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Radioactive Effluents from CANDU 6 Reactors During Normal Operation

Rejets radioactifs des réacteurs CANDU 6 en cours d'exploitation normale

C.R. Boss, P.J. Allsop

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RÉSUMÉ

Pendant l'exploitation normale d'un réacteur CANDU 6, divers déchets radioactifs sont produits sous forme gazeuse, liquide et solide. Le réacteur CANDU 6 et ses systèmes sont conçus de manière à réduire ces déchets au minimum, mais de petites quantités de déchets gazeux et liquides sont continuellement rejetées en concentrations très faibles. Ce rapport traite de la composition de ces rejets gazeux et liquides chroniques.

Du point de vue de la sûreté, les doses reçues par les personnes du public attribuables aux rejets radioactifs chroniques des réacteurs CANDU 6 sont négligeables. De même, les doses reçues par les populations régionales et globales sont négligeables et sont généralement inférieures à 0,001 % du fond naturel de rayonnement. Quoi que bien en-dessous des limites réglementaires, les rejets de tritium, de gaz nobles et d'émetteurs β^{-}-γ bruts ont été les plus importants au point de vue radiologique, tandis que les radio-iodes et les particules se sont révélés les plus dangereux pour ce qui est des doses reçues par la population.

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ABSTRACT

During routine operation of a CANDU 6 reactor, various gaseous, liquid, and solid radioactive wastes are generated. The layout of the CANDU 6 reactor and the design of its systems ensure that these are minimized, but small quantities of gaseous and liquid wastes are continually discharged at very low concentrations. This report discusses the make-up of these chronically generated gaseous and liquid effluents.

From a safety perspective, the doses to individual members of the public resulting from radioactive wastes chronically discharged from CANDU 6 reactors have been negligible. Similarly, doses to the regional and global populations have been negligible, generally less than 0.001% of background. While far below regulatory limits, releases of tritium, noble gases and gross β^--γ have been the most radiologically significant emissions, while radioiodine and particulates have had the greatest potential to deliver public dose.

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1. INTRODUCTION

During routine operation of a CANDU[1] reactor, various gaseous, liquid, and solid radioactive wastes are generated. The layout of the CANDU 6 reactor and the design of its systems ensures that these are minimized, but small quantities of gaseous and liquid wastes are continually discharged at very low concentrations; solid wastes are stored in the station. This report discusses the make-up of these chronically generated gaseous and liquid effluents,[2] their sources, and the radiological consequences of their release.

This report uses the CANDU 6 reactor as the reference plant, so experience from the CANDU 6 reactor is used throughout. It also draws on the experience of other CANDU-reactor designs, when appropriate. There are differences in the radioactive emissions from the different CANDU 6 plants and, where appropriate, the reasons for these differences are presented. This report is not, however, a rigorous analysis of station-specific performance; rather, it examines CANDU 6 reactor performance in broad terms.

2. SOURCES OF GASEOUS RADIOACTIVITY

CANDU 6 reactors are both moderated and cooled by heavy water (D_2O). The principal gaseous radioactive wastes are associated with this heavy water: tritium (hydrogen-3 or T) and carbon-14. Tritium is primarily formed in the moderator and heat-transport (HT) systems, and is released as tritiated heavy-water vapour; that is, the molecules of heavy-water vapour include some DTO molecules. Carbon-14 is principally formed in the moderator system, and released as carbon dioxide (CO_2). Three other types of gaseous radioactive wastes are also released: noble gases, radioiodines and particulate.

2.1 Tritium

Tritium is a radioactive isotope of hydrogen. It undergoes β^- decay to helium-3, with a half-life of approximately 12.33 years. The radiation emitted in the decay process is a low-energy β^- particle, with an average energy of 5.7 keV and an end-point energy of 18.6 keV. This energy is so low that the tritium-β^- has very little range: less than approximately 6 mm in dry air or 6 μm in tissue. While outside the body, tritium contributes negligible radiological hazard, because its β^- particle cannot penetrate the outer layer of dead skin. When tritium enters the body by either inhalation or absorption through the skin, however, the decay energy carried by the β^- particle is deposited in tissue.

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1 CANDU® (CANada Deuterium Uranium) is a registered trademark of Atomic Energy of Canada Limited.

2 Unless explicitly stated otherwise, all references to radioactive wastes and discharges refer to normal operation; accident conditions are not considered.

Within a CANDU reactor, tritium is principally formed by deuterium nuclei in the heavy water capturing a thermal neutron. Tritium is therefore usually bound in a molecule of heavy water as DTO. If heavy water from one of the heavy-water systems or their auxiliaries escapes, it can create a source of tritiated heavy-water vapour in the air. This tritiated heavy-water vapour can be ingested either by inhalation or absorption through exposed skin. If present, heavy-water vapour in the station atmosphere can therefore create an internal, occupational radiation hazard in the station. If air containing heavy-water vapour is discharged from the plant through normal ventilation flows, this discharge will contribute to the station's tritium emissions.

2.1.1 Formation of Tritium in CANDU Reactors

In a CANDU reactor, most of the tritium is formed in the thermal-neutron-capture reaction, $^{2}H(n,\gamma)^{3}H$. The reaction occurs in the heavy water of both the moderator and HT systems. The mass of HT heavy water in the high-flux zones of the fuel channels is small compared with the mass of moderator heavy water inside the calandria. In addition, the average thermal flux inside the fuel channels is slightly lower than in the moderator. As a result, the rate of tritium formation is almost sixty times higher in the moderator than in the HT system. Since the total mass of heavy water in the HT system is smaller than in the moderator system, the predicted ratio of ultimate tritium concentrations for these two systems is about 45:1.[3] The ultimate concentrations depend on the capacity factor achieved. For a capacity factor of 90%, the calculated ultimate concentration in the moderator heavy water is 3.64 TBq/kg after 40 years; in the HT heavy water it is 0.081 TBq/kg.

Extrapolating measured tritium concentrations from currently operating CANDU 6 reactors, the tritium concentrations in the moderator and HT systems should be 3.3 TBq/kg and 0.1 TBq/kg, respectively, after 40 years. The extrapolated concentration of tritium in the moderator is therefore lower than the calculated concentration, while that in the HT system is higher. Similarly, the measured ratio of tritium concentrations in the two systems is lower than the calculated value. At Point Lepreau and Gentilly 2, the ratio is between 30 and 40, but the ratio is lower at Wolsong 1 and Embalse. This discrepancy between calculated and measured ratios is attributed to:

a. an overestimate in the calculated thermal fluxes in the reflector regions of the calandria,

b. the mixing of moderator and HT heavy water during heavy-water-management operations,

c. the formation of tritium from the lithium added to the HT system for corrosion control (see below), and

d. the integral effects of replacing some tritiated heavy water with fresh (low tritium) heavy water each year.[4]

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[3] This ratio ignores the effects of heavy-water loss and make-up.

[4] The exact effects of such replacement will depend on the specific management strategy adopted by each operator.

The decay product from tritium is helium-3. In a thermal-neutron flux, this isotope undergoes an (n,p) reaction to form tritium; i.e., $^{2}H(n,\gamma)^{3}H \rightarrow (\beta^{-}\text{-decay}) \rightarrow {}^{3}He(n,p)^{3}H$ (Holford and Osborne 1979). The thermal-neutron cross-section for the (n,p) reaction is relatively large (5 330 barns), so the probability of this reaction occurring in the moderator — which has the higher tritium concentration — is high. The time constant for this (n,p) reaction, however, is long compared with the half-life of helium-3 in the moderator system,[5] so it is not expected to be a significant contributor to tritium production.

Tritium can also be formed in ternary fission in the fuel, as it is in all other reactor types. Tritium formed in this manner should be retained inside the fuel sheath. This small source of tritium is therefore neglected as a contributor to gaseous CANDU effluents. Tritium is also formed as an activation product of boron and lithium.

Natural boron — as boric anhydride (B_2O_3) — is added to the moderator system for reactivity control. Tritium can be formed from this boron. The concentrations of boron used, however, are typically 0.25 ppm; the formation of tritium from boron is therefore negligible.

Lithium — as lithium hydroxide (LiOH) — is added to the HT system to control corrosion. The lithium concentration is controlled in the range 0.35-1.4 ppm. At these concentrations, the formation rate of tritium from lithium is predicted to be about 18% of the formation rate due to the $^{2}H(n,\gamma)^{3}H$ reaction in the HT system.

2.2 Carbon 14

Carbon-14 is a radioactive isotope of carbon that undergoes β^{-} decay to form nitrogen-14. Carbon-14 has a half-life of 5 730 years. The radiation emitted in the decay process is a low-energy β^{-} particle with an average energy of 49.5 keV and an end-point energy of 156 keV. Carbon-14 becomes a significant radiation hazard only when it has been ingested or inhaled.

