8 - INTERNAL RADIATION HAZARDS

INTRODUCTION

Most radionuclides will give you much more radiation dose if they can somehow enter your body than they would if they remained outside. In this chapter we will look at how radionuclides enter, move around in, and leave the body. Then we'll discuss the internal hazards at Point Lepreau; namely tritiated water, radioiodines and airborne radioactive dusts. At the end of this chapter, you should be able to estimate the radiation dose you would receive when exposed to internal hazards. This ability is essential for doing radiation work.

INTERNAL RADIATION HAZARDS

Among the first people to suffer injury and death from the effects of internal sources were the radium dial painters mentioned in Chapter 3 (page 116). Relatively large quantities of radium entered their gastrointestinal tracts, some radium was absorbed in their bodies, and most of this was deposited in their bones. After death it was shown that some skeletons contained many micrograms of radium.

When dealing with external exposure, protective means are fairly straightforward; usually the radiation level can be measured easily with an instrument, and the exposure can be stopped at any time by walking away from the source. However, the hazard presented by internal sources demands more elaborate precautions for the following reasons:

1. Only some of the radiation emitted by external sources is directed towards the body; the rest will not be able to interact with it. However, if such sources are taken into the body, all of the radiation emitted is capable of interacting with the body.

2. The chemical properties of some radionuclides cause them to concentrate in certain body organs or tissues rather than be spread throughout the body. This means that all the energy of alpha and beta emissions and part of the energy of the gamma emissions are absorbed in these tissues, causing them to receive much more radiation dose than the body as a whole.

   Reasons (1) and (2) explain why the dose rate from sources within the body is much greater than from the same sources outside the body.

3. Internal sources irradiate the body 24 hours a day, seven days a week, until they have been eliminated from the body by excretion and decay. In other words, you can't walk away from them as you can with external sources.

4. While some radionuclides are eliminated fairly rapidly, there are others that remain in the body for years. And in many cases it is difficult to increase their rate of elimination from the body.

5. Finally, we tend to be more cautious in our dealings with internal hazards, because once the radionuclides have entered the body, it is often difficult to estimate the dose they will deliver.

ENTRY OF RADIONUCLIDES INTO THE BODY

Radioactive materials may occur in many physical or chemical forms just as other materials do. They may appear as solids, powders, dusts, liquids, gases, vapours or solutions. Internal contamination can result from the careless handling of such radioactive material. It may enter the body in three different ways:

1) Inhalation (breathing it),
2) Ingestion (eating it),
3) Absorption through the unbroken skin or through wounds.

In the nuclear power industry, inhalation is generally considered to be by far the most likely route for entry of toxic materials into the body. Inhaled material cleared from the lungs often enters the gastrointestinal tract (GI tract) and then a secondary ingestion type of exposure occurs.

The behaviour of inhaled radionuclides in the lungs depends on whether they are soluble or insoluble in lung fluids. The term used by the ICRP to describe solubility in body fluids is " transportability". Transportable radionuclides will readily enter the blood stream from the lung and deposit in body organs. Elimination from the body is mainly through urinary excretion.

For non-transportable radionuclides, the lung is usually the target organ because it retains them for a long time. A small and uncertain fraction is eliminated in the urine, which means that urine analysis is unreliable as a useful bioassay technique for these radionuclides. The greater part of such non-transportable radionuclides is slowly expelled up the respiratory passage by ciliary action, subsequently swallowed and excreted in the faeces. (Ciliary action is the vibrating motion of small hair-like strands in the respiratory passages, which causes all kinds of slime and nasties to be propelled back to the mouth.)

The amount of radionuclide taken up by the body depends on the magnitude of the intake, solubility in body fluids (which in turn depends on the particular chemical form of the radionuclide), and whether the intake is via inhalation, ingestion or absorption through the skin.

Transportable materials which are inhaled or ingested go largely to the blood stream, while non-transportable materials irradiate the respiratory tract or the gastrointestinal tract while they are present.

DISTRIBUTION OF RADIONUCLIDES IN THE BODY

A large percentage of radionuclides that enter the body are eliminated in the first few days. However, a portion will be absorbed in various organs depending on the type of radionuclide. The body deals with elements and compounds on a chemical basis. For example, normal inactive iodine (I-127) concentrates in the thyroid gland. If radioactive I-131 is present as well, the body cannot differentiate between the two isotopes that are chemically identical, and the active I-131 will also concentrate in the thyroid.

Some elements are so closely chemically related, that the body cannot always differentiate effectively between two different elements. For example, chemists say that calcium, strontium, barium and radium are in the same group. Calcium present in the body is largely deposited in the bone, and any radioisotopes of strontium, barium and radium that enter the body will therefore also collect to a considerable extent in the bone. Such radioisotopes (called **bone-seekers)** are excreted at a very slow rate once they have been deposited in the bone. If their radioactive half-life is long, they may therefore irradiate the sensitive bone marrow, as well as the bone, for many years.

The tissue (which may be a body organ) in which radiation is absorbed is known as TARGET TISSUE.

For exposures to radioactive iodine, the main target tissue would be the thyroid gland; for inhaled strontium it is the lung, but for ingested strontium it is the bone.

ELIMINATION OF RADIONUCLIDES FROM THE BODY

In addition to the tendency for a particular element to be taken up by a particular organ or tissue, the main consideration in determining the hazard of a given radioisotope inside the body is the total radiation dose delivered to the target tissue. The most important factors determining this dose are:

1) The amount of radioactive material deposited,
2) The length of time for which it is effective in the body,
3) The type and energy of the radiations emitted.

The time depends on two factors: one is the **radioactive half-life, T_r**, the other is the **biological half-life, T_b**.

The BIOLOGICAL HALF-LIFE is the time taken for the amount of a particular element in the body to decrease to half its initial value due to elimination by biological processes alone.

The biological half-life depends on the rate at which the body normally uses a particular compound of an element. The combination of the radioactive half-life and the biological half-life gives rise to the **effective half-life, T_e**.

The EFFECTIVE HALF-LIFE is the time taken for the amount of a specified radionuclide in the body to decrease to half its initial value as a result of both radioactive decay and biological elimination.

The effective half-life, T_e, is given by the equations

$$\frac{1}{T_e} = \frac{1}{T_r} + \frac{1}{T_b} \qquad \text{or} \qquad Te = \frac{T_r T_b}{T_r + T_b}$$

Serious internal hazards are presented by those radionuclides that have long effective half-lives, such as Ra-226 and Pu-239 with effective half-lives of 45 and 100 years respectively. Once deposited in bone, they remain there essentially unchanged in amount during the lifetime of the individual. The continued action of the emitted alpha particles can cause significant injury, because they deposit their energy over a period of years in a limited region. This was the case with the radium dial painters.

PHYSICAL FORM OF INTERNAL CONTAMINANTS

Particulates

Many of the fission products and activation products that are encountered as air contaminants in a nuclear plant are present as particulates. The radionuclides often collect on small dust particles of other materials, and these particles float about in the air acting as carriers for the radioactivity, which may then be inhaled. If the particulate is collected on a filter paper, so is the radioactivity.

Gases

The radioisotopes of the noble gases are, of course, present as gases. Some of these are argon-41, xenon-133, xenon-135 and krypton-88. The noble gases do not combine with other elements and, fortunately, the body has no use for them. Therefore, they do not become concentrated in the body. The noble gases are considered to be an external radiation hazard rather than an internal one.

Vapours

The radioiodines are usually present in the form of vapours, although some of the iodine may be attached to dust particles. These radionuclides are readily absorbed by the body and are mainly concentrated in one organ, the thyroid. The noble gases and the iodines are the

radioelements which escape most readily from defective fuel elements. Tritium vapour is another internal radiation hazard of considerable importance in heavy water reactors.

Now that you've got a general overview of internal hazards, we'll go on to discuss the various aspects in more detail.

ANNUAL LIMIT ON INTAKE

For internal hazards, we first should distinguish between **intake** and **uptake,** words that are often used interchangeably but shouldn't be.

INTAKE is what you take in.
UPTAKE is what you keep.

Think of a smoker. With each suck on his cigarette, he drags all kinds of crud into his lungs. That's the intake. Fortunately most of this is exhaled, and only a part of the intake remains behind in the lungs as a deposit. That's the uptake by the lung. The nicotine in the cigarette smoke is transportable; it enters the blood stream and is carried to the brain. The tar and other crud is non-transportable and stays in the lungs to be partially cleared by ciliary action and " smoker's cough".

For many radionuclides, the ICRP has calculated the relationship between intake (namely the activity taken into the body) and the dose to the target tissue resulting from that intake. In order to do this, ICRP used a "standard man" in its mathematical models. He is known as **Reference Man,** and is described in great detail in ICRP Publication 23, which contains a lot of interesting anatomical and physiological data. (For example, the average guy's toenails grow 0.25 mm per week, and he has 1.8 m^2 of skin weighing 2.6 kg.)

ICRP has calculated **Annual Limits on Intake** for radionuclides. These differ widely for the various radionuclides that were treated by ICRP.

The ANNUAL LIMIT ON INTAKE (ALI) is the activity of a radionuclide that taken in by itself would commit a person, represented by Reference Man, to 20 mSv of weighted dose, H_W.

The idea behind these ALIs is that in any year, your intake must not exceed that amount that would cause you to receive a weighted dose greater than 20 mSv. Let's use a couple of examples to be sure that you understand what this means.

For I-131, for instance, ICRP has calculated that if you inhale 1E6 Bq, you will receive a thyroid dose of H_T = 400 mSv. The weighted dose equivalent to this is $H_W = H_T w_T$ = 400 x 0.05 = 20 mSv. The ALI for I-131 is therefore 1E6 Bq.

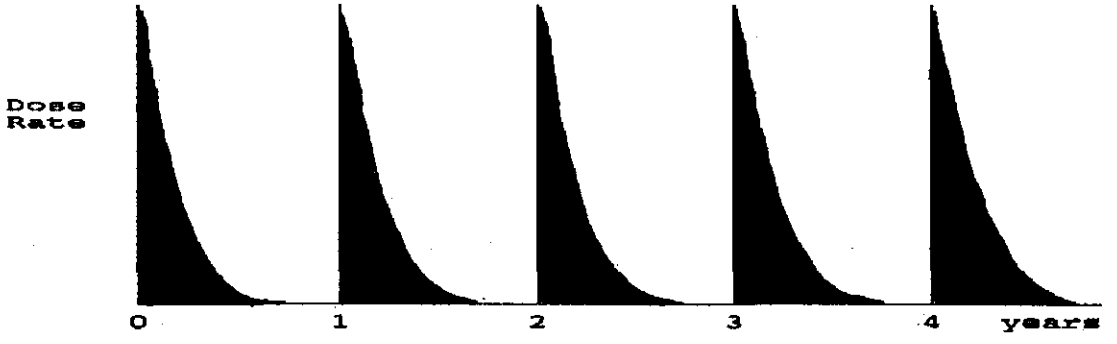
It doesn't matter whether you inhale this in a short time (say an hour) or over a period of one year. In both cases the activity taken into the body is the same, and you will receive a thyroid dose of 400 mSv.

Let's take another example. For Sr-90, the target tissue is the lung which has a weighting factor of 0.12. A tissue dose, H_T, of 167 mSv to the lung will correspond to a weighted dose of $H_W = 167 \times 0.12 = 20$ mSv. Therefore, the ALI for Sr-90 is 6E4 Bq, because this intake would cause you to receive a lung dose of 167 mSv, and hence a weighted dose of 20 mSv.

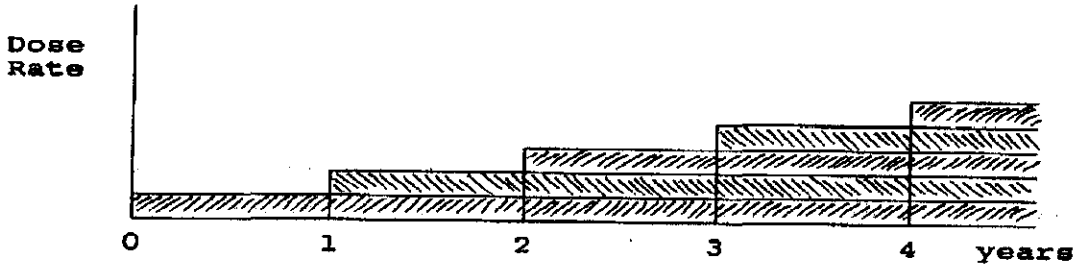
Equally important is the time span over which the doses are delivered. Let's use I-131 and Pu-239 as examples. The effective half-life of I-131 is about 8 days. Its target tissue is the thyroid with a weighting factor of 0.05. The effective half-life of Pu-239 is about 100 years. Its target tissue is bone with a weighting factor of 0.01.

