NUCLEAR SAFETY

AN ANALYSIS OF THE PHYSICAL CAUSES OF THE CHERNOBYL ACCIDENT

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Received June 5, 1989
Accepted for Publication December 18, 1989

The initiating events and propagating mechanisms of the Chernobyl accident are the subject of this analysis. The neutronics and thermohydraulics of RBMK reactors under different regimes are studied. It is found that the reactor response to a loss of pumping power was a reactivity trip that could not be fully overcome by the Doppler effect because of the neutronic importance of hydrogen captures under the conditions before the accident. This very high importance was induced by an incorrect hydraulic regime being established before the accident in order to conduct an electromechanical experiment. This experiment was responsible for the loss of pumping power that triggered the accident.

I. INTRODUCTION AND SCOPE

The accident that occurred at Chernobyl Unit 4 on April 26, 1986, is a major case study in the history of nuclear power. It can be considered from very different viewpoints, as it was a rather complex process that involved design shortcomings, operation violations, fast power transients, and large radioactivity releases. In this paper, the physical causes of the accident and the propagating mechanisms inside the reactor core are analyzed.

The initiating event was an experiment conducted under abnormal conditions involving safety violations, but the accident took place and reached catastrophic magnitudes due to the features of the RBMK reactor.

The accident can be studied through the report[1] on the accident provided by the Soviet nuclear authorities as well as the available information about RBMK reactors.[2-8] On the other hand, the principles and data of nuclear reactions, heat transport, and thermal hydraulics are universal, although the configuration of the problems and the boundary conditions are absolutely dependent on the particular reactor under study. Computer codes currently used in Western nuclear technology are not directly applicable to RBMK reactors for design purposes, and some data, such as hydraulics correlations, are specific to each type of reactor. Nevertheless, the macroscopic evolution of an RBMK reactor can be analyzed with tools used to carry out nuclear and thermal-hydraulic studies of lightwater reactors (LWR) and gas-cooled reactors (GCRs).

Under the foregoing assumptions, the objective of this paper is an analysis of the evolution of the physical parameters of Chernobyl Unit 4 during the accident in order to identify the main initiating events and propagating mechanisms. We intended to determine the physical causes of the accident and to compare the RBMK reactor features with those of LWRs and GCRs to demonstrate that this type of accident is not possible in these reactors. We did not intend, however, to carry out a precise numerical simulation of the history of the accident; this is probably impossible because not all the relevant data were recorded and the standard computer codes would have to be greatly modified to deal with the specific characteristics of the RBMK reactor.

In Sec. II, a summary of the RBMK reactor features and a brief description of the accident are presented. This section contains the basic information through which it was possible to carry out numerical calculations and to make an in-depth qualitative analysis. Some authors[9] consider the information provided by Ref. 1 insufficient for a complete characterization of the accident. This is probably correct if a very accurate quantitative numerical simulation of the accident is sought, but the data are sufficient for the type of analysis done in this paper.

A similar objective has been pursued in several studies,[10-24] some of them done at institutions.[10-12] Most of these papers describe and analyze the sequence of events before, during, and after the accident, but the emphasis is on operator violations rather than the physical causes and propagating mechanisms. All these studies are based on information from Ref. 1, although some present slightly different explanations for the different phases of the accident. It must be remembered that not all the significant data were recorded during the accident. The Soviet report presents a picture based on the actual data but elaborated with a neutronic and thermal-hydraulic model.

The importance of the various physical phenomena at each phase of the accident is a subject of discussion. On the other hand, characterization of such significant items as reactivity coefficients[23] or pump cavitation[24] is difficult to achieve due to the very particular conditions during the accident. This is why several papers[25-27] are devoted specifically to the suitable computing methodologies for calculations on the Chernobyl accident.

Section III presents the central part of this work. The evolution of both the reactivity coefficients and the state variables was analyzed so that all the physical mechanisms involved in the accident could be characterized qualitatively and quantitatively, although some unavoidable uncertainties diminished the accuracy of the computations. Besides the neutronics of the accident, which is obviously of fundamental importance, the thermal hydraulics of the plant and its energy balance are studied in depth. This analysis allows identification of the initiating events and first propagating mechanisms. The overall simulation of the power bursts in this work agrees significantly with the data of Ref. 1, although some parts of our analysis point out the importance of some concepts, such as the pressure drop in the primary circuit, that are not explicitly treated in the report. Section III also contains an assessment of additional energy releases that could play a role in the late phase of the accident.

Section IV is an explanation of the physical causes of the accident as a consequence of the former topical analyses. It also contains a brief description of the critical moments and mistakes in the evolution and the treatment of the accident. It also points out how the accident would have been avoided if a suitable thermal-hydraulic regime had been established before starting the experiment. The paramount role of the neutronic importance of hydrogen neutron captures, which is related to the safety margin measured in terms of control rods inserted in the reactor, is shown. This operational safety margin is currently used in RBMK performance control and must not be confused with the scram safety margin (which is provided by control rod banks totally different from those used in operational control). A third significant item is the saturation of the Doppler reactivity effect due to the first power surge, which increased the fuel enthalpy until fuel fragmentation and phase transition occurred.

A comparison between RBMK and Western reactors is presented in Sec. V in order to assess the possibility of such an accident occurring in Western reactors. It is clearly seen that even in the case of slightly overmoderated reactors, similar initiating events will lead to power bursts much lower than those of the Chernobyl accident. Moreover, the standard operational regimes of LWRs and GCRs have a negative feedback that will quench any transient originated by a loss of coolant pumping energy.

II. A SUMMARY OF THE HISTORY OF THE CHERNOBYL ACCIDENT

The official Soviet report[1] describes the time evolution of Chernobyl Unit 4 operation from some hours before the accident until the end of the significant radioactive release. The most important items to be underlined for the physical understanding of the case are the following:

1. An electromechanical experiment was scheduled to test the ability of a tripped generator to maintain the electrical feed to the main circulation pumps (MCPs) for some seconds or minutes during generator coastdown. A power level margin between 700 and 1000 MW(thermal) had been selected for the experiment. However, when this level was being sought on April 25, 1986, the reactor power went down to 30 MW(thermal). After a transient of >2 h, operators stabilized the power level at 200 MW(thermal) and decided to go on with the experiment.

2. The thermal-hydraulic balance of the plant was very different from the standard regime in RBMK reactors. The main circulation flow was extremely high (with respect to the power level), and the feedwater flow was very small, as can be seen in Table I. Pressure along the primary circuit was smaller than usual. At the reactor inlet, it was slightly above the phase transition pressure corresponding to the inlet temperature. At the outlet, steam quality was very small. Unfortunately, there are no direct records of the reactor inlet conditions during the accident. Thus, some of the data in Table I were calculated starting from recorded data in the steam drum separators, as is described in Sec. III.

3. Operators violated some of the operating rules. Reference 1 lists six main violations that played very different roles in the accident evolution. The first violation, defined as "reducing the operational reactivity margin substantially below the permissible value," is poorly explained in Ref. 1, which does not indicate any relation between the violation and the share of hydrogen captures in the reactivity balance. The concept (and the operator violation) was of paramount importance in triggering the accident and allowing it to rise to catastrophic dimensions, as is analyzed in Sec. III.A.2.

TABLE I

Relevant Data of the Thermal-Hydraulic Balance at Nominal Conditions [3200 MW(thermal)] and Before the Accident [200 MW(thermal)]

	200 MW(thermal)	3200 MW(thermal)
MCP flow (kg/s)	11 280	10 444.4
Feedwater flow (kg/s)	45	1 514.4
Pressure in the steam-separators (MPa)	6.53	7.0
Steam mass quality (%)	0.4	14.5
Reactor inlet pressure (MPa)	7.0	7.7
Reactor inlet temperature (K)	552.5	543
Saturation temperature at inlet pressure (K)	557.5	564
Saturation pressure at inlet temperature (MPa)	6.5	5.55

4. Violations 4, 5, and 6 in Ref. 1 are blocking of the reactor protection system. To perform the experiment and to repeat it if necessary, the reactor was to be maintained in operation in spite of the shutdown of both turbogenerators. Blocking the reactor protection system left the reactor unprotected, without the possibility of an automatic scram that would have been triggered immediately after the start of the experiment. The chain reaction would have been stopped before the loss of pumping power that triggered the accident.

The power evolution before the accident is depicted in Fig. 1. At 1 h, 23 min, 04 s (local time) on April 26, 1986, the experiment was started by shutting the emergency regulating valves of the second turbogenerator (the first was already shut down). Four of the eight MCPs were electrically connected to the tripped turbogenerator to perform the experiment. The pumping power began to go down, as can be seen in the reduction of the main circulation flow rate that was recorded and is presented in Fig. 2. This is the origin of the accident. In Fig. 3, the evolution of some reactor variables is given, according to Ref. 1. It must be noted that there are some discrepancies among the hydraulic data of the report. They probably stem from different recording systems that were disturbed during the accident. These discrepancies are of the order of 10% and do not pose any obstacle for the understanding of the accident.

At 1 h, 23 min, 40 s, the scram button was pressed because a very low neutron period was observed, but it was too late to stop the accident. At 01 h, 23 min, 44 s, the first power burst is depicted in Ref. 1, immediately followed by a second and much bigger burst. According to these data, the energy release was >1.0 TJ and the two consecutive reactivity maxima were 2 and 3 $. The reactor was absolutely destroyed by those explosions, as were the surrounding building and most of the ceiling. Exothermic chemical reactions took place immediately afterward and the graphite fire lasted several hours. Radioactive products were explosively released. Although the reactor core was subcritical due to its disassembly, decay heat maintained the remains at very high temperature, enhancing additional radioactive leakage.

The main conclusion of the Soviet report on the causes of the accident is 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 to enhance the power excursion." In Sec. III, the reactivity coefficient and the other reactivity feedback mechanisms are analyzed to determine why and how the accident took place.