2.2.1 Formation of Carbon-14 in Moderator and HT Systems

Carbon-14 is formed in an (n,α) reaction on the oxygen-17 isotope, a capture process with a thermal-neutron cross-section of 0.24 barns. Oxygen-17 is present in the uranium dioxide fuel, and in the moderator and HT heavy water. In addition, carbon-14 can be formed in the moderator system by a neutron-proton reaction with the nitrogen-14 contained in the gadolinium nitrate used for reactor shut-down. Thus, carbon-14 is formed in the fuel and both heavy-water systems. The fuel sheath ensures that the carbon-14 formed in the fuel cannot contribute significantly to carbon-14 emissions. Gadolinium nitrate irradiation occurs only during brief shut-down and start-up periods, and it contributes far less than 1% of carbon-14 formed by oxygen-17 irradiation. Oxygen-17 irradiation is therefore the primary source of carbon-14 formation in the moderator and HT systems.

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[5] Helium-3 off-gases from the HT system in the degasser-condenser, and from the moderator system in the cover gas.

Most of the carbon-14 formed is in the moderator system, because the mass of heavy water exposed to high thermal-neutron fluxes is greater in that system than in the HT system. In a CANDU 6 reactor, the production rates of carbon-14 in the moderator and HT systems — based on the thermal fluxes given above — are 17.2 TBq/a and 0.30 TBq/a, respectively, for a capacity factor of 90%. Due to the formation of free radicals in the heavy water during radiolysis, this carbon-14 forms molecules of carbon dioxide (CO_2), carbon monoxide (CO) and possibly organic species. Historic evidence shows that most of the carbon-14 ends up as carbon dioxide and is present in the heavy water as carbonate ions. Almost all of this carbonate is immobilized on the ion-exchanged resins used for chemistry control; only a small fraction contributes to carbon-14 emissions.

The processes currently used to produce heavy water enrich that water in oxygen-17. Typically, fresh heavy water has 55% more oxygen-17 than natural water. This enrichment increases the production rate of carbon-14, because oxygen-17 is the primary source of this isotope in a CANDU reactor. New processes for producing heavy water that do not enrich the product water in oxygen-17 will be deployed within ten years. Carbon-14 production rates from all CANDU reactors will therefore decrease as these new processes take over, with resulting decreases in emissions and inventories.

2.2.2 Formation of Carbon-14 in Annulus-Gas System

The annulus between each pressure tube and its calandria tube is filled with carbon dioxide gas at a pressure slightly above atmospheric. The humidity of this annulus gas is monitored; an increase in humidity may indicate a pressure-tube leak.

Within the annulus gas, carbon-14 is formed from the naturally present carbon-13 in the carbon dioxide through the neutron capture reaction $^{13}C(n,\gamma)^{14}C$. As both the abundance of carbon-13 and the cross-section for this reaction (0.9 millibarns) are small, however, the amount of carbon-14 formed in the annulus-gas system by this mechanism is negligible. In a CANDU 6 reactor operating at a 90% capacity factor, the carbon-14 formed by this reaction is estimated to be 43 MBq/a.

Carbon-14 will also be formed from the naturally present oxygen-17 in the carbon dioxide by the $^{17}O(n,\alpha)^{14}C$ reaction. For a CANDU 6 reactor operating with a capacity factor of 90%, the carbon-14 formed in this reaction is estimated to be 510 MBq/a.

Carbon-14 can be formed in a neutron-proton reaction with nitrogen-14, $^{14}N(n,p)^{14}C$. The thermal-neutron cross-section for this reaction is relatively large, 1.82 barns. Nitrogen or air — which is 80% nitrogen — was used as the annulus gas in early CANDU reactors. Those early reactors have now either been decommissioned (NPD Rolphton and Douglas Point) or converted to carbon dioxide (Pickering Nuclear Generating Station A (NGS-A)). Thus one might expect no contribution from this reaction. However, there may be a small nitrogen impurity in the carbon dioxide, leading to the formation of carbon-14 in the annulus gas of CANDU 6 reactors. Usually, the carbon dioxide used in CANDU 6 reactors has a low nitrogen-impurity level, giving a carbon-14 production rate by the $^{14}N(n,p)^{14}C$ reaction of about 10 GBq/a for a 90% capacity

factor. If the reactor were continuously operated at the maximum nitrogen-impurity limit, the production rate of carbon-14 would be 52 GBq/a for a 90% capacity factor.

Along with the ongoing production of carbon-14 in the annulus gas, there can also be a historic contribution. This affects Pickering NGS-A, which used nitrogen for its annulus gas at one time. This led to the deposition of carbon-14 containing solids on the annulus walls. After conversion of the annulus gas to carbon dioxide, this carbon-14 deposit continued to bleed from the calandria-tube walls into the annulus gas as $^{14}CO_2$, thereby increasing the carbon-14 releases from this station. This was particularly evident when the four reactors at Pickering NGS-A were being retubed, as oxygen was added to the annulus gas to accelerate the removal of carbon-14 dust (Figure 1). While carbon-14 emissions from Pickering NGS-A remained far below regulatory limits during this process, they were elevated relative to its historic emissions, or emissions from CANDU 6 reactors.

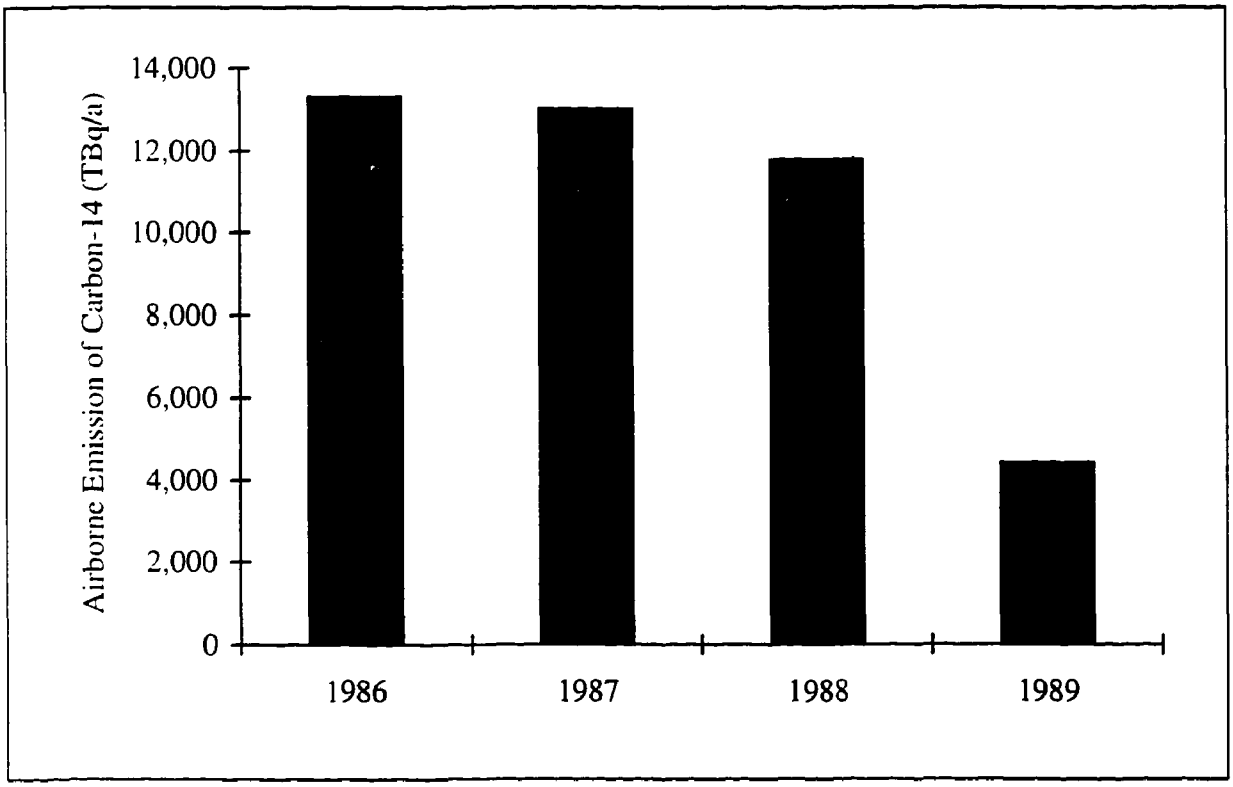

Figure 1 – Airborne Carbon-14 Emissions from Pickering NGS-A
(UNSCEAR 1993)

2.2.3 Other Sources of Carbon-14

In addition to the carbon-14 formed in the moderator, HT and annulus-gas systems, it is possible that any stagnant, gas-filled penetration that goes into the core could contain air. Should that penetration be opened for maintenance, the irradiated air from the penetration would contain traces of carbon-14. There will also be a trace of carbon-14 formed by the $^{13}C(n,\gamma)^{14}C$ reaction in

the carbon of the lubricant used during fuel fabrication, but this will be retained by the fuel sheath. The production of carbon-14 from this source is small.

2.2.4 Total Carbon-14 Production

In a CANDU 6 reactor, the production of carbon-14 from all reactions totals 18 TBq/a. Almost all of this is immobilized on the ion-exchange resins used for chemistry control, and is not released from a station.

