Chernobyl – A Canadian Technical Perspective

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Abstract
On 26 April 1986, the unit 4 reactor at the Chernobyl Nuclear Power Station in the Soviet Union suffered a severe accident which destroyed the reactor core. The reactor design, as it relates to the accident sequence, is reviewed in detail, using information presented in Soviet literature and at the International Atomic Energy Agency Post-Accident Review Meeting in August 1986. The aspects of the design which, in our view, exacerbated the accident are presented and compared to the CANDU[1] reactor design. Key Chernobyl design aspects examined are (in order of importance): capability of shutdown, containment, and the variation of void reactivity with operating state. A number of design issues have been raised for Chernobyl which are less relevant to the accident and which we feel are less important. These include: the sign of the void coefficient, pressure tubes, use of computers in control, multi-unit containment, and fire protection. These are discussed briefly, and compared with the CANDU approach. 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 the plant safety depended too heavily on the skills of operators in maintaining many reactor parameters, especially reactor power and power shape, within a certain operating envelope. By contrast, the ability of the CANDU shutdown systems to shut down the reactor 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, Atomic Energy of Canada Limited (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 high 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) has also been performed.

Résumé
Le 26 Avril 1986, un grave accident s'est produit à l'intérieur du réacteur no. 4 de la centrale nucléaire de Tchernobyl, en URSS: Le coeur du réacteur a été entièrement détruit. La conception du réacteur et son rapport avec le déroulement de l'accident sont étudies en détail dans le présent document, à l'aide des 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 de la conception du CANDU.[1] 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 questions relative à la conception ont été soulevées en relation avec Tchernobyl, mais elles sont moins pertinentes à l'accident et nous semblent moins importantes. Il s'agit du signe du coefficient de vide, des tubes de force, du contrôle-commande par ordinateur, 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 insuffisant dans la mesure où il n'était pas capable de procurer 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 le niveau et forme de puissance, à l'intérieur de certains domaines de fonctionnement. Par contre, dans le cas du CANDU, le fonctionnement des systèmes d'arrêt est indépendent de l'état de fonctionnement de la centrale: en ce sens, la conception est beacoup plus tolérante. Cependant, afin de tenir compte des erreurs de Tchernobyl, l'EACL reverra deux études de conception du CANDU: 1) un réexamen de toutes les

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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 une étude a aussi été executée par la Commission de Contrôle de l'Énergie Atomique, l'organisme réglementaire canadien.

Introduction

On 26 April 1986, the unit 4 reactor at the Chernobyl Nuclear Power Station suffered a severe accident. The core and much of the building were destroyed; all of the noble gases and several per cent 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, [Semenov 1983; Levin and Kreman 1983; Dubrovsky *et al.* 1979; Turetskii *et al.* 1984; Babenko *et al.* 1980; Dollezhal and Emel'yanov 1986], 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. The related reports [USSR 1986; INSAG 1986] are the most authoritative documents 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 in light of the events at Chernobyl, and to see what lessons can be learned.

In this paper the design review done to date in Canada by Atomic Energy of Canada Limited (AECL) is presented. From the Canadian point of view it covers:

1 relevant information on the Chernobyl design and the accident, both as presented [USSR 1986; INSAG 1986; Dastur *et al.* 1986] by the Soviets at the Post-Accident Review Meeting (PARM) held in Vienna from 25–29 August 1986, and as deduced from publicly available Soviet documentation;

2 details of AECL's technical review of the Canadian Deuterium Uranium Pressurized Heavy Water Reactor (CANDU PHWR) 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) has also been performed.

Other related reports produced in Canada to date are a technical review [Howieson and Snell, 1987], an executive summary of the technical review [Snell and Howieson 1987], and a less technical summary for the general public [Snell and Howieson 1986].

Brief Review of the Accident

Accident Sequence

The Post-Accident Review Meeting (PARM) for Chernobyl took place from 25 to 29 August 1986. At the meeting and during the following week, the Soviets presented detailed information on the accident sequence [USSR 1986; INSAG 1986], accident recovery, radiological consequences, and planned design / operational changes for other reactors of the same type. In this section the information presented is summarized.

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: this resulted in an overpressure in the cooling circuit. Pressure tube failures led to pressurization of the core vessel and loading of the 1,000-tonne reinforced-concrete top shield slab, expelling it from the reactor vault. Burning fragments were ejected from the core, starting 30 fires in the surrounding area.

Immediate Effects of Power Runaway

The core expanded into the surrounding space in the reactor vault (i.e., there was destruction of the radial reflector and the water shield), and dispersal of the fuel and the graphite moderator resulted in the core becoming subcritical. The severing of reactor 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.

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. This continued for several days as the graphite burned and the fuel oxidized, with the rate of release falling as the fuel cooled.

To stop the release the Soviets dropped about 5,000 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), and 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: 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 that there is substantial uncertainty associated with these estimates.

Accident Recovery

Firefighting 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 did not occur. To prevent molten materials from falling into the water suppression pools below the reactor, they were drained and replaced by concrete. To prevent ground water contamination, a concrete barrier was built deep into the ground around the area.

As of April 1987, units 1 and 2 have been restarted. The timing of startup of unit 3 is less certain, due to the higher level of radioactivity and the need to check the condition of the equipment.

Radiological Consequences

Onsite staff. There were 2 immediate deaths as a result of the accident. Over the next few weeks there were 29 fatalities from high radiation doses and burns received by station staff trying to bring the accident under control. The dose distribution for these people was as follows:

Dose (rads)	*No. Patients*	*No. Deaths*
600–1,600	22	21
400–600	23	7
200–400	53	1

Offsite – effects of the accident on the surrounding population. Emergency response measures included: 1) distribution of iodine tablets to the population around Pripyat, apparently successfully and with minor side effects; 2) sheltering for residents of Pripyat before evacuation; and 3) evacuation once the plume shifted towards Pripyat.

The Soviets estimated the collective dose *commitment*[2] in the USSR as:

31×10^6 person-rem *external* (over 50 years), and
210×10^6 person-rem *internal* (over 70 years).

In both cases the dose commitment is mostly from caesium. The latter figure is a conservative estimate which was acknowledged verbally as perhaps 10 times too *high* at the PARM and more recently in Soviet media reports [Nuclear News 1987].

Design / Operational Changes for Chernobyl-type Reactors

A number of design and operational changes for Chernobyl-type reactors were presented by the Soviets at the meeting.

Design:

1 Improved effectiveness of emergency shutdown will be achieved in the short term by increasing from 30 to 80 the equivalent number of control rods normally inserted in the core, and also by limiting their uppermost removal position to 1.2 m from the top of the core.
2 Additional operating information will be made available to the operator in the control room.
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, a faster 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 maximum reactivity below prompt critical (for the most severe accident) and also to provide rapid reactor shutdown.

Operational. The areas that will receive emphasis are: 1)adherence to operating procedures, 2)clarification of command responsibilities, and 3)improvement of the man-machine interface.

Design Aspects Relevant to the Accident

This section identifies aspects of the Chernobyl unit 4 design and operation relevant to the accident. More detailed descriptions of the design are given in reports presented in August 1986 [USSR 1986; INSAG 1986].

Conceptual Basis

Chernobyl unit 4 is of the RBMK (roughly translated as 'large reactor with tubes') type, and the most recent of the 1,000 MW(e) series. It is a graphite-moderated, boiling-light-water-cooled, vertical pressure tube design, using enriched (2% U-235) UO_2 fuel with on-power refuelling. It utilizes a direct cycle, to produce electricity from twin turbines (see Figure 1).

