Insights from Chernobyl on Severe Accident Assessment of CANDU Reactors

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Abstract

The accident at the Chernobyl-4 RBMK reactor near Kiev in the USSR on 26 April 1986 is described. The characteristics of the RBMK reactors are compared to those of CANDU reactors. Certain insights on the assessment of severe accidents in CANDU reactors are drawn from the Chernobyl-4 accident. In particular, the importance of the design of the safety shutdown systems in CANDU is recognized. The most significant lesson from the Chernobyl-4 accident is that primary responsibility for the safety of a nuclear power plant must lie with the operating utility itself, and all managers and operators must be fully conscious of their responsibility for worker and public safety.

Résumé

L'accident du 26 avril 1986 au réacteur RBMK Chernobyl-4, près de Kiev en URSS, est décrit. Les charactéristiques du réacteur RBMK sont comparées à celles du réacteur CANDU. Certaines conclusions sur l'évaluation d'incidents sérieux survenus à des réacteurs CANDU sont tirées de l'accident de Chernobyl, en particulier l'importance du système des mécanismes d'arrêt sécurité pour le CANDU. La plus importante leçon apprise de l'accident de Chernobyl est que la responsabilité première d'une station nucléaire doit reposer à la station elle-même et que tous les superviseurs et opérateurs doivent être conscients de leur responsabilités en ce qui a trait à la sécurité des travailleurs et à celle du publique en général.

Introduction

No one in the nuclear power field is ever likely to forget 26 April 1986, the date of the most serious accident ever at a nuclear power plant. The accident on that date to the Chernobyl-4 RBMK reactor near Kiev in the USSR resulted in the destruction of the reactor and building, the deaths of 31 plant and emergency workers, a major release of radioactivity, the evacuation of 135,000 people from a region within 30 km of the plant, and a significant collective dose of radiation to the population of the USSR and other parts of Europe.

The impact of the accident on world-wide public attitudes to nuclear power has been negative, as would be expected. Moreover, a number of nuclear power projects and commitments have been delayed or postponed and others threatened with cancellation.

In this situation, it is incumbent on those in various countries who recognize the present and potential future benefits of nuclear power to assess the accident at Chernobyl and to learn from it. In this way, they can ensure that their own reactor technologies and reactor safety practices provide high confidence that the benefits of nuclear power can continue to be gained at acceptably low risks to operators and to the public.

The purpose of this paper is to evaluate certain severe accident scenarios in CANDU reactors in the light of the Chernobyl reactor accident. No claim is made for completeness of the assessments presented here; rather they are representative of assessments in the areas of CANDU safety in which the author has some experience and with which he is familiar. It is hoped that the insights gained will contribute to the necessary on-going process of learning from the Chernobyl accident.

The Accident at Chernobyl-4

Information on the RBMK reactor and the accident was obtained chiefly from the report of the USSR State Committee on the Utilization of Atomic Energy to the IAEA in Vienna [1], but also from other sources [2, 3], including personal communications with personnel of Atomic Energy of Canada, Ltd., Ontario Hydro, and the Atomic Energy Control Board.

Comparison of RBMK and CANDU Reactors

The Chernobyl-4 reactor was one of four RBMK reactors

Keywords: reactor safety, CANDU reactors, Chernobyl accident.

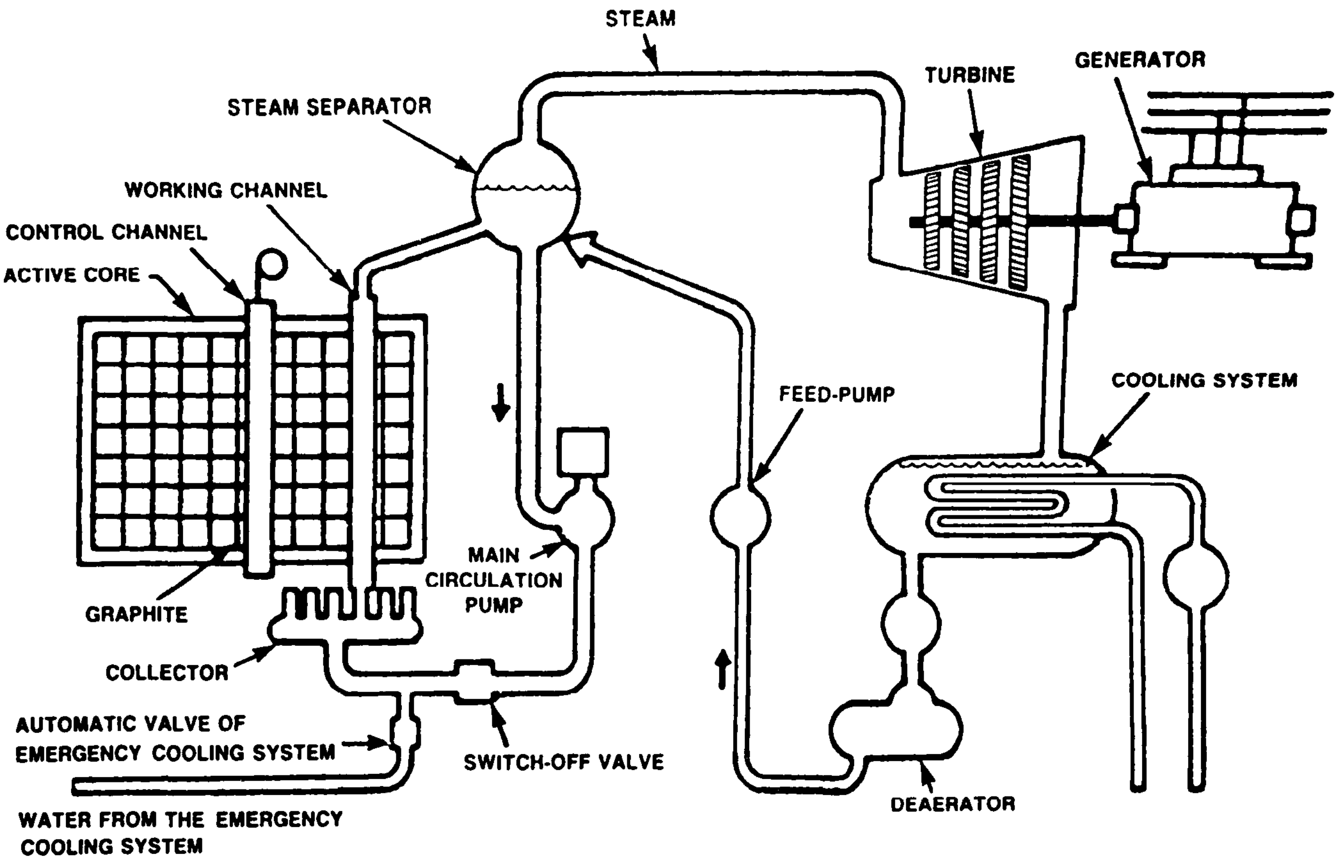

Figure 1. Schematic diagram of RBMK reactor.

