
Chernobyl — A Canadian Technical Perspective

Tchernobyl – Perspective technique canadienne

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Abstract

On April 26, 1986, the #4 reactor at the Chernobyl Nuclear Power Station in the Soviet Union suffered a severe accident which destroyed the reactor core. The reactor design and the accident sequence are reviewed in detail, using Soviet literature and information presented at the International Atomic Energy Agency Post-Accident Review Meeting in August 1986. The aspects of the design and operation which exacerbated the accident, in our view, are presented and compared to the CANDU reactor design. Key design aspects for Chernobyl examined are (in order of importance): capability of shutdown, containment and variation of void reactivity with operating state.

Other concerns raised on Chernobyl (involving features common with CANDU) which we feel are not key design weaknesses for either Chernobyl or CANDU, are reviewed. These are: the sign of the void coefficient, pressure tubes, computer control, spatial control at low power, on-power refuelling, multi-unit containment, and fire protection.

It is concluded that the Chernobyl shutdown system design was deficient in that it did not provide an adequate level of safety for all plant operating states, and that the plant safety depended too heavily on the skills of operators in maintaining many reactor parameter, especially reactor power, within a certain operating envelope.

By contrast the effectiveness of the shutdown systems in CANDU is independent of the operating state of the plant and in that sense the design is much more forgiving. Nevertheless, as a prudent response to Chernobyl, AECL is undertaking two areas of design review for CANDU:

- 1) a re-examination of all possible core configurations to ensure these do not impede shutdown capability, and
- 2) a review of fire protection features in the presence of radiation fields.

Reviews of operational aspects are underway by the Canadian electrical utilities and a review by the Canadian regulatory agency (the Atomic Energy Control Board) is near completion.

Résumé

Le 26 avril 1986, un grave accident s'est produit à l'intérieur du réacteur n°4 de la centrale nucléaire de Tchernobyl, en URSS: le coeur de ce réacteur a été entièrement détruit. La conception du réacteur et le déroulement de l'accident sont étudiés en détail dans le présent document, à l'aide de documents soviétiques et de renseignements présentés lors de la Réunion d'analyse de l'accident de l'Agence Internationale de l'Énergie Atomique, en août 1986. Les aspects de la conception et du fonctionnement qui, à notre avis, ont aggravé l'accident, sont présentés et comparés avec ceux du CANDU. Les aspects-clés de la conception de Tchernobyl étudiés dans le présent document sont les suivants (par ordre d'importance): capacité de mise à l'arrêt, confinement et variation de la réactivité due au coefficient de vide selon la configuration du coeur.

D'autres caractéristiques communes au réacteur russe et au CANDU, qui ne nous semblent pas des faiblesses de conception importantes, ni pour l'un ni pour l'autre, ont été étudiées. Il s'agit du signe de coefficient de vide, des tubes de force, du contrôle-commande par ordinateur, du contrôle spatial à faible puissance, du chargement réacteur en marche, de l'enceinte de confinement multi-tranches et de la protection contre les incendies.

Il a été conclu que le système d'arrêt de Tchernobyl était déficient, dans la mesure où il n'a pas procuré un niveau de sûreté suffisant pour chacun des états de fonctionnement de la centrale, et que la sûreté de la centrale dépendait trop des opérateurs, c'est-à-dire de leur habileté à maintenir plusieurs paramètres du réacteur, et surtout sa puissance, à l'intérieur de certains domaines de fonctionnement.

Par contre, dans le cas du CANDU, le fonctionnement des systèmes d'arrêt est indépendant de l'état de fonctionnement de la centrale: en ce sens, il a une meilleure sûreté intrinsèque. Cependant, afin de tenir compte des erreurs de Tchernobyl, l'AECL reverra deux études de conception du CANDU:

- 1) un réexamen de toutes les configurations possibles du coeur, afin d'assurer qu'elles n'entravent pas sa capacité de mise à l'arrêt, et
- 2) une révision des caractéristiques de protection anti-incendie, dans des champs de rayonnement.

Les compagnies d'électricité canadiennes effectuent actuellement l'analyse des aspects de fonctionnement, et l'étude de la Commission de contrôle de l'énergie atomique, l'organisme réglementaire canadien, est pratiquement terminée.

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PART I INTRODUCTION

The Chernobyl Accident
A Technical Evaluation

I. INTRODUCTION

On April 26, 1986, the #4 reactor at the Chernobyl Nuclear Power Station suffered a severe accident. The core and much of the building were destroyed; all the noble gases and several percent of other fission products were released to the environment.

The reactor design and the accident sequence, have been studied extensively since then. While a reasonable amount of information on the reactor design was publicly available, (References 1 to 6) the specific features of unit #4 design and the accident sequence, were presented by the Soviets at an International Atomic Energy Agency (IAEA) meeting in Vienna in August 1986. That report (Reference 7) is the most authoritative data available to date, and this information is now being used by all countries with a nuclear power program to examine the robustness of their plant design and operation with regard to the events at Chernobyl, and to see what lessons can be learned.

In this report we present the design review done to date in Canada by AECL. From the Canadian point of view it covers:

1. relevant information on the Chernobyl design and the accident, both as presented (References 7, 8) by the Soviets at the Post-Accident Review Meeting (PARM) held in Vienna from August 25-29, 1986, and as deduced from publicly available Soviet documentation,
2. details of AECL's technical review of the CANDU PHWR (Pressurized Heavy Water Reactor) against the background of the Chernobyl accident, and
3. implications of the Chernobyl accident.

Reviews of operational aspects are underway by the Canadian electrical utilities and a review by the Canadian regulatory agency (the Atomic Energy Control Board) is near completion.

An executive summary of this document is given in Reference 10, and a less technical summary for the general public is given in Reference 11.

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PART II CHERNOBYL: THE SOVIET BRIEFING IN VIENNA
August 25-29, 1986

- II-1 SUMMARY**
- II-2 KEY DESIGN AND OPERATIONAL ASPECTS**
- II-3 ACCIDENT SEQUENCE**
- II-4 DAMAGE AND RELEASES**
- II-5 DOSE AND CONSEQUENCE ESTIMATES FOR THE USSR**
- II-6 CHANGES IN DESIGN AND OPERATION TO IMPROVE THE SAFETY OF RBMK-1000**

The post-accident review meeting (PARM) for Chernobyl took place from August 25 to 29, 1986. At the meeting the Soviets presented detailed information on the accident sequence, accident recovery, radiological consequences and planned design/operational changes for other reactors of the same type. In this section we summarize the information presented; in subsequent sections the detailed information is given.

1.1 Design

The Chernobyl site is shown schematically in Figure 1.

The Chernobyl nuclear power station consists of four reactors of the RBMK-1000 type in operation and two more under construction. The initials RBMK stand for "Large Reactor with Tubes". Each operating reactor generates 1000 MW of electricity through two 500 MW(e) turbines. The core consists of vertical pressure tubes, using graphite as a moderator, ordinary boiling water as a coolant, and slightly enriched (2%) uranium dioxide fuel. The steam cycle is direct, i.e., the boiling water from the channels goes through risers to a steam separator and directly to the turbine. There are two independent primary coolant circuits, each containing about 830 fuel channels; two steam separators; and four pumps (one normally on standby). Refuelling is done during operation from the top of the core. A containment structure encloses the inlet piping in the lower portion of the reactor and provides pressure relief to a water pool located beneath the reactor. Control and shutdown are effected by movable absorbing rods in lattice positions. Emergency core cooling is provided for pipe breaks and the system consists of a combination of pressurized water accumulators and electric pumps.

1.2 Accident Sequence

In the process of performing a safety-related test just prior to a scheduled shutdown, a sequence of events occurred which took the reactor outside the permissible operating range and at the same time led to the ineffectiveness of emergency shutdown. The combination of operating conditions, control rod configuration, operator violations of procedures and the inherent core characteristics, led to a large reactivity transient and rapid power rise.

The fuel energy reached a mechanical breakup level causing rapid fuel fragmentation in the bottom portion of the core that caused an overpressure in the cooling circuit. Pressure tube failures led to pressurization of the core vessel and loading of the 1000 Mg top shield reinforced concrete slab, expelling it from the reactor "cavity". Burning fragments were ejected from the core, starting 30 fires in the surrounding area.

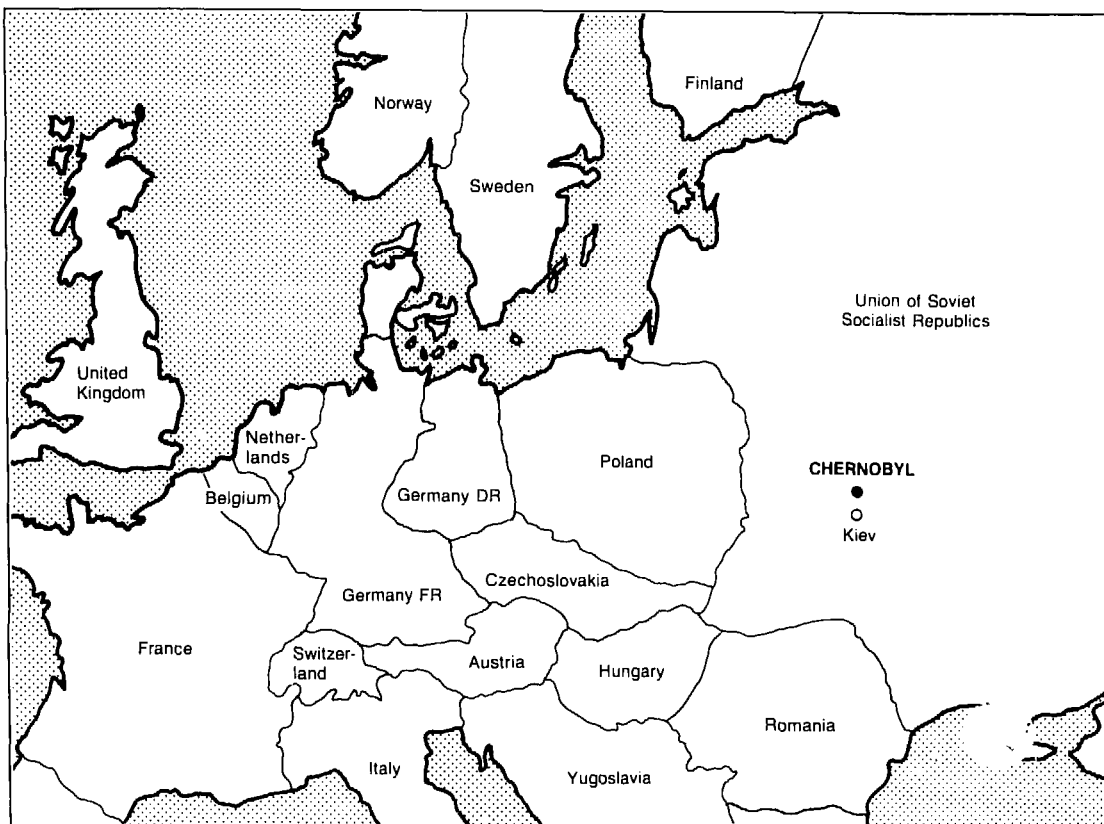


FIGURE 1(a) CHERNOBYL REACTOR LOCATION

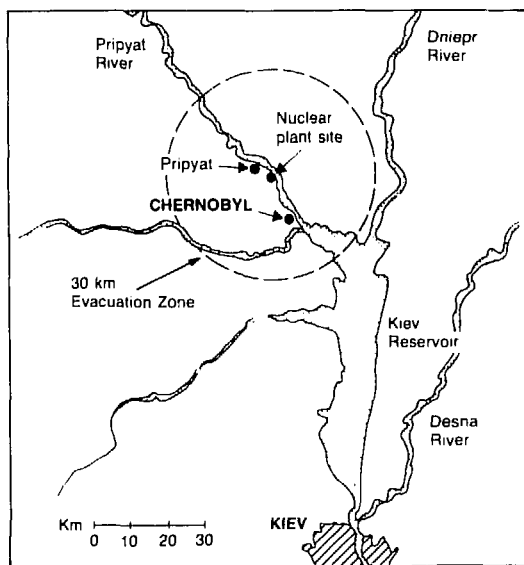


FIGURE 1(b) AREA NEARBY THE CHERNOBYL REACTOR SITE

1.3 Immediate Effects of Power Runaway

The core expanded into the surrounding space in the reactor "cavity" and a mixing of the fuel and moderator resulted in the core becoming subcritical. The severing of inlet pipes and outlet pipes and the destruction of the upper portion of the reactor building led to air access to the core. The graphite began to burn locally; ultimately 10% was oxidized.

1.4 Radioactive Releases

Fragmented fuel and fuel aerosols were expelled in the explosion and taken high (0.8 - 1 km) into the atmosphere by the thermal plume from the hot core.

The continuing thermal plume from burning graphite and fuel oxidation carried aerosols and finely dispersed fuel particles into the atmosphere. Gradually the fuel temperature stabilized as convective air cooling was established and the rate of release fell.

To stop the release the Soviets dropped about 5000 tonnes of material including boron carbide (to ensure shutdown), dolomite (to produce carbon dioxide to try to smother the fire), lead (to absorb heat and provide shielding), sand and clay (to create a filter bed). This led to a rise in fuel temperature as the convective cooling was cut off. The core reached a hot, oxidizing condition (peaking on May 4) and fission product release rates increased again.

At this stage the Soviets fed nitrogen to the bottom of the reactor cavity, cutting off the ingress of oxygen and extinguishing the graphite fire. The fuel temperatures dropped with a corresponding sharp reduction in releases. The core was now in a stable air convective cooling mode.

Total releases were estimated by the Soviets to be: all (100%) of the noble gases, 10-20% of the volatile fission products, and approximately 3.5% of the long-lived fission products. It was acknowledged there is substantial uncertainty associated with these estimates.

1.5 Accident Recovery

Fire-fighting started immediately and external fires were brought under control in four hours.

Extensive cleanup and decontamination began. A "sarcophagus" (reactor burial structure), utilizing a forced-convective air-cooled system with open ventilation and a filtration system, was built around the reactor and turbine hall. The sarcophagus surrounds the reactor and turbine of Unit 4 and reduces the radiation level so that reactor Units 1, 2, and 3 can be operated.

Core meltdown was a Soviet concern during the days following the accident (but it did not occur). Water was drained and replaced by concrete in the original suppression pools, to catch any molten materials. To prevent ground water contamination a concrete wall was built deep into the ground around the area.

As of January 1987, Units 1 and 2 have been restarted. The timing of startup of Unit 3 is less certain due to the higher level of activity and the need to check the condition of the equipment.

In the area, soil was skimmed or covered by concrete. Special sprays (liquid rubber, potassium and sodium silicate and plastic film) were effective in decontamination. Concrete barriers were being erected along the nearby riverbank.

1.6 Radiological Consequences

1.6.1 Onsite Staff

There were two deaths immediately as a result of the accident. Of the 29 fatalities from high radiation doses received by station staff trying to bring the accident under control, the dose distribution was as follows:

<u>Dose (rads)</u>	<u># Patients</u>	<u># Deaths</u>
600 - 1600	22	21
400 - 600	23	7
200 - 400	53	1

1.6.2 Offsite - Effects of the Accident on the Surrounding Population

Emergency response measures included:

1. Iodine tablets, administered to the population around Pripyat, apparently successfully and with minor side effects.
2. Sheltering for residents of Pripyat before evacuation.
3. Evacuation once the plume shifted towards Pripyat.

The Soviets estimated the collective dose commitment* in the USSR as:

31 x 10⁶ man-rem external (over 50 years)
210 x 10⁶ man-rem internal (over 70 years)

* Expected future dose to be received for a person who remains in the area.

In both cases the dose commitment is mostly from caesium. The latter figure is a conservative estimate which was acknowledged as perhaps 10 times too high.

1.7 Design/Operational Changes for Other RBMK Reactors

A number of design and operational changes for other RBMK reactors were presented by the Soviets at the meeting.

1.7.1 Design

1. Improved effectiveness of emergency shutdown will be achieved in the short term by increasing from 30 to 80 the equivalent number of rods inserted in the core and also by limiting their uppermost removal position to 1.2 m from the top of the core.

This reduces the maximum void reactivity from the current level (20 mk) obtained using 30 rods.

2. Additional operating information would be made available to the operator in the control room. Little detail was given but margin to fuel dryout was mentioned.
3. In the longer term the fuel enrichment will be increased to 2.4%. This should reduce void holdup but will require more reactivity from the control rods (more fixed absorbers).
4. Also, in the longer term, an (independent?) fast shutdown system may be added. Poison injection (liquid, gas or solid) into some control rod channels was mentioned as a possibility.

The above mentioned changes were stated to keep the reactor's maximum reactivity below prompt critical (for the most severe accident) and also to provide rapid reactor shutdown.

1.7.2 Operational

The areas that will receive emphasis are:

1. Violation of operating procedures.
2. Clarification of command responsibilities.
3. Improvement of the man-machine interface.

KEY DESIGN AND OPERATIONAL ASPECTS RELEVANT TO THE ACCIDENT

This section identifies key aspects of the Chernobyl-4 station relevant to the accident. Detailed descriptions of the complete design are given in References (1) to (9).

2.1 Conceptual Basis

Chernobyl unit 4 is of the RBMK (roughly translated as "large reactor with tubes") type and the most recent of the 1000 MW(e) series. It is a graphite moderated, boiling light water-cooled, vertical pressure tube design, using enriched (2%) U-235 UO₂ fuel with on-power refuelling. It utilizes a direct cycle, to produce electricity from twin turbines (see Figure 2).

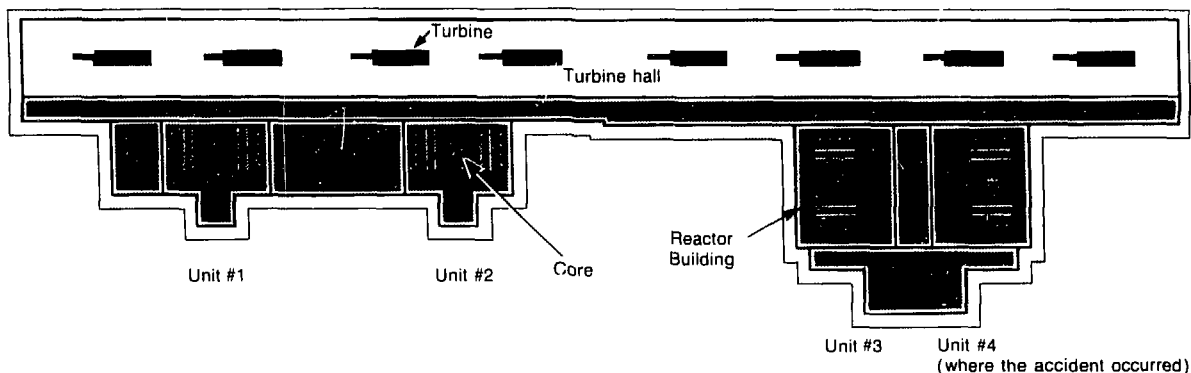


FIGURE 2 LAYOUT OF FOUR REACTOR UNITS

The reactor core is shown in Figure 3. The key reactor physics parameters in the equilibrium fuel state are a positive void reactivity with a strong dependence on the operational configuration of the reactor. The design basis called for a maximum void reactivity coefficient of 20 mk (for 100% voiding of the reactor core water) whereas at the accident conditions it was reported to be 30 mk/100% void. (Note that at their normal operating conditions, i.e. above 20% full power, the void coefficient is about 5 mk) The moderator temperature coefficient is strongly positive for the irradiated core but because of the slow response characteristics of graphite it did not play an important role in the accident.

The large core size is noteworthy since it leads to the potential instability of the power distribution and in the extreme, to local criticality. In the RBMK a spatial control system is required primarily for feedback reactivity induced spatial instabilities. The graphite moderator heat capacity is very large, being at least 400 FPS (full power seconds) above ambient at nominal conditions as compared to that of fuel (11 FPS), and the primary coolant (150 FPS). A distinguishing feature of the RBMK is the use of the primary circuit as the sink

for the moderator heat (5.5% of fission energy). Considerable sophistication has gone into the design of the contact conductance between the pressure tube and moderator and the conductivity of the moderator cover gas.

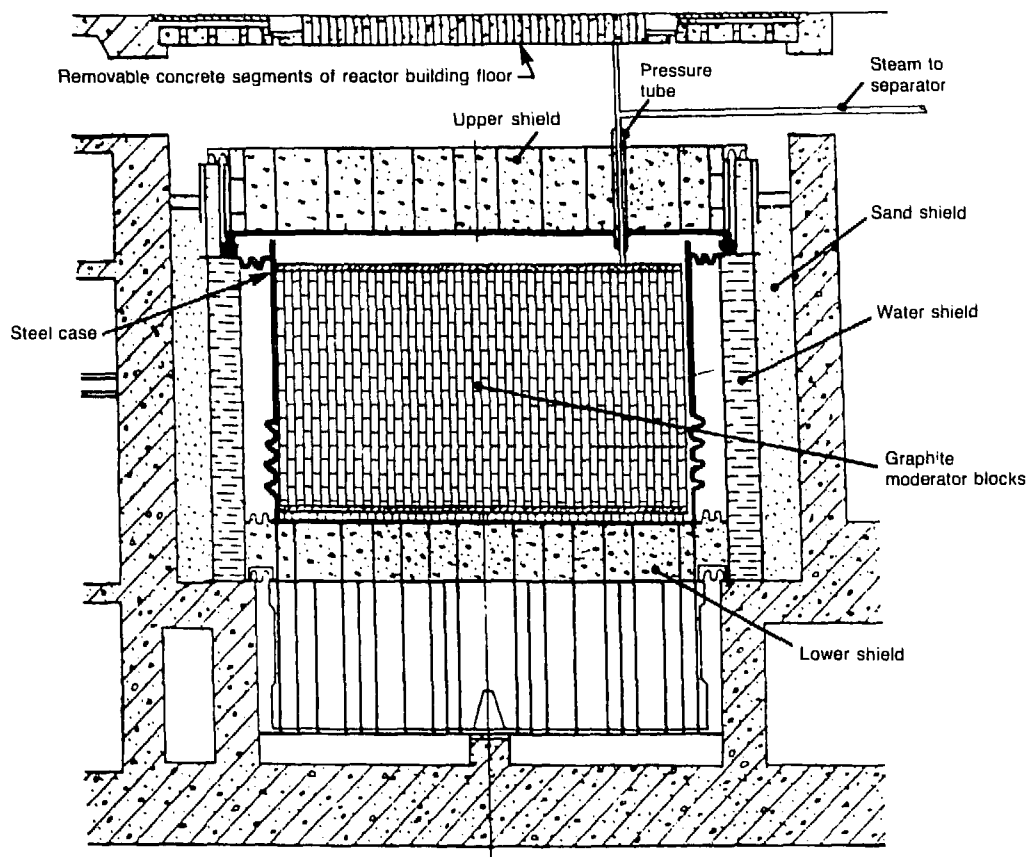


FIGURE 3 CROSS-SECTIONAL VIEW OF REACTOR VAULT

With respect to emergency shutdown the most important features were a slow rate of negative reactivity insertion and a dependence of that rate on the control rod configuration. Administrative controls were required to ensure at least 30 rods equivalent in the core at all times. This heavy reliance on administrative control was traced to early USSR experience in which the operators were more reliable than automatic systems.

