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Consequences of a Single Feeder Break Accident in CANDU Power Reactors

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Abstract. An accident scenario in the Canadian Pressurized Heavy Water Reactor (CANDU) in which the flow in one channel is reduced due to an inlet feeder break is considered. Different break sizes are studied and radioactivity released to the atmosphere (^{131}I and Xe + Kr) in each case is estimated using AECL Computer Codes FIREBIRD, PRESCON, HOTSPOT and CURIES. (The channel with the maximum power of Pickering – B generating station was used in the analysis as an upper limit best estimate).

Six break sizes were used in the analysis of discharge rates. These were 6.45, 10.97, 12.26, 12.9, 14.2 and 25.8 cm^2. Only 10.97 and 12.9 cm^2 sizes were used for further analysis.

The activity released from breaks of smaller size than stagnation were found to cause more activity released because the discharge rate of steam from these breaks is less and therefore the building overpressure period is longer. That is before the dampers open to relief the pressure to the containment, the fuel would have spent longer time at high temperature, and thus more than the free activity of the channel is released to containment.

Both cases of intact (single failure) and impaired (dual failure) containment are analysed. The activity released to the atmosphere in the event of single failure was estimated to be below the permissible regulatory limit. For dual failure (feeder break plus containment impairment), it exceeds this limit. This indicates that unless established means other than the high reactor building pressure signal are employed to open the pressure relief panels and relief the reactor building pressure to containment, the radioactivity released to the atmosphere in the case of dual failure would be more than the permissible limit if such an accident should take place.

1. Introduction

1.1. General

The most important factor in the design of the Canadian Pressurized Heavy Water Reactor (CANDU) as in other nuclear reactors is the public safety, and safety measures have been concentrated upon to achieve this goal.

Although no serious accidents which have any significant environmental impact has ever occured to any of the CANDU reactors, yet scenarios for such accidents are considered since any accident could happen and result in the release of uncontrolled radioactivity to the atmosphere [1,2,3]. One of the most serious accidents which could take place is the Loss of Coolant Accident (LOCA) [4,5,6,7]. Some other accidents which are not as serious may include, Feeder Breaks, End-fitting Failure, Pressure Tube Rupture, etc. [8].

1.2. Scope

The scope of this work is to analyse an accident scenario in one of the CANDU power stations (the Pickering – B power station) in which the inlet feeder to one of the channels breaks down. Before discussing the details of the scenario, a brief discussion is given here about the several barriers in the CANDU system against the release of radioactivity to the environment should an accident occur. These barriers are [9-14]:

(I) Uranium oxide fuel

The fuel retains the bulk of all fission products (the retention factor is ~ 99%).

(ii) Fuel cladding (sheath)

The cladding retains the volatile fission products (iodine and noble gases which diffuse out of the oxide fuel). Retention during an accident depends on the amount of damage which occur to the clad. Should fuel cladding fail during a LOCA, iodine would be released from the fuel into the surrounding fluid, transported with this fluid to the break and then discharged into the reactor building. In the reactor building, iodine partitions between the gas and solution phases depending on the chemical forms present. Iodine behaviour therefore depends critically on the chemical conditions encountered and because iodine is highly reactive, several chemical forms could be present [11]. Previously it was assumed that large quantities of iodine would enter the gas phase. Experience from the Three Mile Island accident shows that, even though a large fraction of the iodine inventory was released from the fuel, airborne iodine concentrations were very small. This emphasizes the complex behaviour of iodine [11,15,16].

(iii) Primary Heat Transport System (PHTS)

In Pickering-B, the primary heat transport circuit is arranged into two figure-of-eights, each containing 190 fuel channels, two pumps, two steam generators, two inlet headers and two outlet headers (Fig. 1). The fuel channels in each *core pass* are individually connected by small diameter feeder pipes (50 to 80 mm I.D.) to the horizontal headers located above the core at each end. This arrangement minimizes the

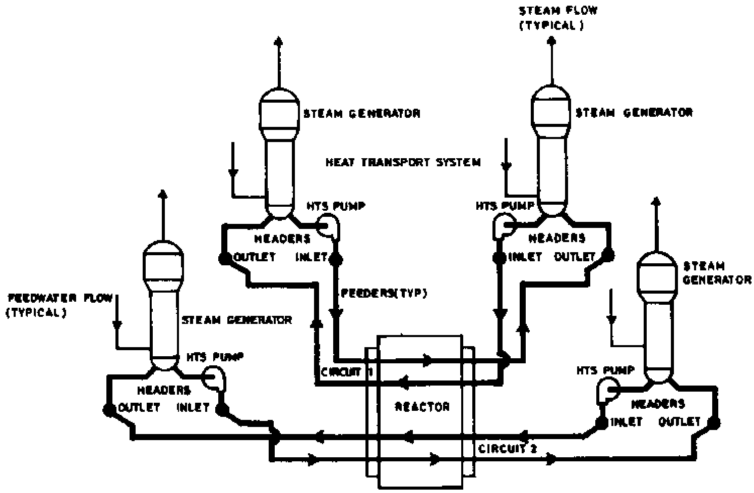

Fig. 1. Heat transport system flow-sheet for candu reactors

amount of large diameter piping: 98% of the piping length has a diameter less than 100 mm [11]. This means that the most likely LOCAs will be initiated by small breaks that are relatively easy to cope with.