An intake of 1 ALI of I-131 will deliver 400 mSv to the thyroid, and hence 20 mSv of weighted dose, within six weeks or so. 1 ALI of plutonium-239 will deliver 2,000 mSv to bone, and will also result in 20 mSv of weighted dose. Neither of these examples is at all likely in practice — certainly not the plutonium — but I'm trying to get you to get a grip on the principles here.

The 20 mSv weighted dose from Pu-239 will be spread over a very long time. Fig. 8.1 shows how the dose from an intake of 1 ALI every year would be delivered. The important thing to realize is that the annual dose limit will never be exceeded, even after 50 years of occupational exposure in the case of Pu-239, provided that you do not take in more than 1 ALI of a radionuclide each year. This of course assumes that this is the only source of exposure.



I-131: Dose from 1 ALI taken in at the start of each year (H_T = 400 mSv; H_W = mSv) is delivered in the year of intake, because the half-life is so short.



Pu-239: Dose from 1 ALI taken in at the start of each year (H_T = 2,000 mSv; H_W = 20 mSv) is delivered over many years, because the half-life is very long.

Fig. 8.1. Doses From 1 ALI of Different Radionuclides Will Be Delivered Over Different Times

The ICRP actually lists two values of the ALI for each radionuclide. One is for inhalation and the other is for ingestion. Since ingested materials are normally cleared from the body much faster than inhaled materials, the ALI for the ingestion pathway is usually larger than the ALI for the inhalation pathway.

The ingestion pathway is not of much concern to us in practice, because food is not permitted in active or potentially active areas, and every effort is made to keep drinking water absolutely free of contamination. All ALIs discussed in this chapter, and indeed throughout the whole course, refer to values that apply to the inhalation pathway.

THE DERIVED AIR CONCENTRATION

The permissible limit for inhalation of a radionuclide is the appropriate ALI. This is quoted in units of Bq.

For practical convenience, we use the ALI to calculate the Derived Air Concentration (DAC).

The DERIVED AIR CONCENTRATION (DAC) for any radionuclide is that concentration in air (Bq/m^3) which, if you work in it for a year (50 weeks at 40 hours per week), will result in the ALI for inhalation.

The DAC is based on the breathing rate of Reference Man when he is engaged in "light activity" at work (ICRP's words, not NB Power's!). This breathing rate is 0.02 m^3 of air per minute.

If we assume that your breathing rate at work is pretty much the same as Reference Man's you will breathe

$$0.02 \frac{m^3}{min} \times 60 \frac{min}{hour} \times 2000 \frac{\text{hours of work}}{year}$$

$$= 2400 \text{ } m^3 \text{ per year at work.}$$

Therefore, if we divide the ALI by 2400 m^3, we will get the Derived Air Concentration. For example, for I-131 the ALI = 1E6 Bq. The corresponding DAC should be 1E6 Bq/2400 m^3 = 417 Bq/m^3. Since ICRP gives the ALIs to one significant figure only, we shall do the same with the DACs. In other words, we'll call the DAC for I-131 400 Bq/m^3.

The ICRP used to publish the DAC values along with the ALIs, but they don't do this any more; I guess they figure if you're smart enough to know what they mean, you're probably able to divide by 2400 as well.

Before we discuss the practical applications of ALIs and DACs, it is worth taking a look at the factors that influence the value of the ALI for any radionuclide.

The ALI for a radionuclide depends on all of the following:

1) Type of radiation emitted.
2) Energy of the radiation emitted (including that of any radioactive daughters).
3) The selective deposition in specific body tissues.
4) The effective half-life.

TABLE 8.1. ALIs OF TRITIUM AND IODINE-131

Radio-nuclide	Radiation Emitted	ALI (Bq)	Target Tissue	H_T From 1 ALI	H_W From 1 ALI
H-3 (DTO)	β only	1E9	whole body	20 mSv	20 mSv
I-131	β, γ	1E6	thyroid	400 mSv	20 mSv

Table 8.1 lists the ALIs of two important radionuclides. What the table tells us is that for an intake of 1E9 Bq of tritium (in water vapour form), you will get a whole-body dose of 20 mSv, and for an intake of 1E6 Bq of I-131 you will receive a weighted dose of 20 mSv (and a thyroid dose of 400 mSv).

Let's look at the four factors above one by one.

1) Tritium emits beta particles only, whereas I-131 emits beta and gamma. This would tend to make the ALI for I-131 smaller than that for tritium, because both radiations will be absorbed to deliver dose.

2) The energy of the beta particle emitted by tritium (E_{max} = 18 keV) is much less than that of the beta particle emitted by I-131 (E_{max} = 600 keV), and so each tritium decay will deliver only about 3% of the energy that is delivered by each I-131 disintegration. This again would tend to reduce the ALI for I-131. We have only addressed the beta energies here because the dose absorbed in the thyroid from the few I-131 gamma photons interacting with it is negligible when compared with the beta dose.

3) The tritium taken into the body is uniformly distributed among all soft tissues. Reference Man has 63 kg of soft tissue spread throughout the body (he weighs 70 kg); so any tritium taken up will be distributed over 63 kg of tissue. The whole body will be irradiated.

   The situation is quite different for radioiodine taken up by the body. About 30% of it ends up in the thyroid gland, and this is a small organ weighing only 20 g. As a result, once in the body, I-131 will deliver a very much larger dose to the thyroid than to the rest of the body. (If you have trouble with this, remember that absorbed dose is energy absorbed in unit mass of material. If the iodine were to be spread throughout the body, its energy would be absorbed in 70 kg, but since 30% of it sits in the thyroid, 30% of the energy is absorbed in only 20 g, thereby leading to a much larger thyroid dose.) Again, this will make the ALI for I-131 more restrictive than that for tritium.

4) Finally, consider the effective half-lives. These are 10 days for tritium and 7.5 days for I-131. The difference is not large enough to be significant. However, imagine that we have two radionuclides A and B that emit the same type of radiation at the same energy and both irradiate the whole body. In other words, we are making factors 1, 2, and 3 identical. Only the effective half-lives are different. A has an effective half-life Te = 10 days; for B it is 100 days. If the ALI for A = 1E8 Bq, what would you expect it to be for B?

   We know that an intake of 1E8 Bq of A will deliver a dose of 20 mSv before radionuclide A is completely eliminated from the body. Radionuclide B has an effective half-life 10 times longer, and so it will be delivering dose for 10 times as long. Therefore, we must restrict its intake to one tenth that of A if we don't want to exceed 20 mSv from B, i.e., the ALI for B is 1E7 Bq.

   What about the case of radionuclide C where the effective half-life is 100 years? (This is the case for plutonium in bone!) If you take in a quantity of Pu-239 this year, you, or rather your skeleton, will be receiving dose for many years after you have ceased to take an interest in these matters. The ALI for such long-lived nuclides is therefore set so that an intake corresponding to 1 ALI will cause the dose limit to be reached over the next 50 years.

   For example, the ALI for Pu-239 is 300 Bq. If you inhale 300 Bq of Pu-239, you will receive a weighted dose of 20 mSv within the next 50 years. Of course, you could be lucky and die next week, in which case you'd get only a very small fraction of this dose.

Table 8.2 lists the ALIs and DACs for some important radionuclides. It is important to remember that these ALIs and DACs apply to the inhalation pathway, which is the only one we are concerned with in practice. You don't need to remember these values, they are given only to show you the sort of range we are dealing with.

If you are ever in a position where for some reason or other you need to know the ALI or DAC of some specific radionuclide not listed above, ask Health Physics. They'll know where to find the answer.

TABLE 8.2. ALIs AND DACs FOR SOME IMPORTANT RADIONUCLIDES

Radionuclide	Radiation Emitted	ALI (Bq)	DAC (Bq/m^3)
Tritium (DTO)		1E9	3E5
Cobalt-60	,	4E5	2E2
Strontium-90		6E4	3E1
Zirconium-95	,	3E6	1E3
Iodine-131	,	1E6	4E2
Iodine-133	,	8E6	3E3
Iodine-135	,	4E7	2E4
Cesium-137	,	2E6	8E2

INTERNAL EXPOSURE AND COMMITTED DOSE

By now you must be wondering what the practical use of ALIs and DACs is, and if, indeed, there is one.

Well, there certainly is. Bear with me for a moment, and it should all drop into place.

Let us use tritium as an example. The ALI is 1E9 Bq. If you take in 1E9 Bq of tritium, you will receive a whole-body dose of 20 mSv. It doesn't matter whether you take in this amount of tritium in a short time (say a few hours) or in a long time (several months). You will still get the same whole-body dose of 20 mSv.

If you go back and look at the definition of the DAC on page 316, you will see that you could take in the ALI by working in a tritium concentration of 1 DAC for 2000 hours. Therefore, exposure to 1 DAC for 2000 h will result in a dose of 20 mSv. If you are exposed to 1 DAC for 1000 hours instead of 2000, you will obviously take in only half as much tritium, and your dose would then be 10 mSv instead of 20 mSv. If you are exposed to 1 DAC for only 1 hour, the whole-body dose resulting from this would be 20 mSv/2000 = 0.010 mSv = 10 μSv.

You have to appreciate that if you work in 1 DAC of tritium for 1 hour, you will not receive the 10 μSv you have coming to you in that hour, but over a period of the next several weeks. This is because the effective half-life is 10 days, and it takes a couple of months until the tritium has been pretty well eliminated. That's why we talk of a **committed dose.** By **exposing** yourself to tritium now, you are **committing** yourself to a dose to be delivered in the future.

Every DAC-h of tritium exposure gives you a committed dose of 10 μSv. Or, looking at it the other way, working in 1 DAC of tritium is equivalent to a committed dose rate of 10 μSv/hour. If you work for 3 hours in an area where the committed dose rate is 20 μSv/h (i.e., 2 DACs), your committed dose will be 3 h x 20 μSv/h = 60 μSv.

We needn't restrict this discussion to tritium. For any radionuclide, working in 1 DAC will result in a committed dose rate of 10 μSv/h. You'll find it useful to think of exposure to 1 DAC as causing the same committed dose in the future as exposure to 10 μSv/h right now.

Working in 1 DAC is equivalent to a committed dose rate of 10 μSv/h.

This concept gives us a useful method of estimating the committed dose **before** the exposure. It is only an estimate, because in practice we often cannot measure the committed dose rate all that accurately. However, it will give us important information for planning the work. For instance, should we wear respirators or a plastic suit? It may even be desirable to ventilate the area thoroughly before starting work there.

TRITIUM

Now that you're a big name in ALIs and DACs, we'll go on to discuss the radiation hazards presented by tritium. In CANDU reactors, tritium can contribute a significant portion of station dose. Historically, it has been responsible for 30 to 40% of the radiation dose received by nuclear station staff.

Let us just review what we know about tritium. It is the isotope of hydrogen with one proton and two neutrons. The symbol is ^{3}H, H-3 or just T, for short. It emits beta particles only (no gamma), and has a half-life of 12.3 years. The maximum beta energy is 18 keV, and the average beta energy is 6 keV.

A beta particle has to have an energy of at least 70 keV to be able to penetrate the dead surface layer of the skin. Therefore, tritium is not an external hazard — only when it is taken into the body will it be able to irradiate live tissue.

PRODUCTION OF TRITIUM

Tritium is produced wherever deuterium is exposed to neutrons, i.e., whenever the moderator and primary heat transport (PHT) heavy water is in the core. The reaction that produces the tritium is radiative capture:

$$ {}^{1}_{0}n + {}^{2}_{1}H \rightarrow {}^{3}_{1}H + \gamma $$

The rate of production of the tritium depends on the average thermal neutron flux to which the heavy water is exposed. The moderator water spends most of its time in the reactor and the PHT water spends most of its time outside the reactor. This is why tritium production in moderator water is much greater than in PHT water.