2.2 Thermalhydraulic Design

The RBMK thermalhydraulic design is based on a boiling water, direct cycle heat transport circuit (see Figure 4). Steam mass qualities range from 11 to 22% at nominal conditions. Provision for individual channel flow adjustment is made and is performed manually a few times between channel refuelling in

order to match flow to power. There are two normally independent primary circuits, which can be interconnected to a single turbine-generator at low power (as at the time of the accident). Moderator heat is transferred to the coolant through the pressure tubes and this large heat source is an important feature during startup and shutdown.

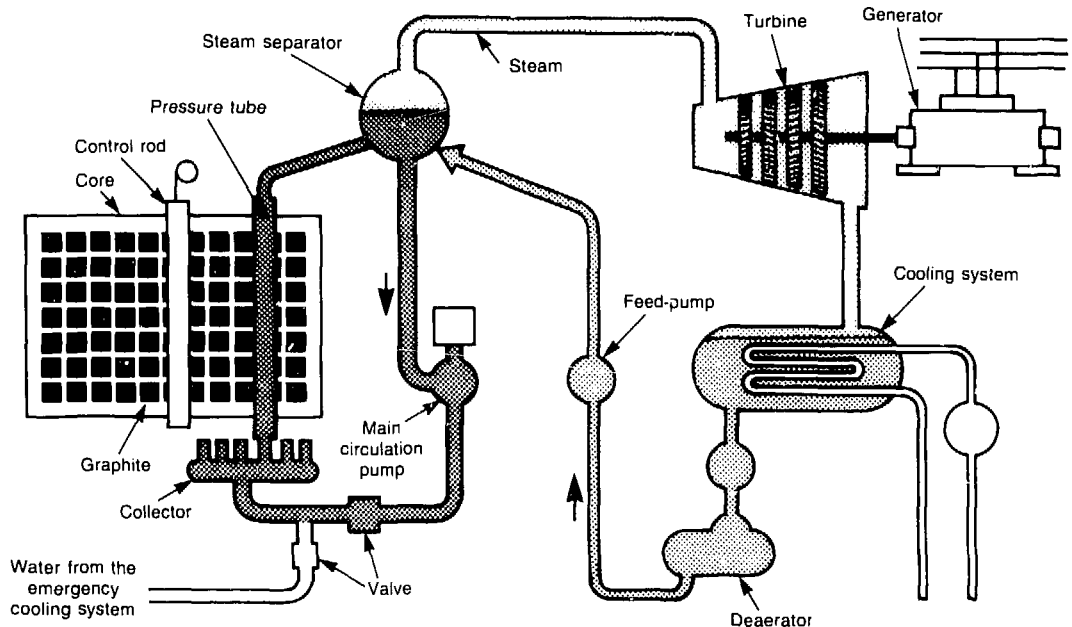


FIGURE 4 SCHEMATIC DIAGRAM OF THE RBMK-1000

The primary circuit flow is induced by three pumps per loop. The pumps have significant rotational inertia that permits a transition to thermosyphoning on loss of power without fuel heat transfer concerns. There is a spare pump in each loop that can be started up at power, but because this leads to a reduction of the net positive suction head, it is not normally used.

The condensate from the turbine is returned to the steam separator, and mixing occurs in the drum. Changes in feedwater flow can therefore have a direct feedback on core inlet temperature (separated in time only by a transport delay).

The design of the emergency cooling (ECC) system will not be described since its removal from service did not alter the course of the accident. Of minor relevance was its relation to the purpose of the test which led to the accident. The turbine mechanical energy was being used to demonstrate that it could provide power to an ECC pump. In fact, four of the main circulating pumps were used as "electrical simulations" of the load of the ECC pump in the test.

The containment "localization" system in Chernobyl was a recent RBMK design (see Figure 5). In this design the containment was divided into local compartments with distinct design pressures and relief/pressure-suppression to the "bubbler pond" in the bottom of the building. Allowance for the removal of limited rates of hydrogen production was provided.

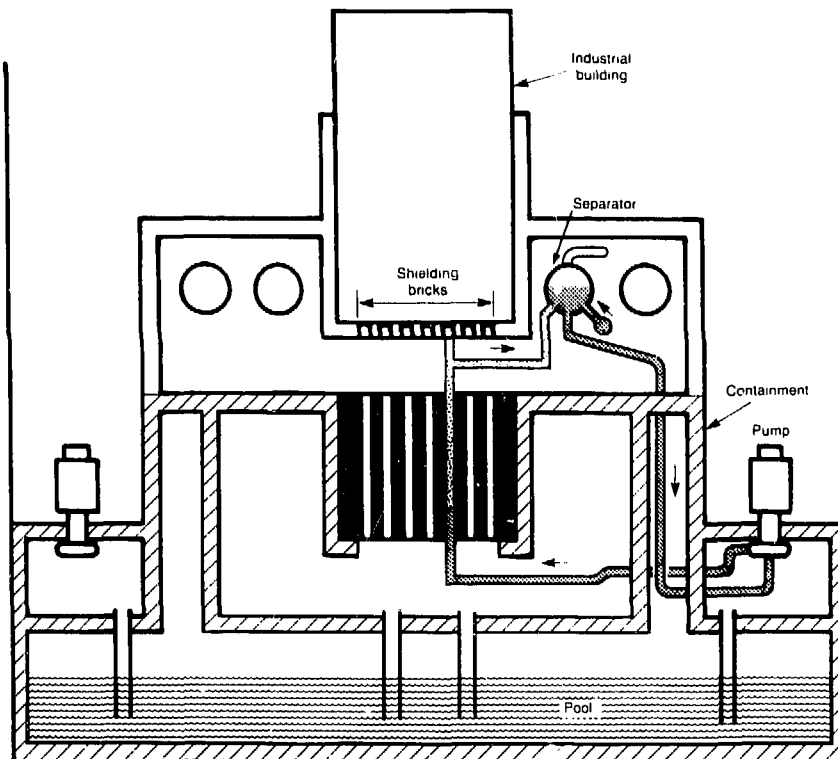


FIGURE 5 CHERNOBYL CONTAINMENT

The top portion of the reactor (risers, separators, steam lines, fuelling machine room) was not within a pressure-retaining containment. For design basis events (like riser tube rupture), it is believed that the Soviets felt that the large fuelling hall was adequate for the limited discharge rates. In any case they stated the impracticality of building a containment of this size.

Relief for the graphite core vessel was provided by eight 30 cm pipes connected to the bubbler pool. Relief capacity was stated to be designed for a single channel rupture.

In the accident, the steam explosion led to multiple pressure tube failures, which caused a pressure rise in the graphite vessel, well beyond design capacity.

Thus, the containment localization system played no real role in accommodating the accident. The basic structural integrity of the lower "containment" compartments was preserved. The upper portions of the building were designed for modest loadings and suffered dramatically from the thermal and possible chemical explosions that occurred.

3.1 Test Description/Rationale

One of the design basis accidents in the USSR is a loss of coolant plus loss of offsite power.

The ECC system which provides water to the broken loop in the short-term, consists of three 50% trains: two 50% trains of pressurized accumulator injection and one 50% pumped train.

Between the time the grid is lost and the time the Class III diesel generators start and are connected to the ECC pumps (25 s), the ECC pumps were to be powered from the inertial energy of the rotating turbine. In earlier tests, designed to prove this operation, the voltage dropped faster than anticipated, and hence, a new voltage exciter had been installed. The purpose of the test which led to the accident was to see if the exciter sufficiently prolonged the period of high voltage. Thus the test, broadly speaking, involved:

1. Reduction of reactor power to 700 - 1000 MW(th)
2. Diversion of all the steam to one of the two turbines
3. Tripping the turbine
4. Using the rotational energy of the turbine-generator to supply in-house load for a few tens of seconds.
5. Observation of the voltage rundown

Note that keeping the reactor at power was not necessary for the test after the turbine was tripped. The reactor was only kept at power so they could easily repeat the test, if needed.

The intent of the test was to have two heat transport pumps in each loop connected to the grid, i.e., continuously operating. This would ensure adequate flow as the other pumps, powered by the turbine-generator, ran down. These pumps were "simulators" of the electrical load of the ECC pump - the operators did not want to test the ECC pump directly because of the risk of spurious injection.

3.2 Event Sequence

The event sequence started with the reactor at full power on April 25 at 1:00 hrs. The test was to be done just prior to a shut-down for a planned maintenance outage.

In Table 1 the event sequence according to the Soviet interpretation is paraphrased. Our own comments are given in square brackets []. We have interspersed observed events with the most reliable portions of the post-accident analysis done by

the Soviets, using a model verified against the observations of plant behaviour during the time before core destruction.

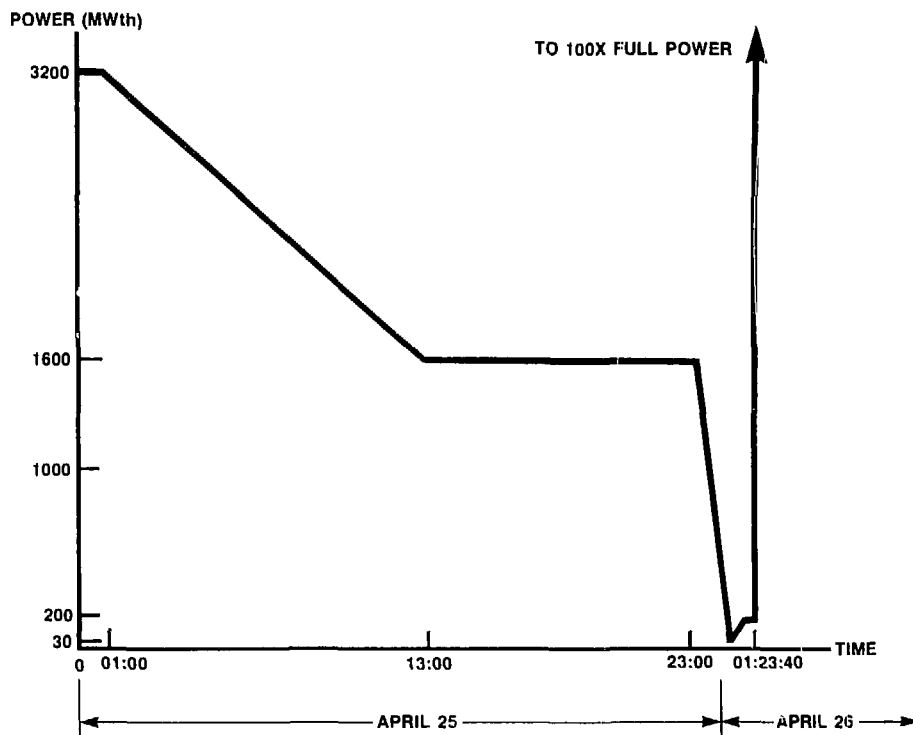


FIGURE 6 REACTOR POWER VS TIME

Table 1: ACCIDENT SEQUENCE

TIME	EVENT	COMMENTS
April 25 01:00	Reactor was at full power Power reduction began. (see Figure 6)	As per test intent and planned shutdown
13:05	Reactor power 50% (1600 MW(th)). Turbine #7 tripped.	All steam flow now directed to turbine #8.
14:00	The emergency core cooling system was isolated. However, the electrical grid controller requested the reactor to stay at 50% power because of electrical demand.	As per test procedure [to avoid a spurious injection]. The delay in power reduction meant that the reactor operated at high power for 9 hours with ECC disabled - a violation of procedure. However, disabling ECC did not affect the outcome of the accident.

23:10	Power run-down continued.	The test called for an initial power of 700-1000 MW(th).
April 26 0:28	Switch from local automatic power control to bulk automatic power control occurred. Reactor power fell to 30 MW(th).	The switch is made because the local automatic control (LAC) is normally ineffective at low power. However, the setpoint of the bulk automatic control (AC) was not matched to the operating power at the point of transfer. The Soviets said the operators was in error in not resetting the AC. [The result of the error was a reduction of void and a build-up of xenon which made it harder to raise power again, a severe power maldistribution and a need for override reactivity.]
01:00	Operator managed to raise power to 200 MW(th)	The implication of the switchover error is that to get up to 200 MW(th), the operator had to pull out more control rods than allowed by operating procedures.
01:03- 01:07	The 4th (standby) heat transport pump was started in each loop	The intent was to have two pumps in each loop operating after the test. However, because of the low power, the void in the core was greatly reduced by the increased flow; hence a drop in separator level and pressure occurred.
01:19	Operator increased feedwater flow	To restore separator level. This caused a further drop in primary circuit void (cold water at core inlet).
Between 01:20 - 01:22	Shutdown system blocked on low separator level. AC control rods went up (removed from core).	Normally the reactor would trip on low separator level. Since the operator was struggling to control this level manually, he

blocked the trip so as not to preclude the test. The disappearance of the void caused a reactivity drop which the AC rods tried to compensate. By now the primary coolant was saturated [with the void almost entirely in the risers and separators]

Between 01:20 - 01:22	Manual control rods removed.	[To give the AC rods some manoeuvring room.]
1:19:58	Operator closed condenser steam dump valve	To keep up system pressure. The separator level was now rising due to the excessive feedwater flow (three times heat balance value)
1:21:50	Manual decrease in feed-water flowrate.	The operator attempted to correct the excessive flow. However, the new flowrate was two-thirds of the heat balance value. The result was that during the next minute, the channel inlet temperature was continually increasing and this added to the sudden increase in void as discussed below.
1:22:10	Void growth began.	Due to warmer water. LAC rods lower to compensate.
1:22:30	The power distribution was printed out.	An operator request. This showed that the reactivity margin was at a value requiring immediate shutdown [six to eight effective rods vs 30 required]. The flux was peaked at the top and bottom of the core due mainly to the absence of inserted rods.

1:22:45	Steam quality in core stopped rising.	[Operator saw a "pseudo-steady state"]
1:23:04	Turbine #8 stop valves were closed.	The test began. The operator also disabled the reactor trip on loss of two turbines. The purpose was to have a chance at repeating the test if necessary. There were still reactor trip signals on high power and high rate of power increase. [The closing of the stop valve meant the reactor had <u>no</u> heat sink except for the main steam safety valves.] The blocking of the trips was a procedural violation and was not required in order to perform the test.
1:23:21	Power rose slowly. AC rods move in.	[Three factors determined the void, and hence the power: the increase in primary circuit pressure (no heat sink), the decrease in flow (four of eight pumps coasting down) and the increase in channel inlet temperature (reduced feedwater flow). The latter two increased the void, the former decreased it. The net void increased and as the channel was saturated, travelled very quickly down the channel. This caused a fast reactivity increase.]
1:23:40	Manual trip button pushed.	Rationale not clear - the operator may have seen the power rise or may have seen control rods moving to offset the power rise or may have decided on a normal test termination. In any case, with most of the

rods withdrawn, it would have taken ~ 6 s to have any significant reactivity "bite". The rods did not fully insert [core damage?] and the operator cut the current so they would go in faster. Two shocks were felt in the control room.

1:23:44 Power surge and fuel breakup. The reactor power increased rapidly and was predicted to have reached about 100 times full power in 4 s. The energy input to the fuel caused the onset of fuel breakup (300 cal/gm (1.3 MJ/Kg) estimated) [approx. 450 cal/gm to fuel vaporization]. A rapid surge of energy to the coolant and a steam explosion in the core followed [probably failing pressure tubes]. Primary circuit relief valves opened.

At this point the event sequence becomes more speculative. The following is one interpretation.

The steam explosion blew the 1000 Mg top shield structure off the reactor. Since all the channels are anchored to this shield, the consequence was failure of any remaining pressure tubes and ejection of control rods, which were mostly out of the core region anyway. About 30% of the fuel disintegrated. A second explosion was heard a few seconds later. The Soviets would not commit themselves to the nature of this explosion, [there is considerable question as to its existence] but the possibilities suggested are:

1. combustion of the hydrogen and carbon monoxide formed by rapid oxidation of the Zircaloy and UO_2 in water and by graphite-steam reaction, or
2. a second power pulse/steam explosion.

The former is believed by the Soviets to be more likely. The explosions were accompanied by burning lumps of material shooting up in the air and landing on the roofs of the reactor and turbine buildings.

The reactor was shut down by:

1. dispersal of some of the fuel into the graphite. The Soviets did a "homogeneous reactor" calculation which showed this would lead to a reactivity decrease, and/or
2. dispersal of the graphite into the reactor vault (i.e., destruction of the radial reflector and water shield) [and ejection of fuel into the upper and lower feeder areas].

II-4 DAMAGE AND RELEASES

4.1 Damage to the Plant

4.1.1 Introduction

The reactor core is housed in a vessel consisting of a thin-walled vertical cylinder capped by thick endshields. The endshields consist a 14.5 m (bottom) and a 17 m (top) diameter plate and are connected by approximately 2000 tubes (through which the fuel and control channels pass) and a short thin-wall cylinder. The plate structures are on the order of 20 - 40 mm thick. The endshields are filled with serpentinite. The cylindrical reactor vessel is surrounded by a series of annular rings consisting of: an empty space about one meter thick, a water-filled tank about one meter thick, about two meters of sand and finally a two meter thick concrete vault wall. This structure sits on top of and is partially enclosed by the containment enclosures. The top is essentially open to the fuelling machine room.

4.1.2 Building and Core Damage

The roof of the reactor building (primarily that portion away from the turbine building) was blown away during the explosion and much of the structure of the reactor building was damaged. The lower pressure suppression chambers housing the pumps and inlet manifolding remained intact. (The pump motors which are outside containment were intact and exposed to view by the destruction.)

Photographs of the installation show substantial destruction. The upper shield (1000 tonnes) can be seen on edge at the top of the reactor in the fuelling machine hall with shreds of channels attached to it. All of the steam outlet (riser) lines were broken by the lifting of the lid. Most of the larger debris from the building fell quite close to the reactor building. Figures 7 and 8 show a schematic view of the reactor building before and after the accident.

4.2.3 Fire fighting & Graphite Fire

Fire crews from the plant, Chernobyl, and Pripyat were called out to fight the fires in the middle of the night (1:30 a.m.). Major fires on the turbine building roof were extinguished within an hour. Other minor fires were extinguished and brought under control by about 5 a.m.

An attempt to cool the reactor with water from the boiler feedwater system was unsuccessful. Steam rose from the reactor for about a day until the water in the vault was exhausted. After the first day, the steam was replaced by smoke, indicating the water was not getting to the fire. A red glow could be seen in videos from helicopters - this could be due to heating from fuel or graphite oxidation. Subsequent dumping of boron, lead, dolomite, sand and clay on the reactor (5000 tonnes

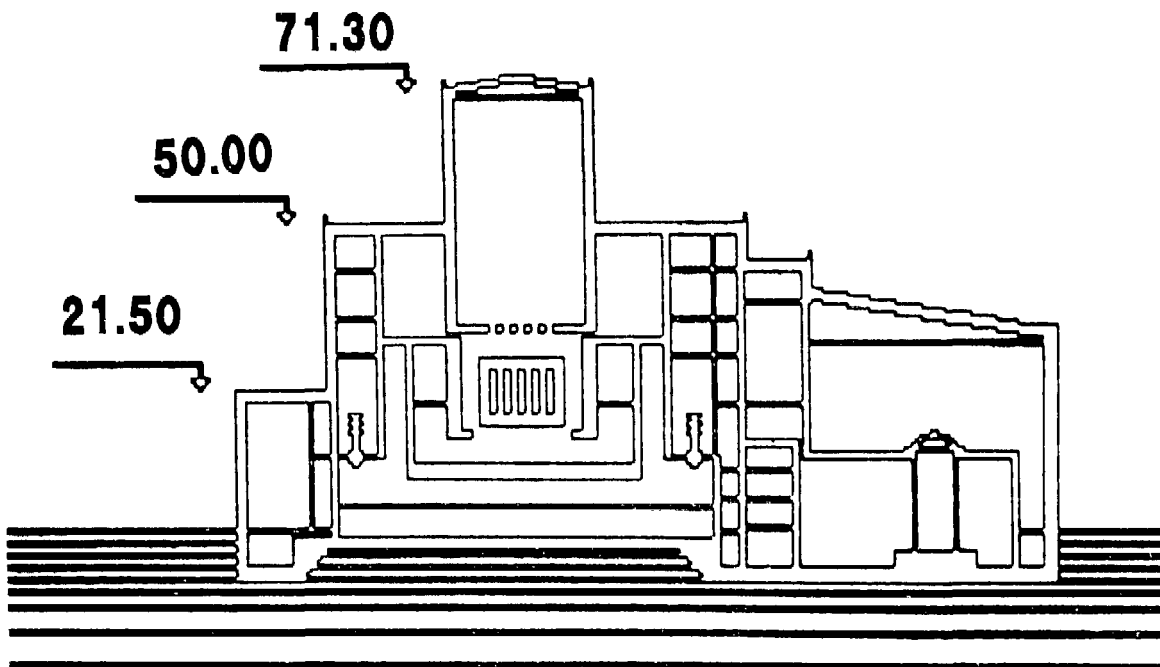


FIGURE 7 VIEW OF THE REACTOR BEFORE THE ACCIDENT

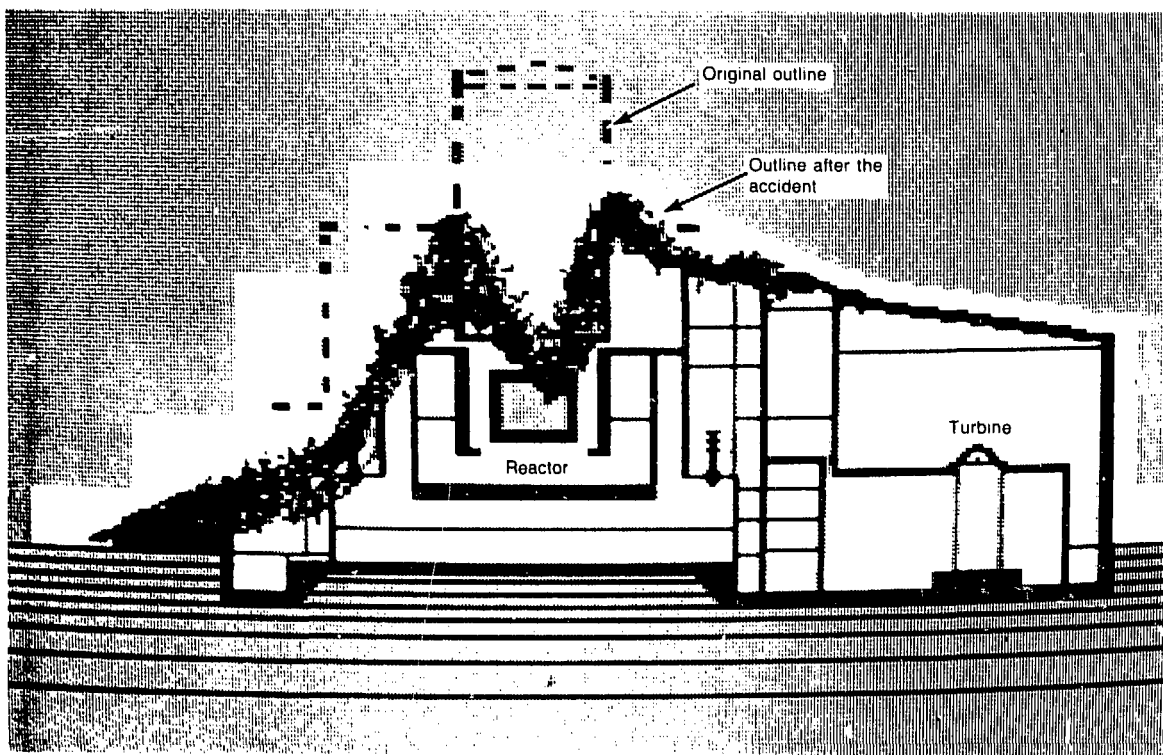


FIGURE 8 VIEW OF THE DAMAGED REACTOR

total), combined with nitrogen injection, finally snuffed out the graphite fire after about 10 days. Approximately 10% or 250 tonnes of graphite is judged to have burned.