If a damage should occur to the fuel cladding during an accident, fission products would be released to the PHTS which retains these fission products and the retention would be complete if the PHTS remains intact. The calandria which contains the fuel channels is filled with cool (about 70°C) heavy water moderator that is separated from the hot reactor coolant by a calandria tube around the pressure tube, but insulated from it by a gas-filled annulus (Fig. 2). The heavy water moderator is continuously cooled, providing a sink for decay heat produced in the fuel if there is a LOCA and a coincident failure of the emergency coolant injection (ECI) system. Because the fuel is separated from the moderator by only the relatively thin pressure and calandria tubes it cannot become very hot and remains insulated from the moderator.

(iv) Containment System

Containment is one of the physical barriers between the large source of radioactivity in the reactor and the public. As with other safety systems, its design is influenced by a hypothesized accident as well as by normal operation.

For a loss of coolant accident, the containment should:

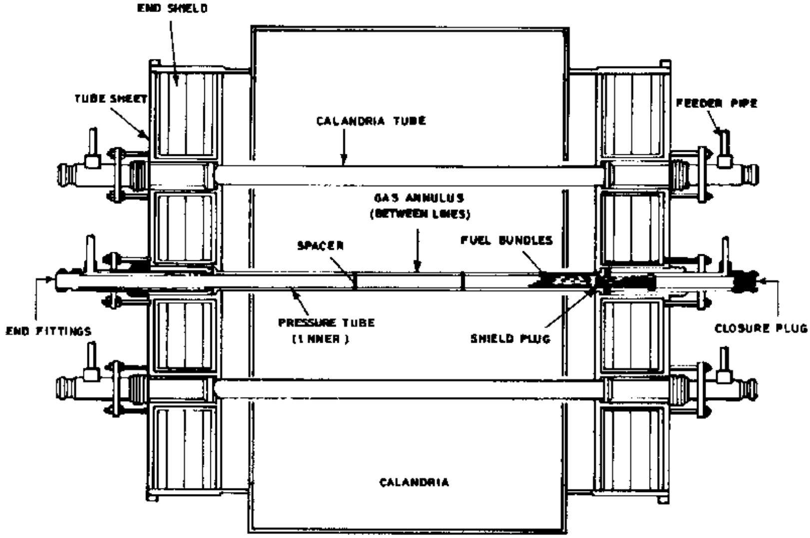

Fig. 2. CANDU reactor core schematic

1) withstand and/or suppress the overpressure caused by the flashing liquid.
2) be sufficiently leak-tight to meet regulatory release limits,
3) contain the activity release (if any) for the duration of over pressure.
4) produce a means of long term cooling and/or staying at subatmospheric pressure.

The Pickering B station has a multi-unit containment, which is a cylindrical domed concrete building (Fig. 3) and can withstand + 40 and −60 kPa(g) pressures. Each reactor vault is connected by a duct through banks of self-actuated valves to a common dousing system, housed inside a vacuum building (Fig. 4). The vacuum building design pressure is −100 to 0 kPa(g) and always stays sub-atmospheric (7-10 kPa) [12] in normal operation and is isolated from the reactor building by a number of large pressure relief valves and one-way louvers. The upper part of the building holds a large tank of dousing water. Upon overpressure the valves open and the steam rushes into the vacuum building. The differential pressure automatically actuates the dousing for subsequent depressurization. The design leakage rate of the containment is ~ 1% per 40 kPa(g)-hour [10,12].

Under normal operation conditions, the pressure within containment is slightly less than atmospheric and the containment ventillation dampers are all open. For very small heat transport system leaks, the building coolers in the containment con-

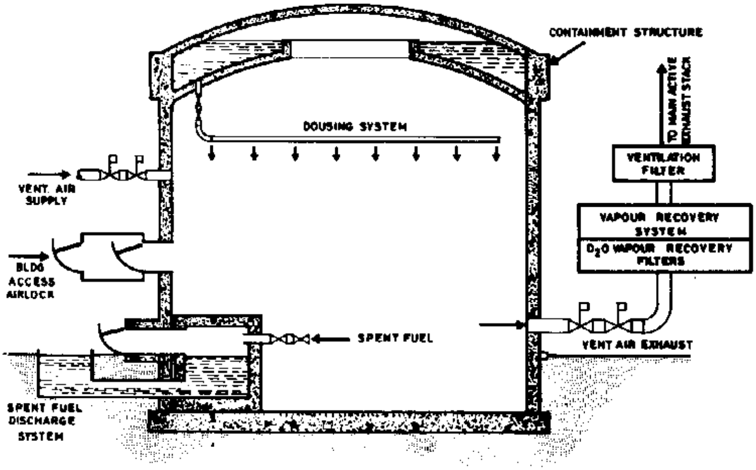

Fig. 3. Simplified diagram of containment envelop for CANDU reactors

dense any steam that is discharged and the building pressure remains atmospheric. For larger breaks, the building pressure rises and at an overpressure of about 3.4 kPa(g) (0.5 psig), containment pressure sensors initiate total containment closure. The containment pressure continues to rise and the dousing system starts to operate automatically at an overpressure of 13.8 kPa(g) (2 psig).

