circulating pumps and feedwater pumps. The rundown concept was accepted and included in the design of plants with RBMK reactors (see, for example, the Technical Safety Report for the second stage of the Smolensk nuclear power plant: "In a design basis accident, involving total loss of power for the unit's internal requirements, cooling water is fed to the damaged part by feedwater pumps powered by the turbogenerator rundown").

According to the design requirements for total loss of power in the event of a DBA, electric power supply to the feedwater pumps of the third subsystem of the emergency core cooling system (ECCS) had to be provided by the mechanical energy of the rundown mode of the turbogenerator. However, Chernobyl Unit 4 was commissioned in December 1983 without having been tested under these conditions. Such tests should be an integral part of the pre-operational testing of the main design basis conditions carried out at different power levels.

In 1982, the corresponding tests were carried out at Chernobyl Unit 3 under an agreement with the firm Dontekhehnergo with the participation of representatives of the General Designer and the S. Ya. Zhuk Gidroproekt Institute. The tests showed that the requirements, in terms of the characteristics of the electrical current generated by the turbogenerator rundown, could not be met for long enough and that the turbogenerator's excitation regulation system had to be improved.

Additional tests with a modernized rundown unit were performed in 1984 and 1985. The 1982 and 1984 programmes provided for the connection of the turbogenerator in rundown mode to one main circulating pump on each side of the reactor, whereas the 1985 and 1986 programmes provided for connection to two main circulating pumps. The 1984, 1985 and 1986 programmes provided for disconnection of the ECCS by manual isolating slide valves.

The Commission considers that it is wrong to regard these testing programmes as purely electrical ones, since they involve a change in the electricity supply to the unit's essential systems and require interference in the protection and blocking system. Such tests should be classified as complex unit tests and should be approved by the General Designer, the Chief Design Engineer, the Scientific Manager and the

regulatory body. However, regulations NSR-04-74 and GSP-82, which were in force at the time of the accident, did not require the plant managers to obtain approval for such tests from the aforementioned organizations.

The main idea of the programme is to test the design basis conditions as realistically as possible and there is nothing wrong with the programme itself. In the light of contemporary approaches to the development of testing programmes for conducting similar tests at nuclear power plants, the programme documentation in question is not entirely satisfactory, primarily in terms of its safety measures. However, the operating documentation as a whole (regulations and instructions), together with the programme in question, provided sufficient basis for the safe testing of the planned operating conditions. The causes of the accident lie not in the programme as such, but in the ignorance on the part of the programme developers of the characteristics of the behaviour of the RBMK-1000 reactor under the planned operating conditions.

An initial coolant flow rate through the reactor which was higher than the nominal rate was a thermal-hydraulic characteristic of the planned mode. Furthermore, the steam quality was at a minimum for subcooling of the coolant to just below boiling point at the core inlet. Both these factors proved to be directly related to the scale of effects manifested during the tests.

I-4.2. Chronology of the events at Chernobyl Unit 4 on 25–26 April 1986

The Commission based its analysis and conclusions on the following chronological sequence of events [*Table I-I*] which it obtained by studying the sources indicated in section 4.3 [*Section I-4.3*].

I-4.3. Information on recorded data used by the Commission

The Commission analysed the pre-accident and accident processes using data recorded by the following instruments and computerized information systems:

- built-in recording devices with corresponding chart strips;
- built-in SKALA computerized centralized control system, comprising the diagnostic parameter recording program (DREG) and PRIZMA program for calculation of reactor parameters which are not measured directly;
- special oscilloscope system to record important parameters of the turbogenerator rundown.

I-4.3.1. Built-in recording instruments

These instruments are for recording relatively slow processes (the tape speed does not exceed 240 mm/h). They therefore make it possible to record peak values of the parameters concerned fairly reliably, but cannot be used to record fast non-steady processes.

TABLE I-I. CHRONOLOGY OF THE CHERNOBYL ACCIDENT

Time	Event
25 April 1986 *(time in operating log)*	
01:06	Start of reactor power reduction; ORM equals 31 manual control rods
03:45	Start of replacement of the nitrogen—helium gas mixture with nitrogen in the gas cooling system for the reactor graphite stack
03:47	Reactor thermal power is 1600 MW
from 04:13 until 12:36	Sequential measurement of the control system parameters and vibration characteristics of turbogenerator No. 7 and turbogenerator No. 8 at constant thermal power of 1500 MW
07:10	ORM equals 13.2 manual control rods
13:05	Disconnection of turbogenerator No. 7 from the system
14:00	Disconnection of the ECCS from the multipass forced circulation circuit (MFCC)
14:00	Postponement of testing programme requested by Kiev power grid controller
15:20	ORM equals 16.8 manual control rods
18:50	Power supply to auxiliary equipment not involved in the tests switched to working transformer No. T6
23:10	Power reduction continued, ORM equals 26 manual control rods
26 April 1986 *(time on printout of DREG)*	
00:05 (in operating log)	Reactor thermal power was 720 MW
00:28 (in operating log)	At reactor thermal power of about 500 MW transfer made from the local to global main range automatic power control (automatic power controllers Nos 1 and 2). During the transfer there was a reduction in thermal power to 30 MW (neutron power to zero), which was not envisaged in the testing programme. Measures to increase the power were taken

TABLE I-I. (cont.)

Time	Event
26 April 1986 (cont.) *(time on printout of DREG)*	
00:34:03	Emergency fluctuations of water level in steam separator drums
00:43:37 00:52:27 01:00:04 01:09:45 01:18:52	
00:36:24	The EPS trip point in response to a pressure drop in the steam separator drums was changed from 55 to 50 kg/cm^2
from 00:39:32 until 00:43:35	DREG program did not work Personnel blocked the EPS signal which would have shut down the two turbogenerators
from 00:41 until 01:16 (in operating log)	Disconnection of turbogenerator No. 8 from the system to determine the vibration characteristics during rundown
from 00:52:35 until 00:59:54	DREG program did not work
01.03	Reactor thermal power increased to 200 MW and stabilized
01:03 (in operating log)	Seventh main circulating pump was put into operation (MCP No. 12)
01:07 (in operating log)	Eighth MCP was put into operation (MCP No. 22)
from 01:12:10 until 01:18:49	DREG program did not work
01:19:39	'One overcompensation upwards' signal on
from 01:19:44 until 01:19:57	'One overcompensation upwards' signal on
01:22:30	The parameters were recorded on magnetic tape (calculations were performed at the Smolensk plant after the accident using the PRIZMA program; ORM proved to be equal to 8 manual control rods)
01:23:04	'Oscilloscope is on' signal was given, emergency stop valves of turbogenerator No. 8 were closed. The rundown was started of four MCPs: MCPs Nos 13 and 23 (section 8RA) and MCPs Nos 14 and 24 (section 8RB)

TABLE I-I. (cont.)

Time	Event
26 April 1986 (cont.) *(time on printout of DREG)*	
01:23:10	Design basis accident button was pressed
01:23:30	'One overcompensation upwards' signal went off (it lasted 3 min 33 s)
01:23:40 (01:23:39 on teletape)	EPS-5 button was pressed; the EPS rods and manual control rods started to move down into the core
01:23:43	Power excursion rate emergency protection system signals on; excursion period: less than 20 s; emergency power protection system signals actuated; power exceeded 530 MW(th)
01:23:46	Disconnection of the first pair of MCPs being 'run down'
01:23:46.5	Disconnection of the second pair of MCPs being 'run down'
01:23:47	Sharp reduction in the flow rates (by 40%) of MCPs not involved in the rundown test (MCPs Nos 11, 12, 21 and 22) and unreliable flow rate readings of the MCPs taking part in the rundown (MCPs Nos 13, 14, 23 and 24); sharp increase of pressure in the steam separator drums; sharp increase in the water level in the steam separator drums; signals 'failures of measuring systems' from both main range automatic controllers (automatic power controllers Nos 1 and 2)
01:23:48	Restoration of flow rates of MCPs not involved in the rundown test to values close to the initial ones; restoration of flow rates to 15% below the initial rate for the MCPs on the left side which were being 'run down'; restoration of flow rates to 10% below the initial rate for MCP No. 24; unreliable readings for MCP No. 23; further increase of pressure in the steam separator drums (left side 75.2 kg/cm^2, right side 88.2 kg/cm^2) and of water level in the steam separator drums; triggering of fast acting systems for dumping of steam to condensers Nos 1 and 2
01:23:49	Emergency protection signal 'Pressure increase in reactor space (rupture of a fuel channel)'; 'No voltage = 48 V' signal (no power supply to the servodrive mechanisms of the EPS); 'Failure of the actuators of automatic power controllers Nos 1 and 2' signals
	From a note in the chief reactor control engineer's operating log: "01:24: Severe shocks; the RCPS rods stopped moving before they reached the lower limit stop switches; power switch of clutch mechanisms is off"

I-4.3.2. SKALA centralized control system and subsystems

The system was designed to calculate the basic reactor parameters every 5 min, this period being determined by the power of the V-3M type computer. This period is also unsuitable for analysing fast processes.

The DREG program is very comprehensive and has a good time resolution. It samples and records several hundred discrete and analogue signals. The time of input into the computer of information on directly measured parameters is less than 1 s. However, the DREG program does not record important reactor parameters, such as power, reactivity and channel coolant flow rates. It records the position of only nine of the 211 RCPS rods, including one rod from each of the three automatic control groups. These parameters are not measured directly and therefore the sampling interval is much longer (1 min). Notwithstanding the short recording time of some parameters (1 s), the sampling time may be rather uncertain because the DREG program has one of the lowest priorities in the SKALA centralized control system. In addition, during the last hour before the accident there were three interruptions in the operation of the DREG program, associated with restarting of the SKALA system. These caused an additional loss of information. Other subsystems of the SKALA system, including the PRIZMA program and magnetic tape records of reactor conditions (RESTART) have a long cycle (5 min). There were also interruptions in these systems caused by restarting of the system and the software characteristics. Furthermore, the results of the PRIZMA program are recorded only on printouts.

I-4.3.3. Oscilloscope system

A special oscilloscope system for fast changing parameters was installed in accordance with the testing programme.

This made it possible to obtain performance parameters of certain equipment items with a good degree of accuracy: turbogenerator No. 8, MCP No. 18, feedwater pump No. 4, sections 8RA and 8RB. One shortcoming of this system is the lack of synchronization between the electrical parameters and the reactor parameters recorded by the SKALA system. However, the documents available — the results of processing the electrical parameter oscillograms and recordings of the DREG program — make it possible to synchronize the reference events fairly accurately. The main events were the closure of the turbogenerator No. 8 stop valves and the time at which the EPS-5 button was pressed by the operator.

According to the DREG records, the time at which the stop valves of turbogenerator No. 8 closed was 01:23:04. This time may be determined by the change in certain parameters on the oscillogram. The signal to actuate emergency protection system No. 5 occurred at 01:23:40, as can also be seen from the oscillogram. Changes in the electrical parameters are recorded on the oscillogram with a high degree of accuracy, and it is therefore possible to determine the times at which the DBA button was pressed and the MCPs were disconnected. The first pair of MCPs

being run down was disconnected at 01:22:46, and the supply to the second pair of MCPs was cut 0.45 s later. This means that these events occurred 6.00 s to 6.45 s after the EPS-5 button was pressed by the operator.

An analysis of the oscillogram shows that the DBA button was pressed 6.6 s after closure of the turbogenerator No. 8 stop valves.

> **Note:** The DBA button is a button specially installed for test purposes in order to simulate a DBA signal, to transmit it to the startup system of diesel generator No. 6 with a step by step load increase circuit, and to connect the turbogenerator No. 8 rundown unit being tested.

I-4.4. Mathematical modelling of the pre-accident and accident processes

The Commission notes that, however comprehensive and reliable the information on reactor parameters recorded by the instruments in an accident situation is, mathematical modelling of the pre-accident and accident processes is needed in order to analyse this type of accident. It is necessary not only in order to fill the gaps in the recorded information and to make extrapolations in the area of the parameters that were not measured, but also in order to clarify the sensitivity of the results in the case of certain very important initial parameters. Without this, it is also impossible to judge the adequacy of the subsequent measures to prevent future accidents.

Having analysed the available information, the Commission is of the opinion that no complex mathematical model has yet been developed which is properly suited to the RBMK-1000 reactor and which has been checked against experimental data. The Scientific Research and Design Institute for Power Technology, the I.V. Kurchatov Institute of Atomic Energy, the All-Union Scientific Research Institute for Nuclear Power Plant Operation, the Kiev Institute of Nuclear Research and certain other organizations have various models covering the different phenomena necessary for the analysis. There are also mathematical models in some foreign organizations and the results of calculations using these models have been discussed with Soviet experts.

By compiling the calculation results of the various fragments of the pre-accident and accident processes which do not contradict each other and which agree with the experimental data available, it now appears that a fairly realistic picture of the accident can be obtained.

One of the first calculational studies after the accident was performed at the I.V. Kurchatov Institute of Atomic Energy using a one dimensional model [3] in which the dependence of reactivity on the position of the RCPS rods was derived from a spatial model. However, despite a generally satisfactory description of the major events starting at 01:19, it is a rough model, since it does not provide a detailed description of the process in the core and consequently cannot give reliable results on the behaviour of the reactivity, power and other parameters. This is confirmed by the fact that there are discrepancies between the calculations and the

recorded data (the absence in reality of a '1 Overcompensation — downwards' signal at 01:23:38, the underestimated value for the flow rate through the MFCC at 01:23:43, etc.), as well as by the lack of correspondence between the behaviour of the reactivity parameter and that of the power parameter.

A version of the distributed high speed neutron physics RBMK model was developed and used in calculational studies of the accident at the Kiev Institute of Nuclear Research. It describes the neutron transport using a non-stationary one-group diffusion equation which is solved on a large grid with 50 cm spacing. The coolant density and the movement of the RCPS rods are taken into account by adjustment of the constants, and the change in temperature is introduced as feedback via the temperature coefficient of reactivity. The results of prognostic calculations (REFUELER) are used to represent the axial distribution of fuel assembly burnup. The one-group constants are drawn from the two-group ones calculated by the WIMS program. This model was used in the DIKRUS software package developed at the All-Union Scientific Research Institute for Nuclear Power Plant Operation as a high speed neutron physics calculation code [28]. The code was used to study the drop mode of the EPS-5 rods for the reactor conditions at Chernobyl Unit 4 on 26 April 1986 at 01:22:30.

The combination of a satisfactory description of the power density kinetics and satisfactory thermal-hydraulic description of the thermal inertia of the fuel elements and of the steam generation growth in the core, together with the preliminary painstaking adjustment of the model using the distributed initial data, makes this one of the best models currently available for analysis of the Chernobyl accident.

The results of the simulation of the process do not contradict the data recorded by the DREG program during the last 9 s (the signals of the power increase rate protection and power protection systems, the increase of pressure and water levels in the steam separator drums and the pressure increase in the reactor space 3, 6 and 9 s after pressing of the EPS-5 button).

However, this model still cannot be regarded as suitable enough for the RBMK reactor, since the one-group approximation in the non-stationary equation on a coarse difference grid probably does not yield the necessary accuracy for this type of reactor. Furthermore, the data taken as initial data were those (on the location of the RCPS rods, currents of the sensors of the physical power density distribution control system, etc.) recorded 1 min 10 s before the start of the conditions being studied. It was about this time that intensive feeding of the steam separator drums stopped, and 34 s later the turbine stop valves were closed. Thus, by the time that the EPS-5 button was pressed at 01:23:40 the parameters might have changed. Nevertheless, the Commission considers that the results of this study are among the most comprehensive ones so far, that they do not contain any major unrealistic assumptions and do not contradict the results of other fragmentary studies. They therefore have a good claim to be taken as a basis for the analysis of the processes that occurred.

The STEPAN computer code developed at the I.V. Kurchatov Institute of Atomic Energy [29] may be considered to be the most advanced of the Soviet neutron physics models of the RBMK reactor.

It is used to solve non-stationary two-group diffusion equations of neutron transport in a three dimensional geometry taking 18 groups of delayed neutrons into account (6 groups each for ^{235}U, ^{239}Pu and ^{241}Pu). The two-group diffusion constants of the RBMK operating cells are presented as a function of five variables: fuel burnup, coolant density, fuel and graphite temperatures and xenon concentration. The initial values of the constants are calculated by the WIMS code.

The Commission notes that no detailed analysis of the origins and development of the accident using the STEPAN code as a neutron physics component of the mathematical model has been made in which the effects of all the factors are considered (critical ORM values, coolant subcooling at the core inlet, etc.).

In addition to the aforementioned characteristics and shortcomings of the different calculation models, there are difficulties caused by incorrect initial data in applying even the most advanced of them. This is because there are no calculations of the axial isotope composition (power density) distribution in the working channels of the built-in centralized control system. The distribution is therefore calculated as a function of the total fuel assembly power density, without taking account of the specific operating conditions of the fuel assemblies. This means that the transient distribution of ^{135}Xe immediately before the onset of the accident process cannot be correctly taken into account. The effect of these factors on the distributed models may well be significant. Hence, the accuracy in determining the parameters of the reactor state (neutron fluxes, power, reactivity, temperature, etc.), the times of events (attainment of maximum reactivity or prompt criticality, temperature limits, etc.) and co-ordinates (maximum neutron flux, power density, fuel damage, etc.) is reduced.

It is the Commission's view that the work on improving mathematical simulation techniques for the RBMK reactor, on verifying them and performing calculational analyses of the Chernobyl accident is being carried out extremely slowly and is regarded as a low priority. As a result, no sufficiently representative quantitative analysis has yet been conducted that takes full advantage of the possibilities of contemporary computer technology and developments in RBMK reactor physics.

I-4.5. Scenarios and possible causes of the accident

The first official version of the accident was formulated on 5 May 1986 at the Chernobyl plant by the Interdepartmental Commission under the Chairmanship of the First Deputy Minister of Intermediate Sized Machinery, A.G. Meshkov [30]. It stated that the accident at Chernobyl Unit 4 was caused by an uncontrolled reactor runaway as a result of steaming of the fuel channels in the core following a break in circulation in the MFCC. The break in circulation resulted from a mismatch between the flow rates of feedwater and coolant in the MFCC.

Somewhat earlier, on 1 May 1986, V.P. Volkov, head of the I.V. Kurchatov Institute Research Group on the Reliability and Safety of Nuclear Power Plants with RBMK Reactors, provided A.P. Alexandrov, Director of the I.V. Kurchatov Institute of Atomic Energy, and subsequently the leaders of the country, in a letter of 9 May 1986, with another version of the accident which attributed it "not to the actions of the operating personnel, but to the design of the reactor core and a lack of proper understanding of the neutron physics processes going on in it." This version suggested that the accident was caused by a positive reactivity excursion following the insertion of the RCPS rods as a result of their defective design and the large positive void coefficient of reactivity.

A subsequent, more detailed, analysis of the thermal-hydraulic operating mode of the MCPs made at the end of May 1986 by the Mechanical Engineering Experimental Design Office (designers of the MCPs), the S.Ya. Zhuk Gidroproekt Institute and the F.E. Dzerzhinskij All-Union Heat Engineering Institute did not confirm the theory about cavitation and disruption in the circulation of the MCPs [31]. It was established that the lowest MCP cavitation margin occurred at 01:23:00, i.e. approximately 40 s prior to the runaway of the reactor, but that it was higher than that at which a disruption in circulation of the MCPs could take place.

At the end of May 1986, after having analysed the available data and performed various calculations, a group of experts of the USSR Ministry of Power (A.A. Abagyan, Yu.N. Filimontsev, V.S. Konviz, V.Z. Kuklin, B.Ya. Prushinskij, G.A. Shasharin, A.S. Surba and V.A. Zhil'tsov) sent an addendum to the Report on the Investigation of the Accident [32], in which they attributed the causes of the accident to the fundamentally faulty design of the RCPS rods; the positive void and fast power coefficients of reactivity; the large coolant flow rate in conjunction with a low feedwater flow rate; violation of the ORM limit by the personnel, with consequent low power level; inadequate safety features in the design and inadequate operating information for the personnel; and lack of indications in the design documentation and the technical regulations on the danger of ORM violations.

