Fundamentals of Power Reactors

Module Three Radiation Protection

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Radiation Hazards in the Nuclear Industry

Table of Contents

Introduction

The Atomic Energy Control Board (AECB) is responsible to the Federal Government of Canada for the administration of the Atomic Energy Control (AEC) Act, and the promulgation of the Atomic Energy Control Regulations (C.R.C. 1978 C.365 AEC Regulations). These regulations are primarily intended to control the use of fissile materials or processes which could be used in a chain reaction or which generate ionizing radiations, in order to safeguard the health, safety and security of Canadians, and the Canadian environment.

The AECB has five directorates. Each is charged with specific responsibilities. One of these, the Directorate of Fuel Cycle and Material Regulation, is charged with the control of atomic and nuclear materials and processes; with the control of certain 'Prescribed Substances' (such as 'heavy water'), and with the control of the Packaging and Transportation of Radioactive Materials. This control function is achieved by means of a strict licensing process which approves the use of the material or process, and specifies the terms and conditions under which the atomic energy material may be acquired, used, and disposed of, packaged and transported.

Regulatory Control is maintained over the following industries:

- Uranium Mines, Mills, Refining, Conversion, and Fuel Fabrication Facilities,
- Power Reactors,
- Heavy Water Production Plants,
- Research Facilities, Research Reactors and Particle Accelerators,
- Medical Uses of Radioisotopes,
- Industrial Uses of Radioisotopes,
- Radioactive Waste Management,
- Packaging and Transportation of Radioactive Materials,

The sale, possession, use and disposal of Prescribed Substances, Prescribed Items, and Devices and Equipment containing Prescribed Substances is strictly controlled.

Licence applicants must submit comprehensive details of the design of the proposed facility and the manner in which it is expected to operate, details of the proposed use of the radioactive material and its effect on the site and surrounding area. The design must meet the prescribed limits on emissions that occur in both normal and incident operation. Compliance inspections are carried out by AECB staff to ensure that the terms and conditions of the licence are complied with.

Radiation Hazards Associated with Activities

Uranium for Power Reactors - The Uranium Decay Chain

The radioactive decay scheme of Uranium-238 is long and complex - a series of 16 nuclear transitions before the final (and stable) nuclide of the chain, Lead-206 (Radium G) is reached. The most important of the decay nuclides is Radium-226. This radionuclide was initially of importance as a source of gamma radiation for the treatment of cancer, but with the easy availability of radionuclides such as Cobalt-60, it is no longer widely used for this purpose.

Nowadays Radium-226 is of importance in radiation protection because it is the parent of the noble gas Radon-222, which is present in all uranium-238 bearing rocks and minerals. Radon-222 is a major radiological hazard. Because it is a gas, and because it has a half-life of 3.85 days, radon is able to diffuse out of the ground into the atmosphere, where it can be detected with relative ease, even though it is present at very low molar concentration.

The presence of radon is easy to detect because it is the parent of a series of four daughters, which decay by both alpha and beta emissions, with half-lives ranging from approximately 2×10^{-4}secs. to 27 minutes. These radioactive daughters are isotopes of polonium, lead and bismuth; they exist, after the radon atom emits an alpha particle, as atoms (i.e. as very small solid particles), which subsequently attach themselves to dust particles suspended in the atmosphere. Dust particles are easily captured on the surface of filter papers when air is drawn through the paper. The attached alpha and beta particle emitters can then be detected by means of detectors as simple as a geiger counter.

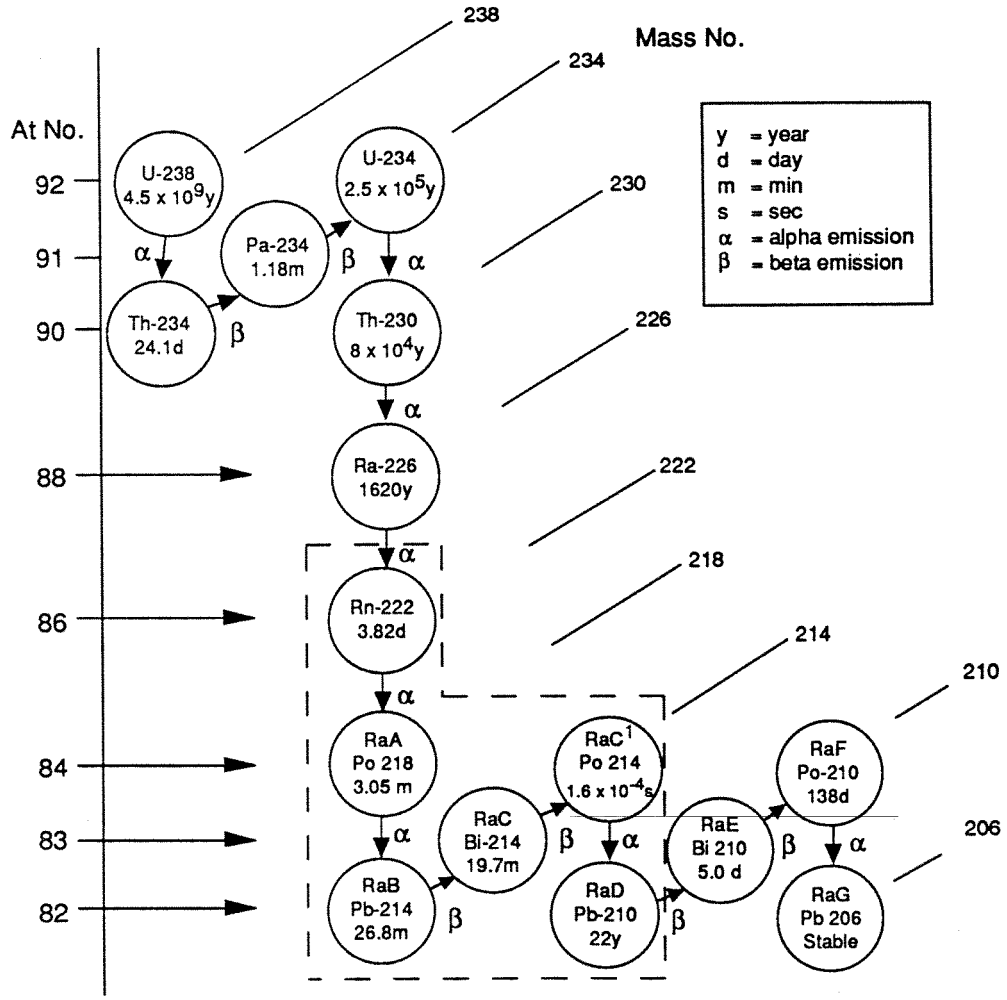
The 5th. nuclide after the initial radon decay (ignoring small percentage branches in the chain) is lead-210. It has a half-life of 22 years and is considered to be the end of the radon segment of the uranium decay chain.

Radiation Hazards in Uranium Mines

Uranium-238 occurs in ores (typically as Pitchblende) at a number of locations across Canada. Because of the demand for uranium for the Power Reactor Program, several mines are in operation. Some mines have already been worked out and have been closed, new mines some with high grades of uranium are being developed.

Because there is a high uranium content in the ores being mined, the concentration of radon in the air of the mines can be relatively high. This leads to high concentrations of radon daughters. These daughter products attach themselves to the dusts generated by the drilling, blasting and haulage operations that are characteristic of mining. Workers in the mine environment inhale this ‘radioactive’ dust, some of which is deposited in the respiratory system.

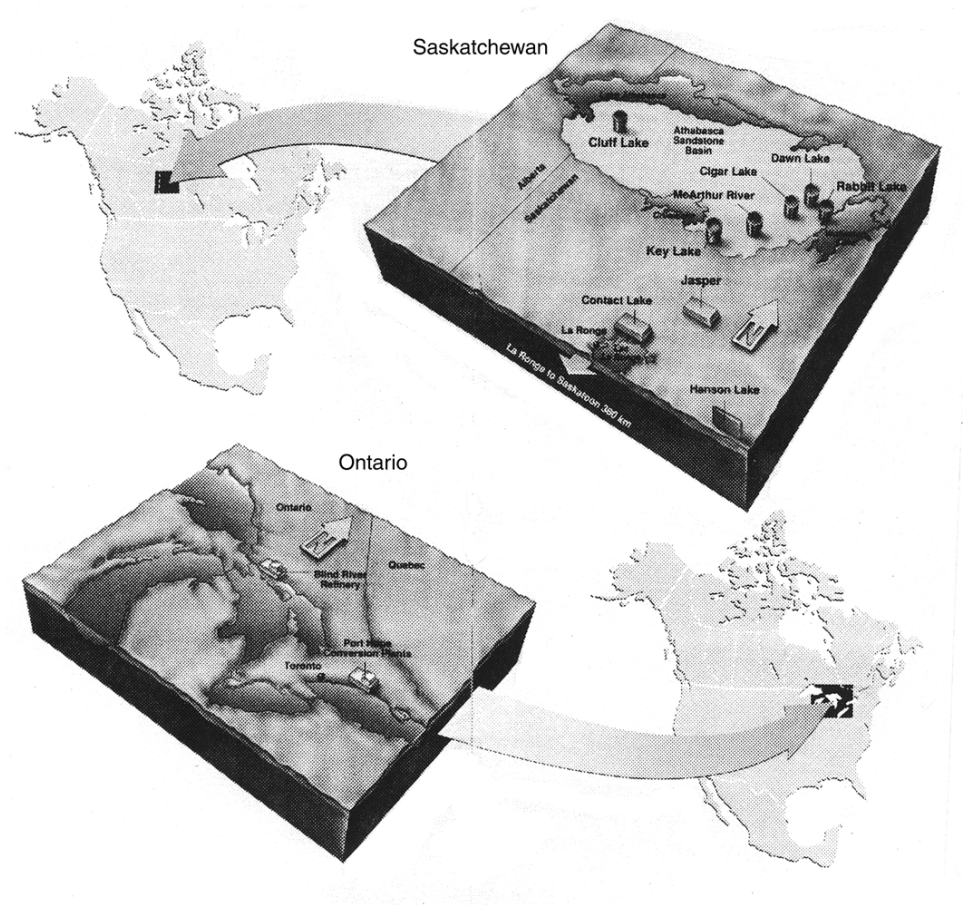
Figure 1
Principal decay scheme of the Uranium series



Alpha particle emissions occurring after deposition in the lung can penetrate into the cells of the lung causing biological damage, with the result that there is a significant incidence of lung cancers in uranium miners who have had significant occupational exposure to radon and its daughter products.

Radon daughter levels in mines can be reduced to acceptably low levels by taking measures to reduce dust concentrations in air (eg.water sprays), by installing efficient, high capacity ventilation systems to purge the radon, and by closing and blocking off side tunnels and stopes which are no longer in use. Coating the walls and surfaces of the mines with materials which reduce the amount of radon diffusing into the tunnels is potentially an effective technique to reduce radon concentrations.

Figure 2
Uranium Mine Locations



Uranium ores may also be associated with other toxic elements such as arsenic and nickel. Many mines in Canada also have ores which are rich in free silica (15% - 65%), leading to a high probability of silicosis. Control measures used for radon are also effective in reducing the silica dust concentration in the mine air. Uranium ores also have a high concentration of Thorium-232 approximately 4.0 lbs/ton as ThO_2. Thorium-232 has a complex decay chain; it too decays to an isotope of radon commonly called Thoron (Radon-220). This isotope has a half-life of 54 secs. so it is more likely to decay before it gets into the mine air. Its first daughter has a half-life of only 0.16 secs. so it does not have much time to attach itself to dust and subsequently be inhaled and deposited in the lung. The next decay product is thorium-B (lead-212), a beta emitter (less hazardous than alpha emitters) with a half-life of 10.6 hours. This means more of the thorium-B can be removed from the mine by the ventilation system before it decays to respirable particulate activity. The overall result is that thoron-220 is significantly less hazardous than Radon-220.

Figure 3

Cameco Uranium Reserves

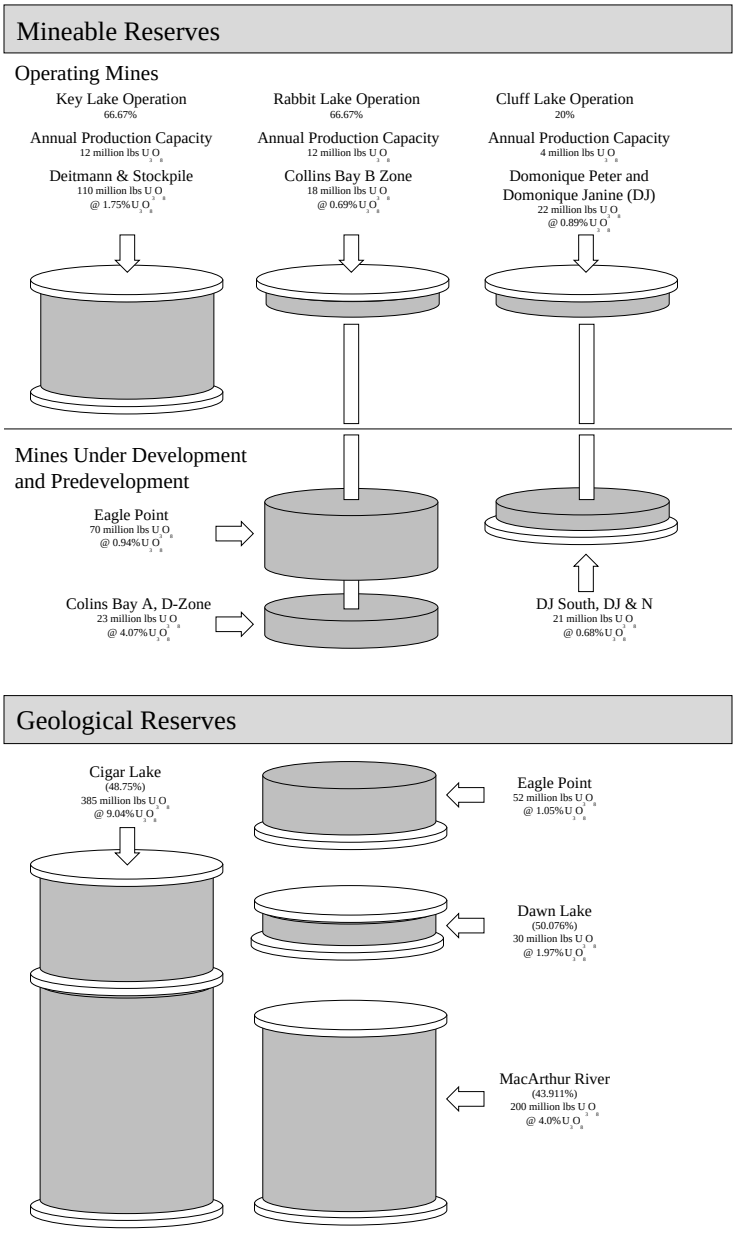
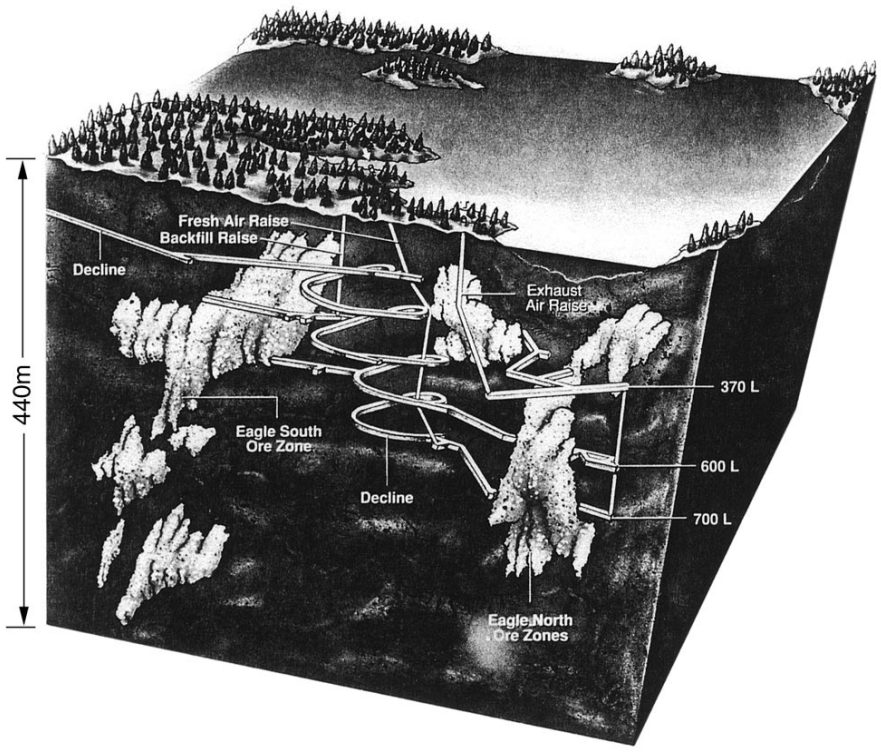


Figure 4

An artist's concept of the Eagle Point underground uranium mine depicts a view of Eagle North and Eagle South orebodies and their proposed development. Construction of the decline will begin in 1991. The depths (L) noted below surface in the mine are measured in feet using lake level as the zero point.



External radiation levels in the mines are of concern. The gamma radiation level in the vicinity of ores producing 6.0 lbs U_3O_8/ton is of the order of 0.016 mSv/h (1.6 mrem/h), and of the order of 0.006 mSv/h in the vicinity of ores producing 2.5 lbs/ton. This latter number, although seemingly small, will if a miner is exposed for 40 hours per week for a year add up to about one quarter of the current allowable annual occupational whole body radiation dose. It becomes more significant with current trends to reduce annual doses to radiation workers towards 10 or 20 mSv/year (1 or 2 Rem/year). Ores in some mines which have yet to be developed may be close to 20 or 30 lbs/ton; in these mines it is to be expected that control of whole body dose will be a major problem.

Figure 5
Ore is transported to the shaft for hoisting to the surface at the Port Radium uranium mine which opened in 1932

Uranium Milling, Refining and Conversion Facilities

Uranium ore is ground to small particles, then subjected to various physical and chemical processes to separate the uranium from the bulk of the ore. In the process, the uranium is purified and separated from the other chemical impurities present in the ore (eg.arsenic, nickel etc.). The process also separates the thorium, and the radioactive daughters of uranium. Since many tons of ore have to be processed to produce one ton of uranium, the process results in large quantities of tailings and chemical wastes which can be rich in nitrates, sulphates, nickel and arsenic. These wastes retain some uranium and also contain the radioactive daughters of uranium, radium 226 being of most concern. At the present time, 'tailings' are contained in engineered ponds, to prevent the dispersion of radium and other isotopes into ground water, lakes, streams and rivers. Uncontrolled drainage from tailings ponds can result in potentially harmful concentrations of radionuclides in waters which are used for domestic consumption some distance downstream. Plans for the eventual disposal of tailings include placing them in the bottom of open pit mines which have been specially engineered to contain these wastes.

The final product of the milling process is U_3O_8 usually called 'Yellow Cake'. This product is low in radioactivity because it no longer contains any radioactive

decay products. However, the Th-234 and Pa-234 daughters, which have half-lives of 24 days and 1.1 minutes respectively, immediately begin to grow into the yellow cake, and may present some problems in later processing, once equilibrium has been established (a period of 3 or 4 months). The decay process essentially stops at the third radionuclide of the chain (U-234) because it has a long half-life of 2.35×10^5 years.

U_3O_8 or 'yellowcake' is sent to a refinery where it is converted to UO_3. This material is then sent to the Conversion Plant.
The Conversion process has two end points:
i) UO_2 for CANDU Fuel
ii) UF_6 for export.

The Conversion Plant takes in UO_3 from the Refinery, treats it with ammonium hydroxide to precipitate a di-uranate which is then reduced in a furnace with hydrogen to produce UO_2.

In the case of UF_6, the UO_3 is treated with hydrogen fluoride to make the gaseous compound UF_6. This material is compressed into strong steel cylinders and exported to the USA where it is enriched in the U-235 isotope and manufactured into fuel for U.S. light water reactors .

Some UF_6 is returned from the U-235 enrichment process in the U.S.A and converted to 'depleted' Uranium metal. This material is used as a shielding material for gamma emitters, and as a material for balance weights, (for aircraft or sail boat keels) when a large weight has to put into a small space (the density of Uranium is about 19 g/cm^3). The conversion to metal may be associated with the release in the furnaces of the Th-234 and Pa-34 daughters of U-238. Both are beta emitters, and may require imposing additional radiation dose, and surface contamination control measures.

In the refining and conversion operations, dust control and ventilation are important to minimize the inhalation hazard associated with the Uranium. Good contamination control is necessary to minimize the uptake of uranium through the mouth (it is primarily a chemical toxin injurious to the kidney) and the spread of contamination to personal clothing and to areas outside the plant. Consideration must also be given to limiting the discharge of uranium from the ventilation systems to the environment.

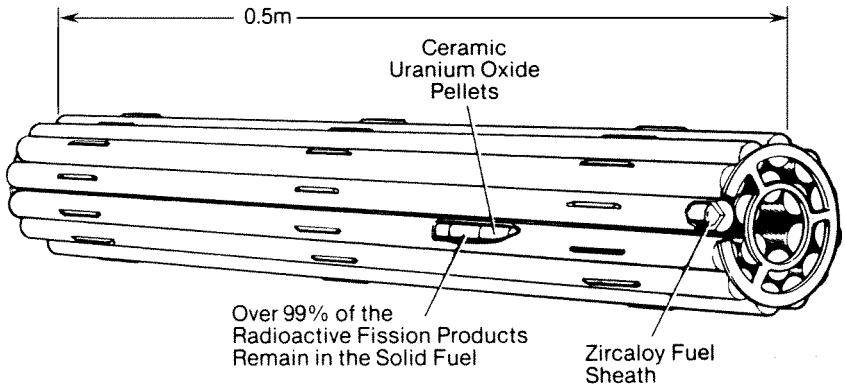
Radiation doses to refinery and conversion plant workers are small; 1.4 mSv/y (140 mrem/y) which amounts to about 2.8% of the current annual dose for radiation workers. Assessment of the amount of UO_2 inhaled is done by direct measurement using a detector placed close to the chest (lung monitoring). UO_2 is classified as a class Y (insoluble) compound. The dose to members of the public due to releases from the refinery is of the order of 0.25 mSv/y (25 mrem)

Fuel fabrication

In Canada, fuel fabrication presents only minor radiation protection problems. This is because the material is UO_2 which has to be pelletized, sintered. and inserted into fuel bundles. The problems are essentially confined to control of extremity (hand and finger) dose, and to dust and surface contamination control. Because UO_2 is an insoluble Class Y compound, lung monitoring is used to assess uptakes by inhalation.

External whole body radiation doses are small; approximately 1.0 mSv/y (100 mrem/y). Public doses are also very small, being of the order of 0.01 mSv/y (1 mrem/y).

Figure 7
Fuel Bundle



Power Reactors

Power Reactors are operated in New Brunswick, Quebec, and Ontario. They are large industrial units producing (typically) 600 to 900 Mw electrical output (1800 to 2700 Mw thermal).

Radiation hazards associated with CANDU power reactors are discussed in more detail in Part 2 of this Lecture.

Research Facilities

The most important nuclear research facilities in Canada belong to Atomic Energy of Canada Ltd. which is a Crown Corporation responsible to the Federal Government of Canada. This organization operates nuclear research reactors, fuel fabrication facilities (including the possibility that plutonium fuels are, or will be studied), hot cells, tritium recovery, storage and waste management facilities. From time to time it is possible that the quantity of fissile material in stock may warrant the installation of criticality control procedures and warning systems. AECL sites have a full range of radiation hazards requiring a full scale

radiation protection program and the employment of professional Health Physics personnel.

Research reactors

Eight university operated research reactors were licensed in 1990. One unit is operated by the Saskatoon Research Council and a second unit is operated by a commercial organization. Seven of these reactors are 'Slow-poke' reactors, one is a 5Mw(t) pool type reactor and there are two sub-critical assemblies.

