The Chernobyl-4 Reactor
and the
Possible Causes of the Accident

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INTRODUCTION

I will limit the description of the Chernobyl plant to the extent required to introduce the accident sequence of events. I will afterwards attempt to delineate some analogies and differences between the Chernobyl power station and better known nuclear plants operated in the Western countries.
And a last part will be devoted to the description of the events which were, at least for a large part, the cause of the accident.

In 1954 the Soviet Union put into operation its first nuclear electrical power plant (6 MWe) in Obninsk. This was 3 years before Shippingport (90 MWe) became the first American PWR operating power plant, but 3 years later than the very first nuclear power plant in the world, which was a fast power plant of 0.2 MWe, the EBR (Experimental Breeder Reactor) built in Arco (Idaho): the first known generation of electricity from nuclear power was realized from the EBR in late December 1951.
The Obninsk power plant is still working; to those amongst you who are familiar with our own reactors, I would say that it is the Soviet BR3. This Obninsk power plant is very interesting in so far that it is the prototype of the RBMK-type reactors of which four working units and two more under construction formed the equipment of the Chernobyl power plant.

Reactor

Bolshoy   Large

Moschnasti   Power

Kipyashiy   Boiling

                                 Reactor

Kokend (water) reaktor met groot vermogen.

Réacteur (à eau) bouillant(e) de grande puissance.

Figure 1

SGHWR

Steam generating heavy water reactor

(RBMK)

Steam generating graphite reactor

TYPES OF REACTORS IN THE U.S.S.R.

RBMK

Steam Generating Graphite Reactor				
6 Exp. unit OBNINSK (1954)	100 - 200 «early plants» TROITSK 1 (1958)	1000 LENINGRAD 1 (1974)	1500 IGNALINA 1 (1984)	(2400) MWe
	1958 - 67 Demonstration plants for commercial power production First series for Pu prod. and for power generation	1974 - 89 Fully commercial plants	1984 - ?	?

VVER

Pressurized Water Reactor				
		1000 NOVOVORONEJ (1980)		1500 MWe

BN

(BR-5)	BN 350 (BOR 60)	BN 600 BIELOYARSK (1980)	BN 800 Start of construction, BIELOYARSK Criticality 90 ?	BN 1600
440				

Heat Generating Plants and Combined Heat/Electricity Generating Plants
VVER type [(range 300-500 MWth) to largest plants (VVER-1000)] plants and modifications to existing plants (LENINGRAD, CHERNOBYL, ...)

High Temperature (1000°C) Reactors, helium cooled

Figure 3

The initials RBMK (fig. 1) are the abbreviation for "Reactor
Bolshoy Moschnosti Kipyashiy": that could be translated by
"Large Power Boiling Reactor", an appellation which is not
very specific and could lead to confusion with our BWR's
(Boiling Water Reactor).

In analogy (fig. 2) with the nearest "Western" design, the
English SGHWR "Steam Generating Heavy Water Reactor", which
we will discuss later, I would propose to qualify the RBMK
as Steam Generating Graphite Reactor.

Fig. 3 gives a survey of the about three to five reactor
types developed in the Soviet Union; in the RMBK series, one
can distinguish 5 stages :
- the progenitor Obninsk, which I mentioned already.
- the "early plants",such as the six 100 MWe units
  commissioned in Troitsk (Siberia) between 1958 and
  1963, which were at the same time demonstration plants
  for the commercial production of nuclear energy and
  plants for the production of energy and plutonium.
- the "fully commercial plants" of the 1000 MWe range of
  which Leningrad 1, put into operation in 1974, was the
  first; all units of the Chernobyl power plant are of
  this type.
- a first extrapolation to 1500 MWe which required a
  minimum of modifications and resulted in the Ignalina 1
  power plant put into operation in 1984, as first reali-
  zation.
- designs - probably still on the drawing-board - of a
  second extrapolation towards even much more powerful
  units; the range of 2400 MWe is mentioned.

The absolute and relative importance of the RBMK power
plants in the Soviet nuclear park can be seen from fig. 4 :
the evolution of the total installed nuclear power in GWe
till the end of 1985 and the previsions of the XII Quinquen-
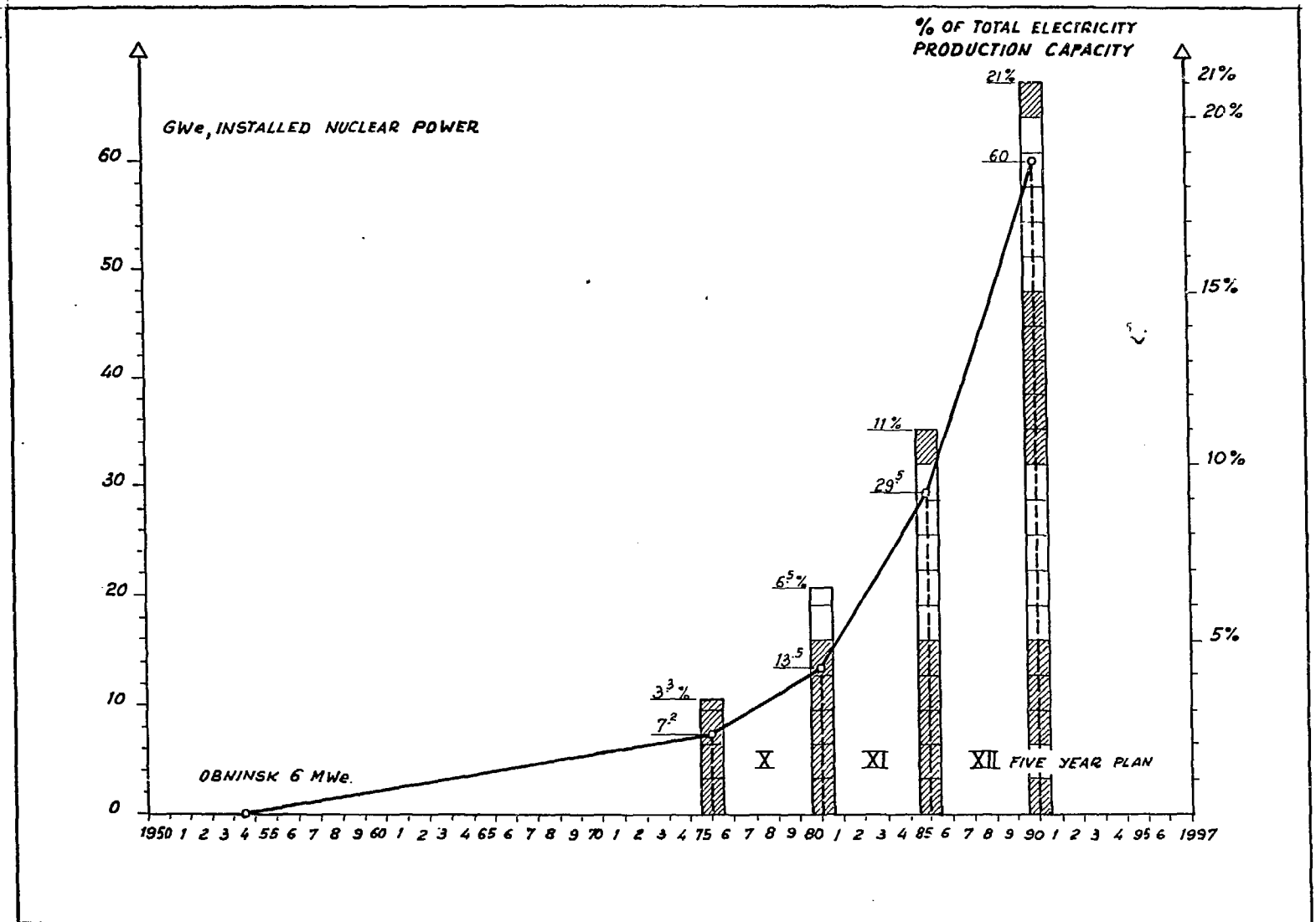nial Plan (1986 - 1990) which should bring the installed

Figure 4 : Evolution of the installed nuclear power in the USSR

power up to 60 GWe, about 21 % of the total electricity production capacity in the USSR.

We can see that till 1976, the importance of nuclear energy for the production of electric energy was rather small : in fact it was considered as a supplementary power source for the industrial areas of the European part of the country which were poor in fossil fuels. The predominating importance was in this period undoubtedly the production of plutonium for military purposes, which rested almost entirely on the reactors of the RBMK-type. The successive accelerations of the X, XI and XII Quinquennial Plans show a remarkable change in the Soviet energy policy. Will the Chernobyl accident produce a certain stagnation or a spreading out of the objectives? That is possible but not necessarily probable in a country with a fully planned economy.

About 60 % of the total nuclear power installed at the end of 85 was covered by the reactors of the RBMK-type which were assuring about 5 to 6 % of the total electricity production.

This seems a very low percentage, but it should be noted that this energy is produced in highly industrialized areas where its economic impact is much more important than is shown by this overall statistic for a country as large as a continent.

The remaining point of this introduction is to locate the Chernobyl Power Plant on the map. Fig. 5 shows part of the Ukraine Republic with its capital Kiev (2.4 millions inhabitants); the Chernobyl Power Plant is located along the Pripyat river between Pripyat and Chernobyl, about 130 km North of Kiev.

Pripyat, the nearest locality (Fig. 6), at a few km's distance, is a community of 49,000 inhabitants which accomodates amongst others the people working on the building site
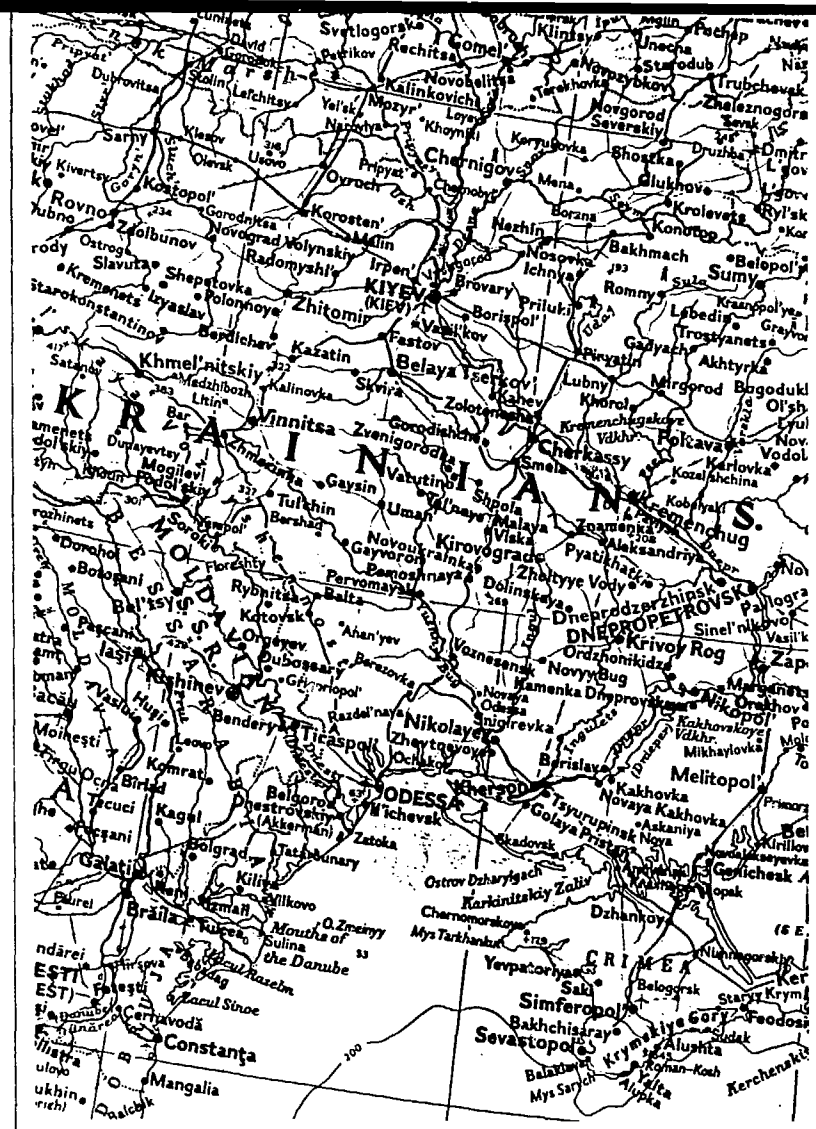
Figure 6 : Enlarged view.
         Region around Chernobyl

Figure 5 : The Ukraine Republic with its
         capital Kiev

SLIGHTLY ENRICHED URANIUM

GRAPHITE MODERATED AND REFLECTED

BOILING WATER COOLED

PRESSURE TUBE

3200 MWth - 1000 MWe (gross)

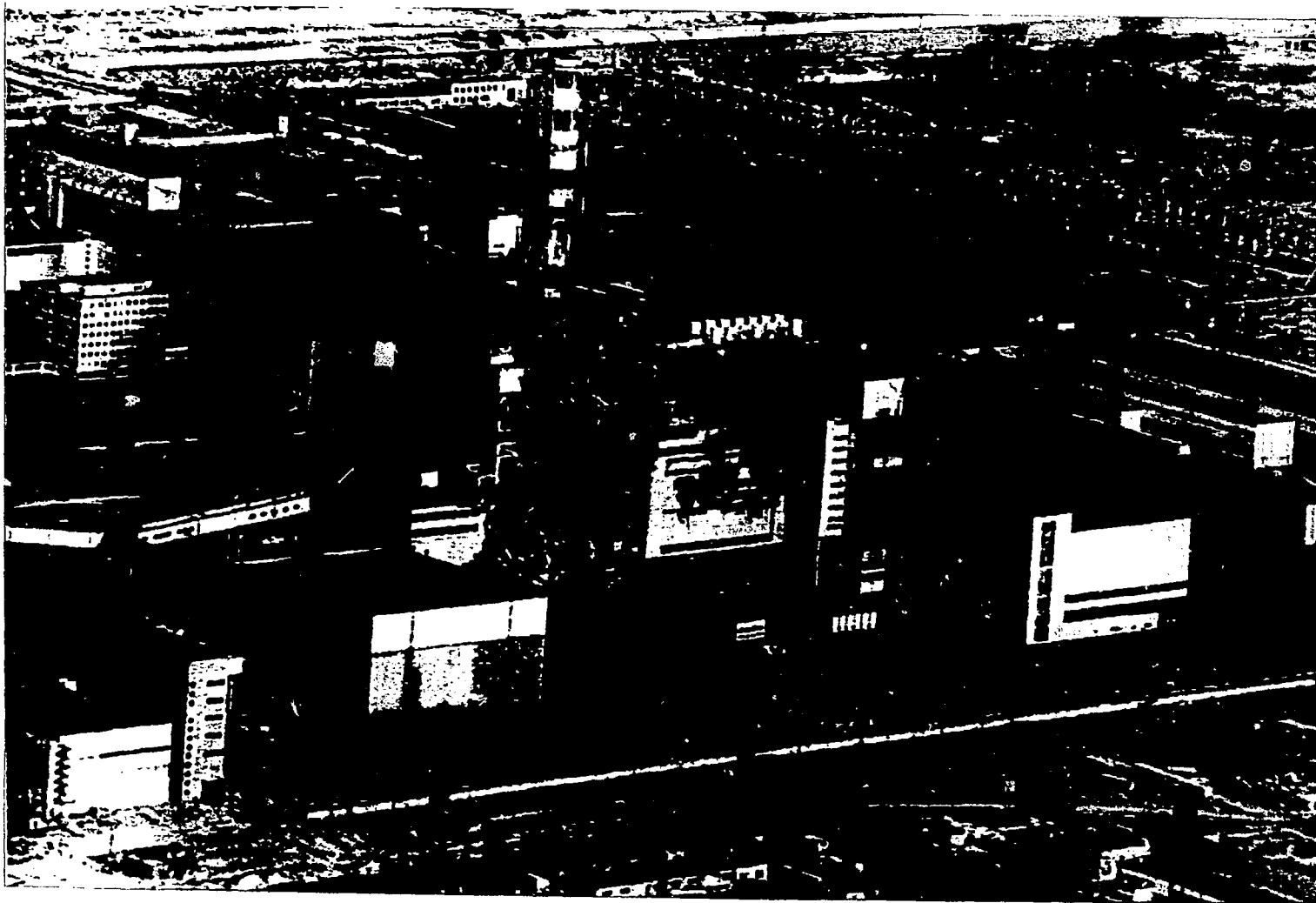|	
START OF CONSTRUCTION	**1975**
FIRST CRITICAL	**11/83**
GRID CONNECTION	**12/83**
COMMERCIAL OPERATION	**FIRST MONTHS OF 1984**

Figure 7 : The Chernobyl Power Station Unit 4

Figure 8 : A view of the Chernobyl power station taken on May 9
(Source : Associated Press)

of unit 5 and 6 of the Chernobyl Power Plant, who make up an important temporary increase of population.

Chernobyl is a small town with 12,500 inhabitants located South-South-East of the power station's site at some 15 km distance. Note the large lake or water reservoir on the Dnieper river, which is fed by several rivers and is used for the water-supply of the town of Kiev.

Fig. 7 is the identity card of unit 4 of Chernobyl. It shows in particular that this unit had a working experience of only 2 years whereas the RBMK generation as a whole had already accumulated a total of <u>245 reactor-years</u> without notorious incident.

Fig. 8 is a reproduction of a photo of the Chernobyl power station taken May 9, with in the foreground the building containing unit 4 destroyed by the explosion of April 26th; one can distinguish a first parallelepiped block topped by a stack where the reactor halls of ·unit 3 and 4 were housed, separated by the central part of the block which contained the common auxiliaries for both units; behind this first block a second building, lower and much longer, can be seen; this is the machine room where the four 0.5 GWe turbo-groups of the two units were housed.

Fig. 9 shows again the general shape of the twin unit building as also the corresponding plan-view, where the general disposition, as outlined above, is schematically illustrated.