At two meetings of the Interdepartmental Science and Technology Council, chaired by A.P. Aleksandrov (2 June 1986 and 17 June 1986), insufficient attention was paid to the calculations made by the All-Union Scientific Research Institute for Nuclear Power Plant Operation which demonstrated that the accident was largely due to deficiencies in the reactor design. In fact, all the causes of the accident were reduced exclusively to personnel errors. The decisions taken by the Interdepartment Science and Technology Council paved the way for the one-sided presentation of information on the causes and circumstances of the accident submitted to the IAEA, a wide range of specialists and the general public.

No mention was made in the report presented by the USSR at the meeting of IAEA experts in Vienna in August 1986 [33] of the version suggesting a disruption in circulation of the MCPs. That report states that: "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 event initiating the accident was not

mentioned. However, the main events of the accident process were presented in the following way'' (see Ref. [33], p. 309).

> ''By the beginning of the tests, namely at 01:23, the reactor parameters were closer to being stable that at any other time. Closure of the turbine emergency stop valve resulted in a slow increase in steam pressure in the steam separator drums at a rate of about 6 kPa/s. At the same time, the coolant flow rate began to fall owing to the fact that four of the eight MCPs were in 'rundown' mode. A minute earlier (at 01:20) an operator reduced the feedwater flow rate.''
>
> **Note by the Commission:** In fact, the feedwater flow rate was brought back to the average value corresponding to a reactor power of 200 MW and equal to about 120 t/h for each side of the reactor.
>
> ''The reduction in flow rate of coolant through the reactor and of feedwater in the steam separator drums, despite the competing effect (in terms of steam generation) of the pressure increase, finally resulted in an increase in reactor power, since there is a positive feedback in the reactor between power and steam generation. Under the test conditions, the steam quality in the core was negligible before the start of the turbogenerator rundown, and it increased much more than during normal operation at rated power'' ([33], p. 309).
>
> ''The increase in power might have prompted the personnel to press the EPS-5 button. Since, in violation of the Operating Procedures, the personnel had withdrawn more than the permissible number of manual control rods from the core, the EPS rods were not effective enough and the total positive reactivity continued to grow'' ([33], p. 311).

As can be seen from the foregoing official version, the closure of the turbine emergency stop valve was the event which initiated the accident, in other words the beginning of the rundown test, exacerbated by the reduction in the feedwater flow rate.

The Commission notes that the official scenario contains no supporting calculations and no additional data. In particular, in the study carried out by experts in the United States of America [34] on the basis of information presented by Soviet specialists to the IAEA, it is shown that the calculations do not confirm the conclusion on the change in power and the explosion during the tests. The same conclusion is contained in the report by the Scientific Research Institute for Power Technology [17] issued in 1990 and in the paper by its Director, E.O. Adamov [35].

In 1986, the I.V. Kurchatov Institute of Atomic Energy analysed possible accident scenarios which could have involved a rapid significant increase in reactivity in the reactor [3].

The aim of the analysis was to identify the contradictions between the expected effect of the accident scenario being considered and the available objective data recorded by the DREG program.

There are 13 such scenarios which have been proposed by different experts at different stages of the investigation into the accident:

(1) Hydrogen explosion in a steam separator drum;
(2) Hydrogen explosion in lower tank of the RCPS cooling circuit;
(3) Sabotage (explosion of an explosive device resulting in destruction of MFCC piping);
(4) Rupture of MCP pressure header or of the distribution group header;
(5) Ruptures of steam separator drum or steam–water lines;
(6) Effect of RCPS rod displacers;
(7) Failure of automatic controller;
(8) Serious operator error in controlling the manual control rods;
(9) MCP cavitation leading to a steam–water mixture supply to fuel channel;
(10) Cavitation of the throttle regulating valves;
(11) Steam capture from steam separator drum to downcomers;
(12) Steam–zirconium reaction and hydrogen explosion in the core;
(13) Penetration of compressed gas from the cylinders of the ECCS.

The study by the I.V. Kurchatov Institute of Atomic Energy shows that all the aforementioned scenarios except one (No. 6) contradict the objective data available.

With reference to this analysis, the Commission notes that, according to the calculations of the All-Union Scientific Research Institute for Nuclear Power Plant Operation [28], given the initial state of the reactor just before the tests and the inherent large positive void coefficient of reactivity of the reactor, a major coolant leakage (with a diameter of more than 300 mm) from the MFCC could have resulted in an accident which would have been just as serious. For some time during the investigation of the accident, it was in fact assumed that the MFCC had been damaged, as a result, for example, of increased vibration of the MCPs caused by their possible cavitation. However, the coolant leakage scenarios (see Nos 3, 4 and 5 in the list) were discarded because in those cases the readings of the instruments indicating the pressure and water levels in the steam separator drums and a number of other parameters would have been different. Furthermore, examinations of the MFCC buildings of Chernobyl Unit 4 carried out during the years following the accident did not reveal any damage to the circuit which could have been the initiating event for the accident.

In view of the foregoing, the Commission thinks that a thorough study should be made of the accident scenario concerning the reactivity effect of the RCPS rod displacers, caused by their design, together with the whole sequence of technical operations during the turbogenerator rundown tests, taking the physical characteristics of the RBMK-1000 reactor into account. No assumptions about improbable events are needed for this scenario.

In a letter of 26 March 1990 signed by the Deputy Director of the I.V. Kurchatov Institute of Atomic Energy, N.N. Ponomarev-Stepnoj, the Director of the Scientific Research and Design Institute for Power Technology,

E.O. Adamov, and the Director of the All-Union Scientific Research Institute for Nuclear Power Plant Operation, A.A. Abagyan, this scenario was not rejected, as can be seen from the following wording:

> "The accident occurred as a result of bringing the reactor to a state which did not conform to the operating regulations. There were several reasons for this, the main ones being: the reduction in the operational reactivity margin to below the value established in the Operating Procedures and the fact that the coolant was only slightly subcooled at the core inlet. *These conditions led to the manifestation of the positive void coefficient of reactivity, deficiencies in the design of the RCPS rods, and an unstable neutron field profile as a result of a complex transient.* The accident ended with a prompt neutron runaway reaction" [36].

No mention is made in the foregoing extract of any external thermal perturbations which might have indicated the major negative effect of the positive void coefficient of reactivity, on the existence of which the Scientific Research and Design Institute for Power Technology insists [17]. Additional work on this obvious unresolved contradiction must be carried out if we are to clarify the causes of the accident.

The Commission does not have a mathematical model describing the course of the accident. However, on the basis of the instrumental measurements made at the damaged unit and the fragmentary calculations carried out and published by other organizations, it considers that it is possible and useful to present the following scenario of the pre-accident and accident processes, together with evaluations of the personnel actions and the effect of the reactor characteristics.

I-4.6. The Commission's scenario on the causes of the accident

I-4.6.1. Period of normal operation and preparations for the tests

The initial reduction of the power of the unit which was commenced at 01:06 on 25 April 1986 and the continued reduction below 720 MW(th) after 00:00 on 26 April did not play an initiating role in the accident, although two violations of the Operating Procedures occurred during that time: reactor operation with an ORM below the permissible value and disconnection of the ECCS.

After 00:28 on 26 April a most important safety significant event occurred. In transferring from the local automatic control system for the core distribution of the power density to the global automatic power regulator, the senior reactor control engineer failed to eliminate fast enough the imbalance that appeared in the measuring part of the global automatic power regulator and allowed the thermal power of the reactor to fall from 500 MW to 0–30 MW (approximately).

Following the unsuccessful attempts to control the reactor, it proved necessary for some of the ORM rods to be withdrawn in order to compensate for the additional

negative reactivity caused by xenon poisoning of the core at low power and during the subsequent power rise to 200 MW. In doing this, the Commission believes that the personnel brought the reactor into a state which did not comply with the operating regulations and in which the emergency protection system could no longer guarantee the termination of the nuclear reaction (see section 4.8 of this report [*Section I-4.8*]). It is not clear whether or not the personnel realized the serious consequences of their actions.

During the period from the beginning of power rise to stabilization of the reactor parameters at 200 MW at about 01:23, normal technical processes and operations were taking place at the unit (except for the switching on of the fourth MCP pair). These included triggering of the fast acting system for dumping steam to the condenser, manual control of the water level in the steam separator drums, overcompensation of the reactor, etc.

At 01:22:30 the reactor parameters were recorded on magnetic tape by the SKALA centralized control system. No operating calculations were performed by the PRIZMA program at this time. These calculations were made after the accident using the magnetic tape from the centralized control system and the PRIZMA-ANALOG code at the Smolensk nuclear power plant. The control room personnel and the SKALA system personnel did not have the operating calculation results and did not know the calculated parameters, including the ORM, at this time.

As part of the evaluation of the causes and scale of the accident, the Commission notes the following characteristics of the operating conditions at this time.

There was a double peaked axial power density distribution over most of the core, the neutron flux being higher in the upper part of the core [20]. This was a perfectly normal distribution for the particular state of the reactor at that time: partially burnt up core, almost all control rods in the upper position and xenon poisoning higher in the central parts than at the edge of the core [1, 33]. As calculations have shown [32, 37], this distribution was a very unfavourable one in terms of kinetic stability in combination with the design of the RCPS.

The thermal-hydraulic operating conditions of the core were characterized by a low level of subcooling of the coolant below the boiling temperature (3°C) and a correspondingly low steam quality, which was observed only in the upper part of the core [28]. Under these circumstances, in view of the low level of subcooling of the coolant below the boiling temperature, a small power increase (for whatever reason) could result in a much higher increase in the volumetric steam quality in the lower part of the core than in the upper part.

Before the tests, the core parameters were therefore such as to increase the reactor's runaway susceptibility in the lower part of the core. The Commission believes that this situation was created not only as a result of a higher than normal flow rate of coolant through the reactor (because eight instead of the usual six MCPs were in operation, and an increased flow rate prevents steam generation), but primarily as a result of the low reactor power level. Similar thermal-hydraulic parameters could occur during any power reduction of the reactor.

The unit's initial condition immediately before the tests at 01:23 was characterized by the following parameters: power of 200 MW(th), ORM (value obtained using the PRIZMA-ANALOG code for the state at 01:22:30) equal to 8 manual control rods, double peaked axial neutron field with a maximum above, coolant flow rate of 56 000 m^3/h, feedwater flow rate of 200 t/h and thermophysical parameters close to stable values.

The unit shift supervisors considered that preparations for the tests had been completed and, having switched on the oscilloscope, gave the order to close the emergency stop valves. These were closed at 01:23:04.

At this time, and for the following approximately 30 s of rundown of the four MCPs, the parameters of the unit were controlled, remained within the limits expected for the operating conditions concerned, and did not require any intervention on the part of the personnel.

However, *it would appear that, under the conditions of reduced ORM, as of 00:30 on 26 April 1986 it would have been impossible to use the emergency protection system of the reactor of this particular design, either in response to emergency signals or manually after completion of the tests, without damaging the core.* This will have to be confirmed by further studies.

I-4.6.2. Implementation of the test programme

The tests, which started at 01:23:04, caused the following processes in the reactor.

The rotational speed and delivery of the MCPs powered from turbogenerator No. 8, which was being run down (MCPs Nos 13, 14, 23 and 24), were reduced. Delivery of the other MCPs (MCPs Nos 11, 12, 21 and 22) was slightly increased. The total coolant flow rate began to fall. Thirty-five seconds after the start of the transient it had fallen by 10–15% of the initial value.

The reduction in coolant flow rate led to a corresponding increase in steam quality in the core, which was to some (small) extent offset by the increase in pressure following the closure of the emergency stop valves of turbogenerator No. 8.

This stage of the process has been mathematically modelled by experts in the USSR [32] and in the USA [34]. The theoretical predictions of the integral parameters agree well with the values actually recorded. Both calculations showed that the released void reactivity was negligible and could have been compensated for by insertion for a short distance (up to 1.4 m) of the EPS rods into the core.

During the rundown of turbogenerator No. 8 there was no increase in reactor power. This is confirmed by the DREG program, which from 01:19:39 until 01:19:44 and from 01:19:57 until 01:23:30 (i.e. prior to and for a substantial period during the tests) recorded the 'One overcompensation upwards' signal, at which time the automatic control rods could not move into the core. Their positions, recorded for the last time at 01:22:37, were 1.4, 1.6 and 0.2 m for automatic regulators Nos 1, 2 and 3 respectively.

Thus, neither the reactor power nor the other parameters (pressure and water level in the steam separator drums, coolant and feedwater flow rates, etc.) required any intervention by the personnel or by the engineered safety features from the beginning of the tests until the EPS-5 button was pressed.

The Commission did not detect any events or dynamic processes, such as hidden reactor runaway, which could have been the event which initiated the accident. The Commission identified a rather extended initial reactor state, during which, if positive reactivity had occurred for any reason, there could have been a power excursion under conditions in which the reactor's EPS would be unable to perform its functions.

I-4.6.3. Development of the accident

At 01:23:40 the senior reactor control engineer pressed the manual emergency stop button (EPS-5).

The Commission was unable to establish why the button was pressed.

Since the rate of development of subsequent events is not compatible with the resolving power of the reactor parameter recording devices, further analysis is only possible using theoretical concepts based on the instrument readings with the necessary time corrections which have to be made because of the characteristics of the recording system (information on which is given in section 4.3 of this report [*Section I-4.3*]).

Reconstruction of the power density field by means of physical calculations [28] has confirmed the axial distribution with an acceptable degree of accuracy and demonstrated that there was also considerable lack of uniformity in the radial power density distribution (power peaking factor of 2.0). Thus, the initial power density distribution over the whole core was extremely non-uniform [20, 28].

There is very satisfactory agreement between the various mathematical models of the power density kinetics performed independently by different organizations [21, 35]. No results have been found which contradict the foregoing conclusions. This enables us to interpret what had happened in the following way.

The movement of the EPS and manual control rods in response to the EPS-5 command caused significant additional deformations of the power density profile. The neutron flux began to decrease in the upper layers of the core where insertion of the absorber parts of the EPS and manual control rods started. In the lower sections of the core where displacement of the neutron absorbing water columns started, the neutron flux increased.

The reactor power recorder, which displays the total current of the lateral ionization chambers located outside the core, recorded a small decrease in power and a subsequent increase. The calculations in Refs [21] and [35] both show that during the subsequent period practically all of the power density shifted to the lower part of the core at a height of about 2 m. Both calculations show that the linear heat flux in the lower parts of the fuel elements increased repeatedly at different rates in differ-

ent parts of the core cross-section. The calculations show that the local growth in power density, after the EPS-5 button had been pressed, was such that the integral reactor power increased in comparison with the initial value several tens of times during a period of about 5 s. The calculations in Refs [17] and [20] show the appearance of all lateral ionization chamber signals only 3 s after pressing of the EPS-5 button. No information on these signals is provided in the calculations in Ref. [28], possibly owing to a lack of interest in this parameter.

The total lack of black absorbers (only one additional absorber) in the core, and the presence of saddle points in the power density profile in many parts of the core, which cause kinetic instability of the axial field, particularly in the event of the introduction of negative reactivity in one part and of positive reactivity in the other, produced strong power density deformations in the reactor volume [17, 20, 28].

From the foregoing results, it follows that, under the initial neutron field conditions, once the movement of the EPS and manual control rods had started, it was bound to cause strong power density deformations in the core with extremely high non-uniformity parameters.

According to the calculations in Ref. [28], the volumetric peaking factor was $K_v = 5.5$. Taking into account the fact that the initial power in the core increased by about 30 times (according to the same calculations), the linear heat flux in the regions of greatest intensity were many times higher than the nominal values at 100% reactor power. Therefore, the fuel element enthalpy reached critical values in some of the fuel channels in the lower part of the core, resulting in different degrees of damage to the fuel elements.

As is shown in the study performed by Japanese experts [35] based on direct experimental research, the fuel elements begin to disintegrate at an enthalpy of 220 cal/g UO_2 (T = 3300 K) [*1 cal = 4.184 J*]. At an enthalpy of 285 cal/g UO_2, the fuel elements rupture, and at 320 cal/g UO_2 they explode and disperse in small pieces.

> **Note:** The RBMK reactor fuel elements are not entirely identical with the fuel elements used by the Japanese scientists in their experiments. However, the Commission believes that although there may be quantitative discrepancies between the critical enthalpies of the model and the real fuel elements, these do not alter the basic conclusion about the disastrous disintegration mechanism, which is also referred to in Refs [1] and [38].

The results of the calculation studies performed four years after the accident by leading organizations in the area of reactor physics, such as the Scientific Research and Design Institute for Power Technology, the All-Union Scientific Research Institute for Nuclear Power Plant Operation, the I.V. Kurchatov Institute of Atomic Energy and the Kiev Institute of Nuclear Research [17, 28], have demonstrated the possibility that insertion of the EPS rods into the reactor may cause a dangerous increase in power of the RBMK-1000 reactor and a multiple increase in local power densities in the core.

As can be seen from the foregoing, *the event which initiated the accident was the pressing of the EPS-5 button when the RBMK-1000 reactor was operating at low power with a greater than permissible number of manual control rods withdrawn from the reactor.*

Note:

Since, as far as the Commission is aware, the possibility of the accident scenario presented here is not currently disputed by any organization and, moreover, it fully corresponds to the accident sequence formulated by the directors of three leading institutes — the I.V. Kurchatov Institute of Atomic Energy, the Scientific Research and Design Institute for Power Technology and the All-Union Scientific Research Institute for Nuclear Power Plant Operation [36] — we think that in order to conclude the analysis, we can present a scenario of the subsequent events which is no longer based on calculations.

Using the data in Ref. [35] relating to the destructive forces of the disastrous process and the data presented in section 3 of this report [*Section I-3*] on the design and characteristics of the reactor, the scenario of the accident process may be described in the following way.

Ruptures and destruction of certain parts of the fuel elements in a limited zone of the reactor caused by large local heat releases led to increased steam generation owing to the direct contact of water with the fuel matrix, and to increased pressure in the corresponding parts of the fuel channels and their destruction as a result of the direct contact of the fuel with the channel tube and local increase in pressure [35].

During the initial phase of the reactor runaway, the specific neutron field configuration with saddle points in the middle sections of the core (this is an objective and unavoidable factor for many reactor states [32]), and the presence of more than the permitted number of water columns in the lower parts of the core (which occurs rarely, is subjective and permitted by personnel) were decisive factors. Once a certain thermal inertia of the fuel elements in the region of highest power densities had been overcome, steam generation began, which, owing to the large local void coefficient of reactivity, made a significant contribution to the non-uniform runaway of the core and rapid fuel damage in the regions of greatest power intensity.

After the initial phase of neutron flux redistribution, caused by the design of the RCPS rods and independent of the thermal-hydraulic state of the reactor and the MFCC, the increase in power density caused the large void reactivity effect inherent in the RBMK-1000 reactor design. With the appearance and growth of steam generation, the zone of increased power density began to expand out of control over the whole core.

Substantial confirmation of the local nature of the initial stage of the runaway is provided by the non-uniform growth in pressure in the left and right steam separator drums. The rapid change in many of the general parameters (signals of the power increase rate protection system and power protection system, increase in pressure, signal indicating an increase in pressure in the reactor space) demonstrates the rapid transition of the local runaway into a general one.

Owing to the design characteristics of the reactor, substantial damage to even a few fuel assemblies (three or four assemblies are enough) can, and in this case did, result in destruction of the reactor itself and failure of its emergency protection system. The rupture of the pipes of several fuel channels caused an increase in pressure in the reactor space and a partial detachment of the reactor support plate from the shroud and consequent jamming of all the RCPS rods, which by that time were only half-way down.

The destruction of fuel channel pipes, initiated at first by only a local neutron power surge, intensified by steam generation in a limited zone of the reactor, caused a new effect from the time at which the channel pipes started to rupture, namely mass generation of steam over the whole core as a result of depressurization of the reactor cooling circuit and release of the large total void reactivity effect inherent in the reactor. However, no DBA signal to bring the ECCS into operation, once depressurization of the MFCC started, was given because the point at which the MFCC ruptured was not in the reinforced leaktight compartments where the sensors are placed, but in the core itself.