Radiation hazards around these units are minor and varies with the applications. Design features such as the level of power produced, thermal neutron fluxes, and cooling systems determine the actual nature and extent of the radiation hazards at each facility. Although unlikely there is the potential of leakage of fission products from failed fuel elements. Other potential hazards are leakage of gamma and neutron radiations at reactor core shield penetrations, activation of core components leading to external gamma radiation hazards when components are removed from the core, and associated risk of a release of surface and airborne contamination. One of the major uses of the slowpoke reactor units is for neutron activation analysis of a wide range of sample materials. These samples will also be associated with the risk of external beta/gamma contamination (on the sample container), and the risk of contamination spread when the samples are handled and analyzed. Heavy water moderated units will have low levels of tritiated water in the moderator and coolant, together with some activation products. Air cooled units will also have some low levels of activation products such as Argon-41.

Particle Accelerators

At the end of 1990 licences were issued for 18 particle accelerators at research facilities, 2 at commercial facilities for the production of radioisotopes), 32 at medical facilities and 4 at facilities for sterilisation of materials. There are several types of particle accelerators; they can range in size from portable units (eg. as Van de Graaf generators incorporated into delayed neutron monitors for oil well logging) to large units installed in relatively complex buildings. Particle accelerators, as the name suggests, accelerate nuclear particles towards a target with the objective of causing collisions between the accelerated particles and the target nuclei. The usual result is the emission of neutrons or other nuclear particles, together with gamma or X-radiation. Large powerful accelerators are therefore usually placed in well shielded rooms, with formalized procedures for entry to the room and for ensuring that the room is empty prior to starting up the machine.

Some of the materials resulting from particle bombardment of a target may be radioactive. Occasionally targets (such as lithium tritide) may have defects which result in localized contamination of the machine or its surroundings.

Heavy Water

Heavy water is not radioactive . Heavy water plants are licenced because heavy water is an excellent moderator and is used in research and power reactors to achieve a chain reaction. The principle hazard at heavy water plants is the Hydrogen Sulphide gas which is used in the separation of heavy water from the very dilute lake water feedstock. Hydrogen sulphide gas is highly toxic and very corrosive, so frequent inspections of the thickness remaining in certain critical piping systems in heavy water plants is necessary. These inspections are sometimes done by means of portable radiography sources leading to potential external radiation hazards to workers.

Figure 8
Silhouetted against the night sky the series of towers of a heavy water plant at Ontario Hydro's Bruce Nuclear Development

Radioisotopes

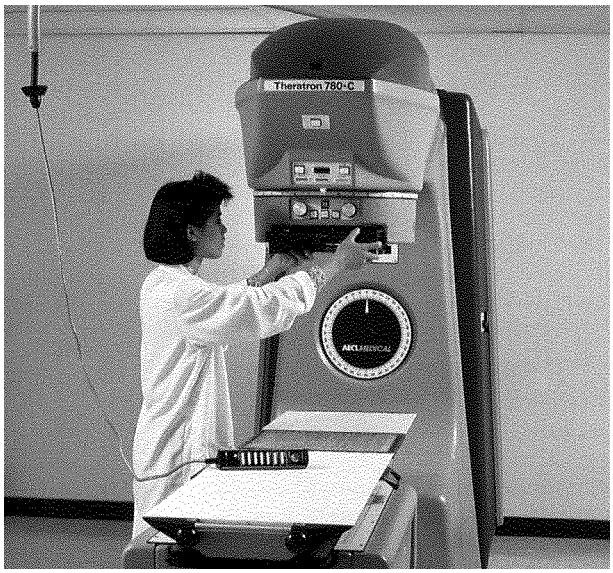
Radioisotopes have numerous applications in modern life. Some applications use massive sources (10^{15} Bq; 10^{5} Ci) of Cobalt-60 for the purpose of commercial sterilization of medical supplies and food irradiation, others use small kilo-Becquerel sized sources for medical diagnostic procedures etc.. Isotopes are used in agriculture, in research, and in industry. In many cases these sources are sealed sources. If the capsule is intact they are external radiation hazards only. If capsule fails the source material may be dispersed. Such incidents may result in the contamination of large areas, personal clothing and skin, and result in internal and external radiation exposures to both local radiation workers, and to members of the public.

Medical Uses of Radioisotopes

Radioisotopes are used both internally and externally for the treatment of, and investigation of, medical conditions. External radiation sources of gamma radiation are commonly used for the treatment of cancers at locations inside the body. Beta sources may be used both externally for the treatment of skin cancers

and cancers of the eye (Sr-90) and internally (implantations for irradiation of the pituitary gland, using separated Y-90, and injections of I-131 for treatment of thyroid cancers).

Figure 9
Medical use of Radioisotopes



Many radionuclides are used in investigative medicine, sometimes in surprisingly large quantities (10^8 Bq, several mCi). Such isotopes include radioiodines which has many isotopes with a wide range of half-lives, and beta and gamma emissions. Other isotopes in wide use include Te-99m, Cr-51, Fe-59, Hg-197, Hg-203, As-74, Ga-78, Tl-201. The radionuclides are used in a wide variety of tests such as thyroid function, renal function brain tumour scanning, blood studies etc.

Radiation hazards to laboratory staff concerned with preparation of sources for use will include both external and internal radiation. External radiation will include both whole body and extremity doses, related to handling small beta and gamma sources (both sealed and unsealed sources). Internal uptakes will be primarily related to ingestion of material from surfaces (related to spills and other careless handling techniques).

Radiation exposures to nurses may also be significant in so far as they must work in the vicinity of patients who are emitting radiation. These exposures will be primarily to external radiation, but there may be some cases where radioactive materials are excreted via the skin, resulting in some contamination transfer when patients are handled.

Industrial Applications

There are many industrial applications of radiation. Large sources are used for the sterilization of medical equipment and in food irradiation facilities. External

whole body radiation is the most important risk - facilities must be fitted with interlocks to prevent unauthorized or inadvertent entry to the exposure room when the source is exposed.

Another industrial application which deserves special mention is the use of portable radiography sources. These sources can be moderately large (Giga-to Tera-Becquerel, 2 to 50 Curie) sources of Cobalt-60 or Iridium-192. They are occasionally lost, mislaid or stolen, and they are sometimes misused or used without regard to proper care, with the result that on occasions the radiographer himself, his fellow workers, or members of the public receive significant whole body or partial body radiation doses.

Waste Management Facilities

Licenced waste management facilities range from the temporary storage ponds for irradiated nuclear fuels located at each of Ontario Hydro's major nuclear stations to permanent disposal facilities such as those located at Chalk River and operated by Atomic Energy of Canada Ltd. Radiation hazards to facility workers are essentially external radiation to the whole body. But there is potential for internal exposure from externally contaminated waste.

Potential routes for exposure of the public is via in-ground seepage and contamination of water supplies. Waste disposal and storage facilities are designed to prevent this occurring through engineering control of drainage and run-off water. In some cases there may be exposure to airborne radionuclides

Figure 10
Canisters: After five years in water storage, used reactor fuel can be stored in above ground concrete canisters licensed by the Atomic Energy Control Board.

such as the exposed dump sites used for refinery wastes (radon and thoron plus radioactive dusts raised by the wind). At Ontario Hydro's low level waste storage facility there may be some exposure to tritiated water diffusing away from absorbent materials (paper etc) which have been exposed to tritiated water in the reactor building environment. Doses to members of the public from these sources are very low.

Packaging and Transportation of Radioisotopes

Packaging and Transportation of radioactive materials is well regulated because the package containing the radioactive material may be sent by public carrier and because radioisotopes are frequently received from, or exported to, other countries. One of the primary responsibilities of the AECB is to ensure that Canadian package and transport container designs meet the requirements established by the International Atomic Energy Agency, and of course the Department of Transport.

Radiation hazards associated with the packaging and transportation of radioactive materials are largely external hazards. There are special regulations controlling the way radioactive materials can be shipped (i.e.the number of, and type of packages per shipment). These are designed to protect both the drivers of vehicles, and members of the public. If a road accident occurs, there may be some risk of internal uptake, but in general this is small because limits are set on the quantities of material that can be sent in certain types of packaging. In addition there are procedures which must be followed by drivers and emergency service Groups.

Very large shipments of high specific activity materials (e.g. irradiated fuel elements) are sent in specially designed - Type B containers, which have been tested in a variety of ways to ensure that they will contain the radioactive material in the worst possible accident.

Introduction: Description of the CANDU Reactor System

Natural uranium is a mixture of three isotopes:

U-238	approx.	99.3%
U-235	"	0.7%
U-234	"	0.005%.

Each of these nuclides has the property of undergoing spontaneous fission, with spontaneous fission half-lives in the range of 10^{16} to 10^{17} years. Each spontaneous fission decay is associated with the release of two or three fast

neutrons (i.e. neutrons with energies up to approximately 17 Mev) so as a result any accumulation of natural uranium is associated with a small flux of fast neutrons.

In the 1940's it was shown that the U-235 isotope would undergo fission readily when bombarded by 'thermal' neutrons (<0.025 ev). U-238 does not undergo fission with thermal neutrons but does absorb thermal neutrons to make U-239. This decays by successive beta emission through Np-237 to Pu-239. Although this process is not helpful in the early stages of reactor operation, Pu-239 ultimately makes a very significant contribution to the total power production in a power reactor.

The CANDU reactor system is an assembly of natural uranium in a suitable, lattice like arrangement, surrounded by heavy water (moderator). The function of the heavy water is to thermalize or slow down the fast neutrons resulting from the spontaneous fissions, so that they can cause fission of U-235. Once the process is started, and if the configuration and purity of the components is such that at least one neutron from every fission causes a further fission, then the process is self sustaining.

Each fission releases about 200 Mev of energy (mostly kinetic) which ultimately appears as heat. The reactor system therefore needs a heat removal system in order to produce useful power. Schematics of the CANDU reactor system are shown in Figures 11 and 12.

Figure 11
The Pickering Reactor System

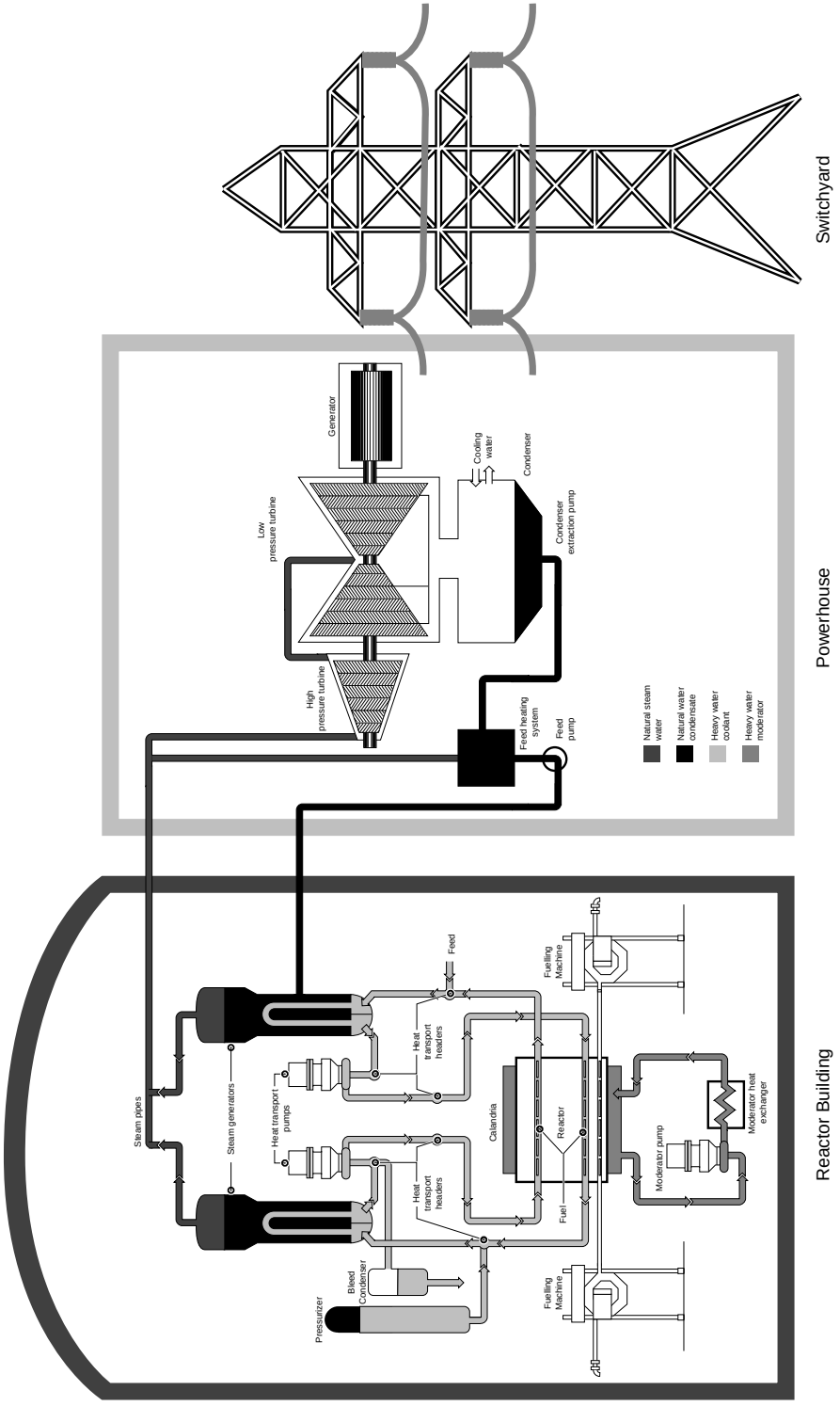
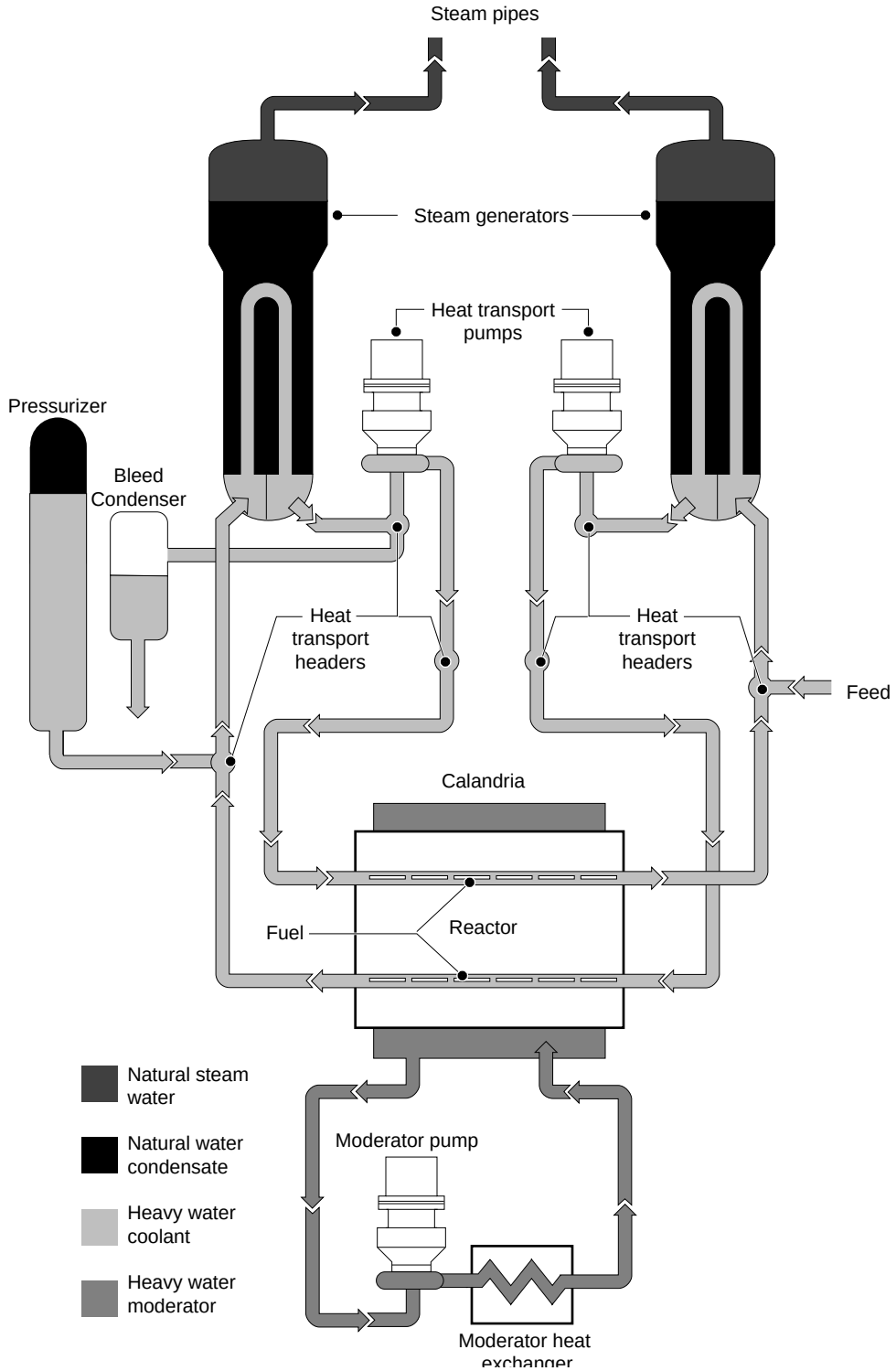


Figure 12
Reactor System Flow



A moderator tank (a calandria) in the form of a horizontal right cylinder contains heavy water. The vertical ends of the cylinder are pierced by a large number (approx.500) of horizontal, parallel tubes. A second independent tube (or channel) containing fuel bundles passes through these calandria tubes. This second tube is a high pressure tube (thick strong walls). The pressure tubes are connected to a header which connects to pumps. These 'Heat Transport Pumps' pump heavy water through the fuel bundles transferring the fission heat to a heat exchanger (or Boiler). In the boiler the hot heavy water converts light water on the secondary side to steam which drives a turbine and electrical generator.

Each fuel bundle, about 50 cm. long, consists of 37, 1.5 cm. diameter zircalloy tubes. Each tube is filled with ceramic natural uranium dioxide pellets. Typically there are 13 fuel bundles in each channel. The design of the fuel bundle is such that the heat transport cooling water can pass over the surface of each tube and remove fission heat efficiently. The operating temperature of the heat transport system is about 256^{o}C, at a pressure of about 11 MPa (1600 psi). The design of the reactor is such that a closure plug at both ends of a pressure tube can be removed by fuelling machines located on each face of the reactor. A new fuel bundle can then be pushed into a channel at one end, and an irradiated bundle is simultaneously pushed out of the channel into the fuelling machine at the other end. This fuelling system is designed to operate with the reactor at full power.

The displaced fuel bundle is highly radioactive, and is physically generating a lot of heat due to the energy of decaying fission products inside each of the UO^2 pellets. For this reason the bundle is continuously cooled and is transferred under water to a temporary 'irradiated fuel' storage pool.

For safety reasons each reactor is installed in a containment structure which has thick concrete walls to minimize radiation hazards from inside the building in the event that an accident causes a release of fission products. Each containment structure is connected to a 'vacuum building' whose function is to receive and condense steam (with entrained fission products) which may be released into the containment structure. This limits the pressure build-up in the containment building so that it will not suffer structural failure.

The reactor system is complex in so far as there are at least two independent emergency shut-down systems (control rods, moderator dump, or poison injection). The reactor power level is controlled by reactivity monitoring systems. There are also Hydrogen recombiners to prevent the build-up of hydrogen (or deuterium) produced by radiolysis, moderator cooling systems, purification systems, and an emergency core cooling system.

The reactor building also contains heavy water recovery systems designed to absorb and recover heavy water which has escaped as vapour from the high

pressure heat transport system, or which has evaporated into the air from spills (both moderator and heat transport). This system is also effective in reducing the concentration of tritiated water vapour in air inside the containment building). An internal air circulation and filtration system is used to minimize the volume of ventilation air which must be passed through the containment building. Charcoal filters intended to remove radioiodines are located outside containment.

Finally the whole reactor system and containment building are designed to withstand the effects of moderate earthquakes.

Radiation Hazards in an Operating System

General Nature of the Radiation Hazards

There are 8 different categories of radiation hazards in an operating CANDU generating station:

- Gamma Radiation
- Beta Radiation
- Neutrons
- Tritiated water vapour
- Radioactive noble gases and their daughters
- Radioiodines
- Airborne contamination
- Surface contamination

A detailed explanation of the nature and properties of each type of radiation is given in lesson 3 of this series.

Gamma Radiation
is emitted in the process of fission, and by the fission products themselves as they decay by radioactive disintegration. Gamma radiation is also emitted after a neutron is absorbed by the nucleus of atoms which form part of the structure of the reactor, or which are present as impurities in the heat transport or moderator systems. These atoms are usually radioactive and are known as Activation Products. They may emit beta and/or gamma radiation as they decay to a stable nuclide.

Beta Particles
are emitted by fission and activation products. Because external beta radiation can only irradiate skin and shallow depth tissues, it is frequently underestimated as a radiation hazard.

Fast Neutrons
are released in the process of fission, and in reactions between high energy gammas and deuterium (photoneutrons)

Thermal (or slow) Neutrons
are low energy neutrons. They are the end result of collisions between fast neutrons and the deuterium in the moderator system. Thermal neutrons react with U-235 to cause fission, and with reactor components and impurities to form radioactive 'activation' products.

Tritiated Water
is formed when moderator water is irradiated by neutrons. Tritium is radioactive and emits low energy beta radiation. In the form of water and water vapour in air, it is easily absorbed into the body, where it is an 'internal' radiation hazard.

Radioactive Noble Gases and Daughter Products.
Small holes in fuel element sheaths allow fission product noble gases to escape from the fuel into the pressurized heat transport heavy water. When the heat transport water escapes from the pressurized system, it carries with it some of these radioactive noble gases. The gases of interest are the isotopes of Krypton and Xenon.

Radioiodines
are fission products which are volatile and which will leak through holes in fuel element sheaths into heat transport system water. From there they may escape to occupied spaces in reactor buildings. They may also leak out of fuel elements in the irradiated fuel storage pool, resulting in dissolved iodines in the water, and airborne radioiodine in the atmosphere of the storage bay rooms.

Long lived Airborne Radioactivity
is particulate radioactive material which becomes airborne. It is hazardous because it may be inhaled and deposited in the respiratory system. It consists of longer lived fission and activation product nuclides.

Surface Contamination
is caused when beta/gamma radioactive materials are deposited on surfaces. The source of deposition may be insidious - such as the accumulation of 'fallout' from low levels of long lived airborne contamination, over long periods of time, or it may be acute as in the deposition from high levels of airborne contamination over short periods of time.

Important sources of surface contamination include fuel machine service areas - highly radioactive heat transport 'crud' from inside the machines is a well known source of fission and activation product contamination. Spills of radioactive liquids may deposit solids as they evaporate - people walking through the spilled liquid or the solid residue can 'track' contamination throughout an entire building in very short periods of time.

Improperly controlled decontamination tasks can also be serious sources of surface and airborne contamination.

Stations Systems and Hazards

Introduction

It is important to realize that radiological conditions at a given location in a reactor building can change very rapidly. The changes may be related to changes in reactor operating conditions, or to activities being carried out by other work groups in other parts of the containment building. Knowledge of work being carried out in the containment building, and the availability (i.e. to hand) of radiation monitoring equipment, can enable workers to keep informed of changing conditions.

Reactor Vaults

The reactor vaults give access to the two faces of the Calandria. It is in this area where re-fuelling operations take place while the reactor is at full power. The vaults are normally inaccessible areas due to high levels of neutrons, mostly thermal, and gamma radiation from the reactor faces, and because of the possibility of high levels of gamma radiation from irradiated fuel in the fuelling machines.

The heat transport/fuel channel end fittings may be leaking heavy water (usually as steam) which may or may not be visible. The vaults frequently contain high concentrations of tritiated water vapour. If there is fuel with defects in the reactor radioactive noble gases and their short lived daughters, radioiodines, and miscellaneous long lived fission products may be present. Carbon-14 is produced in the annulus gas system, the space between the calandria tubes and the pressure tubes. This is encountered in the vaults when pressure tube replacement work is being carried out.

Access to the vault area when the reactor is in operation is not allowed - although carefully controlled access for special purposes may be permissible when the reactor is at low power. Access is normally allowed after reactor shut-down, but only after careful preliminary surveys to determine the radiological conditions and assess the precautions which must be taken.