Fig. 10 is a cross-sectional view perpendicular to the main axes of the two afore-mentioned blocks and passing through the axis of one of the two reactors housed in the building. (it is in fact a view of the Smolensk plan, very similar to Chernobyl).
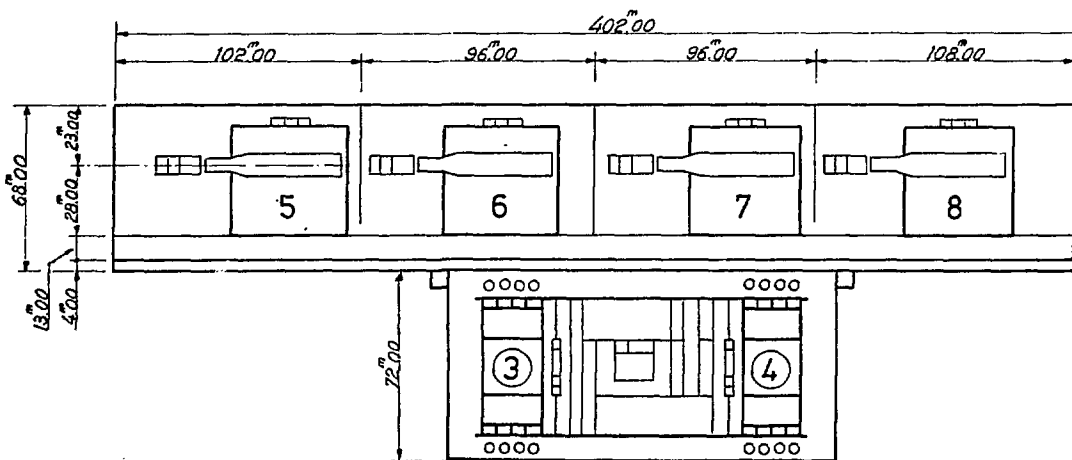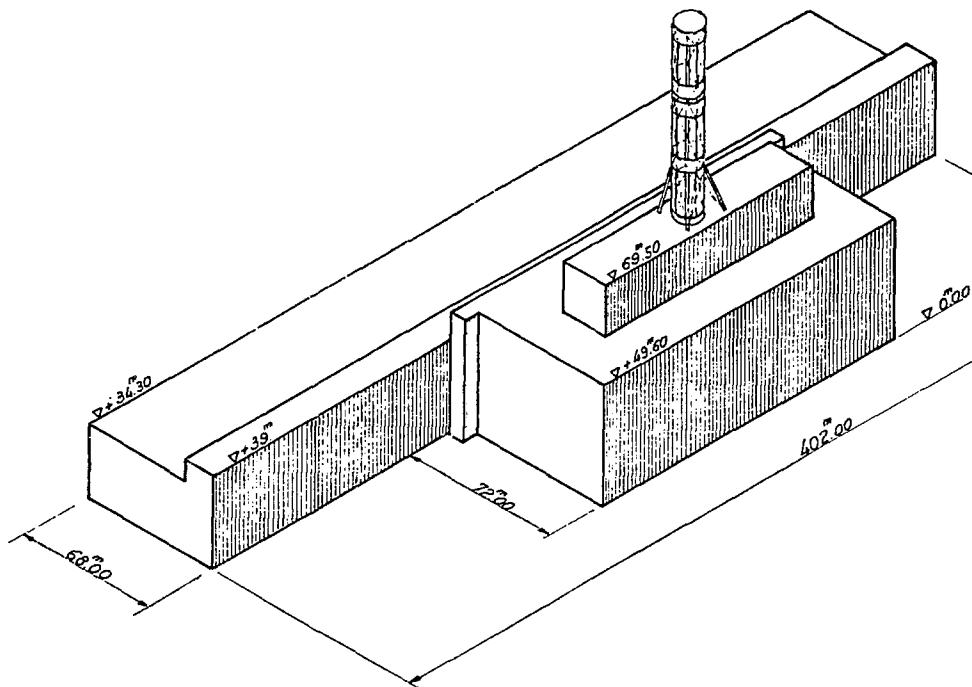
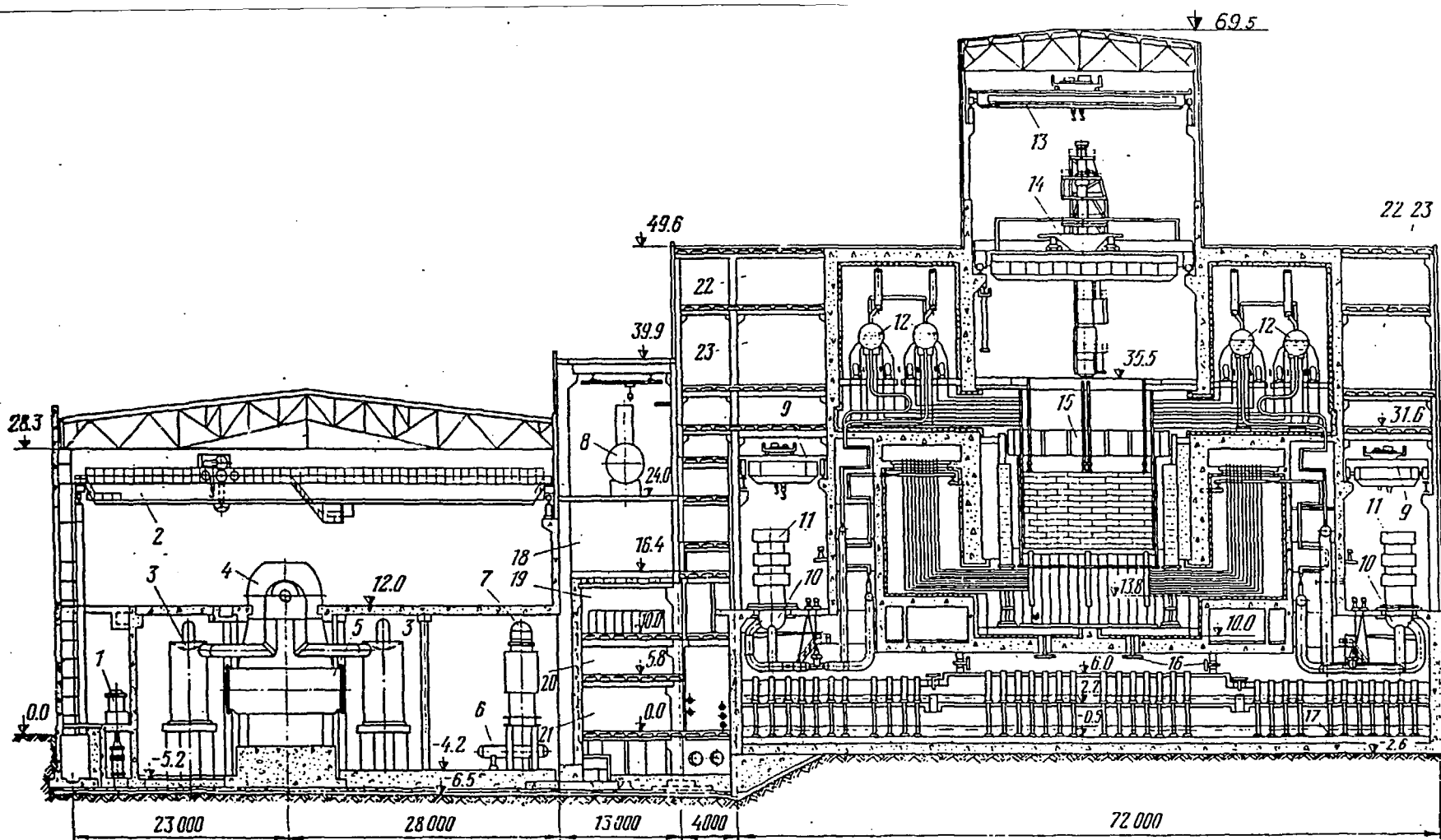Figure 9 : General shape and plan-view of the twin unit
building

Fig. 10   Cross-sectional view of the main building at Smolensk

1—first-stage condensate pump; 2—125/20-t overhead travelling crane; 3—separator-steam superheater; 4—K-500-65/3000 steam turbine; 5—condenser; 6—additional cooler; 7—low-pressure heater; 8—deaerator; 9—50/10-t overhead travelling crane; 10—main circulating pump; 11—electric motor of main circulating pump; 12—drum separator; 13—50/10-t remotely controlled overhead travelling crane; 14—refueling mechanism; 15—RBMK-1000 reactor; 16—accident containment valves; 17—bubbler pond; 18—pipe aisle; 19—modular control board; 20—location beneath control board room; 21—house switchgear locations; 22—exhaust ventilation plant locations; 23—plenum ventilation plant locations
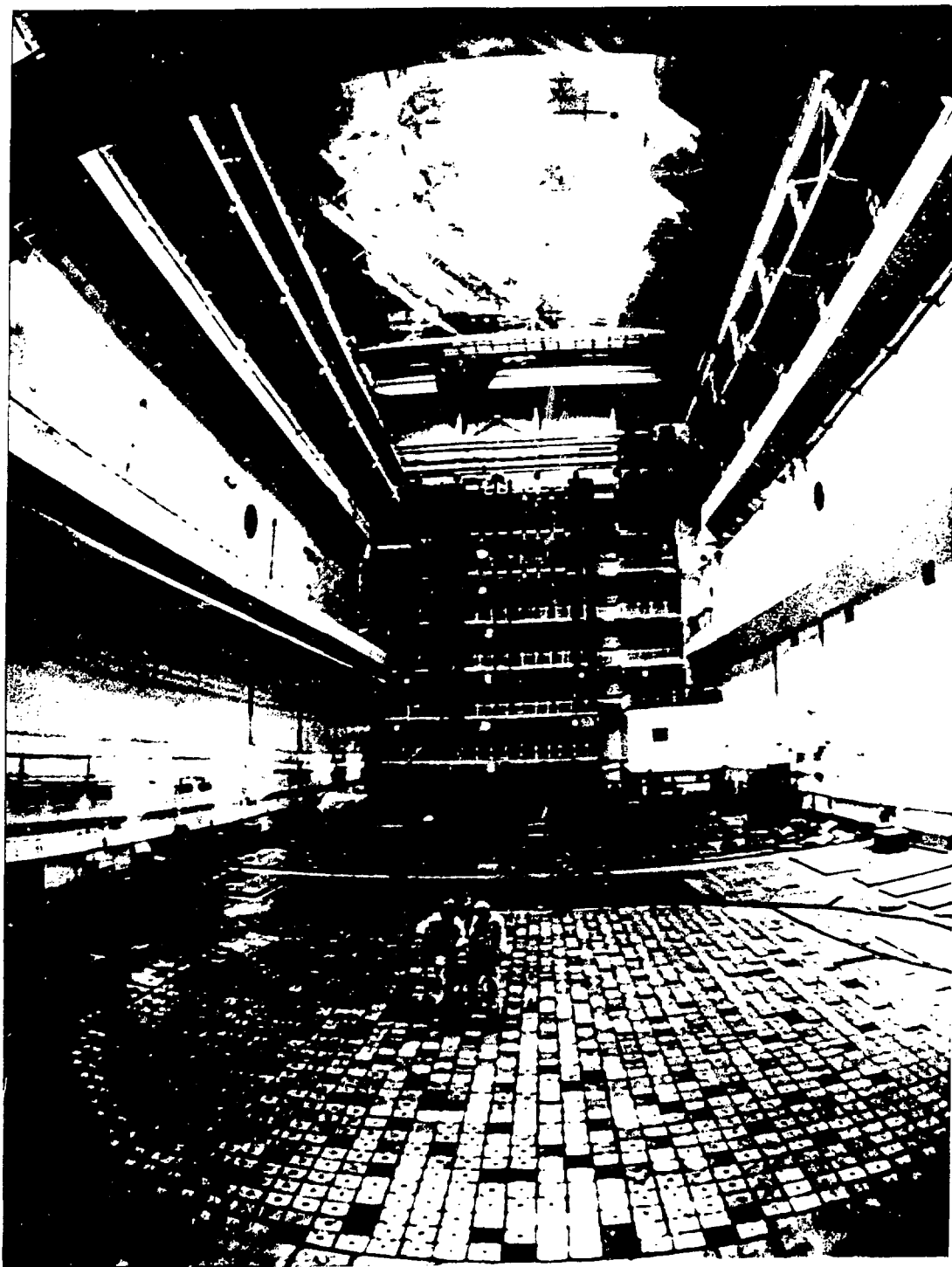
Figure 11 : A reactor hall at the Chernobyl plant
(Source : Associated Press)

Besides the general disposition it is interesting to note :

1) the height of the buildings culminating to approximately 70 m for the reactor hall itself; this height is justified by the dimensions of the reactor refueling machine; the Russians indicate this part of the building with the term "machine hall", though this appellation is used in Western power plants for the hall where the conventional part of the power plant, the turbo-groups, is housed.

2) the localization of the very large control room serving both units.

3) the concrete cells which shelter and separate the different vital components of the circuit : the dryer-separators, the primary pumps, the water supply lines, the reactor.

4) as a contrast to this, the light and conventional character of the superstructure of the machine hall strikes one immediately.

Fig. 11 is an impressive view of the reactor hall showing operators on the working platform above the reactor. This picture is taken on the level marked (35.5) on the previous figure (fig. 10).

1. Reactor
2. Working channel routes
3. Steam-to-water pipes
4. Drum separator
5. Steam headers
6. Downcomers
7. Main circulation pumps (MCP)
8. Group dispensing headers (GDH)
9. Water pipelines
10. Fuel element cladding control system
11. Upper biological shield
12. Lateral biological shield
13. Lower biological shield
14. Cooling pond
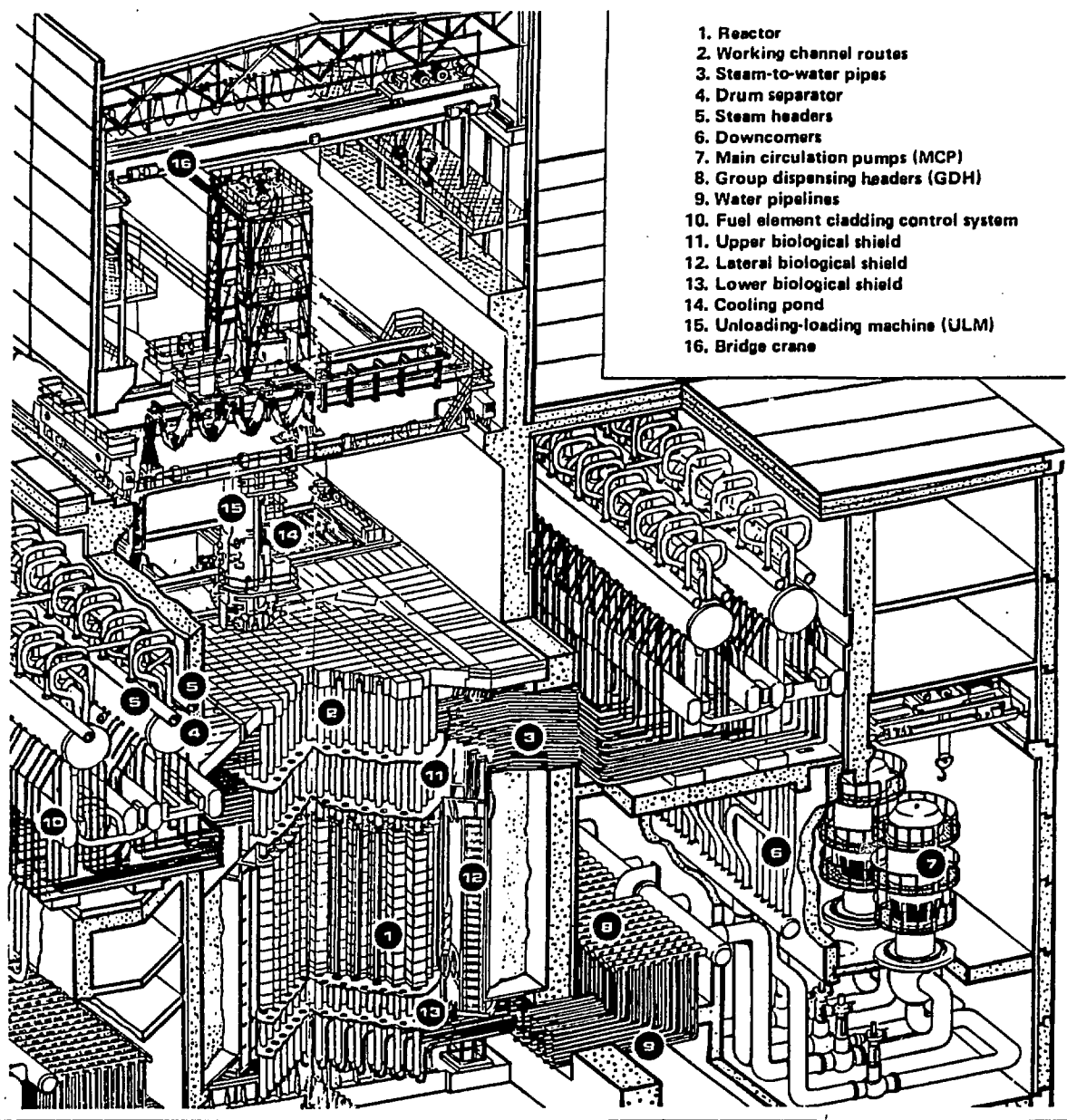15. Unloading-loading machine (ULM)
16. Bridge crane

Figure 12 : a sectional view of the RBMK-1000 reactor

DESCRIPTION OF THE POWER STATION

General Characteristics

The basic equipment of the RBMK-type power plants (fig. 12) consists of a pressure-tube reactor and a forced coolant circulation circuit made up of two loops each comprising two dryer-separators and four main circulating pumps with pipes and distributors from which the cooling water is supplied to the pressure-tubes.

Each reactor feeds two 0.5 GWe turbo-groups and each turbine is provided with a system for drawing off steam for the heating of the power plant, of the nearby residential areas and for other local users.

Fig. 13 shows an enlarged plan-view of one extremity of the building housing the two reactor halls and common auxiliaries. This enlarged plan-view shows the general disposition of the four drum-separators, grouped per two, and of the eight main circulation pumps, grouped per four, with regard to the reactor and to the refueling machine main displacement axis; it is also interesting to locate on this view the access to the deactivation pool in which the unloaded irradiated fuel assemblies are deposited by the refueling machine.

