J. King Saud Univ., Vol. 2, Eng. Sci. (1), pp. 115-129, (1410H/1990)

MECHANICAL ENGINEERING

Radioactive Releases from Candu Power Reactors During An End-Fitting Failure Postulated Accident

Fahmy M. Hussein
Mechanical Engineering Department, College of Engineering, King Saud University, P.O. Box 800, Riyadh 11421, Saudi Arabia

Abstract. An accident scenario in which the end-fitting of one channel of the Pickering-B, Canadian Pressurized Heavy Water Reactor (CANDU) is assumed to fail completely is considered. All of the 12 bundles of the channel are assumed to be discharged to the fuelling-machine vault and onto the floor.

Three cases are analysed; the case of intact bundle, bundle broken into separate intact fuel pins and broken bundle with broken pins.

Release of ^{131}I and Xe+Kr to the atmosphere are estimated for both cases of intact and impaired containment using AECL computer codes HOTSPOT, CURIES, FIREBIRD and PRESCON. For conservatism, the channel with the maximum power was used in the analysis.

The activity released to the environment in both events of intact containment (single failure) and impaired containment (dual failure) was estimated to be below the permissible regulatory limits. However, the activity released in the dual failure case is much higher than that for the single failure.

1. Introduction

1.1. General

The design of the Canadian Pressurized Heavy Water Reactor (CANDU) is based in the first place on safety not only during normal operation but most importantly in the event of an accident which might take place due to failure of any part of system of the reactor components. Accident scenarios for such cases are analysed and activity released to the environment during the course of any of these events are estimated and safety measures are taken to minimize the consequences of these accidents [1, pp. 95–138; 2, pp. 72–80; 3–8]. The use of several barriers to radioactivity release and the consideration of more than one level of defense against any barrier being breached are some of these safety measures. Detailed discussion of these barriers is given in [9].

1.2. Scope

The purpose of this research is to analyse an accident in which the end-fitting of one channel at Pickering-B CANDU Power Station fails, and to compare the activity released with the regulatory limits set by the International Commission on Radiological Protection (ICRP) and the International Atomic Energy Agency (IAEA) for such cases. The ICRP emphasis is to keep all doses As Low As Readily Achievable (the ALARA Principle) and the Atomic Energy Control Board (AECB) ensures that these regulations (which are adopted in Canada) are met.

The regulatory limits are set for two cases, single and dual failures. These are [10-18]:

1.2.1. Single failure

^{131}I	11 Ci
Xe + Kr	2.1×10^4 Ci

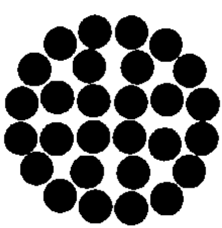
(The equivalent dose limits are 0.5 rem for whole body and 3 rem for thyroid).

1.2.2. Dual failure

^{131}I	930 Ci
Xe + Kr	1.1×10^6 Ci

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

The fuel channel at Pickering-B accomodates 12 fuel bundles (50 cm in length and 10 cm in diameter). The fuel bundle consists of 28 fuel elements (1.5 cm outside diameter with a cladding thickness of about 0.4 mm) which are assembled to form the bundle through the use of spacers and end plates to hold the fuel elements. 29 pellets stacked end-to-end, are sealed in a Zirconium alloy (ZIRC-4) sheath to form each fuel element. The fuel bundle cross section is as follows:



More details about the fuel structure at Pickering-B are given in reference [19].

2. Accident Analysis

An accident scenario which involves only one channel of the Pickering-B station is analysed. In this scenario (for conservatism), the end fitting of the maximum power channel is assumed to fail completely and to fully open the channel. (The end fittings are those portions of the fuel channel assemblies external to the calandria and have connections to the feeders which feed coolant into and out of the fuel channels) [9].

Two cases are analysed; firstly the case of an intact containment (*i.e.* single failure), and secondly the case of end fitting failure coincident with an impaired containment (*i.e.* dual failure).