III. NUCLEAR AND THERMAL-HYDRAULIC FEATURES OF THE CHERNOBYL ACCIDENT

The physics of the Chernobyl accident embodies many different subjects that deserve at least preliminary attention to define their importance in the accident evolution so that the fundamental mechanisms can be identified and analyzed in depth. This overall study is presented in four parts:

1. the general nuclear and hydraulic characteristics of RBMK reactors both at nominal regime and at the conditions during the accident
2. the reactivity feedback mechanisms, such as Doppler effect, void effect, and so forth
3. the external mechanisms, i.e., operating actions and experiment consequences
4. phenomena that could occur in a late phase of the accident and could convey huge energy surges.

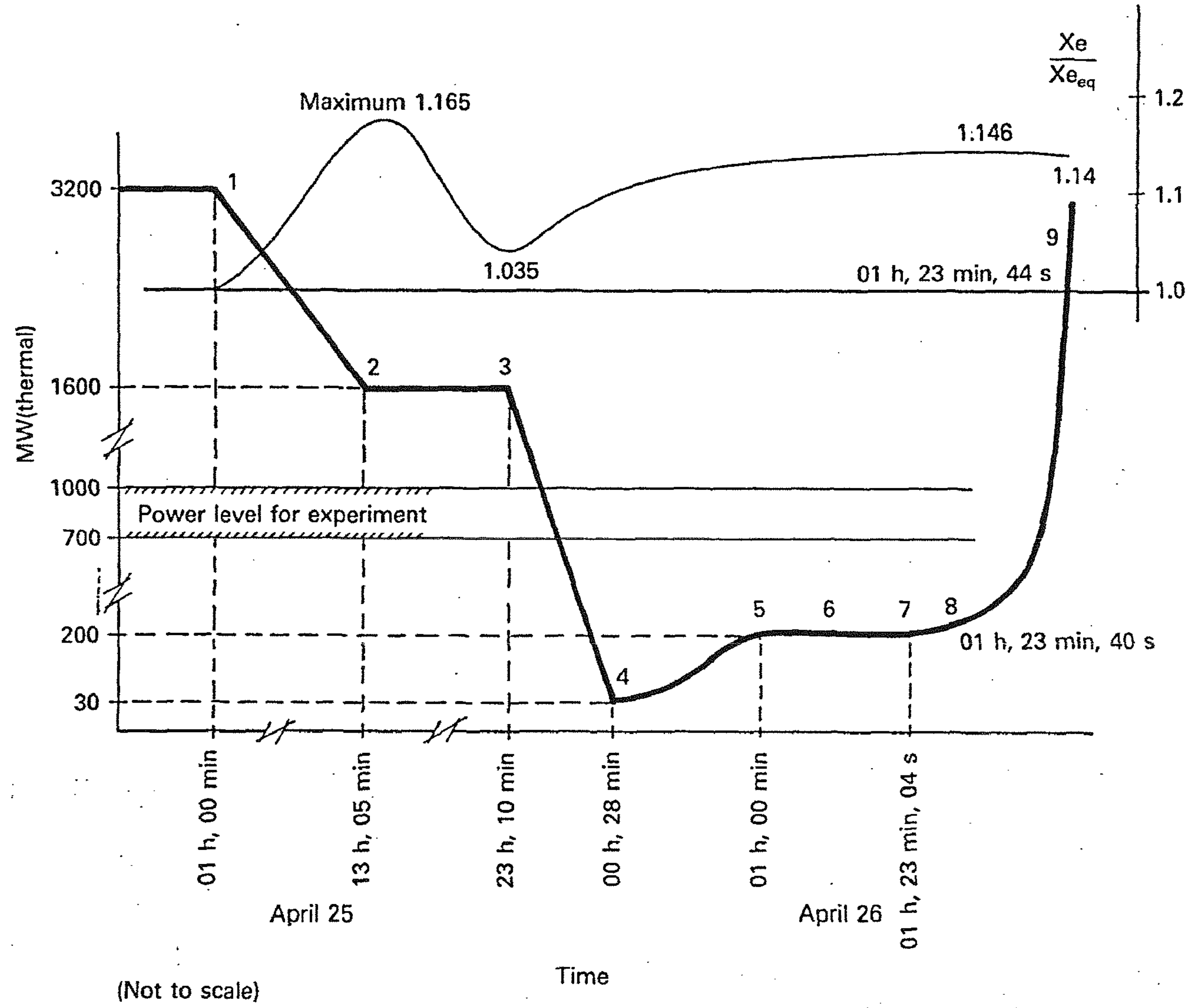

Fig. 1. Power and xenon concentration evolution before the accident.

III.A. RBMK Neutronic and Thermohydraulic Features Relevant to the Accident

Most of our neutronic calculations were carried out with a multicell representation of RBMK configurations using the WIMS-D code,[28] which has been used by other authors[26] for similar purposes. Space effects in both the vertical and horizontal directions were studied with an LWR simulator[29] in the first stages of the accident. Although RBMK reactor cores are very large (~10-m characteristic length), the neutronic migration area is also very large (~400 cm^2), which means that the core is not as large in terms of neutronic coupling. At the beginning of the accident, the output steam quality was very small. Water was practically liquid all along the reactor channels and therefore the reactor was rather homogeneous, neutronically speaking, even though the xenon concentration induced a double-hump shape vertically.[1]

Another important feature of RBMK reactors deserves some comments, although it probably played no significant role in the accident: RBMK reactors are hydraulically divided into two halves. There are two primary circuits, each connected to specific pumps, drum separators, and fuel channels. Recordings from the accident (see Fig. 2) show that both halves underwent similar evolutions, which leads us to consider that the accident was rather symmetric. It does not mean that all the channels suffered similar evolutions, because the power distribution was not perfectly uniform and the central channels could have a faster response to the transient induced by the experiment.

In any case, all the information and considerations from Ref. 1 led us to analyze mainly the reactivity

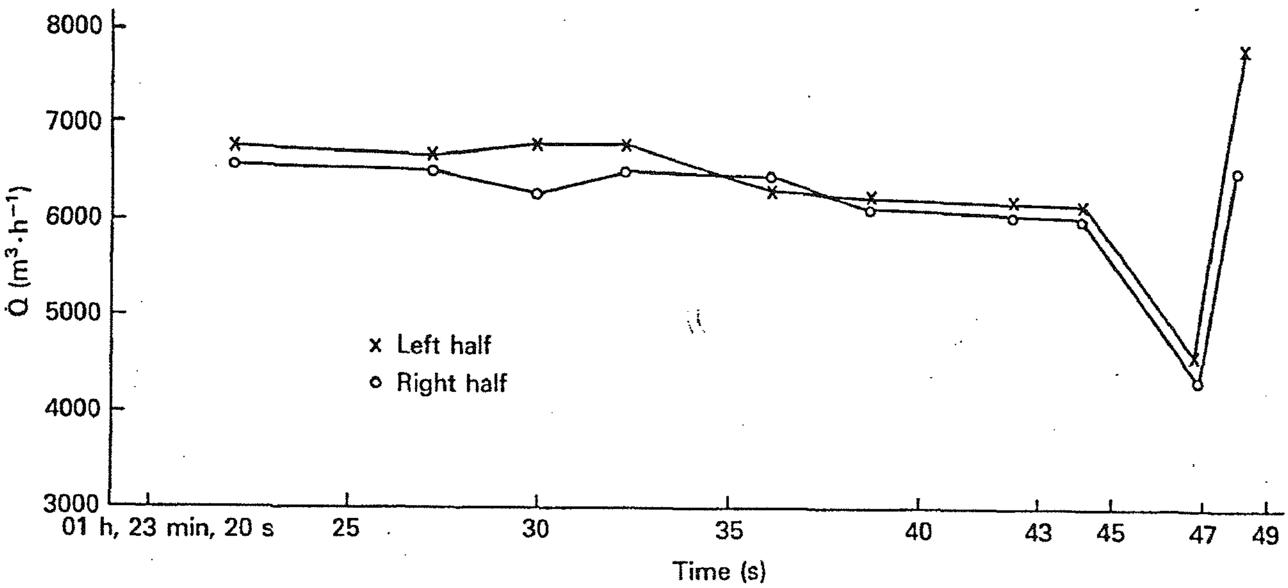

Fig. 2. Main circulation flow rate per pump through both reactor halves.[1]