Subsequently, intense steam generation in the reactor space played a significant role.

The Commission notes that there are studies which contain enough information to clarify the physical processes which occurred in the core during the initial stage of the accident process. These studies were carried out by experts of the All-Union Scientific Research Institute for Nuclear Power Plant Operation, the Kiev Institute of Nuclear Research and the I.V. Kurchatov Institute of Atomic Energy [28] and the Scientific Research and Design Institute for Power Technology [17]. Both papers study the physical processes in the core during the movement of the RCPS rods into the core in response to the EPS-5 command, without reference to external thermal effects such as cavitation of MCPs, depressurization of the MFCC and so on.

As was stated earlier, both calculations are in a good agreement and indicate a shift in the power density field to the lower part of the core and a substantial increase in its overall non-uniformity. However, the conclusions of the calculations contradict each other in their explanation of the causes of the accident. The calculations in Ref. [28] attribute the cause of the accident to a local power excursion, whereas the calculations in Ref. [17], although they confirm these effects, conclude that the quantitative values of local power density surges were not enough to damage the fuel elements. This may be because the description of the core thermal-hydraulics

is inadequate. No information is given on the methods used by the Scientific Research and Design Institute for Power Technology in Refs [20] and [17] to calculate the thermal-hydraulic processes.

The Commission cannot accept the conclusion drawn in Ref. [17], because the authors are unable to guarantee a high degree of accuracy and the reliability of the calculation techniques used. Moreover, the fact that the results are extremely sensitive to the slightest variations in the initial data is pointed out in the studies carried out by the Scientific Research and Design Institute for Power Technology [20] and other organizations [32]. The study in Ref. [28] has found that a slight variation in the initial neutron distribution significantly alters the characteristics of the accident process. Thus, within the limits of a 20% variation in initial power density, the rate of thermal power increase may be 400 MW/s or 1000 MW/s by 6 s to 7 s after the start of the transient. Accordingly, after 6.5 s, the total reactor power may be 31 or 64 times higher than the initial level. The critical fuel enthalpy may be reached for either 5 or 40 fuel assemblies.

In the Commission's opinion, the substantial fuel damage, indicated in Ref. [28] as a possibility in the event of a minor error in determining the initial volumetric power density, did in fact happen. However, although the authors of Ref. [17] confirm the strong dependence of the results on slight variations in the initial data, they did not find initial values which would have given rise to the accident process. They conclude that there must have been some other factors, in addition to the unfavourable reactivity excursion caused by the RCPS rods, in order to account for the accident. The factors suggested include MCP cavitation, entry of non-equilibrium steam at the core inlet, switching off of the MCPs being run down prior to the EPS signal, coolant boiling at the reactor inlet, partial leaks in the lower water channels, and opening of the steam safety valves briefly.

It is possible that, in future, quantitative proof may be provided for these scenarios, which were put forward at the very beginning of the investigation into the causes of the accident. However, during four years of investigation no organization has yet published any such proof. Nevertheless, in order to explain and, more importantly, to correct the design characteristics of the reactor, the Commission thinks that it is sufficient to concentrate on the reactivity aspects of the accident caused by the design of the RCPS rods and the physical and thermal characteristics of the reactor, the most unfavourable aspects of which prompted the actions taken by the Chernobyl plant personnel. The Commission regards the list of organizational and engineering measures which were either immediately implemented or planned at Chernobyl type reactors [18, 25, 26, 39] as confirmation of the correctness of such an approach.

I-4.7. Actions of the Chernobyl plant personnel

The officially published documents on the causes of the Chernobyl accident set the blame mainly on the actions taken by the plant personnel. For this reason, the Commission feels obliged to present its own evaluation of the personnel's actions,

with two aspects in mind. Firstly, it is necessary to establish as full a list as possible of all the violations of the Operating Procedures [40] and other mandatory operating documentation. Secondly, an attempt must be made, using the available data, to assess retrospectively the effect of the violations on the causes and scale of the accident.

The Commission would like to stress that *its evaluations should in no way be regarded as condoning the violations of the regulatory documents committed by the personnel and designers.*

I-4.7.1.

During the decrease in power of Unit 4 on 25 April 1986 (at around 03:00) at a reactor power of about 2000 MW, the ORM fell to less than 26 manual control rods. The Operating Procedures for Units 3 and 4 (Section 9, Ref. [40]) allowed the units to be operated with an ORM of less than 26 manual control rods only after authorization by the Chief Engineer of the plant.

During the further power reduction (at about 07:00 on 25 April), when the reactor power was 1500 MW, the ORM fell to 15 manual control rods. In such cases, according to the requirements of Section 9 of the Operating Procedures, the reactor should be shut down. The personnel did not abide by this requirement. The Commission assumes that the personnel deliberately violated this requirement. The PRIZMA calculation code was found to be unreliable at this time, because it did not take into account the position of the rods of automatic regulators Nos 1, 2 and 3 (a total of 12 rods). There is a note to this effect in the senior reactor control engineer's operating log. The Operating Procedures and other operating documentation did not prescribe the actions to be taken by personnel under such circumstances (in the event of unreliable calculation) and similar circumstances (for example, in the event of complete failure of the PRIZMA code to determine the ORM). Nevertheless, in allowing the reactor to operate at 1500 MW with an ORM of less than 15 manual control rods, from about 07:00 to 13.30 on 25 April, the plant personnel, including senior staff, violated the requirements of Section 9 of the Operating Procedures, although this violation was not the cause of the accident and did not affect its consequences.

Notes:

Section 12 of the Operating Procedures, concerning the planned shutdown and cooling of the reactor, did not contain any requirements regarding the monitoring and maintenance of the ORM.

Section 12 states, in particular, that a power reduction must be carried out "using the set point adjusters of the automatic regulators to 160 MW(th) (5% of nominal power), and then the automatic power control system or the EPS-5 button."

In this connection it is important to make the following points.

Firstly, Section 8.9.1(a) of the Operating Procedures refers to reactivity as one of the important operating parameters which have to be controlled at all power levels. The ORM is not included in the list of important parameters.

Secondly, there was no provision in the design of the RBMK reactor for a device to measure the ORM in terms of effective manual control rods. The operator either had to determine the depth of insertion of rods in the intermediate position from the measuring instruments, correct for the non-linearity of the graduation scale and sum up the results, or instruct the plant computer to make the calculation and wait a few minutes for the result. In both cases, it seems unreasonable to expect the personnel to treat this parameter as a directly controllable one, particularly since the accuracy with which it can be determined depends on the power density field profile.

Thirdly, the Operating Procedures did not draw the attention of the personnel to the importance of the ORM as an essential parameter for ensuring the effectiveness of the emergency protection system.

In fact, post-accident calculation studies have shown that full withdrawal of the manual control rods from the core, which is not prohibited in other reactors, such as WWER reactors, was unacceptable for the RBMK reactor, owing to the design of the manual control rods. Withdrawal of more than a certain number of RCPS rods from the core resulted in the concentration of too much positive reactivity in the lower part of the core in terms of displaceable water columns.

I-4.7.2.

At 14:00 on 25 April the personnel, in accordance with Section 2.15 of the testing programme [41], closed the manual isolating slide valves of the ECCS, thereby disconnecting it from the MFCC, in order to avoid penetration of water into the MFCC in all three ECCS subsystems. Section 2.10.5 of the Operating Procedures states that during heating of the MFCC after a scheduled preventive maintenance outage, when the temperatures exceed 100°C, "the ECCS must be brought to a state of readiness." At the same time Section 2 of the Procedures for Reswitching Keys and Straps of the Engineered Protection and Blocking Systems [42] authorized the Chief Engineer of the plant to switch off automatic actuation of the ECCS. This is tantamount to switching off the fast acting part of the system and, therefore, the whole ECCS. The Commission notes, firstly, that taking the ECCS out of operation is a violation of Section 2.10.5 of the Operating Procedures and, secondly, that switching off the ECCS did not affect the initiation and development of the accident, since the chronology of basic events before and during the course of the accident showed that no signals for the automatic switching on of the ECCS were recorded. Under the specific conditions on 26 April 1986, it is therefore not true to state that

an opportunity of reducing the severity of the accident was lost [30] because the ECCS was switched off.

I-4.7.3.

At 00:28 on 26 April (according to the operating logs), the personnel failed to control the reactor and, as a result, there was an unforeseen reduction in the thermal power of the reactor to 30 MW. On the basis of the incomplete information available it is very difficult to make an unambiguous analysis of the reasons for the power reduction. The senior reactor control engineer noted in the operating log at 00:28: "The working range emergency power increase rate protection system is switched on. The automatic regulator set point has been reduced by the 'fast power reduction' button. Automatic regulator No. 1 is switched on. The unacceptable imbalance with respect to automatic regulator No. 2 has been eliminated. Automatic regulator No. 2 is on standby." Having analysed this note and the data recorded by the DREG program and the RCPS operating algorithm, the Commission presumes that the following events happened during that period:

— For some unknown reason (possibly owing to a perturbation associated with the MFCC: either a variation in the feedwater flow rate or steam pressure in the steam separator drums) the local automatic regulator was switched off and automatic regulator No. 1 came into an automatic regime and, in responding to the negative imbalance, moved to the upper limit stop switch.
— Automatic regulator No. 2, in response to the positioning of the automatic regulator No. 1 at the upper limit stop switch, did not come into automatic mode owing to an unacceptable imbalance in its measuring circuit.
— When all the regulators came out of automatic regime, the working range emergency power increase rate protection system was put on standby and an illuminated indicator 'working range emergency power increase rate protection system is on' appeared on the board of the senior reactor control engineer.
— As a result of the continued 'poisoning', the reactor power started to fall; the unacceptable imbalances in the measuring circuits of automatic regulators Nos 1 and 2 increased; 'failure in measuring circuit of automatic regulator No. 1' and 'failure in measuring circuit of automatic regulator No. 2' signals actuated and the corresponding illuminated indicators were displayed on the board of the senior reactor control engineer and were recorded by the DREG program; the senior reactor control engineer probably reduced the set points of the power regulators using the 'fast power reduction button' at a rate of 2% per second, managed to compensate for the imbalance in the measuring circuit of automatic regulator No. 1 and put it into automatic operating mode.
— Then, by manipulating the set points of automatic regulator No. 1, the senior reactor control engineer began to restore power to create the conditions for carrying out the tests.

Note:

Additional comments on the event that occurred at 00:28 are necessary.

The recording device of the physical power density distribution control system (PPDDCS) did not record the reduction in thermal power below 30 MW. During this time for about 5 min the neutron power recorder recorded zero power, after which the neutron power curve reached a level corresponding to 30–40 MW on the PPDDCS recorder. The low power level and correspondingly low accuracy with which it was determined by the built-in control devices mean that the reactor power actually fell to the minimum controllable level. According to Section 6.7 of the Operating Procedures, a power reduction to any level, but not below the minimum controllable power level, was regarded as a partial unit power reduction. The same section of the Operating Procedures authorized the power then to be restored to the rated value.

Here it is worth drawing attention to the contradictory nature of the instructions in the operating documentation. Section 6.1 of the Operating Procedures defined a short term shutdown as "a reactor power reduction to zero without cooling of the MFCC." However, it does not indicate what type of power is meant. If neutron power is meant, then the personnel violated the Operating Procedures; if thermal power is meant, there was no violation (according to the indications on the tapes of the recording instruments).

The Commission notes that the regulations and operating documentation in force at that time did not contain clear definitions of 'minimum controllable power level' and 'shutdown reactor' as applied to the power manoeuvre that took place.

The authors of this report believe that the drop in reactor power at 00:28 and subsequent power increase were largely to blame for the tragic consequences of the accident. The change in reactor operating conditions between 00:28 and 00:33 gave rise to a new xenon reshaping of the power density fields which the personnel were unable to control (see section 3.4 of this report [*Section I-3.4*]). No calculational studies have been made of the power density field dynamics from this time until the time of the accident.

It is impossible to draw a final conclusion on whether or not the personnel actions were correct under these specific circumstances because of the aforementioned contradictory nature of the requirements in the Operating Procedures, and the inadequacy and contradictory nature of the data recorded by the instruments. No calculational analysis of this situation has been made so far.

I-4.7.4.

The drop in reactor power was accompanied by a reduction in the water level and steam pressure in the steam separator drums. The water level in the steam sepa-

rators fell below the emergency set point of −600 mm without triggering the EPS-5 signal to actuate the RCPS. The Commission notes that during the reactor power reduction personnel did not switch from EPS-1 with a set point of −1100 mm to EPS-5 with a set point of −600 mm in response to the low water level in the steam separators. There are no notes on this in the operating logs. These personnel actions were in violation of Section 9 of the Procedures for Reswitching Keys and Straps of the Engineered Protection and Blocking Systems [42]. However, the Commission notes that another protection system against water level reduction in the steam separator drums below the −1100 mm level existed and was brought into operation. The set point of this protection system did not depend on the power. The statement made in Ref. [1] that "all the thermal parameter reactor protection systems were switched off" is therefore not true.

> **Note:** The reasons for transferring the functions of the EPS to the personnel owing to the lack of appropriate engineered safety features can be seen from the example of the reactor protection system against water level reduction in the steam separator drums. The designers made it clear in Ref. [43] that: "*Automatic reswitching* of the set points of EPS-1 and EPS-5 during emergency fluctuations of the water level in the steam separator drums is not permitted, since during operation of any of the emergency protection systems 1, 2 and 3 the water level falls to the −600 mm set point on the instrument which has a range of +400 to −1200 mm. This, in turn, will result in actuation of EPS-5 and complete shutdown of the reactor." They found an extremely simple way out of this: "*Instead of automatic reswitching* of the set points and automatic actuation (deactuation) of EPS-5 in response to a reduction in the feedwater flow rate *the operator should reswitch them manually using the general key* when alarm signals appear..."

It is not for us to demonstrate the feasibility of solving this problem using engineered safety features (this is feasible), but rather to demonstrate that in cases where there was a choice between complying with the safety requirements and shutting down the unit, or giving priority to economic considerations and keeping the unit in operation, the choice used to be made in favour of the second alternative, with the functions of the emergency protection system being transferred to the operator with a deep faith in the operator's complete reliability as a component of the safety system.

At 00:36:24 the personnel changed the set point for switching off the turbine of the protection system to guard against reduction in steam pressure in the steam separator drums from 55 kgf/cm^2 to 50 kgf/cm^2. These personnel actions were in accordance with requirements of the operating documentation since, according to Section 12 of the Procedures for Reswitching Keys and Straps of the Engineered Protection and Blocking Systems [42], personnel were entitled to select this set point. Contrary to what is stated in official documents, the Commission does not consider

that personnel should be held to blame for having blocked the steam pressure protection system of the steam separators.

> **Note:** It should be stressed that the protection system to guard against a reduction in steam pressure in the steam separators was designed to stop the turbine and was not "a thermal parameter reactor protection system", as described in Ref. [1]. In the interests of objectivity, the authors of Ref. [1] should have pointed out that the design was such that at a turbine power of less than 100 MW(e) the reactor was left without any protection system to guard against pressure reduction. At the actual α_φ it might have resulted in a reactor runaway even at the regulatory ORM (for example, in the event of the opening or non-closing of the main pressure relief valves, or the valves of the fast acting steam dump system, pipe rupture, etc.).

I-4.7.5.

At 00:41 (according to operating logs of the plant shift supervisor, the unit shift supervisor, the electrical workshop shift supervisor and the senior turbine control engineer) turbogenerator No. 8 was disconnected from the system to determine the turbine vibration characteristics during rundown. This procedure was not envisaged in the turbogenerator No. 8 rundown test programme. Measurements of the vibrations of turbogenerators Nos 7 and 8 at different loads were planned in a different programme, which had already been partially implemented by the personnel on 25 April during alternate redistribution of the turbine generator loads at a constant thermal reactor power of 1500–1600 MW. The disconnection of turbogenerator No. 8 from the system, together with the disconnection of the other turbogenerator (turbogenerator No. 7 was stopped at 13:05 on 25 April) without shutting down the reactor meant that the EPS-5 system to protect the reactor in the event of the shutdown of two turbogenerators had to be disabled. The personnel did this in accordance with Section 1 of the Procedures for Reswitching Keys and Straps of the Engineered Protection and Blocking Systems [42], which provided for the disabling of this protection system in the event of a turbogenerator load of less than 100 MW(e). The Commission believes that the personnel cannot be blamed for disabling the reactor protection system which shuts down the reactor in the event of the closure of the emergency stop valves of both turbines.

I-4.7.6.

By 01:00 on 26 April, the power increase had ceased and the reactor power was stabilized at about 200 MW(th). The decision to carry out the turbogenerator No. 8 rundown tests at a reactor power of about 200 MW was a departure from the testing programme. However, neither the design documentation, the regulatory documentation nor the operating documentation prohibited operation of the unit at

that power. Before the Chernobyl accident there were no safe operating limits in terms of minimum permissible thermal reactor power. In none of the documents studied by the Commission relating to the analysis of the operating conditions of the RBMK-1000 reactor do the reactor designers raise the question of the need to limit reactor operation at power levels below a certain level. Moreover, Section 11.4 of the Operating Procedures required personnel to reduce the reactor power to the level corresponding to the unit's internal consumption (200–300 MW(th)) following automatic power reduction in the EPS-3 design mode, or remotely in the event of abnormalities in the power supply system (frequency variations). There was no limitation on the period during which the reactor could operate at the minimum controllable power level.

> **Note:** The Operating Procedures permitted operating conditions similar to those prevailing at Chernobyl Unit 4 on 26 April 1986 and they might have occurred without any intervention on the part of the personnel. We only need to assume a perfectly possible situation in which triggering of EPS-3 occurs when the reactor is operating initially at rated power with an ORM of 26 manual control rods. Under these conditions, approximately one hour after triggering of EPS-3 the ORM could have fallen to less than 15 manual control rods at a reactor power of 200–300 MW(th), and any further action, whether automatic or remote, to shut down the reactor could have led to a similar repetition of the events of 26 April 1986.

The Commission considers that the personnel cannot be held to blame for operating the unit at a power of less than 700 MW.

I-4.7.7.

At 01:03 and 01:07, in accordance with Section 2.12 of the testing programme [41], one MCP from each side (MCP Nos 12 and 22) was also switched on "to cool down the reactor during the test." Before 26 April 1986 no document, including the Operating Procedures, prohibited connection of all eight MCPs to the reactor at any power level. In the Commission's view, the personnel committed no violations by these actions. At the same time, at low power levels when the feedwater flow rate is less than 500 t/h, the Operating Procedures limited the capacity of each MCP to 6500–7000 m^3/h in order to prevent cavitation. On 26 April 1986 the flow rates of certain MCPs actually exceeded the limit (violation of Section 5.8 of the Operating Procedures), but did not cause cavitation of the pumps, as is evident from the DREG program printout and is confirmed by the studies carried out by the Mechanical Engineering Experimental Design Office and other organizations. Reference [31] points out that "both the pumps being run down and those not being run down maintained a steady water supply, even during the runaway and destruction of the reactor."

I-4.7.8.

The Commission's analysis of the actions of personnel during the preparations for and implementation of the tests shows that the personnel committed the following violations of the requirements of the operating and regulatory documentation:

- Reactor operation with an ORM of 15 manual control rods or less from 07:00 until 13:30 on 25 April and from approximately 13:00 on 26 April until the time of the accident (violation of Section 9 of the Operating Procedures);
- Complete disconnection of the ECCS (violation of Section 2.10.5 of the Operating Procedures);
- Change in the set point of the reactor protection system to guard against reduction in water level in the steam separator drums from −600 mm to −1100 mm (violation of Section 9 of the Procedures for Reswitching Keys and Straps of the Engineered Protection and Blocking Systems [42]);
- Increase in flow rates of certain MCPs to 7500 m^3/h (violation of Section 5.8 of the Operating Procedures).

In addition, the personnel made certain deviations from the testing programme (see sections 4.7.5 and 4.7.6 of this report [*Sections I-4.7.5 and I-4.7.6*]). Conclusions about the personnel actions after the drop in power (Section 4.7.3 of this report [*Section I-4.7.3*]) can only be drawn after further studies have been carried out.