at the site, each with a rated net output of 960 MW(e). The RBMK reactor is a graphite-moderated, boiling light water type, operated on a direct-cycle, as can be seen in Figure 1. Certain important characteristics of the RBMK reactor are given in Table 1, where they are compared to those for a typical CANDU reactor of somewhat lower thermal power: a Bruce-B reactor unit. Similarities of the RBMK to the CANDU reactor include the pressure-tube design, with Zircaloy-Niobium pressure tubes, and the use of on-power fuelling.

Among the RBMK characteristics listed in Table 1 which were of significance in the accident are the graphite moderator, the boiling coolant, and the very large core.

The moderator consists of a graphite block structure with the fuel channels running through the centres of the blocks. Heat generated in the graphite is removed by the primary coolant *via* graphite rings between the blocks and the pressure tubes. The resulting moderator temperatures at the design point range from about 270 degrees Celsius to about 700 degrees Celsius, so that there are no Wigner energy problems, which led to the Windscale reactor accident [4]. The graphite block structure is located within a thin-walled metallic container. The voids in the container are filled with a nitrogen-helium blanket to promote heat transfer and to prevent oxidation of the graphite. The moderator both enhanced and mitigated the effects of the accident, as will be explained.

The boiling water coolant introduces voids into the core and results in positive reactivity feedbacks, as

Table 1: Comparison of Chernobyl Reactor with Bruce-B Reactor

	Chernobyl	*Bruce-B*
Type	RBMK	CANDU
Thermal power, MW	3200	2852
Moderator	Graphite	Heavy water
Coolant	Water (boiling, $x_0 = 14\%$)	Heavy water
Cycle	Direct	Indirect
Fuel	UO_2-2% enriched	UO_2-natural
Orientation	Vertical	Horizontal
Core outlet pressure, MPa	7	9.3
Pressure containment	Pressure tubes (Zr Nb)	Pressure tubes (Zr Nb)
Number of fuel channels	1660	480
Core diameter, m	11.8	7.07
Core height or length, m	7.0	5.94
Re-fuelling	On power	On power

Table 2: Comparison of Chernobyl Reactor with Bruce-B Reactor: Reactivity Worths of Control and Safety Systems for Equilibrium Core Conditions

	Chernobyl	*Bruce-B*
Total of automatic control systems, mk	$\sim$6–8[1]	$\sim$33[2]
Safety systems, mk	Total: $\sim$30[1]	SDS #1: 73.6[3] – 32 rods 53[3] – 30 rods[4] SDS #2: >300[5]

[1]Total worth of all automatic and manual control and protective system = 105 mk. Very slow emergency insertion rate ($\sim$0.4 m/s).
[2]Adjusters, zone controllers, control absorbers.
[3]Available within 2 seconds.
[4]With two most effective rods not available.
[5]55 mk available within 2 seconds.

Table 3: Comparison of Chernobyl Reactor with Bruce-B Reactor: Reactivity Coefficients

	Chernobyl	*Bruce-B*
Void coefficient at operating point	$+2.0 \times 10^{-4}$/ vol.% void	$+1.14 \times 10^{-4}$/ vol.% void
Power coefficient (fast) at operating point	-0.5×10^{-6}/MW	-0.73×10^{-6}/MW
Fuel temperature coefficient	-1.2×10^{-5}/K	-4.2×10^{-6}/K
Moderator temperature coefficient	$+6.0 \times 10^{5}$/K	$+6.0 \times 10^{-5}$/K

we will see. The very large core presents problems of spatial stability of power distribution.

Also of significance in the accident were the characteristics of the control and safety shut-down systems, as given in Table 2, which compares the reactivity worths of these systems for the RBMK and CANDU reactors,[1] and Table 3, which compares the reactivity coefficients for the two reactor types. The reactivity worths of the automatic control system and of the safety shut-down system of an RBMK are considerably less than those of a CANDU, and the reactivity insertion rates of the safety shut-down systems of an RBMK are also significantly lower than those of a CANDU. Furthermore, the control and safety shut-down systems are not independent in an RBMK, as they are in a CANDU, which, in addition, possesses two independent safety shut-down systems. Both the reactor types have positive void coefficients, but that of an RBMK is almost twice that of a Bruce-B unit.

The RBMK emergency coolant injection (ECI) system consists of two high-pressure accumulator-driven subsystems, plus one pumped sub-system, to provide emergency cooling for the first one to two minutes. There is a separate pumped sub-system for the longer term. All sub-systems inject into the headers below the core.

Of importance in the accident was the lack of a contaminant over the reactor core in an RBMK unit. Most of the primary heat transport system is located in concrete compartments, called the accident localization system, but not the piping and other components above the core, which are located in the reactor building. The reactor building was not designed as a containment building. There is also a steam suppression pool below the reactor.

Description of the Accident

The accident occurred during a low-power test before a scheduled shut-down to demonstrate the ability of a turbine-generator, disconnected from the grid, to provide power for the short-term emergency-coolant pumped system during the turbine-generator rundown after interruption of the steam flow. This mode of power supply to the ECI is necessary, in the design-basis accident of a pipe break plus loss of offsite power, to run the pumps before the stand-by diesel generators can pick up the load. Such tests had been performed successfully and safely on other RBMK units, and a test had previously been done safely, but not successfully, in Chernobyl-4.