Fig. 8.2 shows how the tritium concentration in moderator and PHT water at Point Lepreau G.S. is expected to build up with time.

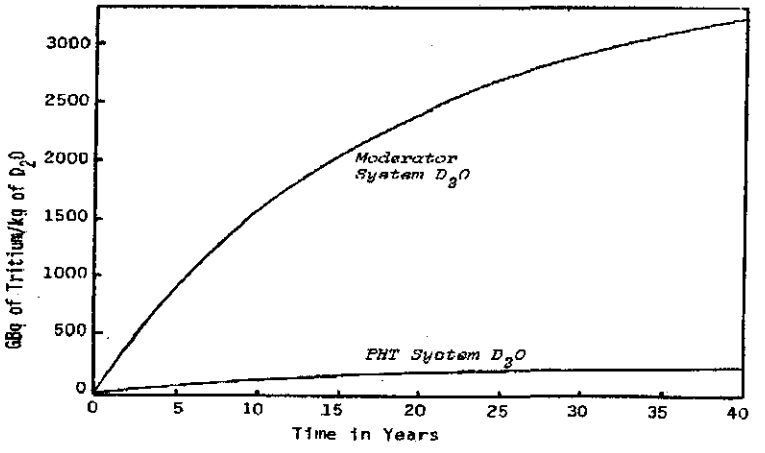


Fig. 8.2. Tritium Build-up at Point Lepreau G.S.

You can see from the curve that, as the reactor continues to operate, the tritium concentration increases relatively quickly at first, more slowly later on, and eventually — after a very long time — it levels out at a maximum concentration. This is called the **equilibrium concentration**, because here the rate at which tritium is being produced is equal to the rate at which it is decaying (the production rate is always constant, the decay rate is proportional to the amount of tritium present).

The tritium concentration is normally quoted in becquerels per kilogram of D_2O (Bq/kg). If Point Lepreau achieves a long-term capacity factor of 80%, the equilibrium concentration will be about 3.5 TBq/kg in the moderator system, and in the PHT system it will be about 1/30th of that.

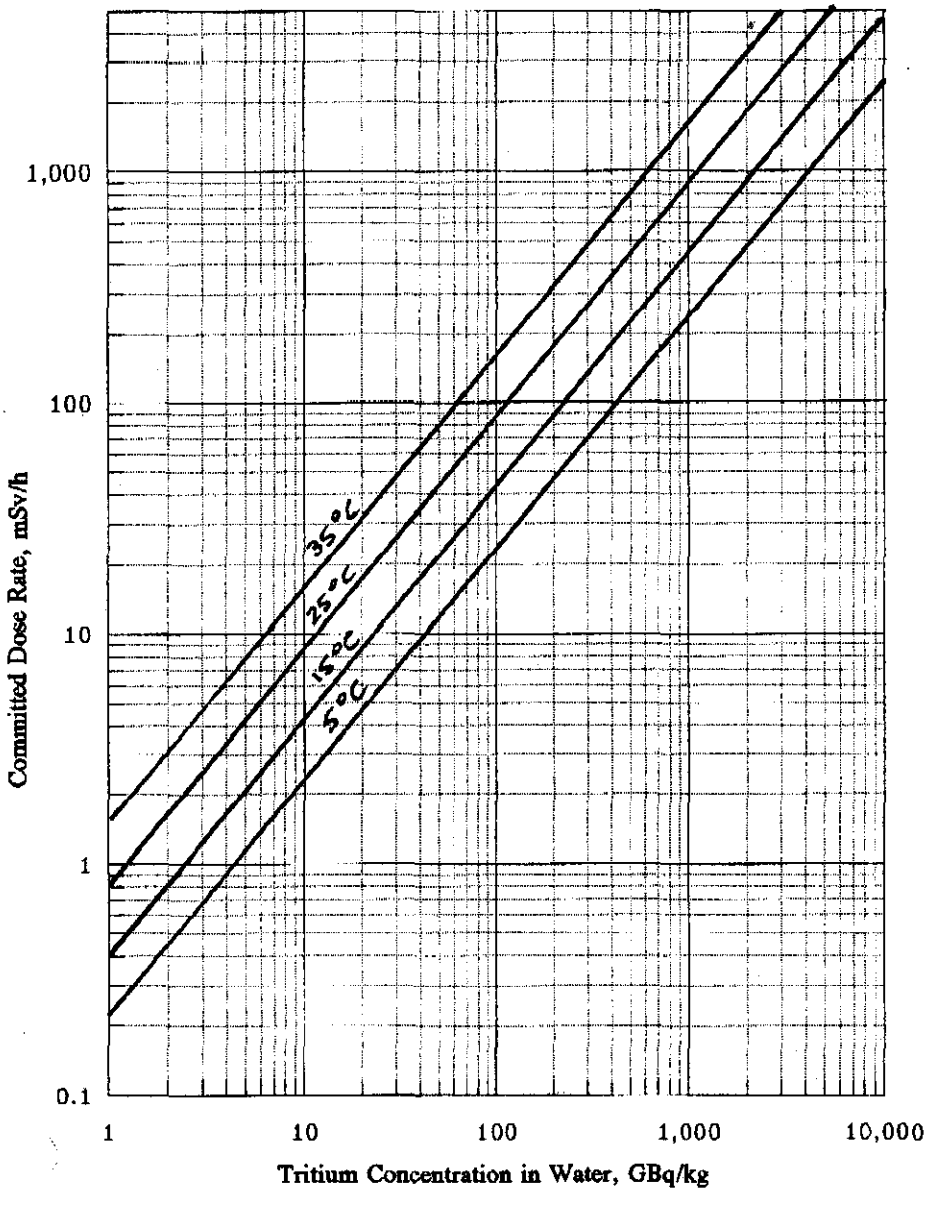


Fig. 8.3. Committed Dose Rate from Tritium in Air Saturated with Water Vapour

PHYSICAL FORM OF TRITIUM

When heavy water (D_2O) is irradiated by neutrons, a small fraction of the total number of deuterons (D) present absorbs neutrons (n, γ) to become tritium atoms (T). Then there will be water molecules in which one of the D atoms has become a T atom, and these molecules will thus be TDO instead of D_2O molecules. TDO is tritiated heavy water which we loosely call " tritium". (You might expect the formula T_2O, but there are thousands of D atoms present for every T atom, so that it is possible but very unlikely for two T atoms to be present in the same water molecule.)

Since the chemical properties of all isotopes of the same element are identical, the behaviour of tritium will be the same as that of water or water vapour. It is safe to assume that all heavy water in nuclear power stations contains tritium. Therefore, whenever heavy water is exposed to air, some of the heavy water and the tritium it contains will evaporate to cause an **airborne tritium hazard.**

The concentration of tritium in the air depends on its concentration in the source water (i.e., the water exposed to the air) and the temperature and the relative humidity of the air to which it is exposed.

Figure 8.3 shows how the tritium concentration in air depends on the concentration in the source water. This graph assumes that the air is fully saturated with water vapour, and that all the water vapour came from the tritiated source water. In other words, the committed dose rates shown will be upper limits — if the air is less than saturated, or if it contains a mixture of H_2O and TDO, the committed dose rate will be lower.

HOW TRITIUM ENTERS AND BEHAVES IN THE BODY

Tritium can enter the body via all three paths of entry mentioned earlier. However, normal and even accidental uptake is usually a result of inhalation and skin absorption.

Inhalation: Almost all of the tritium that is breathed into the lungs (as water vapour in the air) is rapidly absorbed into the blood stream.

Skin Absorption: Tritium diffuses through the skin from tritiated water vapour in air, and from tritiated liquid on the skin. Experience has shown that a person working in an atmosphere containing tritiated vapour will take in half as much tritium by absorption through the skin as he will by inhalation (i.e., one-third through the skin and two-thirds via inhalation).

It may seem peculiar that tritium in the form of water vapour in air enters the body through the skin and lungs, when these organs constantly give off water to the air in the form of perspiration and water vapour in exhaled breath. Yet, that's not the whole story. Tritium enters the body by diffusion — the process by which water molecules in air and in the body constantly move about, bouncing against one another.

Some water molecules diffuse from the air through the skin and the lining of the lungs into the body, while others diffuse out of them into the air. Water molecules containing tritium diffuse through the skin and lung lining into the body at approximately the same rate as ordinary

water. Once inside the body, they are carried away from the skin and lungs by the blood, before having time to diffuse out again. They are so diluted with ordinary water in the body that, under typical conditions, only one tritiated water molecule will diffuse out of the body (during tritium exposure) for every few million that diffuse in.

The blood distributes tritiated water equally among all body fluids, just as it does with normal water. Therefore all tissues in contact with body water will be irradiated by decaying tritium. These tissues are all the soft tissues in the body, and they make up 90% of the body weight. Consequently, any tritium in the body will lead to a whole-body dose.

The average person takes in 3 litres of water a day (2 litres directly as fluids, and another litre as food). Therefore, unless he is going to end up with severe problems, he also has to excrete 3 litres a day. (About half of this comes out as urine and faeces, the rest is by exhalation, diffusion through the skin, and sweat.) A person who has taken up some tritium and is then not exposed any more, will therefore get rid of this tritiated water at the rate of 3 litres per day, while at the same time diluting the remainder with his daily intake of 3 litres of tritium-free water.

It turns out that after 10 days, half of the tritium in his body has been flushed out. In other words, the effective half-life for tritium in the body is 10 days. Bear in mind that this figure is an average — if you drink more than the average person, you will flush the tritium out more rapidly and so have a shorter effective half-life. A person who drinks less will have a longer one. (People who drink less will also live longer, or at least it will seem longer.)

THE ALI AND THE DAC

The Annual Limit on Intake for tritiated water is 1E9 Bq. This value of the ALI applies to both the ingestion and the inhalation pathways, because tritiated water enters the body fluids, regardless of whether you have drunk the tritiated water or breathed it in as water vapour.

The DAC value for tritiated water must make allowance for the fact that a person working in a tritiated atmosphere will absorb half as much tritium through the skin as he will by inhalation. In other words, if he inhales 100 Bq of tritium, he will absorb another 50 Bq through the skin. Therefore, for tritium the DAC is reduced by a factor of 1.5 to allow for this. For tritium, the

$$DAC = \frac{ALI\ (Bq)}{2400 \times 1.5\ m^3}$$

$$DAC = \frac{1E9\ Bq}{3600\ m^3} = 3E5\ Bq/m^3$$

EXPOSURE AND COMMITTED DOSE

We discussed this topic a few pages ago. It is very important, so it won't hurt to mention it again.

Tritium taken into the body during an exposure lasting a couple of hours will still be irradiating the soft tissues many days later. If you know the tritium concentration in the air in DACs (1 DAC = 3E5 Bq/m^3), as well as the length of time spent in this concentration, you can estimate the dose to the whole body that you will receive. The reasoning is as follows:

1) We know that the ALI will lead to a whole-body dose of 20 mSv.

2) We know that a man working in a concentration of 1 DAC for 2000 hours will take in 1 ALI (from the definition of the DAC, page 316).

3) Therefore, exposure to 2000 DAC-h will result in a committed whole-body dose of 20 mSv.

4) Exposure to 1 DAC will then lead to a committed whole-body dose rate of 20 mSv/2000 h = 0.010 mSv/h = 10 μSv/h. This is a very useful relationship for tritium exposures. You've seen it before on page 321.

Every DAC of tritium exposure is equivalent to a committed whole-body dose rate of 10 μSv/h.

Remember, you won't receive the dose you commit yourself to until after several weeks, depending on your effective half-life.

Example: An area has a measured tritium committed dose rate of 80 μSv/h. If you were to work there without respiratory protection (more on this later) for 3 hours, what would be the committed whole-body dose?