4.2.4 Location of Fuel

Post-accident measurements and analyses suggest that fuel has been dispersed through graphite to some extent. Some of the fuel seems to have fallen into the containment enclosure in the inlet piping area of the reactor. Some is in the steam outlet piping area above the reactor.

The Soviets also believe that most of the core material has been forced outward radially into the water, air, sand space at the sides of the core. About 3.5% of the fuel is dispersed in a 30 km radius zone around the reactor. 96% is thought to remain in the reactor vault and immediate vicinity.

4.2.5 Entombment

The Soviets have built an enclosure to cover the damaged reactor and its associated buildings. It has concrete walls about a meter thick and is fitted with a once-through forced ventilation system which is intended to remove decay heat from the remains of the reactor.

Figures 9(a), (b), (c) show general views of the reactor before and after the accident, and the entombment.

4.3 Core Temperature Evolution and Fission Product Release

The Soviets have identified four phases of release of radioactivity from the core (see Figure 10). The first phase was the ejection of fuel and fission-product-containing particles into the air due to the explosion. This released a large quantity of radioactive aerosols whose composition was similar to the fuel composition. The force of the explosion drove the plume at least 800 m into the air and this resulted in long range transport of some of the initial release, despite the fact that non-volatile species made up most of the emission. The Soviets state that, immediately following the explosion, the effective temperature of the fuel was 1600-1800 K for some tens of minutes but that the fuel temperature decreased due to heat transfer to the graphite and structural materials. This high temperature, possibly coupled with the oxidizing conditions, released volatile iodine, caesium and tellurium from the core, especially during the early period following the explosion. Thus, releases on April 26 were characterized by aerosols containing activity in a ratio similar to the fuel composition and by volatile fission products released during the time of high fuel temperature following the explosion. The release on April 26 was 20-22 MCi ($7.4 \times 10^5 - 8.1 \times 10^5$ TBq).

Phase 2 of the releases (April 26-May 2) involved further oxidation of the fuel by air, and the formation of U_3O_8 powder. At the same time the graphite began to burn, creating a

Figure 9(a) Units 3 & 4 before the accident

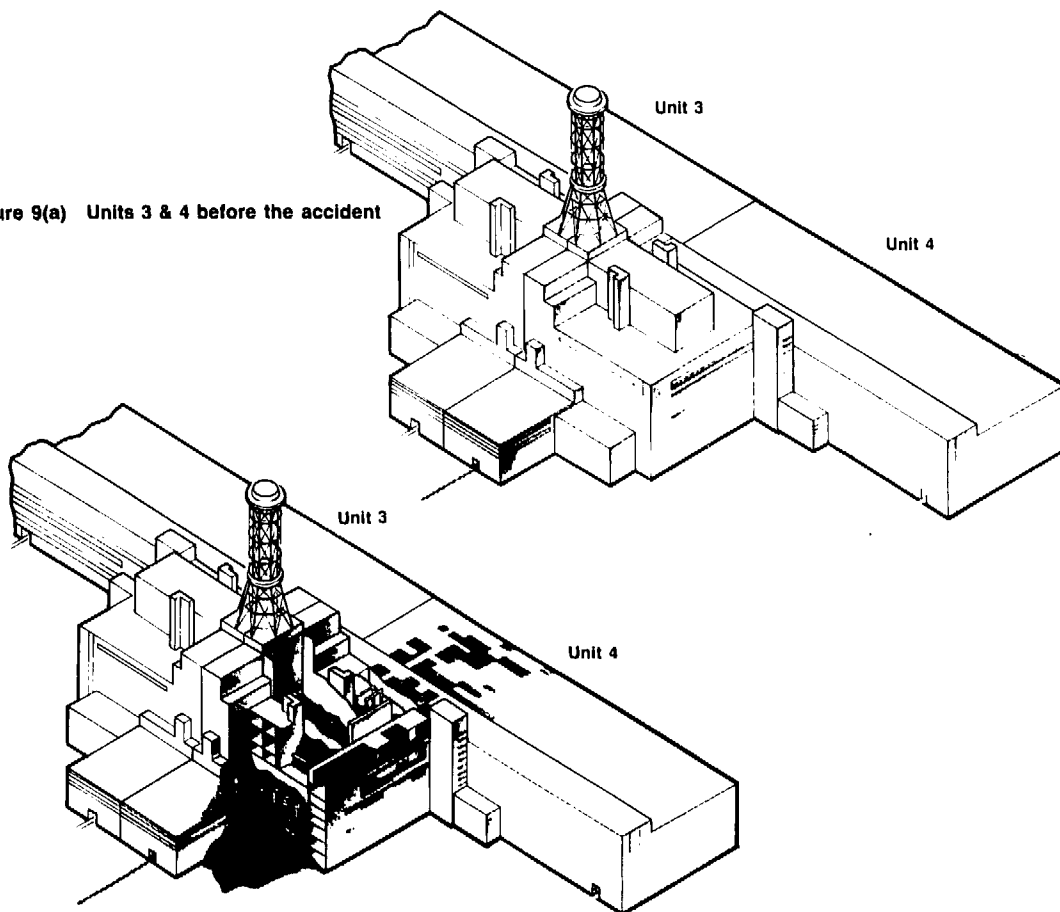


Figure 9(b) Units 3 & 4 after the accident

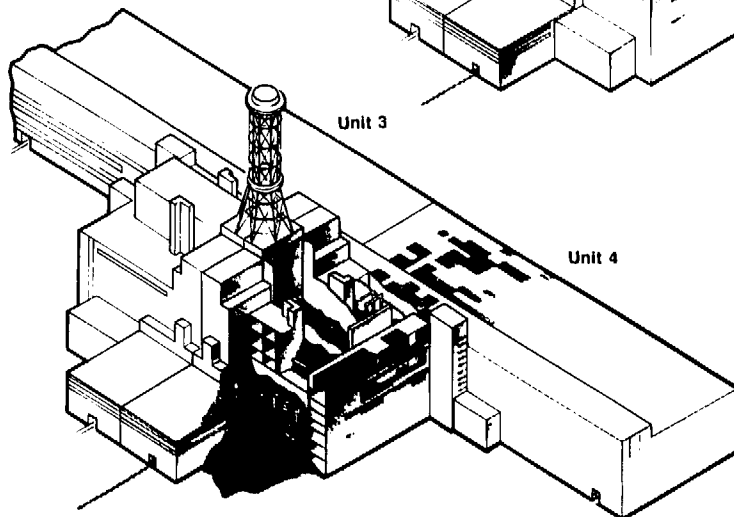


Figure 9(c) Unit 4 burial

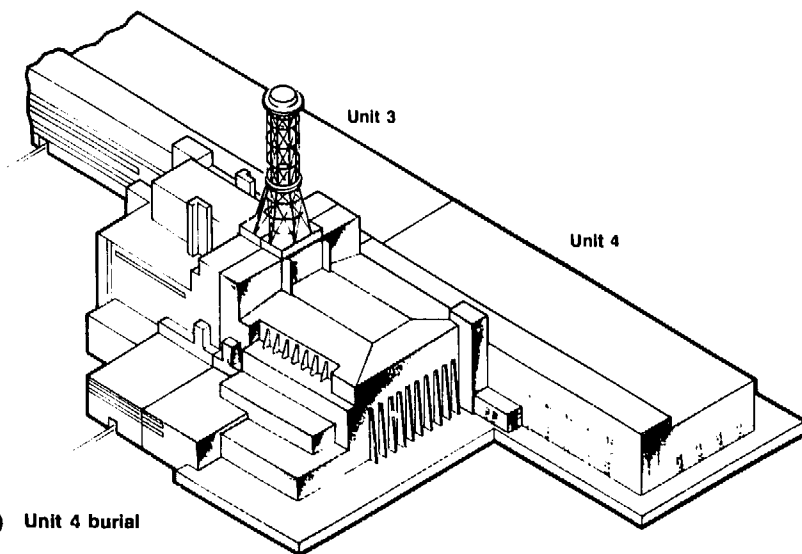


FIGURE 9 DAMAGE FROM THE ACCIDENT

thermal plume that carried combustion aerosols with finely dispersed fuel particles into the atmosphere. The fuel temperature dropped to about 900 K and the releases each day were substantially smaller than the phase 1 releases. As noted, the Soviets began to cover the damaged core by dropping boron carbide, dolomite, sand, clay, and lead from helicopters. In total, about 5000 tonnes of material were dropped in the core between April 27 and May 10.

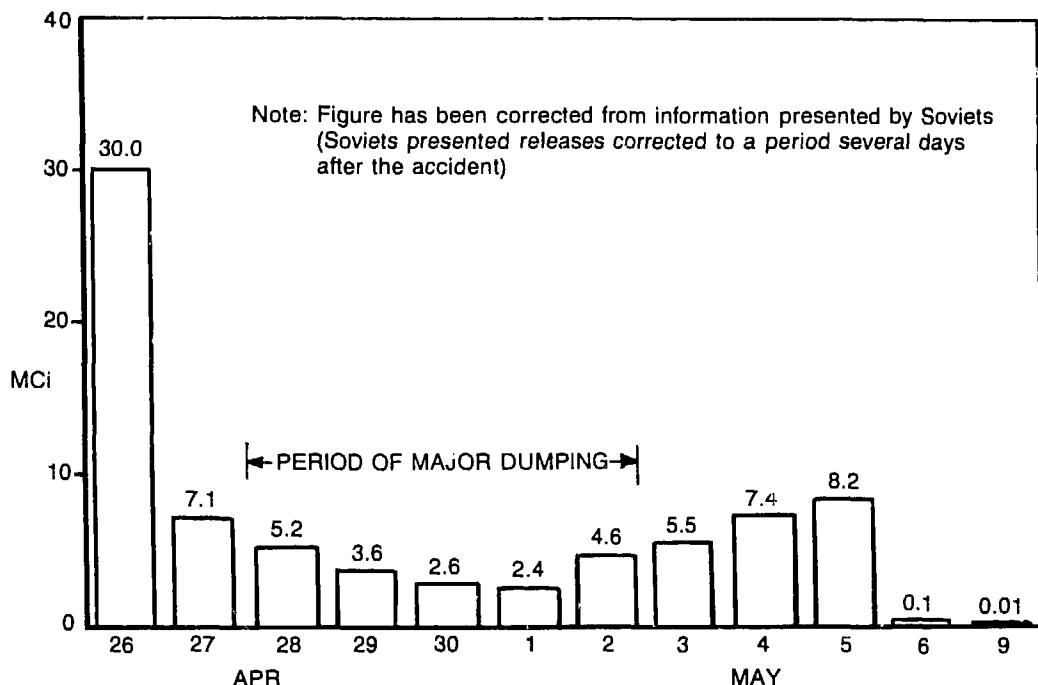


FIGURE 10 DAILY RADIOACTIVE RELEASES INTO THE ATMOSPHERE FROM THE ACCIDENT (WITHOUT RADIOACTIVE NOBLE GASES)

Phase 3 of the releases began about May 2 and was characterized by a sharp increase in the fuel temperature, due to the insulating layer of the material covering the core. The initial releases were dominated by volatile species (especially iodine). Eventually, the core reached approximately 2000 K on May 4. The heat and possibly oxidizing conditions caused an increase in the fission product releases that were similar in composition to the irradiated fuel.

At this point, the Soviets attempted to reduce the temperature, and to exclude oxygen from the core. Nitrogen was pumped from the compressor station into the space beneath the reactor vault and by May 6 the temperature in the core was dropping. This resulted in a sharp drop in the release rates (Phase 4) and by the end of May the release rates were a few tens of curies (about 2 TBq) per day. The Soviets have stated that the fuel was at that point being cooled by a stable convective flow of air through the core.

The total releases were stated to be 50 MCi (2×10^6 TBq) of noble gases (100% release) and 50 MCi (2×10^6 TBq) of other species. The possible error in the release numbers is stated to be about $\pm 50\%$. The 50 MCi (2×10^6 TBq) of non-noble gas releases is stated to be approximately 3.5% of the core inventory at the time of the accident.

For volatile species, the Soviets estimate that 20% of the iodine, 15% of the tellurium, and 10-13% of the caesium were released. In general, 2-5% of other less volatile species were released, including transuraniums.

5.1

Prompt Health Effect

At 2:10 a.m. on 26th April, 29 victims of radiation exposure were admitted to hospital. By 6:00 a.m., 108 persons had been hospitalized; eventually, 203 persons were diagnosed as suffering from acute radiation sickness. Two people died at the site; one at about 6:00 a.m. on the 26th; the other was not found. No member of the public has been observed with symptoms of acute radiation sickness.

The diagnosis and prognosis were based on the time of onset of chemical symptoms (e.g., vomiting, skin reactions), changes in the blood and chromosomal aberrations. The radiation doses that had been received by the victims were estimated from the clinical signs since no dosimeter values were available. The radiation doses appeared in most cases to have been fairly uniform over the bodies as judged by the distribution of chromosomal aberrations.

The distribution of victims and estimated doses was:

<u>Doses</u>	<u>No. of Patients</u>	<u>No. Died</u>
600 - 1600 rads (6-16 Gy)	22	21
400 - 600 rads (4-6 Gy)	23	7
200 - 400 rads (2-4 Gy)	53	1

No one with less severe effects died.

The prognosis and treatment of many of the victims was complicated by extensive skin burns - from beta/gamma radiation and heat. Although the Soviets had made a detailed study of radiation sickness and appeared to have had considerable experience in handling such patients, they had never handled such an extreme combination of burns and radiation exposure.

Internal contamination was important in causing death in only one case although radionuclides were absorbed through damaged skin and by inhalation by most victims. For two cases the doses from absorbed iodine and caesium isotopes were estimated to be 150 and 400 rem (1.5 and 4 Sv).

The absence of short-lived activation of biological tissues was interpreted as indicating that no neutron irradiation of the victims had occurred, confirming reactor shutdown at the time of the explosion.

Extremely detailed biochemical tests were carried out on the hospitalized victims in order to guide treatments. Bone marrow transplants were not effective and none of the 13 recipients have survived - either skin/intestinal damage or reactions to the transplant contributed to their deaths. The latter complication had not been seen before.

The main conclusions drawn by the Soviet medical team were that:

- a) their experience allowed them to develop criteria for handling the victims very quickly;
- b) there was a low need for, and in fact danger from, surgical intervention, particularly with the complication of beta skin burns;
- c) anti-infection therapy was most important; and
- d) observation of chromosomal aberrations was very helpful in the dose range from a few hundred to 800 rad to determine the appropriate treatment.

The good capability and performance of the Soviets in this field seemed to be generally acknowledged.

5.2 Doses to Members of the Public in the USSR

The Soviets have been able to perform only a rough calculation of the doses arising from the plume. For the town of Pripyat, just west of the 3km plant exclusion boundary, they estimated that the exposure was 10 to 15 rem (100 to 150 mSv). At 30km, the exposure was 0.2 rem (2 mSv).

External gamma irradiation from deposited radionuclides was very high close to the plant. Exposure rates in Prip'yat remained at about 1 rem/h (10 mSv/h) for many days. About 40 km² around the plant still had exposure rates above 100 mrem/h (1 mSv/h) a month later.

Doses to individuals were avoided by sheltering, ingestion of potassium iodide pills, and evacuation (see below). As a result, individual doses received by most of the population before evacuation, were believed to be not greater than 25 rem (250 mSv). Some villagers may have received doses as high as 40 rem (400 mSv).

The collective dose from external radiation to the evacuated population (135,000 people) was estimated to be 1.6 million man-rem (16000 man-sieverts).

Doses from inhalation of radionuclides occurred for the relatively short period during which the radioactive material in the air passed over and were considered to be relatively small. Of much greater importance was I-131 in milk and foodstuffs. Generally, milk products were banned at concentrations above 0.1 μ Ci/L (4000 Bq/L) in urban centres but in outlying areas consumption continued at concentrations up to 10 μ Ci/L (400,000 Bq/L) resulting in doses as high as 500 rem (5 Sv) in the thyroid. Nearly 100,000 people (mainly children) were assayed for iodine-131; half had doses up to 30 rem (0.3 Sv).

The Soviets estimate that the extra number of fatal thyroid cancers over the next 30 years would be 1500 in the total Ukrainian, Byelorussian and Russian Federation population of 75 million.

Caesium 134/137 is of long-term concern, both as a source of external radiation from ground contamination and as a contaminant of food produce. The estimated dose from external caesium radiation in the first year to an individual outside the 30-km evacuated zone ranges down from 1 rem (10 mSv); the total for 50 years ranges down from 3 rem (30 mSv).

The collective dose from this external radiation over the 75 million population is estimated to be 8.6 million man-rem (860,000 man-sieverts) in the first year and 29 million man-rem (290,000 man-sieverts) in 50 years. The average individual dose rates are 100 mrem/a and 10 mrem/a (1 and 0.1 mSv/a) for 1 and 50 years, respectively.

The doses that will be received from food products are very uncertain. Because of the high mobility of caesium and high plant uptake from the soils of Ukraine and Byelorussia, doses are predicted to be higher (per unit soil deposition) than usually assumed in models. The Soviets have estimated the collective dose from food products to the 75 million population to be 210 million man-rem (2.1 million man-sieverts) over 70 years (and their data indicate that 40 million man-rem (0.4 million man-sieverts) would be in the first year). They have, however, acknowledged that maximizing assumptions have been made in the calculations. The results of measurements of people, foodstuffs and soils support the idea that the actual doses will be much less - possibly by a factor of 10.

There seems to be little information on strontium-90 yet available. The Soviets suggest its contribution to doses will be less than from caesium but sufficient data will not be available for a few years (for both strontium and cobalt) for accurate dose estimates. Plutonium has been deposited over an extensive area. Air concentrations from resuspension were estimated to be close to 1 mBq/m³ as far away as 30 - 60 km. No doses have been estimated.

In summary, the radiation doses estimated so far for the USSR population could result in additional (fatal) cancers (and serious genetic effects) as follows:

Population Number	Exposure	Collective Dose (Million man-rem) (**)	Extra(*) Cancers	% Normal Cancers
135,000 (Evac. group)	External	1.6	320	1.6
75,000,000	External	29	5800	0.08 to 0.4
	Foodstuffs (Cs)	21-210	4200 to 42,000	
75,000,000	Iodine		1500	1

(*) Using 2 cases per 10^4 man-rem (per 10^2 man-Sv)
(**) 10^6 man-rem = 10^4 man-sieverts

5.3 Emergency Measures

5.3.1 Iodine (as iodate pills) was distributed to station staff and to patients at 3:00 a.m. on 26th April. Distribution to Pripyat and the surrounding population was started at 8:00 p.m. on 26th April. Pills were taken before the local population were exposed to significant amounts of radioactive iodine. The pill contained 0.25 g of iodine. Staff received one pill per day.

The effectiveness cannot be evaluated but there is confidence that the early distribution might have been effective. It certainly had a useful psychological effect. An important observation is that no toxic side effects have been observed; voice thinning out and throat inflammation were noted - but no blood damage. Observations are continuing.

5.3.2 Sheltering and Evacuation

Sheltering (i.e. staying indoors) is credited with 2-5 times reduction in doses from external radiation. It was very important in Pripyat since, although the decision to evacuate was made at 9:00 p.m. on 26th April, there was high risk in starting the evacuation because of the high radiation fields that would have to be crossed (the plume was not then over Pripyat). The population was told to evacuate at 11:00 a.m. on 27th April and the evacuation of the 45,000 inhabitants completed from 2 to 5 p.m. There was a 10 km line of buses.

Exposure information that was used for decisions appears to have been obtained from equipment brought in after the accident. There was no indication of any extensive field array

of radiological and meteorological equipment associated with the site. The closest micro-meteorology monitoring was at Kiev.

The number of medical and support staff involved to treat patients during the evacuation was high - 1,240 doctors, 920 nurses and more than 4000 assistants.

The criteria used by the Soviets for evacuation were very similar to those in other (European) countries. Below 25 rem (250 mSv) evacuation is not considered; from 25 - 75 rem (250 - 750 mSv) evacuation is considered; above 75 rem (750 mSv) evacuation is mandatory.

As could be expected, there were problems with the logistics of the evacuation, with decontamination of people, clothes, and with how to compensate for loss of personal possessions. Evacuation of rural populations was difficult as people were not easily persuaded to leave.

In addition to the evacuation of people, cattle were evacuated immediately on the 26th in open trucks - tens of thousands were removed.

The roles of the military and medical services were very important. A very strong central headquarters with wide powers proved to be essential. Medical brigades went with groups of evacuees to ensure continuity of medical services. Clear, direct decision-making was emphasized and an organization that allowed constructive exchange between medical, dosimetric and other experts proved to be essential.

5.3.3 Other Measures

In addition to banning foodstuffs, a major effort was made to safeguard the water supply for Kiev and surroundings. An alternative supply, from aquifers, appears to have been developed.