However, in this case, an operator error combined with a number of violations of procedures, and with the characteristics of the RBMK, to cause a disastrous accident. A very significant factor in the accident was pressure on the operators to complete the test successfully, since the next opportunity to undertake it would not occur until the next scheduled shut-down in a year's time. The following description of the accident, taken mainly from reference 1, is based on a reconstruction of the events by the Soviet authorities, using instrument charts and real-time analytical simulations of the reactor neutronics, thermohydraulics and control and safety shut-down system actions.

Prior to the test, in preparation for shut-down, the operators reduced reactor power to about 1,600 MW (half-power) and shut-down one of the two turbine generators supplied by the reactor. In accordance with the planned test procedure, the ECI system was blocked, to prevent spurious injection during the test. However, at this point, the grid demand resulted in the unit being required to continue to operate for about nine hours at 1,600 MW, still with the ECI system blocked-off in violation of operating rules.

Power reduction was then resumed, since the test was to be performed at an initial reactor power of 700 to 1,000 MW. However, when the operator switched from local automatic power control to bulk automatic power control, which was required for low-power operation, he failed to establish correctly the controller set-point, with the result that the power fell below 30 MW. Only after some time did the operator succeed in stabilizing the power at 200 MW. Power could not be raised higher because of the build-up of xenon during the long period at part load and the negative effect of the increased water content in the core following power reduction.

Additional primary coolant pumps were started up so that the coolant flow rate would still be adequate to cool the core after the turbine-generator run-down following its isolation from the steam supply. The normal reactor trip which would shut down the reactor with both turbine-generators valved out was also blocked off. These steps were taken, under the pressure to complete the test successfully, so as to enable the test to be repeated with a different type of generator voltage control.[2]

The flow rate through the core was now much higher than pump cavitation limits would normally permit, not only because of the additional pumps but also because of the low power, which reduced steam generation rate, and hence void, and thus core hydraulic resistance. The steam pressure also was dropping because of the reduced steam generation rate. The operators, in attempting to stabilize the operating conditions without tripping the reactor, then blocked the reactor trips for low separator water-level and low separator pressure.

Because of the low core void and the continuing build-up of xenon, the core reactivity continued to drop, which resulted in the automatic control rods being withdrawn, and which also forced the operators to withdraw some of the manual absorber rods. The reactivity margin was now reduced below the level that required immediate shut-down of the reactor. Nevertheless, operation was continued. Just before the start of the test, the operators significantly decreased the feedwater flow rate in an attempt to stabilize the water level in the steam drums. This action resulted in an increasing inlet temperature to the core.

The core was now in a potentially very unstable condition with very little reactivity margin, and under power, flow-rate, and inlet temperature conditions such that there was only a low void near the core exit, giving a high sensitivity of void to power changes.

At this point, the test was begun by closing the stop valves of the operating turbine. This action resulted in the pressure in the steam drums increasing as the steam flow rate decreased, and the coolant flow rate decreasing as the turbine and pumps ran down. The core void fraction was now being influenced by the increasing pressure, the increasing core inlet temperature, and the decreasing core flow rate. The first factor tended to decrease void, the other two to increase it. The net result was a rapid increase in core void fraction, which caused a rapid increase in reactivity,[3] and therefore a rapid increase in power. The increase in power generated more void, which accelerated the power increase, a classic case of positive feedback. The control system could not respond rapidly enough to limit the power surge; the operator activated a manual trip 36 seconds after the test commenced, but this was ineffective because of the axial flux shape, the location of some of the absorber rods and their slow emergency insertion rate. In any case, the safety absorber rods did not insert fully, presumably because of damage to the core by this time. It is estimated that the power surged to 100 times full power in about four seconds.

The fuel overheated and disintegrated, steam and Zircaloy reacted to generate hydrogen, the fuel channels ruptured, which permitted steam to react with graphite to generate hydrogen and CO, and the moderator container ruptured, permitting H_2 and CO to mix with air. Two explosions in rapid sequence were heard, the first apparently associated with the rapid steam formation and resulting fuel channel ruptures, and the second possibly with a chemical explosion (CO, H_2, and air igniting).

Approximately 4% of the fuel was ejected from the core and the graphite moderator ignited, and eventually about 10% of its 2,500 tonnes burned before the fire was extinguished several days later.

The reactor building was destroyed and a number of fires were started around the unit, which were extinguished in a few hours.

Estimated radioactive emissions from the damaged reactor reached about 96 million curies by May 6, after which the releases dropped to minor levels. Data for the estimated releases are given in Table 4.

The adverse health effects of these emissions are not germane to the topic of the paper and so are not discussed here.

Certain potential severe accident sequences in a CANDU reactor will now be assessed in the light of the accident to the Chernobyl reactor.

Table 4: Chernobyl Reactor Accident Estimated Radioactive Emissions[1]

	Emissions,[2] Curies	*Per cent of core inventory*
Noble gases	45×10^6	~100
Iodine	7.3×10^6	20
Tellurium		15
Cesium – 137		13
Cesium – 134		10
Total	96×10^6	~6–8

[1]Radioactivity emitted up to May 6 1986, calculated as of May 6. Only minor emissions after May 6.
[2]Accuracy: ±50%.

Loss-of-Regulation Accidents in a CANDU Reactor

The Chernobyl reactor accident was essentially a loss-of-regulation accident in which the positive void coefficient of the RBMK reactor played a major role. Since the CANDU, being over-moderated like the RBMK, also has a positive void (and moderator temperature) coefficient, the question arises as to the susceptibility of the CANDU to a similar loss-of-regulation power excursion accident. Since recent CANDU reactors operate at low-quality conditions (~4%) at the core exit when at full power, some concern may exist on this point, although it is recognized that the 'stiff' CANDU heat transport system reduces the power-to-void feedback effect below that of the RBMK.

However, a major difference between typical CANDU and RBMK reactors in this respect is the much greater speed of insertion of negative reactivity by the safety shut-down systems of the CANDU compared to the RBMK, as can be seen from Table 2. Also, as shown in Table 2, the total reactivity worth of the two CANDU safety shut-down systems (SDS-1 and SDS-2) is much greater than that of the single RBMK emergency system.