The Reactor

The RBMK reactor is a heterogeneous thermal neutron reactor; its fuel is slightly enriched uranium; it is moderated by graphite and light water and cooled by light water. This reactor can be refueled while in operation. The overall form is a huge vertical cylinder embedded in a concrete cavity; the cylindrical mass is a stack of graphite blocks contained in a thin-walled cylindrical steel vessel. This cylindrical mass is penetrated by numerous vertical tubes, called "pressure tubes", most of these pressure tubes contain fuel
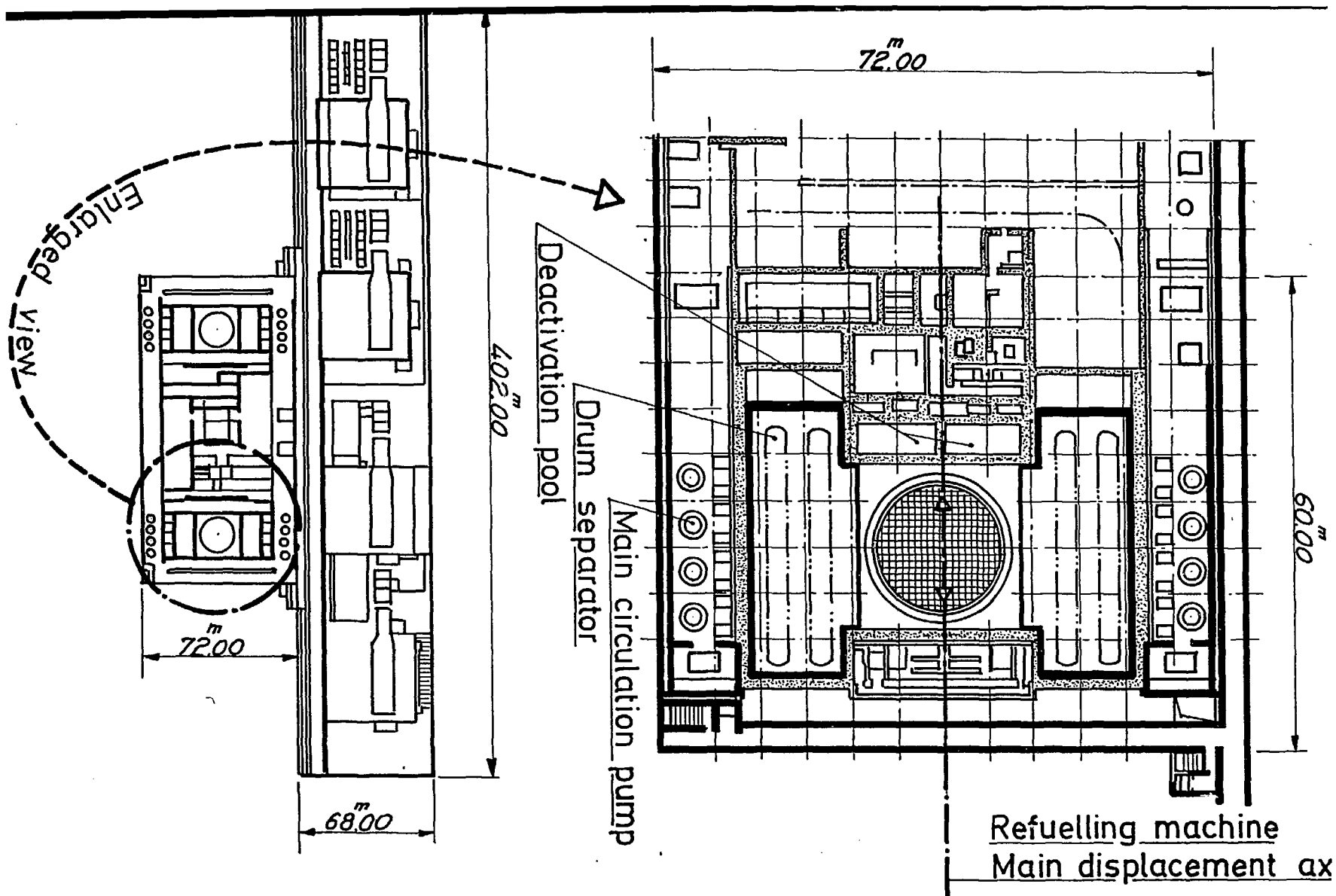
Figure 13 : Enlarged plan-view of the vicinity of the
reactor hall

assemblies and are traversed from the bottom upwards by the heat-transporting fluid which is a mixture of water and steam, with variable steam content, kept under a pressure of about 80 bars.

The Graphite Masonry

Figure 14 gives us a more detailed but schematic view of the graphite masonry.

It is formed by parallelepiped prisms with a horizontal cross section of 250 x 250 mm and a central cylindrical hole such that each vertical column of these blocks can accomodate one pressure tube. In this vertical pressure tube, the fuel assembly which has an active height of 7 m is hung. The central part of the graphite masonry with a height of 7 m and an equivalent diameter of 11.8 m, forms the reactor core, i.e. the part loaded with fuel. The core is surrounded by a reflector which has a radial thickness of 1 m and an axial thickness of 0.5 m at the top and the bottom; this brings the outer dimensions of the graphite masonry equivalent cylinder to a height of 8 m and a diameter of 13.8 m. This stack is leak-tight enclosed in a thin-walled stainless steel cylindrical vessel with a diameter of 14.2 m and a height of 9.75 m. In order to prevent graphite oxidation, this vessel is filled with a gaseous mixture of helium and nitrogen with a volumetric composition of 85-90% He and 15-10% N_2; the purity of this mixture is maintained by means of circulation in a purification unit.

Evacuation of the heat developed in the graphite mass (5% of the reactor power i.e. 160 MW_{th}) is achieved by the cooling fluid circulating in the pressure-tubes. Figure 15 shows how a system of graphite rings assures the thermal contact between the graphite mass and the pressure tubes.

The working temperature of the graphite is high : it attains 750° at the warmest spots.
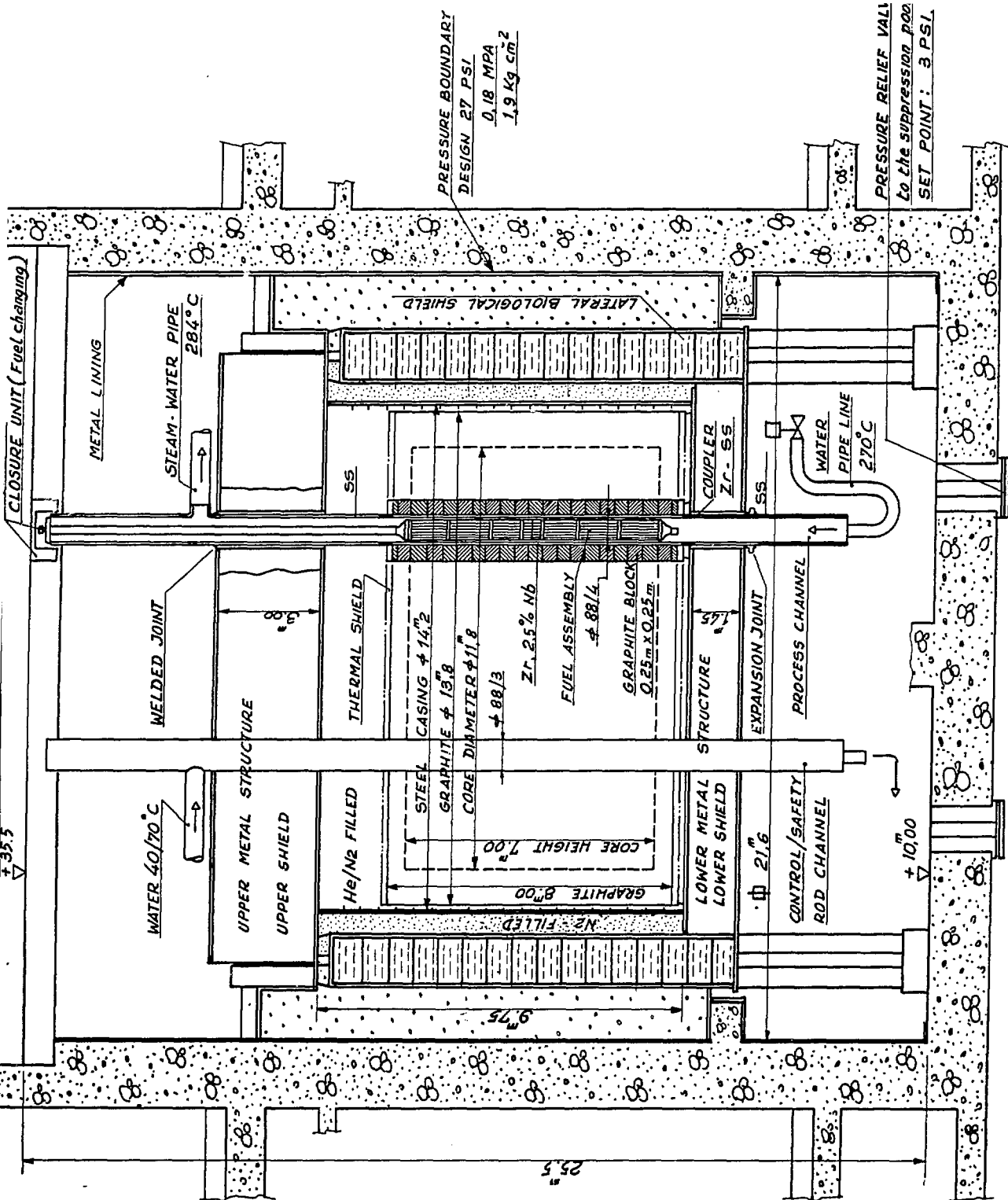
Figure 14 : Cross-sectional schematic view of the reactor vault

Fig. 14 shows also beneath the vessel, a steel supporting structure containing shielding material with a thickness of 1.45 m; above the vessel, an equivalent general disposition is found but with a thickness of 3 m offering to the working floor, at the top of the reactor vault, an adequate biological protection. This biological shielding is completed by a lateral wall, formed by water contained in an annular steel casing.

The exterior of the reactor vessel is surrounded by nitrogen which is kept under a higher pressure than the helium - nitrogen mixture inside the vessel to prevent the possibility of this mixture leaking from the graphite vessel.

The pressure tubes

Fig. 14 shows us also very schematically a pressure tube. This is a metal tube; its central part within the graphite stack is made of a (Zr + 2.5 % Nb) alloy and has an outside diameter of 88 mm and a thickness of 4 mm. It is extended at both sides by upper and lower sections made of stainless steel; the connection between the Zr-alloy and the stainless steel is realized by special couplers which are illustrated on figure 16. At its lower end, the pressure tube is axially connected with the water loop (T_{In} = 270°C) through a water feed pipe.

Each water feed pipe is equipped with an individual flow-meter and an individual valve in order to be able to adjust the flow of the coolant fluid in the pressure tube. The purpose of regulating the water flow of each channel indivi-dually is to ensure that there is a sufficient margin to departure from nuclear boiling in the hottest fuel channels while maintaining the total flow of water through the reactor close to the minimum required level. The regulation of the water flow through each channel is carried out in accordance with the reading of the flow-meter until a theoretically determined flow-rate is reached.

Tube φ 88 x 4

φ 114

φ 91

φ 111
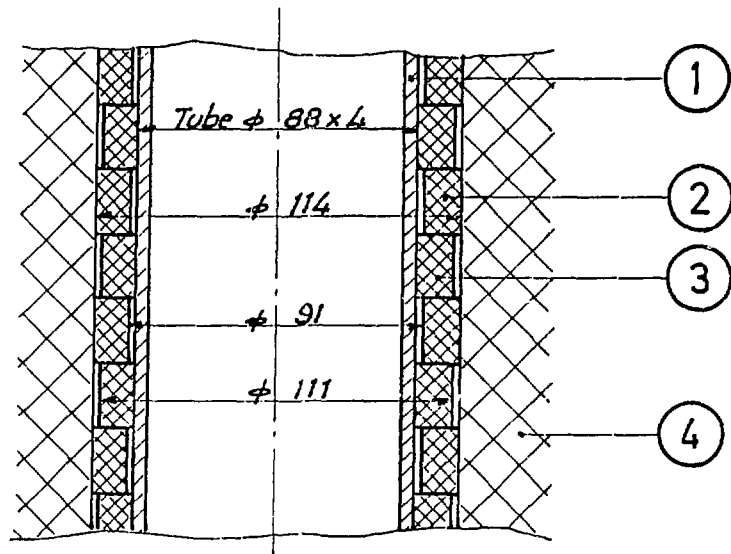
PROCESS CHANNEL
IN GRAPHITE STACKING

1   **Zr $+$ 2.5 Nb alloy tube**

2 and 3 **outer and inner graphite annuli, respectively**

4   **graphite stack**

Figure 15

The water flow is redistributed over the core fuel channels with changes in reactor power level and in the radial power distribution.

At its upper end the pressure tube is connected laterally to a steam-water mixture outlet pipe (T_{OUT} = 284°C, average steam content 15 %). Also, in the axis of the pressure tube a leak-tight plug-system is located which provides the interface with the refueling machine.
Each plug is covered by an individual slab, a metal structure filled with special concrete, forming a removable floor; this floor acts as additional biological shield and heat insulation for the reactor hall.
The distance between the inlet and the outlet of the coolant in a pressure-tube is about 19 m.
The pressure tube is housed in the reactor in tube ducts welded to the bottom steel supporting structure and to the top steel structure; the pressure tube is attached to the upper duct by a welded joint; the lower part of the pressure tube is welded to the lower tube duct through a bellows compensating unit; this makes it possible to compensate for any difference in thermal expansion of the pressure tube and metal structure as well as ensuring reliable tightness of the graphite vessel.
Whenever necessary, a defective pressure tube can be taken out of the reactor and replaced by a new one during a reactor shut-down.

Eventual channel leakage of water from the pressure tubes to the graphite containing vessel is detected by monitoring changes in the relative moisture content and in the temperature of the nitrogen-helium mixture; this gas mixture circulates in a closed circuit, in which it is separated from water vapour and graphite oxidation products; this circuit contributes also to the cooling of the graphite masonry.

Electron beam welding

Vacuum diffusion welding

Argon welding

68 x 4

① ② ③ ④

200
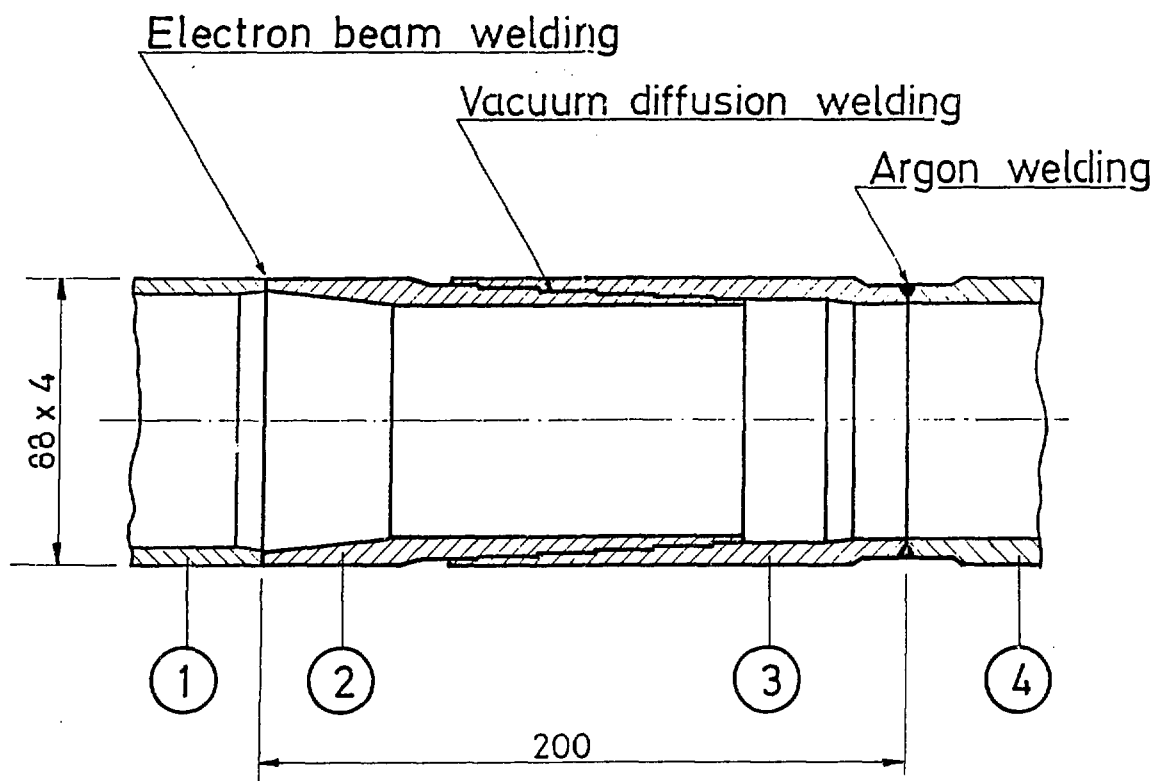
JOINING OF ZIRCONIUM ALLOY
TUBE TO STAINLESS STEEL TUBE

1   **Tube of Zr + 2.5 % Nb alloy**
2   **Nipple adapter of Zr + 2.5 % Nb alloy**
3   **Nipple adapter of 08Cr18Ni10Ti (austenitic) steel**
4   **Tube of 08Cr18Ni10Ti (austenitic) steel**

Figure 16

The channel leakage detection seems to be a critical point
for a single wall pressure tube, especially if the special
character of the Zr/SS connection within the graphite vessel
is taken into account; most probably a weak point.