2.3 Noble Gases

The noble gases released from a CANDU reactor are the krypton and xenon isotopes formed as fission products in the natural-uranium fuel. These noble gases collect in the free space between the pellets of uranium dioxide fuel and the zircaloy-4 fuel sheath. If a fuel sheath fails, because of a small pin-hole defect, the noble gases escape to the HT coolant. These noble gases, and other soluble fission products, are monitored by a gaseous-fission-product monitor to indicate a fuel-sheath defect.

With a severe fuel-sheath defect, small amounts of the uranium fuel may be washed out of the fuel sheath into the coolant. Such traces of uranium dioxide are, usually, readily deposited on the wetted walls of the HT system. These deposits of "tramp" uranium, plus any contamination on the outer surfaces of the fresh fuel-bundle sheaths, will also release gaseous fission products to the coolant.

Gaseous fission products will circulate with the HT coolant until they reach the degasser condenser in the pressure-and-inventory-control system. The noble gases removed by the degassing process are routed either to the vapour-recovery system to recirculate in the reactor building, or to the reactor-building-ventilation system for discharge from the station exhaust stack.

In addition to krypton and xenon, there are also sources of radioactive argon. The argon-41 is formed as an activation product from the argon impurity present in the gaseous-process systems. That is, there is a very small argon impurity in the carbon dioxide gas used for the annulus-gas system, and in the helium used for the moderator cover gas and the liquid-zone-control helium system. There is also argon in any air impurity that could enter any of the gaseous-process systems.

2.4 Iodine-131

Iodine-131 is formed as a fission product, and has a radioactive half-life of 8.05 days. Iodine has particular relevance from a radiation-protection viewpoint, because it is metabolized by one small organ in the human body, the thyroid. As a result, the relative rate of local energy deposition in thyroid tissue (i.e., the exposure rate) is extremely high if this isotope is ingested.

In addition to iodine-131, there are several other iodine radionuclides. However, the half-life, energy released in decay, and fission yield of iodine-131 make this radioiodine the most important as a radiation hazard during normal operations.

The radioiodines are formed by fission in the natural-uranium fuel. A defect in one of the fuel sheaths must occur to allow the iodine produced inside the fuel pellets to reach the HT coolant. The design of a CANDU 6 reactor features a delayed-neutron-detection system to identify the location of any such defective fuel. If a defective fuel bundle is detected, the on-line fuel-handling system allows its prompt replacement. The decision to replace a bundle is made by the operator, considering regulatory issues, plant practices and ALARA[6] principles.

Small amounts of fuel are present on the outside of the fuel sheaths, either as tramp uranium left on the fresh fuel during manufacture, or tramp uranium that escaped from an earlier defect in a fuel sheath. The manufacturing specification for fresh-fuel contamination severely limits this source of tramp uranium. In a CANDU 6 reactor, the maximum mass of tramp uranium that could release iodine-131 to the coolant is 1 gram; this could release 790 GBq/a of iodine-131 to the coolant. Since iodine is very soluble in water (the partition coefficient for iodine between the water and non-condensable phase is large), most of the iodine would remain in the water phase in the degasser condenser. Thus, releases via this path would be expected to be much less than 7.9 GBq/a. Most of the radioiodines would be removed by the HT-purification system.

2.5 Particulate

The term "radioactive particulates" refers to all forms of airborne contamination that have not been characterized above. This group includes all forms of α and β^--γ emitters that, associated with particles of dust, escape from a contained system and become airborne or settle on surfaces. The radioactivity may be that of activation products from the materials of construction, the fuel, a fission product from the fuel, or even an actinide formed in the irradiation of the fuel.

The air-flows from the reactor-building-ventilation exhaust, and the small purge flows from the vapour-recovery systems, are directed through a filter train before being exhausted to the stack. This filter train consists of a pre-filter, a HEPA filter, a charcoal filter and a final HEPA filter. The exhaust air from the spent-fuel-bay area in the service building is similarly filtered, but the exhausts from other areas of the service building have simpler filter trains with a pre-filter and HEPA filter. These filter trains reduce the levels of particulate and radioiodine in the airborne emissions.

3. SOURCES OF LIQUID RADIOACTIVITY

Liquid wastes within a CANDU station are segregated into normal and low-activity waste streams. The normal-activity waste includes wastes from the laboratories, laundries, some service-building drains, upgrader drains, and decontamination centre. The drains from the reactor building, the heavy-water area, the spent-fuel bay, and the resin-storage area are also directed to this normal-activity waste stream. The low-activity waste stream includes wastes from showers and building drains in those areas of the service building that would not normally be contaminated.

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6 As Low As Reasonably Achievable given economic, social and technical considerations.

3.1 Tritium

In a CANDU reactor, any escape of heavy water from the HT and moderator systems or their auxiliaries will involve a release of tritium. The cost of heavy water provides a strong incentive to recover all heavy-water escapes and return them to the appropriate system. Every effort is made to recover these escapes before any light water mixes with the escaped heavy water, downgrading the heavy water. Provided the recovered liquid has sufficient isotopic content, there is usually an economic incentive to extract the heavy water via the upgrader. The upgrader is also used to remove light water from the HT and moderator systems. This upgrading process produces a light-water discharge containing traces of heavy water and tritium.

Other liquid wastes often contain at least a trace of tritium. These may arise from showers or floor drains within specific areas of the service building. There will also be liquid wastes from the laboratories, laundries, other service-building floor drains, upgrading columns and decontamination centre.

3.2 Other

The radionuclides with mixed β^--γ activity present in CANDU liquid wastes arise from activation or fission products present in the process systems. The principal activation products appearing in the liquid waste are zirconium-95 ($t_{1/2}$ = 64 days) and its daughter-product niobium-95 ($t_{1/2}$ = 35 days), cobalt-60 ($t_{1/2}$ = 5.27 years), and chromium-51 ($t_{1/2}$ = 27.7 days). These are all relatively long-lived radionuclides that have little opportunity to decay significantly in the process of escape and collection for radioactive waste management. The ^{95}Zr and ^{95}Nb are activation products from the zirconium alloys used in the fuel channels and the fuel bundles. The ^{60}Co and ^{51}Cr are activation products from stainless-steel components used for the external circuit of the HT system and the stainless-steel components used throughout the moderator system. There are also less abundant radionuclides, such as iron-59 ($t_{1/2}$ = 44.5 days), zinc-65 ($t_{1/2}$ = 243.8 days), cesium-137 ($t_{1/2}$ = 30.2 years) and cesium-134 ($t_{1/2}$ = 2.1 years), whose significance varies from station to station.

4. RADIOACTIVE EFFLUENTS FROM CANDU 6 REACTORS

The design of the ventilation systems in the CANDU 6 reactor ensures that air flowing through any area with airborne radioactivity will be directed to a contaminated exhaust. These contaminated exhausts are directed to the exhaust stack after passing through the filter trains mentioned in section 2.5. Continuous monitoring of the air passing through this stack measures the concentrations of radioiodines, noble gases and particulates. Airborne tritium releases are monitored by laboratory analyses of the heavy-water vapour collected either in adsorbent cartridges or bubblers. Airborne carbon-14 can be monitored by directing a small flow of the exhaust air through a caustic solution to absorb any carbon-14 that may be present as carbon dioxide, then measuring the carbon-14 in the laboratory using liquid-scintillation analysis.

The design of the process systems ensures that any liquid discharges that might contain radioactivity are routed to the radioactive-waste tanks. The liquid discharged from these tanks is

quantified by sampling the radioactive-waste tanks before discharge. These samples are measured for tritium content and any other β^--γ emitter. There is also continuous monitoring of any discharge to the condenser cooling water; if unacceptable levels of activity are detected during a discharge, the pump is stopped.

Radioactive releases from the four operating CANDU 6 stations for the five gaseous and two liquid categories are available from a variety of references. The discussion in this section concentrates on the releases of tritium, because these are the most significant radioactive releases from a public-exposure perspective.

The available data on annual releases of gaseous radioactive effluents from the four CANDU 6 stations, over the period from 1984 to 1994, are listed in Tables 1-5. The annual releases of liquid radioactive effluents are listed in Tables 6 and 7.