Boiler Rooms

Access to boiler rooms is not allowed when a reactor is at full power. High levels of gamma radiation result from the production of the short lived activation products N-16, N-17, and O-19, (T1/2 = 7 , 4 and 23 seconds respectively), produced when neutrons interact with oxygen in D_2O. Photoneutrons may also be emitted from the circulating heavy water. There will also be radiation fields from long lived activation and fission products deposited on piping valves etc..

Steam leaks from heat transport system components in the boiler room area may result in relatively high airborne contamination such as tritiated water vapour, noble gases. radioiodines, and both short and long lived activation products.

Beams of neutron radiation may occur in the vicinity of penetrations in the concrete reactor shield such as the penetrations for the Reactivity Control Mechanisms.

In shut-down conditions, gamma and neutron radiation related to the operation of the reactor will disappear, but other hazards mentioned above will be unchanged. When systems are broken open for maintenance purposes there may be relatively high exposures to beta radiation from contamination deposited on the internal surfaces of the components. This type of contamination may give rise to both airborne and surface contamination hazards, requiring special control procedures. Items giving rise to this type of problem include the heat transport pump impeller, IX columns and filters, fuelling machine maintenance, work inside boilers, and decontamination work on badly contaminated items.

Fuel Transfer Rooms

High levels of gamma radiation, airborne contamination (noble gases, radioiodines, and other volatile fission products) and the potential for high levels of surface contamination are not uncommon in this room. During incident conditions involving dry, overheated fuel, long lived fission products in air may exceed 10^3 times the Derived Air Concentration.

Fuel Machine Maintenance Area

Hazards in this area are related to maintenance of equipment which may be contaminated with activation products and fission products from defective fuel bundles. Bruce 'A' fuelling machines do not have a 'flow through' arrangement for heavy water in the re-fuelling process so these machines can get heavily contaminated internally with activation and fission product 'crud'. This material collects in the dead spot at the end of the fuel channel near to the shield plug. When the shield plug is drawn inside the fuelling machine the crud is also floated into the machine. (Plans are in hand to install a flow through system on these machines.)

Fuelling machines at all stations can become very badly contaminated by damaged fuel bundles, resulting in very high beta radiation dose rates when the machines are opened up for maintenance. There will also be the potential for high airborne and surface contamination.

This system is also a producer of 'hot particles'(small highly radioactive particles). Hot particle problems are really evidence of poor contamination control procedures.

Activity Concentrators

Strainers, filters, ion exchange columns, vacuum cleaners, mops, floor cleaners, air filters, D_2O driers etc are activity concentrators, and may on occasion give rise to some unexpectedly high contamination problems.

Irradiated Fuel

Mishandling of fuel bundles may result in very high radiation levels in irradiated fuel handling rooms and surrounding areas. Damaged bundles may also emit gaseous and volatile fission products, creating airborne and surface contamination hazards.

Failure of Systems and Procedures

The failure of systems such as ventilation, fixed area gamma monitors, access control or radiation procedures may contribute to the development of hazardous situations.

Gas Impurities in Reactor Systems

Air leakage into reactor systems or make-up cover gases in systems can result in gamma hazards from Argon-41 which is an activation product from stable Argon-40 in air.

$$^{40}Ar_{18} + {}^{1}n_{0} > {}^{41}Ar_{18} + \text{gamma radiation}$$

The annulus gas system will generate Ar-41 if air leaks into it.

Radiation Protection Theory & Program for a Typical CANDU Station

Table of Contents

Review of Radioactivity

Radioactive atoms are unstable atoms which emit particles and radiation from their nucleus in an attempt to achieve stability. The atom which is left is usually an atom of a new element and may or may not be stable. The process is known as radioactive disintegration or Radioactivity.

A material which is radioactive is usually known as a radioactive source. The strength of a radioactive source is measured by the number of atoms of the material which are disintegrating in a given period of time. The SI unit for source strength is the Becquerel (Bq). The Becquerel is defined as 1 disintegration per second (1 dps). A source of strength 20 Bq would have 20 atoms disintegrating per second. The Bq is a relatively small unit of radioactivity so that multiples of the unit are in common use. These are:

- kBq (KiloBecquerel) = 10^3 Bq
- MBq (MegaBecquerel) = 10^6 Bq
- TBq (TeraBecquerel) = 10^9 Bq

Sub-units of the Bq are:

- Mbq (milliBecquerel) = 10^{-3} Bq
- µBq (microBecquerel) = 10^{-6} Bq
- nBq (nanoBecquerel) = 10^{-9} Bq

An old unit used for measuring the strength of a radioactive source is the Curie (Ci). It is still commonly used in North America. This was chosen to be equal to the activity of 1 g of radium. It is an activity of 3.7×10^{10} disintegrations per second. This is a large unit compared with the Bq (1 Ci = 3.7×10^{10} Bq) so that sub-units are in common use. These are:

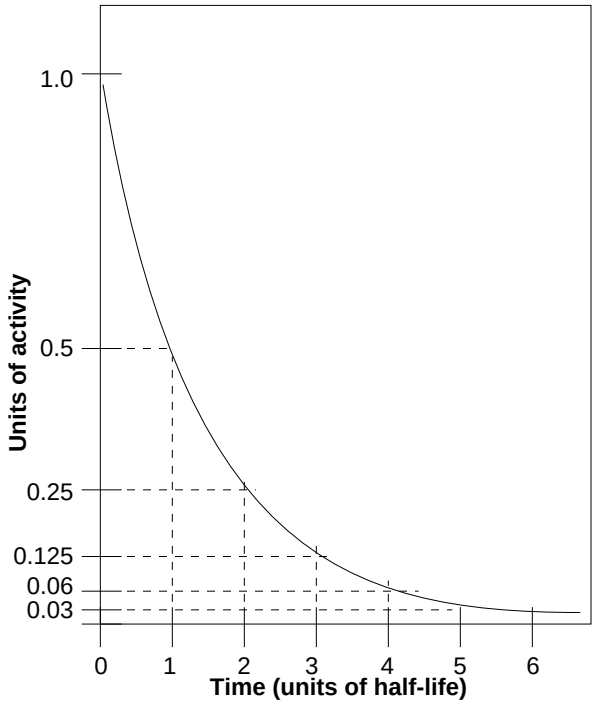
- mCi (milliCurie) = 3.7×10^7 dps
- µCi (microCurie) = 3.7×10^4 dps
- nCi (nanoCurie) = 3.7×10 dps
- pCi (picoCurie) = 3.7×10^{-2} dps
  (2.2 disintegrations per minute)

Half-life

It has been found that the strength of a radioactive source is not constant. It becomes less strong (or decays) as time passes. When the strength of a source is plotted against time the graph shown in Figure 1 is obtained. If the strength of the source becomes reduced by half in 1 hour then after another hour it will become reduced by half again to ¼ of the original strength; after another hour it will be reduced by half again to ⅛ of the original value. This time for a source to lose ½ of its original value is known as the half-life of the radioactive material. Half-lives of radionuclides may vary from fractions of a second to thousands of years.

The half-life is characteristic of a radionuclide. A radioactive material may be identified by its half-life. For example if an unknown radioactive material is found to have a half-life of 5.27 years then it is likely to be cobalt-60. Promethium-147 and sodium-22 both have a half-life of 2.6 hours. Other characteristics of their decay processes would enable the radioisotopes to be separately identified.

Figure 1
Illustration of Radioactive Decay



Ionization

The particles emitted when an atom disintegrates carry electrical charges. As these particles pass through matter, their electrical charge interacts with the negative charges carried by the orbital electrons surrounding each atom. The energy transferred may eject one of the orbital electrons of the atoms of the material. The ejected electron is a called a negative ion; the remainder of the atom is called a positive ion.

Both ions are collectively referred to as an 'ion pair'. The energy transferred to the ejected orbital electron is taken from the nuclear particle, which eventually loses all of its energy in this type of collision.

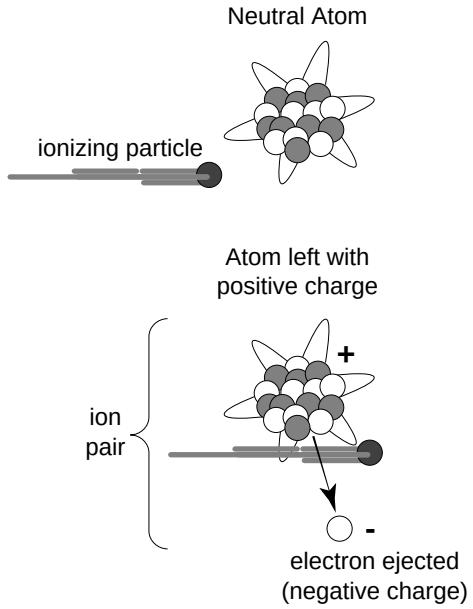
Two nuclear particles are of most interest:

the **Beta particle** is a fast electron; it carries a single negative charge, and has a mass of approximately $\frac{1}{1840}$ of the mass of a hydrogen atom.

the **Alpha particle** is the nucleus of a helium atom; it carries two positive charges, and has a mass of approximately 4 times that of a hydrogen atom.

Figure 2

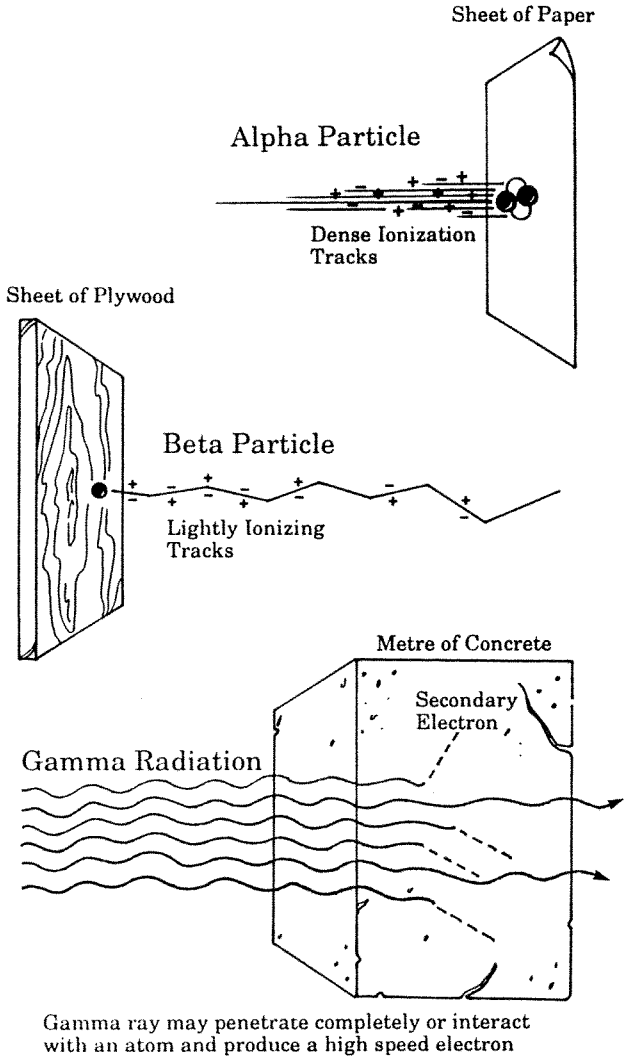
Ionization



Because of its double positive charge and usually slower speed, the alpha particle interacts very strongly with matter to produce dense ionization in its tracks. It produces about 50 ion pairs per micron of track length as it passes through water. The result is that the alpha particle loses its energy very rapidly - it is easily stopped by a single sheet of paper.

Beta particles interact less vigorously with matter because they carry only a single negative charge and are usually travelling at much higher speed. They produce about 3 to 5 ion pairs per micron of track length. The result is that they can travel through greater thicknesses of matter. See illustrations in Figure 3. X and Gamma ray photons interact with matter in a slightly different way but also produce ionization. Gamma photons may pass through matter without interacting with the material at all, but occasionally a photon may interact with an atom and as a result will transfer a large fraction of its energy to the atom. The gamma photon may be completely absorbed at this point or may go on in a different direction with reduced energy. The electron which receives the energy then produces ionization in the same way as a beta particle.

Figure 3
Penetrating Power of Various Radiations



X and Gamma rays are therefore very penetrating. The result is that they can pass through considerable thickness of matter, leaving sporadic clusters of ionization at each point of interaction with an atom.

Neutron radiation may be produced by a variety of nuclear reactions but around a reactor they are normally encountered as a result of fission in the reactor. Initially the neutrons are energetic, or fast neutrons, They are slowed down (or thermalized) by elastic (billiard ball type) collisions with light nuclei such as hydrogen or deuterium atoms.

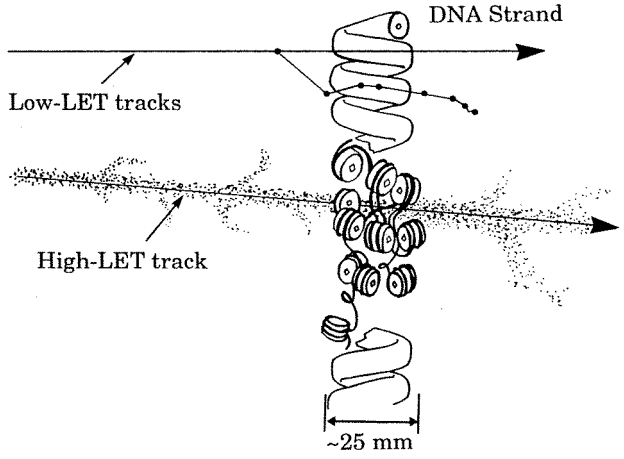
In the case of fast neutrons, in tissue or heavy water, the 'billiard ball' collisions transfer energy to the protons or deuterons, which then behave somewhat like alpha particles, causing relatively dense ionization tracks in their wake, and losing energy rapidly.

When fast neutrons have lost most of their energy, they become slow or thermal neutrons, At this time they carry less than 0.025 ev (electron volts) of energy and have a velocity of less than 2200 m/sec.. These neutrons are absorbed into and 'transmute' the nuclei of nearby atoms, usually with the emission of a gamma photon in an (n,gamma) reaction. The transmuted nucleus is an activation product; it may be unstable, and undergo radioactive disintegration at a later time.

Specific ionization and linear energy transfer

Two terms that are used to describe the way in which particles or radiation deposits energy along their tracks are Specific Ionization and Linear energy transfer. Specific ionization is the number of ion pairs produced per unit of track length in a specified material (frequently air or water). Linear Energy Transfer (LET) is the amount of energy deposited in a material per unit of track length. Because the amount of energy required to produce one ion pair is approximately constant (35 ev/ion pair) LET is approximately proportional to Specific Ionization. The way in which energy is deposited by the ionizing radiation is important in radiation protection because It has been found that the amount of biological damage caused in tissue increases as the LET increases for the same amount of energy deposited. This is known as the Relative Biological Effectiveness of the radiation. It is one of the most important factor considered by the International Commission on Radiological Protection (ICRP) when they make recommendations for the protection of workers and the public.

Figure 4
DNA Damage from Radiation



It is thought that an important consideration is the concentration of damage in the DNA strands of cells when ionizing particles pass through the cell.

Radiation Dose Units

The Roentgen

The roentgen is an old unit but it is occasionally encountered. It was the first precisely defined unit of radiation exposure. It was defined in terms of the amount of ionization or electrical charge it produces in air. It was a useful unit in its time but it has become obsolete because the measured exposure dose in air is not necessarily the same as the exposure dose in other materials. In living tissue, the major difference occurs when low energy photons irradiate bone. Then, the energy deposited in air as a result of an exposure of 1 roentgen may be several times lower than the energy deposited in bone exposed at the same point.

The Rad

The rad is a unit that measures the energy deposited in matter. It is defined as an amount of energy deposited in 1 g. of matter.

1 rad = 100 ergs/g.

This definition avoids the problem associated with the roentgen. However, while the Rad is a measure of the amount of energy deposited in matter, it does not take account of the amount of damage actually produced in living tissue by different kinds of radiation because of factors such as LET.

The Rem

The Rem (rad equivalent man) was introduced to take account of the damage produced by various kinds of radiation. It is a unit of Dose Equivalent.

REM = RAD x QF

where QF is a factor related to Specific Ionization.

The above units (particularly the Rad and Rem) are still widely used in North America. However the officially recognized international units for radiation protection purposes are SI units.

S.I. Units of Radiation Measurement

The new units that have been adopted by the international radiation measurement and protection community are based on SI units of energy and mass. The units are the Gray (Gy) and the Sievert (Sv).

The Gray - The Unit of Absorbed Dose

The Gray (Gy) is the SI unit of absorbed Dose, usually designated D, and represents an energy deposition of 1 joule in 1 kg of a material.

1 Gray = 1 Joule/kg.

Now 1 Joule = 10^7 ergs, and 1 kg = 1000 g.

therefore 1 Gray = 100 rad (of tissue or other matter).

Weighting Factors

The Sievert is the SI unit used for human radiation protection purposes. The Sievert replaces the rem. It takes into account two factors. The first of W_T these take into account the fact that different types of ionizing radiations, even if irradiating the same organ in the body, may have different biological effects because of the differing ways the energy is deposited in the organ (specific ionization). The second factor taken into account in the development of the Sievert is the differing biological sensitivity of the different organs of the body. For the same dose of identical radiation there is a higher incidence of cancer induction in some organs than others. These two factors are applied as weighting factors to obtain the dose that is significant in radiation protection.

W_T and Equivalent Dose

In radiological protection it is the absorbed dose averaged over the organ and weighted for the type of radiation that is of interest. The weighting factor is called the radiation weighting factor WR . Values for the Radiation Weighting Factor are given in Table 1. The organ dose obtained as a result of this averaging and weighting calculation is called the **Equivalent Dose , H_T** .

$$HT = \sum_R W_R * D_{T.R}$$

Where $D_{T.R}$ is the absorbed dose in Gray averaged over the tissue or organ T from radiation R, and W_R is the **Radiation Weighting Factor.** H_T is measured in Sieverts. Since W_R is dimensionless, the equivalent dose (H) has the same dimensions as the Gray (J/kg).

Table 1

Radiation weighting factors (wr)

Type and Energy Range		Radiation Weighting Factor
Photons, all energies		1
Electrons and muons all energies		1
Neutrons	< 10 Kev	5
	10 - 100 Kev	10
	100 kev - 2 Mev	20
	2 Mev - 20 Mev	10
	> 20 Mev	5
Protons	> 2 Mev	5
Alpha Particles, Fission Fragments, Heavy Nuclei		20

Note:

All of the above values relate to the radiation incident upon the body, or for internal sources, emitted from the source.

Effective Dose

Tissue Weighting Factors

In radiation protection not all organs are equally sensitive to the effects of ionizing radiation for the same equivalent dose. The biological effect of most interest is the increased risk of cancer. Another weighting factor, the tissue weighting factor (W_T) has been introduced to enable the effect of a combination of different doses, to different organs or tissues, to be summed in a way which correlates with the biological effect.

Effective Dose

The effective dose (E) measured in Sieverts is the sum of the weighted equivalent doses in all the tissues and organs of the body:

$$E = \sum T \; W_T * H_T$$

where H_T is the equivalent dose in organ T and W_T is the weighting factor for organ T.

Tissue weighting factors are given in Table 2 below.

Table 2

Tissue Weighting Factors.

Tissue or Organ	Tissue Weighting Factor (W_T)
Gonads	0.2
Bone Marrow (red)	0.12
Colon	0.12
Lung	0.12
Stomach	0.12
Bladder	0.05
Breast	0.05
Liver	0.05
Oesophagus	0.05
Thyroid	0.05
Skin	0.01
Bone Surface	0.01
Remainder	0.05

Committed Equivalent Dose

When an internal exposure occurs and radioactive material is taken into the body it stays there for a time depending on the material and how it is used by the body. Some radionuclides may remain for a long time. This is the case for radionuclides that are incorporated in bone. Others such as tritium oxide which becomes part of body water are eliminated quite quickly. An intake of radioactive material will deliver a dose rate to the organ in which it is deposited. The total dose that the organ will receive can be obtained by summing (or

integrating) the dose rate over time. This dose is called the committed equivalent dose ($H_T(\tau)$) where t is the integration time, in years, following the intake. If τ is not specified the value to be used is 50 years for adults and 70 years for children.

The Committed Effective Dose ($E(\tau)$) is similarly defined.

Collective Equivalent Dose and Collective Effective Dose are terms used to describe population dose. This is simply the sum of the dose being received by members of a population e.g. if 10 workers receive 20 mSv and 20 receive 30 mSv the population dose to this group of workers is (10 x 20 + 20 x 30) person mSv or 800 person mSv.

Biological Effects of Radiation

Introduction

The International Commission on Radiation Protection (ICRP); set up in 1928, studies the effects of radiation on man and makes recommendations from time to time, which are based upon a distillation of the current state of knowledge of the effects of radiation on human populations. They provide an interpretation of the best means of protecting people from the adverse effects of radiation, whilst utilizing the benefits and advantages to be obtained from the use of radiation, or processes giving rise to radiation as an unwanted by-product. The ICRP and UNSCEAR, the United Nations Scientific Committee on the Effects of Atomic Radiation regularly review and issue reports on the effects on the biological effects of ionizing radiation.

Deterministic and Stochastic Effects

When ionizing radiation is absorbed in tissue, about half of the energy is used to cause excitation in the molecules; the remainder causes ionization. Ionization changes atoms and may therefore alter the structure of the molecules containing them. Excitation of atoms in molecules may also alter the structure of molecules if the excitation energy exceeds the binding energy of the molecules. If the affected molecules are in a cell then the function of the cell may be damaged. Damage to the DNA in a cell is the worst effect because this may result in a damaged but modified and viable cell.

If enough cells in a tissue or organ are changed, damaged, or killed to the extent that the tissue or organ is unable to reproduce and function normally, then there is a detriment which the ICRP now refers to as **'Deterministic',** that is it ***has a threshold*** for clinical effect, below which damage is not observed. The threshold may have a range but there will be a dose at which no effect is observed in any individual.

A changed or damaged somatic cell may retain its reproductive capacity and give rise to a clone of modified cells which may later result in cancer. Also a modified germ cell in the gonads of an irradiated individual may pass on incorrect hereditary information which may cause severe harm to some of the person's descendants. These somatic and genetic effects are called **'Stochastic',** that is there will be an increase in the frequency of occurrence of the effect in the population exposed with ***no threshold***. The increase in frequency of occurrence is proportional to the size of the population dose.

Data on the effects of ionizing radiation on humans comes from animal studies and studies of human populations that have been exposed. The most important study group is the Japanese exposed in the bombing of Hiroshima and Nagasaki. Studies of their health and causes of death have been ongoing since the second world war and continues. Other sources of effects on humans comes from patients exposed in medical treatment and persons exposed accidentally in industrial accidents e.g industrial radiography.

Deterministic Effects

Some important identifiable effects which have a threshold or are thought to have a threshold are mental retardation resulting from in utero exposure, skin response, cataracts of the lens of the eye, sterility and the acute radiation syndrome.

Mental Retardation

Japanese evidence suggests that there is a reduction in IQ level and an increase in the number of children born with severe mental retardation. The magnitude of the effects are not precisely known but are about a reduction of 30 IQ points/Sv. and an increase of about 0.4 mentally retarded children/Sv

Immediate Skin Effects

The skin will exhibit no effects unless a threshold dose is exceeded. A dose in the range of 5 to 8 Sv will cause reddening of the skin in most persons exposed and a dose of 10 to 20 Sv will cause ulceration. These radiation caused ulcers take a long time to heal.

Immediate Effects of Whole Body Irradiation (Acute Radiation Syndrome)

The **'Acute Radiation Syndrome'** is the term applied to the effects shown by persons exposed to high doses of radiation. If the absorbed doses are high enough, some of the exposed persons may die. Death is usually the result of severe cell depletion in one or more of the vital organs or tissues of the body. The effects that may cause death depend on the level of the dose and are usually classified as effects on the blood system (haemopoetic), on the gut (gastro-intestinal) and central nervous system. The ranges and times at which these occur are given in Table 3.

Table 3

Body Systems Affected by High Radiation Doses

Dose range (sv)	System affected	Time of onset
>10	Central Nervous	5h - 3 days
5 - 9	Gastro-Intestinal	3 - 7 days
2.5 - 5	Blood	1 wk - 2 mths

The level at which death occurs as with most biological effects is variable and occurs in a range. The lethal dose which will result in the death within 60 days of 50% of those exposed is in the range 3 to 5 Gy. Below the lethal range exposed persons will have clinical and observable symptoms - vomiting, loss of hair, reduction of blood cells. These are summarized in Table 4.