2. Accident Analysis

An accident scenario which involves only one channel of the Pickering-B station is analysed in this section. In this scenario (for conservatism) the inlet feeder to the maximum power channel is assumed to break causing coolant flow reduction to that channel. Different break sizes are considered and the activity released in each case is estimated. The break sizes range from small breaks (~ 20% of the feeder cross-sectional area) to a complete rupture of the feeder. A complete rupture of the feeder will reverse the flow in the affected channel very rapidly and maintain adequate fuel cooling. On the other hand, small breaks will cause a small flow reduction. This would not cause too much effect on the fuel cooling. Only break sizes in between need to be considered due to their expected effect since the flow will drop to a very low value and cause fuel and pressure tube overheating. Eventually, the pressure tube may be weakened by the high temperature and fail. Two cases are considered in this work;

a) the pressure tube fails soon after the accident, or

b) the pressure tube does not fail in the first few minutes after the accident.

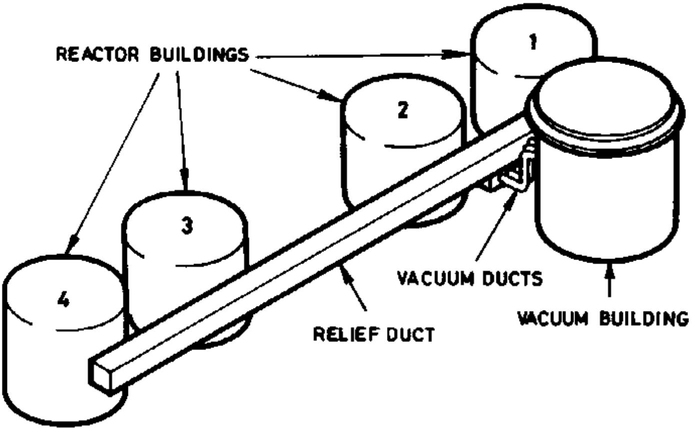

Fig. 4. Multi-unit containment for CANDU nuclear generating stations

2.1. Pressure Tube Fails Soon After the Accident

(i) Sequence of events

In this case, the calandria tube would most likely fail too. Therefore, there would be a coolant discharging inside the calandria. The coolant discharging into the moderator will gradually pressurize it. As the pressure inside the calandria reaches ~ 140 kPa (20 psig), the calandria rupture discs would burst and permit steaming into the building. As the reactor building pressure reaches 2 kPa (0.3 psi), a reactor trip signal will be initiated by high building pressure or low heat transport system (PHTS) flow. The blow-out panels separating the reactor area from the normally accessible (H_2O) areas would rupture when the differential pressure reaches 3.4 kPa (0.5 psi).

At this stage, the pressure of the entire reactor building rises at approximately 0.11 kPa/sec (0.016 psi/sec). As the building pressure reaches 14 kPa(g) (2 psig), the pressure relief panels open to allow flow into the pressure relief duct. As the duct pressure increases upto 4.5 kPa(g) (0.65 psig), the relief valves open and flow begins into the vacuum building.

(ii) Results

Since the pressure tube was assumed to fail soon after the accident, at most the free inventory of the channel is released to containment (since the fuel would have spent a very short time at high temperatures before the pressure tube ruptures). The rupture in all probability would increase the flow in the channel and re-cool the fuel, *i.e.* end the activity release. In this analysis, an inlet feeder break size of 10.97 cm^2

(1.7 inch^2) was chosen to illustrate the procedure since it was found to be the one which causes flow stagnation.

The AECL computer code FIREBIRD [17] was used to simulate the thermohydraulics of the Primary Heat Transport System. The mass discharge rate and enthalpy were thus obtained and then introduced into the containment response AECL code PRESCON [18] to find the integrated overpressure. Figures 5 and 6 give the mass discharge rate and enthalpy for this specific break. The pressure transients obtained from PRESCON are given in Fig. 7. The figure shows that the relief panels open at about 125 seconds after the accident, and that the reactor building goes subatmospheric at about 130 seconds. (Although the simulation period extended to more than 15 minutes which is the time required before the operator action takes place, yet the analysis is concerned only with the overpressure period since it is the only period in which activity will be released from the reactor building to the containment. Moreover, after the reactor trip, there will be decay heat, which is not accounted for since it does not lead to an overpressure of the reactor building). The integrated over-pressure of the reactor building was thus found to be 986 kPa.sec (143 psi-sec). The fractional release of activity is defined as the product of the integrated overpressure and the leakage rate. As mentioned previously [12], the target leak rate is given as 1% of contained volume per 40 kPa-hour whereas a leak rate as high as 2.7% would be acceptable. In this analysis, a 2% leakage rate is used, following Johns [8]. Consequently, the fractional activity release from containment is (Fig. 8);