The reactor core is shown in Figure 2. One of the key

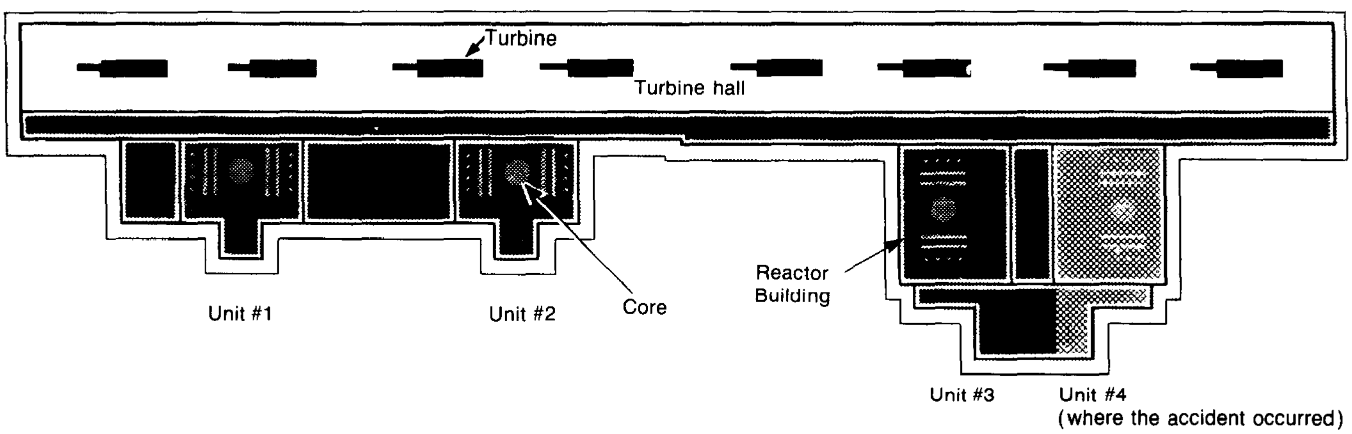


Figure 1 Layout of four reactor units.

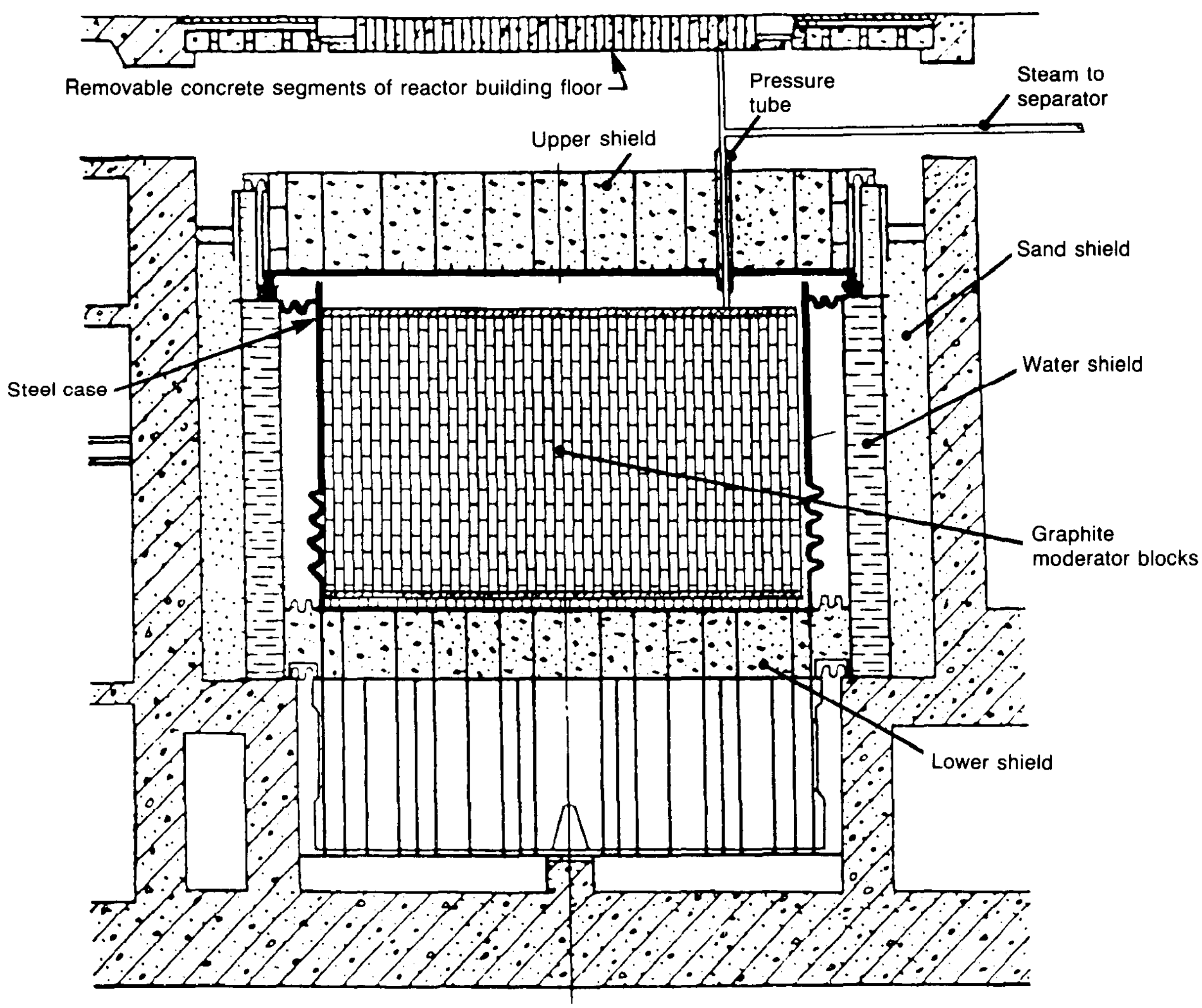


Figure 2 Cross-sectional view of reactor vault.

reactor physics parameters in the equilibrium fuel state is 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 0.2 mk / % void, whereas at the accident conditions it was reported to be 0.3 mk / % void. (Note that at their *normal* operating conditions, i.e., above 20% full power, the void coefficient is about 0.05 mk / % void.) Thus the overall fast power coefficient (which includes both the positive void coefficient and the negative effect of fuel temperature increase) is negative under normal high-power conditions, but positive