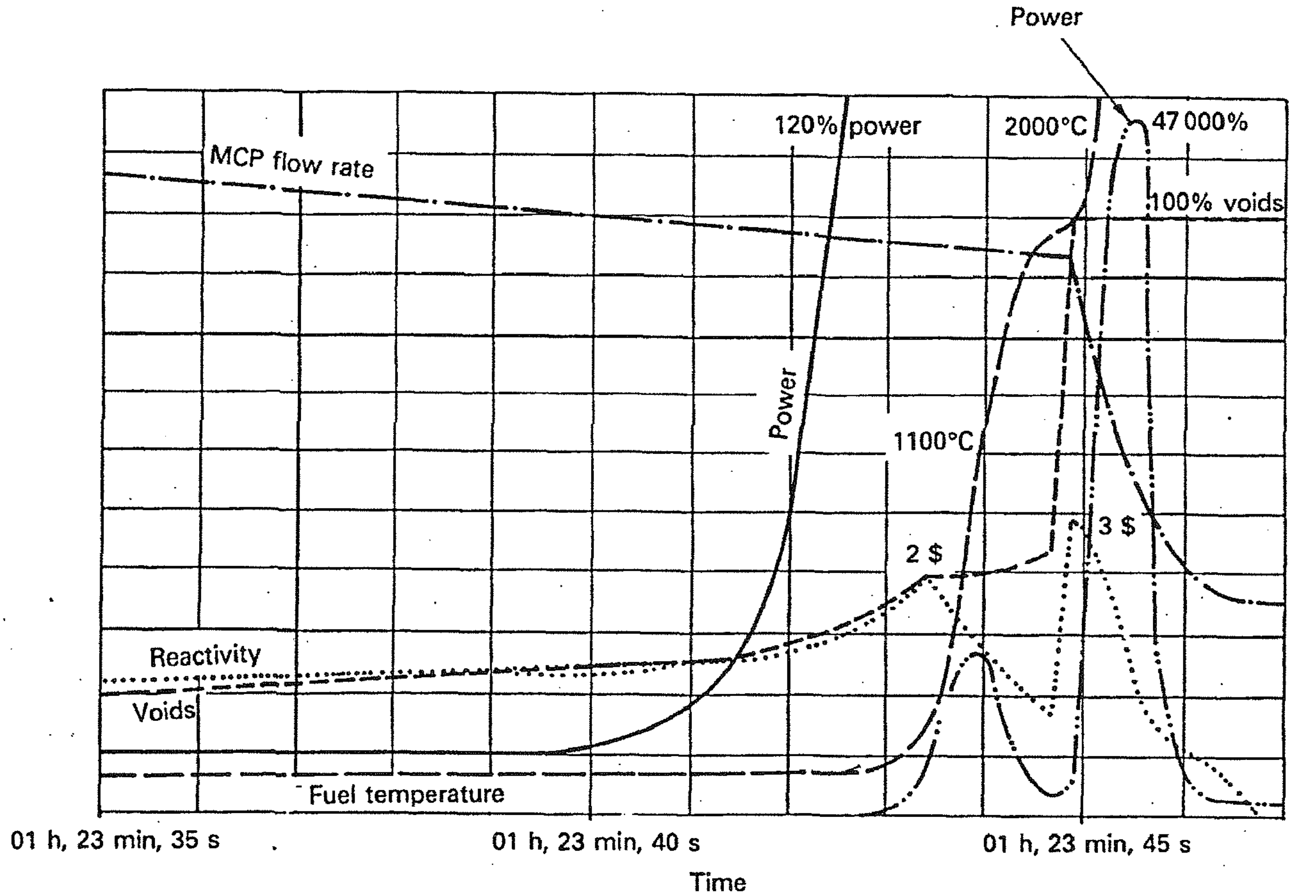

Fig. 3. Evolution of the most significant variables during the accident[1]: reactivity, −1% (lowest line) to +5% (highest line); power (solid line), 0 to 120% nominal; power (dotted line), 0 to 480 times nominal; fuel temperature, 473 to 2273 K; voids, 0 to 120%; MPC flow rate, 4 to 16 m^3/s.

feedback mechanisms in a point reactor approach. Some authors[27] have pointed out that partial voidages of the reactor (particularly in the central part) present higher void reactivity coefficients than homogenous voidages, similar to sodium voidages in a liquid-metal fast breeder reactor (LMFBR). A comparison between reactivity coefficients calculated through multicell approximations and two-dimensional simulations showed that space-dependent effects were of secondary order in relation to the bulk effects studied in the point reactor approach. On the other hand, detailed calculations of spatial nonuniformities would require the development of specific codes to deal with the particular hydraulic regime of RBMK channels. Reference 1 contains some data on two-phase flow correlations, but they are fitted for steady-state conditions and slow transients. Moreover, there are no recorded data of the initial conditions in each channel, which makes a very detailed calculation impossible. Such a calculation would probably not be very significant for our accident analysis.

The vertical evolution of the accident was undoubtedly very important, as the flow transient produced by the experiment moved upward from the reactor inlet. This evolution was studied with the help of three codes (neutronic and thermal hydraulic) plus a simulation of the reactivity feedback. TIMEX, a one-dimensional, time-dependent neutron transport code[30]; COBRA-IV, a thermohydraulic code for rod bundle reactors[31]; and GAPCON-THERMAL, a program to analyze heat transport in a rod plus a channel,[32] were used for this purpose, but the results were of limited significance due to the difficulties of coupling them with the appropriate feedback and in simulating the dynamics of an RBMK channel.

To study the point kinetics evolution of the accident, a time-dependent program (fully implicit in the time variable calculation) was written. Reactivity feedbacks from fuel temperature and water density were embodied. GAPCON-THERMAL was used to calculate the temperature evolution and heat transfer in a channel characterizing the total reactor. Reactivity coefficients were previously computed for many different states, taking into account fuel temperature, graphite temperature, void fraction, xenon concentration, and fuel burnup. The complete series of reactivity coefficients is reported in Refs. 33 and 34.

The main findings of this broad parametric survey are presented in this section when the specific reactivity coefficients are treated. As a general conclusion, it can be said that they show a behavior quite similar to that reported in the Soviet literature,[1-4] although some quantitative discrepancies appear. The main sources of discrepancy are the assumed values of the state variables. For instance, graphite temperature significantly affects all the results, especially those related to burnup. Fuel composition at high burnup levels in our calculation (873 K of effective thermalization temperature) was slightly less reactive than the composition given in Ref. 1. Instead of 2.6 g/kg of ^{239}Pu, we obtained 2.4 at 20 MW·d/kg, but this discrepancy was very sensitive to the graphite fuel temperature used in our calculation, which can be easily explained by the cross sections in the upper thermal and epithermal regions of ^{238}U and ^{239}Pu.

III.A.1. Thermal-Hydraulic Balance of the Plant

The thermal-hydraulic analysis of the accident required accounting for the energy and flow balances of the plant. Figure 4 shows a simplified scheme of the plant (the actual number of each type of element does not appear). The numbers indicate the main points of the circuit used to calculate the thermohydraulic evolution of the plant. Both at nominal power and at preaccident conditions, the balance was easily established, although some data had to be estimated because neither Ref. 1 nor other Soviet literature contained complete information.

The key point in this hydraulic scheme, according to our study, is the reactor inlet. While there are recorded data for the drum separators, there are none for the reactor lower zone, where the pumping transient was first felt.

It has already been pointed out that a very strange hydraulic balance was established just before starting the experiment, as can be seen in Table I. The main circulation flow rate was even higher than the nominal value, although power was just 7% of the nominal level. Reference 1 underlines that the outlet steam quality was very poor, but practically nothing is said about conditions at the inlet. To assess the safety of the

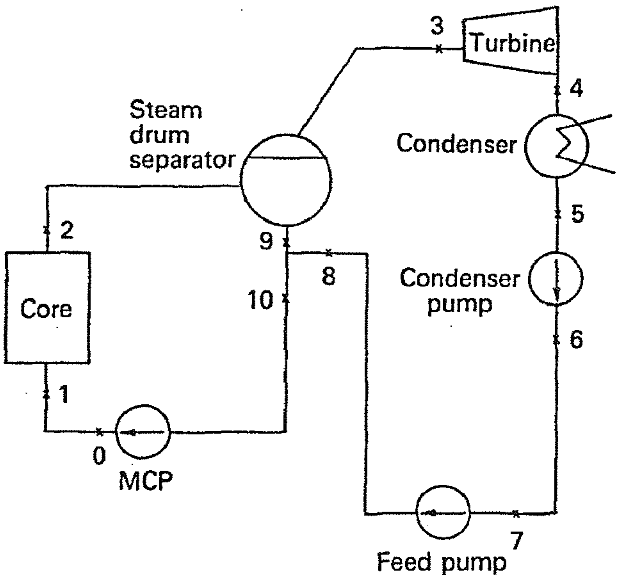

Fig. 4. Hydraulic scheme of the plant.

preaccident balance, we defined two figures of merit: the pressure excess over saturation pressure and the inlet subcooling defined as

$$IS = \frac{H_1 - H_1^0}{1333.3} , \quad (1)$$

where

H_1 = actual specific enthalpy

H_1^0 = liquid at saturation under the actual pressure P_1.

The figure 1333.3 is the specific enthalpy of the phase transition in joules per gram.

According to our data, IS at nominal conditions is −8.3% but it was only −2% when the experiment started. This means the water was too close to the phase transition to steam in the balance established before the accident. The overpressure at nominal conditions was ~2 MPa, but it was only 0.5 MPa when the experiment started. Both data indicate that the reactor inlet was especially sensitive to any change in flow variables and, particularly, pressure.

Figure 5 shows the flow rate versus pressure head of the total set of MCPs in an RBMK reactor. This curve was drawn from data taken from the Soviet literature.[1-3] A second (lower) curve is included to characterize the effect of the pumping power loss corresponding to a steady-state situation with only four pumps working, instead of eight. As the pumps were placed in parallel, the maximum pressure head remains constant in this reduction, unlike the maximum flow rate, which goes down to 50%. A third curve of the type $[h \propto Q^2]$ is plotted to show the pressure loss along the primary circuit in preaccident conditions. The pump characteristic curves cross this curve, indicating the working points for the cases of four or eight acting pumps, respectively. The lower point would be real in the case of steady-state operation with four pumps and a similar hydraulic behavior. It is clear that the pumping transient produced by the experiment did not go from the upper to the lower point just along the $[h \propto Q^2]$ curve. The increase of steam quality (water voids) due to the power rise would increase, as would the pressure loss along the primary circuit; therefore, the $[h \propto Q^2]$ curve would shift to an upper one. Figure 5 also shows a curve corresponding to the nominal regime that reaches 14.5% of steam quality at the outlet. It is difficult to accurately determine the evolution of the pressure head and flow rate as the pumping power went down and the reactor power went up, but it would not be very different from a straight pathway from point 1 to point 2.

The main conclusion of our hydraulic analysis is that the loss of pumping power induced by the experiment caused a rapid fall in the pressure, enhancing the water boiling from the very bottom of the reactor. Figure 5 shows a final pressure drop of ~0.5 MPa in the pump outlet pressure as a consequence of losing 50% of the pumping power, which is equal to the pressure excess over the saturation pressure at the reactor inlet (calculated to be 0.5 MPa in the preaccident balance).