5.4 Followup

The Soviets are proposing to study the exposed population, although from the discussions it seems clear that their objectives and those thought appropriate by Western scientists are quite different. The Soviets emphasize studying individuals in great detail. The West would like to see an epidemiological study on a defined group with an adequate control group.

For follow up studies, only the evacuated population of 45,000 from Pripyat and the iodine-contaminated children were likely to yield useful data for risk estimates. However, the Soviets pointed out that those groups are now well dispersed (although tracked) and finding controls would be difficult. Dosimetry would be difficult for the evacuated population.

There is therefore a need to ensure that a satisfactory protocol is established for the study to be carried out. Collaborative efforts on this were welcomed by the Soviets.

CHANGES IN DESIGN AND OPERATION TO IMPROVE
THE SAFETY OF RBMK-1000

Since the Soviets attribute the cause of the accident to a series of violations of operating rules, their proposal to improve the safety of the RBMK-1000 is based on

- a) Changes to the operating rules which would render such violations virtually impossible.
- b) Changes to the plant design that would provide adequate protection against the full range of design basis events and against any adverse consequences of such violations.

6.1 Operating Rules

The Soviets emphasized human factors and operator error as the sine qua non of the accident. Their evaluation of the reasons for the violation of the numerous operating procedures were:

- 1) The staff felt that the test was a purely electrical one, and therefore were not concerned about, nor knowledgeable about, the hazards on the reactor side.
- 2) Chernobyl was considered a model plant, with less trouble than other RBMK types. Thus staff developed the impression that the plant was less prone to difficulty.
- 3) On the test itself, the reactor was scheduled for a maintenance outage after the test and the next test window would be a year away. Thus the operators were under pressure to complete the test.
- 4) They conceded that during the test it was difficult for the operators to see that the plant was in an un-analyzed state. Furthermore they did not know that the water in the channels was near saturation. They acknowledged that on the older units, in particular, an accident could produce too many confusing signals.

Their remedies proposed at the meeting for these problems were:

- 1) Criminal punishment for operators who violate procedures. Besides firing of the senior staff members at Chernobyl, many senior figures in the nuclear industry have also been discredited. A new Ministry of Atomic Power Engineering has been created to clarify lines of authority and improve operator training and reactor safety.
- 2) Some means of keeping the operator alert during long periods of trouble-free operation.

- 3) Adoption of procedures that will entail checks by several levels of supervision before permission is granted to deviate from the standard operational practice.
- 4) Improved man/machine interfacing to assist operators in decision making under abnormal operating conditions.

6.2 Design Changes

To mitigate any adverse consequences of operator error and violation of rules, the Soviets are taking two approaches in the short term.

- 1) They will improve the capability of the emergency protection system. This will be achieved by changing the limit on the minimum number of rods that are mobilized for emergency power reduction from 30 to 80. (Mobilization of a rod means the partial insertion of the rod from its parked position, which is outside the core, to a position from which an adequate rate of negative reactivity insertion can be achieved). In addition, an upper limit will be placed on the amount these rods can be withdrawn - namely a distance of 1.2 m below the top of the core. This limit is intended to ensure rapid shutdown action.
- 2) They will reduce the coolant void reactivity to the extent provided by the increase in the number of rods from 30 to 80.
- 3) Additional information (such as margin to fuel sheath dryout) will be made available to the operator.

In the longer term two changes have been planned:

- 1) The initial enrichment of the fuel will be increased from 2 to 2.4 wt% U-235. It is expected that this will result in a relatively undermoderated lattice. The moderating role of the coolant will then be enhanced. The loss of moderation on voiding will therefore contribute a negative component to void reactivity.
- 2) A new shutdown system may be added to enhance the negative reactivity insertion. The design of this mechanism was not disclosed but it was mentioned that injection of neutron absorber (liquid, gas or solid) into channels presently containing solid absorber rods may be adopted.

The Soviets have said that by making all these changes they achieve a net negative reactivity insertion rate of 1β per second (1β = delayed neutron fraction) for the most severe of their set of postulated credible accidents. It was not stated how this figure was arrived at or whether it was the average or the maximum negative insertion rate. They stated that with this negative rate, prompt criticality will be prevented for the most severe accident.

Although Units 1 and 2 were not damaged, they were contaminated. Units 1 and 2 were decontaminated and have restarted as of January 1987.

The fate of Unit 3 is less certain. In addition to contamination, the turbine hall was damaged by fire. There is significant contamination of the turbine hall towards the area where the Unit 4 turbogenerators are located. The Soviets did not disclose clear plans for the startup of Unit 3.

PART III KEY DESIGN ISSUES FOR CHERNOBYL

- III-1 VARIATION OF VOID REACTIVITY WITH REACTOR OPERATING STATE**
- III-2 SHUTDOWN SYSTEMS AND REACTOR CONTROL**
- III-3 CONTAINMENT**
- III-4 HEAVY OBJECTS ABOVE THE CORE**
- III-5 GRAPHITE MODERATOR**
- III-6 ASPECTS OF THE HEAT TRANSPORT SYSTEM AND LOCA'S**
- III-7 SOURCE TERM CONSIDERATIONS**

III-1 VARIATION OF VOID REACTIVITY WITH REACTOR OPERATING STATE

1. Background

In Chernobyl, if coolant is lost (voids) from the pressure tubes, there is a positive reactivity addition leading to a rise in power. In fact, the plant was designed to cope adequately with this effect at high power. It was not designed to cope with the effect at low power - because the size of the void reactivity effect was strongly dependent on reactor operating parameters. Because of the unusual conditions of the reactor just prior to the accident (i.e. low reactor power, only 6-8 control and shutdown rods equivalent in the core versus 30 required, high coolant flow through the core), there was an abnormally high void reactivity holdup.

Simulations done at AECL and in the U.S. DOE suggest the existence of a "positive shutdown". Normally the absorber rods have graphite followers to increase their worth. As they are inserted, the rod moves into the high-flux region in the centre of the core which was previously occupied by the graphite, so the rod effectiveness is enhanced. If there were no graphite, the rod would displace water - also an absorber - so the change in reactivity with insertion would not be as great. But in the accident, most of the absorbers were well removed from the core. The flux was peaked at the top and the bottom. Thus when the absorbers first started inserting, the water in the high-flux region at the bottom of the core was first displaced by the graphite follower, leading to a reactivity increase. If these results are correct, then, operating the plant in an abnormal condition resulted in an unusually large holdup of void reactivity exacerbated by a deficient shutdown system design (see Section III-2), which led to the large power excursion and resultant core damage.

2. Chernobyl Design

The characteristics of RBMK-1000 that affect void reactivity are:

- 1) The use of H₂O coolant.
- 2) The relatively high temperature ($\approx 700^{\circ}\text{C}$) of the moderator compared with that of the coolant (280°C).
- 3) A large and hence neutronically decoupled core (i.e. one which behaves like a number of independent reactors), and
- 4) The requirement of significant reactivity hold-down in solid absorber rods due to the use of enriched fuel and the need to be able to override xenon buildup and the impracticality of using soluble poison in a solid (graphite) moderator.

2.1 The Use of H₂O Coolant

The RBMK-1000 reactors are cooled with boiling H₂O. The mean coolant density is about 0.5 gm/cc and the mean exit quality is 14.5%. The relatively high absorption cross section of H₂O results in the reactivity of the coolant due to absorption of neutrons at this density being high. This is a major positive contributor to the lattice void reactivity in RBMK-1000.

2.2 Effect of Moderator Temperature on Void Reactivity

Since in the RBMK reactors, the moderator temperature is significantly higher than the coolant (700°C vs. 280°C), moderated neutrons entering the fuel channel are slowed down further to temperatures closer to the coolant physical temperature. This shift in spectrum is in the opposite direction to that in CANDU and the consequences of voiding affect the spectrum shift differently, in both the thermal and in the epithermal range. Furthermore, the magnitude of the effect is also higher because:

- a) The energy difference between neutrons entering the channel and neutrons being thermalized in the coolant is higher than in CANDU. This is due in part to the larger temperature difference between the moderator and the coolant in RBMK-1000 (700°C vs. 280°C) compared with CANDU (70°C vs. 290°C) and also due to the superior moderating ability of the H₂O vs. the D₂O coolant.
- b) The higher fuel exit burnup due to fuel enrichment in RBMK-1000 (19 MWD/kg(U) vs. 7 MWD/kg(U) for CANDU) leads to a larger variation in fuel composition during the life of the fuel. The effects due to the presence of plutonium are therefore increased. Notable amongst these is the increase on voiding in neutron flux at energies around the 0.3 eV resonance in Pu239.

The lattice void reactivity in RBMK-1000 is near zero for fresh fuel and as Pu 239 is produced becomes positive at equilibrium burnup and under normal operating conditions.

2.3 Neutronic Decoupling and Void Reactivity

The RBMK-1000 core has ≈ 1700 fuel channels. (The average power of a channel is < 2 MW.) In comparison the CANDU 600 has 380 channels (with an average power of > 5 MW). In spite of the large difference in physical size, the migration areas of the two lattices are quite similar in magnitude (396 cm² for RBMK-1000 vs. 370 cm² for the CANDU-600). This makes the RBMK-1000 neutronically large compared with CANDU-600. This is reflected in the relative subcriticality of the higher harmonics of the flux distribution in the two reactors. The RBMK-1000 has a first azimuthal mode subcriticality of between 6 and 7.5 mk. This means that an addition of this amount to the lattice

reactivity would make each radial half of the reactor critical and result in significant power redistribution between the two halves, i.e. the reactor is fairly close to behaving like two independent reactors, one at the top and one at the bottom. In comparison the corresponding value for the CANDU-600 is 17 mk i.e. the reactor behaves much more uniformly. The same is true for other harmonics. Therefore, void reactivity addition in the RBMK-1000 will result in a complex power shape which requires complex trip logic in order to recognize the accident in time.

2.4 Effect of Absorber Rods on Void Reactivity

In the Soviet literature on the RBMK-1000 design it is claimed that the designers have used the effect of flux redistribution on coolant voiding to reduce void reactivity. This is achieved by the combined use of manually operated absorber rods and coolant flow valves (to adjust channel void fractions) and proper fuel management; all to adjust the neutron flux distribution such that on voiding, the flux increases in the several sets of absorber rods (Figure 11). This leads to an increase in neutron absorption in the rods compared to that in the fuel and thereby produces negative reactivity.

The magnitude of the void reactivity coefficient changes with fuel burnup. The reason for this burnup dependence is explained in Section 2.2. According to the Soviets, for the fresh fuel the void reactivity coefficient is negative. For equilibrium fuel it is positive (about .05 mk/% void) for normal operating conditions i.e. with about 80 absorber rods partially inserted into the core. The maximum possible void reactivity coefficient is 0.2 mk/% void at normal operation at the minimum limit of 30 rods inserted in the core. The coefficient was as high as .3 mk/% void before the accident as there were only 8 rods in the core.

3. Comments on the Chernobyl Design

The size of the system void reactivity in RBMK-1000 can be controlled to a large extent by operational constraints. The safety of the reactor therefore is dependent on the competence of the reactor operators and on their adherence to these constraints. The system void reactivity in RBMK-1000 can become significantly higher under abnormal operating conditions. Such conditions include:

- a) reduction in the number of in-core absorbers with concurrent increase in fuel burnup, which is plausible during loss of refuelling capability, and
- b) reduction in reactor power level without a matched decrease in coolant flow rate.

In particular, the RBMK-1000 reactor is very sensitive to item (b). In order to maintain a similar coolant void level in the reactor core, the flow is normally reduced as the power is reduced. However at low powers (i.e., less than 20% full power),

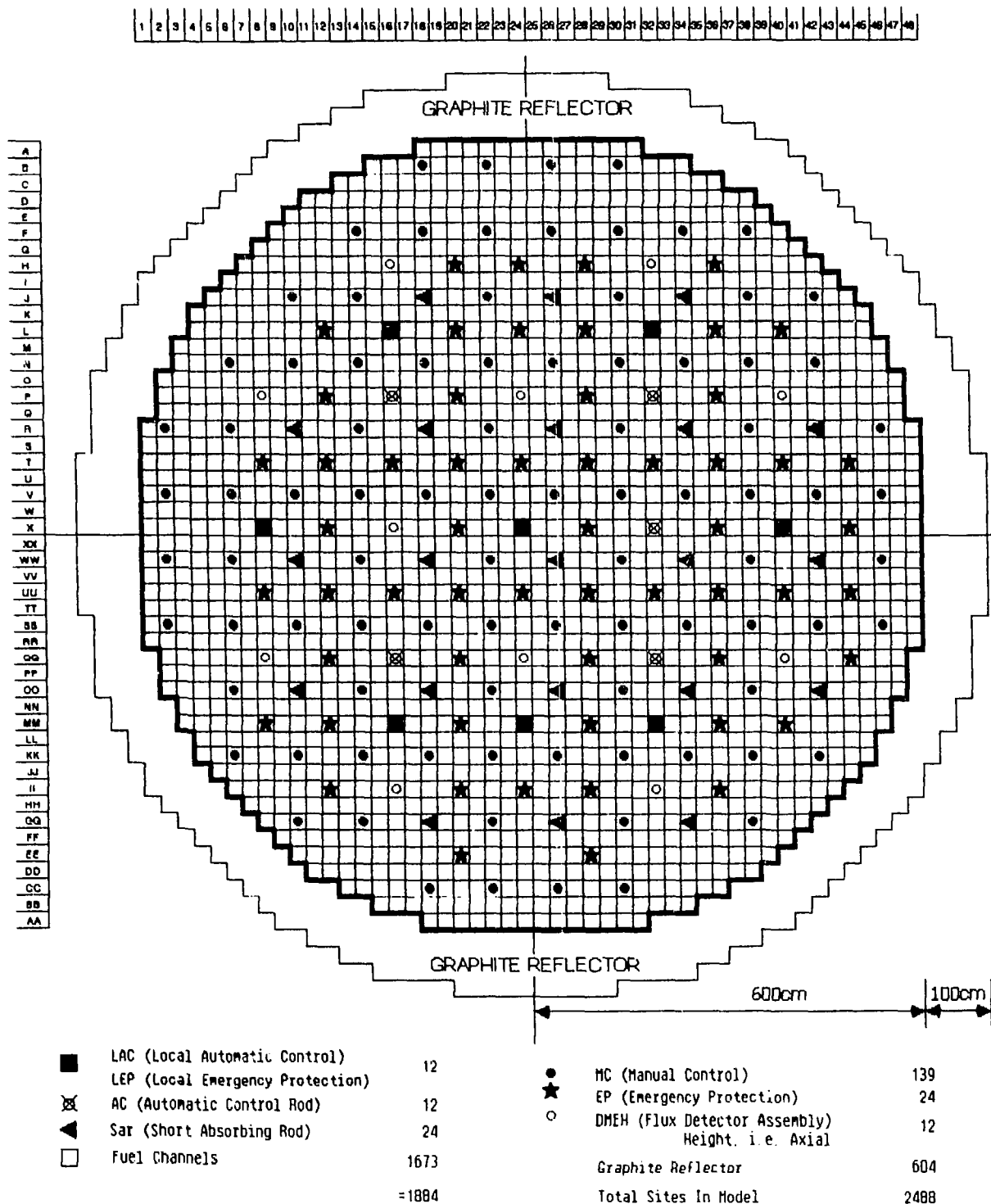


FIGURE 11 REACTOR PHYSICS MODEL OF RBMK-1000 CORE

the flow cannot be reduced to match the power and small changes in coolant conditions can have large effects on coolant void.

In particular (as happened at the time of the accident), at a flow much higher than would match the low power, the channel is almost full of water and the coolant density (void reactivity holdup) is high. Thus a small perturbation in coolant conditions was able to introduce a large void reactivity feedback, far beyond the design capability of the shutdown system.

To summarize then, the weakness of the Chernobyl design is that the void reactivity and the capability of the shutdown system depended significantly on the operating state of the reactor. The Soviets themselves have indicated that operating procedures did not allow operation (other than startup or shutdown) below 20% power.

4. CANDU Design

4.1 Void Reactivity

The heavy water (D_2O) coolant, the heavy water moderator and the natural uranium fuel are the major determinants of the void reactivity of the CANDU lattice.

In total, the lattice void reactivity in CANDU is 16 mk when the fuel is fresh and decreases with irradiation. At equilibrium fuel burnup, it is about 11 mk.

4.2 Changes in Neutron Spectrum on Voiding

The CANDU lattice pitch, which sets the volume of D_2O associated with a fuel channel, is chosen by mechanical considerations to facilitate on-power refuelling and by economic considerations to maximize fuel burnup by adjustment of the rate of neutron absorption in U^{238} (initial conversion ratio) and thereby of plutonium production. As a result, the standard CANDU lattice consists of a 10 cm I.D. fuel channel arranged on a square pitch of 28.6 cm.

The amount of moderator contained in the lattice produces a well-thermalized neutron spectrum in the fuel. Over 95% of the neutron population in the fuel has energies below 0.625 ev. Thus, the role of the D_2O coolant as moderator is not that significant. In fact, since the moderator temperature in CANDU is held between 70 and 80°C, moderated neutrons entering the fuel channel are unthermalized by inelastic scattering in the coolant as coolant temperature is significantly higher, $\approx 300^\circ C$. Thus, if the coolant is lost from the fuel channel, there is a small shift in the neutron temperature (or velocity) of the neutron population in the fuel towards the moderator neutron temperature.

This shift in neutron spectrum in the fuel alters the fuel neutron absorption rates in the thermal and in the

epithermal range. In particular, the resonance absorption in U238 decreases and there is a 6.3 mk increase due to the increase in the lattice resonance escape probability. Since depletion of U238 is minimal during the life of the fuel, this contribution to the void reactivity is almost constant with fuel burnup.

Loss of neutron scattering due to loss of coolant also increases the neutron flux and reaction rates above the 1.4 MeV fast fission threshold for U238. This contributes 5.2 mk to the void reactivity for the standard 37 element fuel bundle design. Due to the negligible depletion of U238 this contribution is also almost constant over the life of the fuel.

The changes in spectrum affect the thermal reaction rates because of the non- $1/v$ behaviour of the uranium and plutonium cross sections. On voiding there is a 3% increase in the U235 neutron production rate which is larger than the increase of 2.5% in its absorption rate. The neutron production per absorption increases by 1.6%. The plutonium cross sections behave differently due to the presence of several resonances; the cooling of the spectrum on voiding reduces the absorption and production rates in the fuel. The net result is a decrease in neutron production per absorption.

So the contribution to void reactivity of the change in thermal reaction rates depends on the irradiation of the fuel because of the role of the plutonium isotopes and of the fission products.

In total, the lattice void reactivity in CANDU is 16 mk when the fuel is fresh and decreases with irradiation. At equilibrium fuel burnup it is 11 mk. This is a relatively small variation compared with other reactor concepts, especially RBMK-1000.

4.3 Effect on Neutron Leakage

The CANDU equilibrium core contains fuel at all stages of irradiation starting with fresh fuel to fuel at exit burnup. Since the lattice void reactivity is relatively insensitive to irradiation, the increase in lattice reactivity on voiding is almost uniform over the voided volume of the core. Because of bidirectional fuelling, the irradiation averaged over a sizeable volume of the core is relatively constant, hence the void reactivity may be considered uniform over the voided volume. This is true for large variations in burnup distribution which can result from the use of different fuelling schemes and also for a temporary cessation of fuelling due to unavailability of the fuel handling system. This property of the CANDU lattice puts a limit on the change in global neutron flux distribution due to coolant voiding and thereby limits the change in neutron leakage. For both fresh and equilibrium cores the neutron leakage on full core voiding changes by less than 1 mk.

4.4 Effect of Absorber Rods on Void Reactivity

In CANDU, the mechanism that leads to a change in void reactivity due to the presence of absorber rods is quite different from that in Chernobyl. Voiding of the coolant in CANDU results in a small decrease in the thermal neutron flux in the moderator. This means that if there are absorbers present in the moderator (such as adjusters), their neutron absorption rate will drop. This effect is included in the 11 mk of void reactivity given in Section 4.1.

5. Comments on the CANDU Design

It should be noted that all power reactors require rapid shutdown capability, regardless of their inherent feedback effects. Thus, a sudden void collapse in a boiling water reactor, or rapid cooldown on the secondary side of a pressurized water reactor, generate reactivity transients which must be quickly terminated. Furthermore, the inherent characteristics of reactivity feedback must be evaluated in the context of other design features. In CANDU for instance, shutdown system action cannot be significantly impaired by a LOCA (since the devices enter the low pressure moderator environment).

In direct contrast to the key weakness in the Chernobyl reactor design, the CANDU reactor physics is such that the shutdown systems can shutdown the reactor, essentially independent of the operating state of the reactor. To confirm this, detailed reactor trip effectiveness studies for the full range of initial power levels and reactor states have been performed, for each shutdown system acting alone. This has formed a part of the licensing assessments of the CANDU reactors. In summary, any reactivity coefficient, whether positive or negative, large or small, fast or slow can be accommodated by the appropriate design. It simplifies the design however to keep them small, as in CANDU, and also by matching the shutdown capabilities to the reactivity coefficients.

1. Background

The accident was characterized by an ineffective shutdown.

2. Chernobyl Design2.1 Overall Philosophy

Reactivity protection (shutdown) and control in the RBMK reactor is quite complex and requires quite a bit of manual involvement, and the separation between protection and control systems is very limited (see Figure 12).