2.1. Case of Intact Containment

As the channel is broken, the discharging coolant might drive all the 12 bundles of the channel into the fuelling-machine vault and onto the floor. One of the following three situations could take place: Upon impact of the bundles with the floor, or the vault walls, the bundle breaks into separate fuel pins; bundle and fuel pins break, and the bundle is discharged but remains intact.

2.1.1. Bundle breaks into separate fuel pins

As the channel is broken, both the fuel and sheath temperatures are expected to rise rapidly. However, there will be a coolant discharge which will likely provide some cooling to the bundle. (The channel is cooled in the long term from the Emergency Core Cooling (ECC)). The AECL computer code HOTSPOT [20] was used to predict the fuel and sheath temperature transients. The bundle with the maximum power (744 KW) with corresponding fuel element rating of 48.52 W/cm was used in the analysis. The results showed that the sheath temperature rises to about 1035° (Fig. 1) high enough that it may cause activity to be released from the sheath (if the sheaths were not already damaged by impact with the floor – the sheath defects if its temperature exceeds 800°C – a number of defects occurs at temperatures between 600 and 800°C. As the temperature increases, the number of defects increases with the irradiation playing an important role in this process – [19]. However, both fuel and sheath temperatures fall so that no extra diffusion of fission products occurs (the release of the activity retained on the sheath depends on the temperature history following ejection from the channel. Diffusion will occcur only as the sheath temperature exceeds 800°C [19]). Thus only the free inventory is estimated to be released. This was confirmed by using the HOTSPOT temperature transients utilizing the AECL computer code CURIES [21], which was developed to predict the inventory buildup and activity releases from CANDU fuel. The results of CURIES indicated that the additional activity released (more than the free inventory) is trivial compared to the free inventory of the channel.

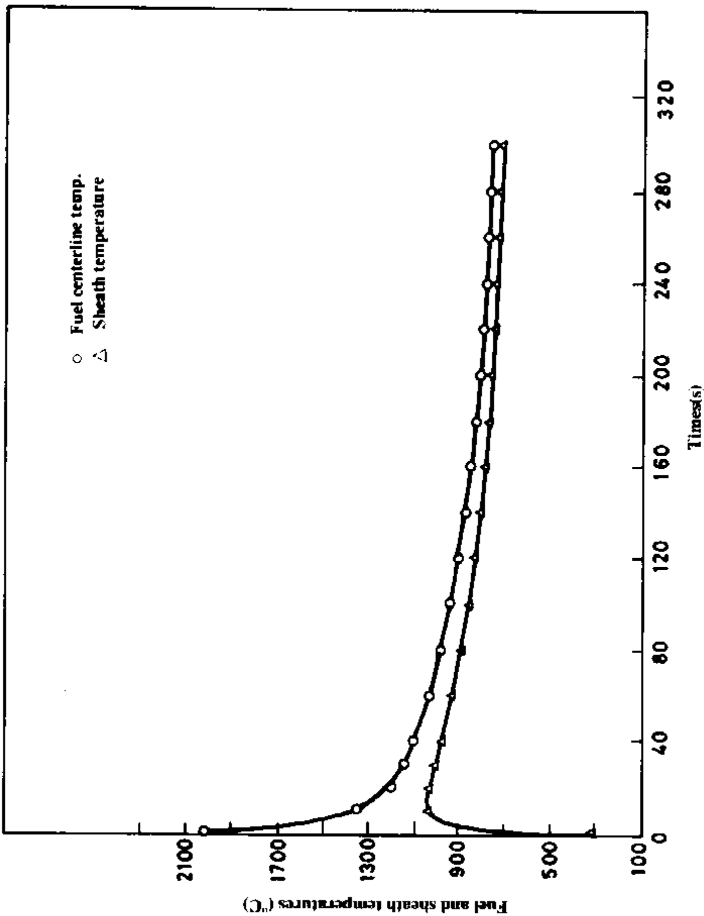


Fig. 1. Fuel centreline and sheath temperatures for hot pin VS time during and end fitting failure accident in a CANDU reactor

Containment response

To find the activity released outside the containment, it is realized that activity releases from the reactor building occur via leakage driven by the overpressure. A leakage rate of activity of 2% per 40 kPa-hour (6 Psi-hr, design pressure) was used in the analysis. 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 [8]).