Table 4

Effects of Single High Doses of Radiation

Dose (gamma gy)	Probable effects
0 - 0.25	No observable effects. Some chromosomal damage can be detected.
0.25 - 1.0	Nausea and some slight blood changes (loss of cells) in a few people.
1 - 2	Nausea, possible vomiting, blood changes, sterility
2 - 3	Vomiting, severe blood effects, two week latent period followed by loss of appetite, diarrhea, possible death but recovery likely for most healthy individuals.
3 - 5	Vomiting, diarrhea, short latent period followed by epilation, (loss of hair), loss of appetite, general malaise, some deaths in first week, death of 50% of individuals exposed for about 4.5 Gy.
>5	All earlier symptoms, death as early as 2nd week with eventual death of all highly exposed individuals.

Stochastic Effects

Long term studies of exposed populations have identified two important stochastic effects. These are:

- genetic effects, that is effects expressed in the offspring of the exposed population and,
- somatic effects - this is effects expressed in the exposed population. The only somatic effect of significance is an increase in the incidence of cancer

Genetic Effects

If germ cells are damaged by radiation this can express itself as disorders in descendants of the persons exposed. These disorders may range from mild to severe malformations with loss of function and premature death. The experimental evidence shows that these hereditary defects will occur primarily

in the first and second generations after exposure. The ICRP in carrying out risk assessments of exposure to ionizing radiation attributes the whole detriment to the exposed individual. This is to avoid the need to assess the detriment to future generations. For low doses and low dose rates the ICRP has estimated that the risk for severe effects and years of life lost over all generations as 1.3×10^{-3}/Sv for the general population and 0.8×10^{-3}/Sv for workers. This means that for 10000 workers exposed to 1 Sv there will be 8 additional children with severe hereditary defects. This is in a natural background rate of about 600.

Increased Incidence of Cancer

The only significant somatic effect observed in exposed populations has been an increased incidence of some types of cancers. There is essentially no data which provides good risk estimates at low doses (< 0.2 Sv) so that risk estimates have to be estimated from the information available from populations exposed to high doses. The risk estimates have been made by extrapolating from high doses assuming a linear relationship between effect and dose, making allowance for the reduction in effect that may occur with lower dose and dose rate. Risk estimates have been made for fatal and non fatal cancers and for the whole population and adult workers. These are summarized on the next page.

Table 5

Probability Coefficients for Stochastic Effects

	Detriment (10^{-2} Sv^{-1})			
Exposed Population	**Fatal Cancer**	**Non-Fatal Cancer**	**Severe Hered. Defects.**	**Total**
Adult Workers	4.0	0.8	0.8	5.6
Whole Population	5.0	1.0	1.3	7.3

The risk factor for fatal cancer means that an atomic radiation worker exposed to .01 Sv (1 rem) per year for 40 years would have an added risk of fatal cancer of $4.0 \times 10^{-2} \times 40 \times 10^{-2} = 1.6\%$. This may be compared with the risk of this occurring naturally of between 20% and 25%.

External and Internal Exposure

Dose limits apply to both **External** exposure and **Internal** exposure. External exposure is where the source of the radiation is outside or external to the body. Internal radiation is where the source has in some way entered the body .

The most common routes of entry are by **Inhalation, Ingestion**, and for a few radionuclides **Absorption through broken or unbroken skin**. Tritium Oxide, large quantities of which are produced in the moderator and heat transport systems of CANDU reactors, is one of the radionuclides which will be absorbed through the skin. Radionuclides may also enter the body through a wound or abrasion of the skin but this is an infrequent occurrence.

Figure 5

External and Internal Exposure

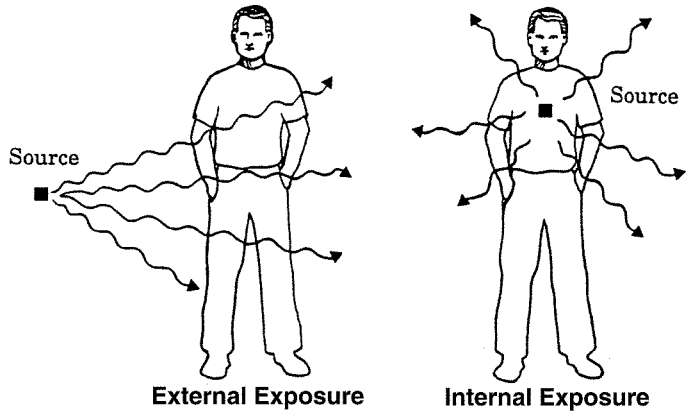
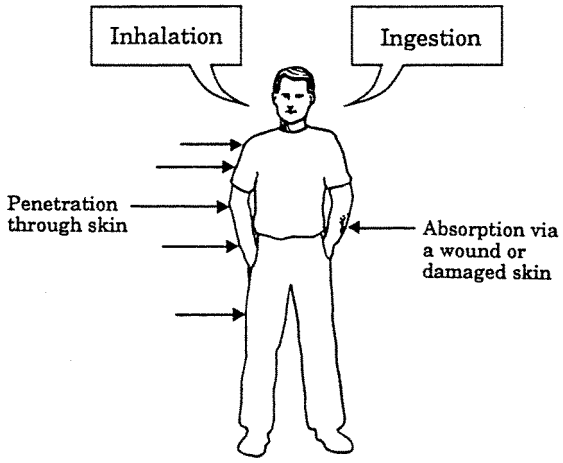


Figure 6

Internal Exposure Routes



The dose that is received from an intake of a radioactive material depends on factors such as:

- the half life of the radioactive material
- the biological half life of the material in the body
- the radiations emitted by the radioactive material
- the organ where the radioactive material eventually deposits (e.g. thyroid for radioiodines)

The System of Radiological Protection of the ICRP

The International Commission on Radiological Protection (ICRP) was established in 1928. It is an independent body of world experts in radiation protection and receives financial support from governments and from bodies such as the World Health Organization, the International Labour Organization and the United Nations.

The ICRP issued its first report in 1928 and has been making recommendations since that time on radiation protection practices and dose limits for workers and the public. The most recent recommendations were made in ICRP Publication 60 in 1990. In this document the ICRP describes a framework and a System of Radiological Protection.

Certain terms are defined. Human activities which increase exposure of individuals and populations by introducing sources, pathways by which these sources may expose individuals and the number of individuals exposed, are called **Practices** (sources in this context are not necessarily physical sources but rather a human activity or installation e.g. a radioactive waste management facility).

Human actions which can decrease exposure by modifying sources, pathways or reducing the number of individuals exposed are called **Intervention.**

The ICRP in its framework also defines three types of exposure:

a) exposure incurred at work and principally as a result of the work - **Occupational Exposure,**
b) exposure received by a person as part of medical diagnosis or treatment - **Medical Exposure,**
c) all other exposure is called **Public Exposure.**

The ICRP has established three principles relating to **Practices** which have been adopted generally by the international radiation protection community. These are:

- The **Justification** of a practice,
- The **Optimization** of a practice,
- Individual **Dose** and **Risk** limits.

Justification of a practice means that no practice should be adopted unless it produces sufficient benefit to the exposed individuals or society to offset the detriment due to the radiation exposure.

By **Optimization** of a practice is meant that doses to individuals and the number of people exposed should be **As Low As Reasonably Achievable (ALARA)** economic and social factors being taken into account.

Dose and risk limits are aimed at ensuring that no individual is exposed to radiation risks that are considered unacceptable.

The ICRP has also established two principles of **Intervention** which are that:

- the proposed intervention should do more good than harm and
- the form of the intervention should be optimized, i.e. the benefit of the reduction in radiation detriment (the ICRP term for harm) less the detriment associated with the intervention should be maximized.

Dose Limits Recommended by the ICRP

The dose limits proposed by the ICRP play a very important role internationally in radiation protection. They are usually adopted by governments and incorporated in some form into national regulations. The objectives of the Commission in their dose limit recommendations are:

1. to prevent deterministic effects
2. to reduce the induction of stochastic effects to levels that are considered acceptable.

Setting limits to achieve this second objective is the much more difficult task. The approach the Commission has taken in ICRP 60 is to consider the three serious delayed effects - the induction of fatal cancers, the induction of non-fatal cancers and the induction of serious genetic defects. Their proposals are summarized below for public and occupational exposure.

Table 6

ICRP 1990 recommended dose limits1

Application	**Dose Limit**	
	Occupational	**Public**
Effective Dose	20 mSv per year Averaged over defined periods of 5 years[2]	1mSv per year[3]
Annual equivalent dose in the lens of		
the eye	150 mSv	15 mSv
the skin[4]	500 mSv	40 mSv
the hands and feet	500 mSv	—-

Notes:

1. The limits apply to the sum of the external effective dose and the 50 year (70 years for children) committed effective dose from intakes in the same period.

2. The effective dose must not exceed 50 mSv in any year.

3. The value may be exceeded in any year, but must not be exceeded as a 5 year average.

4. This limit provides protection against stochastic effects. An additional limit is needed to protect against deterministic effects related to non-uniform exposure. This limit is 500 mSv per year, averaged over 1 cm^2 of skin, regardless of the area exposed, at a nominal depth of 7 mg cm^{-2}. This limit applied to the skin of the face also provides protection to the eyes against radiation such as beta particles. Members of the public are limited to ⅒th. of this value.

The limits proposed will certainly prevent deterministic effects. Continued exposure at the dose limit of 20 mSv per year is considered an intolerable situation. The dose limits must be considered in conjunction with the principle of optimization or ALARA. Application of the principle for most practices generally results in average individual doses being a fraction (0.1 to 0.25 of the limit). The risk associated with doses which are in this range is quite low and may be assessed from the risk values per .01 Sv given earlier (page 13)

Canadian Regulatory Limits

In Canada the Atomic Energy Control Board is responsible for issuing the Atomic Energy Control Regulations, which must be observed by all Corporations and persons in Canada. These regulations specify the maximum radiation dose limits for both atomic radiation workers and members of the public. Current limits are based on earlier ICRP recommendations and are shown in the accompanying table.

Table 7

Current canadian radiation dose limits For atomic radiation workers

Organs of the Body Receiving the Exposure (Critical Organ)	**Maximum Permissible Dose Limit**	
	ECY* Dose (Rem)	**QECY* Dose (Rem)**
Whole Body, Red Bone Marrow and Gonads	5 (50 mSv)	3 (30 mSv)
Skin, Bone, Thyroid Gland	30 (300 mSv)	15 (150 mSv)
Extremities (all parts of forearms, hands, feet and ankles	75 (750 mSv)	38 (380 mSv)
Other single Organs (lungs,lens of the eye, etc.)	15 (150 mSv)	8 (80 mSv)

* *ECY and QECY stand for Equivalent Calendar Year and Quarterly Equivalent Calendar Year respectively. An Equivalent Calendar Year is a period of a year starting on the first of January, April, July or October.*

These limits will be replaced shortly by limits which will be based on the ICRP 1990 recommendations.

Annual Limits of Intake

The limits recommended by the ICRP and in the Canadian regulations apply to both external and internal exposure. In many situations it is possible to prevent internal exposure completely and this is the preferred practice. Internal doses are often difficult to measure and to estimate from the measurements made. The ICRP has provided guidance in the estimation of internal doses in a recent publication (ICRP 61 Annual Limits on Intake of Radionuclides by Workers Based on the 1990 Recommendations). This document lists for about 800 radionuclides the intake which will give a dose equal to the annual effective dose recommended in the 1990 recommendations. For whole body this is 20 mSv. For radionuclides which concentrate in one organ it is the equivalent dose for that organ that determines the intake. The intake which gives an effective dose of 20 mSv depends on whether the intake is by inhalation or ingestion. This is because of the differing behaviour in the body of the radionuclides when taken in by these different routes. For example the lungs may be the most important organ based on effective dose if the intake is by inhalation. The values that have been determined for Annual Limits of Intake (ALI) in ICRP 61 are for adult workers and do not apply necessarily to members of the public. For example in the case of I-131, which is a volatile fission product, the intake for an adult which gives a certain dose will give a much higher dose to a child because the child's thyroid is very much smaller and so the radioiodine that deposits is a higher concentration.

For radionuclides which have a short residence time in the body the dose received from an intake of an ALI may be delivered over a short period of time. If the radionuclide has a long half life in the body, the dose may be received over many years.

A useful measure in occupational radiation protection for radionuclides which have a short residence time in the body is the Maximum Permissible Concentration (MPC). The MPC is the concentration of a radioactive material in air which will if a person is exposed to it for 2000 hours result in an intake of one ALI and a dose of 20 mSv (2000 hours is the number of hours worked in a year by a worker). This measure is very useful in the CANDU program because tritium oxide is formed in an (n,p) reaction with deuterium in heavy water. Tritium oxide becomes part of body water and irradiates the whole body. It has a half life in the body of about 10 days. Exposure to 1 MPC of tritium (as oxide) for 2000 hours will give a dose equal to the annual limit. With the current dose limits of 50 mSv the MPC is 3.7×10^5 Bq/m3 (10 microCi/m^3).

Framework for a Radiation Protection Program in a Nuclear Station

A radiation protection program starts with the highest level of the responsible regulatory authority and extends to the worker in the field. There are a number of ways in which responsibilities and roles of those involved may be discharged. Here we provide a general view of an approach for a typical Canadian station.

Regulatory Authority - the AECB

The AECB is the regulator authority for protection of workers and the public in the handling of radioactive materials. This is somewhat of an anomaly as health and safety both occupational and public are for other toxic agents a provincial responsibility. Even in radiation protection the jurisdiction between provincial and federal authorities is shared. For radiations produced by X-rays and other electronic type devices the responsibility for control of the specifications of the device is federal under the Radiation Emitting Devices Act (the RED Act). The responsibility for operational control is provincial. For nuclear stations the responsibility is clear - it is federal and rests with the Atomic Energy Control Board.

The AECB exerts its authority through the Atomic Energy Control Act and the regulations pursuant to the act. This includes as previously discussed dose limits for workers and the public. Generally the AECB staff in administering its regulations has in the past been non-prescriptive and has requested licensees operating nuclear stations to propose how they intend to operate their program. The AECB then comments on this program and if it is satisfactory approves it. This mode of operation has been changing and more recently the board approach has been more prescriptive. The non-prescriptive approach has been successful in the past as the CANDU radiation protection program has been one of the more successful in terms of occupational dose. In this approach, accountability rests with the utility - the operator. In the prescriptive approach the AECB assumes some responsibility, and therefore accountability, thereby relieving the licensee of this.

The licensing process is the prime control mechanism of the AECB. Through the licensing process the AECB has the opportunity for any proposed activity involving radioactive materials to apply the ICRP principle of **Justification,** i.e. to decide whether a practice should be permitted depending on whether the benefits outweigh the advantages to society.

AECB staff have also in the past carried out an assessment role by auditing the radiation protection program at stations. Compliance has also been a board staff role practiced through inspectors stationed at site. So far little use has been made of the authority of the AECB to lay charges under the Atomic Energy Control Act. This is in contrast to the Nuclear Regulatory Commission in the United States.

Utility - Corporate Role

The optimum role that the corporate group in a utility can play is:

1. to assign a clear responsibility for radiological safety to line management i.e. to the nuclear generation group,
2. to establish an independent radiological safety group assigning to them the responsibility for corporate policies and regulations, program assessment and personal dosimetry services,
3. to monitor performance in radiological safety and to provide direction when disagreements arise between operations and the independent radiological safety group.

Nuclear Production Group/Station Management

The nuclear production group is the line group and is responsible for discharging the radiation safety program as an integral part of producing electricity. Management must provide the systems which are necessary for safe operation. Important considerations are the training of personnel, covering both knowledge of radiation protection and procedures, establishment of appropriate radiation control procedures and provision of appropriate instrumentation and equipment.

Corporate Radiological Safety Group

The important role of this group is:

- to provide a regular assessment of the effectiveness of the radiation protection program to senior management,
- to maintain a watching brief on developments in radiation protection so that the program is able to accommodate changes that may occur to limits, practices etc.
- to provide personal dosimetry services. The reason for this is to provide some assurance to employees that their dose estimation and recording is free of any operational pressures.

Individual Workers

Individual workers have a responsibility to their families, their fellow workers, their company and to themselves to work safely and follow established procedures.

The total framework is summarized in the following two figures.

Figure 7

General Framework for Radiation Protection Management – AECB/Utility

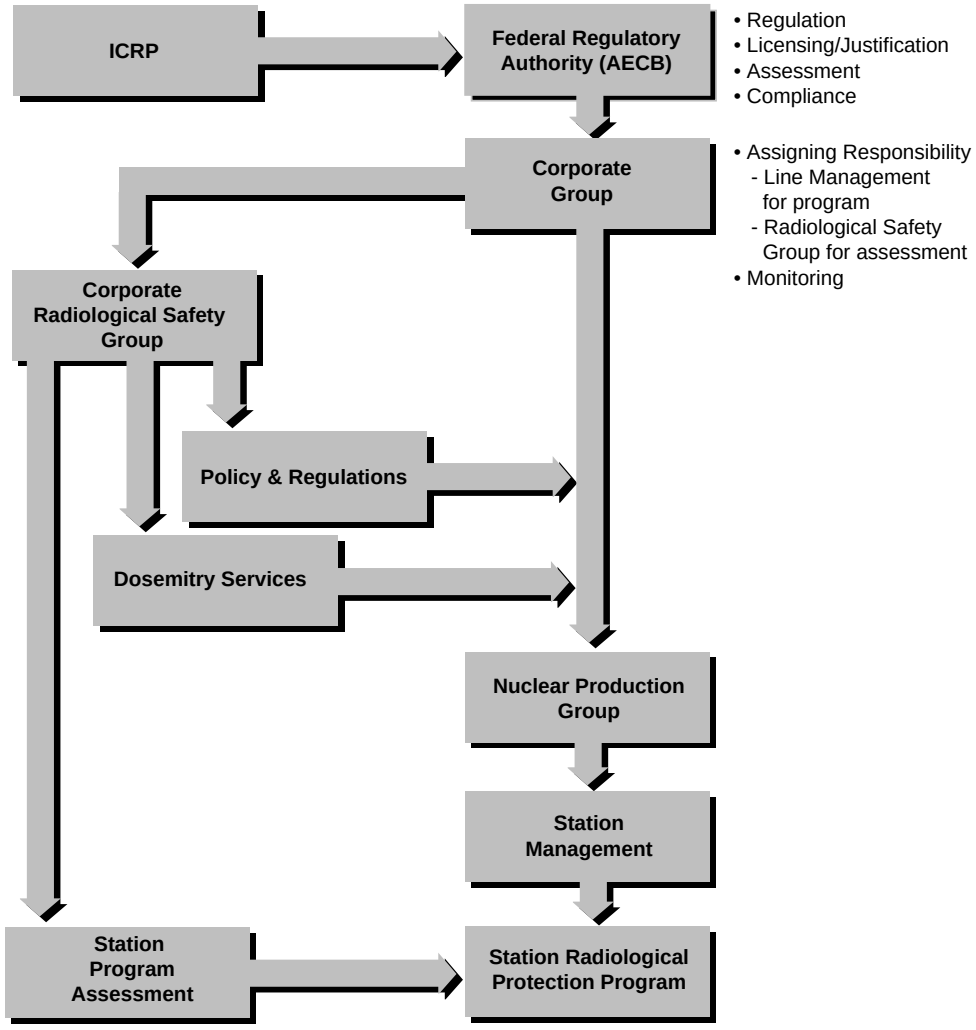
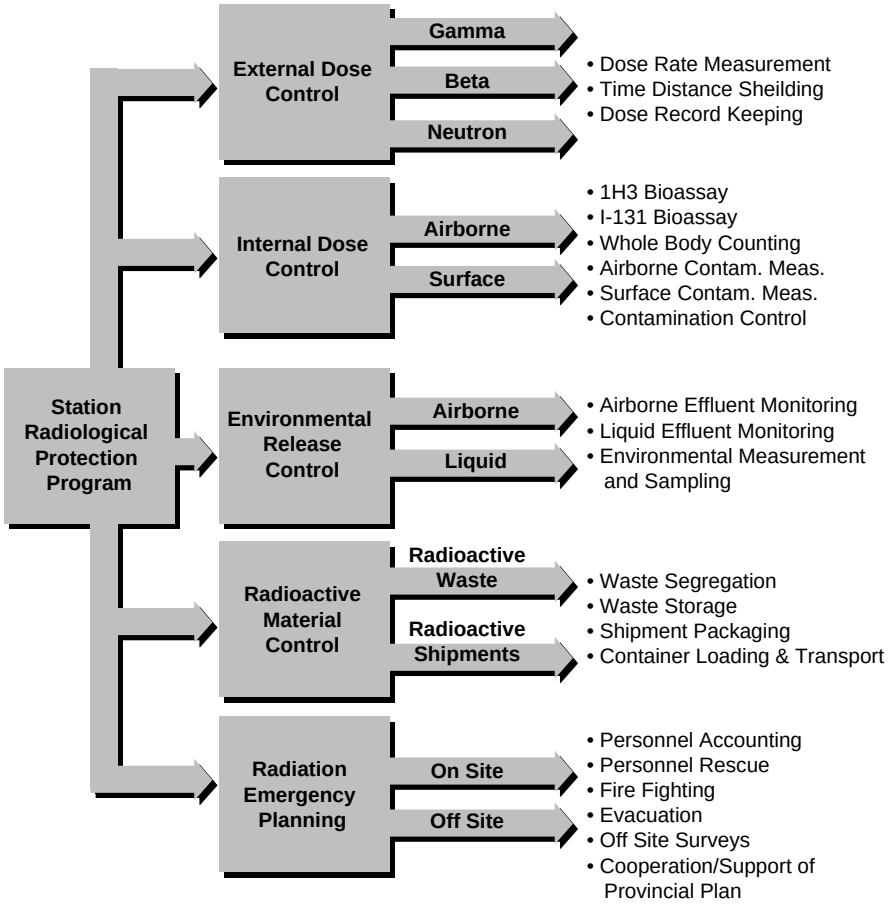


Figure 8

General Framework for Radiation Protection Management – Station Program



External and Internal Dose Control at Nuclear Stations

Table of Contents

Dose and Dose Rate Calculations

External radiation sources can vary from small discrete sources of, say, cobalt-60, to the whole reactor face in a CANDU reactor. The normal methods of protection are **Time, Distance** and **Shielding**.

Time

The dose a person receives in a radiation field is given by:

Dose = Dose Rate x Exposure Time

Calculating the dose received is similar to calculating wages where the hourly rate is multiplied by the time worked, e.g. a plumber works for 10 hours at $20 per hour: he earns $200.00, ($20/h x 10h)

Similarly if a worker spends two hours in a dose rate of 15 mSv/h, his/her dose is given by:

Dose = Dose Rate x Exposure Time
Dose = 15 mSv/h x 2 h
= 30 mSv (3 rem).

The dose may be reduced by reducing the time spent by the worker near the source. For example, if the worker spends only 1.5 h in the area, then the dose received is reduced to:

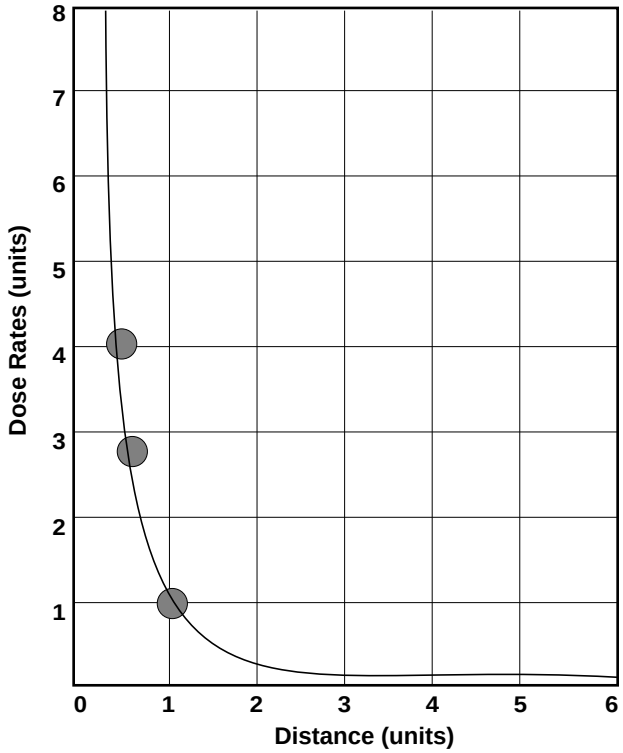
Dose = 15 mSv/h x 1.5 h
= 22.5 msv.