Table 1 – Gaseous Tritium Releases From CANDU 6 Reactors

Year	Annual Releases of Airborne Tritium (GBq/a)			
	Point Lepreau	Gentilly 2	Wolsong 1	Embalse
1984	68 000	49 000	100 000	7 000
1985	110 000	53 000	90 000	30 000
1986	200 000	136 000	242 000	26 000
1987	220 000	123 000	313 000	34 000
1988	220 000	195 000	318 000	49 000
1989	210 000	137 000	226 000	86 000
1990	210 000	227 000	231 000	75 000
1991	170 000	330 000	257 000	55 000
1992	360 000	322 000	388 000	69 000
1993	623 000	200 000	211 000	38 000
1994	555 000	258 000	–	134 000

Table 2 – Gaseous Carbon-14 Releases From CANDU 6 Reactors

Year	Annual Releases of Airborne Carbon-14 (GBq/a)			
	Point Lepreau	Gentilly 2	Wolsong 1	Embalse
1984	–	–	–	–
1985	–	–	–	385
1986	–	–	–	318
1987	330	–	–	472
1988	360	–	–	458
1989	430	–	–	467
1990	480	–	–	–
1991	350	450	–	–
1992	720	7	–	–
1993	580	480	–	–
1994	200	2 915	–	350

Table 3 – Noble Gas Releases From CANDU 6 Reactors

Year	Annual Releases of Airborne Noble Gases (GBq·MeV/a)			
	Point Lepreau	Gentilly 2	Wolsong 1	Embalse
1984	0	26 000	18 000	800
1985	1 000	119 000	17 000	57 000
1986	5 000	53 000	10 000	20 000
1987	100	49 000	12 000	42 000
1988	300	80 000	6 000	15 000
1989	0	73 000	5 000	120 000
1990	0	60 000	8 000	24 000
1991	13 000	54 000	6 000	46 000
1992	11 000	33 000	5 000	150 000
1993	4 790	69 000	12 000	140 000
1994	–	59 000	–	16 000

Table 4 – Gaseous Radioiodine Releases From CANDU 6 Reactors

Year	Annual Releases of Airborne (Iodine-131) (GBq/a)			
	Point Lepreau	Gentilly 2	Wolsong 1	Embalse
1984	–	<0.014	0.037	0.0
1985	0.0042	–	–	0.23
1986	0.029	–	–	2.5
1987	0.0	0.2	0.0	0.002
1988	0.05	0.0	0.0	0.37
1989	0.0052	0.0	4.0×10^{-5}	0.073
1990	0.0	0.0	0.0	1.4
1991	0.016	0.0019	1.5×10^{-6}	1.5
1992	0.0016	0.0	4.0×10^{-4}	0.07
1993	0.0002	<0.004	–	–
1994	0.0089	0.0002	–	0.26

Table 5 – Gaseous Particulate Releases From CANDU 6 Reactors

Year	Annual Releases of Airborne Particulate (GBq/a)			
	Point Lepreau	Gentilly 2	Wolsong 1	Embalse
1984	–	–	0.078	0.0
1985	0.0	0.0	0.006	0.22
1986	0.0	0.0	0.0	0.039
1987	0.0	0.0	0.0	0.95
1988	0.0	0.0	0.0	0.044
1989	0.0	0.0	0.006	0.11
1990	0.0	0.0004	0.0	0.0
1991	0.0	0.0013	0.0	0.011
1992	0.004	0.031	0.0	0.025
1993	0.0012	0.050	–	–
1994	–	0.070	–	0.004

Table 6 – Liquid Tritium Releases From CANDU 6 Reactors

Year	Annual Releases of Waterborne Tritium (GBq/a)			
	Point Lepreau	Gentilly 2	Wolsong 1	Embalse
1984	28 000	14 000	6 600	1 900
1985	24 000	27 000	12 000	16 000
1986	110 000	41 000	37 000	78 000
1987	96 000	80 000	50 000	91 000
1988	120 000	58 000	80 000	170 000
1989	320 000	48 000	60 000	260 000
1990	160 000	163 000	52 000	220 000
1991	110 000	248 000	93 000	519 000
1992	320 000	263 000	42 000	160 000
1993	485 000	208 000	–	200 000
1994	260 000	234 000	–	140 000

Table 7 – Gross ß⁻-γ Liquid Releases From CANDU 6 Reactors

Year	Annual Releases of Waterborne Gross ß⁻-γ Activity (GBq/a)			
	Point Lepreau	Gentilly 2	Wolsong 1	Embalse
1984	13	6.8	5.5	26
1985	1.6	5.8	1.0	1.9
1986	1.6	2.9	2.6	6.5
1987	0.47	11.0	1.5	3.8
1988	1.6	2.3	1.2	3.6
1989	1.0	1.1	0.31	5.9
1990	2.0	4.2	0.20	3.4
1991	4.0	16	0.20	2.0
1992	2.0	4.5	0.30	2.1
1993	66.0	27.6	0.11	1.8
1994	–	14.8	–	1.6

4.1 Tritium Releases

The tritium releases listed in Tables 1 and 6 have increased since the plants were originally placed in-service, presumably because the tritium levels in the moderator and HT systems have built up (Figure 2). Overall emissions have not, however, increased in direct proportion to the tritium levels in the moderator and HT systems. The ratio of increase varies from station to station, and no single value characterizes all stations equally. Between 1986 and 1993, for example, the relative increase in the total tritium emitted by Embalse was approximately 120% of the relative increase in the moderator tritium concentration, but at Wolsong 1 it was 10%.

4.2 Noble Gases

The noble-gas emissions from Point Lepreau have been reported as being zero in some years. However, in those years it would be more accurate to describe them as being below detectable levels. In response to improvements in instrumentation, Point Lepreau commissioned a new noble-gas monitor for stack discharges in 1990. Subsequently, it has had sufficient sensitivity to report measurable values for noble-gas emissions.

The noble-gas emissions reported by Gentilly 2 have varied from year to year. This variation has been associated with recalibration of the noble-gas monitor, which includes new estimates of the background dose rate at the monitor and the efficiency of the detector. Thus, the detection limit of the noble-gas monitor at Gentilly 2 is not evident from the data.

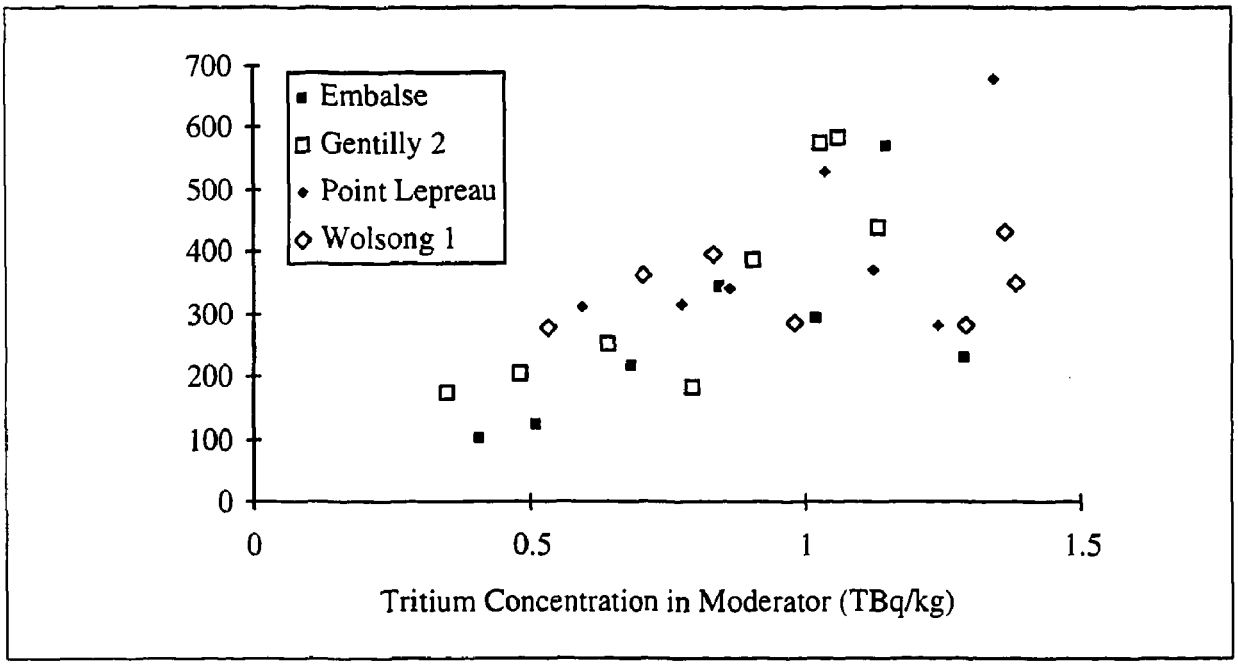

Figure 2 – Evolution of CANDU 6 Reactor Tritium Emissions
(for 1986 through 1993, inclusive)

The reported noble-gas releases from Wolsong 1 are lower than those from other CANDU 6 reactors. This may be due to Wolsong's off-gas waste-management system, a system unique to Wolsong 1. Wolsong 1 has also installed a new noble-gas monitor, but it is unclear when this was placed in service.

5 ESCAPE PATHS FROM PROCESS SYSTEMS

5.1 Heat Transport System

Most of the components in the HT system of a CANDU reactor are connected by an all-welded piping system. There are, however, components with mechanical joints. These include the rolled joints between the pressure tubes and the end fittings, graylocs between the end fittings and the feeders, and end-fitting closures. These are high-integrity, metal-to-metal mechanical joints. In addition to the mechanical joints, there is a gland-seal system on each HT-pump drive shaft. While there are no valves in the main HT circuit, several of the auxiliary systems contain valves, including the feed-and-bleed system, shutdown-cooling system, D_2O -sampling system and purification system. Each of these represents a potential escape path for heavy water.