The fractional release of activity is defined as the product of the leakage rate and the integrated overpressure. It was necessary to find the integrated overpressure first. The mass discharge rates and enthalpy from the AECL computer code FIREBIRD [22] (which is developed to simulate the thermohydraulics of the primary heat transport system), were fed into PRESCON [23] (an AECL computer code used to find the containment response). The mass discharge rates and enthalpy graphs are given in Figures (2 and 3).

The sequence of events is as follows:

i) The blow-out panels separating the fuelling machine (F/M) vault from the reactor area rupture when the flashing discharge pressurizes the F/M vault.

ii) The reactor will trip when the pressure in the reactor building reaches 2 kPa (0.3 Psi). (This is an alternate trip parameter. However, the primary trip which is the high neutron power does not trip the reactor since there is not enough neutron power increase to induce that trip).

iii) The blow-out panels separating the reactor building from the normally accessible areas rupture when the differential pressure reaches 3.4 kPa (0.5 Psi).

iv) At this stage, the pressure of the entire reactor building rises at approximately 0.26 kPa/s (0.038 Psi/s).

v) At about 53 seconds after the accident, the building overpressure reaches 14 kPa (2 Psi) and the pressure relief panels open to allow flow into the pressure relief duct.

vi) As the duct overpressure increases to 4.5 kPa (0.65 Psi), the relief valves open and flow begins into the vacuum building. Results of PRESCON showed that for this scenario the relief panels open at about 60 seconds after the accident, and that the reactor building goes subatmospheric at about 69 seconds. The simulation period extended to more than 15 minutes which is the time before the operator action. However, 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 (the activity is driven out by the overpressure). The integrated overpressure of the reactor building was thus found to be 442 kPa-s (64 Psi-s). Figure 4 illustrates these results.