You would receive an exposure of 80 μSv/h x 3 h = 240 μSv. This calculation is only an **estimate** of the dose that you will receive. In practice, it may be high or low by a factor of 2 or more. Why? Well, there are several reasons:

1) It is unlikely that the tritium concentration would remain unchanged for 3 hours, yet your dose estimate was based on only one measurement. It is obvious that more frequent measurements would be needed for an accurate estimate.

2) It is assumed that you breathe at the same rate as Reference Man.

3) It is assumed that the effective half-life for tritium in Reference Man (10 days) also applies to you. But you may drink more or less than he does.

URINE SAMPLES

The only way of obtaining accurate tritium dose information is by measuring the tritium content of body fluids.

After tritium is inhaled or absorbed through the skin, it is carried to all parts of the body by the blood. It distributes itself equally among all body fluids, such as blood, saliva, sweat or urine. Any of these fluids could be analyzed for tritium. Urine is more convenient to sample than blood, sweat and tears, so that's what we use. The Health Physics Department measures tritium-in-urine concentrations routinely and makes sure that the results are entered into your dose records. Three precautions are required to obtain a urine sample that is truly representative of the concentration of tritium in body fluids:

1) After an exposure it takes about two hours for the tritium in urine to reach the same concentration as exists in other body fluids. A sample given half an hour after exposure may only indicate about 50% as much tritium as one given two hours later. A sample given one hour after exposure may indicate about 75% of the true value. If you are exposed near the end of your shift, give a urine sample before leaving work and another on the next day.

2) Don't wait too long before giving a urine sample. Tritium reaches a maximum concentration about 2 hours after exposure. After that (if there is no further exposure) it is steadily eliminated from the body with an effective half-life of 10 days Hence a sample given one day after an exposure will indicate about 93% of the maximum concentration, and one given four days after will indicate only about 75%.

3) The bladder should be emptied after an exposure, before giving the first sample. This is especially important if the bladder is almost full at the time of exposure. Otherwise, the newly formed urine will be diluted with urine already in the bladder, and the indicated concentration will be lower than that in the rest of the body fluids.

Urine samples may be saved for several days before measurement. The tritium decay in the samples depends only on the radioactive half-life (12.3 years). So even in a week there will be little loss of activity. If you want accurate measurements of the tritium concentration in your body on your days off, take a couple of sample bottles home. Note the date of sampling and turn the samples in to the Health Physics Lab when you come back.

EVALUATION OF DOSE FROM BIOASSAY RESULTS

It can be calculated that a person who has 44 MBq of tritium in his body **all the time** will receive 20 mSv of whole-body dose in a year.

Tritium is spread uniformly throughout the body fluids. Reference Man contains 42 litres of body fluids. Therefore, a tritium concentration of 0.95 MBq/L **maintained** in body fluids will give you 20 mSv of whole-body dose in a year.

Why 0.95 MBq/L? When you divide 44 MBq of tritium by 42 litres of fluids, you will get 1.05 rather than 0.95 MBq/L. Why the difference? About 1% of the tritium does not appear

in body fluids, but is trapped in the body's cells for long enough to contribute 10% to the tritium dose. To include this in the assigned dose, we reduce the value from 1.05 to 0.95 MBq/L.

Therefore, if you **maintain** this concentration of tritium in your body fluids for a whole year, you will receive 20 mSv of whole-body dose in that time. If you maintain this concentration for one day, your dose for that day will be 20/365 mSv = 0.055 mSv = 55 μSv. Please realize that this is quite independent of what your effective half-life actually is, because we said that you are **maintaining** this concentration. How? By taking in enough tritium each day to replace that which is flushed out.

In practice, you will work only a limited amount of time in tritium atmospheres. Between exposures, the tritium in your body will be eliminated with an effective half-life of 10 days or so. Therefore, the tritium concentration in your body is not going to remain constant. It will fluctuate according to the severity of your exposures and the time between them. To keep track of these variations, you provide urine samples at periodic intervals and certainly after each exposure. From these samples the average tritium concentration in your body is calculated. Let's use an example to help you understand how these calculations are done.

In a 2 week period from June 1 to June 15 you submitted several urine samples for bioassay. The results were:

June 1	0.31 MBq/L
June 2	0.64 MBq/L
June 6	0.50 MBq/L
June 12	0.83 MBq/L
June 15	0.69 MBq/L

The average tritium concentrations for the four periods of time would then be calculated (see Fig. 8.4);

(A) (0.31 + 0.64)/2 MBq/L = 0.48 MBq/L (June 1 - June 2)
(B) (0.64 + 0.50)/2 MBq/L = 0.57 MBq/L (June 2 - June 6)
(C) (0.50 + 0.83)/2 MBq/L = 0.67 MBq/L (June 6 - June 12)
(D) (0.83 + 0.69)/2 MBq/L = 0.76 MBq/L (June 12 - June 15)

We pointed out that 0.95 MBq/L will give you a daily dose of 55 μSv. For period (B), for example, the dose for the four days will be

$$0.57 \text{ MBq/L} \times \frac{55 \text{ μSv/d}}{0.95 \text{ MBq/L}} \times 4 \text{ d} = 132 \text{ μSv} = 0.13 \text{ mSv}$$

Now work out the dose received for periods (A), (C), and (D).
The formula for the tritium dose is

$$\textbf{Dose} = \frac{\textbf{Sample 1 + Sample 2}}{\textbf{2}} \textbf{ MBq/L} \times \frac{\textbf{55 μSv/d}}{\textbf{0.95 MBq/L}} \textbf{ x n days}$$

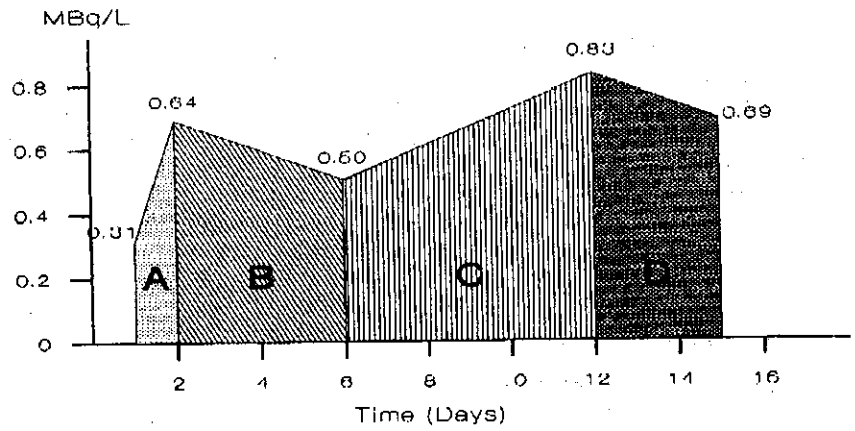


Fig. 8.4. Bioassay Data

If I did it right, the total for the two week period should be 0.52 mSv (0.03 + 0.13 + 0.23 + 0.13 = 0.52 mSv), and this would be entered into your dose records.

These calculations are done by computer. It looks at the bioassay result, searches the file for the previous result, spits out the dose to be added to your dose records, and then updates it. For example, if you give the next sample on June 20, it will look at that and the June 15 result to calculate the dose you received in the five days from June 15 to June 20. It will then add this to your dose records.

If work with significant tritium exposure is planned, and you haven't given a sample recently, it is wise to give one **before** the exposure and one **after** the exposure. The reason for this is obvious from Fig. 8.5. If your last sample was given on day 9, and you take up tritium on day 20, the computer program will just work out the average concentration for the 11 days, and assign you the corresponding dose for that period as shown opposite in (a). If you give a sample a day or two before you get exposed, as shown in (b) for day 18, the program will not have to assign any dose to you that you didn't receive (0.17 mSv in this example).

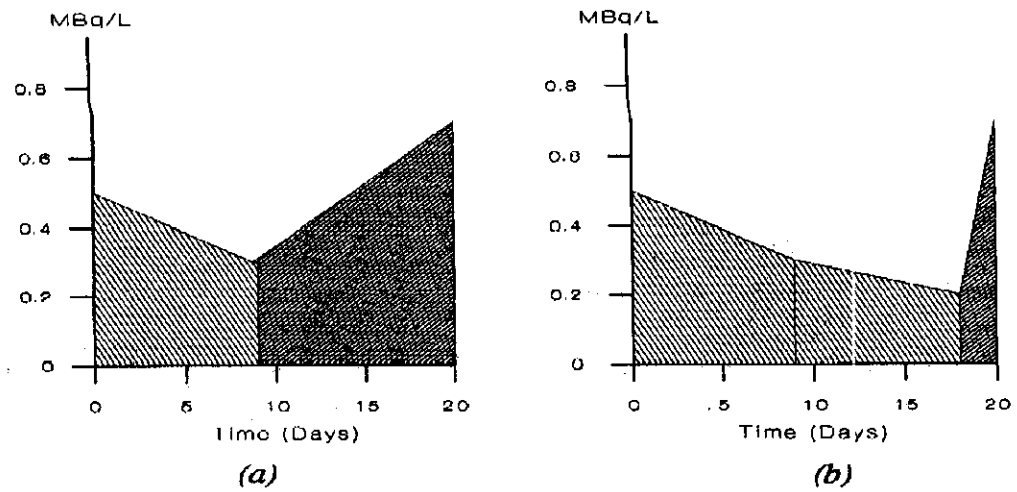


Fig. 8.5. Give a Sample Before and After Planned Exposures

DOSE COMMITMENT

Suppose that you are exposed to tritium so that within a short period of time your tritium-in-urine concentration reaches 0.95 MBq/L. You will initially be receiving dose at a rate of 55 μSv per day. If you are removed from work involving further exposure to tritium, the tritium will be eliminated from your body with an effective half-life of 10 days.

This means that after 10 days the concentration of tritium in your body will have fallen from 0.95 MBq/L to half that. In the same time, your daily dose rate will have been reduced from 55 μSv/day to 28 μSv/day. Figure 8.6 shows the daily doses for the first 30 days. If these are added up until all the tritium has been eliminated, the total accumulated dose you will receive is approximately 0.8 mSv.

We call this the **dose to infinity,** because in theory it takes an infinite time to get rid of all the tritium. In practice, about 90% of the infinity will be received within a month. The infinity dose is abbreviated as H .

Figure 8.6 assumes an effective half-life of 10 days. If you drink more than the average person, your effective half-life will be less than 10 days, because you are going to flush the tritiated water out of your body at a faster rate. For example, if you drink enough (about 5 litres per day) to reduce your effective half-life to 5 days, the tritium will be removed twice as quickly, and then your infinity dose from 0.95 MBq/L will only be 0.4 mSv instead of 0.8 mSv. If you plot the 5-day curve on Figure 8.6, you will be able to appreciate that the infinity dose is halved. In fact, the infinity dose from an acute uptake is directly proportional to the effective half-life.