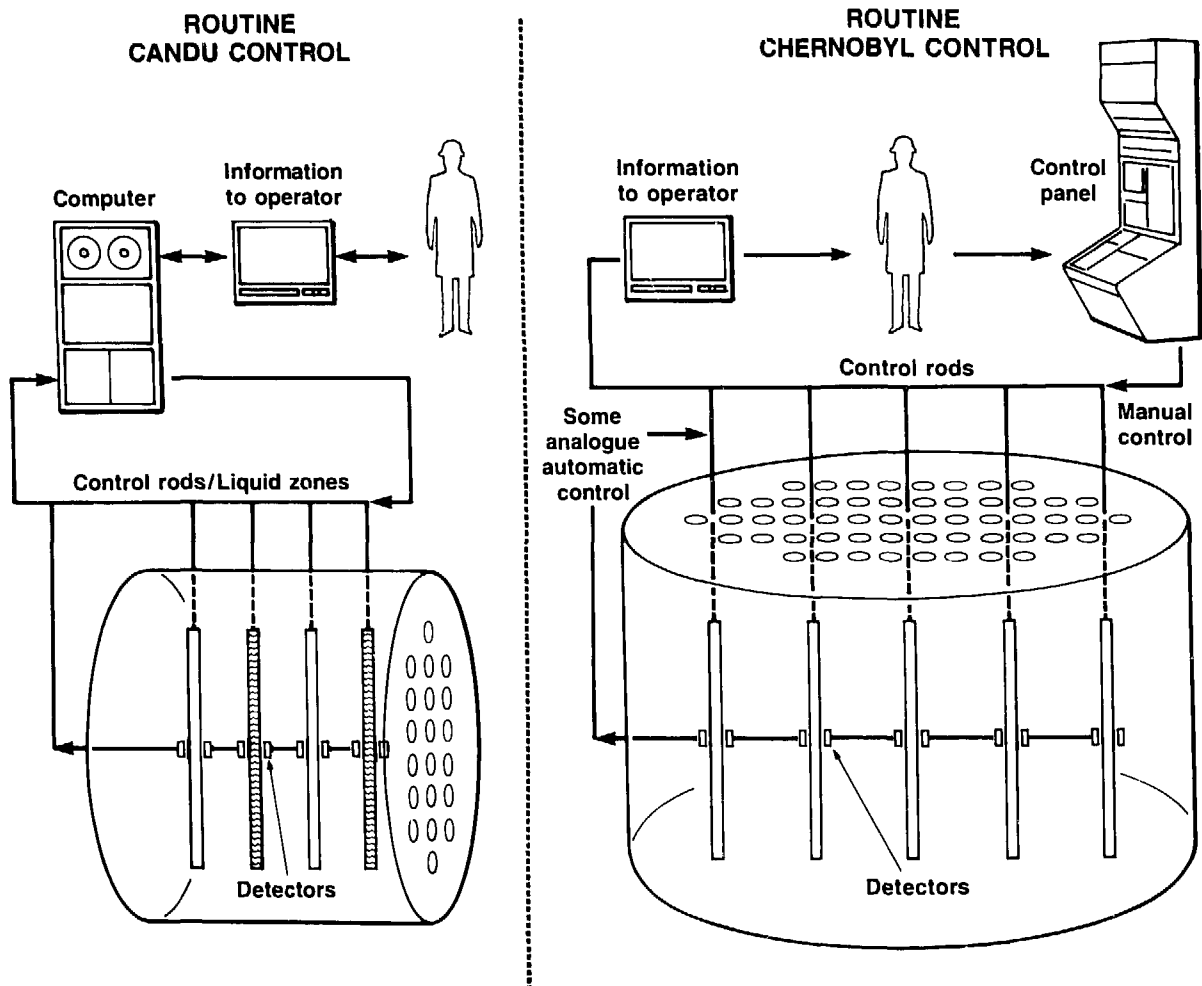


FIGURE 12 CONTROLLING THE POWER

The control function of the RBMK-1000 reactor is divided into:

- 1) Bulk reactivity control for power manoeuvring and for maintaining criticality in the presence of perturbations caused by absorber rod movement or by feedback reactivity.
- 2) Control of flux and power distribution in the radial plane to limit channel power.
- 3) Emergency reduction of total reactor power to safe power levels when necessary.
- 4) Emergency reduction of local reactor power to safe power levels when necessary, and
- 5) Emergency shutdown of the reactor with the insertion of all absorber rods at their maximum speed.

Demands on the absorber rods are made according to certain rules. The auto-control system attempts to meet these demands. If the operator finds that the auto-control system is insufficient, he inserts or removes "supplementary" absorbers manually.

The number of supplementary absorbers present at any time depends on a combination of factors. Some of these are:

1. The extent of power shaping required
2. The neutron poison override capability that was required.
3. The operating value of the coolant void reactivity.

As the demand on the auto-control system increases, supplementary absorbers are driven in or out by the operator to keep the auto-control absorbers in their range of travel. However 24 absorbers are normally kept outside the core to provide reactivity depth on reactor shutdown.

2.2 Required Rod Positions

A significant feature of this mode of operation is that the maximum negative reactivity rate that will be achieved in an emergency shutdown depends on the number of supplementary absorbers that are present in the core and on the neutron importance of their positions. For this reason the equivalent of at least 30 absorbers are always required to be inserted at least 1.2 m into the core. This rule was violated prior to the accident. A significant feature of the rod design is the ingress of water into the bottom of the core that occurs when the absorber is pulled out of the reactor.

2.3 Bulk Control

Automatic control of total reactivity (or total power) is provided over a range of about 0.5% to 100% full power. The control system appears to be entirely analog rather than digital.

2.4 Spatial Control

The majority of the spatial control rods (139) were manually operated. The operator would use recommendations from the plant monitoring computer as well as direct indication of flux distribution from 130 radially distributed and 84 axially distributed (7 at each of 12 locations) in-core flux detectors (see Table 2). Chernobyl also had a limited number of spatial control rods (12) which were automatic (see Table 3).

The automatic spatial control were designed to stabilize the most important radial and azimuthal flux modes. The twelve control rods are moved in such a way that the signals from two fission chambers near each control rod remained at a specified value. This system can operate between 10% and 100% of full power and also controls the total reactor power when it is active.

2.5 Emergency Shutdown

The emergency protection (shutdown) is designed for both bulk and spatial power excursions. Protection is based on three types of signals:

- 1) Ion chambers outside the reflector are used for high flux and high rate trips. One description states that rate is monitored only below 10% FP. Some degree of spatial protection is afforded by tripping if setpoints are exceeded at two ion chambers on the same side of the reactor. A total of eight ion chambers is used by the protection system.
- 2) Two fission chambers are located near each of the automatic spatial control rods. Both chambers near one rod must exceed their setpoint to initiate protective action. There is no reference to a rate trip on these measurements, nor any indication of the power range over which the instruments are effective.
- 3) 130 radially distributed in-core flux detectors (using a silver emitter) are compared to appropriate precalculated setpoints, and a partial forced power reduction is initiated by the protection system if the setpoint is exceeded. This system is stated to be effective only above 10% full power (FP). The detectors have a slow response (25 second time constant), so this system would be of no use during a fast excursion in power.

In summary, the ion chambers would give only poor spatial protection, but their response is prompt. The fission chambers give better coverage, but there are only a few detectors to cover a large core. Fission chambers are usually also prompt in their response. The in-core detectors give very good coverage, but have a slow response and are therefore useless for fast-developing accidents.

3. Comments on the Chernobyl Design

The RBMK protection system is fundamentally different from the CANDU shutdown systems (see Figure 13). In the RBMK design the action is not necessarily a full shutdown; under some conditions only a partial power reduction is initiated (similar to the CANDU power control action called stepback).

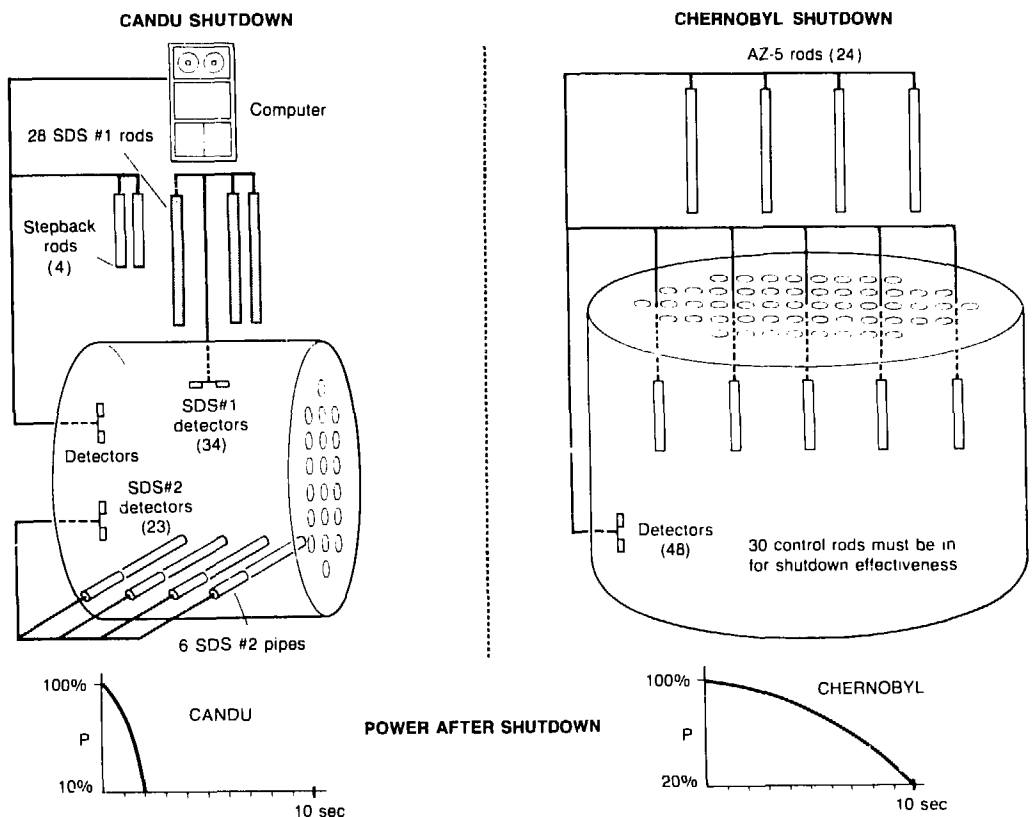


FIGURE 13 SHUTTING DOWN THE REACTOR

The emergency rods are complex devices which can be inserted at various rates, the fastest of which is very slow (about 10 seconds) compared with CANDU shutoff rods (about one second). Trips do not appear to be locked-in; when a flux reading is no longer high, rod insertion is interrupted. Rods do not appear to be rigidly assigned to the control or protection systems; some appear to serve a dual role. Such a system can

probably be made to work adequately but it does have significant weaknesses compared to the CANDU design.

Physics assessments show that the Chernobyl reactor is potentially subject to very local, very large flux perturbations. Less than 10% of the core can sustain criticality. From what we know of the protection system sensors, those which are widely distributed are very slow to respond and would not adequately protect against any reasonably fast power increase, while those which respond quickly are small in number and would not adequately see a very local power increase.

Finally, and most significantly, the protective system action is very slow, so that a power excursion is likely to see a significant overshoot before it is turned around. In addition, as noted earlier, given certain analysis assumptions on fuel burnup distribution, and shutdown system design, it is possible that the shutdown system itself may have exacerbated the accident by inserting positive reactivity during the first few seconds of its initiation.

4. CANDU Design

CANDU stations control reactor power automatically over the entire range from six or seven decades below full power up to full power. Spatial control is done only above about 15% FP because the reactor is spatially stable up to about 25% FP. At low powers, up to about 10% FP, control is based on ion chambers, while at high powers flux detectors are used. Both types of measurement are totally prompt for all practical purposes.

Reactivity control at all power levels, both for bulk and for spatial purposes, is based on the 14 zone controllers (see Figure 12). If their worth is inadequate, rods are available for both positive and negative reactivity addition, again under totally automatic control. Manual reactivity adjustments are limited to poison addition and removal to the D₂O moderator, both of which are very slow and relatively rarely required.

Protection against reactivity insertion accidents is provided partly by the control system itself, via stepbacks on high log rate and high flux, but mostly by powerful, rapid, shutdown. In CANDU 600, shutdown system #1 consists of 28 gravity operated, spring-assisted absorber rods, and shutdown system #2 consist of six liquid injection pipes containing over 200 nozzles. Each system is independently, fully capable of shutting down the reactor for all accidents. Each system has its own detectors, amplifiers, relays, logic, and actuating mechanisms and is independent of the control system and of the other shutdown system.

In particular each has high rate and high flux trips. These trips have been studied quite extensively in terms of their trip coverage (i.e., the range of initial power level and reactivity rate for which trips are effective), and are found to

be fully comprehensive. Any fast power increase would be terminated by the rate trips, while slow increases continue until the high power trip is exceeded, without core damage.

The emphasis on shutdown performance, and independence from reactor control, are hallmarks of Canadian safety philosophy going back to early days of power reactor development in Canada. Indeed the Pickering A units put into operation in the early 1970's (near Toronto, Ontario) have two different shutdown mechanisms (shutoff rods, and quickly draining the heavy water moderator). The shutdown is fully independent of the control, and unlike Chernobyl, capable under any accident conditions of shutting the reactor down. The two shutdown mechanisms are not as powerful as in later CANDU designs (Pickering B, Bruce A and B, CANDU 600 and Darlington A), and the logic is not as separated. Offsetting this, the measured reliability of shutdown in Pickering A is much better than called for in the original design requirements, and shutdown is effective in preventing serious consequences even if a few of the rods do not work. Even the NPD reactor, a 25 MW(e) demonstration of the CANDU pressure-tube concept which went into operation in 1962, has a single shutdown system which is fully independent of the reactor control system and with an availability target of greater than 9,999 out 10,000. There have been no shutdown system failures on test in 27 years of operation, and the predicted future availability approaches the combined target for plants with two independent shutdown systems.

The required response speed and reactivity depth of the shutdown systems are governed by accidents other than loss of reactivity control. As a result, the systems are more than capable of handling any conceivable reactivity insertion, from any initial power level (see Figure 13).

5. Comments on CANDU Design

The CANDU design is especially sound in the area of spatial control (at all ranges of power level) and protection. The CANDU ion chambers and flux detectors give full trip coverage in both shutdown systems; the measurements are very fast; the shutdown action is very fast (less than two seconds) and deep; the shutdown systems are totally independent of the control system.

TABLE 2

SUMMARY OF FLUX MEASUREMENT DEVICES IN CHERNOBYL

3	Start up counters
3	Low power ion chambers
12	Ion chambers for control of total power (used 4 at a time)
8	Ion chambers used for protection
130	Radially distributed silver flux detectors for - computer monitoring - alarm on relative deviation (above 5% FP) - alarm and protection action on absolute limit (above 10% FP)
84	Axially distributed silver flux detectors for - computer monitoring - alarm on relative deviation
24	Fission chambers for - automatic spatial control - local protection

- Notes:
- 1) The silver flux detectors have a full power current of 15 microamps; except for electronic equipment limitations, they should be good down to a few per cent of full power. Their response is about a 25 second time constant for 90% of the signal and a 2.4 minute time constant for 10% of the signal. The burnout rate is about 20% per year, and the expected life about three years.
 - 2) Different versions of RBMK-1000 may have different instrumentation. The above list is indicative only.

TABLE 3

SUMMARY OF CONTROL RODS IN CHERNOBYL

12	Automatic control of total power (used four at a time)
12	Automatic spatial control
24	"Short" rods for manual axial control
139	Regular rods for radial/azimuthal manual control
24	Emergency protection

Note: Different descriptions have different numbers of rods. The above list is indicative only.

III-3 CONTAINMENT

1. Background

Most power reactor designs throughout the world have used the design concept of several barriers (or defense-in-depth) for the prevention of radioactive releases to the public. It is clear that the Chernobyl containment failed to perform its intended containment function. We believe CANDU containment, by virtue of a more comprehensive enclosure would be much more competent (less chance of bypass of containment) in accident situations.

2. Chernobyl Design

The Chernobyl Unit 4 RBMK 1000 reactor was fitted with a containment consisting of:

- 1) enclosures covering parts of the reactor and cooling system designed to withstand approximately 100 to 400 kPa(g) (15 to 60 psig),
- 2) a pressure suppression system which functions by forcing discharged steam through water pools,
- 3) a sprinkler cooling system,
- 4) hydrogen removal systems intended to cope with limited hydrogen production,
- 5) ventilation and filtering systems,
- 6) a very tall stack.

The upper end of the reactor and fuelling machine is not within a pressure-retaining containment enclosure. There is a conventional building covering the fuelling machine area. This building and its ventilation system play a role in collecting small discharges in that area.

2.1 Core Container

Information provided indicates that the core of the reactor including the channels and the graphite is contained in a low design pressure (about 200 kPa (30 psi)) tank filled with inert gas. This tank is fitted with relief valves which lead down into the bubbler pond. A helium/nitrogen mixture is circulated through this tank during normal operation.

2.2 Reactor Building

The fuelling machine and the top of the reactor are enclosed in a building of conventional structure which was blown away during the course of the accident.

Figure 14 shows a schematic of the gas tight enclosure which covers many parts of the reactor. Figures 17 and 18 show features such as:

- 1) double water pools (bubbler ponds) which condense steam from main steam safety valves as well as accidents.
- 2) a complex valving arrangement between compartments. This arrangement swaps the "wet well/dry well" depending on failure location. This design is aimed at minimizing containment volume and design pressure.
- 3) a sprinkler cooling system for cooling of air during normal operation and after accidents.
- 4) a system to remove hydrogen from the enclosure. Sources of hydrogen are controlled by catalytic combustion. The system has a capacity of 800 m³/h and is designed for a postulated release of hydrogen from the oxidation of 30% of the fuel sheaths.

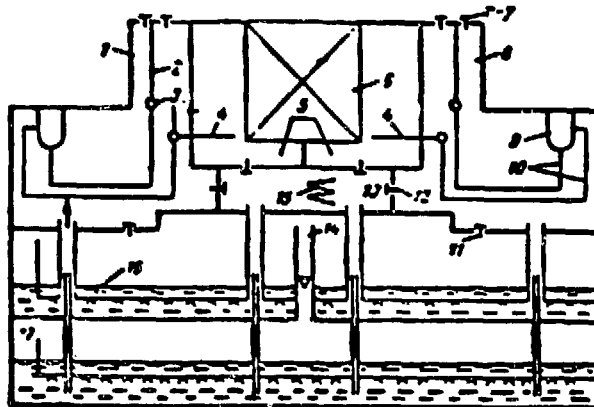


Fig. 1. Schematic diagram of gas-tight box: 1, 8—gas-tight box (emergency and non-emergency halves respectively); 2—downcomer lines; 3—collectors of main circulating pumps; 4—distributing group collectors; 5—lower water line compartment; 6—reactor; 7—safety valve; 9—main circulating pump; 10—main circulating pump lines; 11—bypass valve; 12—check valve panel; 13—lower water line check valve; 14—overflow tube; 15—surface type heat exchangers; 16—steam dump lines following main safety valves; 17—bubbler tank

FIGURE 14 SCHEMATIC OF CHERNOBYL CONTAINMENT

Figures 15 and 16 show additional details. (These figures are of Smolensk - another RBMK design. The same figures are used in the accident report prepared by the Soviets.) Features identified include:

Figures 15 and 16 show additional details. (These figures are of Smolensk - another RBMK design. The same figures are used in the accident report prepared by the Soviets.) Features identified include:

- 1) a concrete structure coincident with the gas-tight box.
- 2) a number of box compartments with a total approximate volume of $60 \times 50 \times 40 \times 1/2 = 60,000 \text{ m}^3$ assuming half of the box is filled with equipment.
- 3) a turbine enclosure.
- 4) a ventilation system which allows for access in some areas and includes aerosol and iodine filters and which discharges to the 150 metre stack.

Figure 17 identifies features of the pressure suppression system. Note that:

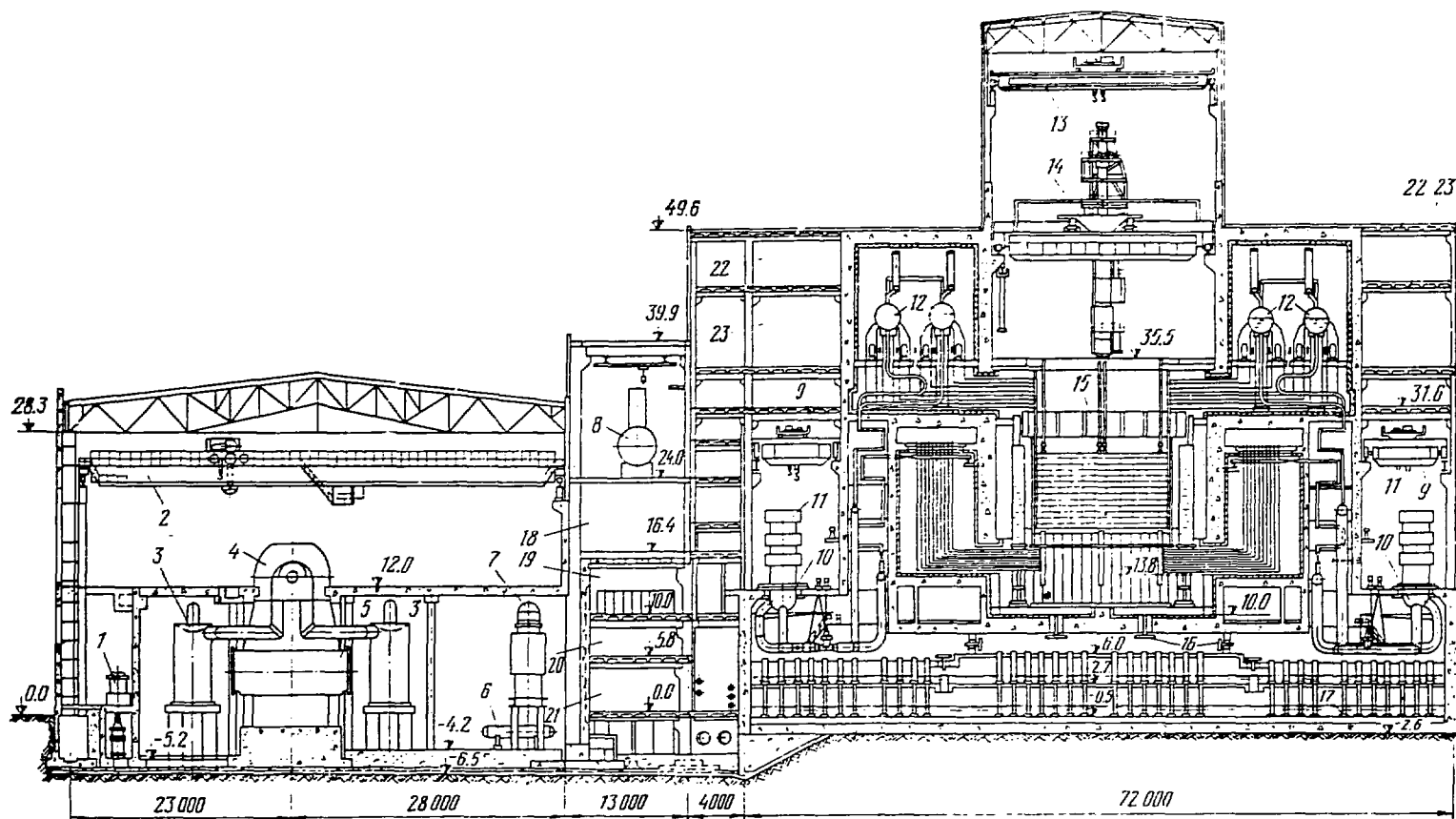
- There is a design pressure of 0.45 MPa for pump chambers and 0.08 MPa for lower feed pipework (Chernobyl #4).
- There is no indication of steam line isolation valves.