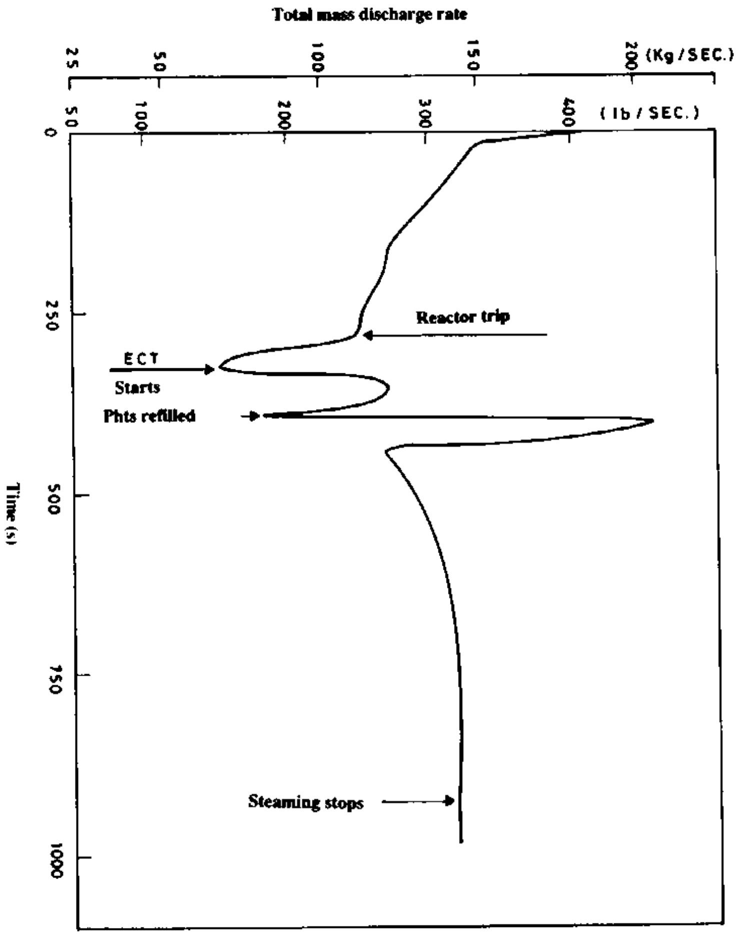


Fig. 2. Total mass discharge rate VS time

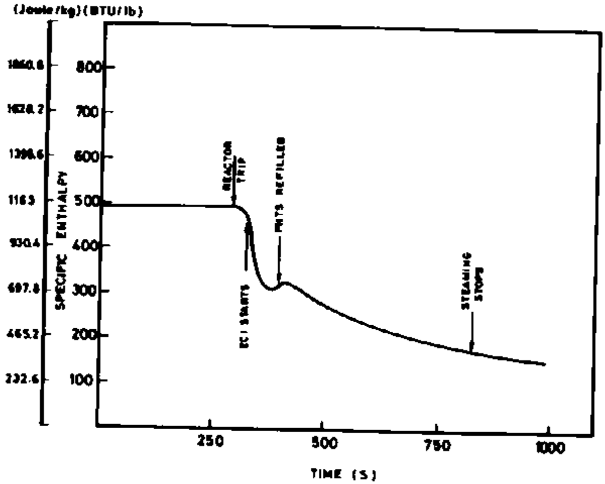


Fig. 3. Specific enthalpy VS time

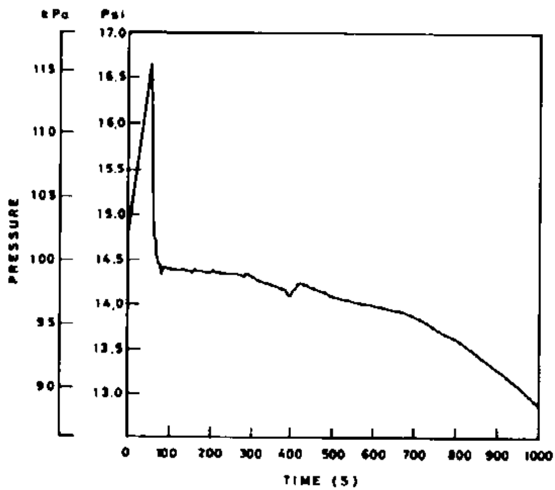


Fig. 4. Pressure transient in the reactor building

The fractional release of activity from containment is found as the product of the leakage rate (2%) and the above value of the integrated overpressure. Thus, the fractional activity release is

$$\left(\frac{0.02\,\text{hour}^{-1}}{3600\,\text{sec/hr}} \times \frac{1}{40\,\text{kPa}} \right) \times 442\,\text{kPa-sec}$$

$$\simeq 6.1 \times 10^{-5}$$

Activity releases outside containment

The entire ^{131}I free inventory of the channel is assumed to be released at once.

From the free inventory estimates of the maximum power channel [9] and the above fractional release of activity, the total activity released outside containment from both ^{131}I and (Xe+Kr) is

^{131}I $\simeq$ $12140\,\text{Ci} \times 6.1 \times 10^{-5} \simeq 0.74\,\text{Ci}$

Xe+Kr $\simeq$ $36724\,\text{Ci} \times 6.1 \times 10^{-5} \simeq 2.2\,\text{Ci}$

2.1.2. Both bundle and fuel pins break into small pieces

This is similar to case (2.1.1) above, except with less acitivity released [as the fuel pins break, the pieces will be cooled faster (by the coolant discharging from the channel) due to the increase of the heat transfer area]. Therefore, the sheath temperature exceeds 800°C for a shorter period than that of case (2.1.1). (800°C is the temperature at which the activity retained on the sheath starts to be released [19]). Thus case (2.1.1) is an upper bound of this case.

2.1.3. Bundle remains intact

In this case the temperatures are considerably higher than the broken bundle with scattered pins (compare Fig. 5 with Fig. 1 for the broken bundle), since radiative heat transfer from inner pins is reduced by the presence of the outer pins. Moreover, there may be a metal – Zircaloy – water reaction since sheaths are becoming hot enough to induce that effect.