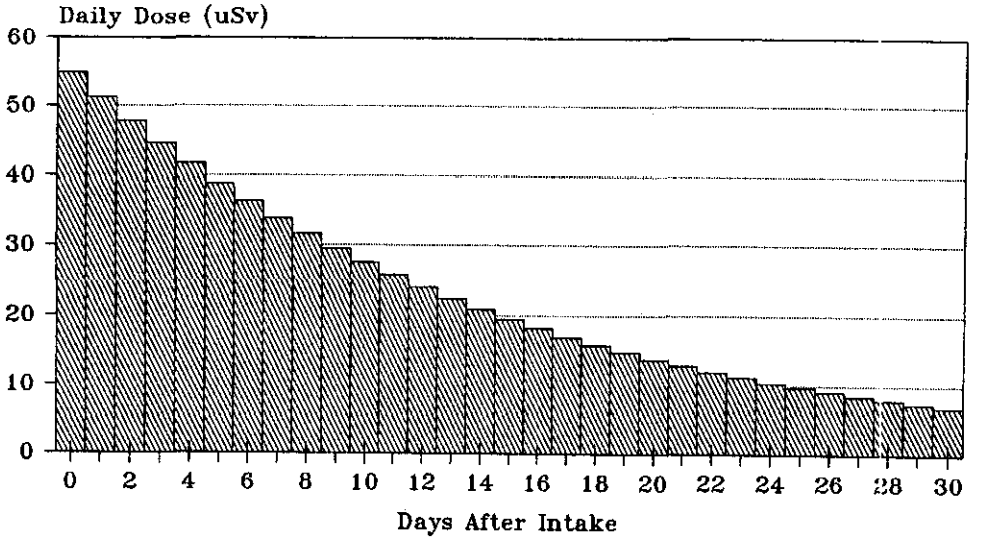


Fig. 8.6. Daily Doses From an Acute Uptake Resulting in an Initial Concentration of 0.95 MBq/L

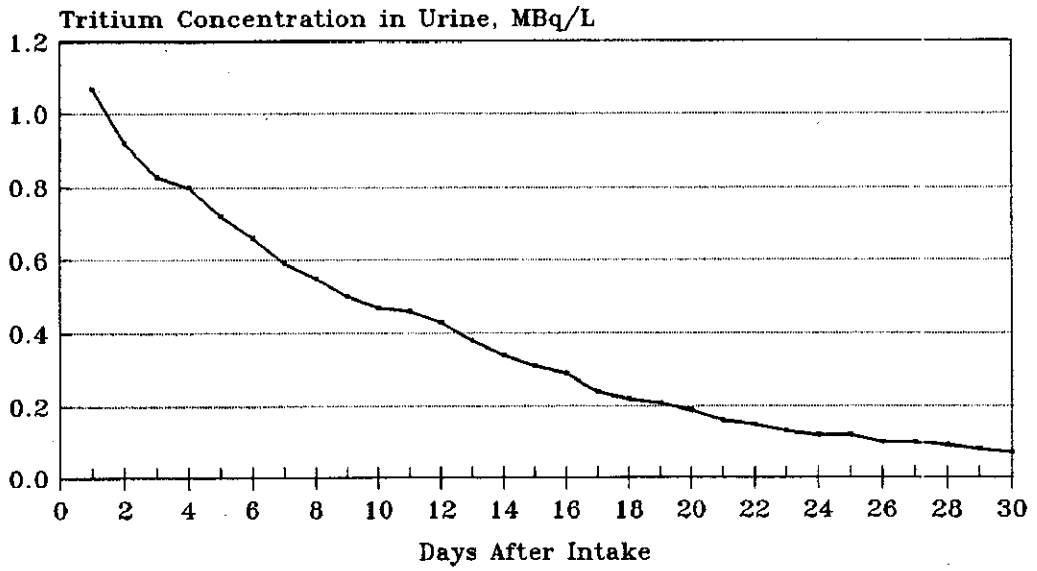


Fig. 8.7. Measured Data Showing the Decrease of Tritium in Urine (Courtesy of Little John)

BIOASSAY UPDATE REPORT

Every day, Bioassay Update Reports are posted in the station. The listing will tell you:

(a) the tritium dose assigned to you from your last sample,
(b) the committed dose, H , yet to come from the last sample submitted.

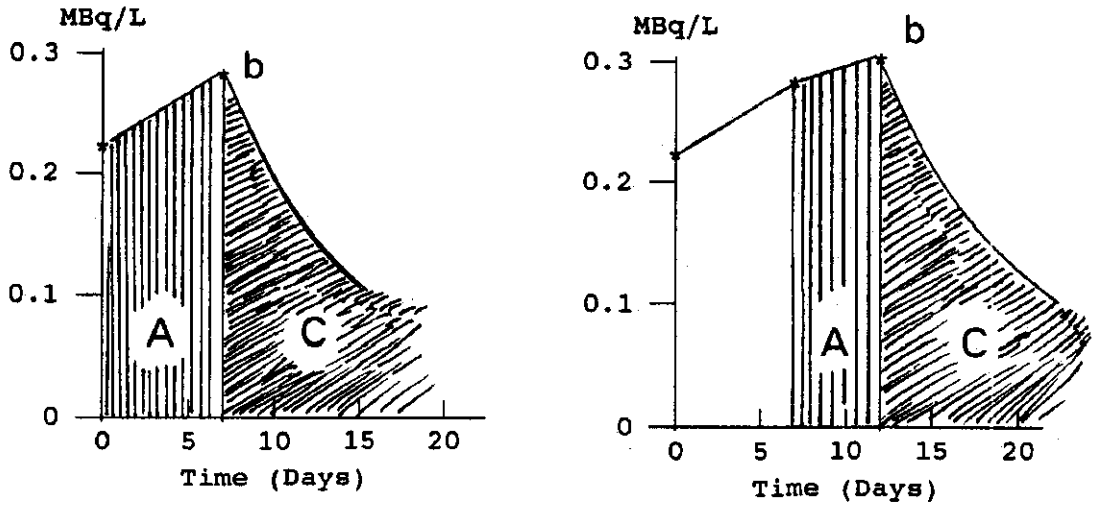
Fig. 8.8 on the next page is an example of how this works. Let us assume that you gave a sample at the beginning of the current Monitoring Period (0.22 MBq/L) and one 7 days later (0.28 MBq/L). The computer will print out:

(a) Dose assigned to you for the 7 days. This is 0.25 x (55/0.95) x 7 = 101 μSv. It will be shown as 0.10 mSv on the report.

(b) Committed dose. This is calculated by multiplying the concentration of the last sample by 0.84 mSv/MBq/L (i.e., 0.80 mSv for 0.95 MBq/L). In this case, that is 0.28 x 0.84 = 0.24 mSv.

Now you give another sample on day 12. It is 0.30 MBq/L. The next Bioassay Update Report will reflect this and the printout will show

(a) 0.08 mSv (for the five days from day 7 to day 12).
(b) 0.25 mSv



area A = 0.10 mSv
point b = 0.28 MBq/L
area C = 0.24 mSV

area A = 0.08 mSv
point b = 0.30 MBq/L
area C = 0.25 mSV

Fig. 8.8. Bioassay Update Calculations

If your sample shows a committed dose greater than 1.00 mSv, you are asked to donate a sample every day until your committed dose drops below 1.00 mSv. The reason for this is so that we can keep accurate track of the dose you receive.

You can see an example of a Bioassay Update Report on page 466.

PROTECTION AGAINST TRITIUM EXPOSURE

In this lesson we have talked about "unprotected" exposures. By dressing a worker in an air-supplied plastic suit we can reduce his tritium exposure by a factor of 100 or more. This subject will be left to the next chapter.

However, once you have been exposed, you can reduce your committed dose by increasing your fluid intake. Such a course of action may well be recommended for accidental large uptakes of several mSv committed, but you wouldn't do it routinely. The infamous 1990 tritiated drink incident, when an ex-worker added tritium to a drink cooler, is an example of this. Eight workers picked up a total of 466 man-mSv of tritium dose through no fault of their own. They were put on a regime of extra fluid intake under medical supervision: this saved about 275 man-mSv.

Increasing your fluid intake (say double the fluids for a week or so) will not give you any problems from a health point of view, but it isn't as easy to do as it sounds. Personally, I've found that significant increases in my fluid intake tend to give me a headache the next morning.

PROBLEM SYSTEMS

Now that we understand something about tritium exposure, let us take a look at our problem systems.

Moderator System

Fig. 8.2 shows that after six years of operation, the tritium concentration in the moderator heavy water should be around 1000 GBq/kg (it was actually 1050 GBq/kg). Air saturated with this water at 35 C can give 1500 mSv/h of tritium to an unprotected worker (see Fig. 8.3). People entering moderator system areas should recognize the enormous tritium risk that can exist there if D_2O leaks are present. To enter these areas without evaluating the tritium concentration in air is extremely risky and is unforgivable. People working with moderator water or performing maintenance on moderator D_2O systems must wear plastics (Chapter 9).

Primary Heat Transport System

The PHT system generally operates at high temperature and pressure (e.g., 300 C and 10 MPa), which makes it more difficult to prevent leaks. Each Fuelling Machine also spills 1 L of D_2O at each channel visit. Therefore the atmosphere of the Fuelling Machine Vaults and the Boiler Room can be expected to have levels of tritiated water vapour of 10 - 50 μSv/h even under normal conditions.

REMEMBER: Since the PHT water passes over irradiated fuel bundles, some of which may be defective, leaks of water vapour from the system may lead to airborne contamination in the form of radioactive noble gases, iodines and particulates. When any one of these hazards is detected in a PHT system area, the possible presence of the other hazards should be anticipated.

Spills of PHT system water can result in very high localized concentrations of tritiated water vapour.

Heavy Water Handling Systems

Work involving the transfer of tritiated water to systems or containers is often associated with tritium uptakes. High local concentrations of tritium can exist while a container is being filled due to venting of excess air from the container. Water spills also occur when lines are being connected or disconnected. Sampling systems are frequently associated with small spills or vapour leaks from unsealed containers. Wetting of the skin with spilled heavy water will cause tritium uptake.

Sponges, mops, etc., used for collecting small heavy water spills must be placed into sealed containers to prevent evaporation of absorbed tritiated water into the atmosphere.

RADIOIODINES

Several radioactive isotopes of iodine are produced in the fuel as fission products. They are very volatile and easily escape from fuel with defective zircaloy sheathing. We can expect to have some radioiodines circulating in the PHT system routinely if efforts aren't made to remove defective fuel as quickly as possible. Any leakage from the PHT system would then give rise to a radioiodine hazard.

Although radioiodine exposures are normally insignificant compared to tritium exposures, it is worthwhile devoting a little time to this subject, because under abnormal conditions the hazard can be significant.

Radioiodines are an Internal Hazard

If you are standing in a room that has radioiodine in the air, the dose you will receive from direct beta and gamma radiation striking your body is about 100 times less than the dose commitment to the thyroid that you will receive as a result of inhaling the radioiodines and absorbing them in your body. We therefore consider radioiodines to be an internal hazard. If we control the internal hazard, the dose from external exposure will be negligible.

THE IMPORTANT RADIOIODINES

There are three: I-131 is the most important, followed by I-133 and I-135 in that order. Table 8.3 on the next page lists the important properties.

There is one important difference between radioiodines and tritium. Although both are produced when the reactor operates, and production of both stops when the reactor shuts down, the radioiodines gradually decay away over a period of weeks, but the tritium does not. For example, if we have been shut down for two weeks, the only radioiodine remaining in the PHT system is I-131 (i.e., about one quarter of the original amount). The other radioiodines will have decayed completely. The tritium concentration in moderator and PHT water on the other hand will hardly have decayed at all because of its long half-life of 12.3 years.

TABLE 8.3. PROPERTIES OF RADIOIODINES

	I-131	I-133	I-135
Radiation Emitted	β, γ	β, γ	β, γ
Radioactive Half-Life	8.0 d	21 h	6.6 h
Effective Half-Life	7.5 d	21 h	6.6 h
DAC (Bq/m^3)	4E2	3E3	2E4

SOURCES AND LOCATIONS OF RADIOIODINE

Before we discuss what happens to radioiodines in the body after we have inhaled them, let's look at how they get into the air. There are three basic ways in which radioiodines can leak into station areas.

Leakage of PHT Heavy Water

If we have defective fuel in the core, radioiodines will escape from it into the coolant where they dissolve. They will exist mainly as elemental iodine (i.e., I_2 molecules) with some organic iodine present, frequently as methyl iodide (CH_3I). (The methyl stuff comes from ion exchange resins in the PHT purification circuits.) If a leak of D_2O steam occurs, the radioiodines enter the air and tend to be picked up by dust particles. Most of them will then exist in particulate form.

Leaks of radioiodines can occur in the following areas of the Station:
1) Fuelling Machine Vaults
2) Fuel Discharge and Transfer Bays
3) Fuelling Machine Valve Stations
4) Fuelling Machine Maintenance Areas
5) Boiler Room
6) PHT Sample Stations

Leakage From Defective Fuel Bundles

Defective fuel bundles in the Spent Fuel Bays will release radioiodines. The radioiodine release causes volatile hypoiodous acid (HOI) to be produced in the bay water. Airborne activity of the order of hundreds of μSv/h of I-131 can be expected in the Spent Fuel Discharge Bay area for several hours after defective fuel bundles have arrived there.

Release From Surfaces of PHT and F/M Components and D_2O Recovery Systems

If these systems are opened for maintenance work, high releases of airborne radioiodines can occur. Welding of PHT and F/M components can cause radioiodines to be released from surfaces to which they had attached themselves. Open D_2O recovery systems can cause significant releases of particulate radioiodine.