I-4.7.9.

To conclude this section, the Commission thinks it necessary to summarize "the most serious violations of the operating documentation committed by the personnel at Unit 4 of the Chernobyl nuclear power plant" [30] in terms of their impact on the causes and the consequences of the accident.

In the Commission's view, the switching off of the ECCS did not affect the initiation and scale of the accident.

It would appear that the connection of eight instead of the usual six MCPs to the reactors if anything hindered the reactor runaway, which was initiated and developed independently of the operating conditions of the pumping group and the temporarily increased coolant flow rates through certain MCPs. Additional theoretical analysis in this area is required.

The changes made to the set points and deactivation of the engineered protection and blocking systems were not the causes of the accident and did not affect its scale. These actions were not in any way related to the emergency protection systems of the reactor itself (relating to power level, power increase rate), which the personnel did not deactivate.

The change in the initial reactor power before the tests and subsequent continued power reduction made it necessary for actions to be taken to control the unit which were not foreseen in the test programme. This increased the risk of incorrect

actions, as demonstrated by the unauthorized reduction of reactor power to the minimum controllable level followed by its increase, which had an extremely negative effect on the subsequent behaviour of the reactor.

The low reactor power level increased the likelihood of the positive reactivity effect which manifested itself at a maximum not only as a result of local power density increases, but also for some other reasons (for example, coolant leakage). The choice of power level therefore affected the scale of the accident. No matter how paradoxical it may seem, low power levels proved to be the most dangerous ones and the safety of the reactor at these levels had not been studied or analysed in the design documentation.

Had the tests been carried out at a power level of 700 MW(th), as was initially planned, the accident might not have happened. However, this theory needs to be tested by studies which have not yet been carried out.

I-4.8. Operating reactivity margin

The ORM is one of the most important aspects of the Chernobyl accident.

In addition to the information presented in sections 4.7.1 and 4.7.3 [*Sections I-4.7.1 and I-4.7.3*] of this report, in which the Commission analyses the actions of the personnel in terms of their compliance with the Operating Procedures, it should be noted that post-accident studies have shown that the way in which the real role of the ORM is reflected in the Operating Procedures and design documentation for the RBMK-1000 is extremely contradictory.

Section 9 of the Operating Procedures, *Normal Operating* Parameters of the Unit and Permissible Deviations, points out that:

> "At nominal power during steady state operation, the ORM should be not less than 26–30 rods.
>
> "Reactor operation at an ORM of less than 26 rods must be authorized by the Chief Plant Engineer.
>
> "When the ORM falls to 15 rods, the reactor should be shut down immediately.
>
> "The scientific managers of the plant must periodically (once a year) examine the specific regulations designed to maintain steady power density fields at the particular unit and, if necessary, change them to make them stricter, in agreement with the Scientific Manager and the Chief Design Engineer."

Note: In the Commission's view, the concepts of "scientific managers of the plant" and "making the specific regulations designed to maintain steady power density fields stricter", which are used in the Operating Procedures but not defined in those Procedures or in the regulatory documents in force at that time, are rather loose concepts.

The contradictory nature of the directives concerning the ORM is also illustrated by the following extracts from the Operating Procedures which are associated with the situation at 00:28 (drop in reactor power):

"6.2. Reactor power rise after a short term shutdown when there has been no iodine poisoning of the reactor is permitted only if there is the required operational reactivity margin, which is determined from the margin prior to the shutdown of the reactor. The required operational reactivity margin as a function of the power level at which the reactor was operating prior to shutdown is given in the Table" [*see Table I-II*].

TABLE I-II. REQUIRED OPERATING REACTIVITY MARGIN AS A FUNCTION OF REACTOR POWER LEVEL[a]

Reactor power level (Percentage of nominal)	Required ORM (Equivalent number of manual control rods)
80–100%	50
50–80%	45
<50%	30

[a] From the Operating Procedures.

"6.6.4. The minimum operational reactivity margin during a power rise following a short term shutdown must not be less than 15 rods.
"If in withdrawing manual control rods while the reactor is being brought to criticality the operational reactivity margin falls to 15 rods and *continues to fall*, all rods must be dropped to the lower limit stop switches..."

From these extracts from the Operating Procedures, the following conclusions can be drawn:

— Firstly, the Operating Procedures unambiguously treat the ORM as a way of controlling the power density field;
— Secondly, the incorrect indication about the possibility of lowering the ORM to less than 15 manual control rods shows that the ORM was not treated as an operational safety limit, violation of which could lead to an accident.

Note: The design documentation also contains equally contradictory directives concerning the ORM. For example, in Ref. [44] it is stated that: "At rated power during steady state operation, the ORM should not be less than 26, and not more than 35, manual control rods. On the authorization of the Chief Plant Engineer, *a reactor may be allowed to operate with an ORM below the mini-*

mum value, but for no longer than three days. Operation of the unit is not permitted at an ORM of less than 10 rods.''

Thus the ORM is not regarded in the Operating Procedures as an indicator of the ability of the EPS to perform its functions. This is natural, since such an understanding would be regarded as an *unlawful transfer of reactor protection functions by designers from engineered safety features to the personnel,* and to its ability to work "in the plant computer operating mode" (see note to section 4.7.1 of this report [*Section I-4.7.1*]). The design documentation also did not treat the ORM as a critical parameter requiring a protection system (see sections 3.3 and 3.7 of this report [*Sections I-3.3* and *I-3.7*]).

However, in the Commission's opinion, the main point is that *having realized the full extent of the danger of reducing the ORM in terms of the ability of the EPS to perform its functions, the designers did not inform the operating personnel accordingly of this fact, who if they had realized the danger might not have taken upon themselves the function given to them by the designers to protect the reactor from a runaway.*

In fact, in 1984, when the positive reactivity insertion effect, which was caused by the design of the RCPS rods and which had not been foreseen at the design stage, manifested itself during experiments, the Chief Design Engineering Organization informed other organizations and all plants with RBMK reactors that it intended to restrict the number of RCPS rods which could be completely withdrawn from the core to not more than 150 rods, while the remaining rods would have to be submerged in the core by at least 0.5 m [22].

In the light of our present knowledge acquired from the post-accident studies, we can interpret the proposed limitation in the following way.

Since the axial power density field distribution in the RBMK reactor can have a specific instability determined by the saddle point at the mid-point of the core (a double peaked field) at which insertion of a manual control rod introduces positive reactivity in the lower part of the reactor and negative reactivity in the upper part (the 'rocker' effect), the total positive reactivity introduced can be reduced by preventing the formation of water columns above a certain permissible value. This can be achieved if the complete withdrawal of the relevant number of rods is prohibited. This decreases the 'explosive' reactivity effect of the manual control rod displacer which replaces the water column in the lower part of the core, while the absorbing part of the rod is located in the neutron flux. The majority of RCPS rods will have the same effect on the reactor reactivity, only this will occur more than one second after the EPS-5 command.

Note: In view of the strong dependence of the reactor's runaway capacity on the number of manual control rod absorbers and water columns under their displacers located in the core, it is rather difficult to add up the lengths of the partially submerged manual control rods in order to calculate the effective ORM (at least for the design of the RCPS rods at the time of the accident).

However, despite the obvious importance of the ORM parameter in terms of the effectiveness of the EPS, the appropriate changes were not made to the Operating Procedures before 1986 and no explanations were given to the personnel of plants with RBMK reactors. In any situation "the personnel had the right to expect, under any operating conditions, that the EPS would function and effectively terminate the chain reaction and prevent runaway of the reactor" [45]. However, that was not the case and *right up to the time of the accident, the personnel at units with RBMK reactors had no idea that the ORM value (for the design of the RCPS rods which existed at the time of the accident) not only determined the ability to control the reactor power density field but, primarily, determined the ability of the reactor's EPS to perform its functions.*

After the RCPS rods had been redesigned (the water columns beneath the displacers removed), the Chief Design Engineer could legitimately state, four years after the accident, that: "with respect to the RBMK reactor this matter (concerning the ORM) has been thoroughly studied and it has been determined that for optimum power density control, an ORM of 26–30 manual control rods is necessary" [36]. Now, this is the case. However, the Commission has to stress that the ORM values now in force (43–48 manual control rods for steady state operation and 30 manual control rods as the limit below which a reactor is to be shut down) differ considerably from those established before the accident.

> **Note:** It is evident that at plants with RBMK reactors many functions of the EPS (including its function when the ORM reaches a critical value) were transferred to the personnel with a deep faith in the personnel's absolute reliability as a component in the complex and ramified reactor safety system. Four-and-a-half years after the accident, representatives of the Scientific Manager acknowledged the mistakenness of this concept: "Many years of accident free operation of military reactors in the USSR led to the deep rooted philosophy that one only has to write correct reactor operating instructions to guarantee safety. It goes without saying that these instructions are mandatory for the personnel. It turned out that the real situation was very different. The first most important lesson to be learnt from Chernobyl is that the safety of nuclear power plants cannot be based on instructions. If a particular parameter exceeds certain limits, the reactor must be shut down automatically without the intervention of the operator. Furthermore, measures must be taken to exclude the possibility of an automatic protection system being deliberately switched off" [14].

To that correct, but late, statement must be added the comment that it is difficult to claim that the RBMK operating instructions in force prior to 1986 were correct.

I-4.9. Causes of the accident

The event which initiated the accident was the pressing by the senior reactor control engineer of the EPS rod drop button (EPS-5) to shut down the reactor for some reason which has not yet been established for certain.

The cause of the accident was an uncontrolled increase in reactor power which initially arose because of the increase in reactivity caused by the displacers of the RCPS rods [17, 28, 35].

The increase in reactivity was not suppressed by the absorbers of RCPS rods, not only because of their slow speed, but also because the operating personnel had withdrawn more than the permitted number of manual control absorbing rods from the core before the tests, thereby creating the conditions for a multiple increase in intensity of the initial reactor runaway, which was predetermined by the design of the RCPS rods.

The initial reactivity increase resulted in a substantial growth in power since there was strong positive feedback between reactor reactivity and steam generation in the core. This process was considerably enhanced by the low initial reactor power, by the thermal-hydraulic characteristics that promoted maximum realization of the steam reactivity effect, and by significant power density irregularities throughout the core.

Note:

The causes of the accident have been analysed in many documents and their complexity has been noted. In particular, Ref. [46] presents a concise view of the causes of the accident:

"An analysis of the Chernobyl accident has identified: the major role played by the displacers; the large void reactivity effect; and the formation of extremely large volumetric power density irregularities in the core during the accident. This last fact is one of the most important ones and is caused by the large dimensions of the core (7 m × 12 m), the slow speed of movement of the non-uniform rods (having absorbers, displacers and water columns) (0.4 m/s), and the large void reactivity effect ($5\beta_{eff}$). All these factors predetermined the scale of the Chernobyl accident.

"The scale of the Chernobyl accident was therefore not determined by personnel actions, but by a lack of understanding, primarily on the part of the scientific managers, of the effect of steam quality on the reactivity of the RBMK core. This led to an incorrect analysis of the operational safety; to a disregard of repeated manifestations of the large void reactivity effect during operation; to a false confidence in the effectiveness of the RCPS which, in fact, failed to cope with both the Chernobyl accident and many others, in particular with DBAs; and, naturally, to the formulation of incorrect operating procedures.

"This inadequate level of scientific management is explained, above all, by the following:

— the extremely low number of studies of the neutron physics processes taking place in the RBMK core;
— a disregard for discrepancies in the results obtained using different methods;
— the lack of experimental studies under conditions close to the natural ones;
— the lack of analysis of specialized publications; and
— in the long run, the transmission to the Chief Design Engineer of incorrect techniques for calculating the neutron physics processes used to analyse the processes taking place in the core and the safety of plants with RBMK reactors.

"It is also an important point that for a long time the USSR Ministry of Power had been operating plants with RBMK reactors with neutron physics instabilities in the core, but did not take sufficient notice of the repeated unusual signals of the emergency power protection system and of the emergency power increase rate protection system following triggering of the EPS, and did not demand thorough investigations of emergency situations.

"...We are bound to conclude that an accident such as that at Chernobyl was inevitable."

I-5. CONCLUSIONS

The Chernobyl accident was examined and analysed by the International Atomic Energy Agency's International Nuclear Safety Advisory Group (INSAG), a body which provides advice to the IAEA Director General [47]. Without going into the details of that report, the Commission notes that in analysing the root causes of the Chernobyl accident, INSAG concluded that the need to create and maintain a 'safety culture' is a precondition for ensuring nuclear power plant safety.

The concept of 'safety culture' relates to a very general concept of dedication and personal responsibility of all those involved in any safety related activity at a nuclear power plant. Inculcation of a safety culture requires that, in training personnel for nuclear plants, particular emphasis be placed on the reasons for the establishment of safety practices and on the consequences in terms of safety of failures on the part of personnel to perform their duties properly. Special emphasis must be placed on the reasons for the establishment of safety limits and the consequences in terms of safety of violating them. Safety culture presupposes total psychological dedication to safety, which is primarily created by the attitude of the administrative staff of the organizations engaged in the development and operation of nuclear power plants [48].

In INSAG publications, the concept of safety culture has been extended beyond the purely operational aspects to cover all types of activities at all stages in the lifetime of a nuclear power plant which may affect its safe operation. It even covers the highest spheres of administration, including the legal and governmental ones which, according to the concept, must create a national climate in which attention is paid to nuclear safety on a daily basis. If the Chernobyl accident is assessed in terms of this safety culture concept, it can be seen that not only those involved in the opera-

tional stage lacked an adequate safety culture, but also those involved in other stages of the lifetime of a nuclear power plant (designers, engineers, constructors, equipment manufacturers, ministerial and regulatory bodies, etc.).

Taking into account the facts presented in this report and the preamble to this section, the Commission arrived at the following conclusions.

I-5.1. Design deficiencies of the RBMK-1000 reactor at Chernobyl Unit 4 predetermined the severe consequences of the accident

The Chernobyl disaster was caused by the choice made by the RBMK-1000 reactor designers of a design which did not take adequate account of the safety issues involved. As a result of that choice, the physical and thermal-hydraulic characteristics of the reactor core contradicted the principles of dynamically stable safe systems. In accordance with this design concept, a reactor control and protection system was designed which did not meet the safety requirements. The unsatisfactory physical and thermal-hydraulic characteristics of the reactor core in terms of safety were aggravated by errors made in the design of the RCPS.

The possible consequences of operating a reactor with such dangerous characteristics were not indicated in the design, the engineering or, consequently, in the operating documentation. Designers at the highest level kept asserting that the RBMK reactor was extremely safe. This resulted in complacency on the part of the personnel with regard to the reactor facility that contradicted the awareness of 'danger' inherent in the concept of safety culture.

The reactor designers were aware that the dangerous property of the reactor they had developed could be a cause of nuclear instability, but failed to estimate quantitatively its possible consequences and attempted to protect themselves by imposing operating limitations which, as it turned out, provided extremely poor protection. Such an approach has absolutely nothing to do with safety culture.

Another point must be made. The aforementioned poor protection system against the very dangerous consequences of an unstable reactor does not fit in with the defence in depth principle on which the development of nuclear power in the rest of the world has been based.

The design parameters and characteristics of the RBMK-1000 reactor on 26 April 1986 violated the safety standards and regulations so seriously that it could only be operated in a country where there was an inadequate safety culture.

I-5.2. The misguidedness of the practice of transferring emergency protection functions to the human operator owing to the lack of appropriate engineered safety features was highlighted by the accident itself: the combination of design deficiencies and the non-total reliability of human operators brought about the disaster

The personnel violated the Operating Procedures and the Commission notes these violations in this report. Some of these violations did not affect the initiation

and development of the accident, others created favourable conditions for the manifestation of the negative design characteristics of the RBMK-1000 reactor. The violations were largely the result of the poor quality of the operating documentation and its contradictory nature caused by the poor quality of the RBMK-1000 reactor design.

The personnel were unaware of some of the dangerous features of the reactor and, therefore, did not realize the consequences of the violations. This fact in itself demonstrates the lack of safety culture, not so much on the part of the personnel, but rather on the part of the reactor designers and the operating organization. It is worth looking at another approach to the analysis of the causes of an accident and the role of personnel actions in its initiation and development. After the serious accident at the Three Mile Island plant in the USA in 1979, the designers did not seek to blame the personnel, since "they [the engineers] may analyse the first minute of an accident for hours or even weeks, seeking to understand what happened or trying to project what will happen next if parameters are manipulated", whereas an operator has to deal with "hundreds of thoughts, decisions and actions he takes during a transient" (see Ref. [49], pp. 644–645). Experts in the USA understood that *"some transients can be avoided completely through good design. If a transient can be imagined, a contingency can be designed to cope with it"* ([49], p. 644). E.R. Frederick, the American operator who made erroneous decisions on the night of 28 April 1979, but was not prosecuted for them, writes: "How I have wished to go back and change those two decisions. But the event cannot be undone — and it must not happen again. *An operator must never be placed in a situation which an engineer has not previously analysed. An engineer must never analyse a situation without observing an operator's reaction to it"* ([49], p. 647).

The ambiguities associated with the problem of human operators and the reasons for operator errors are beginning to be understood by Soviet experts: "One has to admit that, unfortunately, the study of human nature is still given low priority by developers of new technologies and, probably, by engineering circles in general. It is very difficult for a technocratic mind to understand the fact that the psychological motivation for an operator's actions is different from that motivating the actions of someone carrying out research, manufacturing equipment, or making adjustments or carrying out repairs. This failure to understand an operator's errors of course applies not only in the field of nuclear power" [50].

The priority given to economic factors and electricity production has always been, and still is, a dominating principle of nuclear power management. The system of incentives and penalties for operating personnel, still in existence at the majority of nuclear power plants, is based on this principle. In the event of a conflict between economics (planned power production) and safety, this system encourages operating personnel to decide in favour of the former. This also played a role on 26 April 1986 at the Chernobyl plant when the personnel encountered difficulties with the test programme and certain violations of the Operating Procedures were committed as

a result of a long standing habit of ensuring that the goal that was set was achieved at all costs.

I-5.3. The system of legal, economic and sociopolitical correlations that existed prior to the accident and still exists in the field of nuclear power has no legal basis, and did not and does not meet the requirements of ensuring the safe utilization of nuclear power in the USSR

This conclusion is based on the fact that when there is no law governing the utilization of nuclear power, no one bears the full responsibility for the safety of operating nuclear power plants. All those involved in the development and operation of nuclear power plants are responsible only for those parts of the job which they perform themselves. According to international standards and practices this overall responsibility should be borne by the operating organizations. So far, the USSR does not have any such organizations. Their functions of making the most important general decisions concerning a plant as a whole were and are usually performed by the corresponding ministries, which are government authorities. As a result, the decision making is separated from the responsibility for the decisions. Moreover, following the repeated reorganization of government authorities, those bodies which made crucial decisions earlier no longer even exist. As a result, there are dangerous facilities for which no one is responsible.

According to general international practice formulated in the IAEA recommendations [51] and officially recognized by the USSR [52], the final responsibility towards the population and the country as a whole for the safe operation of a plant always rests with the operating organization. However, responsibility cannot be borne without the corresponding rights. The system which existed, and still exists, does not give any rights either to the plants themselves, or to the higher authority, which together perform the duties of an operating organization.

According to the current standards and regulations, these organizations are not empowered to make any crucial decisions (and after the Chernobyl accident also not any less important ones; in fact, to all intents and purposes, no decisions) without clearance by the Chief Design Engineer, the Scientific Administrator, the General Designer and the regulatory body. At the same time, all these organizations which force owners to make decisions and do not allow them any choice apart from termination of the operation of the plant in the case of a disagreement, carry no responsibility themselves for the decisions made (except for the regulatory body, which is also wrong).

This report identifies many violations in the design of Chernobyl Unit 4 of the safety standards and regulations in force at the time of the design, construction and operation of the plant. Nevertheless, the design was approved and authorization given for construction by all the relevant authorities and regulatory bodies. This demonstrates the lack in the USSR of a well organized group of experts endowed with its own resources, rights and responsibilities for its decisions.