$$\left(\frac{0.02\ \text{hour}^{-1}}{3600\ \text{sec/hr}} \times \frac{1}{40\ \text{kPa}}\right) \times 986\ \text{kPa-sec}$$
$$= 13.7 \times 10^{-5}$$

(iii) ***Activity releases outside containment***

As mentioned previously the free inventory of the channel is assumed to be released to the containment. Estimates of the fission product inventories of this channel as obained from AECL computer code CURIES [19] are given in Table 1.

Table 1. Maximum power channel inventories in CANDU reactor at pickering B station

Maximum channel power KW	Maximum bundle power KW	Element	Channel free activity (Ci)	Maximum bundle free activity (Ci)	Maximum channel total inventory (Ci)	Maximum bundle total inventory (Ci)
6088.4	744	^{131}I	12140	3380	164873	20947
		Xe+Kr	36724	12960	1526154	193900

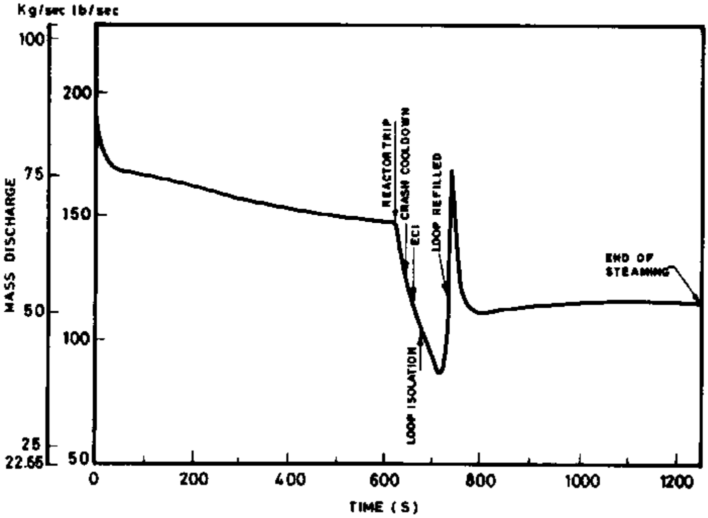

Fig. 5. Mass discharge rate for inlet feeder break Vs time

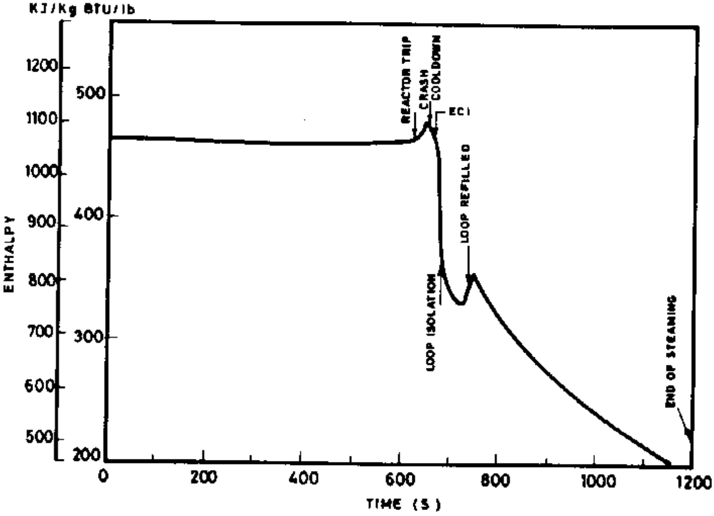

Fig. 6. Enthalpy for inlet feeder break Vs time

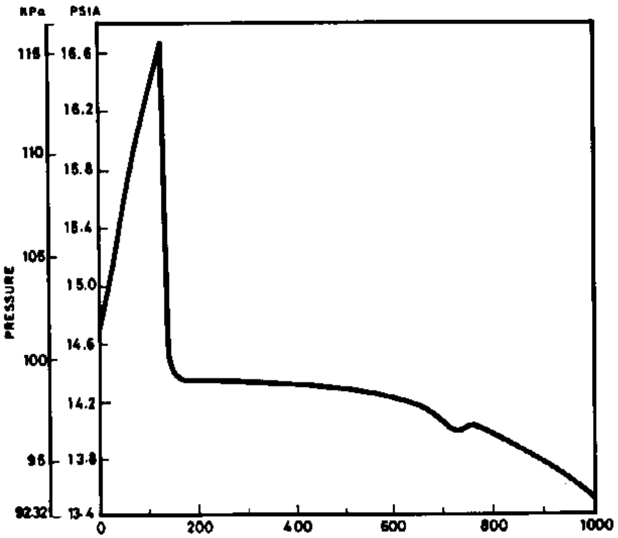

Fig. 7. Pressure transient in the reactor building for an intact containment

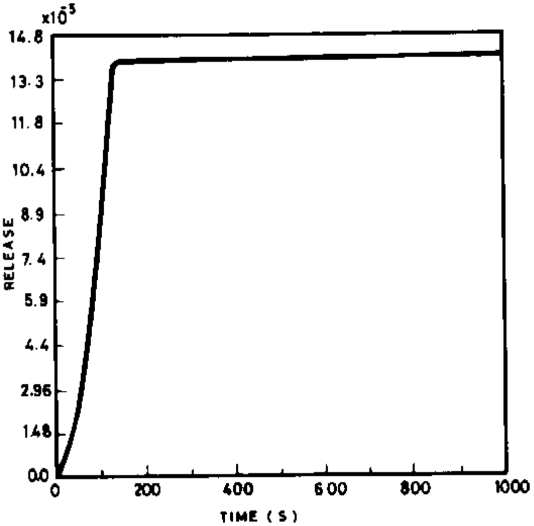

Fig. 8. Fractional activity release from intact containment

Using the value of the fractional activity released together with the free inventory of the maximum power channel, the total activity releases outside containment are estimated to be:

^{131}I : $12140\ \text{Ci} \times 13.7 \times 10^{-5} \simeq 1.7\ \text{Ci}$

Xe+Kr : $36724\ \text{Ci} \times 13.7 \times 10^{-5} \simeq 5\ \text{Ci}$

2.2. Pressure Tube Does Not Fail in the First Few Minutes

As in case a) above, the discharge from the break pressurizes the building and opens the pressure relief panels and the reactor building goes subatmospheric. If the pressure tube fails after that, there will be no contribution to the activity releases outside containment, since the building would not be overpressured.