HOW RADIOIODINE ENTERS AND BEHAVES IN THE BODY

About half of the radioiodine inhaled is absorbed through the lungs into the blood stream. Absorption through the skin is negligible compared with inhalation. Roughly 30% of the radioiodine in the blood stream is deposited in the thyroid gland within 24 hours after the intake. The remaining radioiodine in the blood stream is eliminated from the body by urine in a couple of days (in fact it drops to 10% after one day).

Iodine in the thyroid has a biological half-life of 120 days. For I-131, with a radioactive half-life of 8 days, the effective half-life is

$$T_e = \frac{T_r \times T_b}{T_r + T_b} = \frac{8 \times 120}{8 + 120} = 7.5 \text{ days}$$

The effective half-lives of I-133 and I-135 are the same as their radioactive half-lives (see Table 8.3).

DOSE COMMITMENTS FROM RADIOIODINE

When they do occur, most radioiodine exposures are acute rather than chronic, namely several tens of μSv/h for an hour or two rather than a few μSv/h over a long period of time.

The same principles apply for radioiodine exposures as for tritium. That is, 2000 DAC-h will cause you to receive a weighted dose H_W equal to the annual limit of 20 mSv. Working in an atmosphere contaminated with 1 DAC of radioiodine is the same as committing yourself to 10 μSv for every hour you work there, i.e., 1 DAC is equivalent to 10 μSv/h.

Of course, if you are working in an atmosphere contaminated with radioiodines, you will normally not only be taking in I-131, but also I-133 and I-135. For instance, if the air contains

50 μSv/h of I-131
20 μSv/h of I-133
and 10 μSv/h of I-135

this corresponds to 80 μSv/h of radioiodine. If you work in this atmosphere for 1 h, you will be committed to 80 μSv of weighted dose from radioiodine.

As with tritium, knowledge of the exposure in μSv/h only gives you an **estimate** of the dose you are likely to receive. To measure the dose from radioiodine, we need to know how much is in the thyroid. Once we know that, we can work out the dose commitment to the thyroid (H_T), and then the committed weighted dose ($H_W = H_T w_T$).

You will be relieved to learn that you don't have to give us a thyroid sample. Since all the radioiodine isotopes emit gamma photons, we can use a scintillation counter to measure how many gamma photons of what energy are emitted each second from the thyroid gland. This gives us the activity of each radioisotope. We have two of these thyroid counters at Point Lepreau.

One is a self-serve thyroid monitor located outside the station Health Physics Lab (see Fig. 8.9). If you know or suspect that you have been exposed to radioiodine, you merely go over there, rest your chin on the detector and see if a 30 second count causes the scaler to exceed the limit posted beside the instrument.

If not, O.K. If yes, wash your neck and repeat the count (why?). If still yes, inform the Shift Supervisor and request the deluxe treatment, namely a chair count. You sit in a supremely comfortable, padded arm chair while the lads in the Health Physics Lab measure your thyroid activity.
The results from this and follow-up counts will be used to calculate the dose to the thyroid, H_T. This information will be stored in your dose records file. The weighted dose, $H_W = H_T w_T$, will also be calculated and added to the whole-body dose in your dose records.

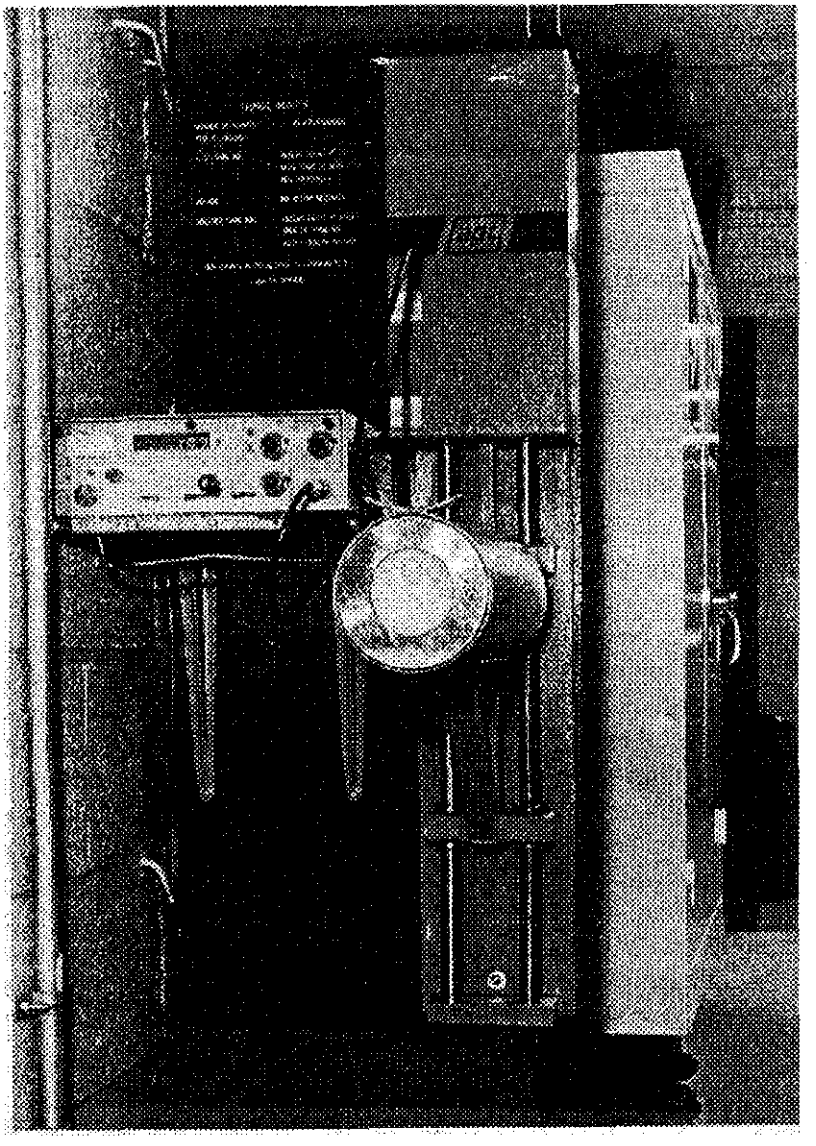


Fig. 8.9. The Self-Serve Thyroid Monitor

REDUCTION OF IODINE DOSE

To perform its functions, the thyroid takes up only a small fraction of a milligram of iodine from the blood each day. If radioactive iodine is inhaled, it reaches the blood quickly and will be taken up by the thyroid.

We can reduce the thyroid uptake of radioactive iodine by flooding the blood with stable iodine just before an exposure or shortly afterwards. The radioiodine will have to compete with the stable iodine for entry to the thyroid. Since the thyroid can't distinguish between the two, a smaller fraction of radioactive iodine is taken up. This is commonly known as "thyroid blocking"; it is a misnomer since the thyroid is never completely blocked. This "thyroid blocking" (iodine dilution of blood) is done by taking a 130 mg potassium iodide (KI) pill. These pills are harmless to most people (for instance, potassium iodide is added to ordinary table salt).

Stable iodine is not able to dislodge radioiodine that has already accumulated in the thyroid. Therefore, ideally, the KI pill should be taken before the exposure, or if not, the sooner after the exposure, the better. If taken before the exposure, it will reduce the radioiodine uptake of the thyroid by more than 98%. After a delay of 3-6 hours, it will reduce the dose by 60-30%, and after 12 hours it does hardly anything for you — i.e., the thyroid has already picked up almost all of the radioiodine it is going to pick up. For a single exposure, only one pill is needed to provide adequate protection.

KI should not be used as a substitute for proper respiratory protection. If you know you are going to be exposed to radioiodine, you should wear a suitable respirator or perhaps an air-supplied suit. These protective measures will be described in Chapter 9.

KI pills are under the control of the Shift Supervisor. He will issue them to workers who may have committed themselves to more than 0.5 mSv of weighted dose from a radioiodine uptake. You should not take KI pills if you suffer from thyroid problems or food-related allergies (particularly salt).

PARTICULATES

Radioactive particulate materials can exist as dust deposited on a surface or dispersed in the air. The radioactive particles can be fission products (e.g., Cs-137) or activation products (e.g., Co-60). The main radiation hazard from particulate forms arises when airborne particulates are inhaled and collect in the lungs. Even surface dusts can become a lung problem when they are released from the surfaces to become airborne. Let's take a look at the internal radiation hazard presented by airborne particulates.

SHORT-LIVED AND LONG-LIVED PARTICULATES

The particulate activity that we occasionally find at Point Lepreau falls into two categories; short-lived, with half-lives of less than an hour or so, or long-lived with half-lives of several days and more. This short-lived activity comes from the daughters formed by the decay of fission product noble gases.

The long-lived activity is normally due to activation products and fission products released from fuel and D_2O systems. The hazards associated with short-lived and long-lived particulates are quite different. If you inhale equal activities of both, you could be receiving dose from the long-lived particulates for years, whereas the short-lived particulates will have decayed within hours. In other words, the long-lived particulates present a much greater internal hazard than the short-lived particulates. For this reason, we deal very cautiously with particulates, and treat them all as long-lived unless we know for sure that they are short-lived.

Because short-lived particulates that you have inhaled will deliver dose for only the short period of time until they have decayed, their DACs tend to be quite high compared to long-lived particulates. Indeed, if you do the sums, it turns out that for half-lives of less than an hour or so, the external beta-gamma dose you get from standing in the cloud is generally higher than the internal dose committed by inhaling the particulates. Therefore, short-lived particulates are largely an external hazard rather than an internal hazard.

SHORT-LIVED PARTICULATES

Short-lived particulates often occur in active areas of the station. The activity is usually due to rubidium-88 and cesium-138. These two radionuclides are the daughters produced by the fission product noble gases krypton-88 and xenon-138:

$$^{88}_{36}\mathrm{Kr} \xrightarrow{\beta^-,\gamma} {}^{88}_{37}\mathrm{Rb} \xrightarrow[T_{1/2}=18\,\mathrm{min}]{\beta^-,\gamma} {}^{88}_{38}\mathrm{Sr}$$

$$^{138}_{54}\mathrm{Xe} \xrightarrow{\beta^-,\gamma} {}^{138}_{55}\mathrm{Cs} \xrightarrow[T_{1/2}=32\,\mathrm{min}]{\beta^-,\gamma} {}^{138}_{38}\mathrm{Ba}$$

The ratio of Rb-88 to Cs-138 is usually in the range of 10:1 or so. This mixture has a half-life of about 20 minutes. In practice, we consider Rb-88 and Cs-138 to be an external hazard and we don't worry about the internal dose. This is because the internal dose is very hard to measure for such short-lived activity and it should normally be less than the external dose in any case. We assess the external hazard with beta and gamma survey meters.

LONG-LIVED PARTICULATES

Here the story is quite different. A significant fraction of the inhaled long-lived particulates will remain as an internal hazard for a long time.

Long-lived particulates can be any of the following radionuclides:

Ce-144	I-131	Sr-89	Mn-54
Ce-141	Ru-106	Zn-65	Cr-51
Ba-140/La-140	Ru-103	Co-60	C-14
Cs-137	Zr-95/Nb-95	Fe-59	
Cs-134	Sr-90		

They are all beta-gamma emitters except Sr-90 and C-14, which emit betas only.

The intake pathway for these radionuclides that exist in particulate form is inhalation. What happens to them after they've reached the lungs depends on whether they are transportable or non-transportable. We've dealt with this before, but it won't hurt to repeat it:

Transportable Radionuclides are relatively soluble in the lung fluid, and are readily transferred into the blood stream. From there they are transported to the various tissues where they are deposited. In our list of long-lived particulates, these are typically transportable: Cs-137, Cs-134, I-131, Zn-65, Sr-90, Sr-89, and C-14.

Non-Transportable Radionuclides tend to stay in the lungs with biological half-lives of about one year. Transfer through the lung lining is slow. An appreciable fraction of the activity is eventually eliminated by travelling up the respiratory tract as phlegm. It is then swallowed to give a little bit more dose to the gastrointestinal tract on its way out.