The USSR State Committee for the Supervision of Nuclear Power Safety was established only three years before the Chernobyl accident and, notwithstanding the safety culture concept, it could not be regarded as an independent body, since it was part of the same state authorities responsible for the construction of nuclear power plants and electricity generation. Since the accident, a number of major changes have been made to the system for the supervision of safety in nuclear power. However, since the regulatory bodies have no legal basis, no economic methods of control, and no human and financial resources, and since it is very difficult to set up an institute of independent experts in this country, the system that existed and still exists is one consisting of many links providing step by step control and finicky supervision of nuclear power plants, rather than a full blooded regulatory system for the safe use of nuclear energy in the interests of the whole population.

One of the most important lessons learned from the Chernobyl accident is not only the need to improve the specific parameters and operating conditions of the RBMK reactors, no matter how important this is in itself, but also the need to incorporate the requirements of the 'safety culture' concept into all aspects of nuclear energy applications in the USSR.

I-5.4. The studies of the causes and circumstances of the Chernobyl accident must not be regarded as complete and they must be continued to establish the truth and to learn important lessons for the future

Since 26 April 1986, considerable efforts have been made to analyse the causes and circumstances of the accident. However, this work must not be regarded as complete. A vast number of calculations and possibly experimental studies will have to be carried out "in order to ensure that no safety significant event goes undetected and to introduce the necessary design improvements to prevent a recurrence of safety significant anomalous events anywhere, regardless of where they originate" [48].

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ЯДРИХИНСКИЙ А.А., Ядерная авария на 4-ом блоке Чернобыльской АЭС и ядерная безопасность реакторов РБМК (1989).

Annex II

REPORT BY A WORKING GROUP OF USSR EXPERTS[1]

Causes and Circumstances of the Accident at Unit 4 of the Chernobyl Nuclear Power Plant and Measures to Improve the Safety of Plants with RBMK Reactors (Moscow, 1991)

This report was prepared by a Working Group comprising the following members:

Abagyan, A.A.	Director General, Scientific Production Group Ehnergiya; All-Union Scientific Research Institute for Nuclear Power Plant Operation (VNIIAES)
Adamov, E.O.	Director, Scientific Research and Design Institute for Power Technology (NIKIET)
Bol'shov, L.A.	Director, Nuclear Safety Institute (IBRAE)
Chukardin, Eh.I.	Chief Expert, State Committee for Science and Technology of the USSR Council of Ministers (GKNT)
Petrov, V.A.	Director, Scientific and Technical Centre of the USSR State Committee for the Supervision of Safety in Industry and Nuclear Power (GPAN)
Velikhov, E.P.	Director, I.V. Kurchatov Institute of Atomic Energy (IAE)

[1] This report was translated into English by the IAEA.

The report was approved by a committee comprising the following members:

Abagyan, A.A.	VNIIAES
Afanas'ev, A.M.	IBRAE
Asmolov, V.G.	IAE
Bol'shov, L.A.	IBRAE
Borshchev, V.P.	NIKIET
Burlakov, E.V.	IAE
Cherkashov, Yu.M.	NIKIET
Chukardin, Eh.I.	GKNT
Filimontsev, Yu.N.	VNIIAES
Kalugin, A.K.	IAE
Kisil', I.M.	VNIIAES
Krayushkin, A.V.	IAE
Kroshilin, A.E.	VNIIAES
Malkin, S.D.	IAE
Nikitin, Yu.M.	NIKIET
Novosel'skij, O.M.	NIKIET
Petrov, A.A.	NIKIET
Petrov, V.A.	GPAN
Petrunin, D.M.	GPAN
Podlazov, L.N.	NIKIET
Ponamarev-Stepnoj, N.N.	IAE
Vasilevskij, V.P.	NIKIET
Voronin, L.M.	VNIIAES

II-1. BRIEF DESCRIPTION AND FEATURES OF THE RBMK-1000 REACTOR OF CHERNOBYL UNIT 4

Operating experience with pressure tube type reactors in the USSR amounts to over 580 reactor-years. This type of reactor was used in the world's first nuclear power plant in Obninsk, in Units 1 and 2 of the Beloyarsky nuclear power plant, in the Bilibino and Siberian nuclear power plants, and finally in a large group of plants with RBMK-1000 and RBMK-1500 reactors. Such is the history of these pressure tube type uranium–graphite reactors.

Chernobyl Unit 4 was a heterogeneous pressure tube type thermal neutron reactor with a graphite moderator and a boiling light water coolant. There were 1660 fuel channels in the vertical ducts of the graphite columns; these channels were tubes 80 mm in diameter made of zirconium alloy. Each fuel channel contained a fuel assembly incorporating 18 rod type fuel elements which were 13.6 mm in diameter and were enclosed in zirconium alloy cladding.

The heat flow diagram is typical for single circuit boiling water reactors (Fig. II-1). The multipass forced circulation loop consists of two parallel loops each of which cools half of the fuel channels in the reactor. The coolant is circulated by the main circulating pumps (MCPs). Individual pipes supply subcooled water to each channel and remove the steam–water mixture. The steam and water are separated in the horizontal steam separator drums at a pressure of about 7 MPa. The saturated steam is sent to two turbines and the steam condensate is returned to the separators, after preheating and deaeration; it is then mixed with the separated saturated water and sent to the reactor inlet by the MCPs.

The reactor control and protection system (RCPS) consists of 211 movable solid absorber rods in special channels cooled by an independent water circuit. In normal operating modes, and in design basis accidents (DBAs), the RCPS automatically maintains the preset power level; initiates a swift power reduction using the automatic and manual control rods if there are signals indicating main equipment failures; terminates the chain reaction using the rods of the emergency protection system (EPS) if there are signals indicating dangerous divergences in unit parameters or equipment failures; compensates for reactivity changes during heatup and when the reactor is being brought up to power; and controls the core power density.

The independent control rods are inserted into the core at a speed of 0.4 m/s when the EPS is triggered.

The RCPS has a local automatic control (LAC) subsystem and a local emergency protection (LEP) subsystem. Both these systems respond to signals from in-core ionization chambers. The LAC system automatically stabilizes the main harmonics of the radial–azimuthal power density distribution, while the LEP system protects the reactor by ensuring that the preset fuel assembly power level at specific core locations is not exceeded. The axial fields are controlled by shortened absorber rods which are inserted into the core from below (24 rods).

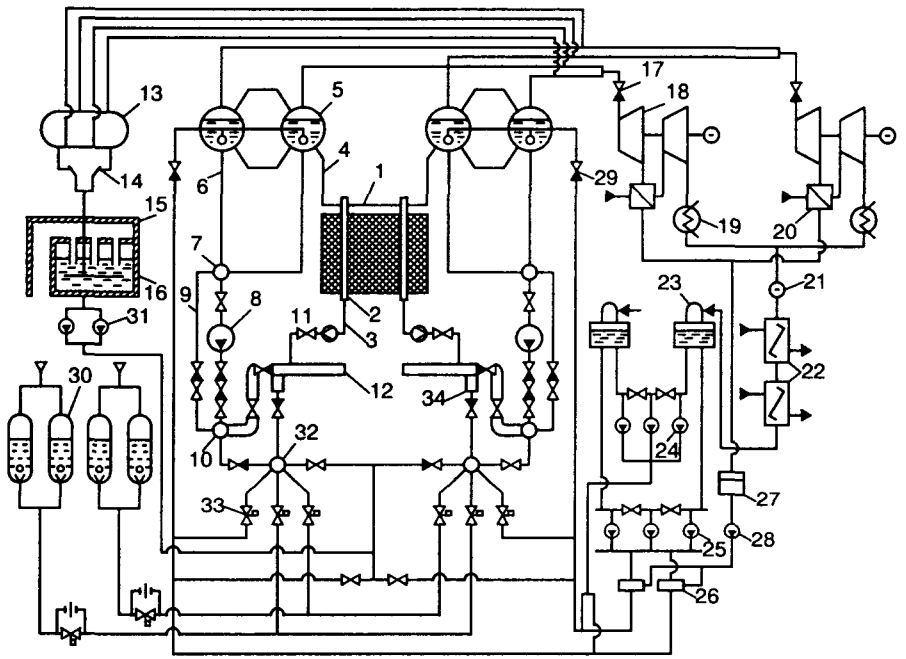


FIG. II-1. Schematic diagram of an RBMK plant. Key: 1: graphite reactor core; 2: fuel channel; 3: in-core instrumentation tube; 4: feedwater pipe; 5: steam separator drum; 6: downcomer; 7: intake header; 8: main circulating pump; 9: bypass; 10: high pressure header; 11: stop valve; 12: distribution group header; 13: steam header; 14: steam dump valve; 15: accident localization system; 6: ECCS water reserve tank; 17: pressure controller; 18: turbogenerator; 19: condenser; 20: moisture separator/reheater; 21: condensate pump; 22: preheater; 23: deaerator; 24: emergency electric feedwater pump; 25: electric feedwater pump; 26: mixing preheater; 27: condensate collecting tank; 28: moisture separator/reheater condensate pump; 29: level controller; 30: ECCS hydraulic accumulator; 31: ECCS pump; 32: ECCS header; 33: ECCS fast acting valve; 34: leak limiter.

The RBMK 1000 reactor also incorporates the following monitoring and control systems:

- radial (over 100 channels) and axial (12 channels) power distribution monitoring systems employing self-powered detectors (SPDs);
- startup monitoring systems (reactivity meters, removable startup chambers);
- a coolant flow rate monitoring system for each channel employing ball type flow meters;
- a fuel rod leak monitoring system, which detects the short lived activity of volatile fission products in the steam-water lines sequentially at the outlet of each channel in the optimum energy ranges ('windows'), using a scintillation detector and a photomultiplier moved by a special dolly from one line to the next;

— a pressure tube integrity monitoring system, which monitors the humidity and temperature of the gas circulating through the channels.

All the data are sent to the computer. The information is presented to the operators in the form of signals indicating divergences, readouts (which may be called up as required) and data from the recording equipment.

The RBMK 1000 units are intended for base load operation (at constant power).

The main design characteristics of the RBMK reactor in Chernobyl Unit 4 are as shown in Table II-I.

TABLE II-I. MAIN DESIGN CHARACTERISTICS OF THE RBMK REACTOR IN CHERNOBYL UNIT 4

Thermal power (MW)	3200
Fuel enrichment (%)	2.0
Mass of uranium in fuel assembly (kg)	114.7
Number/diameter of fuel rods in fuel assembly (mm)	18/13.6
Fuel burnup (MW·d/kg)	20
Axial power peaking factor	1.40
Radial power peaking factor	1.48
Maximum design power of fuel channel (kW)	3000
Void coefficient of reactivity α_φ at the working point ($\%^{-1}$ (δk/k) void)	$+2.0 \times 10^{-4}$
Fast power coefficient of reactivity α_w at the working point (MW^{-1})	-0.5×10^{-6}
Fuel temperature coefficient of reactivity α_T ($°C^{-1}$)	-1.2×10^{-5}
Graphite temperature coefficient of reactivity α_C ($°C^{-1}$)	6×10^{-5}
Minimum efficiency of RCPS rods (%)	10.5
Efficiency of manual control rods (%)	7.5
Effect (on average) of replacing a spent fuel assembly by a new one (%)	0.02

One important physical characteristic from the point of view of reactor control and safety is the operational reactivity margin (ORM), which is expressed in terms of a specific number of equivalent fully inserted RCPS rods, depending on the axial neutron field.

According to the Operating Procedures for Units 3 and 4 of the Chernobyl Nuclear Power Plant [*see Annex I, Ref. [40]*] (1Eh-S-11, pp. 34, 46), the ORM at rated power and in steady state mode should be 26–30 rods.

Reactor operation with an ORM of less than 26 rods is permitted if authorized by the Chief Engineer at the plant.

If the ORM decreases to 15 rods, the reactor should be shut down immediately.

Raising the power after a short term shutdown (when there have been no iodine poisoning effects and the pre-shutdown power level was less than 50% of rated power) is allowed if the ORM prior to shutdown was at least 30 rods.

The reactivity margin during a power raise following a short term shutdown must be at least 15 rods.

If the reactivity margin drops to 15 rods and it continues to decrease when the RCPS rods are withdrawn while the reactor is reaching criticality, then all the automatic control, manual control and emergency protection overcompensation system rods must be inserted up to the bottom limit switches and the shortened absorber rods must be inserted into the zone of their greatest efficacy. The duration of the outage is determined from the depoisoning curves.

The dependence of the effective multiplication factor on the coolant density in an RBMK is largely determined by the various kinds of absorbers which have been placed in the core. During initial loading of the core (which includes approximately 240 additional absorbers containing boron), coolant vaporization in a fuel channel produces a negative reactivity effect. The increase in steam quality which occurs during normal on-line refuelling at rated power with a 30 rod reactivity margin causes a positive reactivity rise.

The Unit 4 RBMK reactor was equipped with safety systems. These included the following:

- reactor control and protection system (RCPS);
- emergency core cooling system (ECCS);
- main coolant circuit overpressure protection system;
- reactor space overpressure protection system;
- the accident localization systems, which include:
  - system of leaktight compartments
  - isolation valve system
  - pool type pressure suppression system;
- executive safety systems;
- control safety systems;
- radiation monitoring system.

Thanks to the aforementioned reactor characteristics, and the safety systems (including the protection, localization and executive safety systems) (Fig. II-2 and Table II-II), reliable and effective operation of RBMKs in all authorized regimes and safety for the whole range of DBAs have been maintained, in accordance with the approved design documentation.

RBMK reactors have the following drawbacks:

— insufficient automatic engineered protection of the reactor to prevent its being put into an unauthorized regime;
— the void coefficient of reactivity α_φ varies, and there is a coolant vaporization effect owing to decreased coolant density in the core;
— the emergency protection system response is too slow, and a positive reactivity insertion can occur when there is an unacceptable decrease in the reactivity margin.

Chernobyl Unit 4 was brought into operation in December 1983. When the unit was shut down for medium term maintenance, which was scheduled for 25 April 1986, the core contained 1650 fuel assemblies with an average burnup of 10.3 MW·d/kg, one additional absorber, and one empty fuel channel. Most of the fuel assemblies (75%) were from the first core charge and had a burnup of 10–15 MW·d/kg.

Unit 4 was designed in accordance with the safety codes and standards which were in force in the USSR in the late 1960s and early 1970s. The engineering design for the plant was developed in 1974. The design for the second stage of the plant was approved in the normal way. The design documentation on plant safety was approved by the USSR State Committee for the Supervision of Nuclear Power Safety, the USSR State Supervisory Committee for Industrial Safety and Mining Inspection, and the USSR State Committee for Public Health Supervision in 1975.

II-2. LATEST VIEWS ON HOW THE CHERNOBYL ACCIDENT OCCURRED AND DEVELOPED

The analysis of the circumstances surrounding the Chernobyl accident started on 27–28 April 1986 when specialists gained access to documents giving the main parameters in Unit 4 prior to the accident and during its first stage (until the measurement and recording systems failed). At least two factors were striking: first of all, the high speed of the process; secondly, the fact that there were hardly any control rods in the core during the run-up to the accident. It was more or less evident that the process was uncontrollable and that the reactor runaway had been caused by a positive reactivity effect.

The first version of the events, which was put together in situ, was based on the assumption that the accident had been caused by an uncontrollable reactor runaway triggered by oversteaming in the core fuel channels as a result of a breakdown in coolant circulation in the multipass forced circulation loop. This breakdown in coolant circulation was in turn caused by a mismatch between the feedwater and coolant flow rates.

After more detailed analysis of all the recorded information it became clear that, for a correct understanding of causes of the accident, computerized simulation

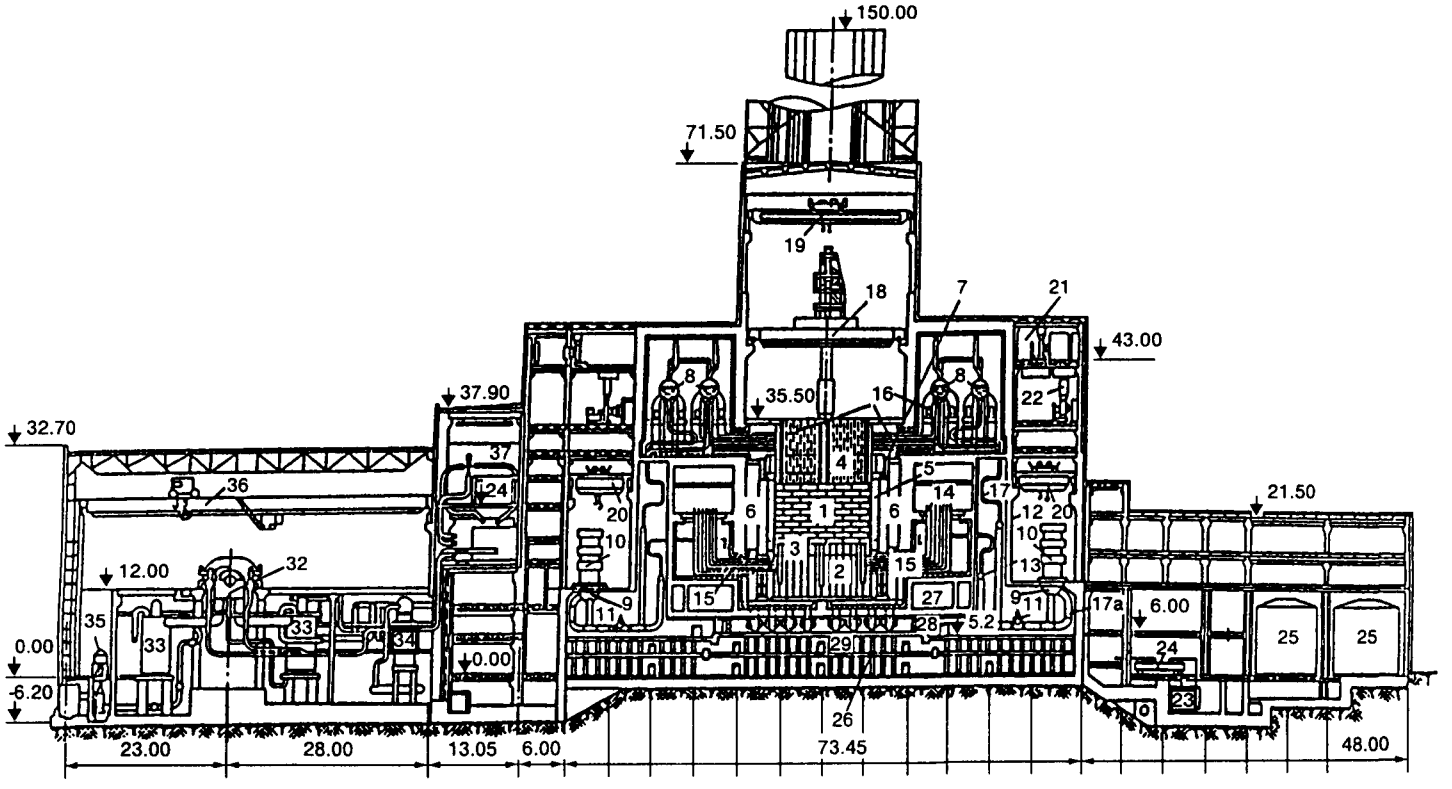


FIG. II-2. Vertical section through the main building of an RBMK unit, including the localization zone. [Numbers refer to itemization of equipment and components in Table II-II. Dimensions are given in metres.]