A break size of 10.97 cm² (1.7 inch²) was again chosen to illustrate the method of calculation. Fuel centerline, fuel surface and sheath temperatures were calculated using the AECL computer code HOTSPOT [20] for this size and the results are given in Fig. 9. The figure shows that sheath fails in less than 20 seconds after the accident takes place (it reaches 1650°C at about 15 seconds). The transients from Fig. 9 were

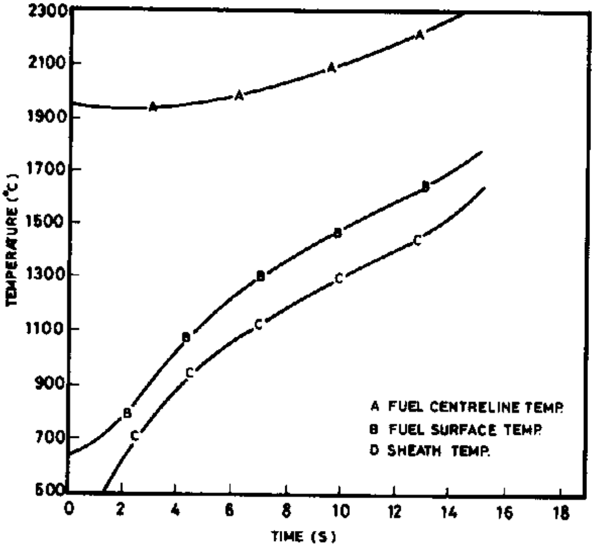

Fig. 9. Fuel centerline, fuel surface and sheath temperatures Vs time - feeder break area = 10.97 cm² (1.7 inch²)

used to determine the additional (more than the free) activity released, shown in Fig. 10. It may be seen that the additional activity released from the bundle in 15 seconds is about 110 Ci or 3.3% of the free inventory of the hottest bundle (Table 1). The figure also shows that the pressure tube will fail in less than 30 seconds (Experimental evidence shows that the pressure tube fails at a temperature lower than 1200°C depending on the mode of contact with the calandria tube. If the internal pressure is low, the pressure tube sags into contact (at about 1000°C) under the weight of the fuel; if the pressure is high it deforms radially (balloons) into contact (between 650°C and 1050°C depending on the internal pressure and the heat-up rate). Canadian tests in a lattice of pressurized tubes [21] showed that a single tube rupture will not rupture other pressure tubes either by pressure impulse or fragments. Indeed, failure of a pressure tube under normal operating conditions may not even rupture the surrounding calandria tube. If it does, the consequences are at most the collapse of other calandria tubes on to their pressure tubes, and possibly minor damage to the calan-