HOTSPOT results, however, showed that sheath melting (if any) does not occur until after ~ 100 seconds after ejection from the channel. Also, the overpressure period in this case is only 64 seconds. Therefore, there is insufficient time for the fuel bundles to release more activity beyond their free inventories before the reactor goes subatmospheric. CURIES was modelled for the transient case in order to find the additional activity releases. In the modelling of CURIES, the fuel element (Pin) is radially divided into 3 rings for temperature homogenity, the center, the inter-

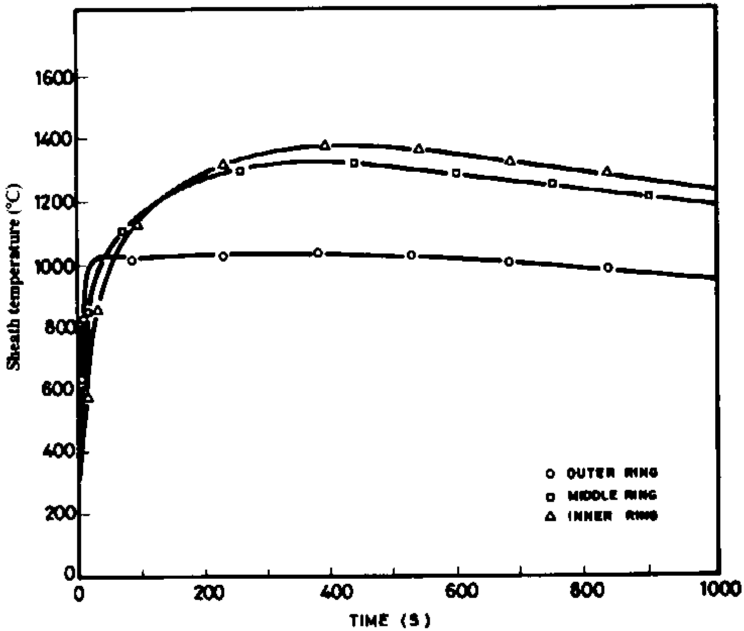


Fig. 5. Sheath temperature for different fuel element rings VS time (intact bundle)

mediate and the outer. The normalized pin powers for Pickering-B- *i.e.* 28 fuel element (and used in this analysis) are: Center 0.791, Intermediate 0.903 and Outer 1.101. The fuel centerline and fuel surface temperatures for a given rating are given as inputs to CURIES. The code then calculates the fuel temperatures of other pins (having different ratings) by assuming a linear variation. The results are summarized in Figure 6. It may be noticed that the additional activity released is zero up to about 60 seconds and then increases to 1.2 Ci for the inner ring and is almost at a constant value of 0.4 Ci for the outer ring over the whole transient period of 220 seconds (the simulation period).

The total additional activity released (beyond the free inventory) was thus found to be 0.042% of the free activity. Consequently only the free inventory of the channel is assumed to be released to the containment. Releases outside containment are thus the same as calculated in case (2.1.1) above, *i.e.*