3. Comments on the Chernobyl Design

There are several clear weaknesses with the Chernobyl containment, in that there are a number of pathways by which activity released from fuel in the reactor core could directly affect the reactor operators or public.

- i) Failures in the steam separators or reactor outlet piping can allow fission products to escape via the removable shielding blocks which form the floor of the reactor hall. It is possible to assume that the Soviet rationale is that large piping (and the steam separators) is unlikely to fail, and would likely leak before break in any case. Breaks in the reactor outlet piping would be limited to one channel, and the affected channel and other channels could reasonably be expected to be cooled by the emergency core cooling system. If so, significant numbers of fuel failures would be unlikely.
- ii) Since the reactor is a direct-cycle design, failures in steam lines or main steam safety valves can allow fission products to escape. There are no obvious ways to isolate the reactor from these pathways (eg. main steam isolation valves). Failing open of the main steam safety valves is covered as they relieve to the pressure suppression pool which could handle the discharge for some period of time.

FIGURE 15 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-500-05/3000 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

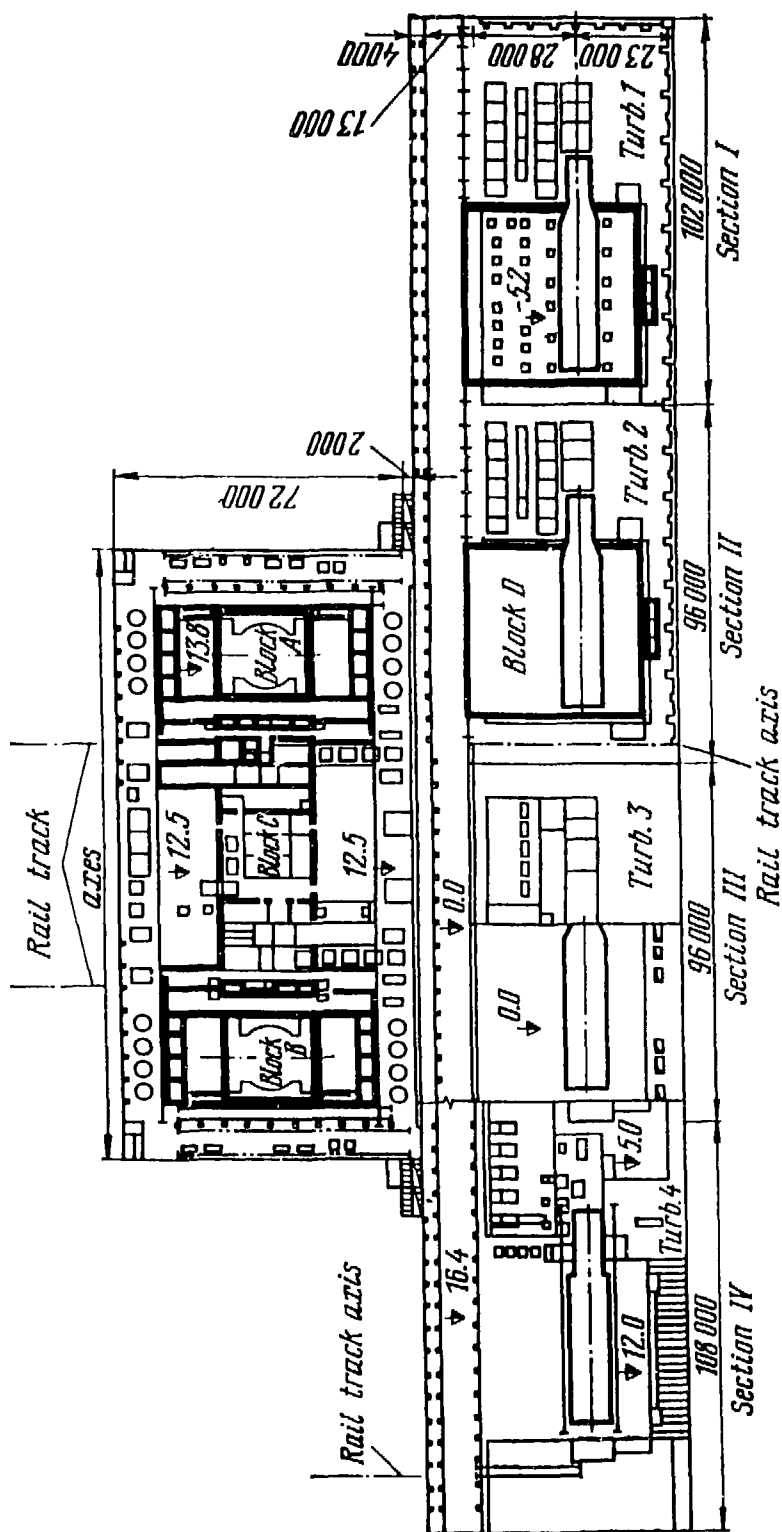


FIGURE 16 PLAN OF THE MAIN BUILDING AT SMOLENSK

- iii) Failures of the cooling of spent fuel in the fuelling machine would not be contained but the consequences are limited to one or two channels worth of fuel.

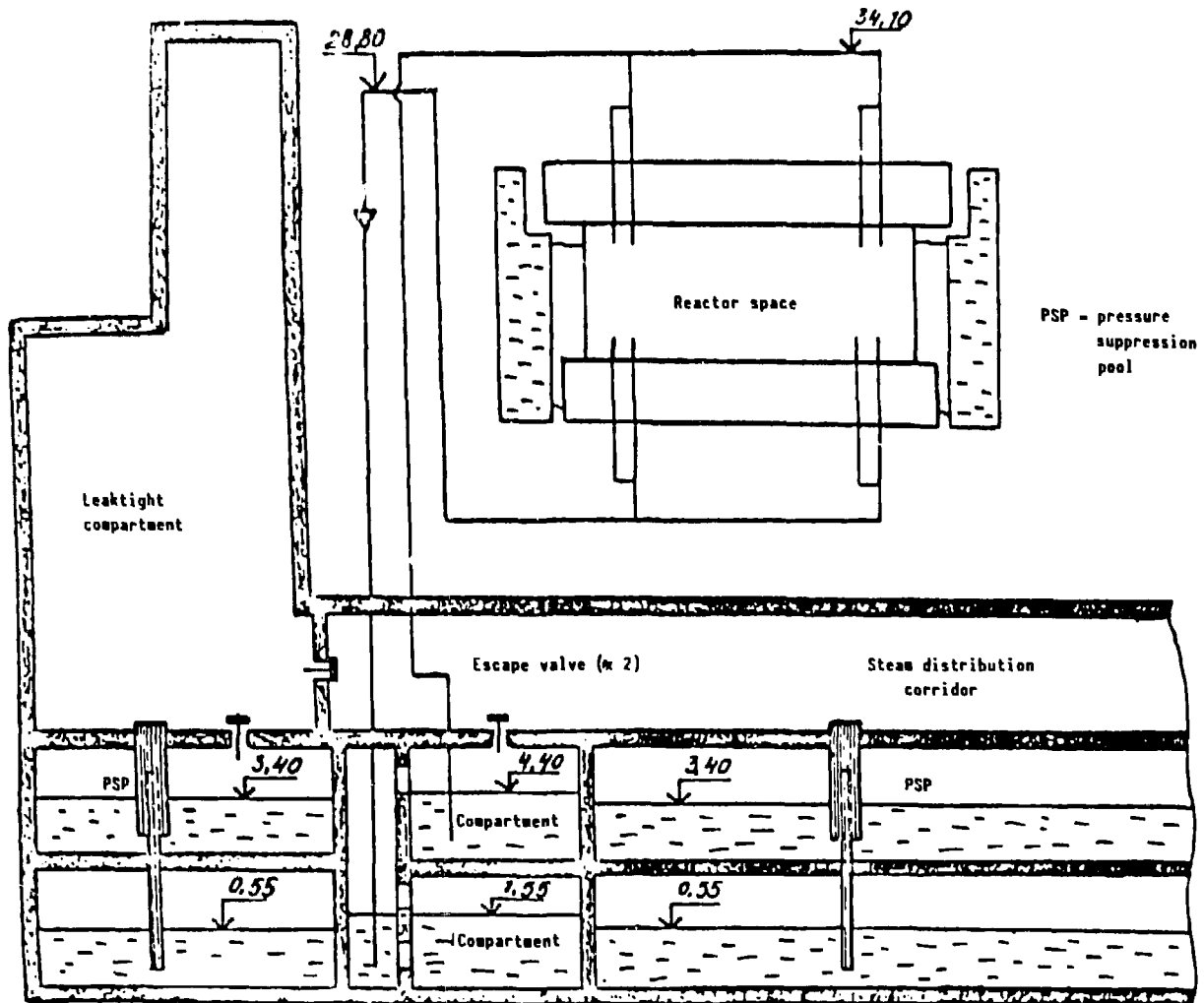


FIGURE 17 SYSTEM TO PROTECT THE REACTOR SPACE FROM EXCESS PRESSURE

4. CANDU Design

There are three different and effective containment designs used for CANDU plants:

- i) The single unit containment envelope, Figure 19, encompasses the reactor core, all major components of the primary and secondary coolant systems, the moderator system and the refuelling mechanisms. Some lines (such as ventilation) may be open to the outside atmosphere during normal operation. These lines are closed should an accident condition be detected.

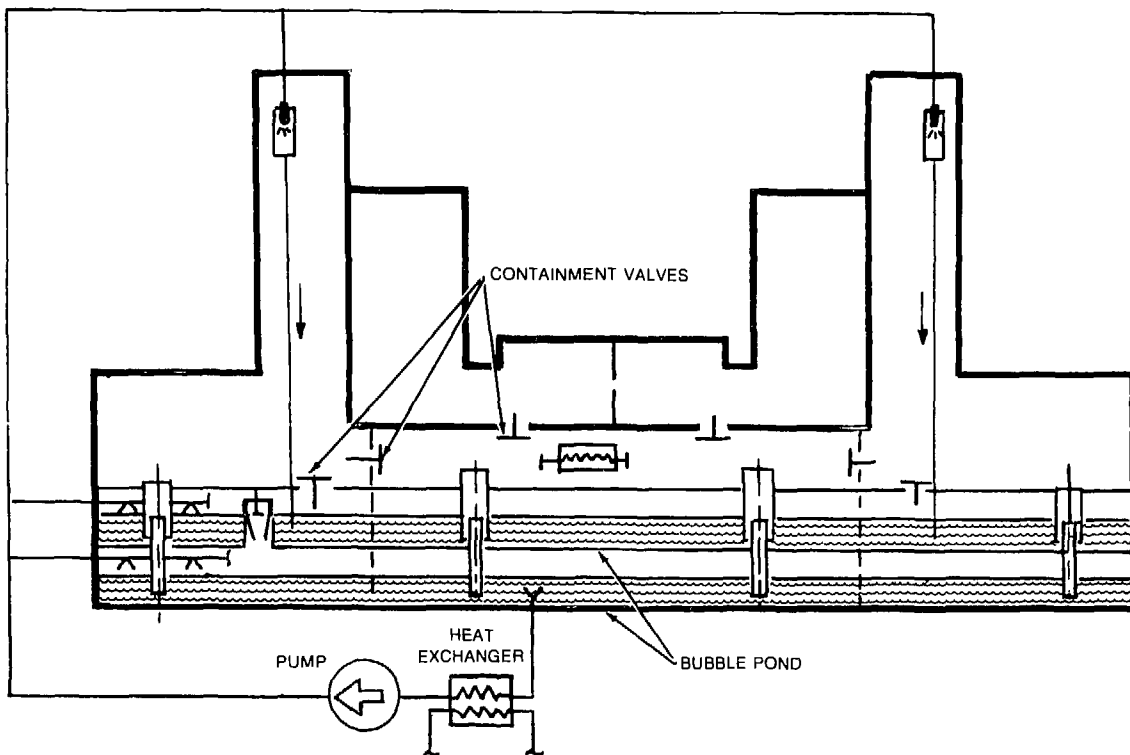
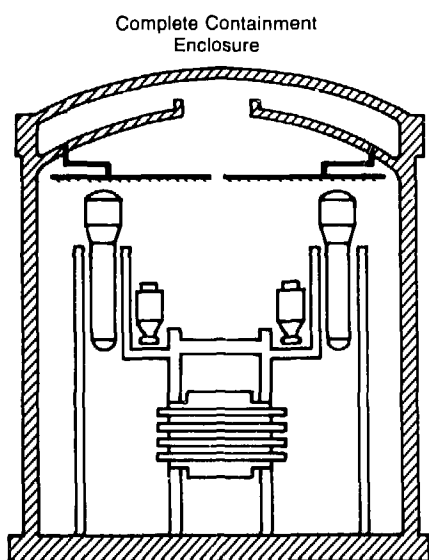
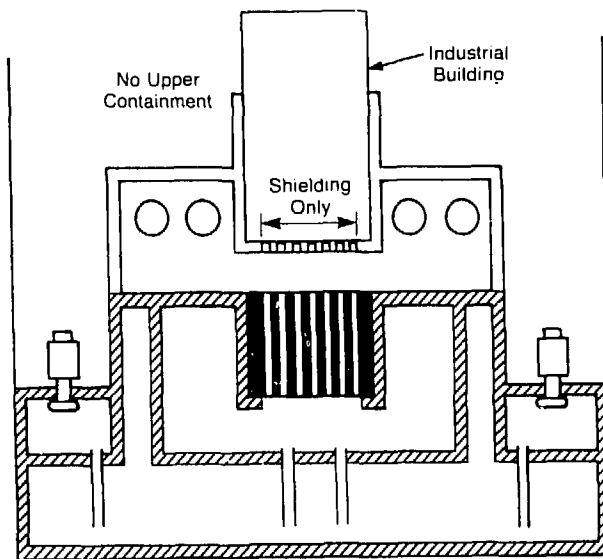


FIGURE 18 SCHEMATIC DIAGRAM OF THE CONFINEMENT SYSTEM



600 MW CANDU



Chernobyl

FIGURE 19 CONTAINMENT STRUCTURES

- ii) The multi-unit reactor stations all have negative pressure containment systems with a vacuum building which takes the enclosure below atmospheric pressure after an accident.

Bruce and Darlington designs also enclose most of the reactor auxiliary equipment. The primary coolant pumps and primary piping systems are inside the containment enclosure, but the pump motors are outside containment and the drive shaft seal forms the containment boundary.

- iii) In all CANDUs the steam cycle is indirect and thus the light water in the secondary side does not circulate through the core. Hence, escape of fission products from the reactor core past the containment boundary requires a breach of the boiler tubes. Single boiler tube failures are innocuous due to the tiny size of the tube and multiple boiler tube failures are extremely improbable. Even if they were to occur, most of the fission products would be retained in the turbine system.

- iv) The containment system for the NPD reactor is a pressure suppression/relief system rather than a pressure suppression/containment design. Its dousing system suppresses pressure and washes out fission products as in all CANDU's. However, for large piping failures which exceed the capacity of the pressure suppression, steam overpressure is initially relieved to atmosphere. Following relief of the initial discharge of steam, the building isolates to trap any fission products which may be generated as a result of an accident.

Release of these from the fuel would be delayed relative to the steam release.

Containment strength and volume (or their product) are a measure of retention capability. In CANDU practice, there is a defined design pressure, a test pressure about 15% above design pressure, a cracking pressure when the first through-wall cracks occur, and a failure pressure when the reinforcing bars yield.

In the case of the CANDU 600 reactors, e.g. Lepreau, these values are:

design pressure	:	124 kPa gauge	(18 psi)
test pressure	:	143 kPa gauge	(21 psi)
cracking pressure	:	~330 kPa gauge	(48 psi)
failure pressure	:	~530 kPa gauge	(77 psi)

The containment is designed for rupture of the largest main cooling pipe, an accident which has a predicted frequency of one in 10,000 years per reactor. The maximum pressure inside containment for this accident is predicted to be less than 70 kPa(g) (10 psig), well below the design pressure.

A hypothetical power runaway in a CANDU 600 (as occurred in Chernobyl) could only happen if there were:

- failure of a normal control system
- PLUS failure or incapability of stepback
- PLUS failure of shutdown system #1
- PLUS failure of shutdown system #2

Such an accident has an estimated frequency of less than 1 in 10 million years per reactor in CANDU 600 - much less frequent than in Chernobyl because of CANDU's stepback and its redundant and independent shutdown systems. Accidents of such low frequency are not specifically designed for anywhere in the world; for example, in a Light Water Reactor (LWR), used in many countries in the world, the core melt frequency is between one in 100,000 and one in 1,000,000 years, and no specific design provision is made or required, as the frequency and consequences together are judged an acceptable social risk. Nevertheless, although a hypothetical severe power excursion could damage the CANDU 600 reactor core, the energy would be released into a large containment volume (50,000 m³) compared to about 100 m³ for the core container in Chernobyl and pressures in the CANDU 600 containment would be much lower. Analysis of such events is quite speculative and depend on the containment design but even if the CANDU 600 containment cracking pressure were exceeded, the resulting pressure relief would make it unlikely to attain the failure pressure. The CANDU 600 containment would retain much of its effectiveness even for such a severe and improbable accident, another "forgiving" property of CANDU 600.

The CANDU 600 moderator tank relieves to the containment enclosure through four relief pipes with a total relief area of 0.66 m². The relief pipes are sealed by rupture discs with a 138 kPa gauge (20 psi) break pressure. All CANDU's employ the same concept and have generally similar relief areas and pressures. In fact, the CANDU moderator system is tolerant of more than one postulated pressure tube failure. Several pressure tubes would have to fail before a major calandria failure would occur.

In CANDU 600 containments, the maximum estimated quantity of hydrogen generated during a loss of coolant/loss of emergency core cooling accident can lead to average concentrations of about 3% in containment. The production of hydrogen is limited by the effectiveness of the moderator heat sink so that very little of the pressure tube reacts. Buoyancy flow and cooling fans mix the hydrogen quite rapidly throughout

the containment volume and reduce local concentrations in compartments quickly below flammability limits. Even if flammable concentrations were generated, the overpressure from a burn would not result in containment cracking.

The multi-unit stations have a more complex internal geometry and a lower design pressure. Most of the multi-unit stations are now equipped with hydrogen igniters and the remaining stations will be similarly outfitted by May, 1987. The objective of the igniters is to burn any existing flammable mixtures before their concentration can rise to the level at which a burn might represent a significant challenge to the multi-unit containment integrity.

Table 4 provides a comparison of the CANDU and Chernobyl containments.

5. Comment on CANDU Containment

The enclosure provided by CANDU containment systems is much more complete than that of the Chernobyl system in that all of the major primary cooling pipes and the reactor core are within the containment. Refuelling is also accomplished inside the containment. Pickering and the CANDU 600 reactors also include much of the secondary cooling system and auxiliary systems inside the containment enclosure although this is for layout convenience rather than safety necessity.

The containment enclosures of Bruce and Darlington are surrounded by buildings of conventional structure housing auxiliary systems. The calandria vessel boundary coincides with the containment boundary in the housing for the reactivity mechanisms. A rotating seal on the pump shafts closes containment at the coolant pumps.

Thus all major CANDU reactors are fitted with an enclosure completely surrounding the systems containing fuel.

TABLE 4: CONTAINMENT COMPARISON SUMMARY

Containment Item	Multi-unit CANDU's				Single Unit CANDU's	
	Pickering A + B	Bruce A	Bruce B	Darlington A	Gentilly-2	Lepreau
Containment volume (m ³) (1)	594,700	212,900	212,900	305,100	48,500	48,500
Reactor building design pressure: cracking pressure: (kPa gauge)	41	69	82.7	96.5	124 331	124 331
Wall condensation area (m ²)	61,300	57,500	57,500		22,300	22,300
Dousing water volume (m ³)	9,200	9,900	9,900	10,000	2,500	2,500
Sensible cooler capacity (MW) (2)	21.3	11.8	11.8	9.2	2.9	2.9

Chernobyl data on next page

Notes: 1. Includes vacuum building (VB) volume x 1.9.

2. Only coolers on Class III power credited.

TABLE 4 (CONTINUED)

Containment Item	CHERNOBYL					
	Main Cooling Pump Compartment	Lower Space Water Piping Volume	Steam Separator+Outlet Piping Space	Relief Tunnel + Suppression Pool Space	Reactor Hall	Total
Containment volume (m ³)	14,000	8,700	13,900	25,400	67,000	129,000
Design pressure (kPa gauge)	350	180	0	350	7	
Wall condensation area (m ²)						200,000
Suppression pool water (m ³)				5,000 x 2		10,000

TABLE 4 (CONTINUED)

Reactor Vault (Chernobyl) or calandria (CANDU) data	CANDU				Chernobyl
	Pickering	Bruce	Darlington	600 MW	
Relief pressure (kPa gauge)	138	138	138	138	185
Yield pressure kPa (gauge)	Estimated 1.0 - 1.2 MPa				Estimated 0.7 MPa

1. Background

One mechanism of severe core-wide damage, that could potentially affect a number of systems, is mechanical damage due to falling objects. The Soviets have stated that the refuelling machine in Chernobyl, fell over due to the explosion.

2. Chernobyl Design

The fuelling machine is located above the reactor core in the fuelling hall and is moved over the face of the core and to the spent fuel storage pool in the same building, by a gantry. The walls of the fuelling hall are 1.2 m thick concrete for a height of 17 m, to support the weight of the fuelling machine and the gantry whose rails are attached at this level. The gantry rails have a span of 23 m, and the weight of the fuelling machine is 200 tonnes. In addition, near the top of the refuelling hall, 28 m above the face of the reactor, there is a 50 tonne service crane.

The fuelling machine duty in RBMK-1000 reactors can be as much as four to five channels a day, so that in equilibrium operation, the fuelling machine is suspended over the core for much of the time.

3. Comments on Chernobyl Design

The boundary between the reactor core and the fuelling machine is for shielding and not containment purposes. Thus an accident in the refuelling hall has the potential to propagate into the core or vice versa.

4. CANDU Design

CANDU reactors have a service crane, which is entirely within containment for Pickering and CANDU-600 and outside containment for Bruce and Darlington. The service crane in the boiler room handles such heavy items as the primary heat transport pump motor (45 - 65 tonnes) and reactivity mechanism/cobalt adjuster flasks at 25 - 30 tonnes. These are infrequent uses and normally the crane is parked away from the top of the reactivity mechanisms deck.

5. Comments on CANDU Design

The fuelling machines in CANDU access the side of the reactor and are entirely within the containment structure. Thus even severe mechanical failure of a fuelling machine would not affect more than a few channels and the releases would be inside the containment.

Dropping a heavy object on the reactivity mechanism deck during power operation would combine two infrequent events - moving a heavy object over the core and failure of the crane.

Damage of the mechanism deck is possible if a heavy object were dropped onto the core, so administrative controls are in place to limit any such movements across the top of the deck.