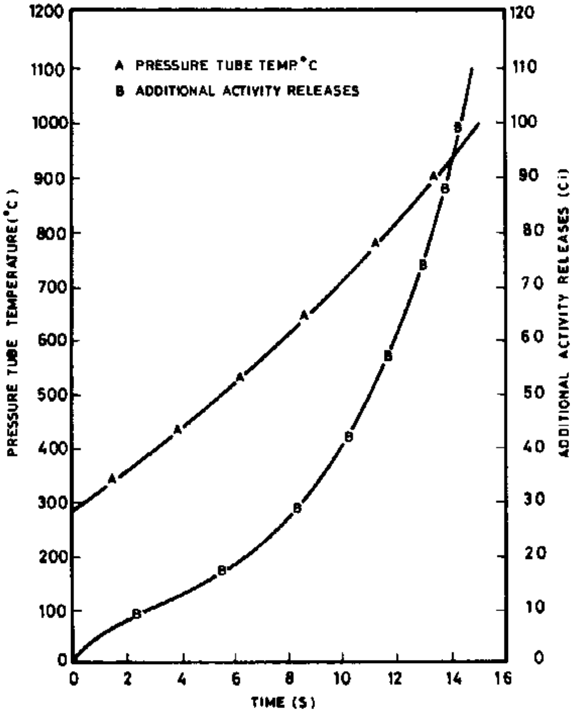

Fig. 10. Additional activity releases and pressure tube temperarture Vs time feeder break areas of 10.97 cm^2)

dria from fuel fragments [11,12]. However, assuming 30 seconds to be the time until the pressure tube ruptures, the additional activity released from the whole channel would be ~ 2400 Ci or as much as 19% of the free inventory for the whole channel. Therefore an upper bound (conservative assumption) of twice the free inventory was used.

Smaller breaks than stagnation would lead to longer overpressure transients in the reactor building since the relief panels would take longer time to open. Therefore smaller breaks than stagnation can result in greater fractional activity release from containment. On the other hand for smaller than stagnation breaks, coolant will flow in the forward direction causing much of the activity released to be diluted in the Heat Transport System.

For larger break sizes than the 10.97 cm^2 (stagnation break), the temperature rise is slower and it takes longer time for the pressure tube to fail. Therefore the additional activity released may be more than that for the 10.97 cm^2 break (as may be seen from Fig. 11) by about 6% of the channel free inventory.

On the other hand, for larger breaks, the increased steaming rates into containment (it is higher for the 10.97 cm^2 than for the 12.9 cm^2 by ~ 9.5%) will offset the

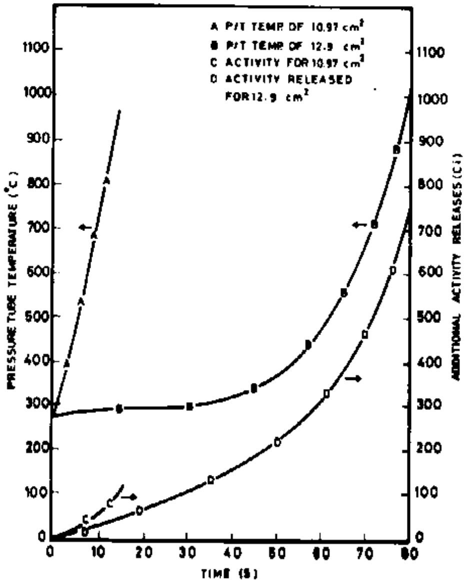

Fig. 11. Additional activity releases and pressure tube temperature Vs time for feeder break areas of 10.97 and 12.9 cm^2

increased activity release. To determine the break size which leads to the highest activity releases to containment, simulations were carried out for break sizes between 6.45 cm^2 (1.0 $inch^2$) and 25.8 cm^2 (4.0 $inch^2$). Results of the analysis are given in Figs. 12 and 13.

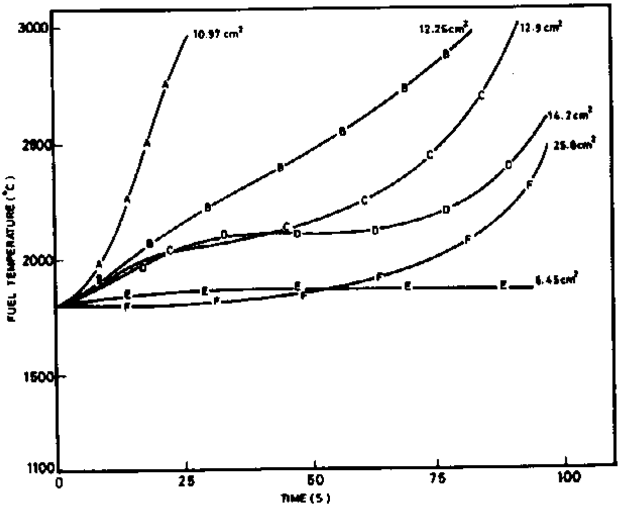

Fig. 12. Center region fuel temperature for various break sizes Vs time

Therefore, the upper bound for the activity released to the containment in this case would still be twice the free inventory of the channel. Consequently, the activity released outside containment is estimated to be

^{131}I : 2×12140 Ci $\times 13.7 \times 10^{-5} \simeq 3.3$ Ci

Xe+Kr : 2×36724 Ci $\times 13.7 \times 10^{-5} \simeq 10.1$ Ci

(where 13.7×10^{-5} is the fractional activity releases calculated in (a) above).