To illustrate the importance of the speed of response and the worth of the CANDU safety shut-down systems, it has been calculated that, had the Chernobyl-4 reactor emergency protective system had the same worth and insertion rate as a Bruce reactor SDS-1, assuming that the manual trip occurred at the same instant, the reactor power would have been turned around at about 15% over-power, and probably no serious damage would have resulted [5].

Furthermore, the existence of the two separate, independent, completely redundant and diverse shut-down systems in CANDU, which are also independent of the automatic control system, would provide much greater emergency shut-down reliability for the CANDU than for the RBMK. Indeed, the main reason for the provision of the two independent shut-down systems in CANDU reactors is to ensure reliable emergency shut-down in all accident conditions, particularly considering the positive void coefficient of reactivity.

Therefore, it can be concluded that the rate of insertion and depth of the two independent shut-down systems in CANDU, and their independence from the automatic control system, would prevent, with a very high level of reliability, power excursion accidents similar to that at Chernobyl. The wisdom of providing two independent fast-acting, high-worth safety shut-down systems in CANDU reactors would seem to be vindicated by the Chernobyl accident.

Impairment or Blocking of ECI in a Large LOCA in a CANDU Reactor

As we have seen, the emergency coolant injection system of the Chernobyl-4 reactor was blocked to prevent spurious injection during the planned test.[4]

The question arises of the consequences of the blockage or impairment of the ECI system during a severe accident in a CANDU reactor.

The dual-failure accident of a large stagnation LOCA plus a loss of emergency coolant injection (LOECI) is a design-basis accident for a CANDU reactor in Canada. The AECB requires for this case, as for all dual-failure accidents, that the maximum dose to an individual in the public not exceed 0.25 Sv and that the collective public dose does not exceed 10^4 person-Sieverts [6].

Considerable analytical and experimental work has been expended over the years in Canada to demonstrate that these requirements can be met.

In such an accident, the fuel and pressure tubes overheat and the pressure tubes deform into contact with the calandria tubes. Depending on the size and location of the pipe break or other event causing the LOCA, the deformation of the pressure tube will consist of a uniform radial ballooning or an eccentric sagging, as shown in Figure 2. If pressure-tube overheating occurs early in the blowdown transient following LOCA, when internal pressure is high, it will deform by ballooning. If overheating occurs late in the transient, when internal pressures are low, it will deform by sagging. In either case, deformation of the pressure tube will provide a heat flow path of relatively low thermal resistance from the fuel to the separately cooled, low-temperature moderator, which thus provides a back-up heat sink for the stored and decay heat and heat generated by the exothermic Zircaloy-steam reaction.

The computer simulation codes, CHAN and CHAN-2, have been developed by AECL and Ontario Hydro to predict the thermal behaviour of a fuel channel for the case of a pressure tube ballooning or sagging into contact with a calandria tube [7, 8]. There is considerable experimental verification of the models used in these codes [9, 10]. Results obtained using the CHAN code for the thermal behaviour of the hottest point along a high-power fuel channel in a Bruce reactor, for pressure tube ballooning following a LOCA plus LOECI, are given in Figure 3, taken from reference 11. The results shown are for the worst residual steam flow conditions in the channel, considering the exothermic steam-Zircaloy reaction and cooling produced by the steam flow. Figure 3 shows that there will be no gross melting of the fuel in a CANDU reactor, in spite of the loss of ECI, and that the maximum pressure tube temperature remains low enough to ensure its integrity.[5]

A computer simulation code, IMPECC, has been developed at Carleton University, under contract to the AECB, to predict the thermal behaviour of a fuel channel for the case of a pressure tube sagging into contact with a calandria tube [12, 13]. The model used for the non-conforming contact thermal resistance between the pressure and calandria tubes in IMPECC has experimental confirmation [13, 14, 15]. Results obtained using IMPECC for the thermal behaviour of the hottest point (circumferentially and axially) along a high-power (7.5 MW) fuel channel in a Bruce reactor following a LOCA plus LOECI are given in Figure 4, taken from reference 16. Again, it can be seen that there will be no gross melting of the fuel and that the maximum