Not all of the pressure tubes are devoted to contain a fuel
element, as explained up to now.
There are 1693 tubes of the type already described:
-   during the first core loading, 1452 pressure channels
    are loaded with a fuel assembly.
-   241 channels are loaded with absorbers to compensate
    for the initial excess of reactivity of a fully fresh
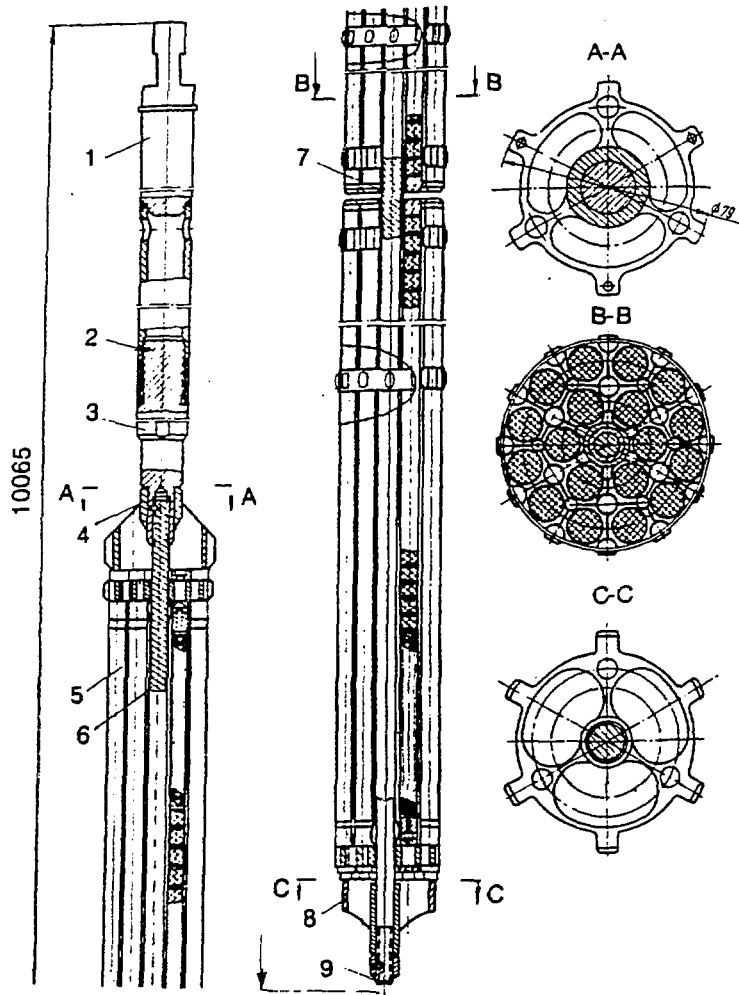   reactor core.
    In our PWR plants, use is made for the same purpose of
    fixed burnable poison rod clusters which are introduced
    in an adequate number of fuel assemblies, for the
    duration of the first operation campaign.
In the RBMK, quasi immediately after the plant start-up,
absorbers are regularly removed and replaced on-load by fuel
assemblies.
At the time of the accident, 1659 process tubes contained a
fuel element; most of the fuel assemblies (75%) were still
first load bundles (core average burn-up : 10.9 MWd/kg)

179 other tubes parallel to the pressure tubes and distri-
buted amongst them, play a different role; for the same
external diameter (88 mm) and same material, they have a
thinner wall thickness (3 mm) and they are traversed in the
inverse direction, from top to bottom, by water at a lower
temperature (40 to 70°C) to cool the control and safety rods
which move in the tubes.
These tubes are also surrounded by graphite rings for heat
conduction between the graphite block and the tube walls;
these tubes are therefore introducing some colder regions in
the graphite blocks, regions where some Wigner energy could
eventually be stored : this question is not evoked in the
available literature.

10065

A-A

B-B

C-C

A

B

C

1

2

3

4

5

6

7

8

9

1 Hanger
2 Pin
3 Transition piece
4 Stem
5 Fuel element

6 Supporting rod
7 Sleeve
8 Nose piece
9 Nuts

Figure 17 : The fuel assembly

To be complete, I have to mention 12 additional tubes of this last type also cooled by low temperature water to accomodate some in-core detectors , as it will be explained later.

In total, there are thus approximately 1900 channels in the graphite stack :

    - 1,693   (88 x 4 mm)   process tubes for fuel or
                              fixed absorber
    -   179   (88 x 3 mm)   rod accomodating tubes
    -    12   (88 x 3 mm)   in-core instrumentation tubes.
    _____
    - 1,884

The fuel assembly

The reactor fuel (fig. 17) is low enriched (2 % U_{235}) uranium oxide in the form of sintered pellets contained in (Zirconium - 1% Niobium alloy) tubes with the following dimensions :

    external diameter : 13.5   mm
    wall thickness   : 0.9   m
    length   : 3.644 m

Eighteen tubes of this type held by 10 stainless steel gridspacers form a sub-assembly.

Two fuel sub-assemblies maintained by an axial supporting tube form a fuel assembly with approximately 7 m of active length and an overall length of 10.065 m.

The overall diameter of the fuel assembly on the gridspacers is 79 mm; this fuel assembly is designed to fit a fuel channel of 80 mm internal diameter.

The leak tightness of the fuel elements is checked by scanning the gamma activity of the coolant entering the steam separators, using gamma energy spectrometers mounted on a moving platform which can scan each fuel channel outlet tube at the separator.

There is also a sampling system for monitoring the activity
of gaseous fission products in the separated steam in each
drum separator; this system allows to monitor, continuously
but collectively, the condition of the fuel assemblies of a
quarter of the reactor core loading (4 drum separators).

Control and safety rods

The control rods, 179 rods in total, all have the same form
of a hollow cylinder made of 3 coaxial tubes : an inner tube
(external diameter 50 mm, thickness 2 mm) and an outer tube
(external diameter 70 mm, thickness 2 mm) - both made of
aluminium alloy - forming a leak proof annulus in which the
third tube, a boron carbide sleeve (external diameter 65 mm,
absorber thickness 7.5 mm) is contained.
These 179 rods are divided functionally into four groups :
 - 146 manual control rods
   - 57 of which are safety rods used for emergency
     protection; during the reactor operation, they are
     fully withdrawn; the fall time for complete
     insertion into the core is 20 sec.
   - 89 control rods, which are used for the control of
     the rate of power increase and for the control of
     the power distribution, essentially radial profile
     control.
 - 12 automatic regulating rods for controlling the
   reactor average power.
The rods of these first three functional groups have an
overall length of 6.170 m; they are made of six absorber
elements of approximately one meter length; these rods are
withdrawn from the core upwards.

The fourth functional group is made up of
 - 21 short manual control rods for controlling the axial
   power distribution; they consist of three absorbing
   elements and have an overall length of 3.050 m.
   They are withdrawn from the core downwards.

At the end of each rod of the four groups, is a number of elements which do not contain neutron absorbers (rod followers) to prevent water occupying the space vacated by the rod; these water-displacer elements prevent neutron-absorption by the water reducing the effective reactivity controlled by the rod.

Neutron detection

Besides a classical neutron detection system made up of ionisation chambers and fission chambers surrounding the reactor, there is an in-core nuclear instrumentation system allowing a three dimensional monitoring of the neutron distribution.
This system is based on self-powered neutron detectors, silver being used as beta-emitter.

The radial monitoring is done by 130 detectors mounted in the central thimble, the central structural tube of 130 fuel assemblies distributed all over the reactor core loading. The axial monitoring is done by 84 detectors in 12 groups of seven detectors; each group is distributed over the height of the core in 12 channels of the same design as the control rod channels; these channels are distributed over the central region of the core.

The reactor circulating system

The cooling water at a temperature of 270°C and under a pressure of 79.6 kgf/cm^2 is distributed (figure 18) at the bottom of each reactor process channel; it is transformed into a steam-water mixture leaving the reactor with a mean mass steam quality of 14.5%, a mean outlet temperature of 284°C and a pressure of 75.3 kgf/cm^2.
This steam-water mixture flows through the top of the channel and the steam-water communication line to the four drum separators.
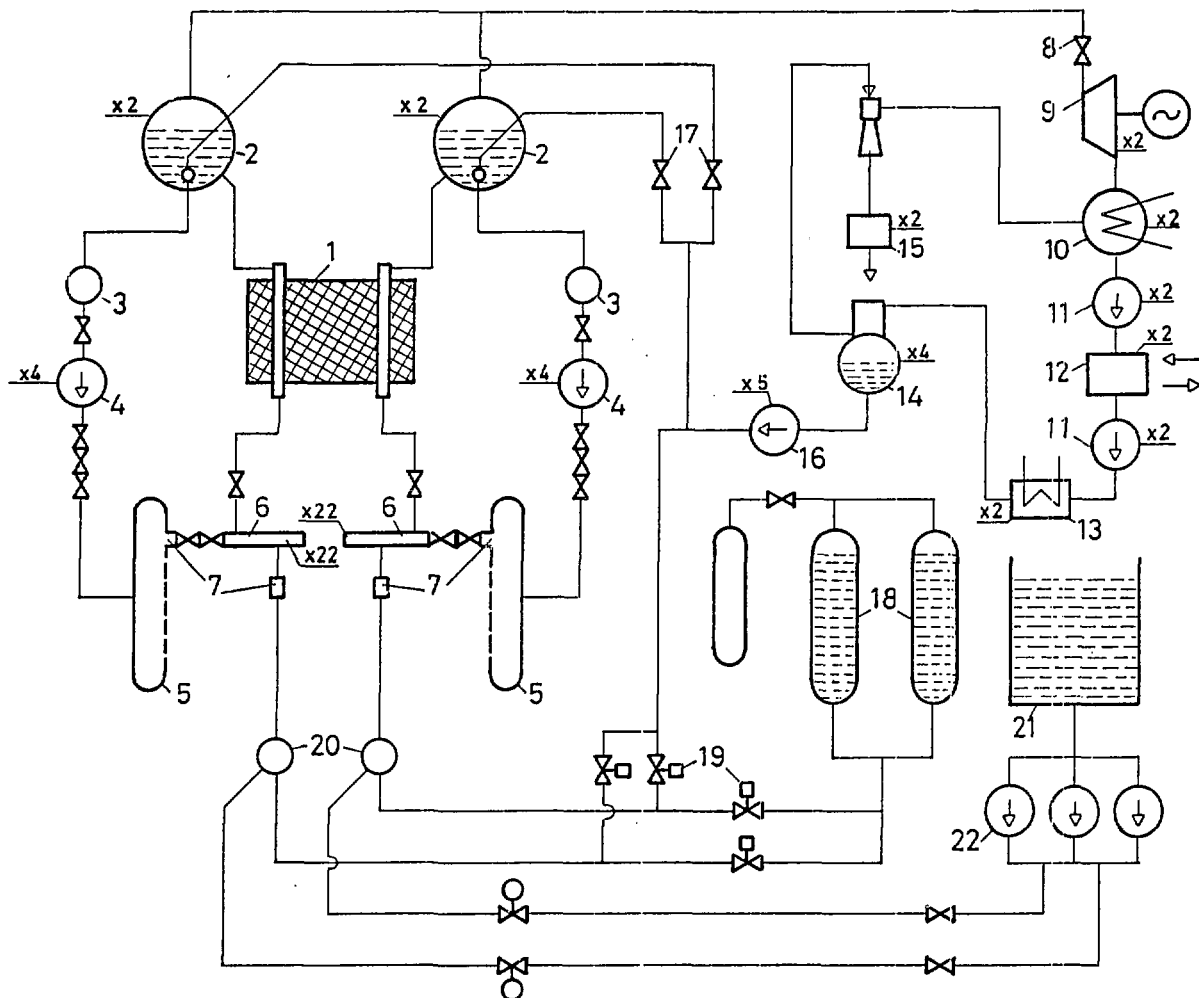
RBMK - 1000

Schematic diagram of main circulation system

1. Reactor
2. Steam separator/drier
3. Feedwater header
4. Main coolant pump
5. Pressure header
6. Distribution group header
7. Leak limiters (restriction nozzles)
8. Pressure regulator
9. Turbo-generator
10. Condenser
11. Condensate pump

12. Condensate purification
13. Pre-heater
14. Degasifier
15. Apparatus for burning explosive mixture
16. Feedwater pump
17. Level controller
18. ECCS water accumulation unit
19. Quick-acting ECCS valve
20. ECCS header
21. Water stock in condensation device
22. ECCS pump

Figure 18

The dry steam (less than 0.1% moisture content) separated in each drum is fed at a pressure of 70 atm. via two steam pipes (8 pipes in total) to the two turbines.
For the nominal power of 3200 MWth, the reactor produces 5,400 tons of vapour per hour.

There are two separators (SS cladded mild steel) per loop; four separators per reactor; each separator is a cylinder approximately 30 meters long with a diameter of 2.3 m and a weight of 200 tons; the rupture of the separators is not considered in the safety studies; it is basically the same philosophy as for the PWR plants pressure vessel where the quality standards and inspection practices are such that a catastrophic rupture is not considered as credible.

With a temperature reduced to 70°C at the turbine outlet, the vapour is condensed; an independent cooling system evacuates the heat losses into the environment via wet refrigeration towers.
The condensate is returned to the steam separators by feedwater pumps where, after mixing with the recirculation water, it is fed through the downcomers to the suction headers of the main circulation pumps; 37,500 tons of water are circulated in one hour.

At the pump outlet, the water enters the common header; there is one common header per loop, two per reactor.
The maximum design-base accident is the sudden transversal, cross-sectional, rupture of a common header (diam. 900 mm).

From the common header, the water is distributed to 22 distributing group headers (diam. 300 mm) feeding half the reactor process channels. The flow rate through each process channel is determined by means of an isolating and regulating valve in accordance with a flow-meter reading.

A by-pass line, normally closed, between the suction and the discharge of each main circulation pump facilitates natural circulation through the reactor with the main circulating pumps not working.

Both loops of the circulating circuit have a common water clean-up system with mechanical and ion-exchange filters, with 4% of the main flow by-passed for clean-up (160 t/hr).

It is worth mentioning that when the steam shut-off valve of one or both turbines is closed, the steam may be dumped into the condenser via pressure reducing valves.

ECCS

The emergency core cooling system (ECCS) is a protection safety system designed to draw off the residual heat release, after reactor shut-down, through the timely feeding of the appropriate volume of water into the reactor channels, in the event of accidents induced by damage to the core cooling circuit.

The ECCS comprises three independent sub-systems; each sub-system includes a fast-acting part and a part providing prolonged afterheat removal.
The fast acting parts of two ECCS sub-systems take the form of a system of vessels filled with water under nitrogen pressure (10.0 MPa); in our plants, that part of the ECCS system would be called the passive sub-system by opposition to the active sub-system where operation of pumps is required.

The fast acting part of the third ECCS sub-system is of particular interest in the framework of the Chernobyl-4 accident; it is a unit for supplying water from the electric feedpumps; in the event of a loss of coolant accident coinciding with the loss of off-site power, the supply of

ECCS water from the feedpumps is assured for a first period of 40 - 50 sec. by electrical energy delivered to these pumps by the running down turbogenerators; the relay is then taken over by the diesel generators which in the meantime have been started.

The prolonged afterheat removal part of the ECCS system comes into operation after the moment at which the fast-acting part of the ECCS ceases to operate.
A redundant system of pumps and heat exchangers is designed to ensure a supply of water at a rate of approximately 500 t/h; the water is drawn by the pumps from the pressure suppression pools.

Emergency condenser system

The bubbler ponds or suppression pools (fig. 19) are located directly beneath the reactor shaft and the adjacent compart-ments of the multiple forced circulation loop.
Two ponds are superposed : a volume of water intended to condense the steam released from a large pipe break is stored at the two levels.
The various compartments are connected to each other and to the suppression pools by valves of different types; basically by bubbling through the water layer, the released steam is condensed.
Each pond is equipped with a spray system and a circulation system on heat exchangers intended to condense the steam and to cool the water, so that the residual heat can be withdrawn through the bubbler pond until the reactor is cooled down completely.

In an emergency situation, the volume of an undamaged compartment can also be used through opening the connecting valves, to reduce pressure in the damaged compartment and in the air-space of the condensation system.
As already explained, the bubbler ponds may also be used as a (third) source of water, for the emergency core cooling system.
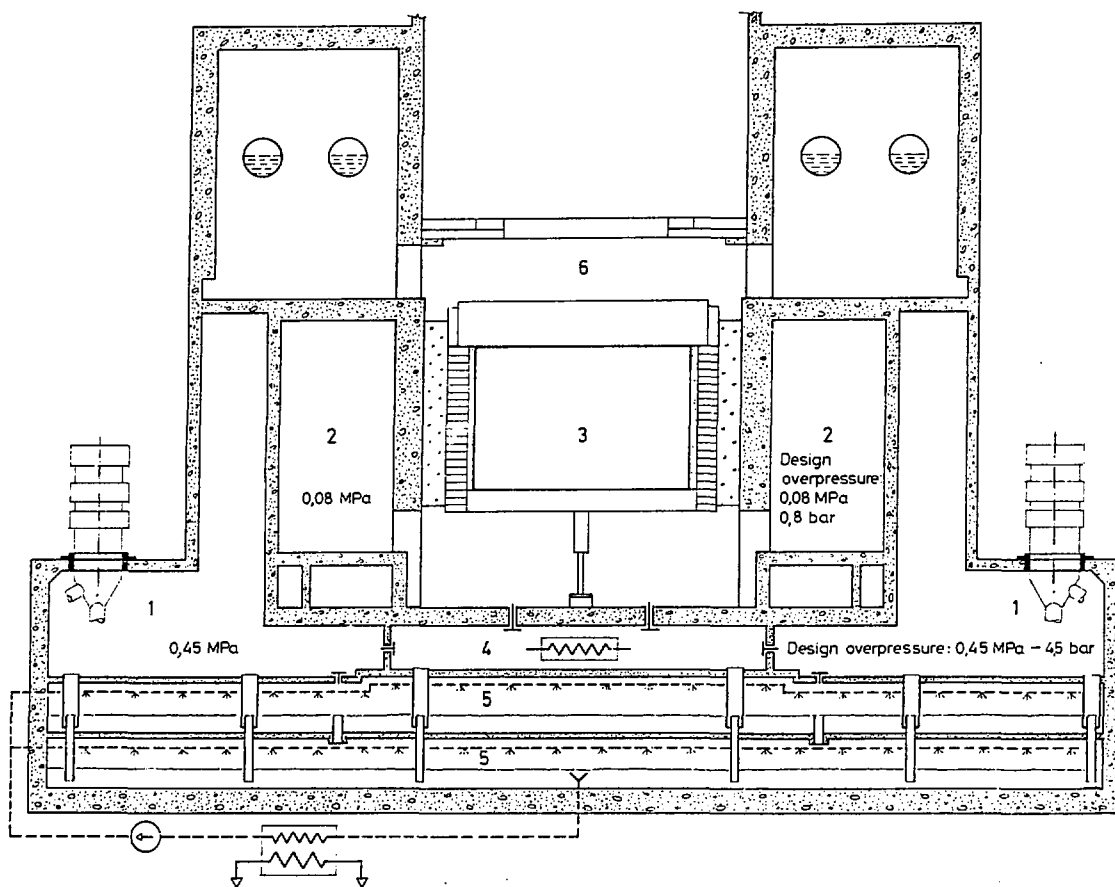
PRIMARY AND SECONDARY CONTAINMENT LAYOUT

1 PUMP COMPARTMENTS
2 GROUP HEADER COMPARTMENTS
3 STEEL REACTOR VAULT
4 SURFACE CONDENSER TUNNEL
5 SUPPRESSION POOLS
6 STEAM SEPARATOR COMPARTMENT (SECONDARY CONTAINMENT)

Figure 19 : The RBMK emergency condenser system (suppression pools)

The two ponds will also, in the course of the accident sequence of events, play another role, not foreseen by the plant designers, in the frame of the emergency measures taken with the objective of limiting the accident consequences.