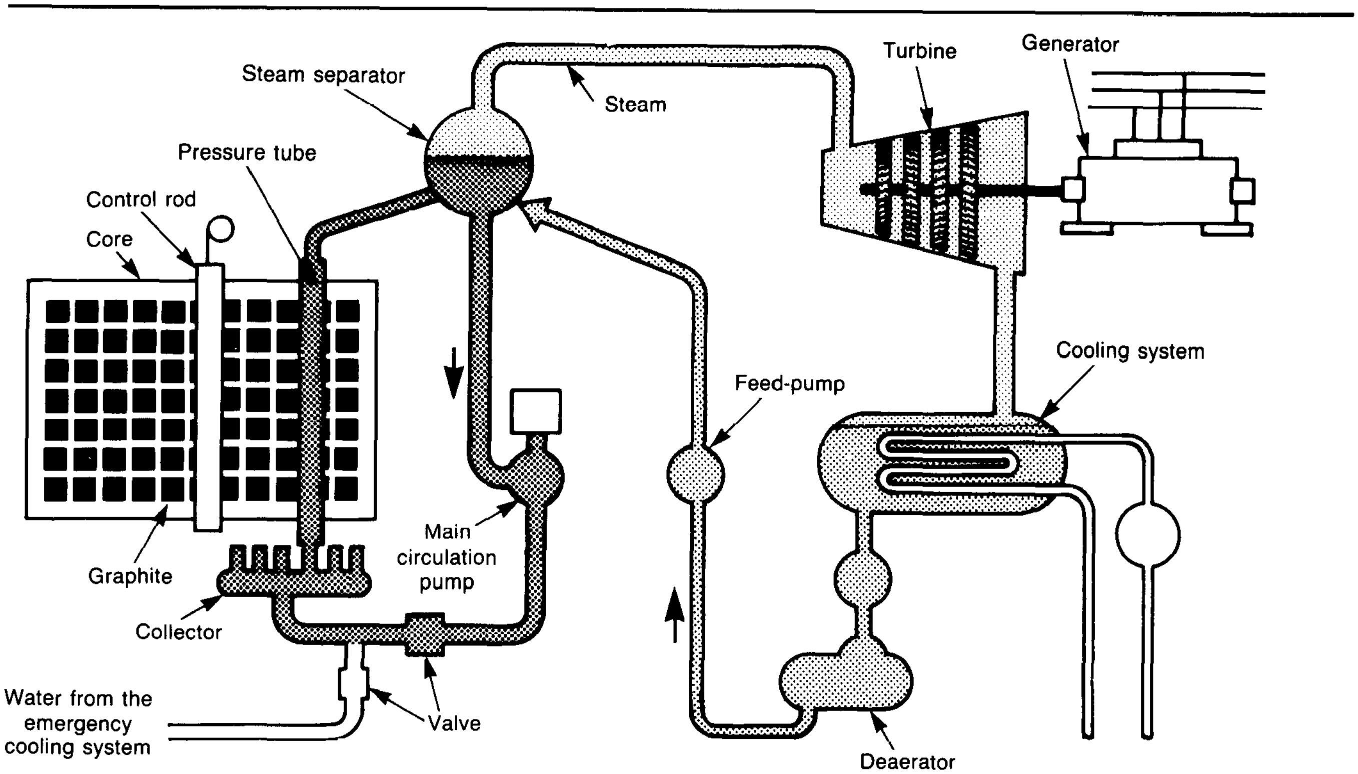


Figure 3 Schematic Diagram of the RBMK-1000.

at low power (below ~ 20%), as was the case just before the accident.

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 power distribution, and, in the extreme, to local criticality. In the RBMK reactor 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 reactor design is the use of the primary circuit as a 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.

With respect to emergency shutdown, the most important features are 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 equivalent rods were in the core at all times. This heavy reliance on administrative control was traced to early USSR experience in which operators were more reliable than automatic systems.

Thermalhydraulic Design

The RBMK thermalhydraulic design is based on a boiling water, direct-cycle heat transport circuit (see Figure 3). 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 circuit loops, which can be interconnected to a single turbine-generator at low power (as at the time of the accident). The primary circuit flow is driven 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).

Containment Design

The containment 'localization' system at Chernobyl was a recent RBMK design (see Figure 4). In this design the containment was divided into local compartments with distinct design pressures and relief / pressure-

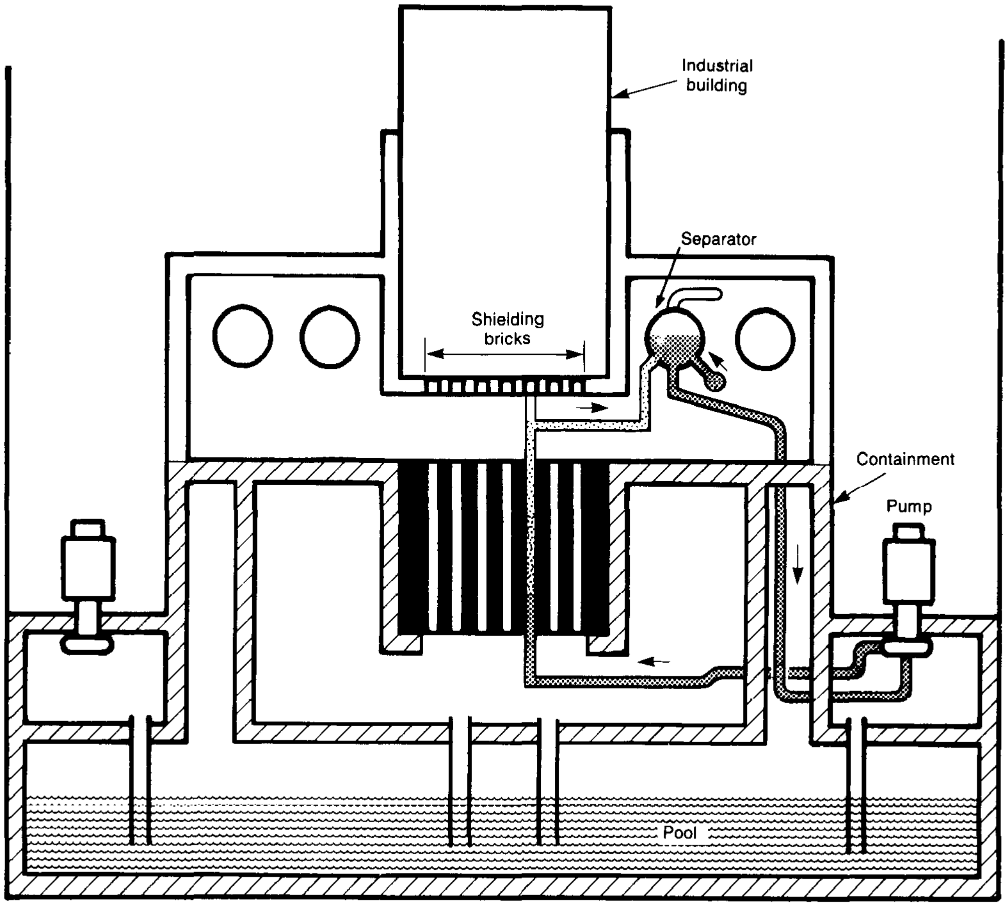


Figure 4 Chernobyl containment.

suppression to the water-filled 'bubbler pond' in the bottom of the building.

The top portion of the reactor (risers, separators, steam lines, fuelling machine room) was not within a pressure-retaining containment. For small pipe breaks in this group (e.g., a riser tube rupture), it is believed that the Soviets felt the large fuelling hall was adequate for the limited discharge rates and low expected levels of radioactivity. In any case, they stated the impracticality of building a containment of this size.

Pressure relief for the graphite core vessel was provided by eight 30-cm pipes connected to the bubbler pool. Relief capacity was stated to be capable of handling a single channel rupture.

In the accident, the steam explosion led to multiple pressure tube failures, which caused a pressure rise in the reactor vault, well beyond design capacity. Thus, the containment localization system played no real role in accomodating 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 possibly chemical, explosions that occurred.

Key Design Issues for Chernobyl

Variation of Void Reactivity with Reactor Operating State

Background

At 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 at the U.S. Department of Energy suggest that *positive reactivity was also added by the shutdown system* [Chan *et al.* 1987; U.S. DOE 1986]. Normally, the absorber rods are attached to graphite displacers or followers, to increase their worth. As

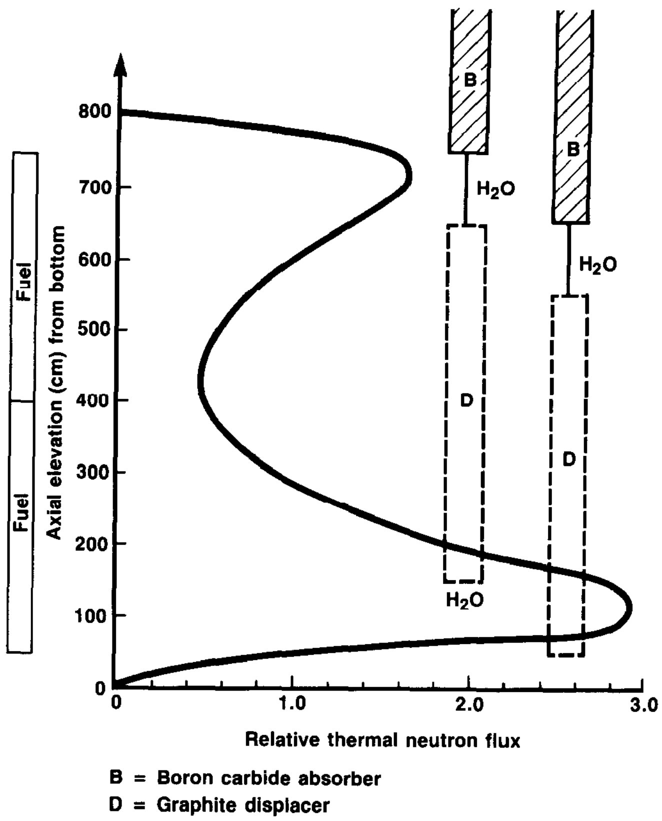


Figure 5 Axial flux distribution preceding accident.