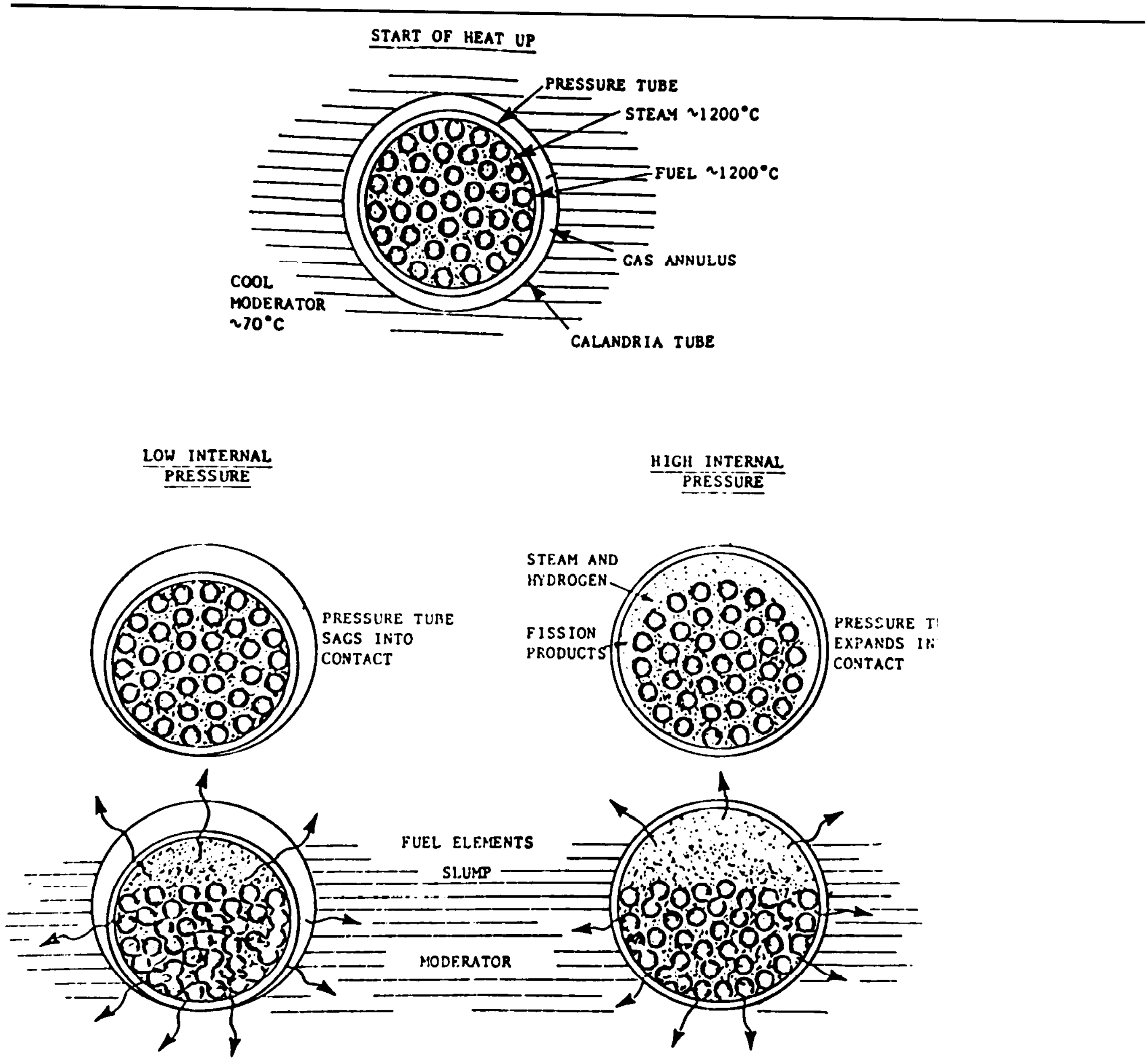

Figure 2. Deformation of CANDU pressure tubes in a LOCA plus failure of ECI.

temperature of the pressure tube is well below its melting point. Brown, *et al.*, using the code CHAN-2, also show that there will be no gross melting of the fuel and that pressure tube integrity will be maintained in the case of pressure tube sagging in a CANDU reactor fuel channel at an initial power of about 6 MW [17].

It has been concluded from these studies that there will be no gross melting of the fuel, although some fuel damage would certainly occur, and that pressure tube integrity will be maintained in CANDU reactors in the event of failure or blockage of emergency coolant injection following a loss-of-coolant accident, because of heat transfer to the low-temperature, independently cooled moderator.

The Chernobyl accident provides an interesting insight into the effectiveness of a separate moderator as a heat sink in a severe reactor accident. Figure 5 shows the estimated fuel temperature in the Chernobyl reactor as a function of time after the accident. It can be seen that, after an initial excursion, the temperature dropped to about 800 degrees Celsius and remained close to this value for a considerable time, then rose again to a peak of about 2,200 degrees Celsius before dropping off. This behaviour is attributed by the authors of reference 1 to the effect of the graphite moderator and structure acting as a heat sink, as well as the loss of some fuel particles and fission products from the core. It is noteworthy that the moderator acted as a heat sink, even though a significant portion was burn-

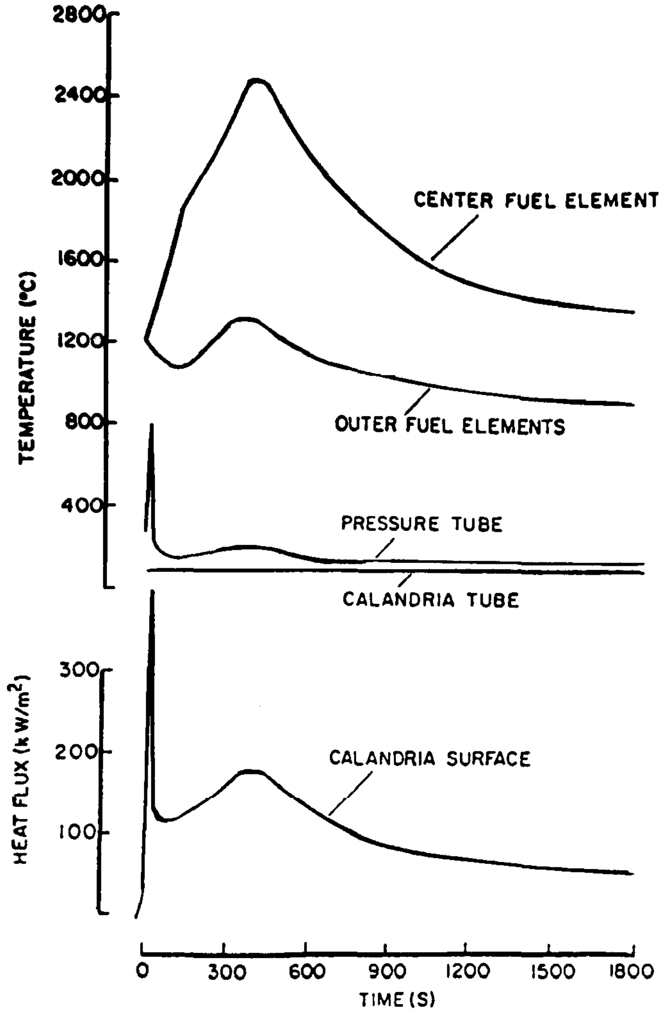

Figure 3. Thermal behaviour of a CANDU fuel channel in a LOCA plus failure of ECI. Pressure tube ballooning.

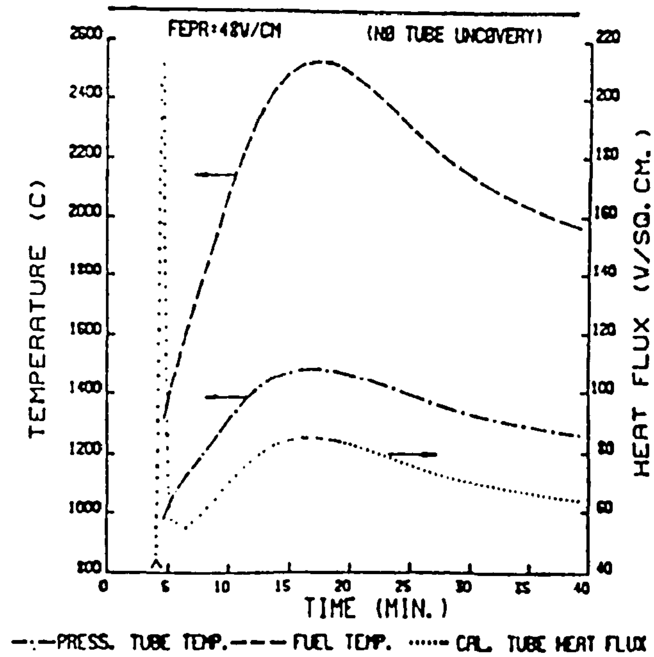

Figure 4. Thermal behaviour of a CANDU fuel channel in a LOCA plus failure of ECI. Pressure tube sagging.