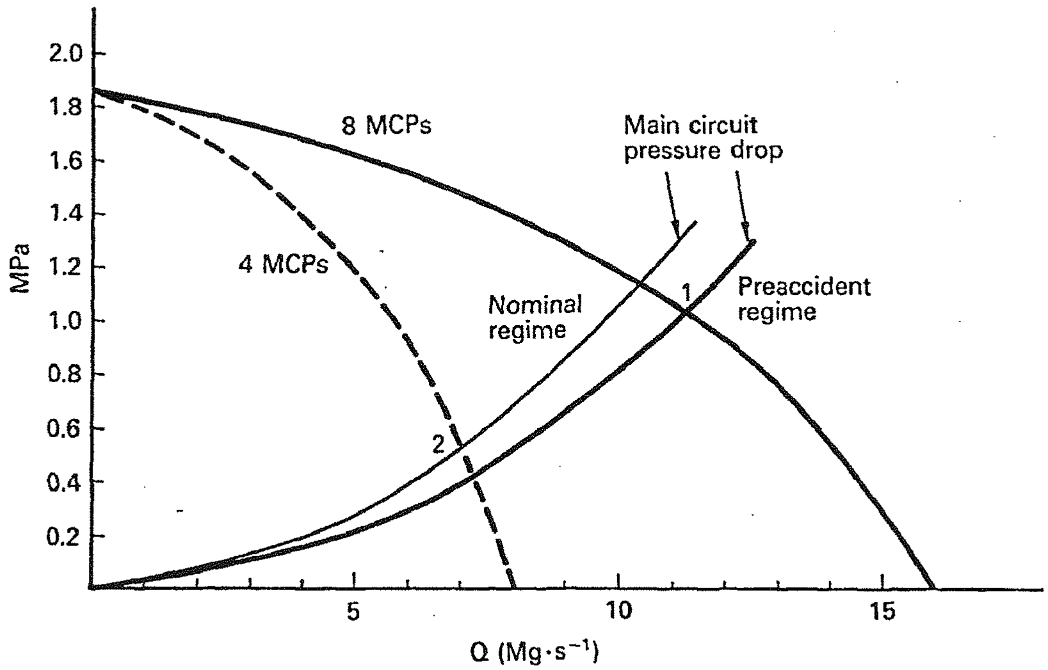

Fig. 5. MCP pumping performance and hydraulic regimes of the primary circuit.

Therefore, water could begin to boil from the reactor inlet itself.

This was the main thermal-hydraulic consequence of the experiment. The reactivity feedback through the water void effect launched a very powerful reactivity trip that was responsible for the first power burst. In Sec. III.B, the void reactivity coefficient is treated, as well as the Doppler effect that was able to stop this first explosion.

III.A.2. Neutronics of the Accident

Reference 1 declares that before the accident the reactor was operating with an excess of reactivity of "6–8 rods, in other words, not more than half of the minimum permissible value laid down in the operating regulations." This is the first of the operator violations listed in the report. Before the accident, the operating regulations established that a minimum of 30 rods had to be inserted in the reactor during operation. Unfortunately, this operational reactivity margin is poorly explained in the report. First of all, it must not be confused with the scram reserve margin, i.e., the number of rods out of the reactors that are available to stop the chain reaction if necessary. The operational reactivity margin is an indirect but integral indicator of the neutronic balance inside the reactor. To understand it properly, it is necessary to bear in mind that in RBMK reactors there are two movable neutron absorbers: water and control rods. Any change in water density is a change in the neutron absorption rate. Water is highly important for thermal neutrons because it lies between the moderator and the fuel. Thermal neutrons diffusing from the former to the latter must go across the coolant, where hydrogen has the advantage over uranium in absorbing those neutrons.

When a small number of rods are required to maintain criticality, there is a high capture rate in water. Of course, there are other variables, such as burnup degree and fuel and graphite temperatures, that affect reactivity, but water and control rods absorption are the final methods to meet the equilibrium between neutron productions and losses.

Just before the accident, the void fraction in the core was very small, i.e., the water density was very high. A correspondingly high absorption rate in hydrogen induced the control rod banks to move out. In this situation, the reactivity worth of the coolant was very high. In Sec. III.B.2, the void reactivity coefficient is analyzed, but it can be said in advance that the reactivity worth could be estimated at ~5 \$.

A sudden boiling of the coolant produced by depressurization and flow rate reduction triggered a very strong reactivity trip that was initially overcome by the fuel temperature effect. However, the Doppler antireactivity margin was lower than the positive void effect and the fuel reached the enthalpy of disaggregation just being able to quench the first reactivity trip. A few tenths of a second after the first power burst, the bulk of the energy (initially deposited in the fuel) was transferred to the water in a very fast, nonreversible process very similar to a steam explosion. The heat transmission rate from the fuel to the coolant was so high that convective streams could not develop within the water. The steam film and bubbles produced in the clad surface grew and expanded much faster than the boiling of the bulk of the water. The internal pressure of the bubbles increased so rapidly that the water was suddenly expelled from the reactor. The "dried-up" reactor was much more reactive than the wet one, and a second reactivity trip occurred. At that moment, the fuel was practically at phase transition or partially disaggregated. In any case, the Doppler effect could do nothing; this second power burst was stopped only by the very destruction of the reactor.

It is important to take into account some integral figures about the accident evolution. According to Ref. 1, the first power burst yielded 200 GJ of energy. A similar figure was calculated in this work using the point kinetic approach with feedback (including the program GAPCON-THERMAL to calculate the thermodynamic evolution of a characteristic fuel channel). This is the reference energy level to be compared with the three main physical consequences of the accident:

1. *Fuel heating and fragmentation:* The total energy required to reach the phase transition point for all the fuel in the reactor (205.7 Mg) was 250 GJ (assuming 1.2 $kJ \cdot g^{-1}$ of specific enthalpy, which is probably overestimated because it neglects the initial specific enthalpy of 75 $J \cdot g^{-1}$ and does not take into account possible shock waves and nonreversible mechanisms in the fuel). On the other hand, reactivity feedback does not depend on the bulk temperature but on the effective temperature of ^{238}U resonant captures, which was much lower. In any case, it is clear that the Doppler effect was practically saturated in the sense that an increase in energy would produce a phase transition, but no significant additional heating.

2. *Water boiling:* Under the accident conditions, 40 Mg of water needs 50 GJ to boil from saturated liquid. Thus, boiling all the water in the reactor would not be enough to remove the energy deposited in the fuel during the first burst. However, heat transmission at that moment was a nonreversible thermodynamic process and did not include the boiling of the bulk of water.

3. *Water expulsion from the reactor:* The steam bubbles and films around the clads acted as pressure pistons, pumping out the liquid water against the drum separators and the rest of the circuit. Our calculations indicate that this process could remove as little as 0.4 GJ from the reactor energy. The total energy removed by the water after the first power surge is very difficult to determine, but it must lie between 0.4 and 50 GJ,

possibly closer to the former. This means it was an order of magnitude lower than the burst energy, or even lower. Hence, the fuel was not cooled significantly by the thermohydraulic reaction of the coolant. This is why the Doppler effect was saturated (ineffective) in preventing the second reactivity trip.

It is obvious that the operators did not understand the physics of RBMK reactors and disregarded the danger of an incorrect hydraulic balance that could produce such a high reactivity worth of the coolant. The meaning of too few control rods inserted in the reactor was not clear to them. It is curious, although merely anecdotal, that many people hearing the official Soviet explanation[1] for the first time thought that the first operator violation was not a violation at all, because in Western reactor technology there is no such limit: The scram reserve margin is expressed as a minimum number of rods out of the reactor, not in. However, the overmoderated condition of the RBMK reactor or, more precisely, the special role of water as an absorber requires a *minimum number of operational control rods to be inserted* in the reactor. Otherwise, it becomes controlled by a vaporizable material (water) that can be removed suddenly by thermohydraulic forces.

III.B. Reactivity Coefficients

Fuel temperature, water density, graphite (thermalization) temperature, and xenon poisoning (prompt burnup evolution) have a significant influence on the neutronics of an RBMK. The standard way to characterize this influence is through reactivity coefficients. In our case, these coefficients were calculated as derivatives of the series of values of k_{eff} in a broad parametric calculation carried out both in a zero-dimensional (multicell) representation and a nodal simulation of the reactor. The following independent variables were used as parameters:

1. burnup degree (between 10 and 14 MWd/kg)
2. number of control rods inserted in the reactor (the multicell representation was varied to allocate different ratios of fuel channels to control rods)
3. xenon concentration
4. graphite temperature
5. void fractions
6. fuel temperature.

The parametric survey was stopped when very low or very high k_{eff} values were found. For instance, a very large number of control rods in conveyed very low k_{eff} values for the burnups studied. However, this region was not of interest because the reactor at the accident had <30 control rods. Besides k_{eff}, other significant items such as migration area and the effective percentage of delayed neutrons were calculated, even though their dependence on some of the independent variables was in some cases negligible. It is worth quoting that a β_{eff} value of 0.55% was found for 10 MWd/kg, slightly decreasing with higher burnups.

III.B.1. Doppler Reactivity Coefficient

This coefficient was always negative, but slightly dependent on the fuel temperature. The rest of the independent variables had much less influence. Coefficients carried out with the nodal approximation were somewhat less negative than those obtained with the multicell approach, and closer to the value given in Ref. 1, −1.2 pcm/K. Our calculations show a value of −1.65 pcm/K at 573 K, but it goes down to −1.25 pcm at 1173 K and to −0.95 pcm at 1973 K. Assuming a temperature jump from 573 to 2773 K, the corresponding reactivity insertion was −2400 pcm, that is, −4.4 $. As the effective temperature of resonant absorption would be lower than the phase transition, the total worth of the Doppler effect during the accident was bound to the last figure and it was probably about −3.5 $ (corresponding to 1973 K of effective temperature).

III.B.2. Void Coefficient

It was well known much before the accident that this coefficient is negative for clean reactors [at beginning of life (BOL)] and becomes positive at high burnups. The reason for that lies in the neutronic importance of the hydrogen captures. An indirect measurement of this importance is the number of control rods inserted in the reactor, which must be larger than a specified number to guarantee reactor stability.

It is worth remembering that steam density at reactor conditions was only 4.5% of the liquid density. Hence, any change in the void fraction represented a significant reduction of the water density and therefore of the hydrogen neutron captures.