TABLE II-II. MAIN EQUIPMENT FOR AN RBMK UNIT (*see Fig. II-2*)

Number in Fig. II-2	Equipment/component	Unit	Mass (t)	Number per unit
Reactor section				
1	Graphite stack	Set	1850	1
2	System S metal structures	Set	126	1
3	System OR metal structures	Set	280	1
4	System E metal structures	Set	450	1
5	System KZh metal structures	Set	79	1
6	System L metal structures	Set	592	1
7	System D metal structures	Set	236	1
8	Steam separator drum	Item	278	4
9	TsVN-8 main circulating pump	Item	67	8
10	MCP motor	Item	33	8
11	Main isolation valve, 800 m nominal bore	Item	5.7	8
12	Intake header	Item	46.0	2
13	High pressure header	Item	46.0	2
14	Distributing group header	Item	1.3	44
15	Lower water lines	Set	400	1
16	Steam–water lines	Set	450	1
17	Downcomers, 300 mm nominal bore	Set	16	1
17a	Main coolant circuit 800 mm nominal bore piping	Set	350	1
18	Refuelling machine	Set	450	1
19	Bridge crane in central hall (Q = 50/10 tf)	Item	121	1
20	Bridge crane in MCP room (Q = 50/10 tf)	Item	176	2
21	VDN type suction ventilator at level +43.0	Item	3.5	30
22	Exhaust ventilator at level +35.0	Item	3.5	50
23	Controlled leakage tank	Item	1.4	2
24	Controlled leakage system heat exchanger	Item	0.2	2
25	Planned preventive maintenance tanks	Item	25	4
26	Metal structures and piping in accident localization zone	Set	270	1
27	Non-return valves in lower water lines area	Set	2.5	11
28	Accident localization system bypass valve	Item	2	8
29	Accident localization system condensers	Item	3.7	36
30	Container wagon[a]	Item	146	1
31	Crane in gas activity reduction system area[a] (Q = 30/3 tf)	Item	45	1
	Carbon steel piping	Item	1170	1
	Stainless steel piping	Item	760	1
Machine hall				
32	Turbogenerator, type K-500-65/3000	Item	3500	2
33	Moisture separator/reheater, type 500	Item	15	8
34	Low pressure preheater	Item	37.5	4
35	Condensate pumps	Item	2.5	6
36	Bridge crane in machine hall (Q = 125 tf)	Item	211	1
37	Deaerator	Item	4.5	2
	Carbon steel piping	Set	3825	1
	Stainless steel piping	Set	1300	1

[a] *[Not shown in the cross-section in Fig. II-2.]*

of the accident sequence would be required. This work was undertaken inependently by three organizations (the I.V. Kurchatov Institute of Atomic Energy (IAE), the Scientific Research and Design Institute for Power Technology (NIKIET), and the All-Union Scientific Research Institute for Nuclear Power Plant Operation (VNIIAES)). In addition, much experimental work was carried out at test facilities and in operating reactors.

The significance of the void reactivity effect was assessed more closely, and the way in which the additional absorbers and control rods affect it. It was found that, for a double peaked axial neutron flux distribution, inserting the control rods from the top position could induce a negative reactivity insertion during the first second and then a reactivity rise as rod insertion continued owing to displacement of the water columns in the lower part of the reactor by the graphite displacers.

In August 1986, an analysis of the accident was performed using an integrated model. This analysis formed the basis of the USSR's report to the IAEA. This document states that 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", and that "the accident assumed catastrophic proportions because the reactor 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" [*see Annex I, Ref. [1], Part I, p. 23*].

The subsequent analysis of the accident employed distributed neutron physics models with thermal hydraulics feedback. These models were validated against available experimental data.

The results which were submitted to the IAEA in September 1987 showed that axial–azimuthal effects had played a significant role. At the same time, the research using integrated models continued; with these, it was easier to assess the effects of the various physical processes and individual factors on the accident sequence.

In October and November 1989, various aspects of the Chernobyl accident and the effectiveness of the measures to improve the safety of plants with RBMK reactors were discussed in detail at the first meeting of the international working group on severe accidents and their consequences held in Dagomys (USSR). At this meeting, specialists from the USSR and other countries presented papers in which they analysed the accident sequence using three dimensional neutron physics models, which took into account feedback on thermal-hydraulics. It was generally agreed that the accident was attributable to "reactor instability caused both by design faults and the operating regime." The measures that had been implemented to improve reactor safety were deemed effective.

Foreign participants noted that the actual details of the accident sequence might be different from those produced by the Soviet specialists, and it was recommended that the research continue. So far, three full scale models of the combined physical and thermal-hydraulic processes in an RBMK reactor have been developed, the details of which are given below.

II-2.1. Overview of the test programme that led to the accident

The accident occurred during rundown tests on turbogenerator No. 8 in Unit 4 with the auxiliary power system on load.

The aim of the tests was to verify whether cooldown of the core by forced circulation could be maintained in the event of a total power loss.

In the event of a total power loss during a DBA, the feedwater pumps, which form part of the third subsystem of the ECCS, are supposed to be powered from the rundown current of the turbogenerator.

In 1982, some similar tests were performed in Unit 3 of the Chernobyl plant which showed that the excitation control system of the turbogenerator would require retrofitting if an acceptable current level was to be maintained for the required period of time during rundown of the turbogenerator. Additional rundown tests with the retrofitted excitation system were carried out in 1984 and 1985. In the 1982 and 1984 test programmes, one MCP from each side of the reactor was connected up to the turbogenerator in rundown mode, whereas in the 1985 and 1986 programmes two MCPs were connected up from each side. In the 1984, 1985 and 1986 test programmes, the ECCS was isolated using the manually operated valves. In the light of present day approaches to the drawing up of such test programmes, this programme is unsatisfactory primarily as regards the safety measures employed.

It is incorrect to qualify these tests as purely electrical since they involve changes in the power supply circuit for main equipment in the unit and interventions on the normal protection and interlock systems.

According to the Operating Procedures for Units 3 and 4 of the Chernobyl Nuclear Power Plant [*see Annex I, Ref. [40]*] (1Eh-S-11, p. 11), the plant management is responsible for preparing any programmes, drawings and technical documentation relating to assemblies, sensors, absorbers and other devices intended for use in the reactor while it is operating at power that are not covered by the reactor design; it is also supposed to clear them with the Scientific Manager, the Chief Design Engineer and the Chief Designer, and have them approved by the USSR All-Union Production Group Soyuzatomehnergo. Any modifications to design components and systems in the reactor must be cleared with the Scientific Manager, the Chief Design Engineer and the Chief Designer, and approved by Soyuzatomehnergo. Any changes which have nuclear safety implications must also be cleared with the USSR State Committee for the Supervision of Nuclear Power Safety.

One specific thermal-hydraulics feature of the test regime was the increased initial coolant flow rate through the reactor over the rated level. The steam quality was at minimum level and the coolant temperature at the core inlet was slightly below boiling point. Both the above factors turned out to have a direct bearing on the scale of the effects which occurred during the tests.

II-2.2. Chronology of events at the Chernobyl plant on 25–26 April 1986

[*Table II–III presents the chronology of events at the Chernobyl plant on 25–26 April 1986.*]

TABLE II-III. CHRONOLOGY OF EVENTS AT THE CHERNOBYL PLANT ON 25-26 APRIL 1986

Time	Event
25 April 1986	
(time in operating log)	
01:06	Start of unit power reduction; operational reactivity margin (ORM) equals 31 manual control rods
03:45	Replacement of nitrogen-helium mixture used to cool the graphite stack with nitrogen commences
03:47	Thermal reactor power 1600 MW
07:10	ORM equals 13.2 manual control rods
13:05	Turbogenerator 7 disconnected from the grid
14:00	ECCS isolated from the circulation circuit; test programme postponed by request of the grid controller Kievehnergo
15:20	ORM equals 16.8 manual control rods
18:50	Auxiliary power equipment not required for tests switched over to operating transformer T-6
23:10	Unit power reduction continues; ORM equals 26 manual control rods
26 April 1986	
00:05	Thermal reactor power 720 MW; steady unit power reduction continues
00:28	With thermal reactor power at around 500 MW, switchover from local power control system (LAC) to main range automatic power controllers 1 and 2; an unplanned drop in thermal power to 30 MW was permitted during this switchover (neutron power dropped to zero); after a pause lasting 4 to 5 min, a power raise was initiated;
(time on printout of DREG)	
00:34:03 00:43:37 00:52:27 01:00:04 01:09:45 01:18:52	Level in steam separator drums went beyond the triggering set point for the emergency protection system of −600 mm (emergency set point for a level drop was left at −1100 mm)

TABLE II–III. (cont.)

Time	Event
26 April 1986 (cont.)	
00:41–01:16 (in operating log)	Turbogenerator 8 disconnected from grid to check vibrational characteristics on no load running
00:43:37	The emergency protection system which shuts down both turbogenerators was disabled
01:03 (in operating log)	Thermal reactor power raised to 200 MW and stabilized at this level
01:03 (in operating log)	MCP-7 switched on
01:06	Feedwater flow rate raised to 1200–1400 t/h to restore level in steam separator drums
01:07 (in operating log)	MCP-8 switched on
01:09	Sudden reduction in the feedwater flow rate to 90 t/h on the right hand side and 180 t/h on the left hand side for a total flow rate in the circuit of 56 000–58 000 t/h; as a result, the temperature at the MCP inlet reached 280.8°C (left side) and 283.2°C (right side)
01:18:52	DBA signal (according to the DREG program)
01:22:30	Parameters of the SKALA centralized monitoring system recorded on magnetic tape; non-measurable parameters at the Chernobyl plant were not calculated; after the accident, the ORM was calculated using the standard axial power distribution curve in the PRIZMA program, which yielded a value of 1.9 manual control rods; the calculations using real data on the axial power distribution yielded an ORM value of 6–8 manual control rods
01:23:04	'Oscillograph on' command given, stop valves for turbine No. 8 closed, turbine rundown operation commences; according to the oscillograph that was recording the electrical parameters of the MCPs, the MCPs that were connected to the turbogenerator in rundown mode operated for 36.2 s; the DBA button was pressed[a]; the exact times at which the DBA button was pressed and at which the oscillograph was turned on are not known

TABLE II-III. (cont.)

Time	Event
26 April 1986 (cont.)	
01:23:40	EPS-5 (emergency protection system 5) signal recorded; according to the explanatory notes written by personnel, the EPS-5 button was pressed; insertion of emergency protection and manual control rods into the core commenced
01:23:43	All the lateral ionization chambers generated emergency protection signals for a reactor runaway and overpower
01:23:47	Sudden drop in flow rate (by 40%) in MCPs not involved in the rundown operation, and misleading readings for the flow rates in the MCPs which were involved in the rundown operation; sudden rise in the pressure and level in steam separator drums; signals for a measurement fault in both the main range automatic controllers
01:23:48	Flow rates in MCPs not in the rundown operation restored almost to initial levels; continued rise in the level and pressure in the steam separator drums (left side 75.2 kg/cm^2; right side 88.2 kg/cm^2); triggering of fast acting steam dump to condenser systems for turbine
01:23:49	Emergency protection signal for a pressure rise (fuel channel rupture) in the reactor space; 'No voltage equals 48 V' signal (RCPS rod drive couplings de-energized); signals indicating a fault in the actuators for main range automatic controllers 1 and 2
01:24 (in operating log)	Severe shocks recorded; RCPS rods stopped before reaching limit switches; power supply to RCPS rod drive coupling switched off

[a] A DBA button had been specially installed for the test programme to simulate the DBA signal. The signal was to be transmitted to the startup circuit of the diesel generator and to the turbogenerator rundown system to start that. The button was to be pressed when the stop valves closed.

II-2.3. Data on information sources

The processes prior to and during the accident were analysed using data from the following instruments and computer systems.

II-2.3.1. Standard recording equipment

This equipment is used to record relatively slow processes (maximum tape speed 240 mm/h). It can record extreme parameter values rather accurately but is not suitable for reconstructing fast transients.

II-2.3.2. SKALA centralized monitoring system and subsystems

This system calculates the main reactor parameters approximately every 5 min (this limitation is due to the capacity of the V-3M type computer). Of course, this cycle is too long for the analysis of fast processes.

The diagnostic parameter recording program (DREG) scans and records several hundred discrete and analogue signals. However, it does not record the main reactor parameters such as the power and reactivity levels, the coolant flow rate in the channels and other mass parameters. The positions of only nine of the 211 RCPS rods are recorded, including one rod in each of the three groups of automatic controllers. These parameters are not directly measurable and their scan cycle is therefore much longer (1 min). Although the recording cycle for some parameters is short (1 s), the length of the scanning interval may be rather uncertain as the DREG program has a very low priority in the SKALA system. Moreover, during the hour preceding the accident there were three interruptions in the running of the DREG program caused by a SKALA system restart. This resulted in additional losses of information. The cycle for other results produced by the SKALA system is longer (5 min), including the PRIZMA program and the recording of reactor status data on magnetic tape (RESTART); and there are interruptions caused by system restarts and certain software features. In addition, the results produced by the PRIZMA program are recorded only on printouts. The process parameter data produced by the DREG program are given.

II-2.3.3. Oscillography

A non-standard system employing oscillographs to record rapidly varying parameters was installed for the test programme. It monitored the operating parameters of certain items of electrical equipment. However, these electrical equipment parameters were not synchronized with the reactor parameters recorded by the system.

II-2.3.4. Other information obtained from operational sources

This category includes entries in the operating logs and the tape recordings of telephone calls made by personnel. This information and the explanatory notes written by personnel did not turn up any further essential data above and beyond that recorded by the instrumentation. In some cases, the accounts of the sequence of

events given by personnel were highly inaccurate. Most importantly, the majority of people reported that there had been two explosions. The operating logs also contain entries which confirm the chronology just given of events before and after the accident.

II-2.4. Initial explanations of the accident

Once the nature of the damage and some details of the accident became known, a number of explanations and scenarios were put forward in an attempt to understand and explain what had happened. Some of these were rather fanciful and others were quite convincing. Most of them were rejected when additional information on the damaged unit became available, or pursuant to more detailed analysis.

The analysis centred around attempts to identify contradictions between the expected effect of a particular explanation of the accident and the available factual data recorded by the DREG program.

II-2.5. Computerized simulation of the accident

II-2.5.1. One dimensional simulation

A one dimensional program was used to simulate the power reduction transient which took account of iodine–xenon kinetics and feedback on the temperatures of graphite and fuel and on the coolant density.

Figure II-3 shows the power reduction curve (the power reduction started at 01:06 on 25 April). Figure II-4 shows the time dependence of the ORM and indicates the actual ORM values where these are known.

It is clear that the ORM dropped initially below the minimum permissible level of 15 manual control rods owing to the 50% power reduction. By 22:00 on 25 April, it had been restored almost to its initial value. A further power reduction soon followed, and the ORM started to fall reaching a value of no more than 6–8 manual control rods prior to the accident. Figure II-4 also shows the axial peaking factor K_z as a function of time. The shape of the axial field is shown in Fig. II-5. Prior to the accident, the field had a double peaked shape with a large upper maximum.

II-2.5.2. Characteristics of the three dimensional models

Model 1 (developed by the I.V. Kurchatov Institute of Atomic Energy)

The neutron field is described by a system of two-group equations with a point for each cell in the finite difference grid. The WIMS code is used to calculate the neutron cross-sections. The initial status can be adjusted to match the readings from the in-core detectors. The thermal hydraulics module describes the section of the circulation circuit from the high pressure header to the steam separator drums. The

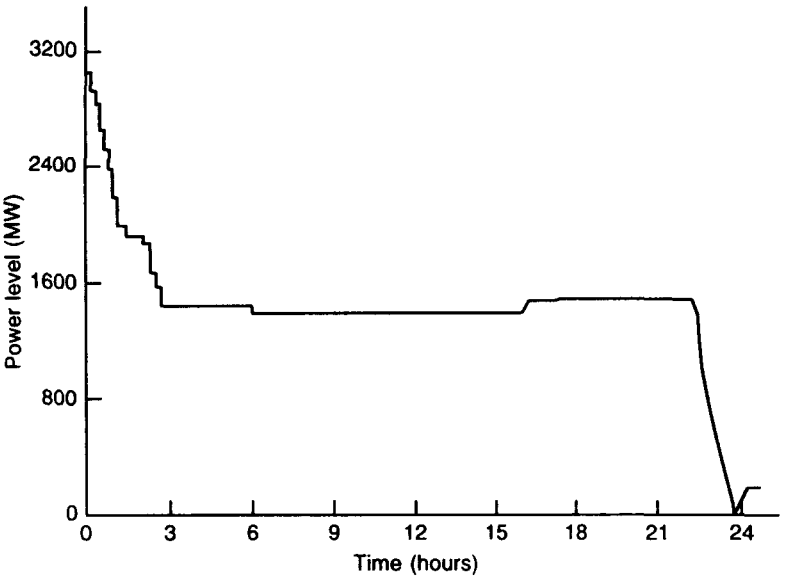


FIG. II-3. The Chernobyl accident: change in power level with time (from initiation of the power reduction).

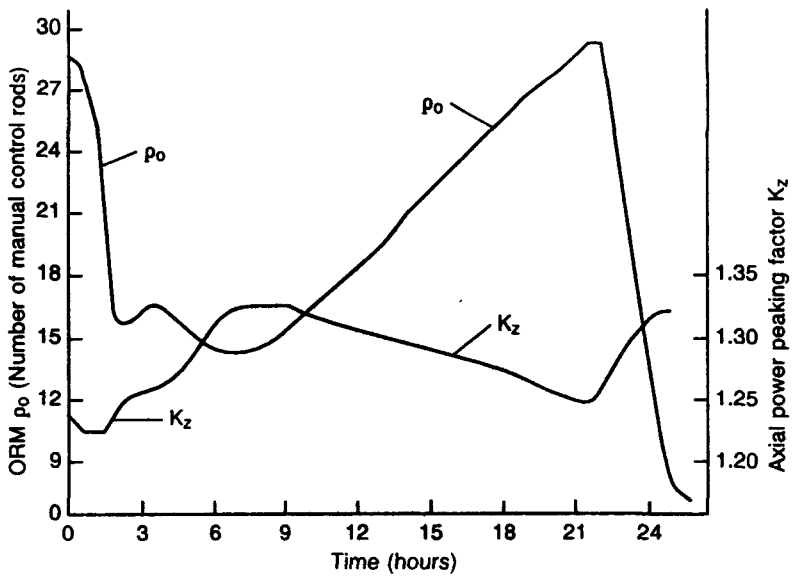


FIG. II-4. The Chernobyl accident: change in the operating reactivity margin ρ_o and the power peaking factor (non-uniformity factor of the axial field) K_z during the transient.

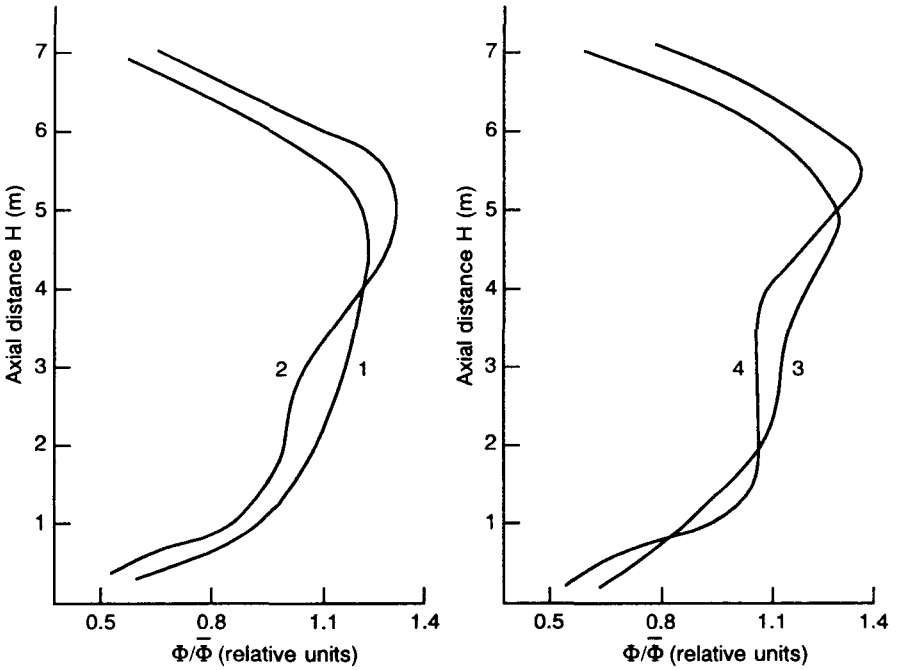


FIG. II-5. The Chernobyl accident: axial distribution of the thermal neutron flux density in the core at various times during the xenon transient preceding the accident: 1: 00:00; 2: 07:00; 3: 22:00; 4: 24:30 (before the accident).

model has been tested against a large number of steady state modes and transients in operating reactors.