they are inserted, the absorber rods move into the high-flux region in the centre of the core, which was previously occupied by the graphite, so the absorber rod effectiveness is enhanced (see Figure 5). 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, where most of the reactor power was being generated. Thus, when insertion of the absorbers first started, 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. Thus operating the plant in an abnormal condition resulted in an unusually large holdup of void reactivity, exacerbated by a deficient shutdown system design (see below, Shutdown Systems and Reactor Control), which led to the large power excursion and the resultant core damage.

Chernobyl Design

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

1 the use of H_2O coolant;
2 the relatively high temperature of the moderator (~700°C), 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 which, indeed, just before the accident, was decoupled into top and bottom halves; and
4 the requirement of significant reactivity hold-down in solid absorber rods due to the use of enriched fuel and due to the need to be able to override xenon buildup and the impracticality of using soluble poison in a solid (graphite) moderator.

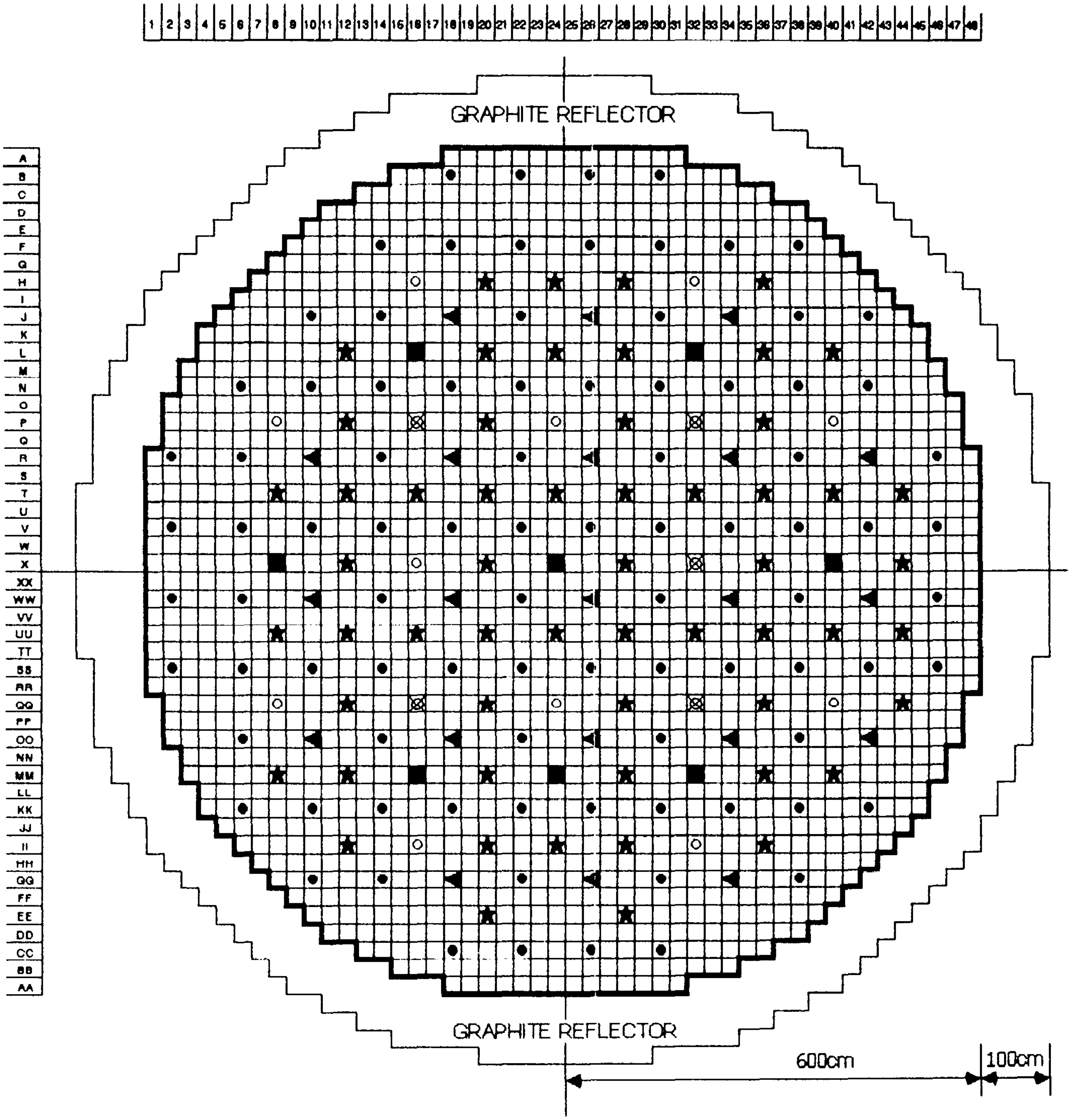
The use of H_2O coolant. The RBMK-1000 reactors are cooled with boiling H_2O. The mean coolant density is about 0.5 kg / L and the mean exit steam quality is 14.5%. The relatively high absorption cross-section of H_2O means that the reactivity of the coolant, due to absorption of neutrons, is high. This is a major contributor to the positive lattice void reactivity in the RBMK-1000.

Effect of moderator temperature on void reactivity. Since the moderator temperature is significantly higher than the coolant temperature (700°C vs 280°C), moderated neutrons are slowed down further after entering the fuel channel.

Neutronic decoupling and void reactivity. The RBMK-1000 has been calculated at AECL to have a first azimuthal mode subcriticality of between 6 and 7.5 mk [Gulshani, Dastur, and Chexal 1987]. 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. This was particularly the case just prior to the accident. (By comparison, the corresponding value for the CANDU 600 is 17 mk [Dastur 1981]; i.e., the reactor behaves much more uniformly.) The same phenomena are true for other harmonics. Therefore, void reactivity addition in the RBMK-1000 results in a complex power shape requiring complex trip logic to recognize the accident in time.

Effect of absorber rods on void reactivity. In the Soviet literature [USSR 1986] on the RBMK-1000 design, it is stated that the designers have used the effect of flux and spectrum changes due to the presence of absorber rods (see Figure 6) 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; the purpose is to adjust the neutron flux distribution such that, on voiding, the flux increases in the several sets of absorber rods and the role of the coolant as moderator are enchanced. 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. According to the Soviets, for fresh fuel the void reactivity coefficient is negative. For equilibrium fuel it is positive (about 0.05 mk / % void) for normal operating conditions, i.e., with about 80 absorber rods partially inserted into the core.



■	LAC (Local Automatic Control) LEP (Local Emergency Protection)	12	●	MC (Manual Control)	139
⊠	AC (Automatic Control Rod)	12	★	EP (Emergency Protection)	24
◀	Sar (Short Absorbing Rod)	24	○	DMEH (Flux Detector Assembly) Height, i.e. Axial	12
□	Fuel Channels	1673		Graphite Reflector	604
		=1884		Total Sites In Model	2488

Figure 6 Reactor physics model of RBMK-1000 core.