The coefficient values obtained with the nodal representation are slightly larger than those of the multicell approximation. Unlike the Doppler coefficient, the void coefficient depends on many independent variables, such as burnup, number of control rods, void fraction, and fuel temperature.

In Fig. 6, some characteristic curves of this coefficient are depicted. Note that Ref. 1 gives a value of 20 pcm/%, which is very close to the result obtained in calculations with the nodal representation. The coefficient increases for high void fractions, mainly at high fuel temperatures. It has been estimated that the total reactivity jump produced by a complete water voidage is >5 $. This is higher than the total antireactivity that could be provided by the Doppler effect. Therefore, any significant distortion of the equilibrium achieved just before the accident would lead to a final

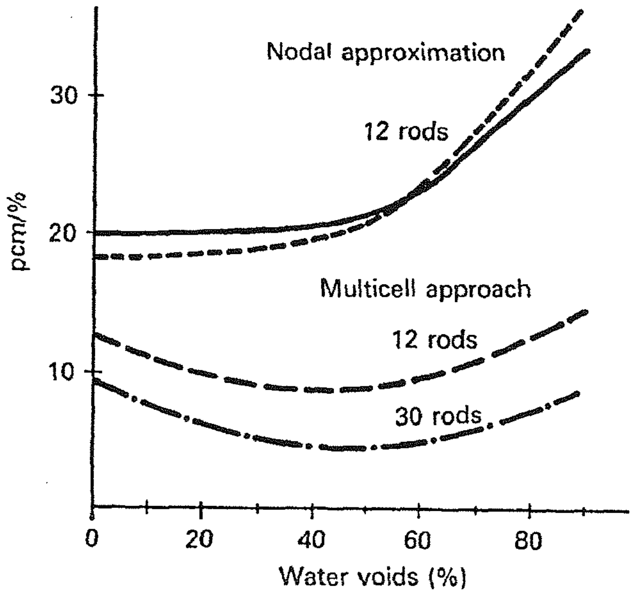

Fig. 6. Reactivity coefficient of water voids, depending on the number of control rods inserted in the reactor: burnup = 12 MWd/kg; fuel temperature = 573 K (300°C), except for the dashed upper line (nodal approximation) where it is 1173 K (900°C).

supercritical state, if we consider the thermodynamic evolution of the state variables. In other words, the reactor had lost its internal capability for self-stabilization. An immediate reaction through the Doppler effect could initially stop the excursion toward a *hot and dry* reactor, but it was unable to stop it definitely.

The Ref. 1 reactivity coefficients were calculated under the assumption of homogeneous water density in the entire reactor. Our calculation could not study the effect on individual channel voidages. However, partial voidages in the vertical direction were simulated in the nodal representation, and it was found that some types of voidages (particularly those affecting the central, lower part) could represent higher reactivity worths. A similar result has been reported by others[18,20] and a parallel can be established between this effect and that of sodium voidages in LMFBRs (Ref. 35), where the central voidages are much more reactive than the total voidages. Nevertheless, the existence of so many water channels (>1600) does not allow a detailed calculation of this effect. On the other hand, the coolant conditions, very close to saturation, represent a fair homogeneity of the water density at the beginning of the accident. In the second phase, related to the sudden expulsion of the water after the first burst, the density was presumably also very homogeneous, because the entire reactor became dry.

As a summary, a short explanation of the physics of the void coefficient can be added. This coefficient is negative at BOL and becomes positive for poisoned (less reactive) cores. This is because the value of the coefficient is the result of two opposite neutronic effects of the cooling water. On the one hand, it improves the thermalization of the neutrons, as it is colder than the graphite and closer to the fuel. On the other hand, it absorbs thermal neutrons. In the case of a very reactive core (with a large number of control rods in), the first effect is dominant. Hence, the absence of water reduces the ability to multiply neutrons by fission, because this reaction is mainly dominant with well-thermalized neutrons. This is why the void coefficient is negative in those cases.

In less reactive cores, the percentage of total neutrons absorbed in water is higher, and the second effect dominates. That is, water acts more as a neutron shielding around the fuel than as an additional moderator. In this case, the void coefficient becomes positive, which was the situation in the accident.

III.B.3. Graphite Temperature Coefficient

This coefficient is negative at BOL, but it increases with the plutonium concentration, due to the fission resonance of ^{239}Pu at upper thermal levels. At 10 MWd/kg, the coefficient is ~5 pcm/K and it goes up to 5.5 pcm at 12 MWd/kg. This very high figure must be considered from the perspective of the reactivity feedback.

Somewhat more than 5% of the energy released is deposited in the graphite, mainly by neutrons and gamma. Because of the Wigner effect and chemical reactivity, graphite is cooled by a helium-nitrogen independent system that transfers heat to the water and maintains a steady temperature level in the graphite.

During a sudden power increase, graphite temperature tends to rise, but at a moderate rate because of the large graphite thermal inertia. Taking into account only the graphite blocks in the core, there is 1700 Mg of graphite with a specific heat of 1.7 J/g·K. A 1 K increase in temperature requires 2.8 GJ; therefore, the energy deposition in the fuel would be ~50 GJ, which would produce a fuel temperature increase of 850°C in round numbers.

A relevant parameter to compare the efficiency of feedback mechanisms through fuel and graphite is the product of the reactivity coefficient times the rate of temperature rise. This figure is ~200 for the fuel and 1 for the graphite, which clearly indicates the secondary role of the graphite during the power surges. Another item is the exact graphite temperature during the accident, taking into account the prior power history. If the graphite had become colder during the 2 h before the accident at 30 MW, there would be a different starting point for the accident. Such graphite cooling could introduce a negative reactivity insertion with the immediate consequence of a control rod extraction; i.e., it could induce a worse reactor state.

III.B.4. Fuel Fragmentation

An optimum moderation configuration can be found if a fuel rupture occurs. Fuel disaggregation requires very high energy yields, of the order of or higher than 250 GJ, but it can happen also at a local level with lower energy yields. In any case, this mechanism is a sequel of a first power burst and can produce a second prompt supercritical burst.

The thermomechanical processes conveyed in fuel disaggregation include thermal shock waves, cladding rupture, and formation of fuel fine particles, which is very difficult to estimate qualitatively and to model for quantitative calculations, as is seen in Sec. III.D.1

III.B.5. Xenon Poisoning

This phenomenon played a significant role in setting the initial conditions of the accident, but did not contribute to the accident evolution itself, which was very fast. During the power surges, the xenon concentration decreased slightly (as a consequence of the higher fluxes) by ~0.5% of the equilibrium concentration, which represents 13 pcm of reactivity insertion, much less than the feedback from fuel temperature and water voidage.

III.C. The Initiating Events

The thermal-hydraulic conditions of the reactor just before the accident are described in Sec. III.A.1. The reactivity feedback in this type of reactor has been analyzed, and it has been found that any reduction in the water density would represent an important reactivity insertion. Such a reduction took place as a consequence of the experiment. There is no direct evidence of a pressure drop at the reactor inlet as the pumping power went down, but it can be deduced from the coastdown of the flow rate. Both parameters are related to each other in Fig. 5, according to our estimates elaborated from data from Ref. 1. Figures 2 and 3 from that report depict the evolution of the main circulation flow rate. The coastdown is not very steep, but the parallel effect of pressure reduction and the associated pump cavitation[36] must be taken into account. The water boiling was not due to an increase in power, which would have conveyed a rise in the fuel temperature and a negative reactivity feedback through the Doppler effect. The boiling was created by the loss of pumping power, and this is the starting point of the accident.

Figure 7 shows the evolution of reactivity and power according to Ref. 1 and as a case calculated by the point kinetic approach. Our results depend significantly on the lumped parameters chosen to characterize the reactivity feedback, such as the reactivity coefficients and the β_{eff} value. The parameters used in the case reported in Fig. 7 correspond to 12 MWd/kg, 873 K of thermalization temperature, 553 K of initial fuel temperature, and 0.91 $g \cdot cm^{-3}$ of initial water density. In spite of the dependence on those parameters, all the cases show the same behavior, with certain discrepancies in the numerical results: a first power surge followed immediately by a second and much bigger burst. The time delay between the power surges was critically dependent on the thermal diffusivity of the fuel-gap-clad-water system, which was calculated with GAPCON-THERMAL (Ref. 32). In Table II, the thermal data of the first power ramp are shown. It can be seen that energy is initially stored in the fuel and is later transferred to the coolant. Presumably, the temperature profile became so steep that a thermal wave developed in the clad/water (steam) interface. The situation was so nonreversible that it could not be simulated with the available codes, as discussed in Sec. III.A.2.

Another mechanism is the "positive scram." According to some authors,[20-24] this played a role in triggering the accident. A positive scram would have been induced by the removal of water from the control rod channel when the scram was switched on. Control rod followers that are too short could give an initially positive reactivity worth to the scram control rods. According to our calculations, the removal of water from the control rod channels would not have a measurable positive reactivity feedback through the void coefficient. The positive void coefficient is related only to the water of the fuel channels and is due to the high neutronic importance of the water location, just between the moderator and the fuel. The higher thermal flux seen by thermal captures in hydrogen with respect to fuel absorption is of paramount importance in making the void coefficient positive. The water in the control rod channels has much less neutronic importance. Due to the lack of reliable information about the follower design and the difficulties in simulating it with the nodal computer code used in our calculations, an accurate numerical assessment of the antireactivity rate introduced by a scram has not been made. Nevertheless, the map of the neutronic importance (adjoint function) in a simplified core representation shows clearly that the positive void coefficient is mainly produced by the fuel channel water.