Refueling machine

The RBMK reactors are reloaded on-load by a special refueling machine which can be coupled to each reactor fuel channel.
In 24 hours, the refueling machine can carry out five operations of fuel channel unloading and reloading with the reactor in power and not less than 10 such operations in a shut down reactor.

The refueling machine is attached to the channel to be refueled; a pressure equal to that in the fuel channel is created in a machine inner chamber and the channel is unsealed; water at 30°C is pumped from the chamber into the channel to prevent the penetration of steam and hot water from the channel into the refueling machine. After removal of the spent fuel element, the channel is sealed and the pressure in the inner chamber reduced to atmospheric pressure. The machine is disconnected from the channel and sent to the location where spent assemblies are stored.
The height required for the handling of very long pieces (more than 10 m for a fuel assembly) is high and the roof of the reactor hall is situated at more than 30 meters above the working platform.
The refueling machine has a weight of 25 tons and rolls on two tracks which are part of the hall lateral walls; an area in the hall is intended for parking the machine in the periods between refueling operations.
This hall is also equipped with a conventional crane for the handling of heavy pieces of material; this crane moves over the refueling machine.

Main data for RBMK 1000

This table summarizes the main data of the Chernobyl-4 characteristics.

Reactor system

Gross electrical output :	1000. MW
Gross fission heat :	3200 MW
Active core diameter	12 m
Active core height	7 m
Reflector diameter	14 m
Reflector height	8 m
Channel pitch	250 mm (square)
Number of fuel channels	1693 (max)
Channel tube material	Zr/2.5% Nb alloy
Channel tube external diameter	88 mm
Channel tube wall thickness	4 mm
Number of control rods	179

Fuel

Enrichment	2 wt%
Number of sub-assemblies per fuel assembly	2
Number of fuel pins per sub-assembly	18
Cladding material	Zr/1% Nb alloy
Pin outer diameter	13.5 mm
Cladding thickness	0.9 mm (min.:0.825 mm)
Active fuel length	3.490 mm
Pin overall length	3.644 mm
Spacer SS grids	10
Fuel assembly overall diameter	79 mm
Mass of uranium per fuel assembly	115 kg
Weight of uranium in core	$\sim$ 200 t (max)
Mean unloading fuel burn-up	22.3 MWd/kg
Isotopic composition of unloaded fuel (at 20 MWd/kg)	
U_{235}	4.5 kg/t
U_{236}	2.4 kg/t
Pu_{239}	2.6 kg/t
Pu_{240}	1.8 kg/t
Pu_{241}	0.5 kg/t

Power

Limiting theoretical channel power	3.250 kW
Peak heat flux	83 W/cm^2
Peak linear thermal power	360 - 385 W/cm
Peak fuel temperature	2100°C
Peak clad temperature, outer/inner	295/325°C

Coolant

Coolant flow	37,600 t/h
Mean steam content at reactor outlet	14.5 wt%
Max. steam content at channel outlet	20.1 wt%
Channel inlet temperature	270°C
Channel inlet pressure	79.6 kgf/cm^2
Channel outlet temperature	284°C
Channel outlet pressure	75.3 kg/cm^2
Steam pressure in the separators	70 kgf/cm^2
Steam flow at nominal rating	5,400 t/h

Coolant circuit

Number of loops	2
Number of steam drums	4
Drum diameter	2.3 m
Drum length	30.7 m
Drum wall thickness	104 mm (SS clad)
Drum weight	200 t
Number of main circulation pumps	8
Pump flow per unit	7,000 m^3/h
Number of common headers	2
Common header diameter	900 mm
Number of distributing group headers	22 per loop

SOME COMPARISON ELEMENTS

After the plant description, it seems interesting to compare
the Chernobyl-4 plant or the modern RBMK reactors with other
well known reactor systems used in the Western countries; a
kind of critical comparison.
This comparison will be focused on the following six items :
- Containment
- Pressure tube
- Positive void coefficient
- Reloading mode
- Hydrogen production, dangerous chemical reactions
- Similar general lay-out (SGHWR) and similar concept
  (Hanford reactor).

Containment

The absence of a containment building enclosing the whole
reactor, like for instance in the case of our PWR electric
plants, is the short-coming of the RBMK reactors most widely
discussed.

The Soviet design

The RBMK reactors are enclosed in a set of massive concrete
cells which can be considered as a partial containment
system.
These cells are designed to localize and contain radioactive
emissions in accidents involving loss of integrity in any of
the pipes of the reactor coolant circuit, with the exception
of (I quote the Soviet report) :
- the pipework of the steam-water communication lines
- the upper part of the fuel channels (and closure plugs)
- that part of the downcomers which is situated in the drum
  separators compartment.
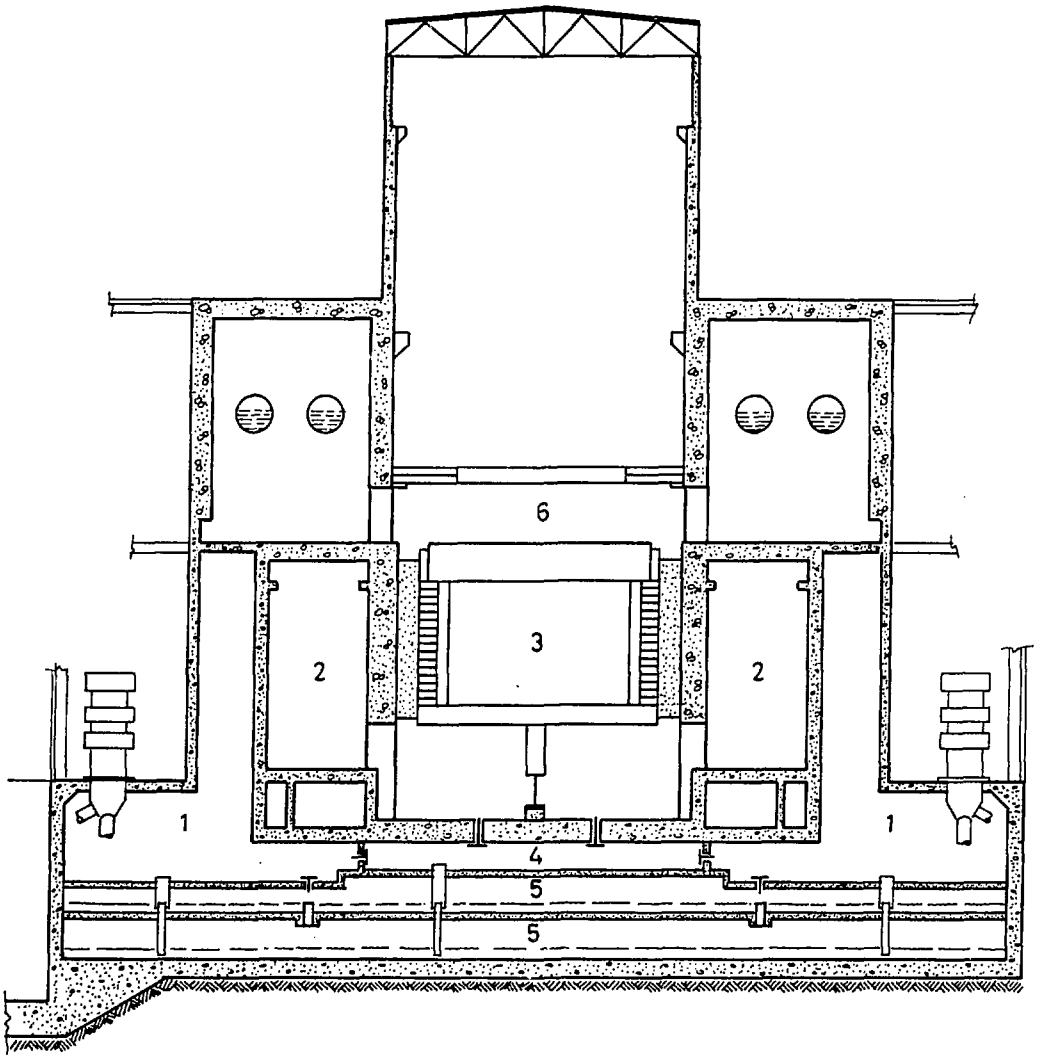
1   PUMP COMPARTMENTS

2   GROUP HEADER COMPARTMENTS

3   STEEL   REACTOR VAULT

4   SURFACE CONDENSER   TUNNEL

5   SUPPRESSION   POOLS

6   STEAM SEPARATOR   COMPARTMENT
        (Secondary   containment)

Figure 20 : The RBMK partial containment system

That makes a lot of exceptions : all these equipments are
situated in the upper compartment which is named by the
Soviets (fig. 20) "Steam separator compartment (secondary
containment)".

In my opinion, there is no justification for considering
this compartment containing all the closure plugs of the
fuel channels, as part of a plant "secondary" containment.

This compartment is effectively part of a tall building with
concrete walls and a fairly conventional type of roof; the
ventilation system is designed to maintain this area at a
slightly reduced pressure to encourage inward leakage.

This structure will not withstand more than about 2 psi
overpressure nor the effects of fire. This type of cons-
truction also does not offer special protection against
outside impacts.

Elements of comparison

The nearest equivalent to the RBMK reactor buildings are
those for the gas-cooled reactors developed in the U.K. and
in France and also used in Italy, Japan and Spain.

This kind of reactor also has a building of conventional
construction over the working area at the top of the reactor
where refueling can be carried out while the plant is on
line.

Each of these reactors also has a huge mass of graphite
moderator in the core.

But there is a fundamental difference between a reactor
cooled by water or by boiling water - which flashes to steam
in the event of a depressurization - and a gas-cooled
reactor where depressurization causes a loss of cooling
efficiency but where some cooling is possible right down to
athmospheric pressure.

In addition, the graphite moderator in a gas-cooled reactor
is run at a much lower temperature - no more than 400°C
compared to 700/750°C at Chernobyl - and acts as a heat
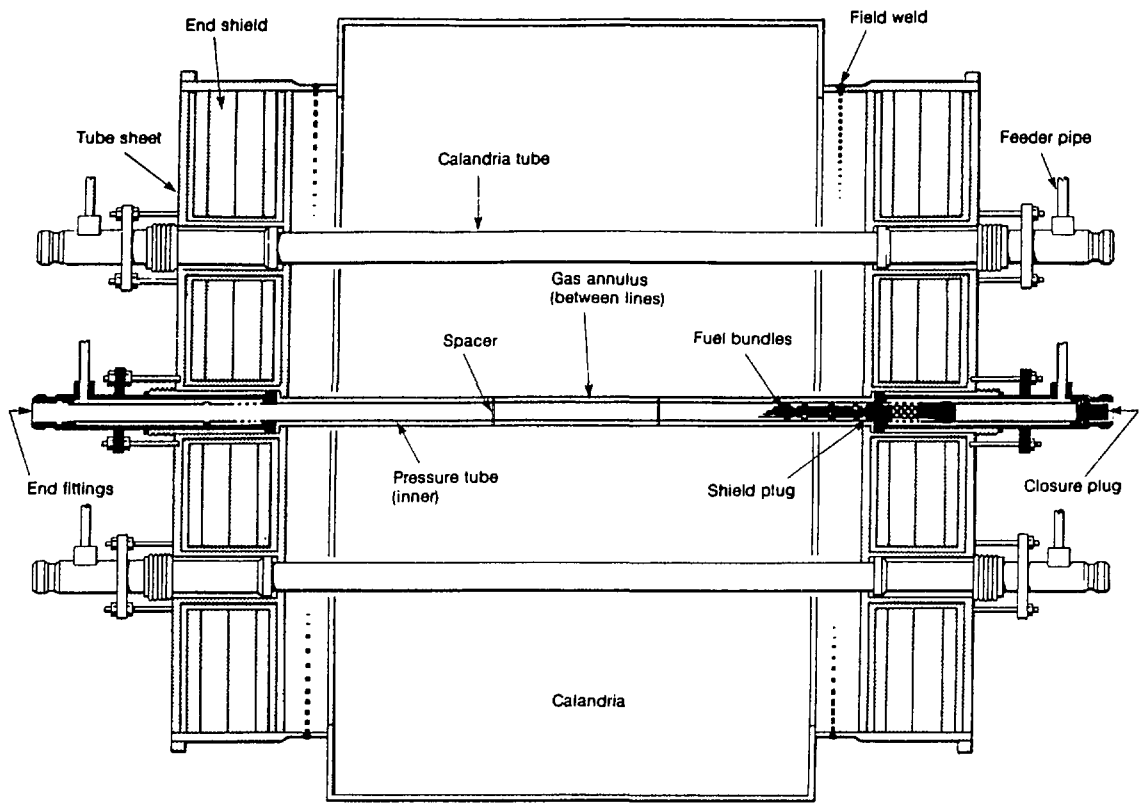sink, delaying heat-up of the fuel to the melting point of

Figure 21 : Cross-sectional schematic view of a CANDU reactor

the cladding by several hours, even in the event of total depressurization.
Nevertheless, those gas-cooled reactor buildings also do not offer protection against outside impacts.
The Belgian PWR plant containers are designed to withstand impact by the largest long-carrier aircrafts.

Pressure tubes

We will not argue about the relative merits of pressure tubes or pressure vessels for reactors cooled with high pressure water. Both approaches are valid and able to produce a perfectly satisfactory safety case.
The big advantage for the Soviets of the pressure tube design was that no special large facilities with modern machinery and well established quality assurance programmes were required. In the Canadian literature, the following statement is made : "One inherent advantage of the CANDU pressure tube approach is that there is no specific limit to the size of such a system; there is no fundamental limitation in size up to at least 2000 MWe." The Russians express similar opinions.

We will focus on a major difference between the RBMK and the Canadian CANDU reactor.

As seen in fig. 21, the CANDU reactor has double-walled fuel channels with central pressure tubes surrounded by the calandria tubes of the heavy water moderator tank; the calandria tubes are also designed to resist the full system pressure.
The space between the walls provides a thermal barrier between the hot fuel channel and the cool moderator and also provides a means for leak detection.

In contrast, the RBMK fuel channels are single wall channels and the pressure tubes have tight fitting sleeves of graphite around them, these sleeves being designed to give a good thermal contact with the hot graphite so that the water in the channels can cool the graphite as well as the fuel.
As already said the graphite temperature is very high (700°C) as a result of the energy dissipated in the graphite by the neutron reactions (slowing-down and absorption) and also by gamma interactions and of course as a result of this relatively poor method for cooling the graphite block.

Failure of the pressure tubes therefore poses the problem of the production of an explosive mixture of hydrogen and carbon monoxide from the well known reaction of steam with hot carbon.
The system for monitoring the integrity of these pressure tubes is based on changes in the relative moisture content and in the temperature of the nitrogen/helium mixture filling the casing containing the graphite masonry; the main role of this cover gas is to prevent the oxidation of the graphite; this gas mixture circulates in a closed circuit - which maybe provides also some additional cooling of the graphite - a closed circuit in which water vapour and graphite-oxidation products are separated and the desired composition of the mixture is maintained.
It is difficult if not impossible to imagine how it is possible with such an integral system to localize, to identify a leaky pressure tube.

But between the CANDU and the RBMK reactors there is a large similarity on what seems to be a weak point in the design of the pressure tube : the upper and lower sections are made of stainless steel while the middle section lying in the active part of the reactor is made of a Zr-Nb alloy; these two metals can not be joined easily. Fig. 16 shows the special construction mode used to join the upper and lower sections to the middle one; these two junctions are situated near the external upper and lower boundaries of the graphite moderator stack.

It is also mentioned by some authors that the nitrogen in the moderator gas cover causes the channel tube external wall to become nitrided, leading to considerable changes in both mechanical properties and resistance to corrosion by moisture in the graphite matrix.

In conclusion of this short discussion on fuel channels, it can be said that the Russians have adopted solutions which seem to be less satisfactory than those retained by the Canadians.

For what concerns the high temperature of the graphite masonry in RBMK, due to the absence of a separate and efficient cooling system (comparison with Hanford N reactor is interesting from this point of view) I think it judicious to say that this characteristic forms with the large positive void coefficient and associated shut-down margin problematics, the two main weaknesses of the reactor design. The fact that at this working temperature, Wigner energy is not stored is a very small advantage amongst many big disadvantages.

Positive void coefficient

The combination of an efficient moderator, like carbon, and boiling water coolant produces a positive void coefficient of reactivity. A positive void coefficient creates significant stability problems; if the temperature rises, the boiling increases, there is less absorbing light water in the reactor; that produces a reactivity increase resulting in a rise in reactor power and temperature; the reactor is unstable.

In a stable reactor, reactivity drops as temperature rises.

A positive void coefficient is not, in itself, impossible to control; in a large core, such as that of the RBMK, this control implies a very good in-core instrumentation and a

computer control system to be able to intervene without delay on every local disturbance of the power level, to limit drastically the amplitude of any such disturbance.
But in such large cores, a good and fast spatial control of the power is not easy to realize; the computer systems and the instrumentation in the Soviet Union have developed considerably in recent years but their performances seem to be still poor in comparison with those of the equipment available to the Western plant operators.

The Canadian CANDU reactors operate with a positive void coefficient but with a lower amplitude, using hot heavy water in the process channel, cool heavy water as moderator and natural uranium fuel; nevertheless it is for that reason that the Canadians were the first to introduce direct on-line computer control of their reactors to achieve good spatial control of power in the cores of their reactors.