3. Activity Releases from an Impaired Containment (Dual Failure)

The term dual failure refers to the coincident failure of a process system and a safety system (in this case an impaired containment, *i.e.* the inlet dampers of the reactor building which are used for vent air supply fail to close).

Details of the accident are as above. Following the same arguments, one of the following two situations may take place.

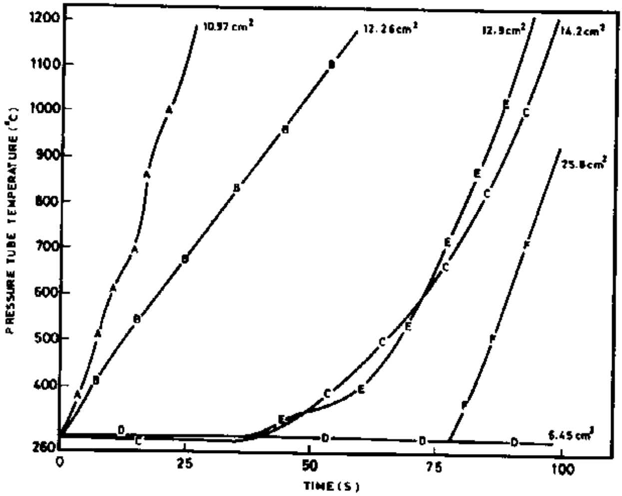

Fig. 13. Pressure tube temperature for various break sizes Vs time

3.1. The Pressure Tube is Postulated to Fail Soon after the Accident

The sequence of events is the same as discussed before. Only the free inventory of the channel is assumed to be released to containment since the fuel would have spent a short time at high temperatures before the pressure tube ruptures.

The overpressure period for this case was found to be very long (~ 840 seconds) as may be seen from Fig. 14. Consequently, the fractional activity releases in this accident are very high ($\sim 29 \times 10^{-2}$) as indicated in Fig. 15. In other words about 30% of the activity released to containment is released outside. Since only the free inventory of the channel is assumed to be released to containment, the total activity releases outside containment are estimated to be

^{131}I : $12140 \times 29 \times 10^{-2} \simeq 3520$ Ci

Xe+Kr : $36724 \times 29 \times 10^{-2} \simeq 10650$ Ci

3.2. The Pressure Tube Does Not Fail in the First Few Minutes

In this case, as in (I) above, twice the free inventory of the channel is assumed to be released to containment. Consequently, the total activity releases outside containment are estimated to be

^{131}I : $12140 \times 2 \times 29 \times 10^{-2} \simeq 7040$ Ci

Xe+Kr : $36724 \times 2 \times 29 \times 10^{-2} \simeq 21300$ Ci

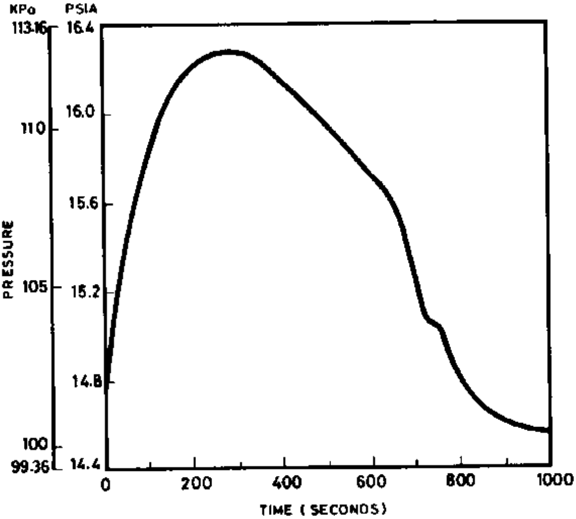

Fig. 14. Pressure transient in the reactor building (impaired containment).

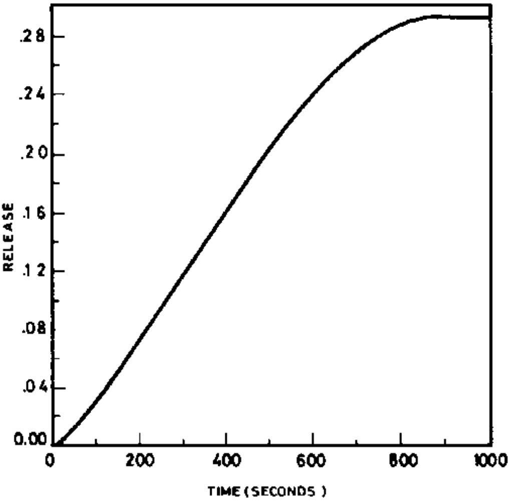

Fig. 15. Fractional activity release from containment (impaired containment)