The very slow movement of the control rods in RBMK reactors is also worth mentioning. It takes 10 s for the total introduction of the scram rods into the reactor. As the scram switch was turned on at 1 h, 23 min, 40 s, it is clear that only a fraction of the rods had entered when the first explosion began 2 s later. The follower movement produces a substitution of water by graphite in the lower part of the control channels of the reactor just in that period, but this effect occurs in every RBMK scram and it has never been reported as a significant reactivity transient due to the positive scram effect, in spite of all the scrams produced in RBMK reactors under different conditions

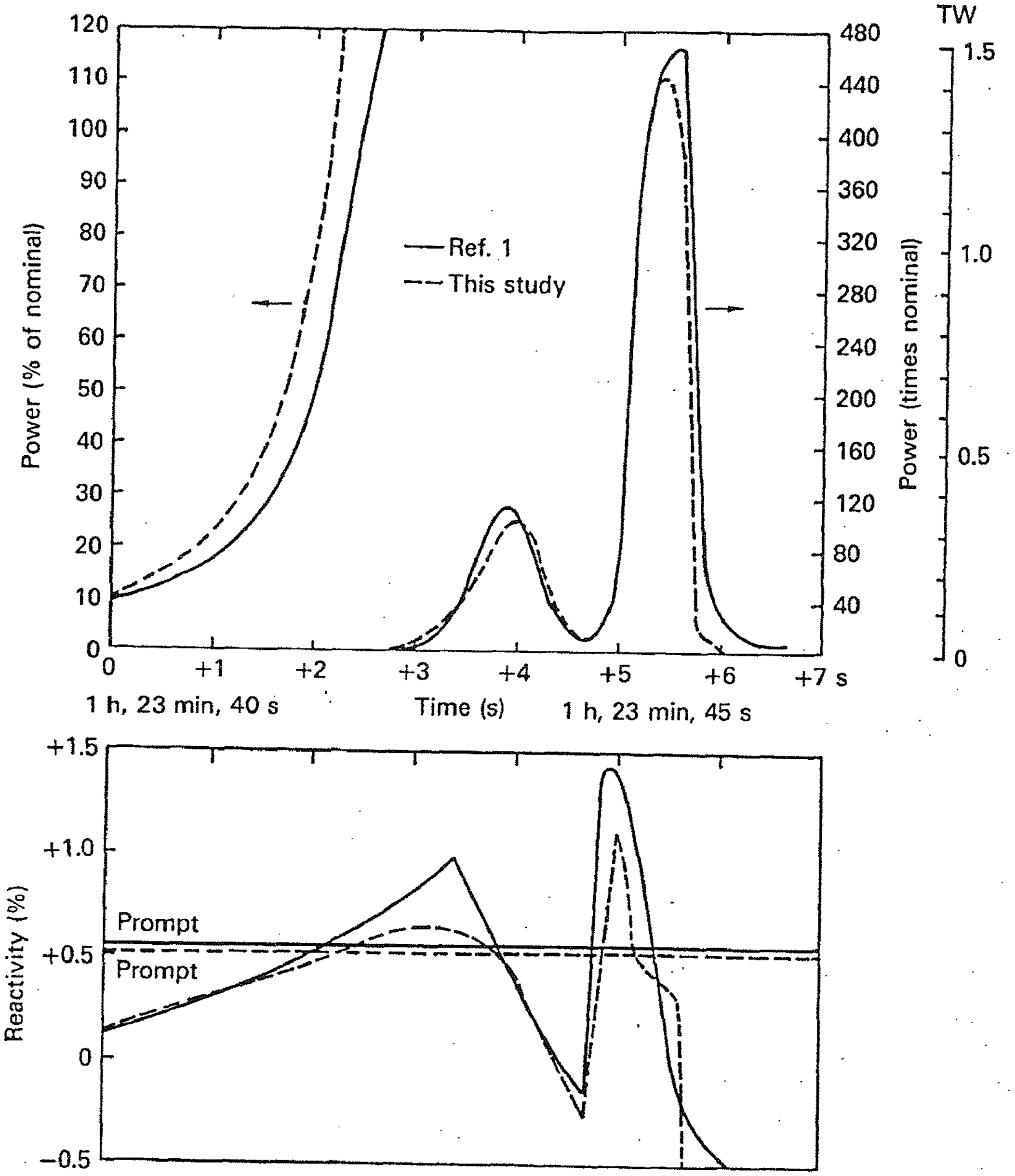

Fig. 7. Evolution of power and reactivity along the accident.

and flux shapes. In terms of reactor physics, the substitution of water by graphite is of secondary importance as compared with the water removal from fuel channels, which was the real origin of the very low reactor period that forced the operators to switch on the scram. Moreover, the operational control rods that were inside the reactor at that moment had been moving in as the water voids produced by the pumping power loss were inducing a reactivity insertion. As the speed of the control rods was very small and the scram was switched on too late to be effective, control rods of all kinds played a rather marginal role in the development of the accident.

It can be concluded that the accident-initiating mechanism was the loss of pumping power produced by the experiments. Both the pressure drop and the flow rate coastdown unbalanced the thermal hydraulics of the coolant and increased boiling. The pump transient is not perfectly known from actual data recorded during the accident, but the pressure drop could have induced cavitation and two-phase flow in the pumps. Even in the case of a much smoother transient, as suggested by Fig. 2, the thermohydraulic unbalance was important enough to trigger the accident through the positive void coefficient.

This effect has already been studied, tested, and

TABLE II

First Power Transient Evolution

Time[a] (s)	Total Power (MW)	Water Power (%)	Total Energy (MJ)	Water Energy (%)	Fuel Energy (%)	Fuel Temperature (K)
0.00	0.2000E+03[b]	100.00	0.0000E+00	100.00	0.00	554.0
1.00	0.2240E+03	90.62	0.2120E+03	95.04	4.96	554.1
2.00	0.3200E+03	68.88	0.4840E+03	85.37	12.86	555.0
3.00	0.5760E+03	48.08	0.9320E+03	71.02	29.58	557.9
3.50	0.8640E+03	38.04	0.1292E+04	62.95	37.69	560.7
4.00	0.1600E+04	27.03	0.1908E+04	52.60	48.60	566.6
4.50	0.3520E+04	19.66	0.3156E+04	40.24	60.38	579.5
4.85	0.6400E+04	16.98	0.4772E+04	33.04	67.10	596.6
5.00	0.1280E+05	11.18	0.6212E+04	28.42	72.02	613.3
5.25	0.5760E+05	5.71	0.1421E+05	16.22	83.89	709.3
5.35	0.8320E+05	5.97	0.2125E+05	12.77	86.96	792.1
5.50	0.1600E+06	6.15	0.3917E+05	9.63	89.87	997.9
5.60	0.2368E+06	7.03	0.5901E+05	8.59	90.89	1216.6
5.65	0.2752E+06	7.99	0.7181E+05	8.40	91.01	1353.2
5.68	0.2906E+06	8.93	0.8030E+05	8.40	91.19	1440.8
5.71	0.3059E+06	9.91	0.8924E+05	8.51	93.96	1524.9

[a]Beginning at 01 h, 23 min, 38 s.
[b]Read as 0.2000×10^3.

published in the Soviet literature. In Fig. 1 of Ref. 3, the neutronic transient produced by a loss of pumping power in the RBMK-type reactors of Beloyarsk is depicted. It clearly agrees with the explanation of the roots of the accident given here.

III.D. Additional Exothermic Mechanisms

The strong energy yield of the accident destroyed the reactor configuration and drastically changed the chemical conditions of its components. Such significant modifications could induce additional mechanisms, contributing to increase the total energy release. In the aftermath of any accident, one of the critical issues is the decay heat cooling. In the Chernobyl accident, it took several days[1] to set up a suitable cooling system to reduce the reactor temperature to almost environmental levels and help stop the radioactive emissions. Other potential causes of energy release directly related to the material properties modifications are analyzed in the following.

III.D.1. Fuel Fragmentation

The energy yields of the first and second power surges are estimated to be 0.2 and 1.0 TJ, which are greater than the total energy required to disaggregate all the fuel in the reactor [0.25 TJ (assuming 1.2 $kJ \cdot g^{-1}$)]. This means that the fuel was severely damaged. Considering that each power surge lasted ~1 s and the power jumped several orders of magnitude in that interval, the thermodynamic evolution of the fuel had to be extremely nonreversible, including shock waves, cladding rupture, and fuel fragmentation. There are several theories[37,38] and computer codes[39,40] that could be used to study this phase, at least in a simplified way, because the particular geometry of RBMK fuel channels would require specific mesh descriptions and boundary conditions.

To determine whether the fuel fragmentation across the cooling channel could induce a supercritical configuration, a parametric analysis was carried out. Results are listed in Table III, which corresponds to a multicell calculation without control rods. A dry reactor is assumed in all the cases. The main parameter is the smeared fuel density, the percentage of fuel mass that is smeared uniformly across the channel. The rest of the fuel mass is assumed to have been expelled from the channel due to the overpressure produced by the power surges. Taking the original configuration case as the reference point, it can be seen in Table III that the system is less reactive if all the fuel is smeared. However, if part of the fuel is removed from the channel, more reactive configurations can be found. For instance, a 65% smeared fuel density corresponds to a prompt critical state (its reactivity being >1 $ from the reference state).

Of course, the actual evolution of the fragmented

TABLE III

k_{eff} of Different Configurations Assuming Fuel Fragmentation*

Fuel Temperature (K)	Solid Rods	Smeared Fuel Density (%)				
		100	65	60	55	50
973	1.0629	1.0593	1.0707	1.0706	1.0697	1.0697
1473	1.0566	1.0525	1.0648	1.0649	1.0642	1.0623
1973	1.0513	1.0467	1.0598	1.0600	1.0595	1.0578
2473	1.0465	1.0415	1.0554	1.0558	1.0554	1.0539

*Multicell calculation, no control rods.

fuel was probably rather heterogeneous, but this parametric study shows at least the possibility of an additional power surge at the very moment of the disassembly accident. Similar conclusions have been found in other works.[41] Nevertheless, the practical impossibility of simulating the fuel fragmentation evolution prevents estimation of the energy yield of such an additional power burst, but it would be surely much lower than the first two because the maximum reactivity would be ~1.5 $.

III.D.2. Thermochemical Reactions

Exothermic chemical reactions can occur in the last phase of the accident due to the temperature rise. Other phenomena such as fuel and cladding fragmentation help enhance the reaction rate of those processes, although they have time constants much longer than the neutronic period of a super-prompt-critical excursion.