1. Background

The moderator had two roles in the accident. It acted as a heat storage mechanism once the fuel reached temperatures higher than the graphite. However once the graphite started burning it provided a continuing source of energy to distribute fission products up to 1000 metres above the reactor.

2. Chernobyl Design

The moderator consists of 1700 tonnes of graphite bricks stacked in the shape of a vertical cylinder 11.8 m in diameter. Each graphite column is composed of 25 cm by 25 cm blocks. The main blocks in the core are 60 cm high; shortened blocks 50 cm high are installed in the top and bottom reflectors for a total graphite height of 8.0 m. The graphite blocks have vertical holes to accommodate fuel channels (about 1670), control rods (211), and instrumentation (142). The reflector is cooled through 156 channels in the peripheral row of the graphite columns. Twenty vertical holes of 45 mm diameter contain thermocouples to monitor graphite temperature.

The moderator and reflector columns are located in a sealed vessel which serves as a gas barrier and structural restraint for the graphite. The atmosphere is a circulating mixture of 40% helium and 60% nitrogen at a pressure of 1.5 kPa. For startup, it is understood that the composition of this mixture is changed to pure nitrogen, to decrease the cooling, so that the graphite temperature is similar to full power operation. This avoids the large reactivity changes from changes in graphite temperature as power is varied.

In normal operation, heat is removed from the graphite partly through gas cooling in the outer channels but mainly by conduction to the pressure tubes and to the primary coolant. That is, the graphite is a heat source for the channels. Conduction is designed in by a series of graphite rings on the pressure tube, which are alternately tight on the moderator graphite and tight on the pressure tube. It is likely that the pressure tubes are inserted and removed with all these graphite rings attached, so that even for the rings which fit tight on the bulk graphite, there must be some clearance - some papers suggest a 0.04-0.05 mm gap. The maximum local graphite temperature has been stated as 750°C.

It is reported that leaks in pressure tubes can be detected by sampling the moderator gas.

3. Comments on Chernobyl Design

The effectiveness of heat removal from the graphite must be very sensitive to the local conditions at the graphite rings on the pressure tubes. On the one hand, one can postulate that dimensional changes in these rings and in the bulk graphite,

as the reactor ages, alter the heat transfer conditions - this was the point made by a U.K. review of RBMK 10 years ago (Reference 9). In addition, the bulk graphite is poorly served with temperature monitors - 20 thermocouple holes in 1700 tonnes, or one per 85 tonnes, suggest it is difficult to detect local graphite hotspots. On the other hand, the Soviets have had lengthy experience with the RBMK type and have not declared any problems related to graphite overheating.

The fact that the graphite is a heat source for the channels affects the course of postulated accidents. The graphite has a large amount of stored heat that must be removed during cooldown after a loss of coolant accident. For severe accidents involving potential pressure tube deformation, the graphite can actually act as a heat sink if the channel temperature rises above the local graphite temperature, because of the large mass of graphite. In contrast to CANDU, the channels will be at higher temperatures for a severe accident (e.g., loss of coolant/loss of emergency core cooling) and therefore more of the zirconium will be able to react with steam to form hydrogen. This will of course be exacerbated if the graphite catches fire.

The response to a pressure tube rupture is key, yet not well understood. On the one hand, pressure tube rupture has been considered in the design, as demonstrated by design provisions for relief from the reactor vessel and the Soviets acknowledge having had channel failures and having replaced them. The restraint provided by the graphite rings should preclude unstable rupture of the tube but not necessarily the growth of a large leaking crack. On the other hand, it is difficult to see how the steam pressure from anything other than a small leak could be relieved - because of the very small clearances between the pressure tube and the surrounding graphite and the fact that escaping liquid from the ruptured tube, on hitting the hot graphite, will flash to steam and increase the pressure in the tank. The U.K. review pointed out that in the absence of a clear escape path for the steam, it would go between the graphite bricks and cause radial and axial forces on the moderator structure. There is no published Soviet accident analysis we have seen on pressure-tube rupture.

Combustion of the graphite has been highlighted as a contributor to the severity of the accident. Simple kinetics calculations done by WNRE show that graphite oxidation in air is exothermic, with ignition around 650 - 750°C. In steam, the reaction is endothermic, becoming significant around 1100 - 1200°C, but requiring an external heat source to keep going. The latter reaction produces hydrogen and carbon monoxide, which burn exothermically in air. In contrast, tests on Hanford reactor graphite cubes (heated in air in a furnace) and bars (heated by an oxyacetylene torch until white-hot) and, crucibles heated by thermite, showed no flame and slow sublimation at the highest temperatures. This suggests geometry (heat losses through conduction) could be significant in any extrapolation of small scale tests to a large essentially adiabatic graphite block;

access of air could also be limiting, and this would depend on the extent and nature of the damage to the core.

The graphite has a large positive reactivity coefficient with temperature. This influences reactor control strategies but not fast accidents, due to the large heat capacity of the graphite mass (bulk heatup will be slow). For severe accidents, with graphite overheating, it imposes a requirement on the reactivity depth of the shutdown systems - it is not known how this is dealt with.

4. CANDU Design

The CANDU moderator is heavy water at an average temperature of 60°C, and a low normal operating pressure up to 21 kPa(g). It is cooled by a separate system of pumps and heat exchangers, since normal heat flow is from the channels to the moderator and from direct gamma and neutron heating. This, together with direct neutron and gamma heating, amounts to about 100 MW(th) in the CANDU 600, or about 5% full thermal reactor power.

The moderator is separated from each pressure tube by an annulus filled with an insulating gas, and a Zircaloy calandria tube. The annulus gas is monitored for moisture, to detect a pressure tube leak. The localization is not to each individual tube, but to groups of tubes, whereafter other methods are used to locate the specific leaking tube.

The calandria is provided with four relief pipes, which discharge into containment and have rupture disks set at a calandria pressure of 138 kPa. They are sized based on a sudden double-ended rupture of a pressure tube, with no credit for the strength of the surrounding calandria tube.

5. Comments on CANDU Design

The amount of heat removed from the moderator in normal operation is the same as fuel decay heat a few tens of seconds after reactor shutdown. Thus the moderator is capable in emergencies of removing fuel heat following a loss of coolant and loss of the emergency core cooling. In such a circumstance, the pressure tube will either sag on to the surrounding calandria tube as it overheats, providing a conduction heat path from fuel to moderator (in addition to radiant heat transfer), or more likely expand under the influence of residual coolant pressure in the channel. The expansion is arrested by the cool calandria tube and the tube-to-tube contact provides a conduction path to remove decay heat.

In either case the UO_2 fuel does not melt and the pressure boundary failure is limited to the initiating break. Equally important, the pressure-tube temperatures are limited by heat conduction and radiation to the calandria tube, so that the amount of hydrogen that can be produced from fuel sheaths or pressure tubes is limited by the metal temperature. For a loss

of coolant/loss of emergency core cooling accident, CANDU 600 analysis indicates that about 35% of the sheaths and less than 1% of the pressure tubes can be oxidized.

A spontaneous pressure tube failure may or may not cause a failure of the surrounding calandria tube - both types have occurred (the accident which led to a failure of the calandria tube though, occurred at a very high coolant subcooling). If the calandria tube does fail, the steam discharge will be largely condensed by the moderator liquid - i.e., the moderator reduces the potential overpressure in the calandria instead of increasing it. In addition for a severe pressure-tube failure, the calandria tubes themselves can absorb some of the energy in the pressure wave by collapsing onto their internal pressure tubes. Thus a pressure-tube failure is not predicted to cause further pressure-boundary or calandria failures.

1. Background

A brief review of selected aspects of the heat transport system (HTS) and the ECC follows.

2. Chernobyl Design

The reactor (Figure 4) has a vertical pressure-tube boiling-light-water-cooled core. The coolant circuit is arranged in two parallel loops, with each loop having two steam separators, 24 downcomers which connect to a common pump inlet header, four pumps (one on standby), a common pump outlet header and then distribution headers, each feeding 40 inlet feeders. The core exit quality is 14% average, 20% maximum. The two-phase outlet goes to the steam separators, with the steam going to two turbine generators and the liquid going down the downcomers. The condensate from the turbines is mixed with water from the separators and returned to the inlet of the main circulating pumps. The steam mains are connected to a common header via check valves. There is no evidence of other provision for loop isolation. By and large, each loop serves one side of the core.

Each inlet feeder has a manually-operated, remote controlled flow control valve, used to equalize channel exit conditions and to isolate the channel for pressure tube replacement. Typically each valve is adjusted twice between refuelling. There are also main flow control valves at the pump exits, pump isolation valves, etc. A pump bypass is provided to assist thermosyphoning.

The emergency core cooling system consists of a two-stage system: injection from pressurized water accumulators, followed by pumped injection from a dedicated tank. The design basis is the guillotine rupture of the pump discharge header, 900 mm diameter. The accumulator pressure is about 10 MPa, or above HTS operating pressure and the accumulators capacity is 200 m³. Injection is initiated on a combination of high building compartment pressure and low separator level or low pressure drop, and is effective within 15 seconds - it is conceivable that injection occurs before shutdown. Flow restrictions are installed between the distribution header and the reactor inlet header to reduce loss of water directly through the break. The design target for ECC is prevention of fuel damage.

3. Comments on Chernobyl Design

The provision of individual channel flow control valves optimizes channel flow but increases the chance of an operator error causing a flow starvation in a channel.

Bulk flow is normally controlled to match power - i.e., at lower powers, the flow is gagged to keep void in the core and so limit the total void reactivity available. This action is

done by the operator, but at very low powers (less than about 20% full power) it is difficult to match the flow and power.

The accumulator capacity seems inconsistent with the maximum pipe break used for design. The total capacity of the tanks is 200 m³ or the same as CANDU 600 but the maximum pipe size is 900 mm compared to 440 mm and the decay heat will be almost twice that of a CANDU 600. The peak flow rate in Chernobyl is estimated as half that of CANDU 600.

After medium-term ECC injection from dedicated tanks, there must be long-term recovery of spilled water. The dedicated tanks probably last for 30 min for the largest break but the recovery path thereafter is not clear.

The Soviets have reported analysis of ECC performance for pump suction and pump discharge header breaks and for the maximum break size only. It is not known what further analysis they have done but not reported. In Canada, a complete break survey covering all pipe locations and all break sizes is done. This is because the largest break is not necessarily the one which causes the lowest core flow.

4. CANDU Design

The CANDU primary heat transport system consists of one (Bruce) or two figure-of-eight loops. Each loop consists of a core pass, at least one pump and steam generator, another core pass and at least another pump and steam generator, for a complete circuit round the loop. Loop isolation is automatic after receipt of a loss-of-coolant signal. There are no flow control valves in any CANDU; Pickering has isolation valves upstream and downstream of the HTS pumps and steam generators to cater for a pump or steam generator out of service.

Emergency core cooling is used to provide alternative cooling to the fuel in case of a pipe break in the main cooling system. It has the capability to refill the core for a break in the largest pipe. The pressure and flow capabilities vary with the reactor size and power. The NPD reactor has emergency core cooling gravity-fed from an elevated tank; Pickering-A initially injects moderator water (Pickering-A is now being fitted with high-pressure ECC pumps drawing from a separate water source); and other reactors have a high pressure injection from a separate source of water, driven by either pumps or pressurized gas. For long term cooling, the water is collected in the sumps, cooled and recirculated through the main cooling system within the reactor building.

5. Comments on CANDU Design

Because all large heat transport system (HTS) piping is within containment, the spilled water from a HTS pipe break can always be recovered in the sump.

Isolation of the two loops on a LOCA minimizes both the amount of potential fuel damage, and the amount of hydrogen that can be produced if ECC is impaired.

Accident analysis considers a pipe rupture in any location for the purposes of ECC design. The break area is taken as a parameter up to the size of a double-ended pipe break; a deliberate search is made for the stagnation break size.

In addition, postulated LOCA's are combined with impairments of the safety systems: 1) failures of each ECC subsystem (injection, cooldown, loop isolation), 2) failures of the containment envelope or of each active subsystem and 3) only the less effective of the two independent shutdown systems is credited in each case.

1. Background

The accident at the Chernobyl reactor pointed out a significant effect of the lack of a complete containment. During the Chernobyl accident, oxidizing conditions occurred such that fission products that are volatile at 1700°C (Iodine, Caesium, Tellurium) were released as elemental gases. In the case of a severe accident in CANDU we expect that reducing conditions would occur and that these fission products would be released to containment as chemical compound aerosols.

2. Chernobyl Design

In general, the composition of the aerosols released during the accident were reported to be characteristic of the irradiated fuel composition, except for enhanced release of elemental iodine, cesium, and tellurium.

The initial reactivity excursion is reported to have shattered the fuel in the bottom 30% of the reactor. The hot fuel and cladding particles interacted violently with the coolant. The explosion probably released fuel particles and fission products into the air. Once the reactor vessel was breached, oxygen entered the core and some of the remaining fuel may have oxidized. Oxidization could have destroyed the fuel matrix and could have led to the production of small fuel particles containing fission products. The fission products that are volatile at 1700°C (I, Cs, Te) would be released as gases, while other less volatile species would be released as aerosols.

A further effect of oxygen is on fission product behaviour. The hot, oxidizing conditions in the core region would either destroy CsI or would prevent its formation and we would expect a substantial fraction of the released iodine to be volatile I₂ gas. As the I₂ cooled, it would attach to aerosols (for example, from combustion of the graphite) and would be transported along with other core material.

Another phenomenon that could have had some effect on the releases at Chernobyl, is the potential interaction of graphite with fuel. The explosion could have mixed graphite and hot fuel particles. At high temperatures (i.e., 1500°C), graphite and fuel can react to form a uranium oxy-carbide. This could have contributed to the destruction of the fuel matrix and further enhanced the release of fission products.

3. CANDU

The releases during the accident at Chernobyl are in marked contrast with the release of Iodine and Caesium in a heavy water reactor (or light water reactor), where the hot reducing conditions in the core would result in CsI formation. The CsI would encounter oxidizing conditions only in the containment building, where temperatures are too low for extensive oxidation

of the CsI. Thus we would not expect to form large quantities of volatile I_2 in a CANDU reactor.

CsI is easily absorbed into water in the containment thus significantly reducing (10 to 100 times) the amount of Caesium and Iodine released. The effect of the wet atmosphere inside a containment is demonstrated by the differences between the releases to the environment from TMI and Chernobyl, even though the former was not completely isolated from the environment for the early part of the accident.

Although there was a similar level of releases to containment for TMI (Table 5), there was a significant attenuation factor for all forms of fission products released. The chemical and physical processes connected with a "wet" containment, like TMI, also would occur for an accident in a CANDU reactor. Even if the containment building were leaking, major attenuation of the biologically significant radioactive species releases would occur.

TABLE 5

THREE MILE ISLAND AND CHERNOBYL RELEASES COMPARED

	TMI-2		CHERNOBYL
	Outside the core	to Environment	to Environment
Noble Gases (Xe, Kr)	48%	1%	100%
I	25%	$3 \times 10^{-5}\%$	20%
Cs	53%	not detected	10-13%
Ru	0.5%	not detected	2.9%
Ce(group)	NIL	not detected	2.3-2.8%

Reference: CHERNOBYL, a paper prepared by J.G. Collier,
L. Myrddin Davies, Central Electricity Generating
Board, Gloucester, England.

PART IV OTHER CONCERNS RAISED

- IV-1 PRESSURE TUBES**
- IV-2 COMPUTER CONTROL**
- IV-3 ON-POWER REFUELLING**
- IV-4 MULTI-UNIT CONTAINMENT**
- IV-5 FIRE PROTECTION**

1. Background

In this section we deal with the pressure tube design of Chernobyl and of CANDU.

2. Chernobyl Design

In Chernobyl the channels are located vertically in the graphite moderator and either contain low-enriched uranium oxide fuel or are used as locations for control rods and instrumentation. (Figure 11)

Fuel bundles are made of 18 elements connected to a central support tube. There are two fuel bundles in each fuel channel. The fuelling machine operates above the reactor and is designed to refuel channels on full power.

The Chernobyl design has about 1670 fuel channels located in vertical holes in square graphite blocks forming the reactor core. Figure 20 shows a typical fuel channel assembly. Each fuel channel is made up of a Zirconium-2.5% Niobium pressure tube, connected at upper and lower ends to stainless steel extensions via a transition piece of Zirconium-1% Niobium or Zirconium - 2½% Niobium alloy as shown in Figure 21. The transition piece is joined to the pressure tube by an electron beam weld. The transition piece is then joined to the stainless steel by a tapered threaded (or grooved) section incorporating a diffusion weld. There is another argon-arc weld between the stainless steel hub and the connection to the upper and lower steel housing. A permissible rate of heatup and cooling of 10° to 15°C per hour has been established by thermal and strength tests of the transition joint.

The pressure tube has an 88 mm (3.46 in.) outside diameter with a wall thickness of 4 mm (0.158 in.). A series of graphite rings are stacked and fitted alternately around the pressure tube to improve the heat transfer from the graphite blocks to the outer surface of the pressure tube (Figure 22).

A mixture of helium and nitrogen fed from the bottom end of the reactor flows between the graphite columns. It provides a heat conducting media for transmitting the graphite heat to the fuel channel and is also monitored for moisture to detect leakage from the tubes.

The top end of the fuel channel is welded to the top housing sleeve and at the other end a stuffing-box assembly seals between the extension pipe and the bottom housing sleeve. Small changes in the length of the pressure tube are accommodated by movement through the stuffing-box seal.

The outlet top end of the channel is sealed by a nozzle plug which can be removed by rotation during the refuelling