5.1.1 Heavy-Water Escape from Main HT Circuit

Each fuel-channel is terminated by end fittings used during refuelling. Leakage past the rolled joints connecting each pressure tube to its end fittings allows HT heavy water to enter the annulus-gas system, and therefore does not necessarily represent an emission path. However, leakage past the grayloc seals or end-fitting closures would be directly to the fuelling-machine-vault atmosphere. When the HT system is hot and pressurized, the seal faces in the closures are forced against the seal rings, so such leakage is normally expected to be smaller than when the HT system is cold and depressurized. Due to the pressure and temperature of the HT system, however, problems with an end-fitting closure could inject heavy-water vapour into the fuelling-machine vaults.

The fuelling-machine-vault atmosphere is circulated through a system of vapour-recovery dryers. These desiccant dehumidifiers adsorb heavy-water vapour from the vault atmosphere, preventing its escape from the station. The desiccant is regenerated at intervals to maintain drying capacity, and the regeneration condensate is routed to the D_2O-management system. Operating experience shows that these dryers are extremely effective at recovering heavy water. In a CANDU 6 reactor there are four such dryers for the fuelling-machine vaults, supplemented by an exhaust dryer that polishes a purge flow taken from these main dryers. This purge flow is discharged up the stack, minimizing vapour migration from the fuelling-machine vaults to adjacent areas (Figure 3). This ensures that little of the heavy water escaping from the main HT system to the fuelling-machine vaults is discharged from the station. Consequently, leakage past the end-fitting closures is considered to contribute less than 1% of the airborne-tritium released.

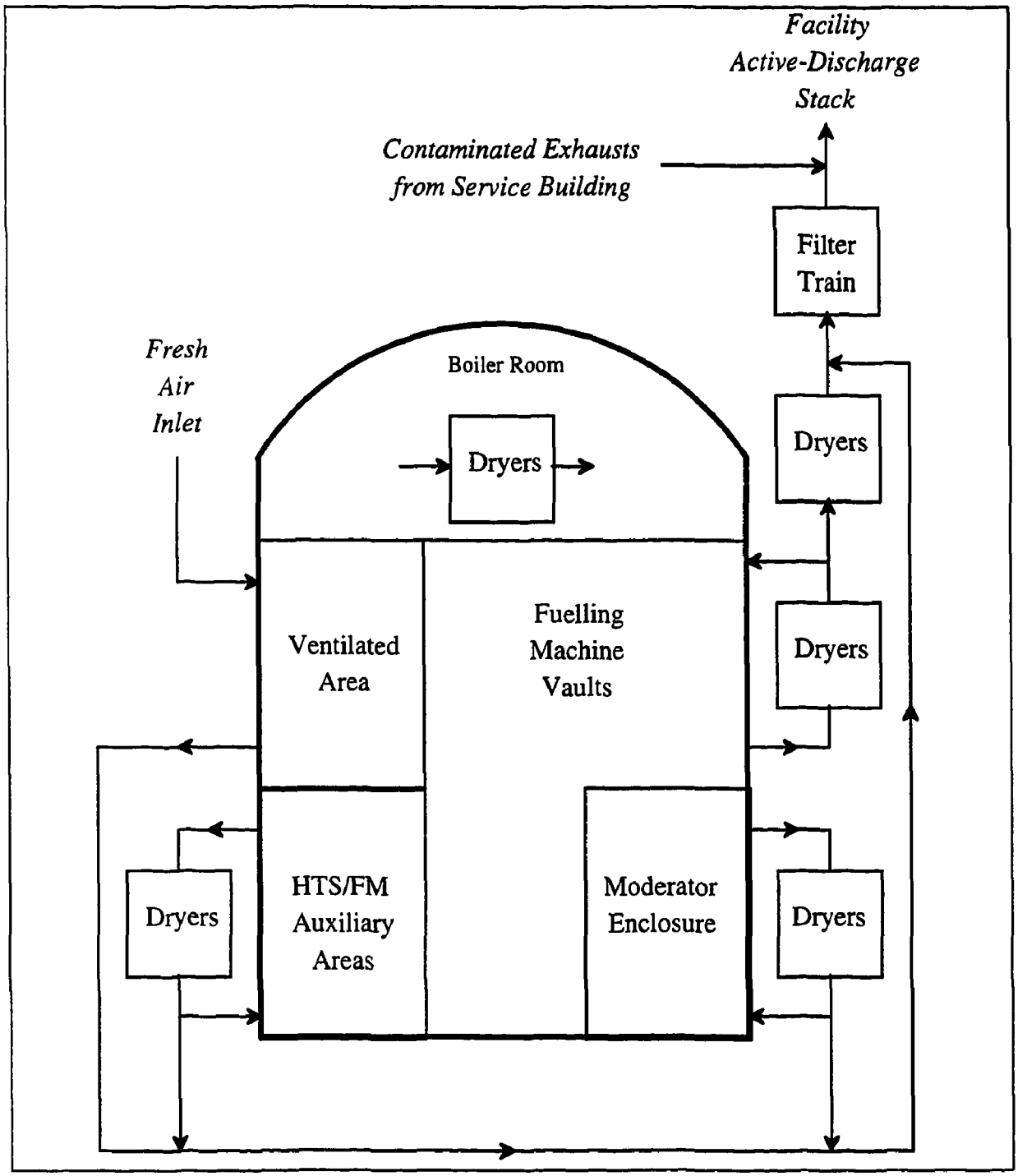

Figure 3 – Simplified Schematic of Ventilation and Vapour Recovery

The HT-pump gland seals are served by a dedicated, pressurized and cooled heavy-water supply, to prevent leakage out of the main circuit. Some of this D_2O flows into the HT system; the remainder is collected at low pressure and returned to the D_2O storage tank. The gland seals have their own backup seal. There is no evidence to suggest that these gland seals provide a significant leakage path from the HT system.

During normal operation, trace levels of tritium are measured in the blow-down water from the secondary side of the steam generators. This tritium reaches the secondary side from the HT system by diffusion through the steam-generator tubes. It is only measurable because of the very large heat-transfer area of the steam generators. While minor, this pathway does contribute to waterborne tritium emissions, and is monitored.

5.1.2 Heavy-Water Escape during Fuelling Operations

During refuelling, a temporary mechanical joint is created at either end of a fuel channel between the fuelling machine and the end-fitting. The leak-tightness of these joints will depend on the surface finish of the interfacing components, which is very strictly controlled during manufacture. A few channels may, however, experience sealing problems after prolonged operation, or due to debris accumulation. Until repaired, these would raise the escape of heavy water to the fuelling-machine vault. In addition, the interspace in the snouts of the fuelling machines will hold a small mass of heavy water that will flash to steam as the fuelling machines unlatch. Finally, several mechanical joints in the structure of the fuelling machine itself may leak. All these sources of heavy-water escape will be collected by the dryers. Consequently, their contribution to the station gaseous effluents will be small.

The fuelling machines are supplied with HT heavy water by the HT auxiliary system. Any heavy-water escape from this system would principally be to an atmosphere served by a vapour-recovery dryer. This route is not expected to be a significant contributor to a station's gaseous effluents.

5.2 Moderator System

In a CANDU reactor the fuel channels are installed in a heavy-water-filled calandria vessel. About 5% of the nuclear energy released in the core is deposited directly in the moderator heavy water, appearing as heat. To remove this heat, the heavy water in the calandria is circulated through an external circuit of heat exchangers and pumps.

The moderator system is a low-pressure, low-temperature system, so the opportunities for heavy-water escape are small. However, the main circuit has a number of valves. These valves and the other components in the external circuit are connected using mechanical or flanged joints. Live-loaded, double-packed stem valves are used for the large valves, and bellows stem seals are used for small valves, to reduce leakage. The pump shafts have double mechanical seals with a back-up atmospheric seal. Thus, every effort has been made to reduce heavy-water escape at the design stage. Nevertheless, small rates of heavy-water escape from the moderator systems are detected. It is the responsibility of the operators to ensure that moderator heavy-water escape from these mechanical joints is minimized.

The moderator-enclosure atmosphere is circulated through a dedicated vapour-recovery-dryer system, similar to that used for the fuelling-machine vaults. The opportunity for environmental releases of tritium from equipment in that enclosure is therefore small. The moderator enclosure, however, sits in the moderator room, an area that is contiguous with the boiler room. Leaks from moderator equipment housed in this room and outside the moderator enclosure could result in moderator tritium reaching the boiler room. If this occurs, it is probable that some fraction of this tritium passes into the ventilated area of the reactor building from the boiler room, and is thereby released.

5.3 Annulus-Gas System

At intervals, the annulus-gas system is purged or vented to the atmosphere. This purge gas passes through vapour-recovery dryers and the filter systems noted in section 2.5, before discharge, and therefore should not significantly contribute to tritium emissions. However, these dryers are not designed to remove carbon-14 or gaseous fission products.

5.4 Process-Systems Auxiliaries

The auxiliary systems for both the HT and moderator systems can be sources of heavy-water escape. Those systems housed in areas equipped with vapour-recovery dryers should not contribute significantly to tritium emissions, but not all areas are so equipped. In particular, there are systems within the service building and ventilated areas of the reactor building. Leaks from these would lead directly to airborne tritium emissions.