There are two main potential sources of chemical energy in a scenario similar to that of the Chernobyl accident: hydrogen and graphite fire. Hydrogen can induce explosions in a relatively short time. Graphite fire is mainly related to the evolution of the accident at longer term. A brief analysis is presented in order to estimate the order of magnitude of the potential energy releases and to compare it with the first explosions.

Hydrogen can be generated by water radiolysis, graphite oxidation by steam, and zirconium oxidation. The latter is by far the most important mechanism under accident conditions. The surface oxidation rate depends mainly on the temperature and it becomes very important over 1473 K, a level that was passed in the accident. Zirconium is the main component of both the cladding and the pressure tubes: There are ~50 Mg in the cladding with an external surface of 9000 m^2 and 80 Mg in the tubes with a surface of 3000 m^2.

According to our estimates, in the 4 s following the first burst, 200 kg of hydrogen was produced by the oxidation of 4.8 Mg of zirconium. These figures depend to some extent on the assumed hypothesis about the cladding fragmentation, because the zirconium external surface is a parameter of primary importance. Other authors[42,43] assume a finer fuel and cladding fragmentation (~100 μm) and find a production rate of 300 kg of hydrogen in the first second. In any case, the maximum amount of hydrogen generated by the total oxidation of the zirconium was ~2 Mg, but the hydrogen actually available for detonation would be a small fraction of that. In any case, the explosion of 2 Mg of H_2 would have produced 240 GJ, which is equivalent to 57.4 t of TNT. This figure is of the order of the first power burst and about one-fourth of the second one, and it represents the maximum energy available. Our estimates indicate that only 10%, ~25 GJ, could be available to explode by sudden oxidation. There is no direct evidence of hydrogen detonations after the accident, but it is very likely they were not a single large explosion but many small ones as the hydrogen escaped from the reactor remains.

The graphite fire was a classical combustion triggered by the very high temperatures achieved after the power surges. Some graphite blocks were expelled from the reactor but most remained with the reactor debris. Reference 1 reports the efforts to quench the fire, including covering the unit with lead and clay. The chemical energy contents of the 1700 Mg of graphite was 70 TJ. This figure is much higher than the largest energy burst (~1 TJ), but it must be taken into account that this energy cannot be released suddenly. It is a combustion process that needs a surface contact between the hot graphite and the oxygen. An upper bound of the graphite fire power has been estimated as 100 MW. This means that it could be higher than the decay heat power (~25 MW), which was not a fundamental parameter during the accident itself (because it evolves at a very low speed as compared to the neutronic evolution) but later became a fundamental source of heat.

In summary, the exothermic chemical reactions did not play any role at all during the central part of the accident but contributed in the aftermath both to enhance the radioactive leakage and to worsen the unit conditions.

IV. A DESCRIPTION OF THE PHYSICAL CAUSES OF THE ACCIDENT

The previous sections describe the individual behavior of several variables that determined the power evolution in Chernobyl Unit 4 during the accident. This information can be easily assembled to describe why and how the accident occurred and reached catastrophic dimensions. This work deals only with the physical phenomena and not with the human factor, i.e., the operator violations of several safety principles and procedures. Similarly, the experiment is not analyzed. It seems that no one on the operating shift understood the physics of the experiment and they intended to maintain the reactor critical for the sake of repeating the experiment if necessary, in spite of the strong anticipated transients in the thermal-hydraulic balance and in the neutronic power of the reactor.

In just one sentence, it can be stated that the accident took place because a loss of pumping power was produced when the reactor was unstable both in a linear sense (immediate response) and in the final state achievable.

The linear instability had a double root:

1. The thermal-hydraulic conditions just before the accident were very close to saturation even at the reactor inlet. The pressure excess over the saturation point was ~0.5 MPa, much lower than in normal conditions. Subcooling (undersaturation) at the reactor inlet was also very poor. This was due to a very high circulation flow rate compared to the energy level. These conditions caused a very large change in water density, which, because of transition to steam, could be produced by any significant loss of pumping power. Figure 5 depicts this situation. The total failure of half of the pumps could induce a pressure drop of the order of the pressure excess over saturation, i.e., the water could boil from the very inlet of the reactor.

2. The reactor at that moment had a very large positive reactivity coefficient. An indirect but reliable measurement of that coefficient was the operation reactivity margin, i.e., the number of control rods inserted in the reactor. Just before the accident, this number was ~10, much lower than the permitted minimum value of 30. This figure has since been raised to 80 in the safety specifications for RBMK reactors. Such a low number of rods in was a clear indication of the very high neutronic importance of the hydrogen captures in the cooling water.

These mechanisms induced a first reactivity trip mainly governed by the loss of pumping power. This first power burst was overcome by the Doppler effect, but the fuel temperature increase needed to counterbalance the former reactivity feedback was so high that the fuel reached the disaggregation enthalpy level. The fuel evolution was nonreversible along this power burst, which lasted ~1 s, i.e., 2000 times the neutron lifetime. The total energy released during this burst was ~200 GJ, which means there was enough energy to produce fuel and cladding fragmentation in some of the fuel channels.

The energy of the first burst, originally mainly deposited in the fuel, was transmitted to the water with a delay of ~1 s (see Table II). The process was also nonreversible, quite similar to a steam explosion, i.e., the sudden and violent expansion of overheated steam in contact with a very hot surface. The water was expelled from the reactor in an overpressure transient that was clearly recorded in the steam drum separators. This process led the reactor to the final state, a hot and dry reactor. Due to the great importance of the neutron hydrogen captures in the neutronic balance, which maintained criticality before the accident, the final state was fairly above prompt critical. In other words, the total reactivity available through the water voidage feedback (5 \$) was larger than the antireactivity produced by the maximum fuel temperature increment (about −4 \$). A second power burst with an energy yield of 1 TJ occurred and practically destroyed the reactor.

This short summary presents the physics of the accident. Of course, the final cause was the mismanagement of the reactor by operators who seemed not to understand the physics of the plant.

The accident could have been avoided if the neutronic importance of hydrogen captures had been diminished by establishing a much lower flow rate, a higher steam outlet quality, and cooler water at the reactor inlet (with a pressure excess over saturation much higher than the actual 0.5 MPa). For instance, with 645 $kg \cdot s^{-1}$ of flow rate through the reactor, the steam flow would had been 93.5 $kg \cdot s^{-1}$, which means an outlet quality of 14.5%. The inlet subcooling would have been −8.3% instead of −2% [see Eq. (1)]. In this situation, the number of control rods in the reactor would have been >30, as required. In any case, the hydraulic performance would not have been suitable to test the experiment (anticipated to be performed at much higher power, ~1000 MW). It seems the shift did not comply with all the experimental specifications, and neither the experiment designers nor the shift understood the safety implications of such a test.

V. CONCLUSIONS

It can be said that the accident occurred because of the operator safety violations and mistakes, but it is also clear that those violations and mistakes had a great significance and a catastrophic consequence due to the physics of RBMK reactors.

All reactors have some theoretical limits to guarantee stability, but it must be emphasized that those limits are very far from the actual situations of the plant in Western reactors, with the potential exception of LMFBRs.

Light water reactors are designed to be undermoderated, which is a guarantee of stability. The total water voidage leads the reactor to a very subcritical state (it is converted into a very poor fast assembly) where the main concern is the decay heat cooling. A pressurized water reactor can become overmoderated if very large soluble boron concentrations are used. This phenomenon is well known and can be avoided by the use of solid absorbers. Nevertheless, it must be taken into account in the design and licensing of each burnup cycle.

Boiling water reactors present a very strong coupling between hydraulics and neutronics, but they are perfectly stable at nominal conditions. Reactivity perturbations (originating from hydraulic perturbations, for instance) can induce power transients that can become important in some regimes that are very far from the permitted one.[44] In any case, a dry reactor (as a consequence of a steam explosion, for instance) is very subcritical.

Gas-cooled reactors can have a positive graphite temperature reactivity coefficient, but the evolution of this variable is much slower than the fuel temperature evolution. The coolant does not have any neutronic significance.

Heavy water reactors and LMFBRs deserve a deep analysis that lies beyond the scope of this paper, because the first is a channel-type reactor where the coolant and the moderator are different fluids and the second can present coolant voidages configurations that lead to supercritical states.

Comparison of RBMK features with those of Western reactors has been the subject of several works.[15,45,46] All of them point out the different neutronic behavior of RBMK due to its overmoderated characteristic. A general conclusion of those works is the lack of need to change any design or operational specification in Western reactors as a consequence of the Chernobyl accident. However, all the works underline the importance of abiding by the rules established to properly operate nuclear reactors.

In the Western nuclear community, much attention has been paid to decay heat removal systems and loss-of-coolant accidents, *radioactivity* being the major concern of nuclear safety because *reactivity* accidents are considered virtually impossible in the very stable Western reactors. The prototype of a *decay heat* accident is Three Mile Island Unit 2, which was orders of magnitude more benign than Chernobyl Unit 4, the prototype of a reactivity accident. This can be considered as an indication that a reactivity accident cannot happen in Western reactors.

ACKNOWLEDGMENTS

The late Counsellor Prof. F. Goded is particularly acknowledged.

This work was performed under a research contract with the Spanish Nuclear Safety Council.

REFERENCES

1. "The Accident at the Chernobyl Nuclear Power Plant and Its Consequences," information compiled for the Experts' Mtg., August 25-29, 1986, Vienna, Austria, INIS-mf-10523, International Atomic Energy Agency (1986).

2. I. YA. EMEL'YANOV, B. P. BASILIEVISKII, B. P. VOLKOV, P. A. GABRILOV, A. YA. KRAMEROV, S. P. KLUZNETZOV, E. P. KUNEGIN, and N. Z. RIVAKOV, "The Safety Problems of Nuclear Power Plants with Channel-Type, Graphite, Boiling-Water-Reactors," *Proc. Int. Conf. Nuclear Power and Its Fuel Cycles*, Salzburg, Austria, May 2-13, 1977, Vol. 5, paper CN-36/341, International Atomic Energy Agency (1977).