During a short first period of an operating campaign, a freshly reloaded PWR reactor core can also be characterized by a positive moderator temperature coefficient and a positive void coefficient when the boron (boric acid) content of the water coolant is high to compensate for the reactivity investment at the beginning of the core life. In that situation, a reduction of the water density by temperature increase or by some nucleate boiling appearing at the reactor hottest points has a positive reactivity effect, the reduction of the neutron absorption by the boronated water density reduction exceeding the negative effect on the reactivity of the reduction of the neutron slowing-down power of the reactor lattice inherent to the reduction in hydrogen atoms partial density. This situation disappears quickly with the core poisoning and with the fuel burn-up.
A method broadly applied in our PWR's to avoid this situation consists in mixing with the fuel a burnable poison (Gd_2O_3) which, at the beginning of an operating campaign,

compensates part of the reactivity investment allowing to reduce the water boron concentration; being less neutron absorbing, a reduction of the water density has a negative effect on the reactivity; being "burnable", i.e. converted by a neutron capture in a non-absorbing atom, the burnable poison atoms disappear progressively as the fuel burn-up proceeds and the corresponding reactivity investment is made available.

These references to the CANDU and even to the PWR reactor show that the problem of a positive void coefficient is not specific to the RBMK design but that techniques and methods have to be applied to eliminate this situation or to operate safely even with a small positive value for this coefficient.

As it will be outlined in the third part of this communication, the Russian engineers were well aware of this problem and had already improved the situation but most certainly in an insufficient degree, as the economy of the reactor operation was adversely influenced by the correcting measures.

Reloading mode

Semi-continuous refueling with the unit on line is a solution adopted in many plants.
All the Magnox gas-cooled reactors are refueled on-line; it is the same situation for the CANDU reactors.
Various systems are used; for reactors with vertical channels, bottom refueling or top refueling like in RBMK, is adopted; in the case of the CANDU reactors with horizontal fuel channels, two refueling machines are working in parallel as they are coupled to each end of a channel during the refueling operation.

Criticism has been formulated against this system on the occasion of the Chernobyl accident saying that the system implies a frequent agression on the pressure barrier.
I think that this criticism is unfounded; the on-line refueling also has advantages; I think the most relevant are:

1. A close control of the core reactivity : the core reactivity investment can always be kept as low as possible.
2. A leaky fuel assembly, when identified - which is particularly easy in a pressure tube reactor - can be removed and taken out of the core at an early stage of its sickness; that avoids reaching a high contamination level in the coolant system with the associated radioactive effluents problems.
3. A very efficient utilization of the fuel.

Hydrogen production

Chemical reactions susceptible to produce dangerous gases in the development of an incidental or accidental situation can be at the origin of a catastrophic situation.
The contribution of a gas mixture (H_2/CO) to the Chernobyl disaster is not completely clear, although the Soviets seem to consider as well established that the second explosion was a chemical explosion.
The contribution of each of the two reactions Water/Carbon and Water/Zirconium to the formation of the H_2/CO mixture is not known, at least up to now and it will probably remain extremely hasardous to delineate the exact role of each one.

It seems interesting to elaborate a little on a comparison of the potentialities for H_2 formation offered by a RBMK and by a modern PWR.

The era of Stainless Steel clad PWR is not so far in the past and the French-Belgian plant of Chooz-A is still SS-clad.

In parallel with the efforts made by most of the fuel
manufacturers to offer "all zircaloy fuel assemblies",
replacing the Inconel grids by Zircaloy ones, it is
interesting to note that French and German designers foresee
a return to stainless steel cladding for the Advanced
Pressurized Water Reactors forming the next generation of
the PWRs, and that, for a lot of reasons which would take
too long to develop here but have fundamentally nothing to
do with the worry of avoiding H_2 formation risk.

About the comparison between the potential risk of
generating H_2 in a Zircaloy clad fueled reactor, it has been
said that the Russians used such a large Zirconium inventory
in their RBMK's that the situation in a modern PWR is very
different.

The comparison of the Zirconium inventory in Chernobyl-4 and
in Doel-4 or Tihange-3 is as follows :

Chernobyl	Doel 4 or Tihange 3 With AFA fuel assemblies) (all zircaloy)
Pressure tubes (Zr, 2.5% Nb) : 100 t Fuel cladding (Zr,1% Nb) : 58 t	Zr-4 : 27 t
158 tons	27 tons

Indeed Chernobyl 4 contained approximately 6 times more Zir-
conium than our most modern PWR plants.
But it must be considered that the complete oxidation of 1 kg
of Zirconium produces half a cubic meter of hydrogen gas, at
normal temperature and pressure (1 m^3 H_2 $\div$ 2 kg Zr).
In both cases (RBMK and PWR), the Zirconium inventory is
large enough to avoid any discussion on the comparison of
the relative levels of risk.

The main reason for using Zirconium in the PWR plants is a better neutron economy, reducing the price of the energy produced ; a second and important aspect of shifting from SS to Zr is the permeability of the SS cladding to the tritium formed in the fuel as a fission product. Zirconium has therefore a beneficial influence on the radioactive effluents of the plant.

But considering the risk associated to the use of this "hydrogen producer" (remember the situation at Three Mile Island where a large H_2 explosion in the plant container was feared, leading to a partial evacuation of the local population) I think that presenting this hydrogen producer as an interesting tritium barrier is as wise as advising you to walk along the middle of the road to avoid twisting your ankles walking on the grass verges.

Many systems (spark-plugs, hydrogen removal system and recombiners, ...) have been developed to avoid reaching conditions for a hydrogen explosive reaction (deflagration or detonation) in a PWR plant container. But it remains of course difficult to guarantee the efficiency of these systems in real accidental conditions; on the other hand, the resistance offered by the reactor plant container to the shock-wave associated to a large hydrogen detonation has a technical limit, and it can not be ascertained that conditions leading to a plant container tearing can not be encountered.

With a stainless steel cladding, the problems associated with H_2 formation in accidental conditions will not be completely avoided as it is well known that H_2 could also be formed by oxidation of very hot steel by steam but this reaction takes place at a higher temperature than the zirconium/water reaction and is therefore less preoccupant.

Work is in progress by all PWR designers on reactor containment concepts and new and more satisfactory designs will most probably be proposed and implemented in the future.

Similar general lay-out and similar reactor concept

SGHWR

The general lay-out and more particularly the coolant circuit of the British steam generating heavy water moderated reactor (SGHWR) is very similar to the Soviet steam generating graphite moderated reactor (fig. 22).
The big difference lies in the fact that the moderator is heavy water cooled in a tank.

Two prototypes operate according to this drawing : one in the U.K. at Winfrith (92 MWe) and one in Fugen, Japan with 148 MWe.

It is interesting to note that during the process of developing the project of a commercial unit based on this formula, the U.K. industry had been interested by the Soviet realizations and had expressed some safety concerns after paying visits to the Leningrad RBMK plant.
The British finally decided to abandon their own project because they came to the conclusion that the cost of engineering to reach satisfactory safety standards was too high and that therefore building such a plant would be uneconomic.
The difficulty was namely to realize an ECCS system that could cool all of the fuel channels effectively in all postulated accident conditions.

According to the SGHWR team, the Russians took calculated risks in deciding not to add emergency spray water cooling for the fuel elements nor a secondary shut-down system.
The Russian engineers said that a fracture of the steam drums or of the large bore coolant pipes was considered incredible and therefore there was no necessity to consider consequences.
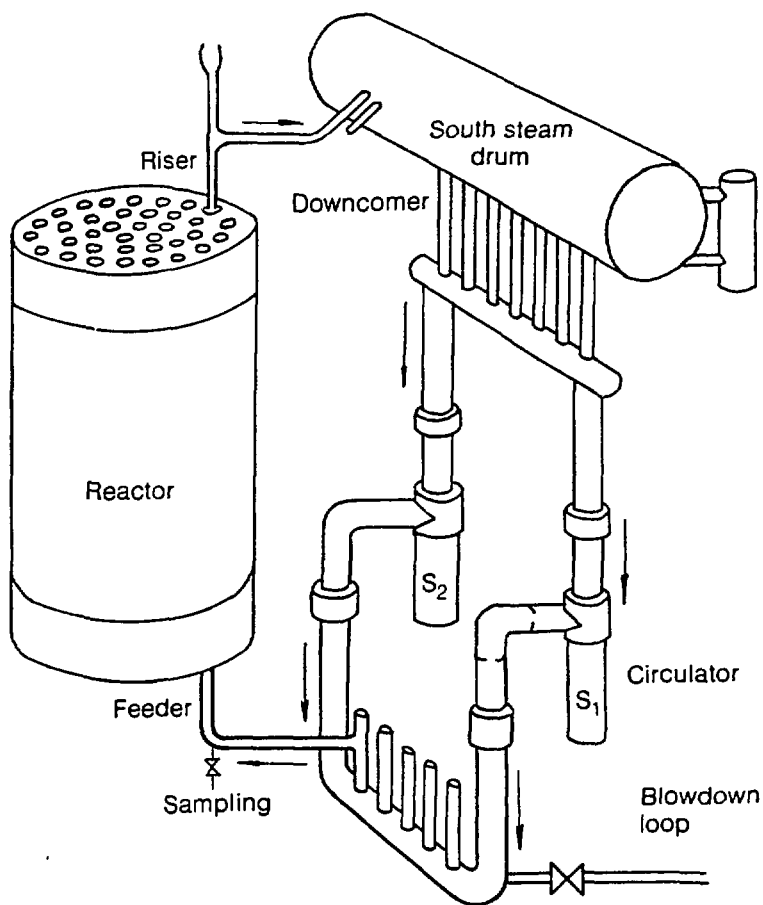
Figure 22 : General lay-out of the SGHWR coolant circuit

It is in such accidental conditions that an emergency spray system would become necessary, an important part of the pressure tubes - up to the half of these tubes - would be no longer fed by water.

On the other hand, the transverse and abrupt rupture of a 900 mm diameter header has been taken as design basis accident to calculate the concrete wall thickness for the isolation cells of the partial containment system and to calculate the system of the two suppression pools designed to condense the steam released by such a large pipe break; but no provisions have been made to cope with the core cooling crisis in such a large accident. This attitude seems to be a little inconsistent.

It remains interesting to note that even with D_2O as moderator, the British industry has decided not to extrapolate its Winfrith prototype experience.

N Reactor

The most interesting reactor to compare with Chernobyl-4 is the so-called N reactor at Hanford, Washington State in the U.S.A.
There is not so much published about this reactor which is used for the production of "nuclear materials", more frankly plutonium for military use; nevertheless a comparison limited to the main characteristics of both plants is, I think, interesting and, more particularly, raises the question why use graphite and boiling water to produce plutonium for the same utilization.
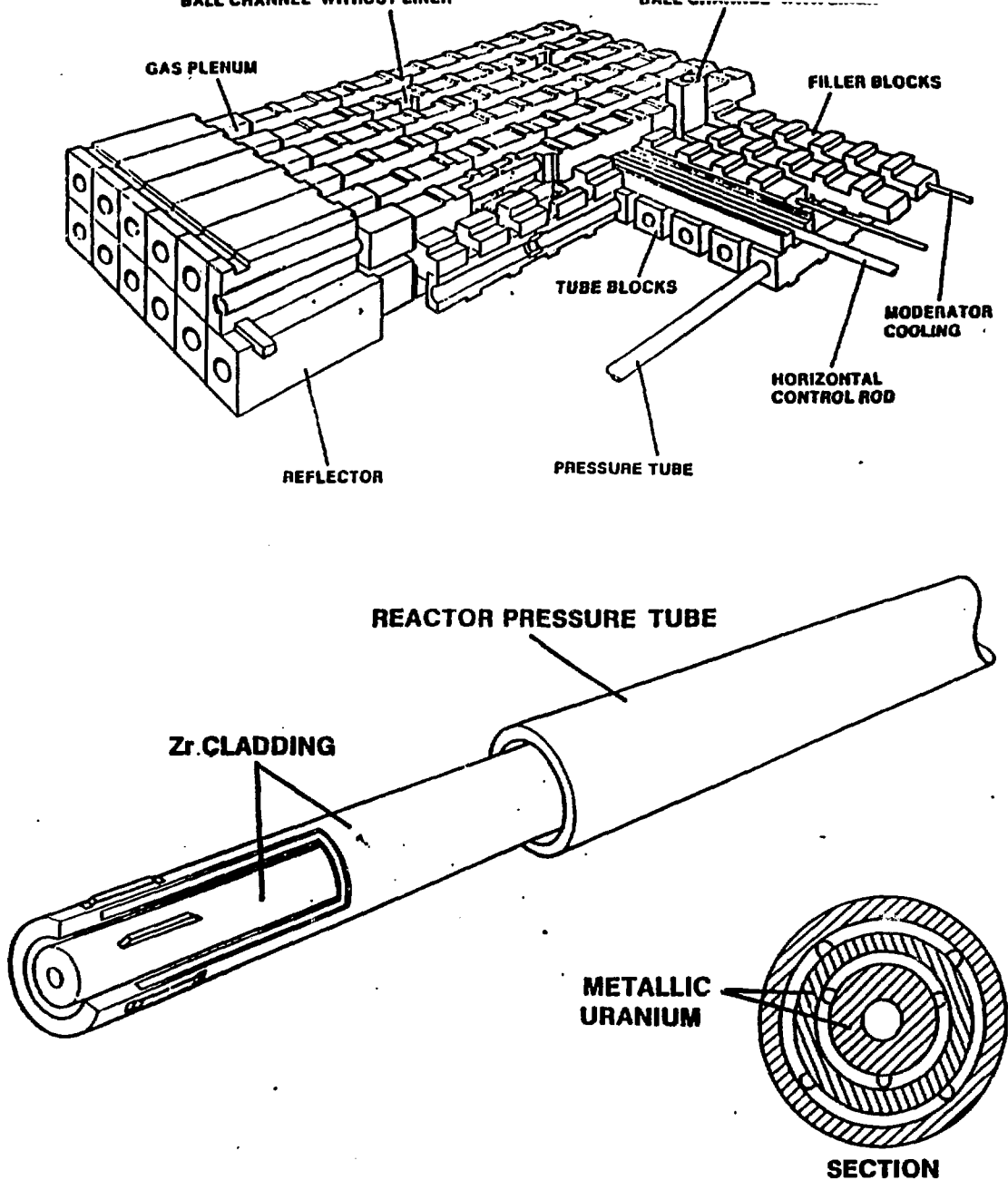
THE HANFORD N REACTOR

Slightly enriched uranium
Graphite moderated and reflected
Boiling water cooled

Figure 23   4000 MW th

First, the comparison :

Fig. 23 gives some data already showing the similarity as to the materials used and as to the plant size.

I have calculated 1900 tons of graphite at a density of 1.67 g/cm^3 for Chernobyl-4 and I found 1800 tons of graphite at 1.7 g/cm^3 for the N reactor.

The graphite reactor core is penetrated by 1003 zircaloy pressure tubes which contain 0.94% enriched uranium metal also zircaloy clad; each fuel element is also made up of two sub-assemblies; each pressure tube contains 16 fuel elements; the piping arrangement for distribution of the water to each process tube is very similar to Chernobyl with supply header, etc.

As to the similarities: the reactor is not situated in a tight containment building but is surrounded by an exclusion area of 16 km diameter (to be compared with the 60 km diameter of the evacuated zone at Chernobyl).

As to the differences : they are important; I noted the following ones as relevant :

- the graphite masonry is cooled by an independent and probably efficient water cooling system. I don't know the graphite operating temperature. The blanket gas on the graphite is helium.
- the reactor has two independent emergency shutdown systems :
    - 84 horizontal hydraulic driven control rods used for the control of the reactor but providing also rapid emergency shutdown capability
    - a completely independent emergency shutdown system provided by a neutron absorbing ball safety system; a total of 106 vertical channels penetrate the reactor and can be filled by these balls when activated by an automatic trip circuit.
- the reactor is off-load refueled.

- the boiling primary water is not driven to the turbine; it flows through heat exchangers where it is cooled and pumped back to the reactor core. The primary circuit is divided into 6 loops with 6 primary pumps and 12 heat exchangers.

    The primary coolant loop is pressurized to 1600 psi (110 bars) under full flow conditions.

I think it can be concluded from this elementary comparison, that the Americans have avoided a lot - but not all - of the weaknesses of the Soviet RBMK design.

Second, the question :
Why use carbon moderated and boiling light water cooled reactor to produce plutonium?

The natural descendants of the first U.S. plutonium producer, the X-10 reactor, are most certainly the (French and U.K.) gas-graphite reactors.
Why this shift from air or gas to boiling light water?
The reason is certainly not a neutronic one; on the contrary, an over-moderated reactor is a bad Pu producer when compared to a sub-moderated reactor, a poorly thermalized spectrum favouring neutron capture in the U_{238} resonances.
Of course, you have to use natural or slightly enriched uranium and the good recipe for the production of a plutonium which is poor in Pu superior isotopes, which the military authorities do not like, is to unload and reprocess your fuel at low burn-up; an on-load refueling system is therefore interesting from this point of view.
But you can manage that as well with gas as with water as coolant.

The big difference and the large benefit of going from gas to boiling water lies in the power density.