5.5 Fuel-Handling Systems

Beyond leakage past the seal between the fuelling machine and end-fitting, heavy water can escape from the HT system during fuel handling. A small amount of water adheres to the fuel bundles when they are transferred out of the fuelling machine. In addition, some HT water may leak past the fuelling-machine ram as irradiated fuel is pushed from the fuelling machine onto the cradle in the spent-fuel bay. If released inside the fuelling-machine vaults, these leaks should not contribute significantly to station tritium emissions. HT system water reaching the spent-fuel bay, however, will contribute to tritium emissions; water evaporating from the bay is discharged through the service-building contaminated exhaust. The tritium-emission rate via this route will strongly depend on the fuelling-machine-maintenance protocols adopted by the reactor operator.

Refuelling operations could also be a source of air, and consequently argon, in-leakage to the HT system. Some of the air left in the fuelling machine from discharge operations at the spent-fuel port will enter the HT system as a result of the injection flow from the machine into the end-fitting. The argon component of this air will be activated in the HT system, and ultimately collect in the degasser-condenser.

6. CONSEQUENCES OF EFFLUENT ESCAPE

6.1 Occupational Exposures

The escape of heavy water from the HT and moderator systems can create tritium-in-air hazards in some areas of the service building and reactor building. These tritium-in-air hazards contribute to the radiation exposures of station staff. Other gaseous effluents can also contribute. In particular, noble-gas fission products have been observed in the reactor building, as has carbon-14 in reactors that have used nitrogen as an annulus gas. The noble gases are from the degasser condenser in the HT system. The non-condensables from this unit are fed to the vapour-recovery system, which recirculates them through the reactor building; this is intended to reduce emissions by allowing these nuclides to decay before discharge. Not all stations do this; some discharge the non-condensables through the reactor-building-ventilation system.

6.2 Public Exposures

Individual members of the population, and the population as a whole, receive very small radiation exposures due to the radioactive effluents from a CANDU station. It is impracticable to monitor these exposures using personal dosimeters, because the levels and doses are so small. Instead, potential doses are estimated using mathematical models during a "pathways analysis".

In a pathways analysis, the transport of radioactive effluent through the environment from the point of release to the point of uptake is modeled mathematically. The maximum dose resulting from any uptake is modeled based on the type and the amount of any uptake. The type of uptake refers to the ingestion or inhalation of any activity, or any external exposure from immersion in a cloud of activity or contact with activity deposited on a surface. The amount of uptake will depend on the age, lifestyle, and diet of the individual. In addition, the population group being studied affects the type of pathways analysis performed.

The population group chosen for study in a pathways analysis can significantly affect the calculated results. Populations can be divided into the most exposed individual, the population group surrounding a facility, the local/regional population and the global population. Typically, the dose (Sv/a) to the most exposed individual (member of a critical group) is used for safety-assessment and licensing purposes, while the integral dose (pSv/a)[7] to the local/regional or global populations is used to assess overall environmental impact.

6.2.1 Exposures to an Individual

For Canadian regulatory purposes, annual exposures to individuals are calculated for people in so-called "critical groups". Each critical group is that segment of the local population that, in theory, could receive the highest average individual dose from a given radionuclide. A nuclear facility may have more than one critical group, reflecting local conditions and the various

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[7] person-Sv/a is the sum of all the individual doses (Sv/a) acquired by the population in a specified area.

pathways that different radionuclides may take in reaching the public. The members of each critical group are selected based on their location, age, lifestyle and diet (CSA 1987). Excluded from these critical groups are individuals who have very atypical lifestyles or diet.

Derived release limits (DRLs) are used to quantify the relationship between releases of radioactivity and doses to public-critical groups. DRLs are based on pathways analyses conducted for each station. DRLs are the radioactive releases that over a year would expose any member of the public to the regulatory annual dose limit. In Canada that limit is 5 mSv/a, and the DRLs are calculated on that basis. Canadian CANDU-reactor operators, however, aim to keep releases far below this limit; for example, keeping all emissions below 1% of the DRLs (calculated doses below 0.05 mSv/a) is a common target. [8] Canadian operators measure their releases against this lower target level, which is substantially below the recommended annual limit for members of the public.

Many conservative assumptions are made when calculating DRLs. Commonly, if the consumption of potentially contaminated food is involved, it is assumed that the food is produced, prepared and consumed in the manner leading to the highest concentrations of radioactivity. For gaseous emissions of particulate activity and liquid emissions of gross β^--γ, the DRL is selected from a list of DRLs for twenty or more radionuclides that could be expected in those emissions (a group). The smallest (i.e., most restrictive) DRL is selected from each group, and all radionuclides in the group are treated as emissions of this most-restrictive radionuclide. In practice, the emissions of this most-restrictive radionuclide might be a small fraction of the total emissions in the group. A similar treatment is used for the radioiodine group, which uses the DRL for iodine-131. The DRLs for other radioiodines should be much larger, because emissions are less significant from a public-dose perspective. All radioiodines emitted, however, are treated as iodine-131, leading to an overestimate of public dose.

The DRLs for the Point Lepreau, Gentilly 2 and Wolsong 1 NGS's are summarized in Table 8 (NB Power 1976; R. Blagoeva 1995;, KEPCO 1991). Similar values are available for Embalse, but these values and the assumptions used in deriving them were not available to the authors for publication. The Canadian regulator has yet to endorse a DRL for carbon-14 emissions from Point Lepreau, so the value listed in Table 8 should be treated as an estimate only.

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[8] For comparison, the global-average dose to an individual due to natural background radiation is 2.4 mSv/a; the local-average dose may be significantly higher. The dose from a single medical chest X-ray is approximately 70 mSv (UNSCEAR 1993).

Table 8 – Derived Release Levels for Airborne and Waterborne Effluents at CANDU 6 Stations

Type of Release	Derived Release Limit (GBq/a)		
	Point Lepreau[a]	Gentilly 2[a]	Wolsong 1[a]
Airborne			
tritium	1.6×10^{9}	4.4×10^{8}	3.9×10^{8}
carbon-14	1.6×10^{7}	8.8×10^{5}	–
noble gases[b]	3.4×10^{8}	1.7×10^{8}	8.3×10^{7}
radioiodine	2.7×10^{3}	1.3×10^{3}	7.8×10^{2}
particulate	7.8×10^{3}	1.9×10^{3}	1.9×10^{3}
Waterborne			
tritium	4.7×10^{10}	1.2×10^{9}	1.8×10^{8}
gross β^{-}-γ	1.1×10^{5}	5.3×10^{3}	9.7×10^{3}
carbon-14	–	1.0×10^{5}	–

a Based on a dose limit to a member of the public-critical group of 5 mSv/a.

b GBq·MeV/a

In Table 8, the DRLs in any particular group show large variations between the different sites. This reflects geographic, atmospheric and population-distribution variations between the sites.

The DRLs for liquid releases from the Point Lepreau site are larger than those for the other sites listed in Table 8. These DRLs were based on the Canadian public-dose limit of 5 mSv/a, and the operator has demonstrated a dilution factor of twenty in the coastal waters diluting liquid effluents. The Wolsong 1 DRLs make no allowance for dilution in its coastal waters. The Gentilly 2 DRLs take account of a critical group that drinks water taken from the Saint Lawrence river 3 km downstream of the station, a location that affords a dilution factor for liquid releases of ten.

All sites take account of the proximity of a local population. For the Wolsong 1 site there is a village immediately outside the plant property. For Gentilly 2 there is a farm nearby. In contrast, the Point Lepreau site is relatively remote. As a result, the DRLs for Point Lepreau tend to be larger than those for other sites.

The theoretical-maximum fraction of the annual dose limit received by an individual in any critical group can be estimated from the releases and DRLs. The quotient of the annual release of activity and the DRL in a release group multiplied by the public-dose limit gives an estimate of the maximum dose from that release group. The sum of these doses gives the absolute-

maximum dose to a member of the public.[9] These estimates for the years 1984 to 1994 are listed in Tables 9 to 11 for Point Lepreau, Gentilly 2 and Wolsong 1. In these tables, the DRL for iodine-131 was considered to deliver a dose of 15 mSv, the annual dose limit for the thyroid (NB Power 1976), to the thyroid of a child. The whole-body dose listed in Tables 9 to 11 was calculated using the weighting factor of 0.03 for the thyroid (ICRP 1977). Also included are these same dose estimates relative to the global-average background dose to an individual, and the dose attributable to a single chest X-ray for medical purposes.

Tables 9 to 11 show that the total, maximum annual exposures to any individual member of the public from the three stations considered are very small. The largest dose was 0.036 mSv for Gentilly 2 in 1994, which was approximately 1.5% of the dose due to natural background (2.4 mSv/a (UNSCEAR 1993)). The actual doses were undoubtedly far below these maximum doses, reflecting the dose-estimation method used and assumptions made in calculating DRLs.