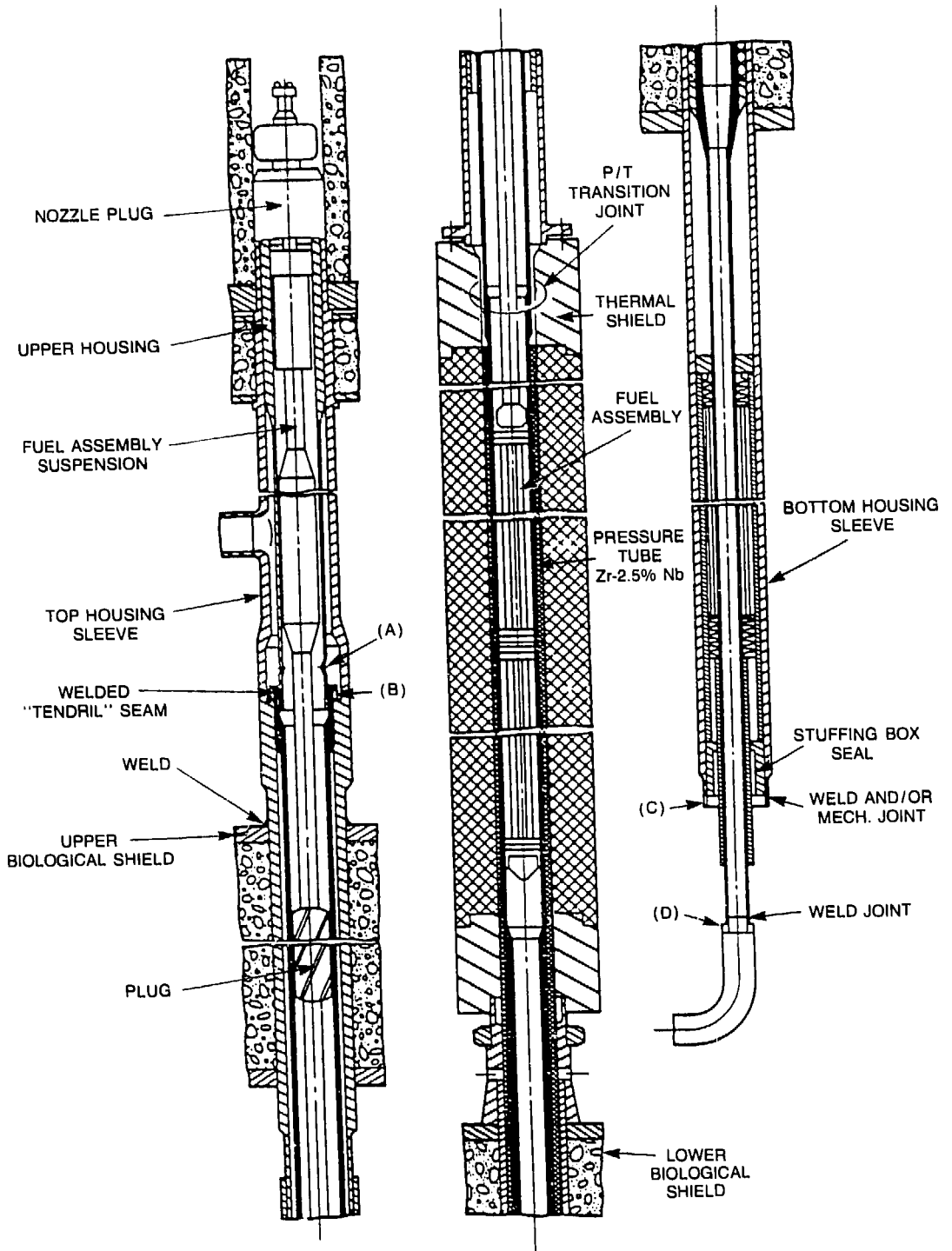


FIGURE 20 CHERNOBYL-4 FUEL CHANNEL

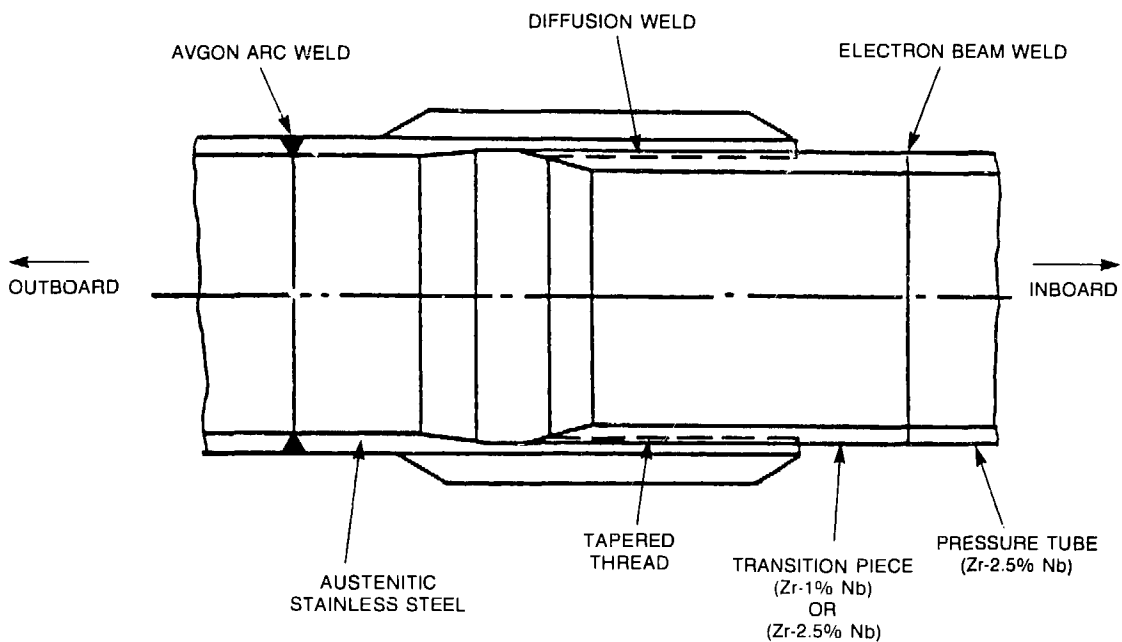


FIGURE 21 TRANSITION JOINT Zr-2.5%Nb/PRESSURE TUBE TO STAINLESS STEEL

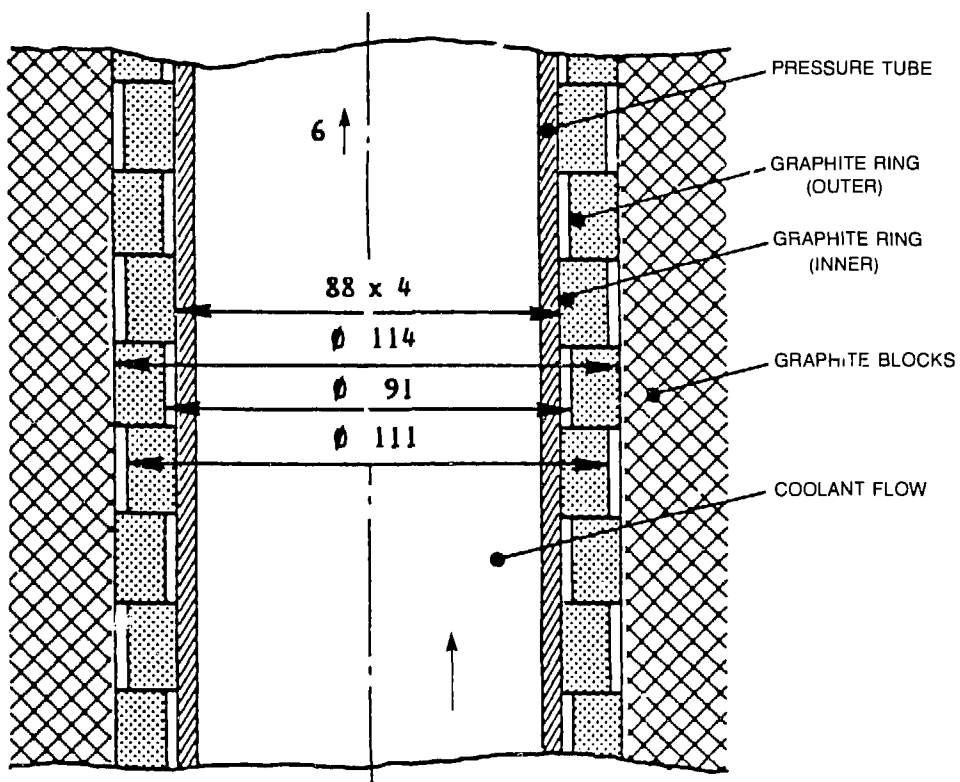


FIGURE 22 GRAPHITE/PRESSURE TUBE ARRANGEMENT

operation, Figure 23. The inlet end of the channel is connected directly to the coolant pipe by means of a welded connection.

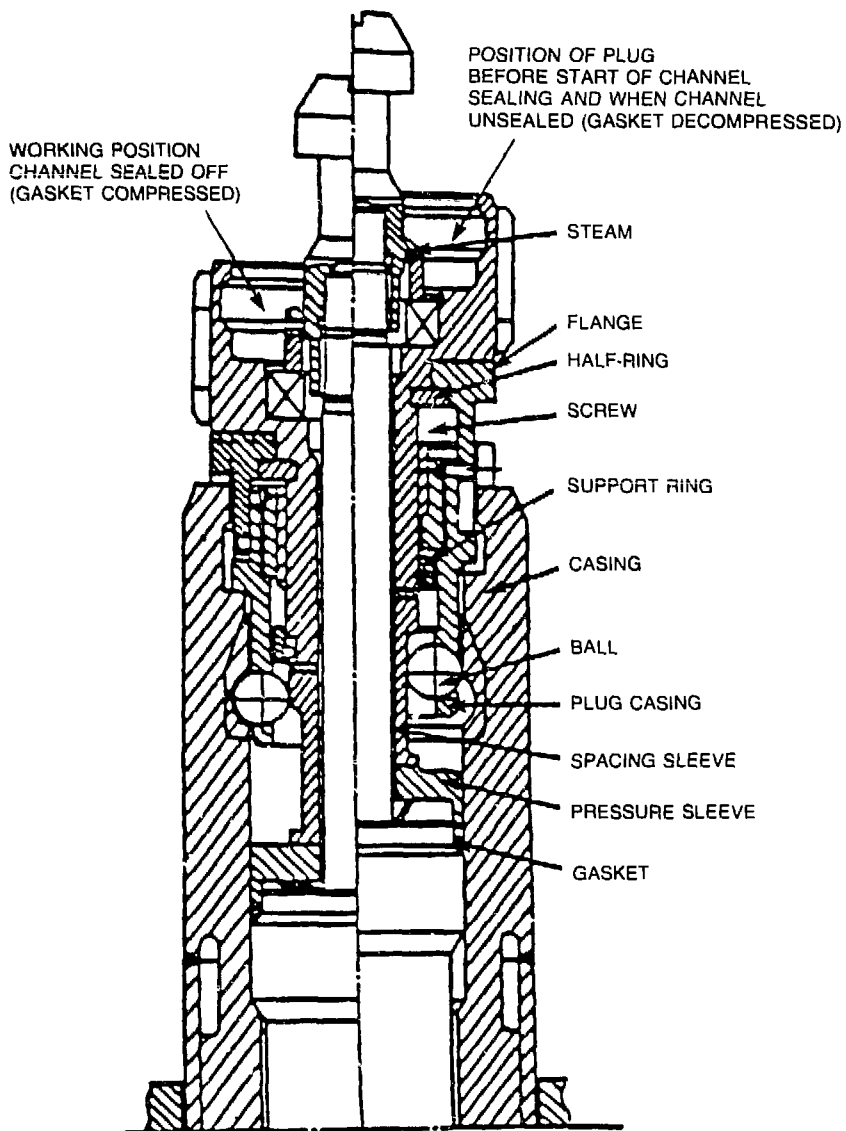


FIGURE 23 CHANNEL NOZZLE PLUG

The service life of the fuel channel is estimated to be 25-30 years (reactor design life is 30 years) and the channel is said to be replaceable during shutdown with remote tooling.

3. Comments on Chernobyl Design

There are several key features of the Chernobyl pressure tube design:

- (i) Heat is removed from the graphite to the channel. The graphite is always hotter than the coolant in the channel (graphite is about 700°C and transfers heat to the channel coolant at a temperature of about 280°C).
- (ii) There does not appear to be any fundamental design problems with the pressure tube itself. The diffusion joint appears to limit maximum allowable heating and cooling rates to 10°C to 15°C/hour. This is likely required to ensure a long design lifetime. The joint is quite strong; however, it is uncertain whether the diffusion joint or the transition section is as strong as the remainder of the pressure tube.
- (iii) As indicated earlier (Section III-5, Graphite Moderator), the response to a pressure tube rupture is key, yet not well understood. On the one hand, pressure tube rupture has been considered in the design (for example the relief from the "calandria") and the Soviets acknowledge having had channel failures and having replaced them - they do not comment on any damage to the rest of the core. The restraint provided by the graphite rings should preclude unstable rupture of the tube but not necessarily the growth of a large leaking crack. On the other hand, it is difficult to see how the steam pressure from anything other than a small leak could be relieved - because of the very small clearances between the pressure tube and the surrounding graphite and the fact that escaping liquid from the ruptured tube, on hitting the hot graphite, will flash to steam and increase the pressure in the tank. The U.K. review pointed out that in the absence of a clear escape path for the steam, it would go between the graphite bricks and cause radial and axial forces on the moderator structure. There is no published Soviet accident analysis we have seen on pressure-tube rupture.

4. CANDU Design

CANDU is a pressure tube, heavy-water (D₂O)-moderated, heavy-water (D₂O)-cooled reactor. The fuel channels consist of two concentric tubes, (the pressure tube and calandria tube) with a space in between. These channels are located horizontally in the heavy-water moderator, and contain natural uranium fuel. The channels and heavy water moderator are all contained in a large tank called a calandria vessel.

Fuel bundles are typically made of 37 elements of short length (about half a metre), and there are typically 12 bundles in each fuel channel. The fuelling machines refuel by coupling onto a fuel channel at both sides of the core (thus the machines are never over the core). CANDU design has typically about 380 to 480 fuel channels. Each fuel channel is made of a Zirconium-Niobium pressure tube (similar in composition to that in Chernobyl), and is connected by "rolled joints" (i.e. no

welding), to stainless steel end fittings which serve as a connection to the fuelling machine and to the external feeder piping through a side part.

In CANDU reactors, the annular space between the pressure tube and calandria tube is filled with a gas which is monitored to detect any moisture in the space. The dewpoint of the gas provides a preliminary indication of a pressure tube leak. Monitors in segments of the reactor annulus system aid in locating a leaking channel.

5. Comments on CANDU Design

There are a number of significant advantages of the CANDU pressure tube design, over Chernobyl, and over pressure vessel design.

- (i) For most conditions, pipes, including pressure tubes, leak before they break. The CANDU design has two separate tubes, the pressure tube and the calandria tube. The calandria tubes can withstand a very high (basically full system) pressure. Thus, should the pressure tube leak, the leak can be detected by the gas in the space between the two tubes and the reactor shut down and the pressure tube replaced. The calandria tube therefore serves as a second, pressure boundary although credit for this is not taken in CANDU safety analysis.
- (ii) Surrounding each of the channel assemblies is the cool (about 70-80°C) water moderator. If, for some reason, the pressure tube heats up, it expands or sags to contact the surrounding calandria tube, and heat is transferred to the cool water. Subdividing the core into many pressure tubes allows this possibility. This cool surrounding water provides an inherent safety defense to prevent significant fuel melting. It also means that fuel and pressure tube temperatures are kept low so that there is little formation of hydrogen for a large range of severe accidents. These two benefits combine to create third benefit. For many types of severe accidents in CANDU, there is no direct challenge (via core melting, steam explosion, or hydrogen explosions) to containment integrity.
- (iii) We have established the capability of the CANDU design to tolerate fuel heatup due to channel blockage or flow reduction in a channel. Severe fuel heatup or fuel melting is an unlikely event, since it could only occur in a highly unusual combination of circumstances. Flow blockage severe enough to damage the channel requires a blockage area > 90% of the channel flow area and has never occurred in CANDU. Such a blockage could fail both pressure tube and calandria tube and result in discharge of coolant to the moderator. The calandria and other channels are designed to remain intact

following such a failure. Indeed, the reactor can tolerate several simultaneous channel failures.

- (iv) There have been two major pressure tube ruptures due to defects; one at Pickering A and one at Bruce A. In both cases the damage was limited to one channel, which was replaced.
- (v) The rolled joints used in CANDU reactors have generally performed well. There were pressure tube leaking problems in the rolled joint area in Pickering A and Bruce A, associated with delayed hydride cracking of some tubes in high stressed areas resulting from improper rolling of the joint. This has been corrected in subsequent CANDU reactors.

Finally, the first two units at Pickering A have been entirely retubed due to premature sagging of the Zircaloy-2 pressure tubes used in those units. The tubes were replaced with tubes of the Zr-Niobium material which was used in all other reactors. While retubing was not expected to be needed so soon, the contribution to the station lifetime unavailability will be less than 10% and the fact that the core pressure boundary can be replaced is a unique advantage of pressure tube reactors.

1. Background

Direct computer control was not used for Chernobyl - the Soviets reportedly felt it was not sufficiently reliable based on their early experience.

2. Chernobyl Design

The actual control of Chernobyl appears to be mostly analogue; from 0 to 0.5% power, the control is manual with special low power ion chambers; from 0.5% to 6% power, the control is non-redundant automatic control of four rods based on four ion chambers; above 6% power, control is dual redundant automatic control with each redundant portion having four rods and four ion chambers.

Spatial control is mostly manual, using 139 absorber rods but there is a rudimentary automatic spatial control system using 12 absorber rods. For the latter, two fission chambers near each rod are used as feedback sensors.

There is an extensive monitoring programme (PRIZMA) in an on-line station computer (SKALA). This program monitors in-core flux measurements, individual channel flows, control rod positions and many other variables, then calculates reactor power distribution, margins to dryout, etc. and issues instructions to the operator to guide him in manual spatial control and flow control. There is apparently no direct digital control of the devices. It also appears that there is only one such station computer. The PRIZMA program runs every 5 to 10 minutes so is relevant for very slow power changes only.

3. Comments on Chernobyl Design

At Chernobyl most of the basic spatial flux control is manual (i.e. 139 absorber rods). In addition each of the 1670 coolant channels has an inlet control valve which is adjusted roughly twice between refuelling of the channel. While it is possible to use operators for these kinds of control it assumes a high reliance on the part of the operator. There is obviously a significant chance for error.

4. CANDU Design

CANDU stations make extensive use of direct digital control; this encompasses all reactor controls and all major process loops. The configuration consists of two identical computers running continuously in active/hot-standby mode. Internal self-checks and external watchdogs transfer control if failure of the active computer is detected. If both computers fail, all control circuits are isolated and go to their designed state which is either failsafe or neutral. For example, the reactivity control absorbers would be inserted and cause a rapid reactor power decrease if both computers failed.

Flux mapping for purposes of refuelling is done off-line, as in Chernobyl.

5. Comments on CANDU Design

The dual computer concept has served well - after some initial poor experience in Pickering A (early 1970's) there have been only a few instances of computer failure. Dual computer failure, although it has occurred, has been very rare and has always been ended by a safe shutdown by the (independent) shutdown systems.

From a safety point of view, the key is that the shutdown systems are completely independent of the control computers, in terms of sensing devices and shutdown mechanisms and have the capability to overcome any computer-induced positive reactivity insertion. Thus even a massive adverse computer failure (e.g., driving all reactivity devices in a positive direction) can be easily terminated.

1. Background

In this section, we discuss on-power-refuelling and in particular the use of the refuelling machines.

2. Chernobyl Design

The Chernobyl fuelling machine is a massive (200 tonne) flask which refuels the reactor from the top, above the deck plate. It is held by a carriage supported by a 100 tonne crane on rails. It can refuel five channels per day on power, and 10 per day during shutdown. In operation it first seals over the outside of the fuel channel nozzle; pressurizes the seal; removes the nozzle cap; removes the nozzle plug, shield plug, suspension rod and fuel assembly into the pressurized cylinder of the flask; rotates a cartridge within the cylinder to permit insertion of a channel inspection gauge; then lowers in the fresh fuel assembly. Water flows from the fuelling machine to the channel during this operation to cool the spent fuel. The steps above are reversed to remove the machine and the spent fuel is transported to the spent fuel bay. It is believed the operation is largely manual.

3. Comments on the Chernobyl Design

Apart from points made earlier - that the fuelling machine is outside containment, and is positioned over the core, there is nothing unusual in the refuelling operation, which resembles CANDU apart from the length of the fuel element and the use of one as opposed to two machines. It also generally resembles gas-cooled reactor fuelling machines. Large failures of the fuelling-machine-to-channel connection would seem unlikely due to the mass of the machine; i.e., hydraulic forces would not likely move it. We would expect there to be interlocks which prevent moving the machine before the refuelling action is complete.

4. CANDU Design

The CANDU fuelling machines operate horizontally, inside containment and are located at each face of the reactor: one machine inserts new fuel bundles while the other one receives the spent ones. About 14 channels (110 bundles in total) are refuelled each week. Connection on to the channel, pressurization, removal of the closure plug, shield plug etc. are similar steps to Chernobyl. The spent fuel is discharged through a containment penetration to the spent fuel bay. The process is almost entirely computer-controlled.

5. Comments on CANDU Design

The advantages of on-power refuelling are:

1. Increased plant availability, by at least 5% relative to light water reactors. The fuelling machine

contribution to unavailability has been very small - 0.5%.

2. Ability to remove defected fuel bundles, thus reducing man-rem exposure by keeping the main coolant "clean".
3. Ability to optimize fuel channel power and burnup.
4. Ability to limit the amount of reactivity holdup.

From a safety point of view, the fuelling machine in operation is an extension of the primary system pressure boundary and the same accident analysis is applied to it as to the rest of the heat transport circuit. The added risk from the fuelling operation compared to a primary system pipe break, (which all reactors assume for analysis), is at most a single channel's worth of irradiated fuel. Thus we analyze such accidents as:

- failure of heavy water supply and return hoses to the machine, thus jeopardizing spent fuel cooling, both when the machine is on-reactor and when it is off-reactor
- massive failure of the snout-to-end-fitting connection with assumed ejection of all the fuel bundles in the channel (this is probably not physically possible)
- either of the above with an assumed impairment in the containment envelope or in the emergency core cooling system

and these are shown to meet the relevant dose limits for the public.

In actual experience, massive failures of the connection have never occurred in CANDU - indeed, the emergency core cooling system has never been necessary. Leaks in the hoses and at the snout-to-end-fitting connection (due to a failed O-ring) have occurred in the past; there has also been mechanical damage to some bundles in the channel which were then removed. Given the limited damage which can occur, and the fact that it is contained, there is no evidence that fuelling machine events are a significant risk contributor.

1. Background

Both Chernobyl and most of the nuclear stations in Ontario are multi-unit plants on the same site. The accident forced a shutdown of all the other operating units at the site.

2. Chernobyl Design

There are four operating units at Chernobyl, plus two more under construction. There is no sharing of containment facilities, but the operating units share a common turbine hall and some electrical services.

3. Comments on Chernobyl Design

An accident which spreads contamination as widely as Chernobyl will restrict access to other units on the same site. An effective containment is key to preventing such damage. Because the reactor is direct-cycle, there is a possibility of contaminating the common turbine hall since there is apparently no steam main isolation capability.

4. CANDU Design

The multi-unit plants in Ontario have a linked containment structure, wherein the containment around each reactor is linked by a large duct to a common vacuum building kept at reduced pressure. In the event of an accident, steam and radioactivity will be sucked into the vacuum building, and the entire structure will stay below atmospheric pressure (leakage will be in, not out) for many hours.

Since the coolant does not run the turbines directly, the extent of contamination on the turbine side is limited to that from an accident with a prior leaking boiler tube.

5. Comments on CANDU Design

The vacuum concept has been analyzed for the usual spectrum of accidents such as large loss-of-coolant, but as part of the Canadian safety philosophy must also meet public dose limits for dual failures such as a loss-of-coolant plus a failure of the emergency core cooling water flow, or plus an impairment in the containment envelope. The vacuum concept, because of its forced in-leakage, is very powerful in limiting short term releases for such impairments. In the long term (hours to days), the emergency filtered air discharge system can be used to vent containment and at the same time to filter and remove activity from the containment atmosphere. Typically 99.9% of the core inventory of iodine is contained, and doses to the most exposed member of the public are limited to less than 25 rem.

Source terms from accident analysis are used to study the habitability of the control room after an accident; the units

could also be safely shut down and monitored (if necessary) from the secondary control area in Pickering-B, Bruce-B, and Darlington.

Given the powerful containment and the severity of failures analyzed to meet the dose limits, it is very unlikely that damage in one unit would prevent effective control of the others by station staff.

There are other safety advantages to the multi-unit design:

1. ability to use the electrical and water supplies of the other stations in emergencies
2. presence of a large operational staff familiar with all the units on site

1. Background

The dramatic graphite fire at Chernobyl, in combination with a conventional fire in the fuelling machine hall and turbine hall, has further raised awareness of fire as a reactor safety issue.

2. Chernobyl Design

It has been reported that the fire protection system consists of "hydrants inside and outside of the turbine building and a system to cool the trusses and roof of the machinery room. An automatic water-spray fire-extinguishing system is provided in the cable and transformer rooms. The pumps and automatic valves of this system are connected in three independent subsystems, which are in turn connected to the emergency diesel generators. The water supply for each system consists of three tanks with a capacity of 150 m³. These tanks are filled from the plant general fire-fighting system".

3. Comments on Chernobyl Design

The fire protection system in the Chernobyl design is of quite a high standard. It is clear though that the accident was well beyond the capability of the fire protection design.

4. CANDU Design

In CANDU there are no automatic fire suppression systems in the reactor buildings; fires there are expected to be limited in extent because of the absence of large quantities of flammable material and are fought with portable fire extinguishers. Limiting the safety consequence of local fires is achieved by the two-group philosophy: that is, the plant can be shutdown and monitored and decay heat removed, by either of two independent and spatially separated groups of systems. Fire suppression systems outside the reactor building are conventional sprinkler systems, CO₂ systems, halon systems and fire standpipe systems. Manual firefighting using fire hoses and portable fire extinguishers are relied on for areas of lower fire hazards.

5. Comments on CANDU Design

Of course there is no combustible graphite in the vicinity of the core. Combustible sources in the reactor building are mainly the lubrication oil in the pump motors and electric cables. Due to the physical separation of the combustible sources and the reactor core, it is improbable that a fire could induce direct core damage. The dousing system in single-unit containments could be used for some fires, e.g., a pump lube oil fire, but it does not cover the entire reactor building volume and has a severe economic penalty associated with its operation. Further review of the adequacy of fire-fighting systems in CANDU plants is underway.

PART V IMPLICATIONS OF CHERNOBYL ACCIDENT

- i) The threat posed by reactivity accidents has long been recognized in our program, starting with the 1952 NRX accident. CANDUs have always been provided with safety shutdown systems which are independent of the regulating system and powerful (rate and depth).

Nonetheless, it is prudent to review, in depth, the adequacy of our defenses. In particular, a review is underway to ensure that there is no conceivable combination of distorted flux shape, reactor power, control system action (automatic or operator), coolant conditions, etc., which could result in a reactivity excursion exceeding the capability of our shutdown systems.

- ii) The consequential fires (besides the graphite fire) at Chernobyl were well handled, under extreme circumstances (particularly radiation), by the fire-fighting crews. CANDUs have all included fire protection programs in the design and operation of the reactors. It is prudent, however, to review the fire protection design adequacy, particularly in the presence of radiation, to determine any possible lessons learned.

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