Model 2 (developed by the Scientific Research and Design Institute for Power Technology (NIKIET))

The neutron field is described by a one-group equation. The number of difference grid nodes in the reactor plan varied from 140 to 2000. The constants were derived from the two-group constants that were obtained using the WIMS code. The thermal hydraulics module describes the section of the circuit from the distributing group header to the steam separator drums. A single fluid homogeneous model is used for one dimensional flow of the two phase coolant with empirical corrections for phase slipping.

Model 3 (developed by VNIIAES in collaboration with the Nuclear Research Institute of the Academy of Sciences of the Ukrainian SSR, Kiev)

The neutron field is described by a one-group equation with one point for every four cells. The constants were derived from data that were obtained using the WIMS

code. The thermal hydraulics diagram of the reactor circuit is described in detail, including the two coolant circulation loops. The equations from the single temperature variant of the flow model with slipping are used. The slipping functions and the flow regime map are taken from the TRAC-PIA code.

Thus, these three models differ significantly from one another. On the other hand, the fact that we have three independent models and the opportunity to compare results enhances the validity of the main conclusions drawn pursuant to the analysis of the accident sequence.

II-2.5.3. Calculation results

First of all, the positions of all the controllers (Fig. II-6) and the power density field for 01:22:30 on 26 April 1986 were reconstructed. The radial–azimuthal distribution is shown in Fig. II-7, and the readings from the axial distribution sensors in Fig. II-8. At the point in time corresponding to 01:23:40 on 26 April 1986, the insertion of all the RCPS rods into the core pursuant to the EPS-5 signal was simulated.

All three models indicate a drop in reactivity during the first one second. Then, the neutron field is redistributed, the maximum shifts to the bottom of the core

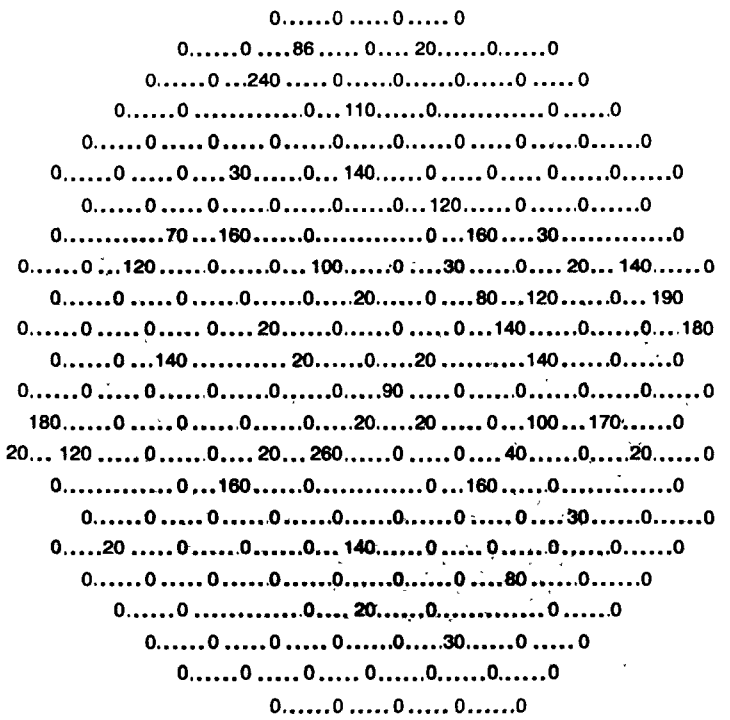


FIG. II-6. The Chernobyl accident: plan of the core showing the insertion depths of the RCPS rods in centimetres.

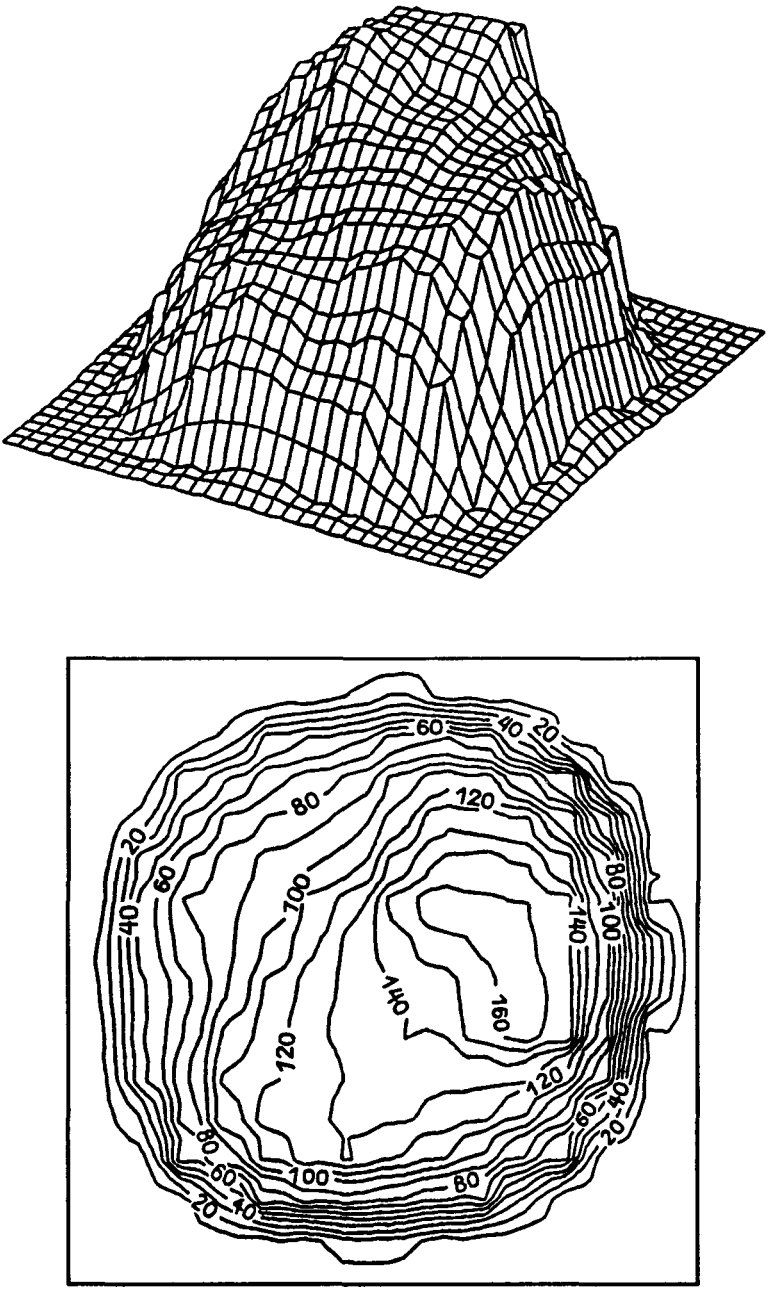


FIG. II-7. The Chernobyl accident: radial distribution of the initial neutron field in the core.

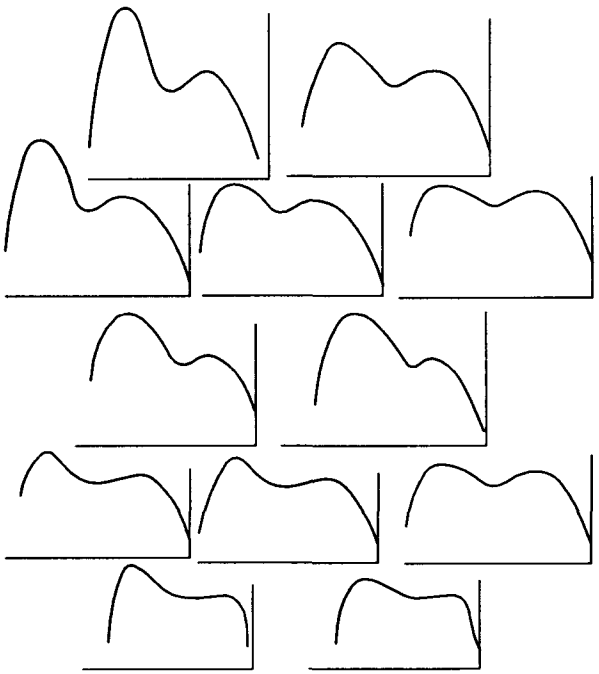


FIG. II-8. The Chernobyl accident: distribution of the power density profile over the core volume as recorded by the axial detectors.

(Fig. II-9), and the reactivity and the integrated power level start to increase. Under abnormal conditions, this effect is caused by control rod design features: when the rod is inserted into the core from the top position (Fig. II-10), the multiplication factor in the bottom of the core increases owing to replacement of the neutron absorbing water column by the graphite displacer. The changes in the reactivity and the integrated power level which were generated by Model 1 are shown in Fig. II-11.

Owing to the low departure from nuclear boiling (DNB) ratio of the coolant at the reactor inlet immediately prior to the accident, the effect which the positive void coefficient of reactivity had on the power rise was significantly enhanced.

The various models yield strikingly different values for the absolute level of the integrated power surge, ranging from 3.5 times the initial value in Model 2 to approximately 80 times the initial value in Model 1; Model 3 yields an intermediate value of around 9 times the initial value. All three models display a high level of non-uniformity of the volumetric field (approximately 6.0), even though the deformation of the radial field differs significantly from model to model; this is typical for large scale reactors.

However, one fact cannot be avoided: none of the three models reproduces the reactor runaway when, three seconds after EPS-5 is triggered, signals are generated indicating that the emergency set points for the power level and the rate of power rise have been exceeded.

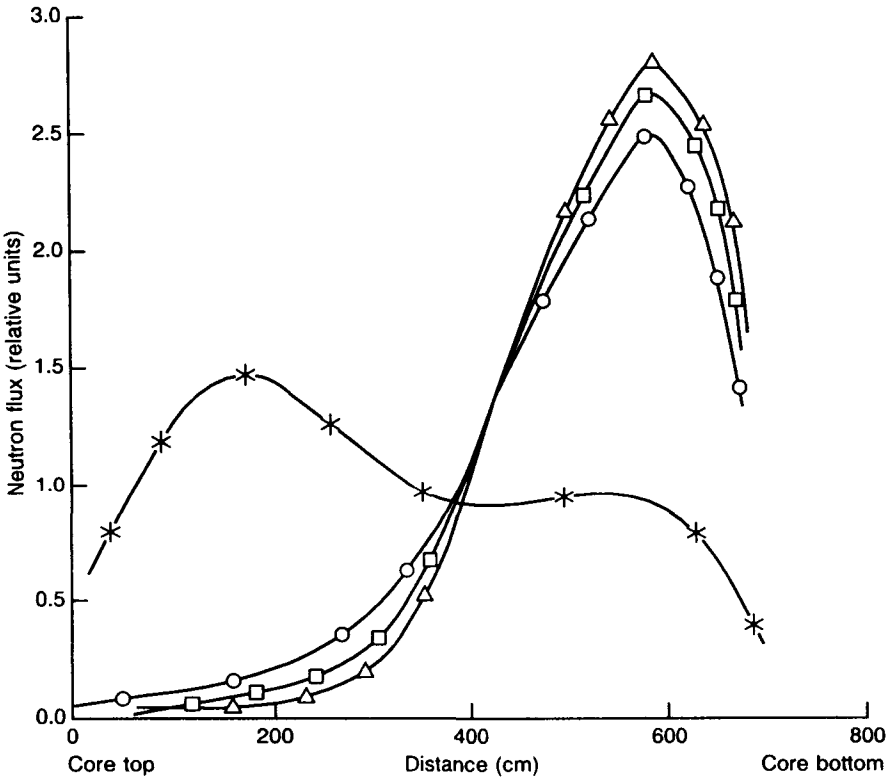


FIG. II-9. The Chernobyl accident: distortion of the axial neutron fields during rod insertion at the time when the accident occurred. The displacement of the neutron flux maximum to the lower part of the core is clearly shown. Key: time after EPS-5 button was pressed: ∗: 0 s; o: 4 s; □: 6 s; Δ: 7 s.

The models need to be further refined both with regard to their neutron physics and their thermal-hydraulic elements to achieve a better fit between the calculation results and the experimental data.

One highly important point which was noted by all the groups of research workers is the very strong dependence of the characteristics of the transient on the potential error level of the initial axial distribution, the void reactivity effect value, the change in the reactivity level when the water column in the RCPS channel is replaced by the graphite displacer, and many other factors.

By eliminating the major deficiencies which cause reactor instability we can preclude the possibility of an accident of this kind recurring. Indeed, calculations using the same models and for the initial reactor states already analysed, but with present day parameters, show that the reactor is shut down sufficiently quickly. Thus, all the models can be said to yield equivalent results.

Nevertheless, the results obtained from the analysis of the first phase of the accident cannot be viewed as final: the models could probably be made to agree more

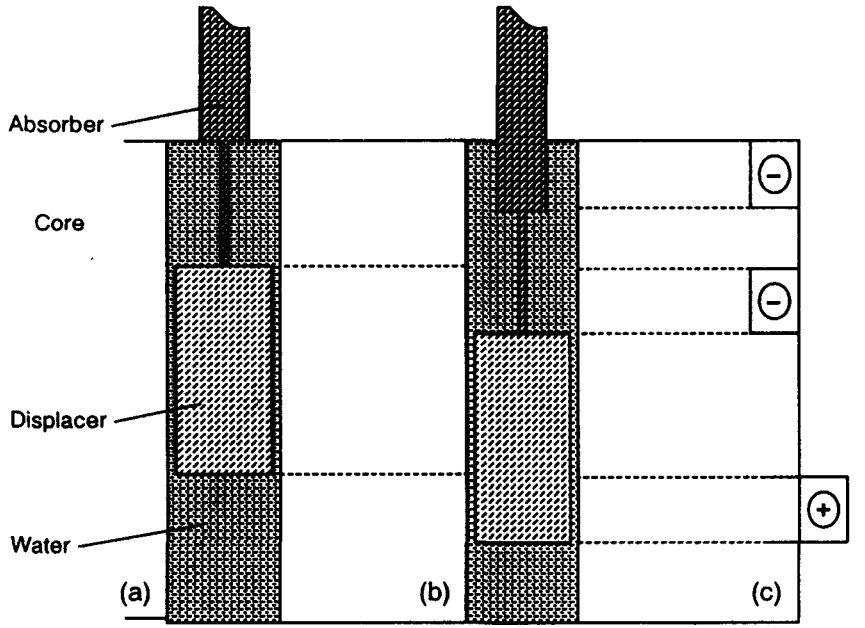


FIG. II-10. The RBMK reactor: schematic representation of the reactivity insertion caused by inserting the RCPS rods from the top position. (a) Manual control rod withdrawn; (b) manual control rod partially inserted; (c) the change in the theoretical reactivity insertion δk_∞ as the rod is inserted.

closely if the thermal-hydraulic modules were improved since, in recent years, greater emphasis has been placed on neutron physics models (although the methods used to generate group constants also need to be improved).

Much less attention has been paid to those stages of the accident where the damage to the reactor and the unit buildings occurred. However, in order to reach an understanding of these processes the second phase of the accident would need to be studied in greater depth, particularly since a large number of data on the damaged unit have accumulated in the interim.

Calculations performed independently using the three three-dimensional dynamic models have shown that spatial factors played an important role during the accident. Also, all the available data have to be included in the accident analysis; in particular, the pre-accident status of the reactor must be reconstructed in detail starting from the point when the power reduction was initiated one day before the accident occurred.

The main factors that affected the accident sequence were the positive void reactivity effect and RCPS design deficiencies which caused the insertion of positive reactivity under the reactor conditions prevailing prior to the accident. These conclusions underlie the measures which were elaborated to improve the safety of plants with RBMK reactors.

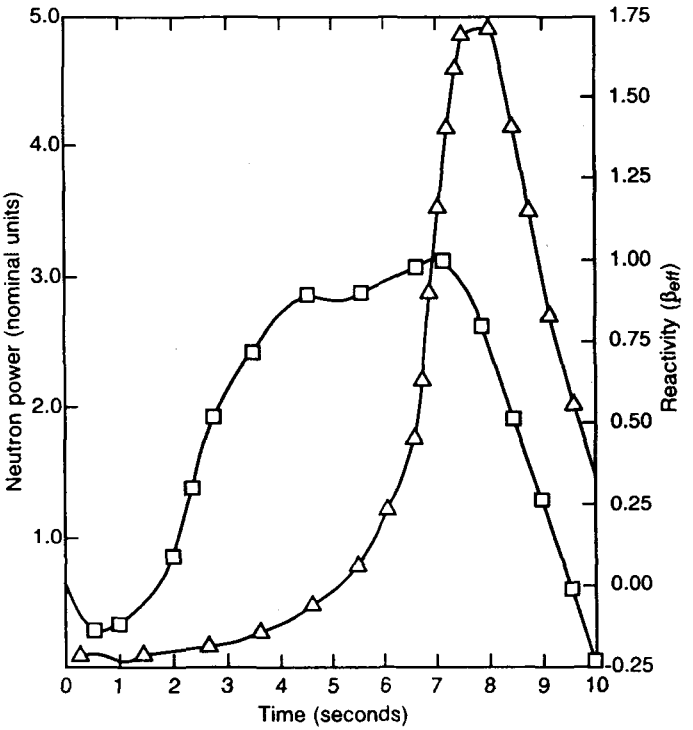


FIG. II-11. The Chernobyl accident: time dependence of the reactivity level and the neutron power during the initial phase of the accident. □*: reactivity;* Δ*: neutron power.* [Reactivity is expressed in units of effective delayed neutron fraction β_{eff}.]

II-3. MEASURES TO IMPROVE THE SAFETY OF PLANTS WITH RBMK REACTORS

Pursuant to the analysis of the causes of the accident in Chernobyl Unit 4, organizational and technical measures were developed to improve the safety of operating plants with RBMK reactors.

First of all, measures were developed and put into effect which aimed at:

- reducing the positive void coefficient of reactivity;
- improving the response efficiency of the emergency protection system;
- introducing ORM calculation programs which output a numerical indication of the current ORM value on the operator's control panel;
- preventing the emergency protection systems from being shut down while the reactor is operating at power;
- avoiding regimes which cause a reduction in the DNB ratio of the coolant at the reactor inlet.

The reduction in the void coefficient of reactivity was implemented in two steps. First of all, its value was reduced to $+\beta$ by installing additional absorbers in the core (80–90) and increasing the ORM to 43–48 RCPS manual control rods. Table II-IV gives the measured values for the void and fast power coefficients in all units with RBMK reactors currently in operation. Figure II-12 shows the dependence of the reactivity level on the coolant density both before and after the safety measures were implemented, and the design basis dependence. The second step in reducing the void coefficient of reactivity involves converting all the reactors to use fuel with a 2.4% ^{235}U enrichment level.

Three independent retrofitting operations were carried out to improve the efficiency and response speed of the emergency protection system.

First of all, the manual control rods were replaced with ones of improved design which do not give rise to water columns at the bottom end of the RCPS channels and have a larger absorber section (Fig. II-13).

The RCPS rod drives were retrofitted, reducing the time required to insert the rods fully into the core from 18 to 12 s. The implementation of these first two measures has improved by several times the response efficiency of the emergency protection system during the first few seconds of rod insertion.

As a third measure to improve the performance of the emergency protection system, a fast acting emergency protection system was developed and systems of this kind were installed in all operating reactors. The actuator in this system cools the channel wall with a thin film of water and moves the rod in a gas medium. In 1987–1988, this new system was put through full scale testing at the Ignalina and Leningrad plants. The tests confirmed the design characteristics of the system, i.e. 24 fast acting EPS rods in less than 2.5 s cause an insertion of negative reactivity of over 2β [*where β is the delayed neutron fraction*] (Fig. II-14). All RBMK reactors are now equipped with these systems.

Figure II-15 shows the improved level of response efficiency which has been achieved by retrofitting the RCPS rods and installing the fast acting emergency protection system by comparison with the response efficiency of the system at the time of the accident in Unit 4 in terms of reactivity change curves.