Reduction in working time is a powerful method of protection against radiation exposure.

The calculation can be turned around:
If the worker's dose must restrict to 0.5 mSv, how long can the worker stay in a place where the radiation dose rate is 12 mSv per hour.
Re-arrange formula (1) above:

Exposure Time = dose/dose-rate.
= 0.5 mSv / 12 mSv/h
= 0.042 h
= 2.5 min.

Figure 1
Dose Rate from a Point Source



Distance

Point sources of gamma radiation obey the $1/d^2$ or Inverse Square Law. This law states that the dose rate from a point source of radioactive material decreases in proportion to the square of the distance from the source. As the distance from a source is increased the dose rate decreases markedly and vice versa. This is illustrated in Figure 1.

From the graph it is seen that the dose rate increases very rapidly as the distance from the point source decreases. Dose rates at the surface of small sources can be incredibly high, and this is the reason why small sources must never be picked up in the fingers - always use tongs or tweezers to increase the distance and reduce the dose-rate.

The equation relating dose rate H1 at distance D1 to dose rate H2 at distance D2 is:

$H1/H2 = D2^2/D1^2$

where H1 is the dose rate at distance D1
H2 is the dose rate at distance D2

Example:

The dose rate at 1 foot from a 1 mCi cobalt-60 source is 0.13 mSv/h (13 mrem/h). What is the dose rate at 1 inch, and 3 feet (1 inch = 0.083 ft).

i) The dose rate at 1" is given by:

$H(1\text{ foot})/H(1") = [D(1")]^2/[D(1\text{ foot})^2$

$H(1") = 0.13 \times (1)^2/(.083)^2$

$= 18.9$ mSv/h (189 mrem/h)

ii) The dose rate at 3 ft. $= (0.13) \times 1^2/3^2$

$= 0.14$ mSv/h (1.4 mrem/h)

Exercise:

Calculate the dose rate at ¼", 5 ft and 6 ft from the same source,

Increasing the distance from a source of radiation will almost always reduce the dose-rate.

Not all sources encountered are point sources. On a nuclear station there are many sources of different shapes and sizes. Two simple sources are a 'line' source and a 'plane' source. A line source on a nuclear station could be a long narrow pipe containing radioactive fluid. For a line source for practical purposes the variation of dose rate with perpendicular distance from the line can be taken as inversely proportional to that distance. e.g. if the dose rate at distance a1 is d1 then at distance a2 the dose rate is:

$$d_2 = d_1 \times a_1 / a_2$$

Example:

The does rate at 1 metre from a line source is 6 mSv/h what is the dose rate at 3 meters.

$d_{3m} = d_{1m} \times a_1 / a_2$

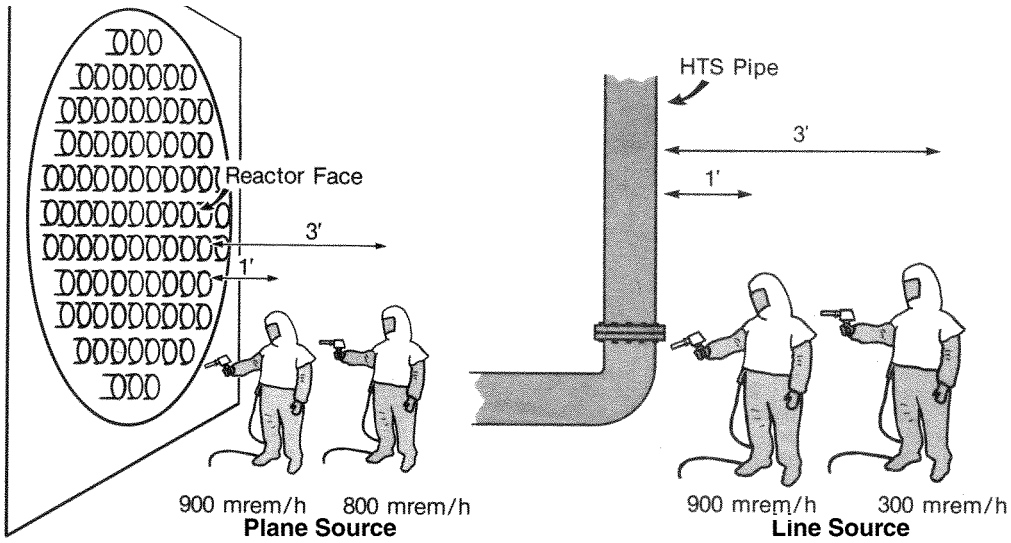
$= 6 \times ⅓$

$= 2$ mSv / h

For a plane source the decrease in dose rate with distance depends very much on the size of the plane source. If the plane source is an infinite size then for all practical purposes the dose rate does not change with distance from the plane. For most practical plane sources encountered in nuclear stations such as the reactor end face the decrease in dose rate with distance from the surface will be small, (see figure 2).

When a work area is entered a quick survey of the whole area should be carried out to detect any abnormally high dose-rates . At the work location the dose rate at the position of body hands head and feet should be measured so that a complete picture of the exposure conditions is obtained.

Figure 2:
Dose Rate vs Distance from Reactor Face and Piping



Properties of Radioactive Materials and Radiation

Radioactive Decay

Another aspect of protection by the use of time which can be very useful in a nuclear station, is to allow short lived radionuclides (fission products and activation products) to decay before entering or working in an area. This form of protection is most useful immediately after the reactor has been shut down when short lived fission and activation products may be present in reactor systems.

Shielding

The properties of a radiation determine the best kind of shielding for that radiation. For example, beta radiation is easily stopped by sheets of plastic or plywood between 1 and 2 cm thick. More dense materials will also stop the beta rays but in doing so will convert some of the beta radiation energy to X-rays called Bremstrahlung (German for braking radiation).

Fast Neutrons are most easily slowed down by hydrogenous materials including heavy water, light water, paraffin wax, polyethylene or concrete. Thermal neutrons are absorbed into the nucleus of atoms; sheets of cadmium, or boron compounds are commonly used.

The best absorbers for gamma radiation are those elements which have high atomic number and are very dense. (e.g. lead, uranium, tungsten). These materials also tend to be very expensive, so they are used when the shield has to be compact (because of space limitations) or when the source has to be portable (e.g. industrial radiography cameras). When space is available, cheaper, lighter materials, such as iron or concrete are used. Water is also used, particularly

when the application requires the ability to see the source, Irradiated Fuel Inspection bays, (which may have 18 or 20 feet of water over the irradiated fuel) are a good example of this application.

The amount of shielding required in a given installation is determined by the energy of the radiation, and the intensity of the radiation. Low energy gamma radiation (e.g. 0.06 Mev photons from americium-241 sources) are almost completely attenuated by ¼ inch of steel. On the other hand the 1.25 and 1.33 Mev photons from cobalt-60 require 1.5 inches of lead, 3.5 inches of steel, or about 12 inches of concrete to reduce the dose-rate by a factor of 10. The decrease in dose rate from a point source with depth of shielding in any material follows an exponential law.

$$D = D_0 e^{-ax}$$

Where D0 is the dose rate at the surface of the material

D is the dose rate at depth x cm. and 'a' is a characteristic of the material known as the absorption coefficient in cm-1.

Shielding ideally should be designed to reduce dose rates from the reactor core and radioactive systems to levels which cause few problems in radiation dose management. Determination shielding thickness in design provides a good exercise in optimization (ALARA). The design target for the dose rate for an area protected by a shield depends on the fraction of the time that the area is occupied and by how many people. Obviously an area continuously occupied by a large number of station staff must have a considerably lower design dose rate target than one for which an entry of a few hours per year is made by one person. For a variety of reasons, occasionally, design shielding provided is underestimated and portable or temporary shielding in the form of concrete blocks, lead bricks or lead blankets is required. Lead blankets which are thin sheets of lead contained within heavy duty plastic are useful for draping over pipes and other pieces of equipment containing radioactive liquids.

At the design stage shielding must not just be considered as that required for normal operation. The requirements during abnormal and accident conditions must be taken into account. Backfitting has been necessary for this purpose and has been necessary at some Ontario Hydro stations. e.g. at Bruce 'A' shielding has been added to the Emergency Cooling Injection System header, vault vapour recovery and Emergency Filtered Air Discharge Systems

Beta Radiation - Special Aspects

Beta radiation may be both an external and an internal radiation hazard. In this section we are primarily concerned with beta radiation as an external hazard. Because it has a limited penetrating ability, beta radiation is primarily a hazard to the skin of the body and the hands. The hands frequently receive more dose

than skin on other parts of the body because they are often closer to a source of contamination (and the $1/d^2$ law applies).

The lens of the eye is potentially at risk, but it is covered by about 3 mm. of less sensitive tissue, which provides protection against beta particles with energies of less than about 2 Mev.

When high energy beta radiation (>2 Mev) is present it may be necessary to wear protective spectacles to protect the lens of the eye (normal safety glasses with a 3 mm thick plastic lens will be satisfactory for most applications).

Decontamination of surfaces is another very effective method of protection against beta radiation.

Neutron Radiation - Special Aspects

Neutrons are sometimes found in unexpected places in vaults and boiler rooms due to 'neutron streaming', (or leakage) through gaps around penetrations, through shielding, or through weaknesses in shielding. The placement of new equipment in these areas can cause scattering of neutrons from a'streaming' source, to produce neutron fluxes in places where previously there were none.

Neutrons may also be found in areas containing Heat Transport and Moderator system components, as a result of photoneutron reactions between high energy gamma radiation (from activation products such as N-16 and O-19) and the deuterium atoms in heavy water.

$$^{1}D_{2} + \gamma\ (>2.2\ \text{Mev}) = {}^{1}H_{1} + {}^{1}n_{0}$$

Radiation Survey Instruments

General

All radiation survey and measuring instruments are based on the detection of ionization, or the detection of the secondary effects of ionization. Electronic circuits are then used to amplify the effect so that it can be displayed in an appropriate format.

To be useful in measuring external radiation, the instrument should ideally display the **effective dose rate** that a person located at the position of the instrument would experience. This is known as having a tissue response. Achieving this is not always easy. For neutron dose this is particularly difficult and it is sometimes necessary to use different types of instruments for differing energies of neutrons. For gamma radiation designing an instrument with a tissue response is difficult for low energy X-rays. Another important requirement of a radiation protection instruments is its response time or 'time constant'. This is the time the instrument takes to reach its final reading.

Obviously this should not be too long or there is a danger of underestimation of the dose rate. Saturation effects are also important. Generally this is not a problem with modern instruments but there have been poorly designed instruments in the past which in very high dose rates would saturate and give an erroneously low dose rate reading.

Operational Tests on Survey Instruments

All survey instruments **must** be checked and exposed to a check source to confirm that they are fully operational and give a correct response, before they are taken into a work area.

If an instrument gives an 'off scale ' response when being used in a work area to assess hazards, **believe it** and leave the work area immediately. Accidents have occurred in the past when a person has not believed an instrument reading and has continued to work in a high field. 'Off Scale' reading must be believed until such time as measurements have been repeated with a reliable instrument. If an instrument reads zero then this is also a concern and it is necessary to check the performance of your instrument, and obtain a replacement if necessary.

Gamma Survey Instruments

Some typical gamma dose rate (or survey) meters used on a nuclear station are:

i) Medium Range — this unit typically has two ranges
(0 - 5 mSv/h) (1-500 mRem/h)
(0 - 50 mSv/h) (.1-50 Rem/h)

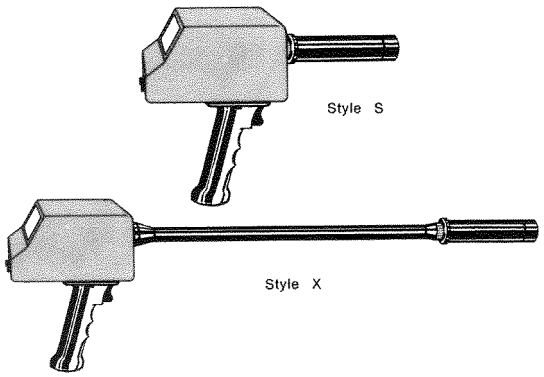
ii) High Range — typically (0.1 Sv/h) - (100 Sv/h)
(10-10,000 Rem/h)

iii) Emergency Gamma Meter
this type of instrument is usually calibrated in rem/min. with a maximum reading of (2 Sv/min) (200 Rem/min).

The first two instruments usually have very small 'high current' geiger detectors; the third frequently uses a solid state detector. Both types of detectors are very small which means that they can be installed at the end of the instrument so that it can be used to get reasonable accurate 'contact' dose-rates from surfaces. These instruments are illustrated in Figure 3.

Skill in monitoring an area for gamma radiation hazards comes with practical experience. Factors of importance include identifying 'beams' through imperfect shielding, hot spots that may be masked by high background levels, and the need for 'contact' readings on surfaces of equipment which may have to be handled. Finally there may be an associated beta radiation field.

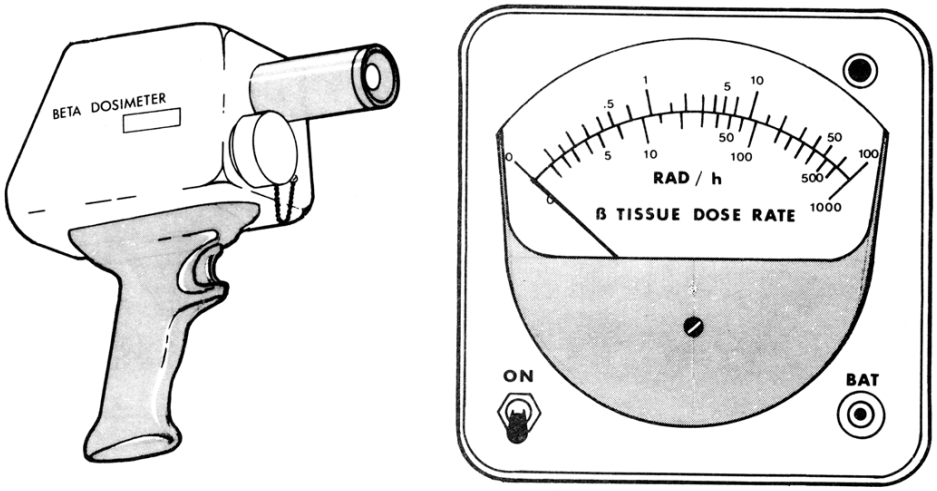
Figure 3:
Typical Gamma Meters Used on Nuclear Stations



Beta Dose Rate Meters

Beta dose rate meters typically have a solid state detector at the end of the snout and a plastic cap which covers the detector. It is necessary to take two readings; one with the plastic cap ON, and one with the plastic cap OFF. The plastic cap is thick enough to absorb beta radiation - so the 'cap ON' reading gives the gamma effect component of the readout. To get the beta dose rate this reading must be subtracted from the 'cap off' reading which is a measure of the beta dose-rate plus the gamma response of the detector. (The response to gamma radiation must not be taken as the gamma dose rate at the location as the instrument is not usually designed to measure gamma dose-rate). Typically an instrument of this type reads beta dose rate in two ranges: 0 to 1 Sv/h (100 rem/h, and 0 to 10 Sv/h (1000 rem/h).

Figure 4:
Beta Meter Showing Cap Off



Important points to remember when carrying out beta surveys include the facts that:

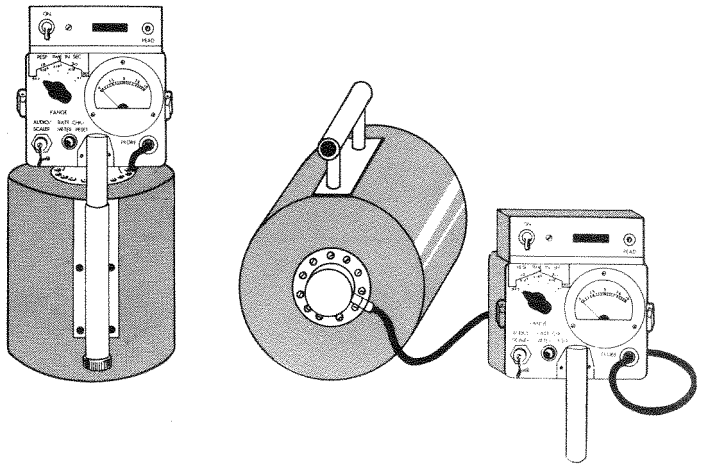
- beta radiation is directional (except when you measure the beta radiation from a gas cloud),
- beta dose rates from fresh fission products may be up to 100 times the gamma dose rate,
- beta dose-rates increase very rapidly as the surface being measured is approached ,
- the meter frequently has a long time constant, 3 or 4 seconds (3 times the time constant is needed to get a good estimate of the dose rate),
- finally there may be very 'hot' particles on equipment that has loose contamination on its surface and these are sometimes difficult to locate.

Neutron Dose Rate Meter

The most widely used neutron dose rate meter is known as the 'Snoopy'. This is a bulky instrument but it can measure the dose-rate up to 10 rem/h, for all neutron energies, from thermal to about 10 Mev.

The instrument uses a proportional BF_3 counter shielded in polyethylene, and incorporating additional 'filters' to even out the energy response. Measuring neutron dose with a personal detector is difficult so the Snoopy is sometimes used to measure integrated dose. For this purpose the meter has an additional component - a dose 'integrator' which is used to accumulate the neutron dose while a person is working in a neutron field. The instrument must be carried into the work area and placed at a suitable location such that it is exposed to the same neutron field as the worker.

Figure 5:
Wide Range Neutron Meter (Snoopy)



Neutrons are mainly of concern when the reactor is operating. But it must be remembered that a fairly intense source of photoneutrons exists in the CANDU reactor even when the reactor is shutdown. Existing neutron fields or beams are usually posted so that people know they are there and can avoid them.

Factors that should be remembered when performing a neutron survey are that neutron beams may be scattered if heavy equipment is placed in the beam, changing the neutron flux distribution in the area. The response time of the instrument changes with each range. The low range is very slow to respond.

External Radiation Dosimeters

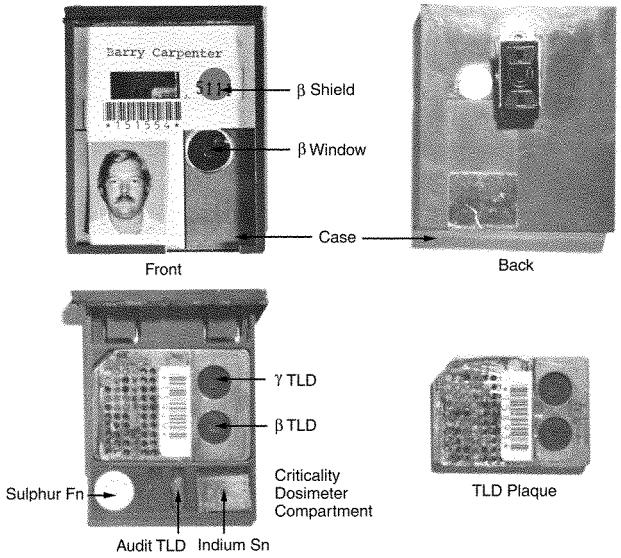
All persons who are exposed to radiation must:
1. have a means to monitor their exposure (both external and internal),
2. have their radiation doses permanently recorded in an approved Dose Record System.

The dose measurement and recording system must satisfy the needs for day to day and long term (i.e. cumulative) radiation dose control.

Personal Dosimetry

The standard means for measuring and recording the external radiation doses received by nuclear station workers makes use of Thermoluminescent Dosimeters, (TLD's) (or Badges). These 'dosimetry badges' are issued for a defined period,(for example, a two week period), but they can be returned for read-out by the worker at any time that dose control concerns warrant this action. A new TLD badge is issued when a badge is returned for readout.

Figure 6:
A Dosimetry Badge



The function of the dosimetry badge is to measure the whole body 'X' or Gamma radiation dose received by the wearer, and to measure the beta whole body dose to the skin.

'Whole body' and beta 'skin dose' TLD's are installed into an aluminum plaque which carries a readable identification code. When the plaque is installed in the badge (or holder) the 'Whole body' TLD is covered (or shielded) by a 2 mm thick aluminum 'filter' (or shield) to prevent irradiation of that TLD by beta radiation. The front surface of the badge has a 'window' (or hole) to allow both beta and gamma radiation to irradiate the second TLD.

TLD badges must be worn in defined 'radiation' area or zones in the plant. Badges must be worn on the front, upper trunk, so that they can be seen by other workers.

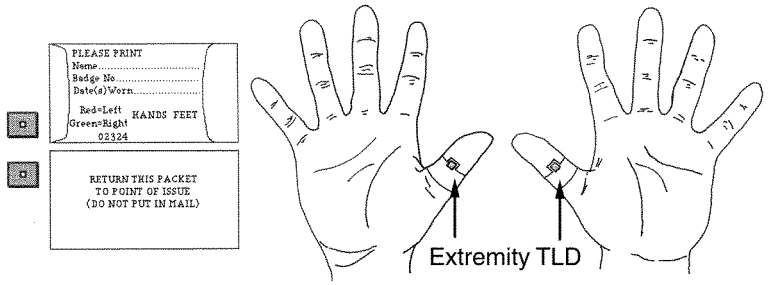
Badges should be protected during work in badly contaminated areas by being placed into a thin plastic bag to prevent the transfer of small radioactive particles to the surface of the badge. Local procedures for the issue of extremity badges must be followed.

If the radiation field is known to be very non-uniform, several badges may be worn simultaneously at several positions on the body. In this case the highest reading on the body would be recorded as the whole body dose. Badges on the head arms and legs are called extremity badges. When these additional badges are obtained, the regular badge is 'handed in', and recovered when the extra badges are submitted for readout.

Extremity Dosimeters

These are small, thin, TLD crystals enclosed in a thin flexible plastic package. They measure total beta + gamma dose. The plastic packages containing the TLD crystals are taped to the thumbs (Figure 7). Extremity dosimeters must be worn when the radiation dose to the extremities (eg. hands, feet, ankles, forearms) will be significantly higher than that recorded by the whole body TLD. These dosimeters must not be worn for measuring the radiation dose to other parts of the body.

Figure 7:
Extremity Dosimeters



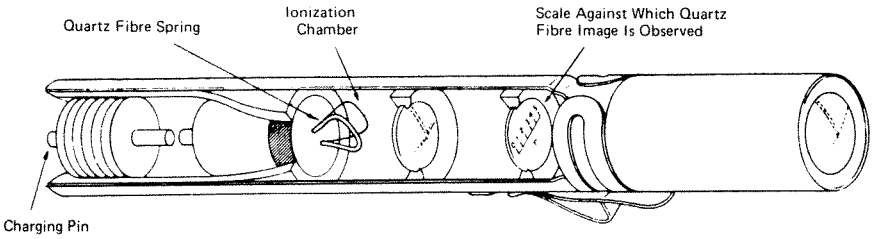
Direct Reading Dosimeters

Direct Reading Dosimeters (DRD's) are used for daily measurement of whole body gamma dose. These devices are not as accurate as TLD's, but can serve as a back-up if a TLD is lost, or mislaid, or damaged. They have the additional advantage that they can be used for immediate read-out when a worker is exposed in a high dose-rate situation.

A drawing of a Direct Reading Dosimeter (DRD) is shown in Figure 8. This type of DRD is available in several ranges.
Useful ranges for current dose limits are:

0 - 5 mSv (500 mrem)
0 - 50 mSv (5 Rem)
0 - 500 mSv (50 Rem),
[reserved for use in station emergencies].

Figure 8:
Direct Reading Dosimeter



The DRD is a miniature ion chamber. The readout scale is marked off in mSv (or mrem or Rem).

When a work program involves exposure to high dose rates, a high range 0 - 50 mSv (0 - 5 Rem) DRD is worn in addition to the standard DRD in order to ensure that an estimate of the exposure dose will be available, even if the exposure is greater than 5 mSv (500 mrem).