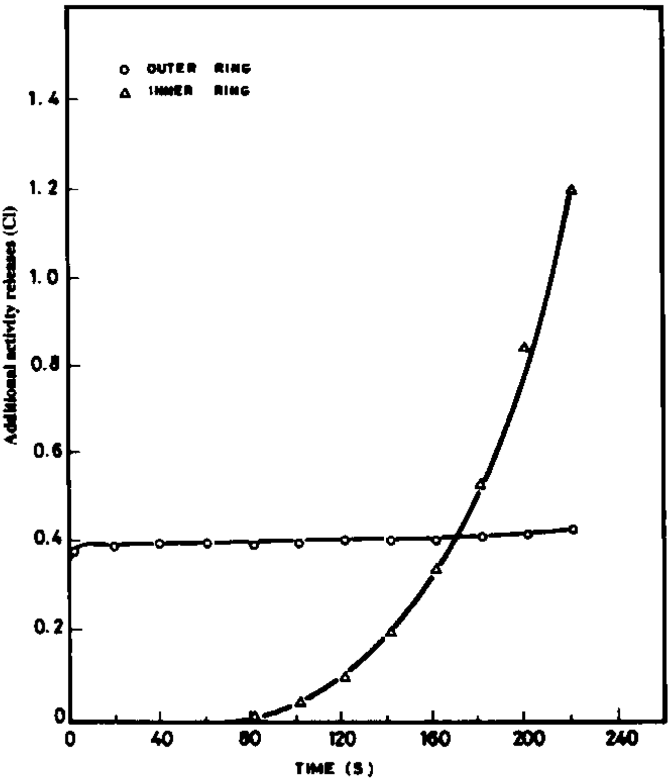


Fig. 6. Additional activity releases for an intact bundle VS time

$$^{131}I \sim 0.74\ Ci$$

$$Xe + Kr \simeq 2.2\ Ci$$

2.2. Activity Releases From Impaired Containment

The term impaired in this analysis refers to the case in which the inlet dampers of the reactor building (which are used for vent air supply) fail to close. The fractional release from containment would be higher than the intact containment case. The analysis proceed in the same manner as section (2.1). Since the reactor building is usually held at –0.25 kPa [8] and for small breaks, the overpressure period ends as soon as the pressure relief panels open to allow flow into the pressure relief duct (they are designed to open at ~ 14 kPa (~ 2 Psig) [8]), hence the maximum overpressure possible is the relief pressure of the panels (14 kPa). The pressure relief panels were thus found to open at 68 seconds after the accident and the overpressure period was ~ 73 seconds (Fig. 7). Consequently the fractional release from containment is estimated as ~ 3×10^{-2} (Fig. 8).

Based on the above values, the total activity releases outside containment were estimated to be:

$$^{131}I \simeq 12140 \text{ Ci} \times 3 \times 10^{-2} \simeq 364 \text{ Ci}$$

$$Xe + Kr \simeq 36724 \text{ Ci} \times 3 \times 10^{-2} \simeq 1100 \text{ Ci}$$

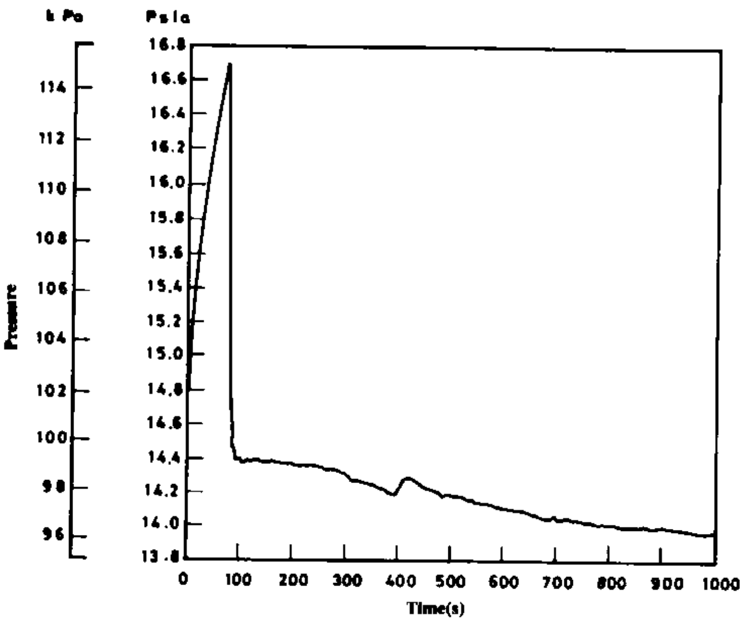


Fig. 7. Pressure transient in the reactor building (impaired containment)