By implementing the above measures to improve the neutron physics characteristics of the reactor and significantly improving the performance of the emergency protection system, we have been able to preclude the possibility of an uncontrolled power rise during a loss of coolant accident (LOCA) and limit the consequences of all DBAs to the permissible levels for radiation exposure of personnel, the general public and the environment.

The operational documentation was updated to take account of the results of the accident analysis and the measures which have been implemented to improve the safety of plants with RBMKs.

Of course, there are six first generation RBMK units which have no reinforced leaktight compartments and no special structures to localize radioactive releases. The main aim in these units must be to reduce the probability of large diameter pipe

TABLE II-IV. OPERATING CHARACTERISTICS OF PLANTS WITH RBMK REACTORS IN OPERATION

Plant	Unit	Measurement date	Mean burnup per fuel assembly (MW·d/assembly)	Number of additional absorbers	Number of water columns	Reactivity margin (rods)	Number of fuel assemblies with a 2.4% enrichment level	Void coefficient of reactivity α_φ (β)	Fast power coefficient of reactivity α_w ($\beta \times 10^{-4}$)
Leningrad	1	29.03.1991	1114	88	2	47.9	917	0.7	−2.7
	2	06.12.1990	1280	80	3	46.5	1305	0.8	−2.8
	3	28.10.1990	1230	80	0	47.0	1367	0.8	−2.7
	4	15.03.1991	1229	80	12	46.0	1444	0.9	−2.7
Kursk	1	20.07.1990	1011	94	4	45.8	579	1.0	−1.2
	2	15.02.1991	1001	99	21	44.5	833	1.0	−1.45
	3	11.08.1990	1079	81	9	44.5	789	1.0	−1.25
	4	04.03.1991	1164	81	0	50.4	1134	0.9	−1.3
Chernobyl	1	27.02.1991	1143	80	13	46.5	774	0.87	−1.85
	2	26.02.1991	1101	81	1	48.4	787	0.66	−1.45
	3	17.02.1991	1140	81	1	46.5	769	0.66	−1.2
Smolensk	1	13.12.1990	1121	81	3	50.6	1009	0.5	−2.58
	2	08.05.1990	1145	81	0	48.5	1145	0.89	−2.98
Ignalina	1	13.10.1990	885	52	2	55.0	0	0.8	−1.9
	2	09.02.1991	865	54	2	54.1	0	1.0	−1.6

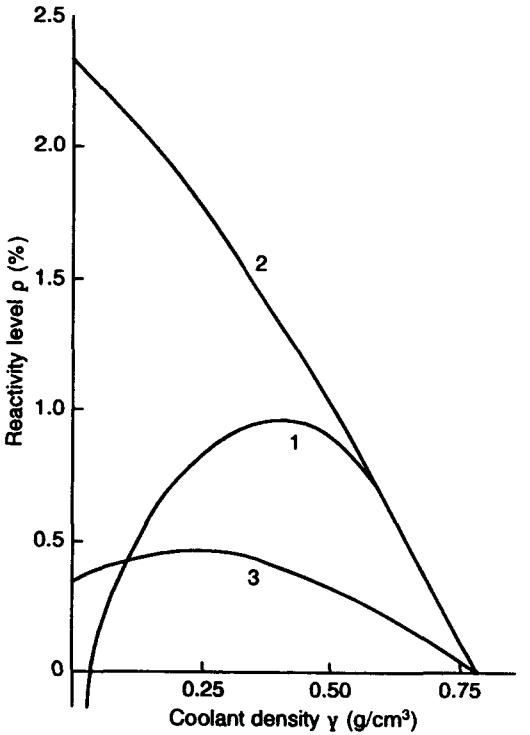


FIG. II-12. The RBMK reactor: dependence of the reactivity level ρ on the coolant density γ. 1: design calculations; 2: actual dependence at the time when the accident occurred on 26 April 1986; 3: current status after improvements.

breaks up to a point where such incidents might be termed hypothetical. With this in mind, some computerized and experimental research was carried out into the processes which cause cracks to appear and propagate in metal, and critical crack sizes were determined for cracks which might cause a pipe break. Pursuant to this research, a set of procedures were elaborated for enhanced monitoring of the metal in large diameter piping. These set out requirements for monitoring equipment, the periodicity of monitoring, the training and qualifications required by flaw detector operators, etc.

Over the period 1986–1990, the welded joints in the 800 mm nominal bore piping and 900 mm nominal bore headers in the circulation circuit were subjected to ultrasonic testing in all units. VNIIAES, in the light of the results of this recent testing, estimates the probability of a break in piping of this kind at 0.75×10^{-6} per reactor-year. 100% monitoring of the state of the metal and hydrostatic tests are carried out once every four years. Work is under way to develop automated non-destructive testing (NDT) systems which would help eliminate the manual work done by the flaw detector operators and thus reduce the probability that faults might be overlooked. We plan to install prototype automated NDT systems in 1991.

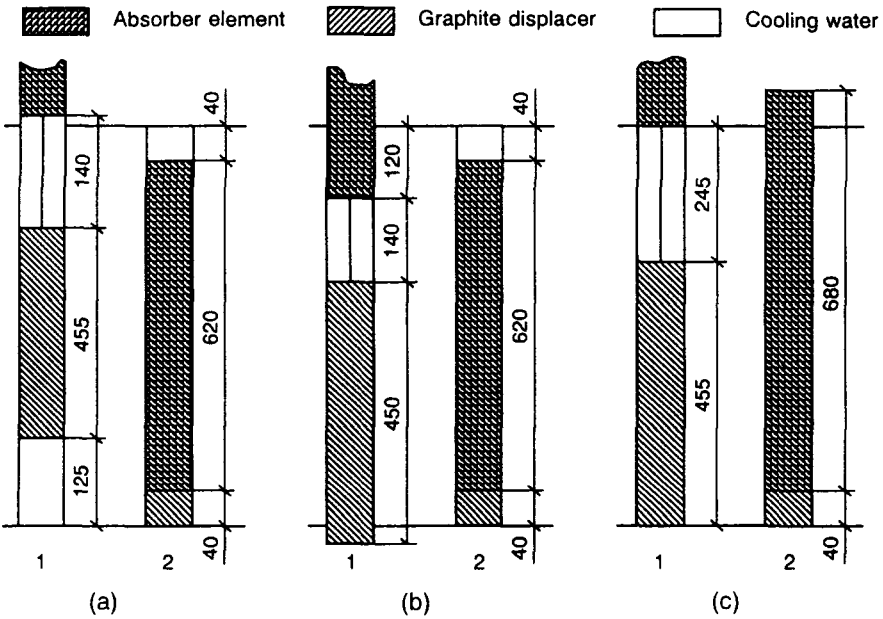


FIG. II-13. Manual control rods in the RBMK reactor. (a) old design rod; (b) old design rod inserted into the core to avoid the insertion of positive reactivity; (c) retrofitted rod. 1: rod withdrawn; 2: rod inserted. [Dimensions are given in centimetres.]

One of the problems of pressure tube type reactors which was widely discussed after the Chernobyl accident is the possibility of a simultaneous rupture of a large number of fuel channels. The existing steam dump system for the reactor space is designed to cope with the steam discharge from a simultaneous rupture of two channels, the probability of such an event being low.[2]

Over the period in which these units have been operating, several incidents involving fuel channel damage have occurred. In two of these incidents (which were caused by local power tilting and an interruption in coolant flow triggered by procedural breaches), the fuel channels and the fuel assemblies in them were almost totally destroyed. However, in both cases there was no damage to the adjacent fuel channel pressure tubes which are all still operating normally. Nevertheless, despite the relatively low probability of an event of this kind, an emergency steam dump system for the reactor space has been installed in the recently commissioned Unit 3 at the Smolensk plant which is designed to cope with the steam discharge from a simultaneous rupture of 9–10 fuel channels. Similar measures will be implemented at all operating units in the course of the planned retrofitting programmes for these reactors.

Retrofitting projects are currently being elaborated for first generation units with RBMK reactors in order to bring them into line as far as possible with recently

[2] NIKIET estimates it at 10^{-8} per reactor-year.

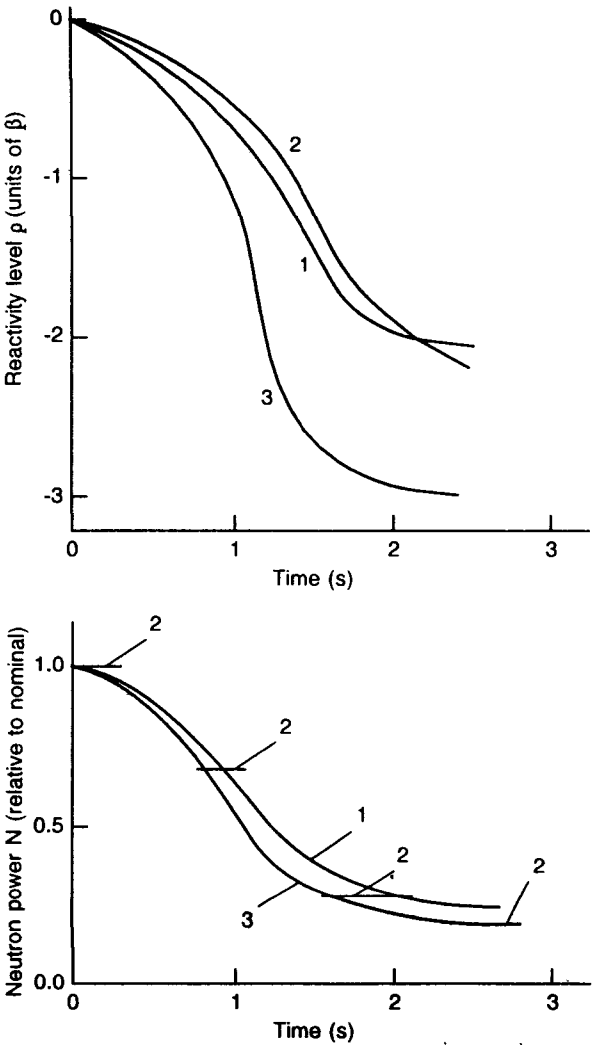


FIG. II-14. The RBMK reactor: reactivity level ρ and neutron power N: testing of the fast acting emergency protection system at the Leningrad and Ignalina nuclear power plants. $N_p = 0.4\ N_{nom}$. 1: theoretical calculations; 2: experiment at the Leningrad plant; 3: experiment at the Ignalina plant.

introduced, more stringent requirements. These projects are based on a retrofitting design which has gone through many stages of discussion and authorization.

The retrofitting design provides for the following:

— development of more efficient emergency core cooling systems;
— total replacement of the reactor control and protection systems and development of a multizone power density monitoring and emergency protection system which works on signals from in-core detectors;

TABLE II-V. PERFORMANCE DATA FOR PLANTS WITH RBMK REACTORS OVER THE PERIOD 1986–1990

Plant/ unit		1986		1987		1988		1989		1990	
		Electricity output (10^9 kW·h)	Load factor (%)	Electricity output (10^9 kW·h)	Load factor (%)	Electricity output (10^9 kW·h)	Load factor (%)	Electricity output (10^9 kW·h)	Load factor (%)	Electricity output (10^9 kW·h)	Load factor (%)
Leningrad	1	6.42	73.3	8.11	92.6	7.17	81.9	4.96	56.7	0	0
	2	7.75	88.4	7.23	82.5	7.38	84.3	6.65	76.0	6.61	75.4
	3	7.53	86.0	6.02	68.7	7.60	86.8	7.56	86.3	8.20	93.7
	4	7.69	87.8	7.32	83.7	6.59	75.2	8.07	92.1	8.45	96.5
Kursk	1	6.14	70.1	7.02	80.2	7.12	81.1	6.09	69.5	5.53	63.2
	2	6.07	69.3	7.20	82.2	6.45	73.5	6.65	75.9	5.24	59.8
	3	5.26	60.0	5.46	62.3	7.26	82.7	6.44	73.5	7.48	85.4
	4	6.84	78.1	6.17	70.4	7.19	81.8	6.68	76.2	6.05	69.1
Chernobyl	1	3.55	40.6	6.41	73.1	6.12	69.9	6.57	75.0	5.65	64.5
	2	3.16	36.0	6.78	77.4	6.62	75.6	6.92	79.0	4.81	54.9
	3	2.36	27.0	—	—	6.95	79.4	6.767	77.2	6.86	78.3
Smolensk	1	3.29	37.5	7.44	85.0	7.21	82.1	7.06	80.6	6.76	77.2
	2	7.20	82.2	6.36	72.7	7.29	83.0	7.17	81.8	7.25	82.8
	3	—	—	—	—	—	—	—	—	5.09	58.1
Ignalina	1	9.88	75.2	6.65	50.6	5.34	48.3	8.94	81.6	8.07	73.2
	2	—	—	3.53	—	7.55	68.9	7.71	70.4	8.95	81.8
Total		84.8		91.0		103.9		104.2		101.0	

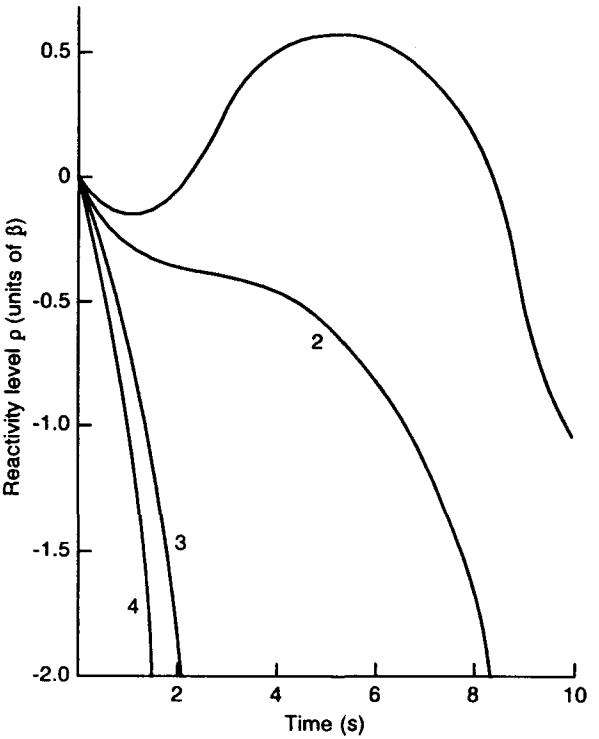


FIG. II-15. The RBMK reactor: efficiency of the emergency protection system. 1: the Chernobyl plant on 26 April 1986, ORM = 7 manual control rods (DBA conditions); 2: ORM as specified in the Operating Procedures (15 manual control rods); 3: retrofitted RCPS, ORM = 30 manual control rods; 4: retrofitted RCPS, fast acting emergency protection system in operation.

— multitrain redundancy of safety systems;
— full-scale introduction of automated metal monitoring systems;
— improvement of the efficiency and reliability of the auxiliary power systems;
— improvement of the resistance of reactor structures and components to earthquakes;
— improvement of the capacity of the steam dump system for the reactor space, and other measures.

During the retrofitting outages we also plan to replace all the fuel channels in all reactors. On the whole, units with RBMK reactors have a good performance record (Table II-V).

As of the beginning of 1991, the installed capacity of plants with RBMK reactors accounted for 16.5 GW, or some 45% of the total installed capacity of nuclear power plants in the USSR (36.6 GW). In 1990, RBMK plants generated 47.8% (101.0×10^9 kW·h) of the total quantity of nuclear electricity generated in the country (211.5×10^9 kW·h).

II-4. CONCLUSIONS

Since the Chernobyl accident, a great deal of analytical work has been done. The aims of this work were as follows: a more accurate determination of the status of Unit 4 prior to the accident; the processing of factual data from the monitoring system on the accident sequence; the analysis of the status of the unit after the accident; and mathematical modelling of the first phase of the accident.

On the basis of this research, and the discussions which have been held at various meetings (including international meetings) on the results obtained, the following main conclusions may be drawn:

(1) The accident was caused by interaction of the following main factors: the physical characteristics of the reactor; specific design features of the control elements; and the unauthorized state into which the reactor was brought.

(2) Thanks to the availability of new, up to date computer programs, the use of powerful computers, and experimental studies which have been carried out on the coolant vaporization effect in RBMKs, we have been able to define more accurately the main physical parameters of the reactor and thus develop new sets of requirements for systems which improve reactor safety.

(3) The safety level of RBMK reactors has been significantly enhanced by a series of organizational and technical measures which have been implemented as part of the 'Comprehensive Action Plan'. These measures include: modification of the physical characteristics of the reactor by installing additional absorbers; conversion to fuel with a 2.4% enrichment level; installation of a fast acting emergency protection system; updating of operational documentation and improvement of staff qualifications; tightening up of the operating procedure requirements.

LIST OF ACRONYMS AND SYMBOLS

AA	Additional absorber
DBA	Design basis accident
DNB	Departure from nuclear boiling
DREG	Diagnostic parameter recording program
ECCS	Emergency core cooling system
EPS	Emergency protection system
FAEP	Fast acting emergency protection
GKNT	State Committee for Science and Technology of the USSR Council of Ministers
GPAN	Scientific and Technical Centre of the SCSSINP
GSP	General Safety Provisions for Nuclear Power Plants
IAE	I.V. Kurchatov Institute of Atomic Energy
IBRAE	Nuclear Safety Institute
INSAG	International Nuclear Safety Advisory Group (of the IAEA)
LAC-LAP	Local automatic control and local automatic protection system
LEP	Local emergency protection
LOCA	Loss of coolant accident
NDT	Non-destructive testing
NSR	Nuclear Safety Regulations for Nuclear Power Plants
MCP	Main circulating pump
MFCC	Multipass forced circulation circuit
NIKIET	Scientific Research and Design Institute for Power Technology
ORM	Operational reactivity margin
PRIZMA	Steady state operational physics program for calculation of reactor parameters
PPDDCS	Physical power density distribution control system
RBMK	Soviet light water cooled graphite moderated reactor
RCPS	Reactor control and protection system
SCSSINP	USSR State Committee for the Supervision of Safety in Industry and Nuclear Power
SKALA	Computerized centralized reactor control system
SPD	Self-powered detectors
VNIIAES	All-Union Scientific Research Institute for Nuclear Power Plant Operation

H	Axial distance
K_v	Volumetric power peaking factor (non-uniformity factor)
K_z	Axial power peaking factor (non-uniformity factor)
N	Neutron power
N_{nom}	Nominal power level
T	Temperature

α_C	Graphite temperature coefficient of reactivity
α_N	Power coefficient of reactivity
α_T	Fuel temperature coefficient of reactivity
α_W	Fast power coefficient of reactivity
α_φ	Void coefficient of reactivity
β	Delayed neutron fraction
β_{eff}	Effective delayed neutron fraction
γ	Coolant density
ρ	Reactivity
ρ_o	Operational reactivity margin
Φ	Thermal neutron flux density

MEMBERS OF THE INTERNATIONAL NUCLEAR SAFETY ADVISORY GROUP

Beninson, D.
Birkhofer, A.
Chatterjee, S.K.
Domaratzki, Z.
Edmondson, B.
González-Gómez, E.
Kouts, H.J.C. *(Chairman)*
Lepecki, W.
Li, Deping
Sato, K.
Sidorenko, V.A.
Tanguy, P.
Vuorinen, A.P.

A. Karbassioun deals with matters relating to INSAG in the IAEA Division of Nuclear Safety.

WORKING GROUP MEMBERS

Vienna, 13–15 November 1991

Abagyan, A.A.	All-Union Research Institute for Nuclear Power Plant Operation, Russian Federation
Brown, R.A.	Ontario Hydro, Canada
Cogné, F.	Commissariat à l'Energie Atomique, France
Guppy, J.	Brookhaven National Laboratory, United States of America
Shteynberg, N.	State Committee for the Supervision of Nuclear Power, Ukrainian Republic
Young, J.D.	Nuclear Electric, United Kingdom

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Vienna, 27–28 July 1992

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Burlakov, E.V.	I.V. Kurchatov Institute of Atomic Energy, Russian Federation
Cherkashov, J.M.	ENTEK, Research and Development for Power Engineering, Russian Federation
Naidenov, M.	Ministry of Atomic Power and Industry, Russian Federation
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