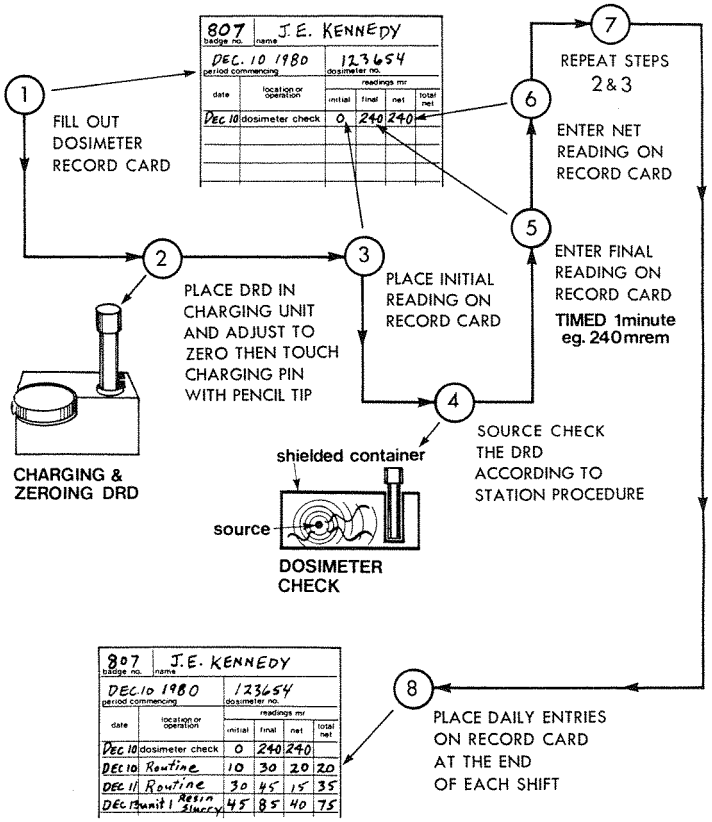
An outline of the station procedure for using DRD's is shown in Figure 9.

The first time a DRD is used in each dosimetry period (issue period for the TLD badge) it must be 'constancy checked' (exposed to a source to confirm that it is working correctly), and the results recorded. The DRD must be worn alongside the TLD badge.

Recorded doses may be in error if the DRD is knocked, or dropped. If a DRD goes 'off scale' for any reason when being worn the TLD badge should be read out immediately.

Discrepancies between the DRD and the TLD badge are generally resolved in

Figure 9:
Direct Reading Dosimeter Procedure



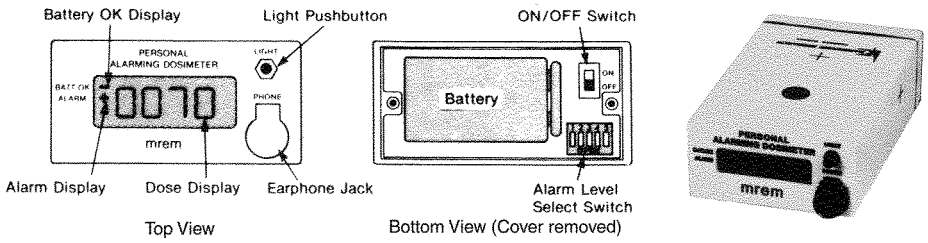
favour of the TLD badge, it is the official dosimetric device. Bigger discrepancies are subjected to an inquiry to try to establish the cause of the discrepancy between the two devices. Discrepancies may be due to errors in setting the DRD to zero, the energy response of the TLD - vs - DRD, or the fact that the TLD dose is corrected to correspond to that of the gonad dose. There may also be charge leakage from the device.

Personal Alarming Dosimeters

These devices are used to control dose when working in high dose-rate areas. They can show the accumulated dose in the range 10 to 9999 mrem, when used in dose-rates up to 100 Rem/h. (1Sv) The dose received is shown on an illuminated scale.

In this model an alarm can be set to trigger at 160, 320, 640, 1280, or 2560 mrem, An earphone is provided to facilitate hearing the alarm in high noise areas.

Figure 10:
Personal Alarming Dosimeter



Neutron Dosimetry

Neutron dosimetry in nuclear stations, or anywhere for that matter, is difficult. The problem is in obtaining a dosimeter that is of reasonable weight and size, that has adequate sensitivity and has a response that is close to the delivered effective dose as the neutron energy changes.

Use of the Snoopy as a Dosimeter

Whole body dosimetry for neutrons is carried out with the Snoopy meter in the integrating mode. This device described earlier in the lesson (Figure 5) displays the radiation dose at the location of the detector from neutrons of all energies, in increments of 10 microSv (1 mrem), in the range 10 microSv (1 mrem) to 9990 microSv (9999 mrem).

The integrator operates separately from the neutron meter, i.e. it must be switched on as a separate instrument.

Neutron Sensitive and Insensitive Tlds

Some stations have available badges containing two types of TLD chips. The TLD chip for normal gamma dose measurement is a neutron insensitive Lithium Borate ($LiBO_3$). The neutron insensitivity is achieved because the Li in the TLD contains only Li-6 and no Li-7. Li-6 does not interact with neutrons. A second dosimeter is again made of $LiBO_3$ but this time enriched in Li-7 and B-10 both of which interact with neutrons in an (n, α) reaction. This dosimeter is very sensitive to neutrons. The difference between the readings of the two TLD chips gives a measure of the neutron exposure. The device is very energy dependent, over responding to thermal neutrons. It is useful in defined neutron fields where the energy spectrum does not change. In such a situation the TLDs can be calibrated using a Snoopy or similar dosimeter which has good neutron energy dependence.

Neutron Bubble Dosimeter

These are a relatively new dosimeter consisting of a water gased polyacrylamide gel containing superheated Freon gas. Exposure to neutrons will give rise to

bubbles along the tracks of the secondary particles produced. These bubbles are counted. The advantage of this type of dosimeter is that it is gamma insensitive and has good neutron energy response. The disadvantage is that at least for the initially available dosimeters the response was quite temperature dependent. This has been corrected to some extent for dosimeters more recently available. The dosimeters have been tested on stations experimentally but are not yet in routine use.

CR39 Track Etch Dosimeters

This type of dosimeter can be used to detect the presence of neutrons above a certain energy. The dosimeter consists of a small plastic film made of polycarbonate, known as CR39. When exposed to neutrons above a certain energy the recoil protons produced damage the plastic film. This damage when etched in XXX will produce a small hole in the plastic. The number of holes produced increases with the neutron exposure. Holes can be counted automatically. Energy dependence is poor for the neutron fields encountered in CANDU stations. The device is useful for determining whether a person has been exposed to a neutron field. Dose may then be evaluated using other methods.

Internal Dose

In a CANDU station the possibility of internal exposure to a variety of radionuclides exists - there are large numbers of fission and activation products present in fuel and in the station systems. Practically however, the internal dose control program is dominated by exposure to tritium oxide.

Annual Limit of Intake

The ICRP has calculated for a large number of radionuclides the quantity, in Bq, which if inhaled will result in a dose of 20 mSv. This is the Annual Limit of Intake. It does not matter whether this is inhaled as a large intake over a short period of time or if it is inhaled in small amounts over an extended period. Now the average person breathes at a rate of approximately 1.2 m^3/h during the working day (a value recommended by the ICRP). This means that about 2,400 m^3 of air is breathed in a year. The uniform concentration of a radionuclide in air which will result in the intake an ALI is therefore given by dividing the ALI by 2,400. This intake means ultimately that the person taking in the radioactive material will get a dose of 20 mSv. The airborne concentration calculated in this way is known as the Derived Airborne Concentration (DAC).

DAC = ALI/2400

The DAC is a useful measure as it indicates the dose that a person who is not wearing respiratory protection will receive if he/she works in that concentration for an hour. Table 1 gives the DAC calculated in this way for several radionuclides commonly encountered in nuclear station work.

Tritium has not been included in this table as the ALI used in Canada differs slightly from that given by the ICRP. This is discussed more fully in the next section.

Table 1

Annual Limit Of Intake For Radionuclides In Candu Stations

Radionuclide	ALI (Bq)	DAC (Bq/m^3)
C-14	4×10^7	1.7×10^4
Co-60	4×10^5	1.7×10^2
I-131	1×10^6	4.2×10^2
I-133	8×10^6	3.3×10^3
I-135	4×10^7	1.7×10^4
Cs-134	2×10^6	8.3×10^2
Cs-137	2×10^6	8.3×10^2

Station Internal Dosimetry Programs

Internal dose measurements are accomplished by a variety of means, urine samples, thyroid monitoring, lung monitoring, or whole body counting. In certain circumstances e.g. for C-14 fecal samples may also be analyzed.

The internal dosimetry programs at Canadian heavy water reactors are similar in principle but currently are quite different in detail for a variety of reasons. All stations have a problem in that the Atomic Energy Control regulations governing radiological safety have not kept pace with changing ICRP recommendations. In Ontario Hydro SI units were adopted in the late 1970s. Radiological units unfortunately were excluded at that time. The four multi-unit Ontario Hydro stations are still using old units rem, rad and Ci and the old concepts of Maximum Permissible Body Burden and Maximum Permissible Concentrations. New Brunswick Hydro Electric Power Commission and Quebec Hydro are using modern units Gy, Sv and Bq.. New Brunswick have implemented fully the most recent recommendations of the ICRP and are using ALI, DAC, equivalent dose and effective dose.

Internal Dosimetry of Tritium (as Oxide)

Tritium Production

Tritium is produced as an activation product by a (n,gamma) reaction on the deuterium atom in heavy water in the moderator and heat transport systems. This results in large quantities of tritium in these two systems. At equilibrium, that is when the production equals the decay, the concentration in the moderator for Point Lepreau for example is about 3 TeraBq/kg (80 Ci/kg) with the heat transport approximately thirty times less. This means that tritium as oxide is encountered routinely in CANDU stations and uptakes are regular. Very high concentrations may be encountered during a spill. Measures are taken to keep doses as low as reasonably achievable, but the regular uptakes that do occur

require a program for routine dose allocation to exposed personnel. Some other radionuclides are encountered, but this occurs infrequently, and for these radionuclides the objective of the internal dose control program is prevention of uptake. This objective has to be tempered in a work situation with the need to minimize total dose. There is little sense in requiring extensive protective equipment to save a few microSv internal dose if the protective equipment doubles working time in a high external radiation field. However care must be exercised when taking this approach to ensure that the intake of a radionuclide that is difficult to measure does not occur. If an uptake does occur then a measurement of the uptake and a dose estimate is required .

Dose From a Tritium Intake

The ease with which tritiated water vapour can be taken into the body by inhalation (100% absorption in the respiratory system) and by absorption through the skin (at about the same rate as by inhalation) makes tritiated water the most important internal radiation hazard on a CANDU station. Once inhaled or absorbed through the skin, it is mixed uniformly with body water within a period of two to four hours. At this point tritium irradiates most body tissues. Because it is mixed with body water, tritiated water is excreted in urine, perspiration and exhaled breath. The tritiated water concentration in the body is diluted every time there is an intake of water (It can be likened to pouring blue ink into a bucket of water and stirring it up. If three times a day a glassful of the blue water is taken out and a glassful of clean water is poured in then after a period of time the water in the bucket will lose its blue colour). The biological half-life (the time it takes to reduce the concentration in the body to half of the initial concentration) is taken as 10 days, but this is easily influenced either way by increasing or decreasing the water intake to the body. Decreasing the biological half-life will have the effect of reducing the committed effective dose related to a given intake. This is sometimes done under medical advice when high exposures occur.

For tritium dose calculations the AECB has published a regulatory guide (AECB Regulatory Document R-100, 'The Determination of Effective Doses from the Intake of Tritiated Water'). This document uses the advice of the ICRP and also additional information. Based on this document which recommends the use of body tissue as the organ on which to base dose and allowing for organically bound tritium (this is tritium which has been incorporated into the molecular structure of body tissues) the sustained concentration of tritium in urine which will deliver an annual dose of 20 mSv is 9.6×10^5 Bq/l (26 microCi/l).

ALI and DAC for Tritium

The ALI for tritium based on R-100 is 1×10^9 Bq (27 mCi) for an effective dose of 20 mSv (2 Rem). This is also the ALI recommended by the ICRP. The ICRP has also recommended that a standard man breathes at a rate of 1.2 m^3/h during working hours and that for tritium the intake via skin absorption is equivalent to

a breathing rate of half that value. This gives a total equivalent intake rate of 1.8 m^3/h or 3600 m^3/year. The DAC is therefore 3 $x10^5$ Bq/m^3 (8 microCi/m^3).

Bioassay for Tritium

The bioassay program for tritium requires that those routinely exposed submit a urine sample on a weekly basis. Sample submission is also required when an uptake is suspected. Samples of approximately 100 ml are collected in 'Dispo' bottles. These bottles are located at the main washroom near to the change room facilities. A bioassay record form (usually pre-punched for regular employees) must be submitted with each sample. Persons who are not regular employees must complete one of these forms.

The tritium in the submitted urine sample is measured using a liquid scintillation technique. A few mls of the urine is mixed in a vial with a chemical (liquid scintillator) which gives off light as the energy of the beta radiation from tritium in the urine is deposited in it. The counter detects and counts these pulses of light. Nuclides other than tritium may also be detected. If other nuclides are detected then a large volume sample of urine will be requested for more detailed analysis.

Persons working, at the station but not regularly exposed to tritiated water vapour should submit a sample prior to exposure in order to establish a 'baseline' from which a the current uptake can be calculated.

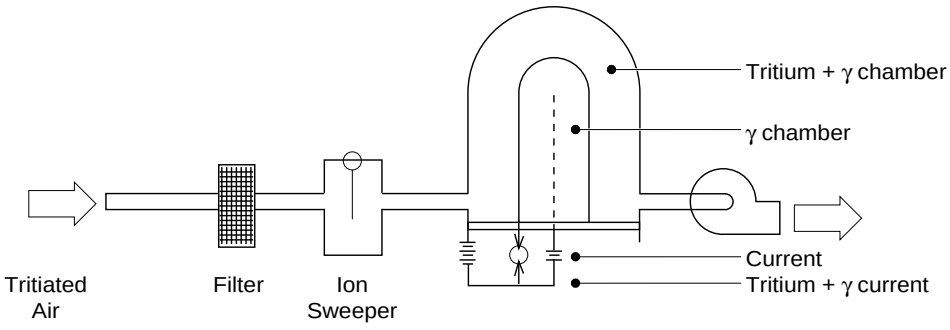
The results of the samples are posted regularly along with dose estimations.

Airborne Tritium Dose Control

Time

As with other radiation hazards, reducing the exposure duration will reduce the total intake. Proper work planning can play an important role in reducing exposure duration.

Figure 11:
Tritium Monitor



Measurement of Airborne Tritium

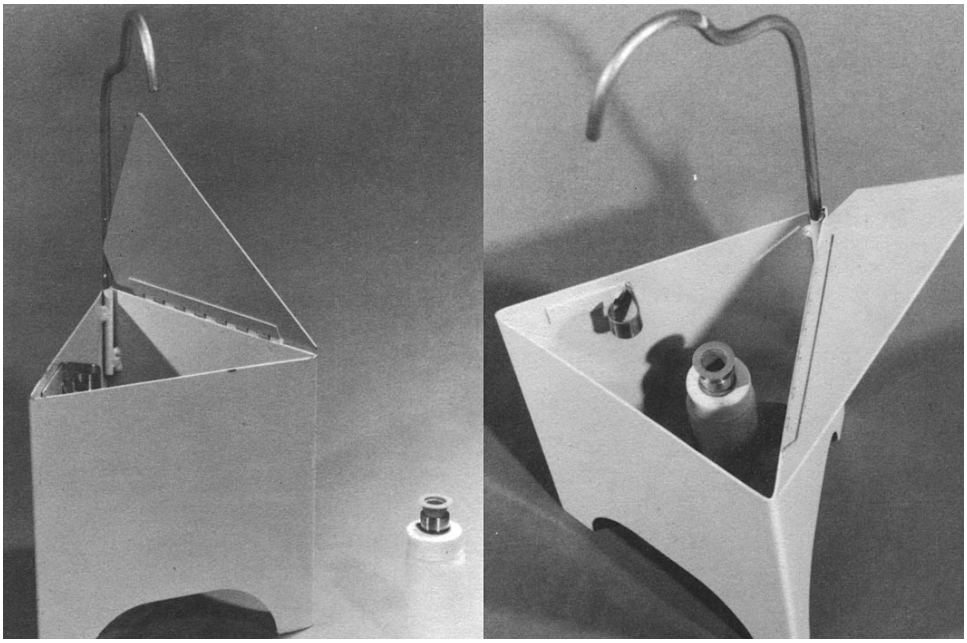
The Tritium Meter

The tritium meter is an electronic instrument which draws a sample of air into an ion chamber and measures the ionization current due to the presence of a radioactive gas inside the chamber. Unfortunately there are factors which complicate this seemingly simple concept. The most important is that external gamma radiation also causes ionization inside the chamber. This is compensated for by having two identical chambers; one being sealed. The ion current from the sealed chamber is subtracted from that of the tritium chamber to get the contribution due to the tritium concentration in the air. This system works unless the gamma radiation field (5mrem/hr max) is non-uniform. However it is possible to carry the instrument into the area of interest, to fill the chamber with a sample of air, then to turn off the pump, carry the instrument to a low gamma background area, and obtain a reading of the HTO concentration in air. Another problem is that the ion current from 1 DAC of HTO in air is very small. It is easily swamped by small concentrations of other radioactive gases in air. To correct for this readings can be taken simultaneously by two instruments, one has a drier in the intake line to remove the HTO vapour. Again the HTO concentration is obtained by the difference of the readings.

The Tritium Bubbler and Collector

The tritium concentration in air can also be measured by means of a 'Tritium Bubbler' or a 'Tritium Collector'. These devices have the disadvantage that a sample obtained from their use must be taken to a laboratory for preparation and counting to obtain the information needed to estimate the airborne concentration.

Figure 12:
Tritium Collector



A tritium bubbler operates by passing air containing tritiated water vapour

through a container holding ordinary water. The tritiated water in the air exchanges with the ordinary water and essentially remains in the water. The amount of tritium in the water is determined in the same way as for tritium in urine and the airborne tritium concentration may then be calculated.

Tritium (Diffuser) Collector

A 'Tritium Collector' operates by allowing tritiated water vapour in the air to diffuse into a liquid scintillation counting vial through an orifice. The vial contains a wet filter paper which quantitatively absorbs the HTO diffusing into the vial. If the exposure time of the vial is known, nominally 10min, the HTO concentration in air can be calculated from the measured count-rate.

Protective Equipment.

Protective equipment is necessary to prevent, or reduce exposure to HTO in air, or as liquid. Protective equipment can provide protection against intake by inhalation, or against intake by skin absorption, or both.

Protection against intake by inhalation requires the use of respiratory protective equipment, of which there are 3 forms:

i) Tritium respirator - an 'air purifying' cartridge attached to either half-facepiece or full-facepiece masks. The cartridge has a useful life time of about 1 hour before HTO vapour begins to penetrate.
ii) Air-line Supplied Respirators - either half, or full- facepiece masks are attached to a compressed breathing-air line.
iii) Airline supplied plastic suit and Hood.

The Protection Factor (PF) of a respirator or suit is defined as the ratio of the uptake when not protected to the uptake while protected.:

$$PF = \frac{\text{Uptake when not protected}}{\text{Uptake when wearing protection}}$$

Both forms of respirator described above prevent the inhalation of tritiated water vapour, but do not prevent uptake by absorption through the skin. They are accorded a protection factor of two.

Airline supplied suits with hood, provide protection factors (PF) of about 500 against HTO uptake by inhalation and absorption through skin when they are supplied with air. When the suits and hood are not supplied with compressed air, they do not provide protection. (i.e PF = 1). It is therefore imperative that the suit user should plug into an airline as soon as possible after he enters his work area. Failure to do this results in a rapid deterioration in the average level of protection afforded by the suit and hood.

Calculation of Dose Commitment from a Tritium Exposure

1) calculate your dose commitment after you have worked for 2.5 hours in a

room where the HTO concentration is 12 DAC. (An exposure of 1 DAC-hour means an effective dose commitment of .01 mSv.)

Dose = Dose-rate x time
Dose Commitment = 12 (DAC) x 0.01(mSv/DAC-h) x 2.5 h
= 0.30 mSv (30 mrem)

2) Calculate how long you could stay in the same room if your dose must remain below 0.5 mSv.

Time = Dose/dose-rate.
Dose = 0.5 mSv, Dose rate = 12 x 0.01 mSv/h
Time = 0.5 mSv/0.12 mSv/h .
= 4 h 10 mins

3) How long could you stay in the above area if you wear a tritium respirator or an airline supplied mask ?.
Both respirators provide a PF of two.

Time = Dose/(dose-rate / PF)
Time = 0.5 mSv/((12 x 0.01 mSv/h)/2)
= 8 h 20 min.

Airborne Activity Other Than Tritium

The following types of airborne radioactivity other than tritium are encountered on nuclear stations:

i) Noble gases
ii) Radioiodines
iii) Particulate activity
iv) Carbon-14

For the radioiodines, particulate activity and C-14 the objective is to prevent intake. This is normally done by establishing an airborne concentration at which respiratory protection is required. This should be some small fraction of the DAC for the radionuclides being encountered.

Noble Gases

The noble gases encountered are primarily Krypton-88, Xenon-133, and Argon-41, Kr-88 and Xe-133 are fission products. Ar-41 is an activation product, produced from naturally occurring Argon-40 when air is irradiated by neutrons. The noble gases are an external beta and gamma radiation hazard.

The noble gases are found in the reactor vaults, in areas housing vault accessories, areas containing irradiated fuel handling equipment and areas containing heat transport system components. The irradiated fuel storage bay may also contain fission product noble gases and their radioactive solid daughter products.
Protection measures are based on measuring the external beta and gamma radiation levels produced by the gases and limiting exposure time. The noble

gases themselves are not absorbed in the body to any extent so are not an internal hazard but their decay or daughter products are particulate activity. However they are short lived (about 20 minutes) and so any associated effective dose is very small compared with the external dose from the parent gases and particulate daughters.

Radioiodines

The radioiodine fission products are of considerable importance in radiation protection in nuclear stations is the radioiodines. Three of ten fission product radioiodines have radiological half lives and fission yields to be of significance. They are I-131, I-133 and I-135. Radioiodines are normally taken up readily by the thyroid which is the critical organ in terms of effective dose.

Table 2

Radiological Properties of the Significant Radioiodines.

Radionuclide	Radioactive Half-life	Effective Half-life	DAC (Bq/m3)	Dose/kBq in Thyroid
I-131	8.05 d	7.4 d	4×10^2	1.7 mSv
I-133	20 h	20 h	3×10^3	0.46 mSv
I-135	6.7 h	6.7 h	2×10^4	0.15 mSv

Annual Limit of Intake for I-131

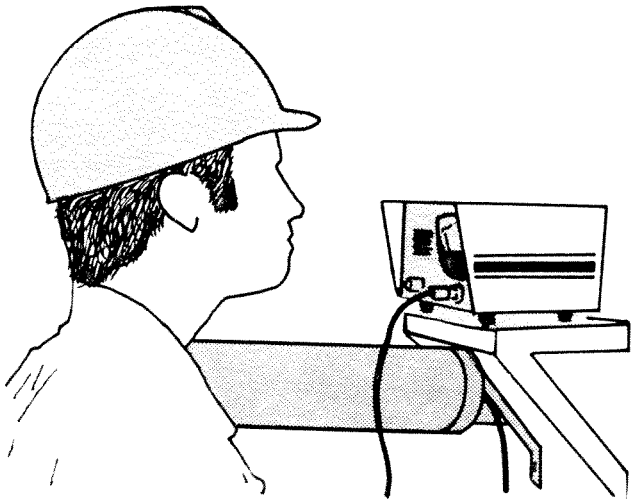
The ALI for I-131, that is the intake which gives an effective dose of 20 mSv is 10^2 Bq. The Derived Airborne Concentration, that is the concentration which will give a dose of 20 mSv if breathed for 2000 hours is $4x10^3$ Bq/M^3. The last column of Table 2 gives the committed dose equivalent to the thyroid for a thyroid uptake of 1 kBq..