THE DAC FOR PARTICULATES

The long-lived particulates are an internal hazard. They have effective half-lives varying from several days to several years. The short-lived particulates are primarily an external hazard and those normally found in CANDU stations have a half-life of 20 minutes or so. This should tell us that the DAC for long-lived particulates will be much more restrictive than for short-lived particulates.

What do we do in practice to assess the hazards from particulates? We take an air sample (Chapter 9 describes how this is done). For the time being it is enough to know that sampling for particulates tells us how much particulate activity is present in the air. We won't know whether this activity is due to long-lived particulates or short-lived particulates. We therefore take the cautious approach and assume that it is the more hazardous long-lived particulates. The long-lived radionuclides (other than radioiodines) that have occasionally been found present in the air in operating CANDU stations are Co-60, C-14, Cs-134, Cs-137, Zr-95 and Nb-95. Their corresponding DACs are listed in Table 8.4 opposite.

At Point Lepreau we have taken the approach that all particulate air sample activity is assumed to be attributable to Co-60. This is a cautious approach. For example, if we take an air sample and measure the activity to be 1000 Bq/m^3, this could correspond to any one of the following, depending on which radionuclide is responsible:

50 μSv/h Co-60,	13 μSv/h Cs-134,	13 μSv/h Cs-137,
10 μSv/h Zr-95,	3 μSv/h Nb-95,	0.5 μSv/h C-14.

Any mixture of these radionuclides would still correspond to less than 50 μSv/h overall. If we base our protection requirements on the assumption that all particulate activity is due to Co-60, we will be on the safe side. Therefore

The DAC for unidentified particulate activity is 200 Bq/m^3.

The DACs for the short-lived particulates we normally find (i.e., Rb-88 and Cs-138) are at least a thousand times larger.

If the particulate activity is identified, then the true committed dose rate can be used. In order to do this, the sample must be analyzed by gamma spectrometry to determine how much of each radionuclide is present. This service is available from the Chemistry Lab if required.

TABLE 8.4. DACs OF LONG-LIVED PARTICULATES TO BE EXPECTED IN CANDU STATIONS

Radionuclide	DAC (Bq/m^3)
Co-60	2E2
Cs-134	8E2
Cs-137	8E2
Zr-95	1E3
Nb-95	4E3
C-14 (organic compounds)	2E4

SOURCES AND LOCATIONS OF PARTICULATES

Long-Lived Particulates

Areas related to the fuel transfer system can become contaminated with high levels of long-lived particulate, particularly after damaged fuel bundles have been handled. High concentrations corresponding to committed dose rates of 5 mSv/h have been measured in a fuel transfer room at Pickering G.S. Leakage from this room resulted in concentrations equivalent to 2 mSv/h in a nearby accessible area.

Maintenance tasks like those described below can also cause significant levels of long-lived airborne particulates:

(a) Replacing spent molecular sieve on closed cycle (D_2O) vapour recovery driers can produce long-lived particulate airborne levels equivalent to almost 1 mSv/h, primarily due to fission product Cs-137 and activation product Cs-134. Levels this high are usually the result of poor work practices.

(b) Maintenance work of a vigorous nature (e.g., sanding, machining, welding) on components that have been in a reactor system (e.g., pump impellers, seals), has frequently resulted in wide-spread particulate contamination due to long-lived fission and activation products. Levels of tens of μSv/h have occurred under these conditions.

(c) Radioiodine in the form of particulate occurs when radioiodine escapes to the atmosphere and adheres to dust particles. Fuel transfer systems are a particularly common source of this hazard.

(d) Work activity with loose surface contamination present will inevitably result in resuspension of active material to form airborne particulate contamination.

Short-Lived Particulates

We know that short-lived particulate activity is mainly due to Rb-88 and Cs-138 and that this form of particulate occurs quite frequently. Any leak in the PHT system can result in a spread of short-lived activity throughout the entire Reactor Building.

For example, wide-spread levels of short-lived particulates throughout the Reactor Building were found when a D_2O collection tank serving the fuelling machines was vented to atmosphere. This caused general fields of 50 mGy/h beta and 10 mGy/h gamma in the Reactor Building basement. Even at the Equipment Airlock they were still at 10% of these values. This problem was solved by re-routing the tank ventilation to the contaminated exhaust system.

The exposure of recently failed irradiated fuel bundles to air during passage through the fuel transfer system results in high localized air concentrations of short-lived particulates. Remember that short-lived particulates present an external radiation hazard, not an internal hazard. The external hazard is measured with beta and gamma survey meters.

CARBON-14

In February, 1985, Ontario Hydro had a nasty surprise while changing the pressure tubes in Pickering A Units 1 and 2. A radionuclide that their interzonal monitors could not detect was found all over the station. Contamination was relatively widespread and was even detected in some employees' homes. This radionuclide was carbon-14.

Physical Properties

C-14 emits low energy beta radiation (E_{max} = 156 keV); there is no gamma emission. The half-life is 5730 years.

C-14 is produced whenever water or air is exposed to neutrons:

$$^{7}_{8}O + n \longrightarrow {}^{14}_{6}C + \alpha \quad \text{(moderator and PHT water)}$$

$$^{4}_{7}N + n \longrightarrow {}^{14}_{6}C + p \quad \text{(air)}$$

Another radionuclide that is produced by the neutron irradiation of air is argon-41. It often signals air contamination in systems where we don't want any air, because there will be a rapid increase in gamma fields caused by Ar-41. Once the reactor is shut down, the argon will decay with its half-life of 1.8 hours.

C-14, on the other hand, slowly builds up in systems that are contaminated with air, and it won't decay away when the reactor is shut down. Since C-14 doesn't emit gammas, it doesn't increase the gamma fields. C-14 contamination should be suspected in in-core gas systems that could be contaminated with air.
C-14 is of concern: it is hard to detect, and we can't depend on it being mixed with other radionuclides that are easier to detect. At Point Lepreau, contamination in one particular job was over 99% C-14.

Detection of Carbon-14

You need an instrument with a window thin enough to allow the low energy beta particles to enter the detector. We have several of these, of which you'll normally use four:

Instrument	Efficiency
Herfurth Full-Body Monitor	about 15%
Herfurth Hand & Shoe Monitor	about 15%
Nuclear Enterprises Frisker (Exit to Bridge)	about 15%
Eberline Frisker	about 4%

As you can see, there isn't much chance that C-14 contamination will escape detection at Point Lepreau. In fact, the Herfurth monitors were purchased as a direct result of the Pickering experience.

If you suspect that some contamination is C-14, you can use this test:

1. Measure the count rate with one of the 15% efficient instruments listed above.
2. Place a piece of paper between the contamination and the detector, and measure the count rate again.

If the count rate in (2) was only 10% of that in (1), you are probably dealing with almost pure C-14.

Behaviour of Carbon-14 in the Body

The behaviour of C-14 in the body is strongly influenced by its chemical form. At one extreme, carbon monoxide and carbon dioxide, because of their short retention time in the body, have DACs that are less restrictive than tritium. For C-14 in organic compounds, the DAC is 2E4 Bq/m^3, and that's the value we use, because it is the most restrictive.

BIOASSAY

Before you work in areas contaminated with airborne activity, you must assess the anticipated exposure. How this is done will be described in the next chapter. However, this assessment is only an estimate. We won't be able to tell how much activity you have taken up until after the exposure. The measurement of your uptake is known as **bioassay.**

> **BIOASSAY is the determination of the type, quantity and location of radioactive material in the body by direct measurement or by analysis of materials excreted from the body.**

As you might realize, from the definition above, there are two methods:

1. Excretion Analysis (urine and faeces)
2. Whole-Body Counting.

EXCRETION ANALYSIS

Urine Analysis

A fraction of the body content of a radionuclide is excreted in urine each day. By analyzing the radionuclide content of urine, we can estimate the body content for some radionuclides if we know the fraction that is excreted each day. Unfortunately, this fraction is usually not well known. The main problem with urine analysis is that only transportable radionuclides are excreted in substantial amounts and in a predictable manner in urine.

Many inhaled radionuclides are non-transportable. For these, a very small and highly variable daily fraction reaches the blood stream from the lung. This makes it very unreliable and sometimes impossible to estimate body contents from urine analysis. However, it is possible to confirm the presence of the non-transportable radionuclides and to identify them.

Tritium represents a special case because it is uniformly distributed in body water. The concentration in urine is the same as in all the other body fluids, so the daily dose rate and the dose commitment may be estimated directly from the urine concentration.

Tritium-in-urine analysis is done by liquid scintillation counting. The normal sampling period for people routinely exposed (i.e., those with a committed dose greater than 0.01 mSv) is one week. This rather short sampling period is necessary because the effective half-life of tritium is only 10 days.

We take advantage of the routine tritium bioassay program to look for the presence of other beta-gamma emitters in urine. Tritium has a rather low beta energy (E_{max} = 18 keV), and spare channels of the pulse height analyzer in the liquid scintillation counter are used to monitor the beta energy region above 18 keV. The presence of transportable radionuclides such as C-14, I-131, Sr-90, Sr-89, Cs-137 and Cs-134 is looked for in these "high energy" channels. If we see anything other than tritium, we will call you in for a whole-body count.

On rare occasions, large volume urine samples may be analyzed by gamma spectrometry or radiochemistry to identify and estimate radionuclide contents with much better sensitivity. But even so, this will still only measure what comes out of the body, not what's in it. Radionuclides that are excreted very slowly can be very hard to detect.

Faecal Analysis

This is not generally done at CANDU stations. It is very messy and doesn't tell us much. Faecal analysis is more useful for ingested material, and ingestion is not a significant pathway in our case. Inhaled transportable nuclides will largely be eliminated in urine; inhaled non-transportable nuclides will be deposited in the lung. You will recall that from the lung they are eventually brought up as phlegm to be swallowed and show up in faeces. Normally it is much better to look for non-transportable radionuclides in the lungs with a whole-body counter. Faecal analysis is superior only when the gamma energy is so low that one cannot detect the gamma emitter in the lungs, or when we suspect that non-transportable pure beta emitters are present (such as C-14 particulates). Other than C-14, internal hazards in this category are rare. This is good news for the folks in the Health Physics Lab.

WHOLE-BODY COUNTING

In whole-body counting, the amounts of gamma emitting radionuclides in the whole body are measured directly using a sodium iodide scintillation detector. To reduce the background and hence increase the sensitivity of the measurement, both the detector and the subject are shielded, usually with lead.

Three versions of a whole-body counter are common: a scanning type, a chair type, and a stand-up type.

In the older scanning type, the person to be monitored lies on a motor-driven bed which passes slowly under a single large sodium iodide detector enclosed in a lead-shielded housing. The time for a complete scan from head to toe is typically around 10 minutes. The counting data are analyzed by gamma spectrometry.

We have a chair counter as shown on page 217. It has three detectors; one looks at the thyroid, one at the lungs and one at the gut. The subject sits in the chair, and the counts seen by each detector are accumulated in separate spectra. All the data are processed by computer. After a 5 minute count, we can normally identify radionuclides and assess their quantity in the thyroid, lung and gut often to much better than 1% of that quantity that would cause you to be irradiated at the annual limits. If real activity (other than the potassium-40 present in everybody) is detected, we would use the results from the whole-body counter to calculate the doses to the target tissue, and then enter these doses into your dose records. Figure 8.10 shows a typical set of lung and thyroid spectra obtained with our chair counter.

The stand-up type of body counter is currently in vogue. It is available with sodium iodide or germanium detectors. You stand in front of the shielded detectors for a minute, and then have the computer analyze the spectrum and cough out the results. With just a few simple inputs from you, it can be used as a self-serve unit. Health Physics would update the dose records, and follow up any non-zero body count results.

PROS AND CONS OF WHOLE-BODY COUNTING

The advantages of whole-body counting are:

1) Whole-body counting measures body contents directly, as contrasted with indirect inferences from excreta measurements (urine and faecal analyses).

2) Non-transportable radionuclides in the lung do not pose a problem for whole-body counting as they do for urine analysis, provided that the gamma photons are sufficiently energetic to be able to reach the detector.