In a Magnox reactor, the peak power density is in the range
of 2.2 to 3.2 kW/kg of fuel, according to the literature.
At Chernobyl, it is easy to calculate with a peak linear
rating of 385 W/cm, a peak power density of the order of
35 kW/kg of fuel.
So, in first approximation with water, from the same kg of
fuel, 10 times more fissions, 10 times more captures and 10
times more plutonium are obtained in the same time.
Moderation differences is a second order difference towards
this factor of 10.
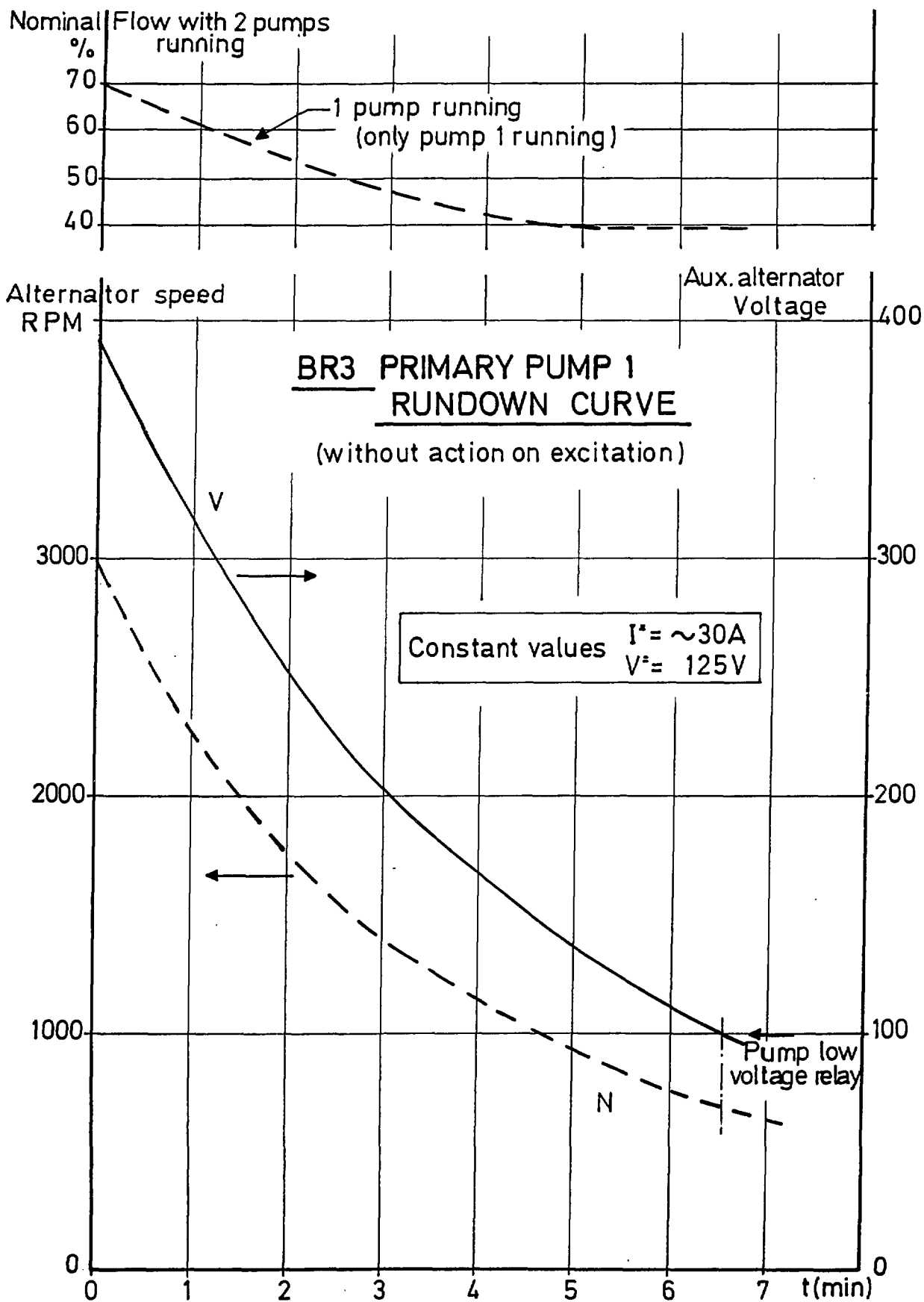It is not the neutronics which governs the choice but the
thermodynamics.

Figure 24

# DISCUSSION OF THE CAUSES OF THE ACCIDENT

The events

A reactor shut-down for a maintenance period was planned for April 25, 1986. Before shut-down, tests were to be carried out on turbogenerator nr. 8 to demonstrate that, in the event of a turbogenerator disconnection with simultaneous loss of off-site power, the mechanical energy of the turbine rotor could contribute to auxiliary electricity supply, before the start-up of the diesel generators. This is a technique used in many plants to provide power to various vital systems during the first instants of an emergency situation.

At BR3, figure 24, following the same line of thought, an auxiliary generator situated at the extremity of the turbo-group shaft provides the electrical power to one of the two main circulation pumps in the primary loop in such a way that, in the same circumstances as those considered at Chernobyl (turbogroup cut-off and loss of off-site power) this primary pump remains powered by the group inertia during the first 6/7 minutes after the incident.

At Chernobyl, the energy generated during the turbogroup deceleration, was used for one of the three sub-systems of the fast acting emergency core cooling system (ECCS): the supply of ECCS water by electric feedwater pumps was assured for a period of 45 - 50 sec as these pumps run down in tandem with the turbogenerator.

Previous tests already carried out at Chernobyl had shown that the voltage on the generator busbars falls off long before the mechanical energy of the rotor was exhausted during the run-down. In the tests planned for April 25, the experimenters intended to use a new special voltage regulation system developed to eliminate this problem.

Figure 25

UNIT 4
SCHEMATIC THERMAL POWER TREND
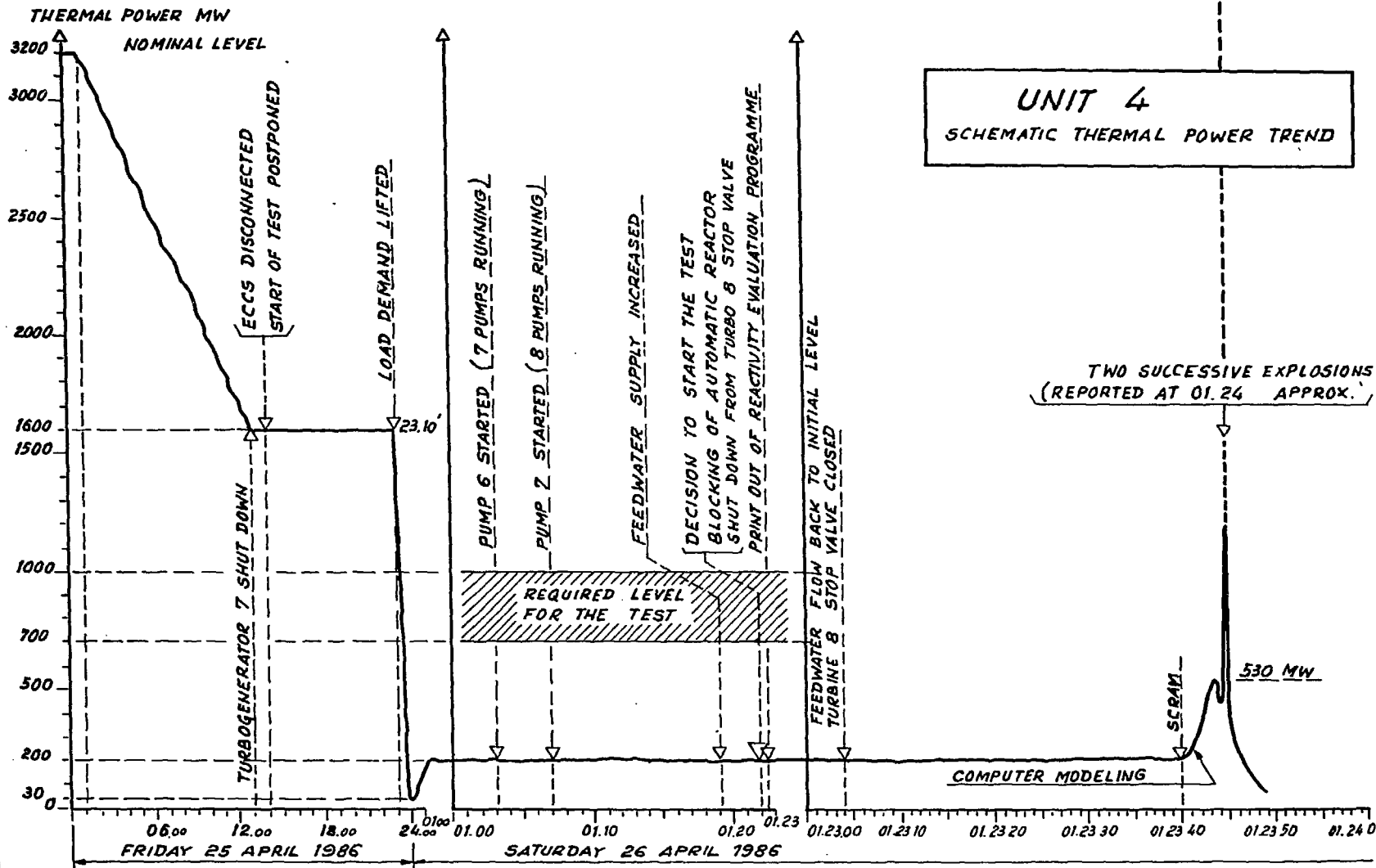
This test was placed under the responsibility of an electrical engineer who was not a specialist on reactor plants; it seems that the quality of the programme prepared was poor and especially that the section on safety measures was drafted in a purely formal way.

The programme made essentially no provision for additional safety measures; the programme called for deactivation of the ECCS so that it would not trip in as the circulation pumps ran down.

In fact, this experiment was not even regarded as safety significant by the station management.

The sequence of events can be reported approximately as follows :

1. On April 25 at exactly 1 a.m. (figure 25) the staff began to reduce the power of the reactor which was operating at nominal power (100% - 3200 MWth), 6 main circulation pumps running (2 pumps in stand-by) - standard situation.

2. At 13.05 hrs, with the reactor developing 1600 MWth, the unwanted turbogenerator (nr. 7) was shut down, turbogenerator nr. 8 being kept running.
   In this situation, four main circulation pumps were connected to the test generator (nr. 8) and the other four to the grid, with still 6 pumps running.

3. At 14.00 hrs, as a preparatory measure for starting the test, the ECCS was isolated, as foreseen.

4. However, the start of the test was then postponed at the request of the local electricity dispatcher.
   The unit then continued to operate and the plant was maintained in an unauthorized state during the following hours, as the ECCS isolation measure was not rescinded; this particular violation did not play any important part in what followed.

5. At 23.10 hrs, after a delay of nine hours, the load demand was lifted; during this delay, the operator's impatience over the test had probably grown considerably; it was Friday night, the beginning of a week-end. The preparation for the test resumed with the power level to be reduced to the required level, between 700 and 1000 MWth.

6. Due to some excessive delay when adjusting an automatic controller system to the required power setpoint, there was an overshoot in the power reduction and the power level fell below 30 MWth.

7. Only at 01.00 hrs on April 26, were the operators able to stabilize the power level back to 200 MWth but it was the maximum they could reach due to the xenon poisoning build-up that had started during the operation at lower power and was still continuing.

    To bring the reactor back to 200 MWth, the operator had pulled far too many of the manual control rods out of the reactor.

    According to the rules, the operation reactivity margin is not allowed to go below 30 rod equivalents without special authorization of the chief engineer of the plant.

    If this margin falls below 15 rod equivalents, nobody can authorize continued operation of the plant.

    The operation reactivity margin, with the reactor power back to 200 MWth, had dropped to between six and eight rod equivalents.

    By "operation reactivity margin" the Russians mean that the available rods which will fall into the reactor on a scram (automatic reactor shut-down) signal produce, during their first second of insertion, a reactivity effect (negative reactivity insertion), equivalent to the full insertion of x rods.

I don't know why they don't speak in dollars per second for instance, as everybody else does; they refer to the reactivity value of a fully inserted rod; a rod, I presume with a "mean" reactivity value i.e. a rod occupying a mean position in the core. A strange idea! (*).

But anyway, at this stage of the test preparation, the reactor safety system reactivity worth was reduced during the first second of insertion into the core to one fifth of the minimum required value without special authorization.

8. In immediate preparation of the test, the operator started at 01.03 hrs the main circulating pump nr. 6 (7 pumps running) and at 01.07 hrs the main circulating pump no. 7 (8 pumps running), so that, at the end of the test, the four pumps connected to the grid would remain available for safe cooling of the reactor, the four pumps connected to the generator having been disconnected from that supply in the normal way when the steam valves on the generator were first shut off, leaving the feedwater pumps drawing energy from the running down turbine.

The utilization of the eight main circulating pumps combined later with the blocking of the automatic reactor shut-down signal from both turbogenerators - we will come back to this point later - was a measure adopted to maintain the reactor on power during the

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(*)
In the Soviet report presented at the Post-Accident Review Meeting, the following information is also given :
- Minimum "weight" of control and protection system rods Δk : 10.5% - This data completes the preceding consideration on the differential reactivity minimum value ($\Delta k/s$) of the rods bank, expressed in rod equivalents per second : in all circumstances, the reactivity associated to the full insertion of all the control and safety rods may not be inferior to 10.5% Δk; if this rods bank falls in a reactor that is just critical, without any other concomitant disturbance of the reactivity balance, k_{eff} would be reduced to 0.895 or less, at the end ($\cong 20$ s) of the rods bank travel.

test, allowing thus the test to be <u>repeated quickly, if necessary</u> through the "opening-up" and "shutting-off" of the steam valves once again, or more, bringing back the tested turbo-group up to its nominal rotation speed, if necessary.

With the eight pumps operating at a substantially lower power level than planned, the hydraulic resistance of the reactor core and of the circulation loop was also lower than planned, and as a consequence, the total coolant flow rate and pump individual flow rate (8,000 m^3/h) exceeded the normal operation level (7,000 m^3/h) defined in the operation rules (danger for pump break-down, pipe vibrations, ...).

But the most serious consequence of the increased flow was a reduction of steam formation and the creation of a coolant condition very close to saturation, with the possibility that a small temperature increase could cause extensive flashing to steam.

The steam pressure and the water level in the steam separation drums had also dropped below emergency levels and after attempting manually to restore more normal values in order to avoid a shut-down of the reactor, the operator blocked the emergency protection signals relating to these parameters.

9.   At 01.19 hrs, the feedwater supply was increased to as much as four times its initial value in an attempt to restore the water level in the steam separator drums. This reduced both the reactor inlet coolant temperature and consequently the fuel channel steam production with consequent negative reactivity effects.

Within 30 seconds, the automatic control rods had fully withdrawn in response to this negative reactivity and the operator attempted to withdraw the manual rods as well. But as they again overcompensated, the automatic rods began to move back into the reactor.

10. At 01.22 hrs, the reactor parameters were approximately stable and the operators decided to start the turbine test. But in case it would be necessary to repeat the test, they blocked the automatic reactor shut-down signal from the turbine stop valve, which they were about to close, in order to avoid a reactor trip.
    This meant a departure from the experimental programme which did not call for the blocking of the reactor protection with the switching off of the two turbogenerators, but only with the switching off of one turbogenerator, turbogenerator nr. 7 which had been shut off on April 25.

11. At 01.22.30 hrs, the operators obtained a print-out from the fast reactivity evaluation programme showing that the operating reactivity margin had fallen to a level that required immediate shut-down of the reactor. But they delayed already long enough to start the test!

12. Just before they shut off the steam to the turbine, they sharply reduced the feedwater back to the initial level required for the test conditions.
    This boosted the coolant inlet temperature, creating a transient situation.

13. At 1.23.04 hrs, the turbine stop valve was closed. With the isolation of the turbine, four of the primary circulation pumps that had been connected to the test turbogenerator were disconnected from their supply and started to run down.
    Shortly after the beginning of the experiment, the reactor power began to rise slowly.

14. At 1.23.40 hrs the scram button (manual shut-down of the reactor) was pushed which would send all control and safety rods into the reactor.

The motivation for pushing the scram button is still not clear, despite some talks with the shift supervisor in the hospital before he died; three hypotheses are presented by the Soviets :

- it was a belated reaction to the print-out (01.22.30 hrs) of the reactivity evaluation programme.
- it was a reaction induced by the observation of the increase in power
- it was an action taken to terminate the test, thinking that enough time (36 seconds) had elapsed to obtain the required observations and that everything was going all right.

The rods fell but after a few seconds a number of shocks were felt in the control room and the operator saw that the absorber rods had halted without plunging fully to their lower stops.

The operator then cut off the current to the sleeves of the servodrives so that the rods would fall into the reactor under their own weight : most certainly without result; during the time (20 s.) required for a safety rod travelling to full core insertion, mechanical distortions associated to the brutal power excursion were already present in the reactor internals.

15. Observers outside the plant reported two explosions, one after the other, at about 01.24 hrs; the second explosion was reported to have taken place three or four seconds after the first one.

Burning lumps of material and sparks shot into the air above the reactor; some of which fell onto the roof of the turbine hall and started a fire.

The mechanism of the accident

The experiment began at 01.23.04 hrs when the staff closed the turbine stop valve.

When 4 of the main circulating pumps began to run down - pumps which had been connected to the test turbogenerator - the flow was reduced, the coolant boiled and the reactor power began to rise (1.23.31 hrs).

It must be remembered that the coolant was very close to saturation with the possibility that a small temperature increase could cause extensive flashing to steam, resulting in a large positive reactivity insertion.

The reactor (fig. 26) went prompt-critical $[\rho = \beta]$ and even super-prompt-critical $[\rho > \beta]$, the reactivity investment reaching 1.5 β in less than two seconds, according to Dr. Kalugin of the Kurchatov Atomic Energy Institute.

The reactor power shot up more than 100 times the initial 200 MW(th) in less than a second.

Taking into account a void coefficient of the order of $x.10^{-4}$ Δ k/% vol. void (a value of 2.10^{-4} Δ k/% vol. void is reported $^{(*)}$ with a control rods content higher than at the moment of the accident; x has therefore a value higher than 2), I was considering a value of 1% reactivity insertion as reasonable.

As shown in figure 27, the parameter values, ℓ, the lifetime of a prompt neutron and β , the delayed neutron effective fraction, are chosen in order to generate results which could not be considered as exaggerated (ℓ is most certainly shorter than 10^{-3}s and β , with a U-5 and Pu-9 containing fuel is certainly lower than 0.0064).

$^{(*)}$ To be complete, it is also worth mentioning that the Soviet analysis reports also a value of $2.0.10^{-6}$ Δ k/% steam as the "void coefficient of reactivity at operating point".
This discrepancy by a factor 100 seems very large; the low void coefficient value correponds maybe to a reactor core with a large absorbers content, like a fresh core containing a large amount of fixed absorbers loaded in the fuel channels.