Unlike tritium, radioiodines are only encountered occasionally. They are frequently found together. Any limit set for the wearing of respirators on the basis of the presence of I-131 must take in to account that I-133 and I-135 may be present.

Bioassay for Radioiodines

Detection and control of uptake of radioiodine by direct measurement is relatively simple. I-131 emits characteristic gamma photons which may be detected by a monitor placed against the neck. (See Figure 13). The detector is a scintillation detector and a gamma photon analyzer set to detect the gamma photons from I-131. The monitor is set at a small fraction of an ALI. The measurement is normally carried out by station staff themselves using a thyroid

Figure 13:
Self Check Iodine in Thyroid Monitor



monitor set up in the change room. If the alarm point is exceeded, the individual must have a more exact measurement of the three significant radioisotopes of iodine which could be present so that the internal thyroid dose may be assessed.

Radioiodine Measurement and Respiratory Protection

Radioiodine in ambient air is measured by collecting the radioiodines present in a measured volume of air. The collecting is done by passing the ambient air through an iodine filter.

Radioiodines may be present in reactor areas as :
i) Free elemental iodine vapour
ii) Elemental vapour attached to particulate
iii) Organic Iodine

The Organic iodine form is the most difficult to collect on filter absorbers (see the discussion on respirator filters).
The collected samples are analyzed on a scintillation detector which detects the characteristic photons given off by the radioiodine isotopes.

Iodine uptake can essentially be prevented by the use of air supplied masks or respirators employing appropriate filters . Air purifying respirator cartridges must contain a special charcoal to be effective in absorbing organic radioiodines from moist air. The filter cartridge must also have a high efficiency particulate filter (similar filter packs are used for iodine sample collection). This type of cartridge can give protection factors (PF) of 10 and 50 respectively with properly fitted half- and full-facepiece masks. Airline supplied masks or hoods can provide PF of better than 1000.

Thyroid Blocking

Thyroid blocking is the term given to the practice of flooding the body with stable iodine either prior to, or shortly after, exposure to radio-iodine. The

objective is to provide a lot of normal iodine in the body so as to reduce the amount of radioiodine getting to the thyroid. The process is effective in reducing the dose to the thyroid if the blocking agent (a 100 mg potassium iodide or potassium iodate pill) is given within about 6 hours of the exposure. Potassium iodide used to be commonly used in cough medicines. There no evidence of any harmful side effects in this treatment.

Airborne Particulate

Airborne particulate activity may arise in a variety of locations and from various activities. Generally airborne particulates may be considered as two classes - Short Lived and Long-Lived.

Short Lived Particulate

Short lived particulate are the solid daughter products of the fission product noble gases. The noble gases escape readily from defective fuel so this type of particulate activity may be found in locations where fuel is handled e.g. fuelling handling rooms and irradiated fuel bays.

The noble gases of interest are:

	β		β	
Kr-88	$\Rightarrow$	Rb-88	$\Rightarrow$	Sr-88
$T_{1/2}$ =2.8 h		$T_{1/2}$ = 18 min		stable

and

	β		β	
Xe-138	$\Rightarrow$	Cs-138	$\Rightarrow$	Ba-138
$T_{1/2}$= 17 min		$T_{1/2}$ =32 min		stable

The beta emitted by Rb-88 is very energetic having an energy of 5.4 Mev. This type of short lived activity is accompanied by the parent noble gases. The combination is an external beta/gamma radiation hazard. The daughter products are a nuisance in that they may cause widespread surface contamination. They are an additional nuisance in that they may mask the presence of a long lived particulate activity. A common practice is to set a limit for respiratory protection based on the possibility that there may be 10% of long lived activity present.

Long-Lived Particulate

This type of airborne activity is frequently related to the state of the fuel in the heat transport system. If badly damaged fuel is or has been present in a system then maintenance work on the system or equipment from the system may give rise to a variety of long lived airborne radionuclides in air. Activation products may also give rise to airborne activity during maintenance.

Measurement Techniques.

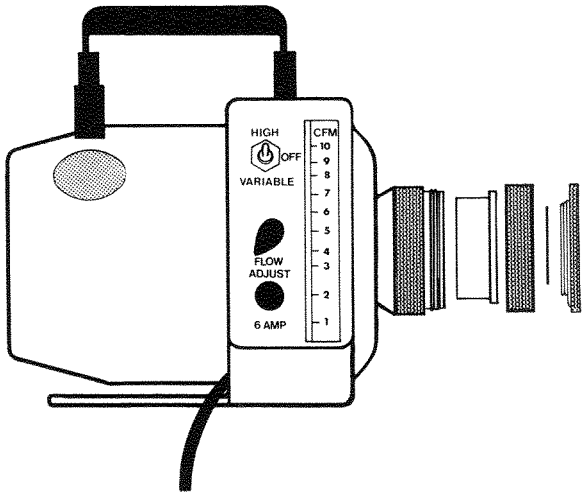
Particulate activity in air is measured by collecting the activity on filters using air

sampling pumps. This can be done continuously or by spot sampling. The latter technique is simple and with knowledge of the volume of air sampled and the amount of activity collected on the filter the airborne activity at the time of collection can be determined. Continuous monitors are of two types, fixed filter and moving filter.

Interpretation of moving filter monitor readings is quite simple. The monitor has a detector in close proximity to a moving filter which is collecting activity from air being pumped through it. The monitor can be regarded as a series of spot samples and the readings are read directly in Bq /m^3 or similar units.

Fixed filter monitors consist of a detector in close proximity to a filter which is fixed. The detector measures the activity which collects on the filter. The activity is read out on a recorder and the readings required some interpretation. If long lived activity is present in the air being sampled the recorder trace will increase with time and the greater the slope of the trace the higher the particulate activity level in the air. If no long lived activity is present the recorder trace will be a level. For short lived activity which is commonly encountered in nuclear stations when activity is first present the recorder trace will increase with time. After a period the trace will level out to a steady unchanging reading. At this stage the activity is being collected at the same rate at which it is disappearing by radioactive decay. Interpretation of fixed activity monitor readings is not simple and these instruments should be employed simply to alert personnel to the presence of airborne activity and identify the need for spot sampling. Sample collected as well as being measured for gross beta-gamma activity should be measured in a gamma spectrometer for identification of specific radionuclides present.

Figure 14
Radeco Air Sampler



The level at which respiratory protection is mandatory for the short lived particulate in Ontario Hydro is 667 Bq/m^3 ($4x10^4$ dpm/m^3).

Particulate Intake - Behaviour in the Body

Inhaled radionuclides are deposited in the respiratory system. Their fate depends upon the particle size, and upon the solubility of the particles. Soluble particles will be transferred to other organs (or as in the case of Caesium distributed throughout the whole body). Insoluble particles deposited in the trachea or bronchi can be elevated to the oesophagus and swallowed, while insoluble particles in the deeper lung may be removed to lymph nodes or may remain where they are and continue to irradiate the lung for a long time.

Methods of Protection

In some cases it is possible to use a continuous air sampler, which alarms if some preset level of activity on the collecting filter paper is exceeded. Spot samples are usually necessary to confirm the indication from the continuous monitor.

If a continuous sampler alarms, then personnel are required to leave the area. A spot sample should then be taken by personnel wearing appropriate respiratory protection to obtain an exact measure of the airborne contamination level.

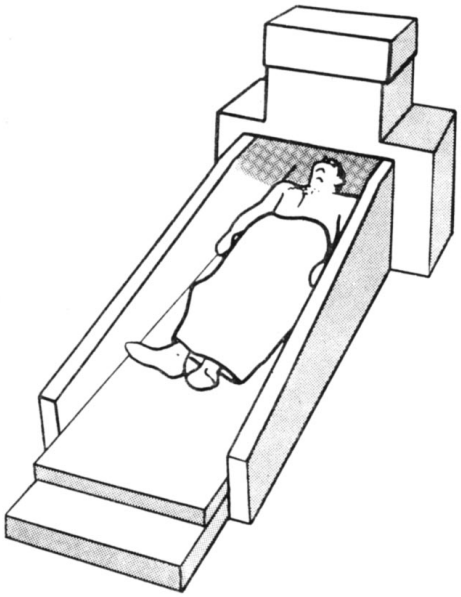
The level at which respiratory protection is mandatory in Ontario Hydro for unidentified long lived airborne particulate activity is 33 Bq/m^3 (2000 dpm/m^3). At NB Power the DAC for Co-60, 2×10^2 Bq/m^3 is used as the level for unidentified long lived particulate airborne activity.

Whole Body Monitoring

The 'Whole Body Monitor' is used as a general surveillance monitor to measure gamma emitting radionuclides which may have been deposited in body organs after an intake. The device is a large shielded scintillation detector (8" x 4" sodium iodide detector) with a bed which moves under the detector. The device may be fixed or transportable in a trailer. The subject lies flat on the bed which then passes underneath the detector. Some of the gamma photons from the radioactive material within the body enter the detector. Light pulses from the crystal are fed to a gamma photon spectrometer which determines the energy of the emitted radiation. Interpretation of the spectrum leads to the identification of the radionuclides present. The height of the peaks can be used to determine the quantity of each radionuclide present. It takes about 7 minutes to complete a measurement.

Because the whole body monitor can not distinguish between contamination present on skin or clothing and that deposited inside the body, the subject being measured is required to shower, wash hair and clean finger nails, and to don a new disposable coverall, immediately prior to being counted.

Figure 15:
Whole Body Counting



Whole body counting is generally done on a six monthly or annual basis, but non-routine measurements may be carried out on request if an intake has occurred or is suspected. Figure 15 shows a person passing underneath the sodium iodide crystal of the whole body monitor.

Carbon-14

C-14 is an activation product which has at times caused difficult problems at nuclear stations. It is produced by neutron irradiation of Nitrogen-14 (n,p reaction). It is found in moderator, heat transport and annulus gas systems. The common form is as CO_2 gas. This form does not cause any significant radiation protection problems. However some of the C-14 produced in the annulus gas system and encountered during retubing at Pickering is an insoluble particulate form. This form is not a serious problem in terms of ingestion, but presents very difficult measurement and dose assessment problems if inhaled.

C-14 is a pure, low energy, beta emitter (Emax= 0.156 Mev). This means that it can only be detected by thin window detectors such as end-window geigers, or scintillation detectors covered with thin aluminized foils.

C-14 in insoluble particulate form is of concern because it is extremely difficult to measure the amount deposited in the lung. The beta radiation emitted is absorbed in a few millimetres of lung tissue and so does not reach any external detector The airborne concentration used in Ontario Hydro for insoluble, particulate C-14 is 333 Bq/m^3 (20,000 dpm/m^3). New Brunswick Power uses a DAC of 2×10^4 Bq/m^3. Exposure at any detectable level is best prevented. The

control measures are therefore designed to prevent inhalation. Respiratory protection, preferably airline supplied hood (PF = 1000) is required at airborne concentrations exceeding 0.02 DAC. In addition personnel working in areas where C-14 exceeds 0.2 DAC are required to use a personal air samplers (sampling air from their breathing zone) to measure their airborne exposure. The results from these personal air samplers are recorded as primary dose information.

Surface Contamination

Liquid leaks from Heat Transport and Moderator systems carry radioactive materials that leave solid residues when the liquid has evaporated. Reactor systems and fuelling machines are contaminated internally; work on these systems or component from these systems may lead to the transfer of contamination to protective gloves and clothing, tools and miscellaneous surfaces.

Airborne contamination (discussed previously) may deposit (or fall-out) on surfaces to form a layer of relatively loose surface contamination, which can build up, over a period of time.

Areas which are normally contaminated to some extent include:
- Irradiated fuel facilities
- Vaults
- Rooms containing Heat and Moderator system components
- Decontamination facilities
- Zone 3 ventilation systems.

Surface contamination may be:
- Loose contamination
  - this is easily wiped off a surface.
- Fixed contamination
  this resists removal even by fairly vigorous decontamination techniques. (techniques which remove the surface are likely to be effective in removing fixed contamination.)

Loose surface contamination presents an ingestion and an inhalation risk as it may give rise to airborne contamination levels. Measures are usually taken to remove loose contamination or to adopt procedures which will minimize the risk of intake and the spread of the contamination to other areas.

Fixed contamination may also be an external radiation hazard (particularly in the form of beta doses to the hands).

Limits have been established for control of beta/gamma surface contamination. These are known as Derived Working Limit (DWL). The DWL for beta/gamma surface contamination is $3.7 \times 10^4 Bq/m^3$ (1 $\mu Ci/m^2$). This level of contamination (as fixed contamination) may be on the surface of items transferred from a controlled area to an uncontrolled area for unrestricted use. Other DWL are sometimes established for protective clothing and tools which may not leave controlled areas.

Methods of Protection

The basic principles used for the control of contamination are:

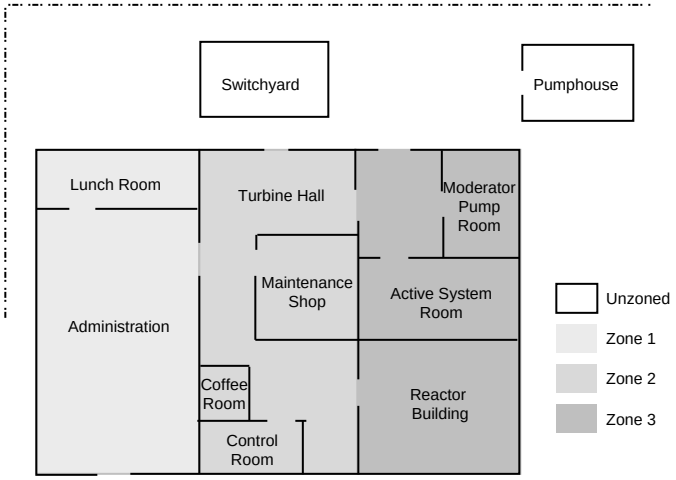
- By Monitoring - that is by using surface contamination monitoring and other techniques (smear techniques etc.) to measure the levels.
- By practising Cleanliness - that is cleaning up contamination when it is found.
- By Containment - that is by keeping identified contamination within tightly controlled areas of the smallest possible size.

Examples of containment range from encapsulation of sources, hot cells, glove boxes, fume cupboards, to zone controls.

The Concept of containment is incorporated into the total generating station environment by establishing Control Zones:

Zone 3 - Radioactive work may only be carried out in Zone 3.
Zone 2 - an area surrounding Zone 3. Radioactive work is not allowed in this area. It forms a 'buffer' area which may be slightly contaminated by inadvertent transfer from Zone 3.
Zone 1 - A clean Zone with no radioactive contamination.

Figure 17
Typical Contamination Zones in a Station



Good ventilation system design also prevents the spread of both surface and airborne contamination. Work involving high levels of contamination should be carried out in fume hoods or in temporary ventilation enclosures. The design intent of the main ventilation in stations is to provide movement of air form zones of lower contamination levels to higher, that is from clean to less clean areas. Problems are frequently encountered with ventilation systems. It is often difficult to establish the correct air flow patterns and after they are set up they can easily be disturbed, for example. by leaving doors open or by starting up fans that are not normally operating. Problems were experienced at Bruce 'A' when certain large overhead doors in the powerhouse building were left open in windy condition;ns. This disturbed the normal air flow patterns and contamination spread from a waste and flask management area throughout the building.

Surface Contamination Monitoring

There are two types of contamination monitors - Fixed and Portable

Fixed Monitors can also be split into two types:

- Portal Monitors:
  These are designed to determine if a person passing through the monitor is carrying contamination.
- Hand and Foot Monitors:
  These are designed to measure the amount of contamination on hands and feet, and to alarm if the amount present exceeds a predetermined value.
- Some facilities employ foot monitors only, for special application

Fixed Contamination Meters

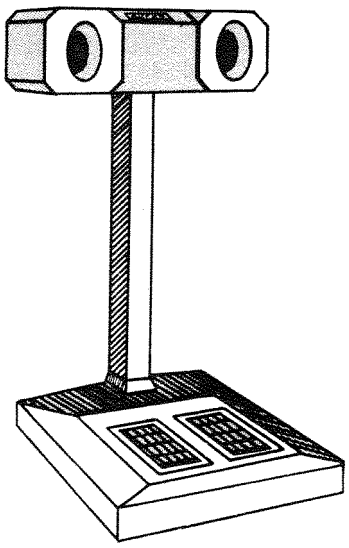
Portal Monitors are installed at the boundary of Zone 2 and Zone 1 the clean (administrative) area. Their function is to prevent people carrying radioactive contamination into the administrative area.

Early forms of portal monitor measured the change in the background count-rate perceived by the monitor. If the change exceeded 4 times the standard deviation of the long term background count rate, then the monitor alarmed. This system could detect gross levels of contamination on a person’s clothing, or a significant intake of a nuclide like radioiodine.

The latest Portal monitors are very sophisticated and sensitive. The instrument scans the front and back of personal clothing, and hands and feet, for beta/gamma contamination, They are also able to detect low energy beta emitters such as carbon-14 and have compensation for background gamma radiation levels.

Hand and foot monitors are installed at zone 3/2 boundaries, and sometimes additionally at zone 2/1 boundaries. Their function is to detect contamination

Figure 18
Hand and Foot Monitor

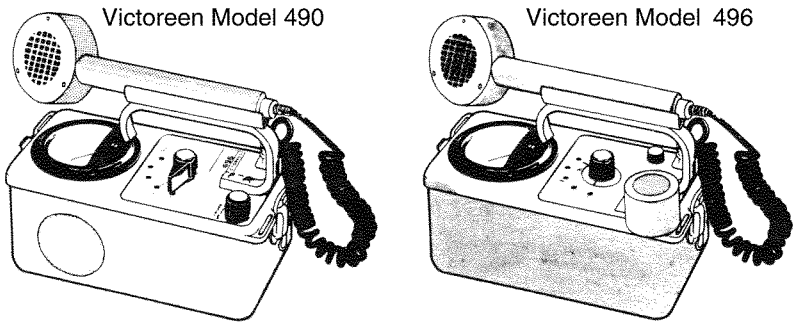


on hands and the soles of footwear, so that it is not distributed into the zone 2 or zone 1 areas. People are required to decontaminate hands and footwear if it triggers an alarm. These monitors also have an attached contamination hand probe which can be used to locate or confirm contamination on hands and footwear. It may also be used to check clothing and miscellaneous items such as papers, books, brief-cases etc.

Portable Contamination Meters

These units are used for monitoring miscellaneous items and surfaces for contamination. Most of these devices are now equipped with thin (7 mg/cm^2) windows which make the detector sensitive to low energy betas (e.g.C-14). It is important to know that this type of detector is primarily a beta particle detector (detection efficiency 5-15% depending on energy and other factors). The sensitivity to gamma photons is relatively low - in the range 0.1 to 0.5%.

Figure 19:
Portable Contamination Meter



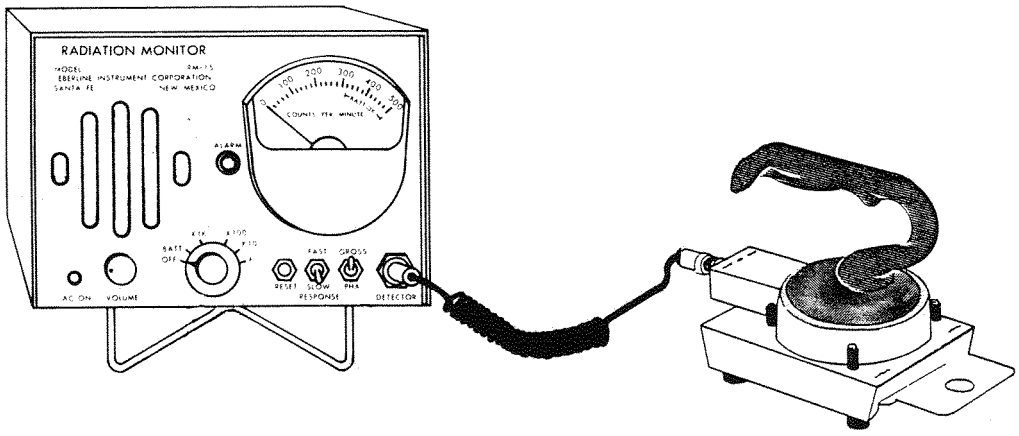
Monitoring for surface contamination is difficult and tedious and should be done in a low gamma background. The probe must be moved at a slow speed over the surface (e.g.less than 10 cm/sec) and the probe to surface distance should be about 1 cm. It is necessary to move the probe over the whole surface. If areas are missed areas contamination may be missed, starting from the area closest to your body (so as to locate any high sorces near your body first) in a paint brush fashion or motion.

There are many surfaces which can not be monitored directly:

- Internal surfaces of tubes or pipes
- external surfaces with many curves
- items or surfaces in high gamma background areas etc.

Smears are also used to detect contamination on equipment which is activated or contains a source, where the radiation field from the material would swamp the detector, e.g. when checking a shipping flask containing a source prior to shipment.

Figure 20:
SmearCounting Arrangement



In these circumstances 'smears' or 'swipes' are taken of the surface of interest. The smears are taken to a 'smear counter' and counted in defined conditions. Assumptions are made about the surface area smeared (cm^2), and the percentage of activity transferred to the smear, and the efficiency of the detector in order to calculate the amount of activity on the smear and determine the level of contamination on the item of interest.

The usual assumptions are 10% counter efficiency, 10% wipe off from the surface of interest, and 100 cm^2 area smeared.

It is normal for those engaged in radioactive work to change routinely into

Protective Clothing. This basic protective clothing is intended to:

i) provide reasonable assurance that the worker's skin and personal clothing will not become contaminated with radioactive material,
ii) provide reasonable assurance that radioactive contamination will not be carried out of the station on personal clothing.

Specific items of protective clothing are also provided to ensure that the worker is adequately protected against common industrial hazards and against specific hazards associated with radioactive materials.

Standard Protective Clothing

Any person who works routinely in a radioactive area or who is doing radioactive work will change from street clothes into standard protective clothing. This consists of a one piece cotton coverall, cotton underwear, socks and safety shoes. The clothes are usually an easily distinguished colour - brown is often used for the coveralls and yellow for the rest of the clothing. A change room is provided in which street clothes are completely removed and stored in their own locker. The normal protective clothing is collected from a clothing crib, then donned in a second change room.

Special Protective Clothing

Many kinds of protective clothing are available in a nuclear station. Most is the type that can be found in any industrial occupation for protection against a wide range of occupational hazards and as far as is possible this type of protective equipment is used against radiological hazards. Even so, there are a number of special items that provide protection against certain specific radiological hazards.

Examples of special protective equipment include:

- Lead Gloves -leaded rubber gloves used to reduce the dose rate to the hands from beta contamination on surfaces.
- Tritium Respirator - an air purifying respirator with a cartridge for absorbing tritiated water vapour from inspired air.
- Iodine Cartridges - air purifying respirator cartridges used to remove radioiodines from inspired air. This will also remove most other forms of radioactive particulate activity.
- Air Supplied Plastic Suits and Hoods - used primarily to protect workers against the intake of HTO by respiration and absorption through the skin.
- "N" type canister/filter mask combination can be used in atmosheres containing mixed airborne radiation hazards.
- Many kinds of rubber gloves are available to protect against surface contamination. Sometimes heavy weight gloves are used as beta radiation absorbers, when the dose rate is not high enough to warrant the discomfort of the 'lead' gloves.

Sources of Occupational Dose in CANDU Stations

Table of Content

Introduction

CANDU Nuclear Generating Stations, were intended to utilize materials and technologies available in Canada. The original 20 MW(e) NPD G.S at Rolphton, owned by AECL, operated by Ontario Hydro, (now shut down) was developed and enlarged through successive stations to the 881 MW(e) units now being installed at Darlington (designed and built by Ontario Hydro). Since the early '70's AECL has concentrated on the CANDU-600; a 600 MW(e) single unit station which was considered most likely to meet the needs of many small to medium sized Canadian and Foreign Utilities.

Although there are now 22 operating nuclear power reactors in Canada, there is enough continuity of design, from one station to another, to be able to discuss the radiation hazards, both internal and external, as they apply to all stations.

New design features are generally associated with improvements in operating efficiency or safety. Some components or materials in the older stations, which were the source of radiation protection problems, have been improved, or removed in the newer stations, resulting in a reduction in radiation hazards.