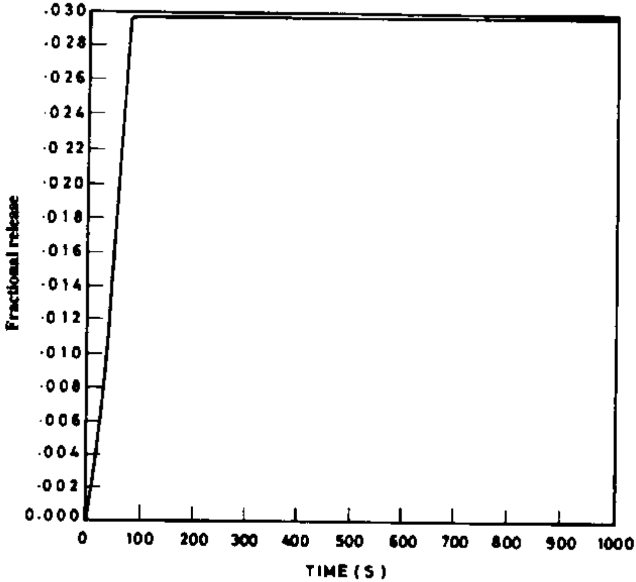


Fig. 8. Fractional acitvity release from impaired containment

3. Conclusion

The above analysis indicate that the activity released from an intact containment during a hypothesized accident of an end fitting failure of the Pickering-B CANDU reactor system is small. The values are lower than the permissible limits set by the International Commission on Radiological Protection (ICRP) and the International Atomic Energy Agency (IAEA) and implemented by the Atomic Energy Control Board of Canada (AECB) for such accidents. For a dual failure, the values are much higher than the single failure (which is expected), yet they are still within the guidelines set by the ICRP and the IAEA.

Acknowledgement: The author carried out part of this research while he was associated with the safety branch of AECL. The cooperation of the safety branch group as well as that of other branches is here acknowledged. Special thanks are due to Dr. V.G. Snell for his valuable recommendations and suggestions and for bringing the results of this work up for discussion with the concerned gorups, especially Ontario Hydro and AECB.

References

[1]Beattie, J.R.; Griffiths, R.F.; Kaiser, G.D. and Kinchin, G.H. *The Environmental Impact of Radioactive Releases from Accidents in Nuclear Power Reactors.* Oxford: Pergamon Press, 1980.

[2] Daw, H.T. *Setting Release Limits of Radioactive Materials Into the Environment.* Radiological Safety Section, Vienna: IAEA, 1984.

[3] Hastings, I.J.; Hunt, C.E.L. and Lipsett, J.J. "Release of Short-lived Fission Products from UO_2 Fuel: Effects of Operating Conditions." *Journal of Nuclear Materials* (North Holland, Amsterdam), No. 130 (1985), 407-417.

[4] Robertson, J.A.L. "The CANDU Reactor System: An Appropriate Technology." *Science,* 199, (1978), 657-664.

[5] Hancox, W.T. "Safety Researh for CANDU Reactors." A paper presented to the *International Atomic Energy Technical Committee Meeting on Thermal Reactor Safety Research in Moscow,* December 1-4 (1981). (Also, an AECL Report # 6835, 1982).

[6] McConnell, L.G. *et al.* "CANDU Operation Experience." Presented at the *IAEA International Conference on Nuclear Power Experience,* Vienna 13-17, September (1982). (Also, an AECL Report # 7762, 1982).

[7] Fee, E.W. and Show, G.E. "Vacuum Containment Systems for Multi-Unit Nuclear Power Stations." Societe Francaise de la Radioprotection, VII^{me} *congress International sur le confinement de la Radioactivite dans l'utilization de l'Energie Nucleaire,* Versailles, Mai (1974).

[8] Pease, L. and Sawai, S. "Heavy Water Moderated Pressure Tube Reactor Safety." Presented at *The International Conference on World Nuclear Power,* American Nuclear Society Washington, D.C., November (1976). (Also, AECL Report # 5856, 1977).

[9] Hussein, F.M. "Consequences of a Single Feeder Break Accident in CANDU Power Reactors." *Journal of King Saud Univ.,* vol.1, *Engineering Science* (1,2) (1989).

[10] International Commission on Radiological Protection (ICRP) *Report of Committee II on Permissible Dose for Internal Radiation.* ICRP Publication 2, Oxford: Pergamon Press, 1959.

[11] International Commission on Raidological Protection (ICRP) *Report of Committee IV on Implications of Commission Recommendations that Doses be Kept as Low as Readily Achievable.* ICRP Publication 22, Oxford: Pergamon Press, 1973.

[12] International Atomic Energy Agency (IAEA) *Basic Safety Standards for Radiation Protection.* 1982 Edition, Safety Series # 9, Vienna: IAEA, 1982.