Table 9 – Estimated Maximum Exposures to Critical Group From Gaseous and Liquid Effluents From Point Lepreau

Year	Maximum Annual Exposures (μSv/a) to Individual in Critical Group							Total Maximum Exposures		
	Airborne Effluents					Waterborne Effluents		μSv/a	Fraction of*	
	tritium	carbon-14	noble gases	radio-iodine	parti-culate	tritium	gross β-γ		Natural Back-ground	Single Chest X-ray
1984	0.21	–	0.0	–	–	0.003	0.59	0.81	0.03%	1.2%
1985	0.34	–	0.015	0.0078	0.0	0.003	0.07	0.44	0.02%	0.6%
1986	0.63	–	0.074	0.0537	0.0	0.012	0.07	0.84	0.03%	1.2%
1987	0.69	0.10	0.001	0.0	0.0	0.010	0.02	0.82	0.03%	1.2%
1988	0.69	0.11	0.004	0.0926	0.0	0.013	0.07	0.98	0.04%	1.4%
1989	0.66	0.13	0.000	0.0096	0.0	0.034	0.05	0.88	0.04%	1.3%
1990	0.66	0.15	0.000	0.0	0.0	0.017	0.09	0.91	0.04%	1.3%
1991	0.53	0.11	0.191	0.0296	0.0	0.012	0.18	1.05	0.04%	1.5%
1992	1.13	0.23	0.162	0.0030	0.003	0.034	0.09	1.64	0.07%	2.3%
1993	1.95	0.18	0.070	0.0004	0.001	0.052	3.00	5.25	0.22%	7.5%
1994	1.73	0.06	–	0.0165	–	0.028	–	1.84	0.08%	2.6%

* Natural background 2 400 μSv/a, Single chest X-ray 70 μSv (UNSCEAR 1993).

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[9] In reality, this estimated maximum dose is high for two reasons: the public-critical group for one radionuclide or emission pathway may not be the same segment of the local population as the public-critical group for a different radionuclide or pathway, and the conservative assumptions made in calculating DRLs.

Table 10 – Estimated Maximum Exposures to Critical Group From Gaseous and Liquid Effluents From Gentilly 2

Year	Maximum Annual Exposures (μSv/a) to Individual in Critical Group							Total Maximum Exposures		
	Airborne Effluents					Waterborne Effluents		μSv/a	Fraction of*	
	tritium	carbon-14	noble gases	radio-iodine	parti-culate	tritium	gross β-γ		Natural Back-ground	Single Chest X-ray
1984	0.56	–	0.76	–	–	0.06	6.42	7.79	0.32%	11.1%
1985	0.60	–	3.50	–	0.00	0.11	5.47	9.69	0.40%	13.8%
1986	1.55	–	1.56	–	0.00	0.17	2.74	6.01	0.25%	8.6%
1987	1.40	–	1.44	0.77	0.00	0.33	10.38	14.32	0.60%	20.5%
1988	2.22	–	2.35	0.00	0.00	0.24	2.17	6.98	0.29%	10.0%
1989	1.56	–	2.15	0.00	0.00	0.20	1.04	4.94	0.21%	7.1%
1990	2.58	–	1.76	0.00	0.00	0.68	3.96	8.99	0.37%	12.8%
1991	3.75	2.56	1.59	0.01	0.00	1.03	15.09	24.03	1.00%	34.3%
1992	3.66	0.04	0.97	0.00	0.08	1.10	4.25	10.09	0.42%	14.4%
1993	2.27	2.73	2.03	–	0.13	0.87	26.04	34.07	1.42%	48.7%
1994	2.93	16.56	1.74	0.00	0.18	0.98	13.96	36.35	1.51%	51.9%

* Natural background 2 400 μSv/a, Single chest X-ray 70 μSv (UNSCEAR 1993).

Table 11 – Estimated Maximum Exposures to Critical Group From Gaseous and Liquid Effluents From Wolsong 1

Year	Maximum Annual Exposures (μSv/a) to Individual in Critical Group							Total Maximum Exposures		
	Airborne Effluents					Waterborne Effluents		μSv/a	Fraction of*	
	tritium	carbon-14	noble gases	radio-iodine	parti-culate	tritium	gross β-γ		Natural Back-ground	Single Chest X-ray
1984	1.28	–	1.08	0.24	0.21	0.18	2.84	5.83	0.24%	8.3%
1985	1.15	–	1.02	–	0.02	0.33	0.52	3.04	0.13%	4.3%
1986	3.10	–	0.60	–	0.00	1.03	1.34	6.07	0.25%	8.7%
1987	4.01	–	0.72	0.00	0.00	1.39	0.77	6.90	0.29%	9.9%
1988	4.08	–	0.36	0.00	0.00	2.22	0.62	7.28	0.30%	10.4%
1989	2.90	–	0.30	0.00	0.02	1.67	0.16	5.04	0.21%	7.2%
1990	2.96	–	0.48	0.00	0.00	1.44	0.10	4.99	0.21%	7.1%
1991	3.29	–	0.36	0.00	0.00	2.58	0.10	6.34	0.26%	9.1%
1992	4.97	–	0.30	0.00	0.00	1.17	0.15	6.60	0.27%	9.4%
1993	2.71	–	0.72	–	–	–	0.06	3.48	0.15%	5.0%
1994	–	–	–	–	–	–	–	–	–	–

* Natural background 2 400 μSv/a, Single chest X-ray 70 μSv (UNSCEAR 1993).

6.2.2 Exposures to Global Population

In addition to members of the critical groups located near a nuclear site, doses to the global population can be estimated. This requires that the pathways analysis be extended to account for the sum of the doses to the local, regional and global populations. This is a challenging task, requiring many assumptions concerning dietary habits, reactor location, etc. The United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) has for several years been working on this problem, and has published several reports estimating the doses involved. The most recent report (UNSCEAR-93) was published in 1993 (UNSCEAR 1993).

UNSCEAR-93 presents estimates of the population doses resulting from the releases of airborne and liquid effluents from all power reactors. These doses were calculated as local/regional doses due to reactor operation, plus doses due to globally dispersed radionuclides, the sum of these giving the total, global dose. Doses were reported by reactor type, with CANDU 6 reactors being grouped with heavy-water reactors (HWRs). These doses are summarized in Table 12. For the five-year period ending in 1989, UNSCEAR assigned a normalized local/regional dose to the twenty-six operating HWRs equivalent to an average individual dose of 46 nSv/a, or 0.002% of natural background. For the same period, UNSCEAR assigned a total global dose equivalent to an average individual dose of 444 nSv/a, or 0.009% of natural background. Unfortunately, these doses were exceedingly high, due to the way UNSCEAR handled missing data.

UNSCEAR's objective is to provide a reasonably accurate estimate of the effects of radiation on the environment. To do this, it must rely on various sources of data and adopt an acceptable approach for dealing with missing data (stations not reporting the emission of identified radionuclides). For power reactors, UNSCEAR has grouped all reactors into broad types. If any station within a type reports the emission of a radionuclide, UNSCEAR assumes that all stations within that type emit that radionuclide in proportion to the station's power output. Stations not reporting a specific radionuclide are therefore assigned the normalized average emission of reporting stations. This is a very conservative approach, and can lead to dramatic overprediction of emissions. This occurs when a few reactors, particularly unusually small reactors or old designs, report a radionuclide while the majority of reactors within that type do not, or if radionuclide emissions increase during maintenance outages when no power is being produced. This was the case for carbon-14 from HWRs reported in UNSCEAR-93.

CANDU 6 reactors have always used carbon dioxide for the annulus gas, as have all Ontario Hydro CANDU reactors after Pickering NGS-A. Pickering NGS-A originally used nitrogen for the annulus gas, which led to elevated carbon-14 production and emission rates. Pickering NGS-A converted to carbon dioxide during the 1980s, but the carbon-14 emission rate remained elevated as residual material was purged from the system (see section 2.2.2). Thus, carbon-14 emissions from Pickering NGS-A have historically been elevated relative to CANDU 6 reactors, particularly during the significant maintenance outages covered by UNSCEAR-93. Since carbon-14 emission rates for the majority of CANDU reactors were not reported to UNSCEAR for use in UNSCEAR-93, however, UNSCEAR's methodology essentially assigned Pickering NGS-A's carbon-14 emissions to all HWRs. This led to an approximate threefold overestimation of the emission rate of carbon-14 from HWRs. Focusing specifically on CANDU 6 reactors, the over-estimate was eightfold (Allsop 1995).

Extrapolating UNSCEAR's approach, the UNSCEAR-93 results for HWRs can be corrected by assigning the reported CANDU 6-reactor emissions of carbon-14 to the non-reporting stations on an electric-power-produced (GW_e) basis.[10] Using this method, the UNSCEAR model and data predict significantly lower doses for HWRs (Table 12). For the five-year period ending in 1989, the corrected local/regional dose is equivalent to an average individual dose of 34 nSv/a, or 0.0006% of natural background. The total global dose is equivalent to an average individual dose of 156 nSv/a, or 0.003% of natural background. For the CANDU 6 reactors, the doses are even lower. The local/regional dose is equivalent to an average individual dose of 9.1 nSv/a, or 0.0004% of natural background. The total global dose is equivalent to an average individual dose of 20 nSv/a, or 0.0008% of natural background.