The maximum possible void reactivity coefficient is 0.2 mk / % void at normal operation with a minimum of 30 equivalent rods inserted in the core. The coefficient was as high as 0.3 mk / % void before the accident, as there were only 6 to 8 equivalent rods in the core.

Comments on the Chernobyl Design

The size of the system void reactivity in the RBMK-1000 reactor 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 this reactor 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), the flow cannot be reduced to match the power, and small changes in coolant conditions can have large effects on coolant void.

To summarize, then, the weakness of the Chernobyl design is that the void reactivity and the capability of the shutdown system depend significantly on the operating state of the reactor (and the ability of the operators to maintain the reactor within an allowable operating envelope). The Soviets themselves have indicated that operating procedures did not allow sustained operation (other than startup or shutdown) below 20% power.

CANDU Design

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.

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 inside diameter 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. If the coolant is lost from the fuel channel, there is a small reduction in the energy (or velocity) of the neutron population in the fuel.

This shift in neutron spectrum in the fuel alters the fuel neutron absorption rates in the thermal and epithermal ranges. In particular, the resonance absorption in U-238 decreases, and there is a 6.3-mk increase due to the increase in the lattice resonance escape probability. Since depletion of U-238 is minimal during the life of the fuel, this contribution to 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 U-238. This contributes 5.2 mk to void reactivity for the standard 37-element fuel bundle design. Due to the negligible depletion of U-238, 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-linear behaviour of the uranium and plutonium cross-sections with neutron velocity. On voiding there is a 3% increase in the U-235 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 the CANDU reactor is 16 mk when the fuel is fresh and decreases with irradiation. At equilibrium fuel burnup it is 11 mk [Rouben 1987].

Effect of absorber rods on void reactivity. In the CANDU design, the mechanism that leads to a change in void reactivity due to the presence of absorber rods is quite different from that at Chernobyl. Voiding of the coolant in the CANDU reactor 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 above (see Changes in neutron spectrum on voiding).

Comments on the CANDU Design

In direct contrast to the key weakness in the Chernobyl reactor design, the CANDU reactor physics is such that void reactivity does not depend on operating state, and therefore the shutdown systems can shut down the reactor, essentially independently 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 [CANDU 600 Safety Report 1984].

Compared to the Chernobyl design, CANDU has a *smaller* void coefficient under abnormal conditions, and the capabilities of the shutdown systems are more successfully matched to the reactivity coefficients (see below).

Shutdown Systems and Reactor Control

Background

The accident was characterized by a power excursion and an ineffective shutdown; the former, as noted above, may also have been initiated or worsened by the shutdown system design.

Chernobyl Design

Overall philosophy. Reactivity protection (shutdown) and control in the RBMK reactor is complex and requires manual involvement (see Figures 7 and 8). The control function of the RBMK-1000 reactor is divided into:

1 bulk reactivity control for power manoeuvering 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 automatic control system attempts to meet these demands. If the operator finds that the automatic 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, and 3) the operating value of the coolant void reactivity.

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

Required absorber rod positions. A significant feature of this mode of operation is that the maximum negative reactivity rate achieved in an emergency shutdown depends on the *number* of supplementary absorbers present in the core, and in *which locations* they are inserted. For this reason, the equivalent of at least 30

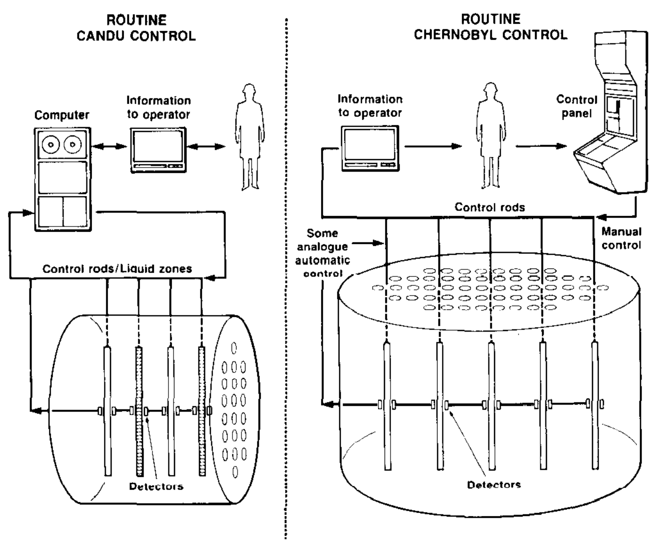


Figure 7 Controlling the power.

absorber rods are always required to be inserted at least 1.2 m into the core and spread reasonably uniformly over the reactor diameter. 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 and its graphite displacer are pulled out of the reactor.

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.

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 1). The Chernobyl design also had a limited number of spatial control rods (12) which were automatically controlled (see Table 2).

The automatic spatial control rods were designed to stabilize the most important radial and azimuthal flux modes. The 12 control rods are moved in such a way that the signals from 2 fission chambers near each control rod remain 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.

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 of the reflector are used for high flux and high rate trips. One description states that rate is monitored only below 10% full power. Some degree of

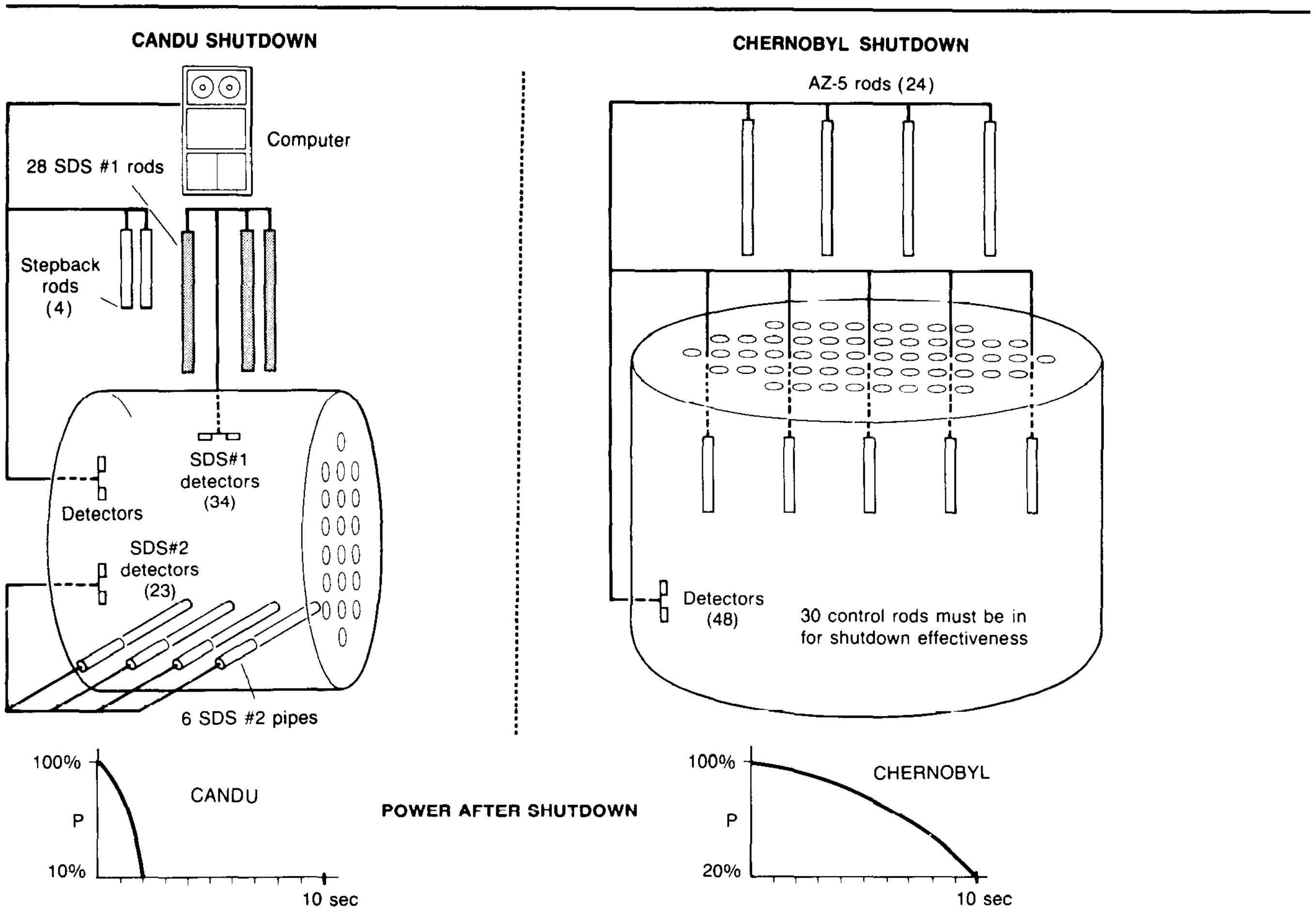


Figure 8 Shutting down the reactor.