[13] International Commission on Radiological Protection. *Radionuclide Release into the Environment: Assessment of Doses to Man.* ICRP Publication # 29, Oxford and New York: Pergamon Press, 1979.

[14] International Commission on Radiological Protection (ICRP). *Limits for Intake of Radionuclides by Workers.* ICRP Publication # 30, Oxford and New York: Pergamon Press, 1982.

[15] International Atomic Energy Agency (IAEA) *Principles for Limiting Releases of Radioactive, Effluents into the Environment.* Safety Series # 77, Vienna: IAEA, 1986.

[16] International Atomic Energy Agency (IAEA). *Principles for Establishing Limits for the Release of Radioactive Materials into the Environment.* Safety Series # 45, Vienna: IAEA, 1978.

[17] International Commission on Radiological Protection (ICRP). *Protection of the Public in the Event of Major Radiation Accidents. Publication* # 40, Oxford and New York: Pergamon Press, 1984.

[18] International Atomic Energy Agency (IAEA). *Derived Intervention Levels for Application in Controlling Radiation Doses to the Public in the Event of a Nuclear Accident or Radiological Emergency.* Safety Series # 81, Vienna: IAEA, 1986.

[19] Page, R.D. "Canadian Power Reactor Fuel" in CANDU Reactors, Lectures presented to *the International Atomic Energy Agency,* October 12-16 (1981) compiled by Zarecki, C.W. (Also, AECL Report # 5609, 1982).

[20] Sokaguchi, R.L. "HOTSPOT: A Computer Code for Predicting The Fuel Bundle Temperature Distribution History Following a LOCA." Nuclear Studies and Safety Department, *Ontario Hydro (OH) Internal Report* (1977).

[21] Keil, H. and Dorbolo, G. "CURIES: A Computer Program to Predict Inventory Buildup and Activity Releases from CANDU Fuel." *Atomic Energy of Canada Limited (AECL) Internal Report,* (1979).
[22] Tong, J.S.C. and Bash, J.M. "FIREBIRD-I-Program Description." *AECL Internal Report* (1976).
[23] Baby, G.B. "PRESCON-Users Manual." *AECL Internal Report* (1978).

(*Manuscript Received:* 7.9.1987; *Accepted:* 18.1.1988)

دراسة لحساب كمية الإشعاعات المنبعثة من المفاعل الكندي في حالة افتراض انكسار في الجزء الرابط لوحدة الوقود

فهمي محمود حسين

قسم الهندسة الميكانيكية، كلية الهندسة، جامعة الملك سعود، ص.ب ٨٠٠، الرياض١١٤٢١، المملكة العربية السعودية

ملخص البحث. في هذا البحث تم افتراض حادثة انكسار في الجزء الرابط لوحدة وقود من المفاعل الكندي في محطة بيكرنج (ب) وتم الحساب على أساس افتراض أن كل الوقود بالوحدة والمكون من ١٢ جزء قد قذف بقوة إلى أرضية المفاعل وفي هذه الحالة يمكن أن يحدث أحد احتمالات ثلاثة:

١) إما أن تظل أجزاء الوحدة متماسكة.

٢) أو ينكسر الجزء إلى الوحدات الأصغر.

٣) أو تنكسر الوحدات الأصغر إلى قطع وشظايا.

وتم حساب المواد المشعة المنبعثة في كل حالة من الحالات سواء في حالة ما إذا كان الحاجز الوقائي سليما أو إذا افترضنا أن الحاجز الوقائي قد تهدم واستخدمت في الحسابات وحدة الوقود ذات أكبر قدرة من باب الأمان.

وأثبتت الحسابات أنه في حالة افتراض الحاجز الوقائي سليما فإن كمية الإشعاعات المنبعثة للجو تكون أقل بكثير من القيمة المسموح بها حسب توصيات الوكالة الدولية أو هيئة الرقابة الكندية وفي حالة ما إذا تم إفتراض انكسار الحاجز الوقائي فإن كمية الإشعاعات المنبعثة للجو تكون أكبر بكثير من حالة بقائه سليما إلا أنها مازالت أقل من القيمة المسموح بها في مثل تلك الحالات حسب توصيات الهيئتين السابقتين.