3. V. P. ANDREEV, A. V. BONDARENKO, V. K. VIKULOV, V. D. KOZYREV, F. M. KRASNOGOROV, V. F. LYUBCHENKO, V. V. MAL'TSEV, YU. I. MITYAEV, A. S. OVESHKOV, E. I. SNITKO, A. G. SHEINKMAN, and S. V. SHIROKOV, "Steam Effects and Reactivity Coefficients of the Beloyarsk Nuclear Power Station Reactors," *At. Energ.*, 50, 381 (1981).

4. V. I. PUSHKAREV, A. D. ZHIRNOV, and A. P. SIROTKIN, "Way of Altering the Coefficients of Reactivity in RBMK Reactors," *At. Energ.*, 46, 386 (1979).

5. V. S. ROMANENKO and A. V. KRAYUSKHIN, "Physical Characteristics of an RBMK Reactor in the Transitional Period," *At. Energy*, 53, 367 (1982).

6. I. YA. EMEL'YANOV, L. N. PODLAZOV, A. N. ALEKSAKOV, V. M. PANIN, and B. V. PARAEV, "Effect of the Shape of the Height Distribution of Neutrons on the Stability of a Reactor to Xenon Oscillations," *At. Energ.*, 50, 385 (1981).

7. A. N. ALEKSAKOV, B. A. VORONTSOV, I. YA. EMEL'YANOV, L. N. PODLAZOV, V. I. RYABOV, and B. M. SVECHAREVSKII, "Deformation of an Energy Release Field in a High-Powered-Water-Cooled Channel Reactor (RBMK)," *At. Energy.*, 46, 227 (1979).

8. S. V. BRYUIN, A. I. GORELOV, V. YA. NOVIKOV, I. K. PAVLOV, and V. V. POSTINOV, "Optimization of the Safety Margin to the Critical Load of the Heat-Releasing Assemblies in a RBMK," *At. Energ.*, 46, 222 (1979).

9. K. YOSHIDA, F. TANABE, M. HIRANO, and A. KOHSAKA, "Analyses of Power Excursion in Chernobyl Accident with RETRAN Code," *J. Nucl. Sci. Technol.*, 23, 1107 (1986).

10. "Summary Report on the Post-Accident Review Meeting on the Chernobyl Accident," IAEA Safety Series No. 75-INSAG-A, International Atomic Energy Agency (1986).

11. "Report on the Accident at the Chernobyl Nuclear Power Plant," NUREG-1250, U.S. Nuclear Regulatory Commission (1986).

12. "The Tchernobyl Accident," IPSN 2/86, Commissariat à l'Energie Atomique (1986).

13. J. H. GITTUS, "The Chernobyl Accident and Its Consequences," *Atom*, **268**, 2 (June 1987).

14. M. CLEMENTE, S. LAGENBUCH, and J. P. WEBER, "Kern-Rechnungen zur Bewteilung des Unfallablaus in der Anlage Tschernobyl-4," Technische Notiz, GRS-A-1289, Gesellschaft für Reaktorsicherheit (Sep. 1986).

15. H. R. DENTON, "The Causes and Consequences of the Chernobyl Nuclear Accident and Implications for the Regulation of U.S. Nuclear Power Plants," *Ann. Nucl. Energy*, **14**, 295 (1987).

16. T. S. KRESS, M. W. JANKOWSKI, J. K. JOOSTEN, and D. A. POWERS, "Chernobyl Accident Sequence," *Nucl. Saf.*, **28**, 1 (1987).

17. C. L. HECK, L. E. HOCHREITER, P. HUANG, and A. STOLMAR, "An Analysis of the Chernobyl Accident," *Trans. Am. Nucl. Soc.*, **54**, 221 (1987).

18. R. KYRKI and R. RAJAMAEKI, "Dynamic Analysis of the Chernobyl Accident with the One-Dimensional Program TRAB," *Trans. Am. Nucl. Soc.*, **54**, 222 (1987).

19. P. GULSHANI, A. R. DASTUR, and B. CHEXAL, "Stability Analysis of Spatial Power Distribution of the RBMK-1000 Reactor," *Trans. Am. Nucl. Soc.*, **54**, 383 (1987).

20. P. LOIZZO, P. AZZONI, and M. G. BORGIA, "Comprehensive Analysis of the Chernobyl Accident," Proceedings Series IAEA-SM-296, p. 199, International Atomic Energy Agency (1988).

21. H. TOFFER and R. W. TWITCHELL, "Dynamic Analysis of the Chernobyl Accident," *Trans. Am. Nucl. Soc.*, **55**, 411 (1987).

22. T. WAKABAYASHI, H. MOCHIZUKI, H. MIDORIKAWA, Y. HAYAMIZU, and T. KITAHARA, "Analysis of the Chernobyl Reactor Accident. Part 1," *Nucl. Eng. Des.*, **103**, 151 (1987) and "Analysis of the Chernobyl Reactor Accident. Part 2," *Nucl. Eng. Des.*, **106**, 163 (1988).

23. K. TSUCHIHASHI and F. AKINO, "Analysis of Reactivity Coefficients of Chernobyl Reactor by Cell Calculation," *J. Nucl. Sci. Technol.*, **24**, 1055 (1987).

24. P. W. W. CHAN et al., "Chernobyl and Pump Cavitation," *Nucl. Eng. Int.*, **33**, 16 (1988).

25. A. BUCCAFURNI, "Performance of the Standard Calculations Tools in the Analysis of the Chernobyl Accident," Proceedings Series IAEA-SM-296, p. 371, International Atomic Energy Agency (1988).

26. C. M. HEEB et al., "Three-Dimensional Diffusion Theory Model of Chernobyl 4," *Trans. Am. Nucl. Soc.*, **54**, 382 (1987).

27. P. LOIZZO, "Neutronics of the Accident at Chernobyl," IAEA-TC-610, p. 152, International Atomic Energy Agency (1987).

28. M. J. HALSHALL, "A Summary of WIMS-D4 Input," AAEW-M-1327, U.K. Energy Establishment, Winfrith (1980).

29. J. M. ARAGONÉS and C. AHNERT, "Lola System: A Code Block for Nodal PWR Simulation," JEN-567, Junta de Energía Nuclear (1984).

30. T. R. HILL and W. H. REED, "TIMEX: A Time-Dependent, Explicit, Discrete Ordinates Program for the Solution of Multigroup Transport Equations with Delayed Neutrons," LA-6201-MS, Los Alamos National Laboratory (1976).

31. G. L. WHEELER, C. W. STEWART, R. I. CENA, D. S. ROWE, and A. M. SUTEY, "COBRA-IV-I: An Interim Version of COBRA for Thermal-Hydraulic Analysis of Rod Bundle Nuclear Fuel Elements and Cores," BNWL-1962, Battelle Northwest Laboratory (1976).

32. D. D. LANNING, C. L. MOHR, F. E. PANISKO, and K. B. STEWART, "GAPCON-THERMAL-3 Code Description," PNL-2434, Pacific Northwest Laboratory (1978).

33. J. M. MARTÍNEZ-VAL et al., "An Analysis of the Physical Causes of the Chernobyl Accident," DENIM-100, Instituto de Fusión Nuclear, Universidad Politécnica de Madrid (1986).

34. J. M. MARTÍNEZ-VAL and J. M. ARAGONÉS, "Reactivity Calculations of the Chernobyl Accident," *Trans. Am. Nucl. Soc.*, **57**, 177 (1988).

35. J. F. JACKSON and M. G. STEVENSON, "LMFBR System Safety Analysis," *Nuclear Reactor Safety Heat Transfer*, p. 647, O. C. JONES, Jr., Ed., McGraw-Hill Book Company, New York (1981).

36. C. D. FLETCHER, R. CHAMBERS, M. A. BOLANDER, and R. J. DALLMAN, "Simulation of the Chernobyl Accident," *Nucl. Eng. Des.*, **105**, 157 (1988).

37. H. K. FAUSKE, "On the Mechanism of Uranium Dioxide-Sodium Explosive Interactions," *Nucl. Sci. Eng.*, **51**, 95 (1973).

38. S. J. BOARD, R. W. HALL, and R. S. HALL, *Nature*, **254**, 319 (1975).

39. L. D. BERTHOLF and S. E. BENZLEY, "TOODY II, A Computer Program for Two Dimensional Wave Propagation," SC-RR-68-41, Sandia National Laboratories (1968).

40. W. T. SHA and T. H. HUGHES, "VENUS: A Two-Dimensional Coupled Neutronics-Hydrodynamics Computer Program for Fast Reactor Power Excursions," ANL-7701, Argonne National Laboratory (1970).

41. M. RAJAMAKI and F. WASATJERNA, "On the Reactivity Effects of Nuclear Fuel Fragmentation with Reference to the Chernobyl Accident," *Nucl. Sci. Eng.*, **101**, 41 (1989).

42. J. L. HENRIE and A. K. POSTMA, "Hydrogen Generation During the TMI-2 and Chernobyl Events," *Trans. Am. Nucl. Soc.*, **54**, 225 (1987).

43. M. ISHIKAWA, S. SHIOZAWA, T. WAKABAYASHI, and N. OHNISHI, "An Examination of the Accident Scenario in the Chernobyl Nuclear Power Station," *Nucl. Saf.*, **28**, 448 (1987).

44. J. MARCH-LEUBA, D. G. CACUCI, and R. B. PÉREZ, "Nonlinear Dynamics and Stability of Boiling Water Reactors," *Nucl. Sci. Eng.*, **93**, 111 (1986).

45. B. L. COHEN, "Nuclear Reactor Accident at Chernobyl, USSR," *Am. J. Phys.*, **55**, 1076 (1987).

46. "Chernobyl and the Safety of Nuclear Reactors in OECD Countries," INIS-XN-81, Nuclear Energy Agency (1987).