spatial protection is afforded by tripping if setpoints are exceeded at 2 ion chambers on the same side of the reactor. A total of 8 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 One hundred and thirty radially distributed in-core flux detectors (using a silver emitter) are compared to appropriate pre-calculated 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. 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 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.

Comments on the Chernobyl Design

The RBMK protection (shutdown) system is fundamentally different from the CANDU shutdown systems (see Figure 8). 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).

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 (less than 2 seconds). This speed limitation is due to the hydraulic drag as the rods are driven or dropped into their water-filled guide tubes. 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.

Physics assessments at AECL show that the Chernobyl reactor is potentially subject to very local, very large flux perturbations [Gulshani, Dastur, and Chexal 1987]. 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

Table 1: Summary of Flux Measurement Devices at 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% full power) – alarm and protection action on absolute limit (above 10% full power)
84	Axially distributed silver flux detectors for – computer monitoring – alarm on relative deviation
24	Fission chambers for – automatic spatial control – local protection

(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 percent of full power. Their response is about a 25-second time constant for 90% of the signal and as 2.4-minute time constant for 10% of the signal. The burnout rate is about 20% per year, and the expected life about 3 years.

Table 2: Summary of Control / Shutdown Rods at Chernobyl

12	Rods for automatic control of total power (used 4 at a time)
12	Rods for automatic spatial control
24	'Short' rods for manual axial control
139	Regular rods for radial / azimuthal manual control
24	Emergency protection

respond quickly are small in number and would not adequately detect a very local power increase.

Finally, and most significantly, the protective system action is very slow, so that a power excursion is likely to experience 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 [Chan *et al.* 1987; U.S. DOE 1986].

CANDU Design

CANDU stations control reactor power automatically over the entire range from 6 or 7 decades below full power up to full power. Spatial control is done only above about 15% full power, because the reactor is spatially stable up to about 25% full power. At low powers, up to about 10% full power, 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 7). If their worth is inadequate, mechanical control rods are available for both positive and negative reactivity addition, again under totally automatic control. Manual reactivity adjustments are limited to poison addition to, and removal from the D_2O 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 lograte and high flux, but mostly by powerful, rapid shutdown. In CANDU 600, shutdown system No. 1 consists of 28 gravity-operated, spring-assisted absorber (shutoff) rods, and shutdown system No. 2 consists of 6 liquid injection pipes containing over 200 nozzles (see Figure 8). 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. Because the shutoff units act in the liquid moderator, they can be inserted very quickly. For example, the shutoff rod guide tubes are full of holes to allow the water to escape as the rods are inserted, reducing hydraulic drag.

In particular, each system 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 [CANDU 600 Safety Report 1984]. 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. The design has evolved since then. The Pickering A units (the first full-size CANDU reactors), put into operation in the early 1970s (near Toronto, Ontario), have 2 different shutdown mechanisms (shutoff rods and quick draining of the heavy water moderator). The shutdown is fully independent of the control, and, unlike the Chernobyl units, capable under any accident conditions of shutting the reactor down. The two shutdown mechanisms were made more powerful in later CANDU designs (Pickering B, Bruce A and B, CANDU 600, and Darlington A), and the logic was fully 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 that is fully independent of

Table 3: Summary of Flux Measurement Devices in CANDU 600

3	Start-up counters (installed temporarily only for initial startup and after very long shutdowns)
3	Ion chambers for control at low power
3	Ion chambers for SDS-1 emergency shutdown
3	Ion chambers for SDS-2 emergency shutdown
28	Platinum in-core flux detectors for control at high power (total power plus flux tilts)
102	Vanadium in-core flux detectors for – calculation of reactor flux shape by the computer every 2 minutes – automatic power reduction on high local flux
40	Platinum in-core flux detectors for SDS-1 emergency shutdown
23	Platinum in-core flux detectors for SDS-2 emergency shutdown

the reactor control system, and with an availability target of greater than 9,999 out of 10,000. There have been no shutdown system failures on test in NPD 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 is set by the large loss-of-coolant accident. As a result, the systems are more than capable of handling any conceivable reactivity insertion due to loss of reactivity control, from any initial power level.

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 2 seconds) and inserts a large negative reactivity; the shutdown systems are totally independent of the control system.

Containment

Background

As an immediate consequence of the accident, 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 (1,000 tonnes) can be seen on edge at the top of the reactor in the fuelling machine hall, with shreds of channels attached to it.

Table 4: Summary of Control / Shutdown Rods in CANDU 600

14	'Liquid Rods' (water-filled chambers) for control of total power and flux tilts
21	Adjuster Rods for control (normally fully inserted, but can be driven out, in banks, for extra positive reactivity)
4	Mechanical Control Absorbers for control (normally fully inserted, but can be driven or dropped in for extra negative reactivity)
28	Shutoff Rods for SDS-1 emergency shutdown
6	Liquid Poison Injection Pipes into moderator for SDS'2 emergency shutdown

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. It is clear that the Chernobyl containment was bypassed by the accident.

Chernobyl Design

The Chernobyl unit 4 RBMK 1000 reactor was fitted with (Figure 9) a containment consisting of: 1) enclosures covering parts of the reactor and cooling system, designed to withstand approximately 100 to 400 kPa(g); 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; and 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.