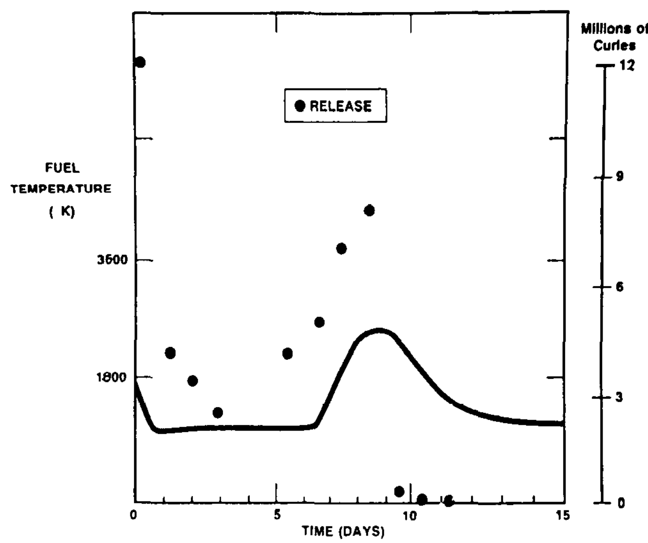

Figure 5. Estimated fuel temperature and radioactivity release in the Chernobyl reactor accident.

ing. The temperature rise from about day seven to day nine resulted from the average temperature of the moderator increasing slowly because of the heat being stored in it and because of the restriction of natural convection flow through the core by material dropped on the reactor from helicopters to reduce radioactive releases from the core. The ultimate temperature turnaround resulted from the effects of fission product decay and natural convection air cooling through the moderator blocks, as well as the introduction of liquid nitrogen below the core. It has been concluded that there was no gross melting of the fuel in the accident, except perhaps for some in the initial power surge.

While the conditions in the Chernobyl accident were greatly different from those for the hypothetical LOCA plus LOECI in a CANDU, the accident does demonstrate that a separate moderator can act as a heat sink in a very severe accident. It is quite probable that the effectiveness of the graphite moderator as a heat sink in the Chernobyl accident prevented even more serious releases of radioactivity than actually occurred, and eventually assisted in controlling the accident.

Fission Product Releases in a Severe Accident in a CANDU Reactor

As can be seen from Table 4, significant fractions of the inventories of the more volatile fission products were released to the environment in the Chernobyl accident. Lower fractions of the less volatile fission products and actinides were also released [1].

Two issues of significance in severe accidents in a CANDU reactor arise here, the first being the mechanism of large releases of fission products from the fuel under conditions of no fuel melting, and the second being the quantities of various fission products that would be released into and from containment in a severe accident in a CANDU.

It has been generally accepted since the Rasmussen study [18] that major releases of fission products other than noble gases, would not occur from the fuel in a severe reactor accident unless gross melting of the fuel occurred. Nevertheless, major fission-product releases from the fuel did occur at Chernobyl in the apparent absence of fuel melting. The initial power surge in the accident, to about 100 times full power, resulted in rapid energy deposition in the fuel to levels of considerably more than 300 cal / gm (1.25 J / kg) [1],[6] the value assumed in the USSR as that causing fuel element and pellet disintegration. The accident report states that fuel particles were carried into the coolant and embedded in the moderator [1]. With such significant disintegration of fuel, much of the grain-boundary inventory, as well as the free inventory of the high-volatile and medium-volatile fission product species (Xe, Kr, I, Cs, Te) would be rapidly released from the fuel, and the release of the grain-bound inventory would also be facilitated. Therefore, the initial release of radioactivity from the fuel was very high, which resulted in very high release to the environment, about 20–22 MCi at the time of release [1], in spite of little or no fuel melting.

For the first few days after the accident, finely dispersed fuel was carried from the reactor by graphite combustion products and hot air. This situation resulted from the oxidizing conditions caused by the air, and possibly CO, which, under the existing temperature conditions, brought about continuing disintegration of the matrix of the fuel elements, thus facilitating continuing release of fission products from the fuel [20]. In this period, the high releases from the fuel resulted in about 80 MCi of radioactivity, as of May 6, being released into the environment. The virtual cessation of fission product releases from the fuel into the environment after May 6 can be attributed to the introduction of liquid nitrogen below the core, which not only reduced fuel temperatures, but provided a nitrogen blanket for the core, which effectively stopped the oxidation process.

Therefore, significant fission product releases from UO_2 fuel can occur even in the absence of fuel melting, given a significant disintegration of a fuel matrix by oxidizing conditions following the accident. Such a situation would not be expected to occur in a CANDU reactor for two reasons. First, as we have already seen, the two fast-acting, high-worth, independent shut-down systems preclude power excursions of the magnitude experienced in the Chernobyl accident. Second, in the most severe accident in a CANDU reactor, a large LOCA plus LOECI, the conditions in the neighbourhood of any damaged fuel would be reducing rather than oxidizing, because of the Zircaloy-steam reaction, producing H_2 gas.

The second issue that arises from a consideration of the fission-product releases in the Chernobyl accident is the quantities of various fission products that might be released into and eventually from containment in a severe accident to a CANDU reactor. The very low levels of iodine (13 to 17 curies) and cesium (virtually none) released to the atmosphere in the Three Mile Island accident [21], in spite of significant core melting [22], led to re-assessment of past experience and existing knowledge of fission-product behaviour, as well as to stimulation of more intensive research in this field. The results of this work have shown, in general, that it would be expected that almost all fission product species except the noble gases would be retained to an overwhelming extent in liquid water in accident sequences in water-cooled and moderated reactors, considering the water-chemistry conditions in these reactors and the chemical forms of iodine and other fission products [11, 23, 24].