The disadvantages of whole-body counting are:

1) It cannot be used in general for radionuclides that do not emit gamma radiation (pure beta emitters). The only important pure beta emitters to be found at a nuclear station are H-3, Sr-90 and C-14. As part of the routine H-3 bioassay program, the presence of C-14 in urine is automatically screened. The strontium isotopes are not present in significant amounts in the station. Typical Cs-137/Sr-90 ratios are 50/1 to 100/1. If a significant amount of Cs-137 is found by whole-body counting, the individual will be requested to submit a large volume urine sample for radiostrontium analysis.

2) The whole-body counter cannot distinguish between radioactivity inside the body (genuine internal contamination) and external contamination on the skin surface or in hair. This can lead to counting results that are somewhat on the high side, i.e., overestimation of the amount of internal contamination. Also, it is difficult to calibrate a whole-body counter, because of differences in organ sites, body size and shape, etc. However, a whole-body counter has high sensitivity so that negative results (i.e., no counts above background) definitely indicate the absence of internal contamination.

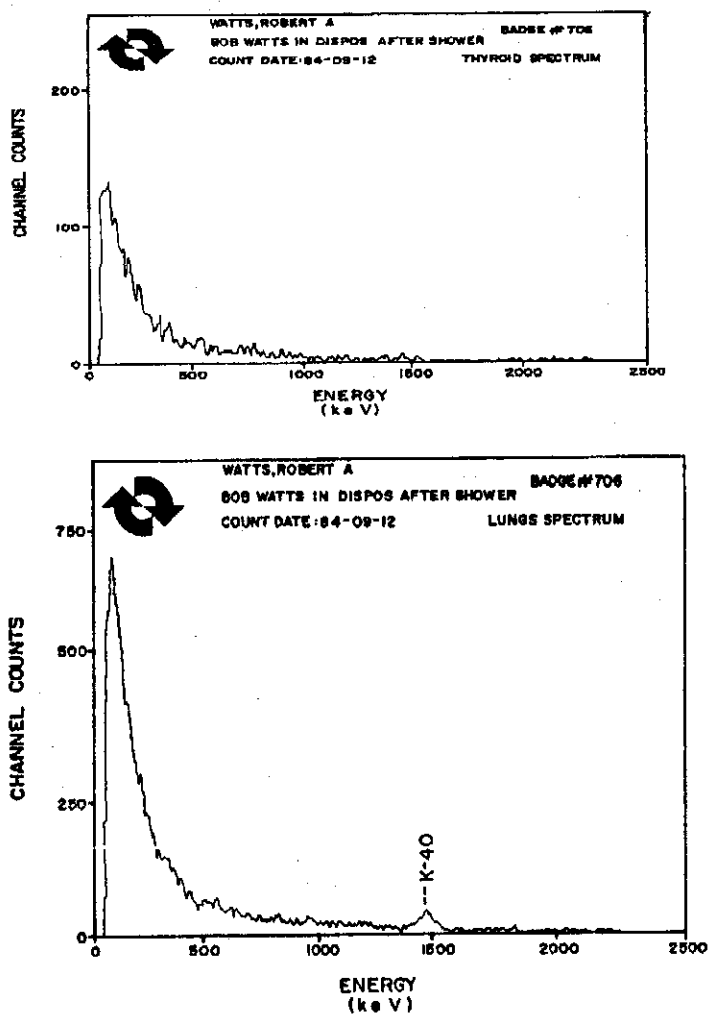


Fig. 8.10. Typical Thyroid and Lung Spectra From Our Whole-Body Counter

Table 8.5 below summarizes the routine bioassay done at Point Lepreau.

TABLE 8.5. ROUTINE BIOASSAY AT POINT LEPREAU

Radionuclide	Bioassay Method	Frequency
Tritium	Urine analysis by liquid scintillation counting	Every 28 days, but daily for 1 mSv H_{∞} and weekly for 0.01 mSv H_{∞}
Radioiodine	a) High energy channel screening as part of routine H-3 program	As for tritium
	b) Self-serve thyroid monitor	Individual preference
	c) Whole-body counter	When required for accurate thyroid dose assignment
Gamma emitters (Co-60, Cs-137, Zr-95, etc.)	a) Whole-body counting	Selected people in exposed groups, or as called for by b)
	b) High energy channel screening as part of routine H-3 program	As for tritium
Carbon-14	Medium energy channel screening as part of the routine H-3 program	As for tritium

SUMMARY

Radiation sources that can enter the body are called internal hazards. They demand more precautions than external hazards, because

1) *all radiation from internal sources can interact with tissue,*
2) *most radionuclides accumulate in certain organs, called target tissues, which will receive higher doses than the rest of the body,*
3) *internal sources expose you continuously,*
4) *it is difficult to increase their rate of elimination,*
5) *except for tritium, it is difficult to estimate the dose.*

Radiation sources can enter the body in three ways: inhalation, ingestion, and absorption through skin or wounds. The first is the most important. Internal sources are removed by biological elimination and radioactive decay with an effective half-life T_e.

The Annual Limit on Intake (ALI) is the activity (Bq) of a radionuclide that would commit you to 20 mSv of weighted dose (H_W).

If you work in the Derived Air Concentration (DAC) of a radionuclide for 2000 hours, you will take in one ALI of that radionuclide. One DAC is equivalent to a committed dose rate of 10 μSv/h.

Tritium is produced whenever heavy water is irradiated by neutrons. When exposed to air, tritiated heavy water leads to airborne hazards in the form of tritiated water vapour. It enters the body by inhalation and absorption through the skin, and mixes with body fluids. A concentration of 0.95 MBq/L of tritium in the body will deliver 20 mSv of whole-body dose if it is maintained for a year, but only 0.8 mSv (the infinity dose) if it is eliminated with a 10 day half-life. Dose is assessed by analyzing the tritium concentration in urine samples.

Radioiodines (I-131, I-133, I-135) are volatile β, γ emitting fission products. Defective fuel allows them to escape into PHT water, from which they can become an airborne hazard through PHT system leaks. The target tissue is the thyroid gland. Radioiodine uptakes can be detected with the self-serve thyroid monitor — accurate measurements are made with the whole-body counter in the Health Physics Lab. A KI pill taken soon after the exposure can prevent the majority of the radioiodine taken into the body from reaching the thyroid.

Particulates are radionuclides attached to dust particles. We classify them as long-lived or short-lived, depending on whether their half-life is greater or less than 30 minutes. Short-lived particulates are an external hazard; long-lived particulates are an internal hazard. If you don't know what they are, you treat them as unidentified particulates with a DAC of 200 Bq/m^3. Long-lived particulates are classified as transportable or non-transportable. The former are soluble in lung fluid, and transported to their target tissue; the latter tend to remain in the lung for about a year.

Bioassay is the determination of radionuclides in the body by direct measurement or by excretion analysis. Urine bioassay is done routinely for tritium, and is also able to detect other radionuclides that may be analyzed with the whole-body counter.

PROBLEMS

1. What is the difference between external and internal hazards?

2. Why do we tend to be much more cautious about internal hazards than external hazards?

3. Name the routes by which radioactive material is unlikely to enter your body at Point Lepreau.

4. What is meant by Target Tissue?

5. Tritium has a T_r of 12 years and a T_b of 10 days. Calculate the effective half-life T_e.

6. Ru-103 (T_r = 40 days) has been deposited in your lungs. Repeated chair counts indicate that the T_e is about 30 days. What is the biological half-life in the lung?

7. What is the difference between intake and uptake? Is uptake always less than intake?

8. Which of the following statements are correct?
   a) The ALI is given in units of dose.
   b) The ALI is given in units of dose per year.
   c) The ALI is given in Bq.
   d) All the dose from 1 ALI will be delivered within one year.
   e) The ALI for inhalation is usually smaller than that for ingestion.
   f) None of the above.

9. Which of the following exposures will result in an intake of 0.1 ALI?
   a) 100 DACs for 10 hours,
   b) 50 DACs for 2 hours,
   c) 1 DAC for 20 hours,
   d) 0.1 DAC for 200 hours,
   e) none of the above.

10. You have been exposed to tritium and radioiodine, and from each of these exposures you have committed yourself to a weighted dose of 1 mSv. Obviously, the total weighted dose is 2 mSv, but what is the committed dose to the thyroid?

11. You work for 4 hours in an area containing 100 μSv/h of tritium, 0.2 mSv/h gamma and 0.1 mSv/h beta radiation. What deep and shallow dose will you receive as a result of this work?

12. You accidentally swallow 50 mL of moderator water with a tritium concentration corresponding to 10 years of operation (see p. 323). How many ALIs did you take in, and what dose commitment would you expect from this?

13. It was mentioned on page 338 that wetted skin will absorb water. In fact, for each square metre of wetted skin, the skin will absorb about 1 gram. Assume that you are working in the Moderator Enclosure fixing a leak, and your right hand and forearm get splashed with moderator water (the same water as in #12). What dose commitment might you expect?

14. Two of you are working in 30 μSv/h of tritium for one hour, but your friend is working much harder than you, and as a result breathes twice as much as you. What dose commitment would you expect each of you to get?

15. You worked in a tritiated atmosphere for two hours and gave a urine sample immediately afterwards, just before the end of your shift. You gave another one when starting work next day. The results were surprising: 0.12 mSv committed dose for the first sample and 0.17 mSv for the second. How come?

16. Workers are required to give a urine sample on their last day at Lepreau. If the tritium concentration of such a sample was found to be 9.5E4 Bq/L, what committed dose does it represent? Would this dose actually be assigned to that person?

17. Why is the tritium concentration in the moderator water so much higher than that in the PHT system, and why do we take considerable trouble to ensure that moderator water is kept out of the PHT system?

18. By drawing a smooth curve through the data points of Fig. 8.7 (on page 334), estimate the effective half-life of tritium in the donor of these samples, and calculate his dose from tritium for the week starting on day 10.

19. In your last two-week monitoring period you had the following bioassay results:
    Day 1 9.5E4 Bq/L, Day 2 3.8E5 Bq/L,
    Day 5 5.7E5 Bq/L, Day 13 2.9E5 Bq/L,
    Calculate the tritium dose you would have been assigned for this period. The Bioassay Update Report published on day 15 has a listing for "committed dose". What will this values be?

20. You have an acute (one shift) uptake of 0.80 mSv of tritium committed. By how much would you expect your tritium-in-urine concentration to increase?

21. Would your answer to the above question be the same if you were exposed to 10 μSv/h of tritium over 80 hours (2 weeks)? Explain.

22. Under what circumstances could you expect a radioiodine hazard to exist in the Reactor Building?

23. Which exposure gives you the greatest risk: 200 μSv/h of I-131, 200 μSv/h of I-133, or 200 μSv/h of I-135?

24. What about 200 μSv/h of tritium and 200 μSv/h I-131?

25. Roughly 100 g of dry dulse contain the same amount of iodine (130 mg) as a KI pill. Will this amount of dulse have the same "thyroid blocking" effect as one KI pill?

26. What should you do if the self-serve thyroid monitor indicates activity in your thyroid?

27. Stable iodine is used for "thyroid blocking". Which of the following mechanisms is responsible for the blocking action?
    a) Flooding the blood with stable iodine causes the body to excrete iodine in the urine at a much faster rate so that all radioactive iodine is excreted in 2-3 days and therefore little reaches the thyroid.
    b) The stable iodine completely satisfies the thyroid's need for iodine, thereby filling it completely so that no radioactive iodine is taken up.
    c) The stable iodine dilutes the radioactive iodine in the blood, so that a much smaller fraction of iodine taken up by the thyroid is radioactive.

28. You have taken an air sample in the Reactor Building. It has a beta/gamma activity equivalent to 500 Bq/m^3, but you have no idea what the radionuclides are. What committed dose rate does this represent? If you wanted more specific information, how would you get it?

29. What distinguishes transportable from non-transportable particulates, and which category is generally more hazardous?

30. What is meant by bioassay?

31. Why do we analyze urine samples routinely, but never bother with faeces?

32. There is a reason why the whole-body counter at Point Lepreau is adjacent to the Change Room. Do you know what it is?

33. Although the presence of I-131 will be detected in liquid scintillation counting of urine samples, this is not an effective way of screening for I-131 uptakes. Why not?