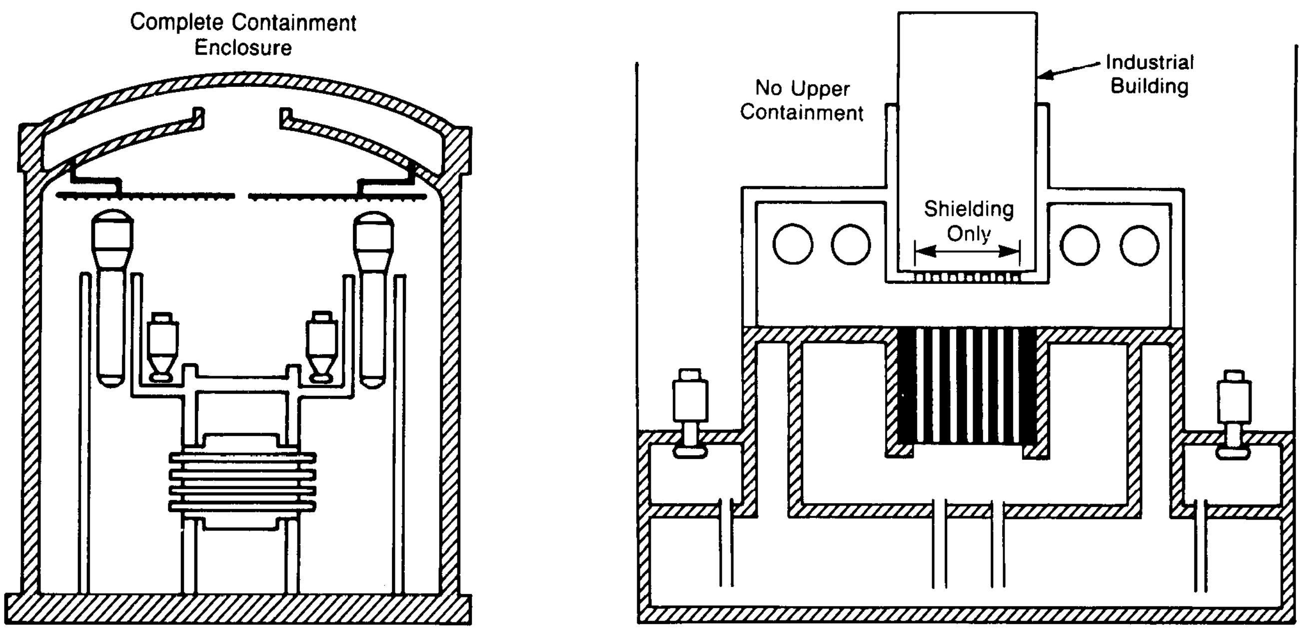
Core Container. Information provided indicates that the core of the reactor, including the channels and the graphite, is contained in a low-design-pressure (about 200 kPa) tank filled with inert gas. This tank is fitted with relief valves which lead down into the bubbler pond. A helium/nitrogen mixture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

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

Containment. The Soviets indicated [USSR 1986; INSAG 1986] that the Chernobyl containment included such features as:

1 double water pools (bubbler ponds) which condense steam from main steam safety valves, as well as from accidents;

2 a complex valving arrangement between compartments that swaps the 'wet well / dry well,' depending on failure location, and whose design is aimed at minimizing containment volume and design pressure;



600 MW CANDU **Chernobyl**

Figure 9 Containment structures.

3 a sprinkler cooling system for cooling of air during normal operation and after accidents; and
4 a system to remove hydrogen from the enclosure. Sources of hydrogen are controlled by catalytic combustion. The system has a capacity of 800 m^3/h and is designed for a postulated release of hydrogen from the oxidation of 30% of the fuel sheaths.

Comments on the Chernobyl Design

There are a number of pathways by which activity released from fuel in the reactor core could directly affect the reactor operators or public:

1 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.
2 Since the reactor is of 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 (e.g., 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.
3 Failures of the cooling of irradiated fuel in the fuelling machine would not be contained, but the consequences would be limited to one or two channels' worth of fuel.

CANDU Design

There are 3 different containment designs used for CANDU plants:

1 The single-unit containment envelope (see Figure 9) 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.
2 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.

The Pickering *reactor* containment is similar to CANDU 600. Bruce and Darlington designs have a smaller reactor containment which encloses 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 seals form the containment boundary.
3 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 CANDUs. 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 is delayed relative to the steam release because of its low power rating.

The CANDU 600 containment has significant capability beyond its design basis. It has 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., Point Lepreau), these values are [MacGregor *et al.* 1980]:

design pressure	:	124 kPa gauge
test pressure	:	143 kPa gauge
cracking pressure	:	~330 kPa gauge
failure pressure	:	~530 kPa gauge

The containment is designed for rupture of the largest main cooling pipe. The maximum pressure inside containment for this accident is predicted to be less than 70 kPa(g), well below the design pressure.

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

- failure of a normal control system,
- *plus* failure or incapability of stepback,
- *plus* failure of shutdown system No. 1,
- *plus* failure of shutdown system No. 2.

Such an accident has an estimated frequency of less than 1 in 10 million years per reactor in CANDU 600 – much less frequently than in the Chernobyl reactor because of CANDU's stepback and its redundant and independent shutdown systems [Snell 1986]. 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 [U.S. NRC 1975; U.S. NRC 1987]. Nevertheless, although a hypothetical severe power excursion could damage the CANDU 600 reactor core, the energy would be released into a large containment volume (for the CANDU 600 the volume is about 50,000 m^3 compared to about 100 m^3 for the core container at Chernobyl) and pressures in the CANDU 600 containment would be much lower. Analysis of such events is quite speculative and depends 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 [MacGregor *et al.* 1980]. The CANDU 600 containment thus is likely to retain much of its effectiveness, even for such a severe and improbable accident.

The CANDU 600 moderator tank relieves to the containment enclosure through 4 relief pipes with a total relief area of 0.66 m^2. The relief pipes are sealed by rupture discs with a 138 kPa gauge break pressure. All CANDUs 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 could 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 [CANDU 600 Safety Report 1984]. The production of hydrogen is limited by the effectiveness of the moderator heat sink, so that very little of the pressure tube zirconium reacts. Buoyancy flow and cooling fans mix the hydrogen quite rapidly throughout the containment volume and quickly reduce local concentrations in compartments 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 these stations are now equipped with hydrogen igniters and the remaining ones will be similarly outfitted in 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 5 provides a comparison of the CANDU and Chernobyl containments.

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. The Pickering and 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 extension 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 5: Containment Comparison Summary

Containment item	*Multi-unit* CANDUS *Pickering A and B*	*Bruce A*	*Bruce B*	*Darlington A*	*Single-unit* CANDUS *Gentilly-2*	*Lepreau*
Containment volume (m^3) (2)	594,700	212,900	212,900	305,100	48,500	48,500
Reactor building design pressure:	41	69	82.7	96.5	124	124
cracking pressure: (kPa gauge)					331	331
Wall condensation area (m^2)	61,300	57,500	57,500	61,100	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) (3)	21.3	11.8	11.8	9.2	2.9	2.9

Chernobyl data on next page.
Notes: 1 Some data are approximate.
2 Includes vacuum building volume × 1.9.
3 Only coolers on Class III electric power are credited.

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
Supression pool water (m^3)				5,000 × 2		10,000

Reactor vault (Chernobyl) or calandria (CANDU) *data*	CANDU *Pickering*	*Bruce*	*Darlington*	*600 MW*	*Chernobyl**
Relief pressure (kPa gauge)	138	138	138	138	185
Yield pressure	Estimated 1.01.2 MPa				Estimated 0.7 MPa

*Chernobyl design pressure 200 kPa.

Heavy Objects Above the Core

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 at Chernobyl fell over due to the explosion.

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 capacity service crane.

The fuelling machine duty in RBMK-1000 reactors can be as much as 4 to 5 channels a day, so that, in equilibrium operation, the fuelling machine is suspended over the core for much of the time.

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.

CANDU Design

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

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 release 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.

Graphite Moderator

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, in addition to being a heat sink, it provided a continuing source of energy to distribute fission products up to 1,000 metres above the reactor.

Chernobyl Design
The moderator consists of 1,700 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 1,670), 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 that at full power operation. This avoids the large reactivity changes resulting 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, i.e., 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 come clearance – some papers suggest a 0.04–0.05-mm gap. The maximum local graphite temperature has been stated to be 750°C. It is reported that leaks in pressure tubes can be detected by sampling the moderator gas.

Comments on Chernobyl Design
The effectiveness of heat removal from the graphite must be very dependent upon local conditions at the graphite rings on the pressure tubes. On the one hand, it can be postulated 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 11 years ago [NNC 1976]. In addition, the bulk graphite is poorly served with temperature monitors – 20 thermocouple holes in 1,700 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. On the other hand, for severe accidents involving potential pressure tube deformation, the graphite can actually act as a heat sink if the channel temperature rises above local graphite temperature, because of its large mass. This is probably why there was no 'meltdown' at Chernobyl after the initial explosion. In contrast to CANDU, the channels are at higher temperatures for a severe accident (e.g., loss of coolant/loss of emergency core cooling), and therefore more of the zirconium is able to react with steam to form hydrogen. Of course, this is 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 [USSR 1986]. 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, flashes to steam and increases the pressure in the tank. The U.K. review points 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 appears to be no published Soviet accident analysis 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 Whiteshell Nuclear Research Establishment (AECL) show that graphite oxidation in air is exothermic, with ignition around 650 to 750°C. In steam, the reaction is endothermic, becoming significant around 1,100 to 1,200°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 is slow). For severe accidents, with graphite overheating, it imposes a requirement on the reactivity depth of the shutdown systems – it is not know how this is dealt with.