Also, experience and studies have shown the importance of the existence of a high-moisture atmosphere in the reactor building following a severe reactor accident in promoting such processes as aerosol formation, adsorption, and deposition for removal of fission products from the atmosphere [23, 25, 26]. Thus, we would expect that very little iodine, cesium, or other fission products would be available in the atmosphere for ready release to the environment in a severe accident in a reactor with large water inventory, as was observed in the Three Mile Island accident and in other accidents in water-cooled reactors [23].

Although the Chernobyl-4 reactor was cooled with boiling water, the water inventory would be relatively low compared to that of water-moderated reactors. The explosions at the moment of the accident, followed by the moderator fire, would have dispersed the water and ensured that the atmosphere around the reactor remained dry. Therefore, processes for fission product removal from the atmosphere would not have been effective. Thus, the high releases of iodine, cesium and tellurium that occurred would be expected. It should be noted that the previous reactor accident which released the greatest amounts of fission product species, other than noble gases, occurred in the Windscale reactor – graphite-moderated and gas-cooled, and thus with no water inventory [4, 23].

We would, therefore expect that, following a severe accident in a CANDU reactor, there would be very low concentrations of fission products, other than noble gases, in the atmosphere in the reactor building – and thus readily available for potential release to the environment – unlike the situation in the Chernobyl reactor accident.

The Role of Containment in a Severe Accident in a CANDU Reactor

There was no containment over the top of the Chernobyl reactor, nor over the steam-water piping and other components above the core. The reactor building was of conventional industrial building design and was not designed as a containment building [1, 3]. Even if the reactor building had been designed as a containment building similar to those used for single-unit stations in other parts of the world [27], it is uncertain whether it would have survived the initial explosions intact, considering their very large energies. Nevertheless, especially since a significant fraction of the explosion energy must have been used in rupturing fuel channels and piping, as well as lifting the 1,000 tonne reactor cover plate several metres, there is at least some probability that a standard-design containment building, while suffering some damage, might not have failed catastrophically. Thus, not only might the initial large release of fission products been reduced significantly, but the presence of the building might have permitted inherent removal processes for air-borne fission products, plus radioactive decay, to reduce subsequent releases also.

The containment system in CANDU reactors is the ultimate line of defence in the defence-in-depth design philosophy. Whether it is of the high-pressure type for single-unit stations, or of the low-pressure, vacuum building type for the Ontario Hydro multi-unit stations, it is designed to cope with the maximum energy release and to prevent or minimize the release to the environment of fission products from a large stagnation LOCA combined with a LOECI. The performance of the containment in such an accident must limit the maximum individual and collective doses to the levels prescribed by the AECB's dual-failure criteria [6].

The containment in a CANDU need not be designed to resist a reactor-power excursion of the magnitude experienced in the Chernobyl accident, because the two independent, fast-acting, high-worth shut-down systems, coupled with the inherent characteristics of CANDU, virtually preclude such accidents under any foreseeable conditions in a CANDU. Nevertheless, the CANDU containment can survive accidents more severe than the above design-basis accident, and thus limit the release of fission products to the environment even in such cases. A study has been undertaken at Carleton University, under contract to the AECB, of severe accidents in which the moderator cooling system fails or the moderator heat sink is lost in a LOCA plus LOECI, both highly improbable sequences of events [28, 29]. The study shows that, although gross fuel melting would eventually occur after several hours in these cases, assuming no operator intervention, the molten core would be effectively contained in the calandria vessel, separately cooled by the shield tank cooling system, and would eventually solidify there. (The calandria vessel acts as an inherent core catcher.) Although fission product releases into containment would be very large in this case, there would probably be no consequent failure of containment, as can be seen from Figure 6. Figure 6 shows the estimated pressure transients in containment following a failure of the moderator cooling system during a LOCA plus LOECI in the Bruce-A station, assuming no dousing after the initial one. The peak pressure is seen to lie between 35 kpag and 55 kpag. The higher of these two values is about the same as the containment design pressure difference (~50 kPa) and is well below the test pressure difference (~80 kPa) [28]. This peak pressure difference would not cause any containment failure [30].

While, obviously, the fission product leak rate from containment in this case would eventually be higher than in the design-basis accident of LOCA plus LOECI, the intact containment would provide continuing effective conditions for natural removal processes for iodine, cesium, and other fission product aerosols from the high-moisture containment atmosphere, and thus would limit releases of these fission products from containment.

General Insight

When we examine the magnitude of the Chernobyl accident, with the complete destruction of the reactor and building, many fuel fragments ejected from the core, and great quantities of fission products released, it appears that this may have been the ultimate reactor accident. It is hard to visualize another situation in which the inherent characteristics of a reactor, coupled with human error and many violations of procedures, could combine in such a way as to produce a greater disaster. If so, even though different weather conditions might have resulted in greater predicted adverse health effects, the large worker death toll, the very high economic costs of population evacuation and foregone crop and land use, and the slightly increased risks of cancer to the general population of the USSR, may represent a real upper limit to the consequences of any power reactor accident. This possibility should be considered in future power reactor risk studies.

Conclusions

The insights in this paper into assessments of severe accidents in a CANDU reactor gained from a study of the accident to the Chernobyl-4 RBMK reactor are, of course, based on a first, rather rapid analysis of the accident. But, while some of the technical details may change on further study, the insights should probably remain valid.