4. Conclusions and Recommendations

The following points may be deduced from the analysis:

4.1. For Intact Containment

a) The activity released to the atmosphere if the pressure tube fails soon after the accident is very low compared to the permissible limits set by the International Commission on Radiological Protection (ICRP) and The International Atomic Energy Agency (IAEA) for such accidents. (The legal standards of radiation protection adopted in Canada are based on the recommendations of the ICRP. The ICRP emphasis is to keep all doses As Low As Readily Achievable (the ALARA principle) and the Atomic Energy Control Board (AECB) which performs the regulatory function ensures that this regulation is met). The individual limits for single failure based on these recommendations are [22 - 30]:
(These limits are adopted by AECL):

$$^{131}\mathrm{I} = 11\ \mathrm{Ci}$$
$$\mathrm{Xe+Kr} = 2.1 \times 10^{4}\ \mathrm{Ci}$$

The equivalent dose limits are; 0.5 rem for whole body and 3 rem for thyroid.

b) The activity released to the atmosphere if the pressure tube does not fail in the first few minutes after the accident is still less than the permissible limit set for single failure accidents but is twice the value for the case when the pressure tube fails soon after the accident.

4.2. For Impaired Containment

For either case of pressure tube behaviour, the activity released (from ^{131}I) to the atmosphere is much more than the permissible limit set for such case. (The individual limits set for dual failure are [22 - 30]:

$$^{131}\mathrm{I} = 930\ \mathrm{Ci}$$
$$\mathrm{Xe+Kr} = 1.1 \times 10^{6}\ \mathrm{Ci}$$

The equivalent dose limits are; 25 rem for whole body and 250 rem for thyroid).

It is therefore recommended that CANDU Reactors should have some means by which the pressure relief panels open sooner to relief the pressure and avoid the long overpressure period, therefore reducing the activity releases. This may be achieved through:

1- Lowering the pressure setting values at which the pressure relief panels open,
2- Installing by-pass valves which open at lower pressure to relief the pressure of the reactor building,
3- Developing the pressure relief panels in such a way that they open on an activity signal (along with a pressure signal - whichever comes first), or
4- A combination of some of the above methods.

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دراسة النتائج المترتبة على حدوث كسر مفاجىء في خط تغذية سائل التبريد في المفاعلات الكندية

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ملخص البحث: في هذا البحث يتم افتراض تناقص كمية الماء اللازمة لتبريد وحدة وقود من وحدات المفاعل النووي الكندي نتيجة كسر مفاجىء في خط تغذية المياه لتلك الوحدة. ولقد تم افتراض قيم مختلفة لنسبة الكسر في هذا الخط وتم حساب نسبة الإشعاعات التي تنتج في كل حالة وقيمة ما يصل منها للجو الخارجي وذلك باستخدام بعض برامج الكمبيوتر التي تم تصميمها وتطويرها في مؤسسة الطاقة الذرية الكندية.

واستخدمت في الحسابات وحدة الوقود ذات أكبر طاقة وذلك لحساب أكبر قيمة ممكنة من الإشعاعات (من باب الأمان). ولقد وجد أن الإشعاعات الناتجة في حالة الكسور أقل من القيمة التي تسبب ركود سائل التبريد تزيد عن مثيلتها في تلك الحالة لأن معدل البخار الناتج يكون أقل وبالتالي فإن فترة تزايد الضغط داخل المفاعل تكون أطول ـ أي أنه قبل أن تفتح منافذ التهوية لتقليل قيمة الضغط فإن الوقود يكون قد استمر فترة أطول عند درجة حرارة عالية وبالتالي فإن كمية الإشعاعات الناتجة تزيد عن الكمية الأساسية والتي تنبعث عند درجات الحرارة المنخفضة. ولقد تم استخدام ٦ قيم لمساحة الكسر في خط التغذية وهي ٦ر٤٥، ١٠ر٩٧، ١٢ر٢٦، ١٢ر٩، ١٤ر٢، ٢٥ر٨سم^{٢} على التوالي في حسابات كمية البخار الناتجة من الكسر إلا إنه لم يتم إلا استخدام ١٠ر٩٧، ١٢ر٩سم^{٢} في حسابات كمية الإشعاعات الناتجة حيث إن هاتين القيمتين تعطيان أكبر كمية إشعاعات.

وتمت دراسة حالتين: الحالة الأولى هي انكسار خط التغذية فقط مع بقاء الحاجز الوقائي سليما، والحالة الثانية هي حدوث أعطال للحاجز الوقائي بالإضافة إلى انكسار خط التغذية. ولقد وجد أن الإشعاعات المتسربة للجو الخارجي في الحالتين أقل من القيم المسموح بها في كل حالة على حدة ـ كما وجد أن الإشعاعات في حالة أعطال للحاجز أكبر بكثير من الحالة الأولى وهذا هو الوضع المتوقع.