LARGE REACTIVITY EXCURSION

(SUPER) PROMPT-CRITICAL REACTOR

(APPROACHING CONTROL BY PROMPT NEUTRONS ONLY)

$$\rho = \beta \qquad\qquad \rho = \frac{l}{Tk_{eff}} + \sum_{i=l}^{m} \frac{\beta i}{i + \lambda_i T}$$

$$\rho - \beta > 0 \qquad\qquad T \cong \frac{l}{k_{eff}} \cdot \frac{1}{\rho - \beta}$$

$$\rho >>> \beta \qquad\qquad T = \frac{l}{k_{eff} - 1}$$

$$\left(\rho = \frac{k_{eff} - 1}{k_{eff}} \right)$$

Figure 26 : Relationships between reactivity and reactor period

$$\Phi(t) = \Phi(O) \cdot e^{t/T}$$

k_{eff}	ρ	T, seconds	$\Phi(0+1\ second)/\Phi(0)$
1.006	0.0064	0.83	3
1.010	0.0099	0.28	34
1.015	0.0148	0.12	5.10^3
1.020	0.0196	0.074	7.10^5

with $l = 10^{-3}s.$, $\beta = 0.0064$

Figure 27 : Neutron flux and thermal power increases in a 1 second
time delay

For a 1% reactivity investment, most exactly 1.5 β , the adequate formula predicts a power increase by a factor of 34 within a second; but with a 50% increase in the reactivity investment, the power evolution shifts from a factor of 34 to a factor of 5,000 : 100 times the initial 200 MWth in less than a second is therefore well in the picture.

On fig. 25 the power evolution during the accident, as resulting from the Soviet analysis on computer modelling of the reactor shows a first power burst to around 530 MWth and only the Doppler effect of the fuel heating-up to an estimated 3000°C pulled the power back down briefly.

But the continuing reduction of water flow during the power excursion led to intensive steam production and the destruction of the fuel.
With the energy reaching and even exceding 300 calories per gram of fuel, the fuel disintegrated, exploded; the particles of destroyed fuel entering the boiling water converted the remaining water droplets in the channels to steam, in a classic fuel-coolant interaction, which blew off the channel plugs and displaced the roof slab.
This displacement ruptured the pressure tube of all fuel channels, exposing the overheated fuel and the graphite to the air in the reactor building.

The explosion, the first one, was calculated to have generated "tens of atmospheres" of pressure when two atmospheres were sufficient to lift the reactor space upper slab.
French specialists have calculated, working back from the energy needed to rupture all fuel channels, an amount of energy released in the first explosion of 200 megawatt-second (200 megajoules).

MECHANISM OF THE ACCIDENT

FIRST EXPLOSION

Prompt critical reactivity excursion
Steam explosion

SECOND EXPLOSION

Various chemical and exothermic reactions
Hydrogen-carbon monoxide detonation
(Some doubts still remain)

Figure 28

But this worth appears to be underestimated :
- With a power peak reaching 15,000/20,000 MWth approxima-
  tely, a total energy release during the peak of the
  order of a few thousands megajoules can be estimated.
- Starting from an energy density exceeding 1260 Joules
  per gram of fissile fuel, at the moment of the fuel
  desintregration, a total energy amount of the order of
  5,000 Megajoules is found, to be reduced to about 3,000
  Megajoules, taking into account a mean core fuel
  burn-up of the order of 10,000 MWd/ton

An order of magnitude of a few thousands Megajoules is
probably correct.

As shown by the computer modelling, as a result of the
internal pressure at the moment of the fuel disintegration,
the check valves on the main circulating pump closed and the
corresponding loss of flow was recorded by the data logging
system.

After the rupture of the fuel channels, following the first
explosion, the flow from the pump would have been partially
restored (cells containing four of the main circulating
pumps are intact) but the water was now directed into a mass
of zirconium and hot graphite.

The ensuing chemical reactions would have produced large
amounts of hydrogen and carbon monoxide which upon contact
with the air above the reactor could have caused the second
explosion (figure 28).

This is at least what the Russians assume; according to some
other experts it could have been a second reactivity excur-
sion and fuel-coolant interaction in another part of the
reactor.

It is estimated that 3.5% to 4% of the reactor fuel (about 7
metric tons) was released from the reactor and deposited in
and around the site, all over the 30 km radius evacuation
zone.

VIOLATIONS OF OPERATING PROCEDURES

VIOLATION	MOTIVATION	CONSEQUENCES
Reducing operational reactivity margin below permissible limit	Attempt to overcome Xenon poisoning	Emergency protection system was ineffective
Power level below level specified in test programme	Error in switching off local automatic control	Reactor difficult to control
All circulating pumps on with flow in excess	Meeting test requirements	Coolant temperature close to saturation temperature
Blocking reactor protection system relying on shutdown signal from both turbogenerators	To be able to repeat the test, if necessary	Loss of automatic reactor shutdown possibility
Blocking water level and steam pressure trips from drum separator	To perform test despite unstable reactor	Protection system based on heat parameters lost
Switching off ECCS	To avoid spurious triggering of ECCS	Loss of possibility to reduce scale of accident

Figure 29

Visual examination confirmed that about 10% of the graphite (190 metric tons) had been ejected from the reactor building. About 25% of total graphite, or about 500 metric tons, was estimated to have been consumed during the graphite fire which followed the two explosions.

REMARKS
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1) The Russians, taking into account the potential danger of the melted core penetrating and perforating the concrete slab below the reactor core have succeeded in evacuating the water from the suppression pools (figure 20) and have injected special refractory concrete (heat resistant to 2000 degrees C) in these two pools. It is known that the concrete slab below the reactor has not been affected; so this measure taken to stop an eventual corium or melted core penetration into the soil proved unnecessary. But nevertheless these pools filled with concrete will play the role of a reinforced foundation for the entombment of the destroy-ed reactor.

2) It is also worth mentioning that nothing happened to the spent fuel pool which remained undamaged in the explosion; the water was still in place and there were approximately 100 spent fuel elements in storage at the time of the accident.

Causes of the accident

Figure 29 "Violation-Motivation-Consequences" is taken directly from the Soviet report.
The Soviets declare that if any one of the first five violations had not been committed, the accident would not have happened.

Concerning violation nr. 1, there is no doubt that the Russian assertion is true.
There may be some doubt left for some other ones.

Taking into consideration the weight of violation nr. 1, i.e. the operation of the reactor far below the authorized limit for reactivity margin, it can be said that relying only on administrative procedure to avoid such situations is a large error of judgment committed at a high level of the plant supervision and/or plant designer.

The Soviets said at Vienna, that there is no aircraft designer who considers it necessary to provide automatic locks to stop a pilot from testing the doors during flight (at high altitude) because nobody could imagine that a pilot would be stupid enough to try.
Firstly, the comparison does not stand up to critical analysis because the example of the aircraft door is too simplistic : it does not imply a process of judgment in the pilot's brain, a process of apprehension of a complicated situation; it is a technical evidence for him that, most certainly, the crew would not survive the test.
Such an evidence was certainly not present in the Chernobyl-4 control room during the early hours of April 26th.

Violations nrs. 4 and 5, both consisted of the blocking of the automatic reactor shut-down actions on three signals of emergency level (turbine stop, low water level in the steam separation drums, low steam pressure in the steam separation drums) to the emergency protection system.
These blockings were not foreseen in the test programme and they were decided on the spot, most certainly by the shift supervisor.
These blockings are catalogued by the Soviet report as "violation of operation procedures or operation rules" and a lot of questions have been raised about the reason for such an erratic comportment by the operation people on duty.
It is certainly an interesting discussion but, in my opinion and according to my experience of reactor operation, the dominant question is : "How is it possible that the by-passing of automatic reactor shut-down actions is left to the appraising of the shift supervisor?"

This is again an error of judgment committed at a high level
of the plant supervision and/or plant designer, error not
detected or covered, the result is the same, by the safety
authorities.

In Belgium, nobody at these three levels of design, operation
and safety-review would consider as acceptable to leave the
possibility of by-passing or deactivating a reactor scram
automatic action in the hands of the operation shift supervi-
sor.
The by-passing of such an action would, in Belgium, be
decided at least at the level of the plant supervisor, after
an in-depth technical analysis of the situation asking for
such a temporary by-passing and of the possible consequences
on the plant safety.
The conclusion of this analysis would be recorded in writing
and the relevant safety authorities would at least be
informed if not associated with the analysis. The action of
by-passing the automatic reactor scram action would be
ordered in written form to the engineer in charge of the
plant instrumentation. This temporary measure would be
rescinded in an equally formal manner.

I would suggest I summarize the picture "Violation-motiva-
tion-consequences" as shown on figure 30.
Starting from what is presented in the Soviet report as the
"chief-motivation" : To complete the test as expeditiously
as possible.
I propose to speak of a "chief-violation" in the following
terms :To leave the plant in the hands of incompetent staff
unaware of the possible dangers and I would summarize the
consequences as indicated on fig. 30; I think indeed that
the Chernobyl-4 accident will teach to the nuclear community
far more in the fields of emergency measures and radiation
induced health effects than in the field of reactor design
or reactor operation.

(CHIEF) VIOLATION	CHIEF MOTIVATION (according to the Soviet Report)	CONSEQUENCES
To leave the plant in the hands of incompetent staff unaware of the possible dangers	To complete the test as expeditiously as possible	- Disaster for site personnel and local population - More or less important set-backs for the nuclear industry - High value experience in emergency measures and radiation induced health effects and long term consequences

Figure 30

CAUSES OF THE ACCIDENT

PRIMARY CAUSE (OPERATORS)

Incredible series of violations
of instructions and operating rules

SECONDARY CAUSE (DESIGNERS/OPERATORS)

Design weaknesses

- Positive void coefficient (stability problems)
- Insufficient shut-down margin
- No secondary emergency shut-down system
- No efficient system for graphite cooling
- Insufficient containment for water cooled reactor
- Too large part of safety left to operating procedures

some aggravated by operators short-comings

(245 reactor-years of RBMK safe operation?)

Figure 31

Tentative conclusions are drawn in fig. 31 on the causes of the accident from what has been reported by the Soviet experts and previously published on the RBMK characteristics.

When speaking of "primary cause", it means that in the absence of this series of violations, the Chernobyl-4 accident would not have happened; "secondary" cause sums up the system characteristics which played an important contributing role to the severity of the accident.
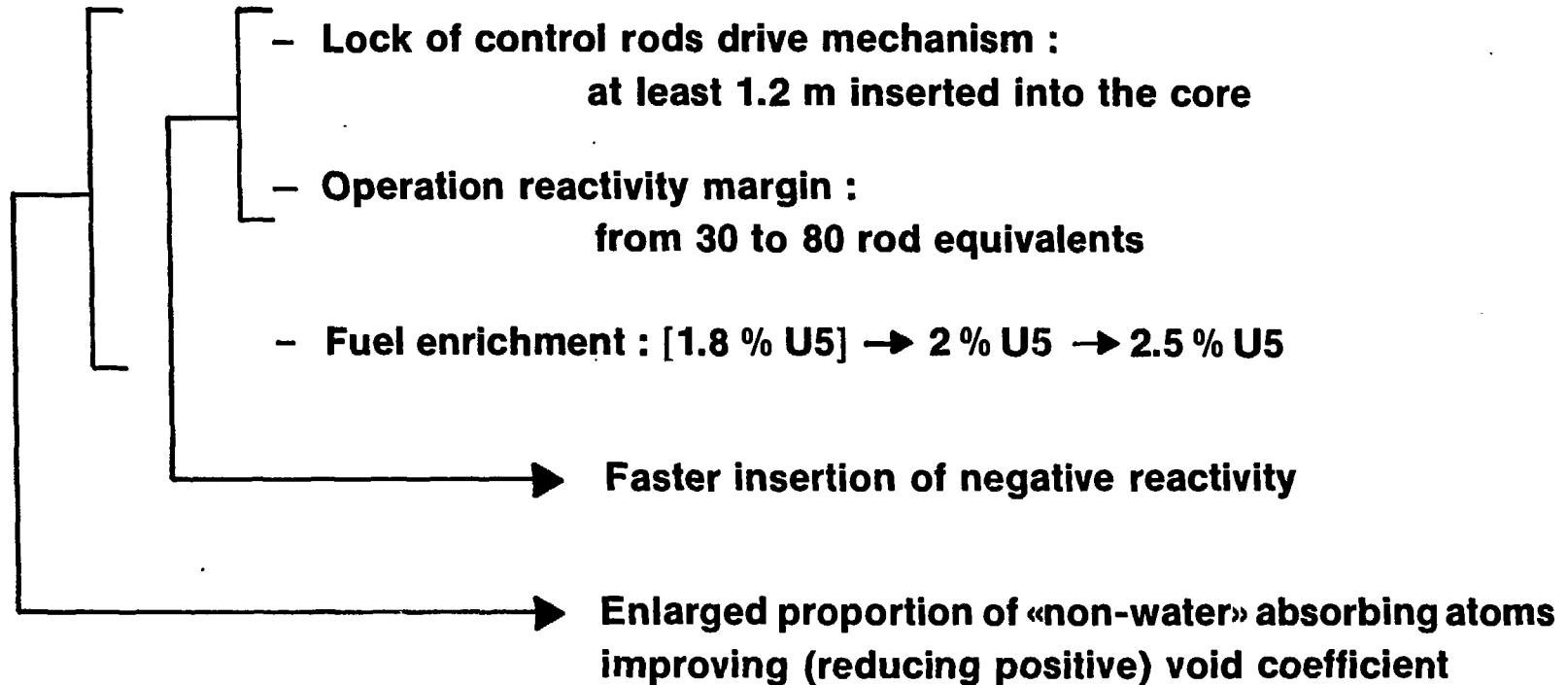
RBMK planned modifications

The Soviets in their report are relatively frank about the few identifiable design weaknesses of the RBMK reactor, but they also assert that modifications which will be, more or less quickly, implemented are sufficient to allow continued safe operation of the other RBMK reactors; this is a first priority question.

The positive void coefficient of the light water coolant in the reactor fuel channels is clearly the most significant safety characteristic of the reactor. In such a large reactor where the losses of neutrons by leakage and by absorption in the various materials within the core (low enriched fuel, moderator, structural materials) are relatively small, the coolant, the light water, becomes one of the more significant absorbers in the core.
Any reduction of the water density by boiling has therefore a significant positive effect on the neutron multiplication; if another absorbant is present, for instance a great number of inserted control rods, the removal of water atoms becomes less "reactivity" significant.
Among the immediate measures being taken on the RBMK reactors, is a lock of the rod drive mechanism that ensures at least 1.2 m of insertion in the core (the total core height is 7 m).

RBMK MODIFICATIONS

– **Lock of control rods drive mechanism :**
           at least 1.2 m inserted into the core

– **Operation reactivity margin :**
           from 30 to 80 rod equivalents

– **Fuel enrichment : [1.8 % U5] ➔ 2 % U5 ➔ 2.5 % U5**

➤ **Faster insertion of negative reactivity**

➤ **Enlarged proportion of «non-water» absorbing atoms improving (reducing positive) void coefficient**

Others, later : – **More control rods**
                   – **A rapid independent shut-down system**
                                 (fluid injection)

Figure 32

Also, the authorized minimum operating reactivity margin is
increased from 30 rod equivalents to 80; it means that in
their first second of insertion, the reactivity effect of
the available rods is equivalent to the full insertion of 80
rods.
These measures will both contribute to a reduction of the
positive void coefficient   and to   a faster   negative
reactivity insertion in case of emergency.
On the longer term, it has been said that more control rods
will be installed.

Another change designed to help reduce the positive void
coefficient will be the introduction of fuel with an enrich-
ment of 2.5% instead of 2.0%; as it can be seen in the
Russian   publications,   a   previous   fuel   enrichment
modification from 1.8% to 2.0% was already realized with the
same objective of helping to overcome the positive void
coefficient; this indicates - if necessary - that the void
coefficient has always been a preoccupation for the RBMK
designers.

> As a matter of example, the hereunder quoted text
> is taken from the article "State of the art and
> development prospects for nuclear power stations
> containing RBMK   reactors"   by   E.V.   KULIKOV
> published in Atomnaya Energiya - December 1984 /
> English translation : Soviet Atomic Energy, Volume
> 56- Number 6, June 1984 :
>> " The less the positive steam coefficient, the
>> higher the power distribution stability and
>> the simpler the reactor control. The most
>> rational   way   of   reducing   the   steam
>> coefficient   is   to   increase   the ^{235}U
>> concentration to the moderator concentration
>> in the core. The reduction in the steam
>> coefficient as a result of going to fuel with
>> 2% enrichment is about 1.3 . (1)
>> These conclusions served as a base   for
>> increasing the RMBK enrichment.(2) "

Notes
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(1) β is the effective proportion of delayed
     neutrons; I assume the quoted reduction is on
     the positive reactivity insertion associated
     to a void volume fraction varying from 0 to
     100%.
(2) from 1.8% U5 to 2% U5.

The apparent contradiction of improving the situation by
putting more fissile material into the core, is also related
to a greater proportion of "non-water" atoms capturing
neutrons in the critical balance equation, reducing the
relative importance of the absorptic   in the water.

Up to now the Russians had not adopted like others, the
Canadians and the Americans for instance, the provision of
an additional fast shut-down system, independent of the
control rods system; they have now decided, unfortunately
too late for the Chernobyl victims, to implement such a
system most probably based on the use of the injection of a
liquid absorber.

All these RMBK planned design modifications are summarized
in the figure 32; the difficult question is to judge if they
are adequate to reduce significantly enough the accident
probability associated to the RBMK reactors operation; they
will also certainly affect very badly the RBMK economy.

CONCLUSION

As a conclusion to my presentation, I want to express the following opinion, which, I hope, you will share :

-   It is unacceptable to produce nuclear electricity at the Chernobyl social cost (203 seriously injured people of which 31 died, 135,000 people evacuated with significant radiation doses and potential health consequences).
-   It is equally unacceptable to build large water dams which do not resist water pressure or geological movements, with the social consequences already experienced.
-   It is equally unacceptable to guarantee the security of people attending a football-match like it is done in Belgium.

These comparisons of energy production or social life risks you can hear or read about every day since Chernobyl - and I will not speak of the consequences of burning fossil fuels - do not in my opinion reduce the unacceptability of a tragedy like the Chernobyl 4 accident and do not attenuate the responsibility of those who contributed to the genesis of the accident - more or less consciously and unfortunately, most probably, less than more consciously.

This is why, each of us, have to remain conscious of the fact that we are sharing, in the nuclear community, the responsibility of elaborating the technical solutions for bringing or keeping the probability of occurrence of such an accident at an "as low as technically achievable level".
It will then be decided by others, if this level is or remains economically or politically acceptable.

Considering the RBMK-1000 design weaknesses and the apparently inadequate technical level of the operation personnel, such a system would never have obtained an operation license in Belgium .

It would be of course irrational to already make now a judgment on the adequacy of the planned RBMK modifications : it is part of the nuclear community's responsibility, hereabove evoked, to acquire a clear conviction on this subject and to make it clearly and broadly known.