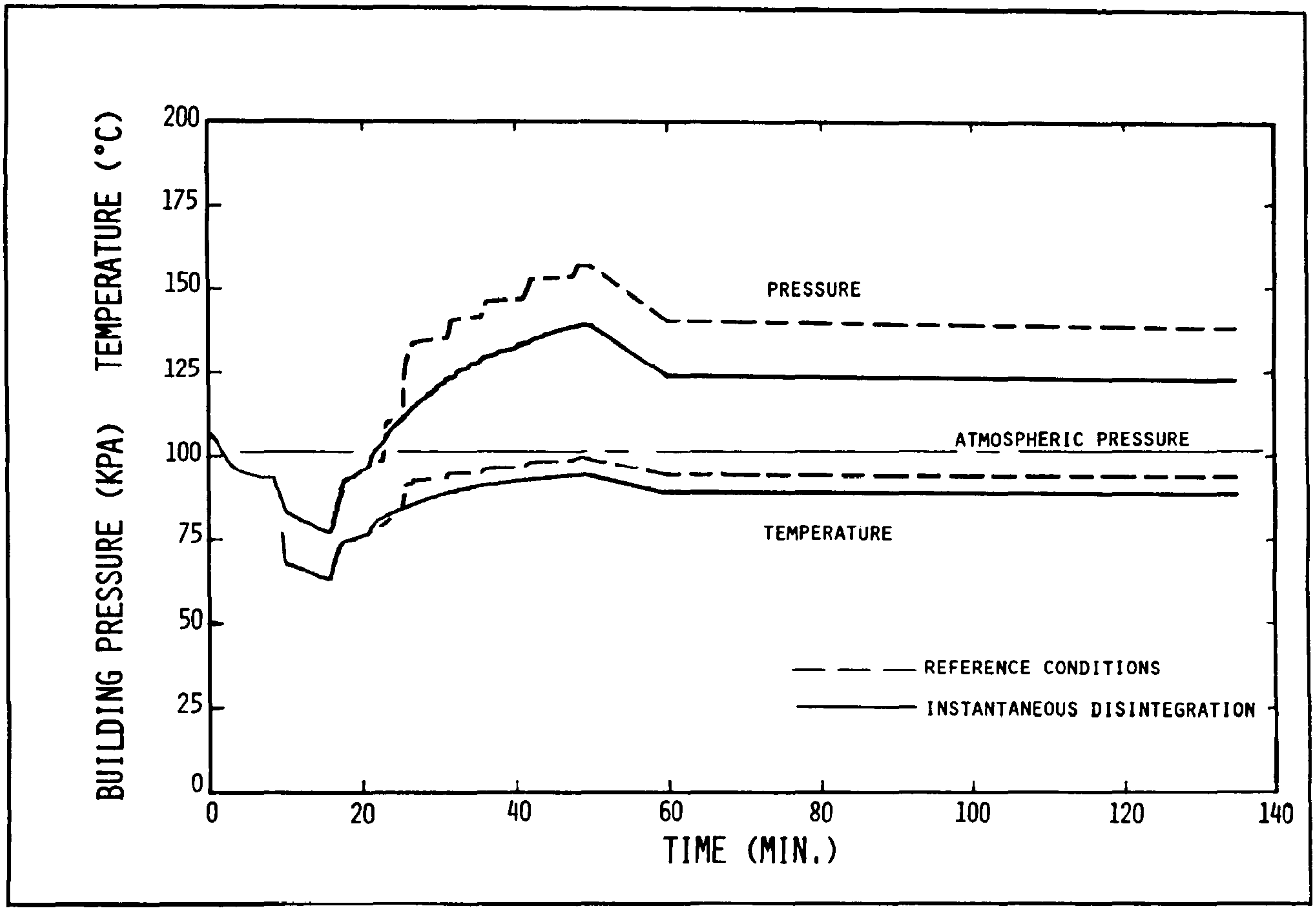

Figure 6. Containment pressure transient in the Bruce-A NGS in a LOCA plus failure of ECI plus failure of the moderator cooling system.

We may conclude that the inherent characteristics and designs of CANDU reactors would preclude accidents of the magnitude experienced at Chernobyl, and that these characteristics and designs, especially the shut-down systems and other special safety systems, would greatly mitigate the consequences of any accidents that might occur, as happened in the TMI-accident. The ability of a CANDU reactor to survive a serious accident with minimal damage and with no adverse health effects to the public was clearly demonstrated by the pressure-tube rupture accident to Pickering-2 in August 1983.

While these insights and conclusions about CANDU safety are heartening, we must recognize the significance of the human element in the Chernobyl accident, as in the TMI-2 accident. While the inherent characteristics of the RBMK resulted in a great disaster, the ultimate cause was a human error, coupled with a number of serious violations of procedures and common sense, as summarized in Table 5. While it would be more difficult, physically, to violate certain of these procedures (e.g., blocking of trip signals) in a CANDU station than it apparently was at Chernobyl, operators can still make mistakes under stress or under pressure, as was the case at Chernobyl. This accident emphasizes, once again, the need for the continuation of thorough operator training and the need for ensuring that nuclear utility employees and management must be very conscious of their responsibilities for public and worker safety at all times. Perhaps this is the most important lesson to be learned from the Chernobyl accident.

Table 5: Chernobyl Reactor Accident: Violations of Procedures before and during Planned Test

1. Inadequate attention to safety in written program for test.
2. Emergency coolant injection system blocked out.
3. Test conducted at 200 MW(t) instead of 700–1000 MW specified (Error in establishing control set-point).
4. Reactivity margin reduced below required level.
5. Total coolant flow rate through core higher than permitted.
6. Trip signals for low separator water level and low separator pressure blocked out.
7. Trip signal for closure of turbine stop valve blocked out.

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Notes

1. There is some uncertainty in the reactivity worths of the control and protective elements for the RBMK reactor as given here. The values given in Table 2 represent the author's interpretation of reference 1 after consulting with Canadians who attended the IAEA Experts' Meeting in Vienna, at which the information in reference 1 was presented.
2. Reference 1 states that the repeat test was to be performed in case the first test failed, not because two different voltage controls were to be tested. However, a Canadian delegate who remained after the official IAEA meeting was informed that the latter was the reason for ensuring two tests could be done. A repeat test required the reactor to continue to operate during the first test.
3. The positive void reactivity coefficient was about 50% greater than normal because of the particular core operating conditions.
4. It is unlikely that the accident consequences would have been mitigated to any significant extent had the ECI not been blocked. However, this is not the question that concerns us here.
5. The nominal melting point of UO_2 is about 2,800 degrees Celsius and that of Zircaloy is about 1,750 degrees Celsius.
6. For the estimated power excursion to 100 times full power in 4 seconds, assuming a linear power ramp, the average energy deposited in the fuel by this excursion was about 800 cal / gm. Note that the level of energy storage to cause fuel disintegration generally accepted in other countries, including Canada, is 200 cal / gm based on a conservative interpretation of TREAT and SPERT experiments [19].

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