Table 12 – Public Dose using the UNSCEAR-93 Model and Data

Dose Component	Normalized	Total Integral	Average Individual	
	p-Sv/[GW_e·a]	p-Sv/a	nSv/a	Fraction of Background
HWR Doses from UNSCEAR-93				
Local/Regional Component	14	138	46	0.0009%
Globally Dispersed Radionuclides	409	3 968	398	0.0076%
Total Global Dose	423	4 106	444	0.0085%
Corrected HWR Doses using UNSCEAR-93 data and model				
Local/Regional Component	10	101	34	0.0006%
Globally Dispersed Radionuclides	125	1 210	122	0.0024%
Total Global Dose	135	1 311	156	0.0030%
CANDU 6 Doses Derived from UNSCEAR-93 data and model				
Local/Regional Component	4	9.1	9.1	0.0004%
Globally Dispersed Radionuclides	51	109	11	0.0005%
Total Global Dose	55	118	20	0.0008%

Great caution must be used in interpreting the global doses derived in Table 12. The UNSCEAR dose model assumes a reactor site that is not necessarily typical of a CANDU reactor. The UNSCEAR model assumes that the local/regional population is 250 million within 2 000 km of the site, with 3.1 million people within 50 km. The majority of CANDU reactors are in locations where the local and regional populations are lower. On average, the actual local/regional dose from CANDU reactors will therefore be lower than predicted by UNSCEAR. Further, there are

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10 There is a much smaller tritium correction that can also be applied.

statistical problems with the UNSCEAR method that affect the reliability of its estimates (Allsop 1995). As a result of these two factors, the doses calculated for HWRs and CANDU 6 reactors using the UNSCEAR method are undoubtedly high. For reactor types where the majority of stations are located in areas more closely matching the UNSCEAR model — such as BWRs,[11] PWRs[12] and GCRs[13] — the UNSCEAR-calculated doses may be more accurate.

6.2.3 Relative Importance of Radionuclides

The data from Tables 8 through 11 provide information that can be used to establish the relative importance of each radionuclide class to public-critical dose. From Table 8, the dose factor for each radionuclide (the public dose resulting from the release of unit of activity) can be calculated. A similar calculation can be performed using the local/regional component of the pathways model employed in UNSCEAR-93. These dose factors are summarized in Table 13, while Table 14 lists the normalized dose factors (normalized to the radioiodine factor for each station or the local/regional component). These factors show that airborne releases of radioiodines and particulates have the greatest potential to deliver public-critical doses at each station, but that particulates and airborne carbon-14 have greater local/regional potential. In addition, carbon-14 and tritium emissions are weighed much more heavily in the local/regional calculation than in the public-critical calculations, while waterborne β^--γ and airborne radioiodines are weighed more lightly. These factors are taken into consideration in designing CANDU stations.

Due to the design features of the CANDU 6 reactor and operator diligence, the radionuclides having the greatest potential to deliver public dose do not necessarily deliver the greatest dose. Table 15 lists the average calculated doses from 1984 to 1994 to members of the public-critical groups at each station, and the local/regional doses derived from UNSCEAR-93. Table 16 lists the same data, but normalized to the radioiodine dose from each station. While radioiodine had the greatest potential for delivering public-critical dose, airborne and waterborne tritium, airborne noble gases and waterborne β^--γ were more significant during operation. For local/regional dose, airborne tritium and carbon-14 were the dominant dose contributors, with waterborne tritium also being important.

In conclusion, the relative importance of each radionuclide depends on the population group being considered, the geographic and atmospheric features of the reactor site, and the type of scenario being analyzed. If the risks associated with unplanned-events or accidents are being analyzed, the potential of each species to deliver dose must be considered. For licensing purposes, the public-critical dose is usually considered the important factor. Local/regional and global doses, such as those presented in UNSCEAR-93, are primarily of use in comparing the overall impact of nuclear power to that of other energy or radiation sources. Generally, local/regional and global doses, such as those listed in Table 12, are so small as to be of little use

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[11] Boiling Water Reactors.

[12] Pressurized Water Reactors.

[13] Gas Cooled Reactors.

in assessing the impact of CANDU 6 reactors on the public.[14] For this reason, the public-critical dose is also the more conservative measure of the maximum impact of a CANDU 6 reactor on the public. Tritium, noble gases and waterborne ß⁻-γ were therefore the most significant radionuclides during normal operation at the existing CANDU 6 stations.

Table 13 – Derived Dose Factors for Effluents from CANDU 6 Stations

Type of Release	Dose Factors			
	Public-Critical (μSv/GBq·a)			Local/Regional[a]
	Point Lepreau	Gentilly 2	Wolsong 1	(p-Sv/GBq·a)
Airborne				
tritium	3.1×10^{-6}	1.1×10^{-5}	1.3×10^{-5}	1.1×10^{-5}
carbon-14	3.1×10^{-4}	5.7×10^{-3}	–	1.8×10^{-3}
noble gases[b]	1.5×10^{-5}	2.9×10^{-5}	6.0×10^{-5}	1.2×10^{-7}
radioiodine	1.9	3.8	6.4	5.1×10^{-4}
particulate	6.4×10^{-1}	2.6	2.6	5.4×10^{-3}
Waterborne				
tritium	1.1×10^{-7}	4.2×10^{-6}	2.8×10^{-5}	8.1×10^{-7}
gross ß⁻-γ	4.5×10^{-2}	9.4×10^{-1}	5.2×10^{-1}	2.0×10^{-5}
carbon-14	–	5.0×10^{-2}	–	–

a From UNSCEAR 1993. b μSv/GBq·MeV·a

[14] It is difficult to quantify the impact on the public of doses approximately one one-millionth of natural background.

Table 14 – Normalized Dose Factors for Effluents from CANDU 6 Stations

Type of Release	Normalized Dose Factors			
	Public-Critical			Local/Regional[a]
	Point Lepreau	Gentilly 2	Wolsong 1	
Airborne				
tritium	1.7×10^{-6}	3.0×10^{-6}	2.0×10^{-6}	2.2×10^{-2}
carbon-14	1.7×10^{-4}	1.5×10^{-3}	–	3.5
noble gases	7.9×10^{-6}	7.6×10^{-6}	9.4×10^{-6}	2.4×10^{-4}
radioiodine	1	1	1	1
particulate	0.35	0.68	0.41	11
Waterborne				
tritium	5.7×10^{-8}	1.1×10^{-6}	4.3×10^{-6}	1.6×10^{-3}
gross $\beta^-\gamma$	2.5×10^{-2}	0.25	8.0×10^{-2}	3.9×10^{-2}
carbon-14	–	1.3×10^{-2}	–	–

a From UNSCEAR 1993.

Table 15 – Maximum Doses For Effluents at CANDU 6 Stations

Type of Release	Average Doses to Specified Group			
	Public-Critical Groups (μSv/a)			Local/Regional[a]
	Point Lepreau	Gentilly 2	Wolsong 1	(p-Sv/a)
Airborne				
tritium	0.84	2.1	3.0	6.4
carbon-14	0.13	5.5	–	2.3
noble gases	0.05	1.8	0.60	0.050
radioiodine	0.02	0.11	0.03	0.00044
particulate	0.00037	0.04	0.03	0.0018
Waterborne				
tritium	0.02	0.52	1.34	0.36
gross $\beta^-\gamma$	0.42	8.32	0.67	0.00025
carbon-14	–	–	–	–

a Derived from UNSCEAR 1993.

Table 16 – Relative Doses For Effluents at CANDU 6 Stations

Type of Release	Relative Doses to Specified Group			
	Public-Critical Groups			Local/Regional[a]
	Point Lepreau	Gentilly 2	Wolsong 1	
Airborne				
tritium	39	19	89	15 000
carbon-14	6.3	49	–	5 200
noble gases	2.4	16	17	115
radioiodine	1	1	1	1
particulate	0.017	0.36	0.77	4.0
Waterborne				
tritium	0.92	4.7	39	833
gross ß⁻-γ	20	75	19	0.58
carbon-14	–	–	–	–

a Derived from UNSCEAR 1993.

7. CONCLUSIONS

This report has reviewed the sources of gaseous and liquid radioactive emissions from CANDU 6 reactors. It has discussed how those sources escape from the plant to become part of the radioactive emissions. The report has shown that, from a safety perspective, the doses to individual members of the public resulting from these emissions have been negligible.

The report has also reviewed the data in UNSCEAR-93, and shown that this UNSCEAR report exaggerates the releases of carbon-14 from HWRs in general, and from the CANDU 6 reactor in particular. The releases of other radionuclides assigned to HWRs in UNSCEAR-93 were also higher than those typical of a CANDU 6 reactor, reflecting the grouping of the CANDU 6 reactor with other designs.

For the operating CANDU 6 reactors, the report has shown that in the gaseous pathway the releases of tritium and noble gases have been the most radiologically significant. In the liquid pathway, the releases of gross ß⁻-γ have been more significant. In contrast, radioiodine and particulate emissions have the greatest potential to deliver dose to public-critical groups.

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