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. The total 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 4 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 and associated calandria tube, with no credit for the strength of the surrounding calandria tube.

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 either sags 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, expands 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 no significant melting of the UO_2 fuel occurs and the channels remain intact. 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 [CANDU 600 Safety Report 1984].

A spontaneous pressure tube failure at normal operating pressures may or may not cause a failure of the surrounding calandria tube. 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, some calandria tubes 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.

Source Term Considerations

Background

The accident at Chernobyl pointed out a significant effect of the lack of a complete containment. During the accident, oxidizing conditions occurred, such that fission products that are volatile at 1,700°C (iodine, caesium, tellurium) were released as elemental gases. In the case of a severe accident in CANDU it is expected

Table 6: 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%	3×10^{-5}%	20%
Cs	53%	not detected	10–13%
Ru	0.5%	not detected	2.9%
Ce(group)	nil	not detected	2.3–2.8%

that reducing conditions would occur and that these fission products would be released to containment as chemical compound aerosols.

Chernobyl Phenomenology

In general, the composition of the aerosols released during the accident was reported to be characteristic of the irradiated fuel composition, except for enhanced release of elemental iodine, caesium, 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 1,700°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 a substantial fraction of the released iodine would likely be volatile I_2 gas. As the I_2 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., 1,500°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.

CANDU Phenomenology

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, large quantities of volatile I_2 would not be expected to form in a CANDU containment system.

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 Three Mile Island unit 2 (TMI) and the Chernobyl unit, 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 6) [Collier and Myrddin Davies 1986], 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, would also occur for an accident in a CANDU reactor. Even if the containment building were leaking, major attenuation of the biologically significant radioactive releases would occur.

Other Concerns Raised

Pressure Tubes

Background

In this section the pressure tube designs of the Chernobyl unit and of CANDU are discussed.

Chernobyl Design

In the Chernobyl unit 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.

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

A mixture of helium and nitrogen, fed from the bottom end of the reactor, flows between the graphite columns. It provides a heat conducting medium 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 operation. The inlet end of the channel is connected directly to the coolant pipe by means of a welded connection.

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

Comments on Chernobyl Design

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

1 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).

2 The diffusion joint appears to limit maximum allowable heating and cooling rates to from 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.

3 As noted in the previous section, the response of the surrounding structure to a pressure tube rupture is important, yet not well understood.

CANDU Design

CANDU is a pressure tube, heavy-water-moderated, heavy-water-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 at 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 an inert 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.

Comments on CANDU Design

1 For many 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 tubes, the reactor can be shut down, and the pressure tube replaced. The moderator vessel is nevertheless designed to withstand a sudden channel rupture (both pressure and calandria tube).

2 Surrounding each of the channel assemblies is the cool (about 70-80°C) water moderator. If the pressure tube heats up to temperatures in the range of 650°C to 800°C, 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 defence 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 [CANDU 600 Safety Report 1984].

3 Severe fuel heatup or fuel melting due to channel blockage or flow reduction in a channel 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 greater than 90% of the channel flow area [CANDU 600 Safety Report 1984] and has never occurred in a CANDU reactor. 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.

4 There have been 2 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.

5 The rolled joints used in CANDU reactors have generally performed well. There were leaking pressure tube problems in the rolled joint area in Pickering A and Bruce A, associated with delayed hydride cracking of some tubes in highly stressed areas, resulting from improper rolling of the joint. Subsequent CANDU reactors have used an improved pressure tube installation procedure.

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 zirconium-niobium material which is used in all other CANDU 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 feature of pressure-tube reactors.

Computer Control

Background

Direct computer control was not used at Chernobyl – the Soviets reportedly felt it was not sufficiently reliable based on their early exerience.

Chernobyl Design

The actual *control* of Chernobyl appears to be mostly analog; from 0 to 0.5% full power, the control is manual, with special low-power ion chambers; from 0.5% to 6% full power, the control is non-redundant automatic control of 4 rods, based on 4 ion chambers; above 6% full power, control is dual redundant automatic control, with each redundant portion having 4 rods and 4 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, 2 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 1 such station computer. The PRIZMA program runs every 5 to 10 minutes, so is relevant for very slow power changes only.

Comments on Chernobyl Design

At Chernobyl, most of the basic spatial flux control is manual (i.e., 139 absorber rods). While it is possible to use this kind of control, it assumes a high reliance on the operator.

CANDU Design

CANDU stations make extensive use of direct digital control; this encompasses all reactor controls and all major process loops [for a detailed description see Ichiyen 1982]. The configuration consists of 2 identical computers running continuously in active / hot-standby mode. Internal self-checks and external checks 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 at Chernobyl.

Comments on CANDU Design

The dual computer concept has served well – there have been only a few instances of computer failure [Ichiyen 1982; Ichiyen and Yanofsky 1980]. 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.

Multi-Unit Containment

Background

Both the Chernobyl reactors and the majority of current CANDU reactors are multi-unit plants on the same site. The accident at the former forced a shutdown of all other operating units at the site.

Chernobyl Design

There are 4 operating units at Chernobyl, plus 2 more under construction. There is no sharing of containment facilities, but the operating units share a common turbine hall and some electrical services.

Comments on Chernobyl Design

The *physical* damage was apparently restricted to unit 4. However, an accident which spreads contamination as widely as Chernobyl did will restrict access to other units on the same site. An effective containment is key in 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.

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 are sucked into the vacuum building, and the entire structure stays below atmospheric pressure (leakage in *in*, not *out*) for many hours.

Since the primary 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 steam generator tube.

Comments on CANDU Design

The vacuum concept has been analyzed for the usual spectrum of accidents, such as a 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 [Hurst and Boyd 1972; Domaratzki 1984]. 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.

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) an ability to use the electrical and water supplies of the *other* units in emergencies, and 2) the presence of a large operational staff familiar with all the units on site. These two factors were doubtless true for Chernobyl as well.

Fire Protection

Background

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

Chernobyl Design

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 3 independent subsystems, which are in turn connected to the emergency diesel generators. The water supply for each system consists of 3 tanks with a capacity of 150 m^3. These tanks are filled from the plant general fire-fighting system.

Comments on Chernobyl Design

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

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 2-group philosophy: that is, the plant can be shut down and monitored and decay heat removed by either of 2 independent and spatially separated groups of systems. Fire suppression systems outside of the reactor building are conventional sprinkler systems, CO_2 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.

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 lubricating oil in the pump motors, and the 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 lubricating 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 firefighting systems in CANDU plants is underway.

Conclusions

The threat posed by reactivity accidents has long been recognized in nuclear programs worldwide. The 1952 NRX accident at Chalk River [Lewis 1953] spurred the development of fast, powerful, and independent shutdown systems in the Canadian nuclear program.

Nonetheless, it is prudent to review, in depth, the adequacy of our defences. 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 condition, etc., which could result in a reactivity excursion exceeding the capability of the CANDU shutdown systems.

The consequential fires (besides the graphite fire) at Chernobyl were well handled, under extreme circumstances (particularly radiation), by the firefighting crews. CANDUs all have fire protection programs included 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 to be learned.

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Notes

[1]CANadian Deuterium Uranium.

[2]Expected future dose to be received for a person who remains in the western USSR, but outside of the evacuated area.

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