The CANDU Generating Station

The CANDU generating station design is illustrated in Figures 1A and 1B on the following pages.

The Reactor Core

The Reactor Core is housed in a Calandria (or tank) which is essentially a horizontal right cylinder with vertical end faces. The tank is filled with high purity heavy water which serves as the Moderator. Two sets of horizontal tubes pass through the tank. The first set, the calandria tubes, penetrate through the vertical faces. They are sealed to the faces. Between 390 and 480 such tubes (depending upon the power output of the reactor) are installed in a lattice arrangement. The second set of tubes, the 'fuel channels', are installed inside the calandria tubes. Twelve or thirteen fuel elements (the actual number depending on design) are inserted into each fuel channel. Heavy water at high temperature and pressure (250 to 265C; 1500 to 1750 psig pressure) is pumped through these channels to remove the heat generated by the fission process.

The Heat Transport System

The fuel channels form part of the Heat Transport System, which transfers the heat from the fuel elements to a steam generator. Here the heat is in employed to boil light water forming steam to drive the turbine. There is no fluid transfer

Figure 1A
CANDU Reactor Simplified Flow Diagram

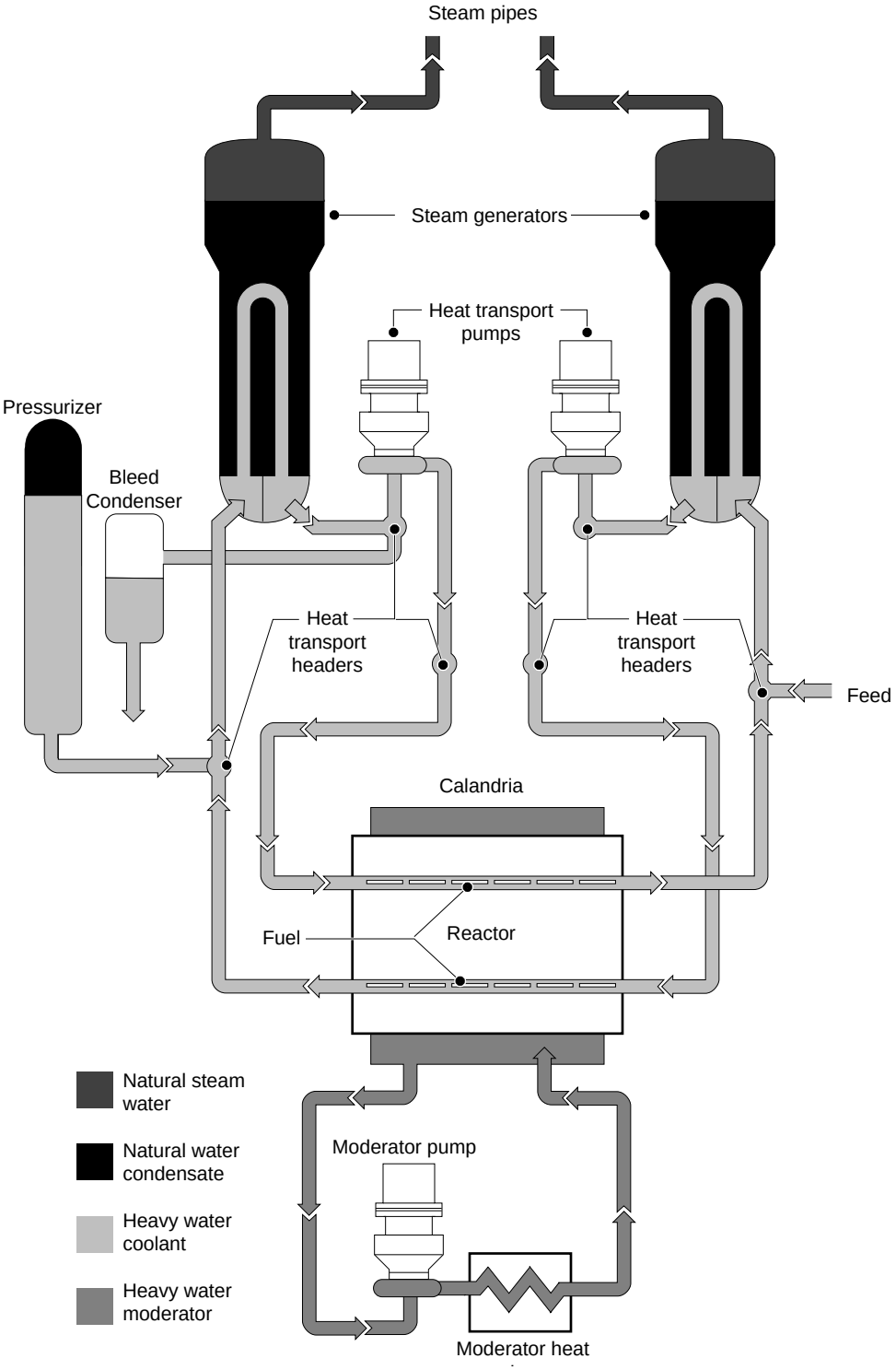
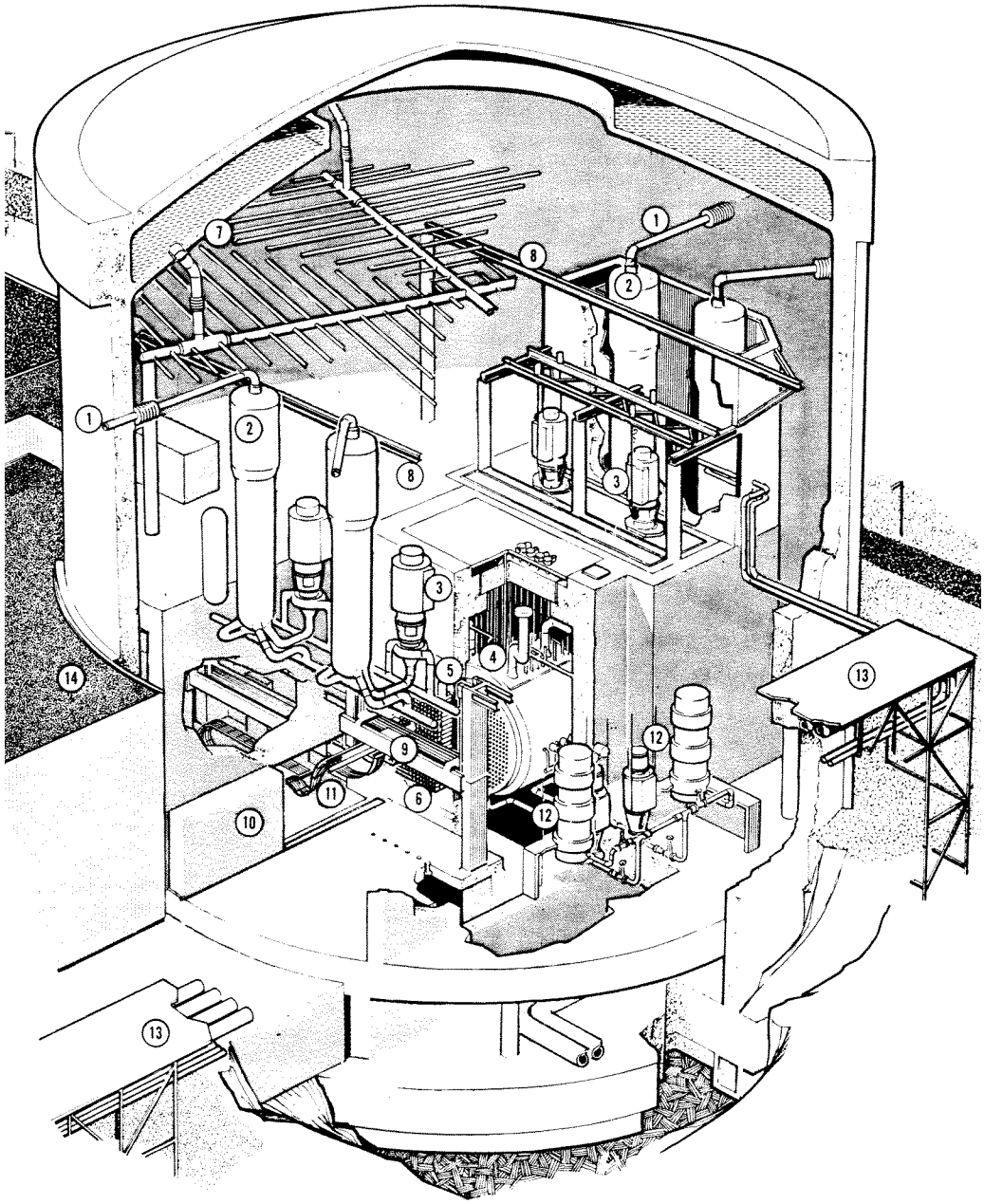


Figure 1B
CANDU Reactor Building



1 MAIN STEAM SUPPLY PIPING
2 BOILERS
3 MAIN PRIMARY SYSTEM PUMPS
4 CALANDRIA ASSEMBLY
5 FEEDERS
6 FUEL CHANNEL ASSEMBLY
7 DOUSING WATER SUPPLY
8 CRANE RAILS
9 FUELLING MACHINE
10 FUELLING MACHINE DOOR
11 CATENARY
12 MODERATOR CIRCULATION SYSTEM
13 PIPE BRIDGE
14 SERVICE BUILDING

between the heavy water (primary side) and light water (secondary side) of the steam generators (or boilers).

A purification system is installed to remove fission and activation product impurities from the circulating water.

The Moderator System

The heavy water moderator in the calandria tank is heated by radiant and conductive heat from the fuel channels. It is therefore equipped with its own pumps and heat exchangers to remove this heat. Moderator water is maintained at a temperature in the range 70-85 C. The system has its own purification circuit to remove activation products generated by neutron irradiation of impurities in the moderator water.

On Power Refuelling

An important feature of the CANDU Reactor is 'On Power' refuelling. Fuelling machines can be attached to each end of a fuel channel. Plugs in the end of the pressure tube are removed by the fuelling machines and stored in a rotating magazine. New fuel can then be inserted at one end of the channel by one machine while irradiated fuel is simultaneously removed and stored in the machine at the other end.

Irradiated Fuel Storage

Irradiated fuel is removed from the reactor in the fuelling machines, and transferred via the 'irradiated Fuel Transfer Mechanism' to the 'Irradiated Fuel Storage Bay'.

A facility for servicing fuelling machines is provided at each station.

Containment Building

Each reactor and its associated moderator and heat transport systems are enclosed in a 'Containment Building' The containment building is intended to prevent any leakage of fission or activation products from the damaged reactor to the environment. The containment building of single unit stations includes a light water dousing system which is activated when building pressure rises. This reduces the ultimate pressure rise in the building and makes the containment structure simpler. The Pickering, Bruce and Darlington designs utilize a vacuum building . This is a large building maintained at about 0.1 atmospheres pressure (with an integral water dousing system) connected by a duct to each reactor building. If a major heat transport system break occurs and the pressure in a reactor building increases, large valves in the duct, open and connect the reactor building to the vacuum building. The final resulting pressure is less than atmospheric.

Auxiliary Systems

Each reactor has many secondary systems, essential to the operation or safety of the reactor. Such systems include:

- Heat Transport Purification System,
- Reactivity Mechanisms e.g. Control Rods, Liquid zone control systems,
- Emergency Shut-down System e.g. Shut-down Rods, Poison injection system,
- End-shield cooling system,
- Annulus Gas system,
- Vault ventilation system and driers for D_2O recovery,
- Boiler Room Ventilation System,
- Reactivity Monitoring system,

General Radiation Hazards in a CANDU Station

Introduction

All of the radiation hazards arising in the nuclear station are derived from the fissioning of uranium in the reactor core. The direct hazards from the core include:

Fission Neutrons

- fast neutrons escaping from the core become a radiation hazard in the areas immediately surrounding the reactor.

Prompt Fission Gammas

- gamma radiation during the fission process.

Photo-neutrons

- neutrons emitted from the deuterium atom when it reacts with a high energy (>2.2 Mev) gamma photon.

$$\frac{\gamma}{2.2\ \text{Mev}} + {}^{2}H_1 \rightarrow {}^{1}H_1 + {}^{1}N_0$$

Capture Gamma Radiation

- a neutron may interact with the nucleus of an atom and be captured. The new nucleus formed may immediately emit a gamma ray; this is known as an (n,gamma) reaction.

Activation Products

- a neutron may interact with the nucleus of an atom in many ways being absorbed to form an unstable atom which immediately emits either a proton, an alpha particle or a gamma ray. The nuclide left may be radioactive and is an 'Activation Product'

Fission Products

- most of the product atoms produced when U-235 atom fissions are radioactive; they emit beta and gamma radiation.

Fission neutrons and prompt fission gamma rays disappear shortly (within two minutes) after a reactor is shut down. They are important only in so far as they cause high radiation levels when the reactor is operating, in the vaults, boiler

rooms, and some closely associated areas. These areas are not accessible (i.e.the doors are locked and fitted with security devices which prevent entry or will trip the reactor if they are opened when the reactor is at power). They may be entered for special reasons and purposes when the reactor is at low power, but the entry is always made under close supervision.

Photoneutrons

A distinctive feature of a heavy water moderated reactor is that even on shutdown there is still a sizable neutron flux in the reactor due to neutrons produced by the gamma radiation from fission products interacting with the heavy water of the moderator and heat transport systems. These are known as photoneutrons. This source of neutrons is often forgotten but it may still activate any material which penetrates the reactor core during a shutdown. A flux, usually small, of photoneutrons may also be produced where N-16 a short lived but high energy gamma emitter is present. N-16 is produced in the heat transport and moderator system heavy water e.g in the HT boilers. Photoneutrons may be found there.

Fission Products

Fission products are produced in very high concentrations in fuel bundles during the fission process. Care must be taken when working on or in the vicinity of systems which handle irradiated fuel as life threatening fields may be associated with the equipment if fuel bundles are in the system. Some of the fission products decay very rapidly but others are very long lived.

If defective fuel is present in a system then it will have high associated radiation fields. Rooms and work areas associated with that system may also have high levels of airborne activity due to volatile fission products such as the noble gases and radioiodines escaping from the defective fuel and the system. The irradiated fuel handling systems and the primary heat transport system with its auxiliary systems may at times be contaminated with fission products.

Table 1 gives the volatile and soluble fission products which are commonly found in systems when defective fuel is present.

Table 1

Soluble & Volatile Fission Product Found in Ht System

Nuclide	Half-Life	Nuclide	Half-Life
Kr-85m	4.5 h	Xe-133	2.3 d
Kr-88	2.8 h	Xe-138	17 m
Rb-88	18 m	Cs-138	32 m
I-131	8.1 d	Cs-134	2.06 y
I-133	21 h	Cs-137	30 y
I-135	6.7 h		

Other less volatile fission products may be found in the heat transport or fuelling systems if the damage to a fuel element is severe. These nuclides are listed in Table 2:

Table 2

Additional Fission Products in Systems if Fuel Damaged Severely

Nuclide	Half-life
Sr-89	53 d
Sr-90 + daughter,Y-90	28 y, 64 h
Zr-95	65 d
Nb-95	35 d
Mo-99	66 h
Ru-103	40 d
Ru-106 + daughter, Rh-105	1y, 30 s
Te-132	78 h
Ba-140 + daughter, La-140	12.8 d, 40 h
Ce-141	33 d
Ce-144 + daughter, Pr-144	285 d, 17 m
Nd-147	11 d
Pm-147	2.7 y

Very high levels of fission products have at times been measured in irradiated fuel handling systems as a result of serious fuel failure. Surface beta dose rates on the inside of these systems may reach extremely high levels.

Current practice of lubricating the inside surface of the fuel sheath with graphite has reduced the incidence and severity of damage to fuel sheaths. Also attention to the chemistry of the HT system water has reduced the mobility of the fission products. In addition current operating practice is to minimize temperature cycling maintaining the system at high. This minimizes water being drawn into the fuel sheath and subsequently expelled along with soluble fission products

Activation Products

Heavy Water Activation Products - N-16, N-17 and O-19

Three isotopes of oxygen are present in heavy water - oxygen-16 oxygen-17 and oxygen-18. Each of the isotopes is able to absorb a neutron to form activation products:

$$^{16}O_8 + {}^{1}n_0 = {}^{16}N_7 + {}^{1}p_1 + \text{gamma}$$
$$^{17}O_8 + {}^{1}n_0 = {}^{17}N_7 + {}^{1}p_1 + \text{gamma}$$
$$^{18}O_8 + {}^{1}n_0 = {}^{19}O_8 + \text{gamma}$$

N-16 and O-19 decay with half-lives of 7 and 27 seconds respectively and emit very high energy gamma radiation. When the reactor is operational these two activation products produce a high gamma radiation field and small photo-neutron field in the vicinity of the moderator and heat transport system

equipment (boilers heat exchangers etc.). This hazard disappears three or four minutes after reactor shut-down.

Heavy Water Activation Products - Tritium

Tritium is an activation product formed when a deuterium atom absorbs a neutron:

$$^{2}H_{1} + {}^{1}N_{0} = {}^{3}H_{1} + \text{gamma photon}$$

The tritium is part of a molecule of water, DTO, and behaves like water. Tritiated water concentrations in the moderator D_2O can be high - 1 TBq/kg (about 20 Ci/kg) is not uncommon and if not removed the equilibrium concentrations could be twice as high. Heat transport water is generally about ten times lower in tritium concentration. Fig.4 shows the number of DAC's of tritiated water vapour, in saturated air, versus the concentration of DTO (TBq/kg (Ci/kg)) in heavy water, at various temperatures.

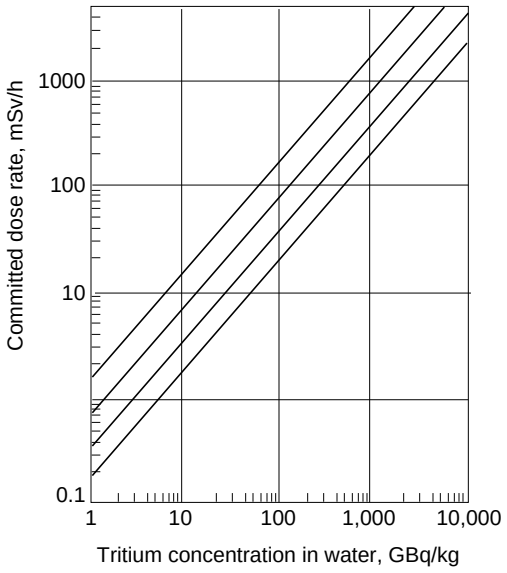


Figure 2
Tritium Derived Air Concentrations in Saturated Air

Up to 2 to 4 x 10^4 DAC (Derived Air Concentrations) are possible. Unprotected exposure for 1 hour at these levels will result in committed equivalent doses of the order of 0.5 or 1 Sv (50 to 100 rem). Experience dictates that entry to moderator areas requires 'double plastics' (a disposable plastic suit over an air supplied suit) to provide better contamination control in the event that moderator water is spilled onto a suit.

General Activation Products

Impurities in the moderator and heat transport heavy water result from corrosion of pipework and other components of the system.

A list of activation products which can be found in the moderator and/or heat transport systems is given in Table 3.

Table 3

Activation Products Found in Moderator/Ht Systems

Nuclide	Half-Life
Cobalt-60	5.3 y
Cobalt-58	71 d
Iron-59	45 d
Manganese-54	300 d
Copper- 64	12.8 h
Chromium-51	28 d
Zirconium-95	65 d
Antimony-124	60 d
Tungsten-187	24 h

Most of the activation products listed in Table 3 are beta emitters. All of them are gamma emitters. They tend to plate out on the internal surfaces of pipes, valves and pumps in the system, and may present significant radiation fields for persons working in the vicinity of these components.

In general, in the first 24 hours after reactor shut-down there will be a reduction in the gamma radiation levels related to activation product activities - perhaps by as much as a factor of 10. This is very much dependent upon the identity of the major activation products present in the system. Cobalt is of great importance because of its production cross-section and its half life. The reduction of cobalt has been a major thrust in the CANDU program (see the section on ALARA).

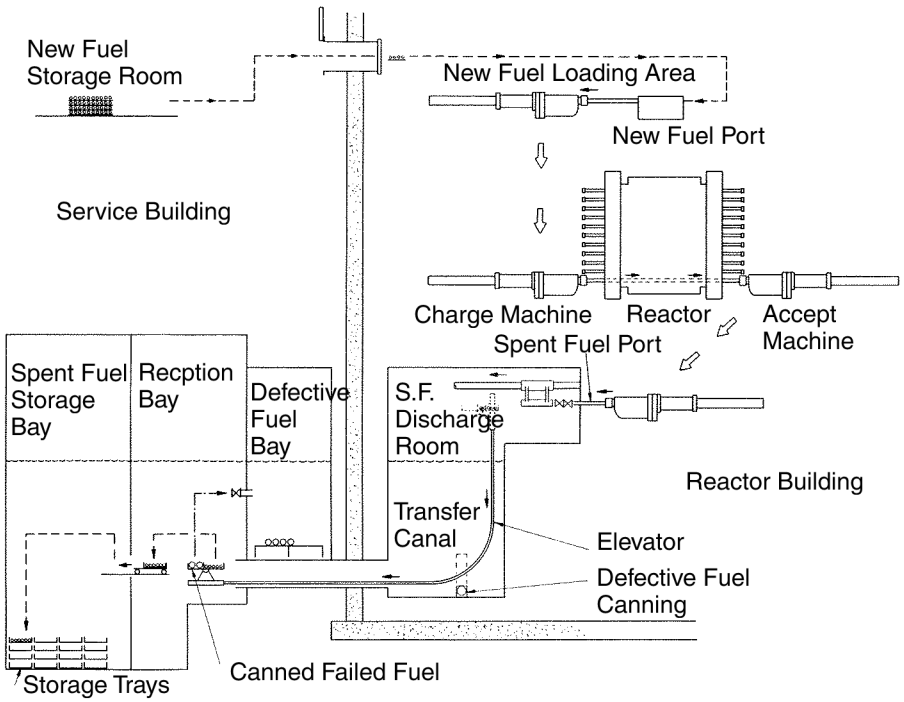
Specific Radiation Hazards Associated with Refuelling

Reactor Re-Fuelling Systems

The radiation hazards associated with new fuel bundles are relatively minor. No array of new fuel bundles is capable of reaching a critical state, so criticality control is not a problem. Dose rates from fuel bundles in a new fuel storage area are of the order of a few tens of microSv/h. There is a surface contact beta-gamma field on a fuel bundle but it is quite low and may be reduced to minor levels if cotton or leather gloves are worn during handling.

CANDU Reactors have a computer controlled, on-power re-fuelling system, in which the machines are raised up into the Fuelling Machine (FM) Vault from a Fuelling Machine Service Area. In the Pickering and Candu-600 units, when the FM is not in the FM vault, a heavy shield can be rolled over the opening in the

Figure 3
Fuel Handling Sequence



floor through which the machine is raised. This shield must stay open while the machine is in the FM vault. When the shield is in the open position, external radiation may shine directly or be scattered into the FM service area. Also airborne contamination from the FM vault may be transferred by air movement into the FM Service Area. In the Bruce and Darlington generating stations the shield is only installed during reactor maintenance shutdown work, when it is desirable to segregate the shut down reactor from possible contaminants introduced into the FM duct by the other operating reactors.

In the process of pulling the end shield plug from the fuel channel, and transferring irradiated fuel into the fuelling machine, radioactive crud (predominantly particles of insoluble metal oxides impregnated with fission and activation products) deposited on the plug and on the fuel element surfaces, is drawn into the fuelling machine where some of it is deposited on the internal surfaces of the machine. The fuel is kept cool while it is in the machine by passing heavy water through the machine. At this time, (with fuel in the machine) the gamma radiation dose-rate from the machine will be dangerously high.

At Pickering, and for CANDU-600 units, the fuel machine is then moved to a 'Fuel Transfer Port', where the fuel elements are pushed into the 'Fuel Transfer Mechanism'. This lowers the fuel down an 'elevator' on to a conveyor belt. While the fuel is in the elevator it is not cooled by water (i.e.it is air cooled) so

the element will experience a rise in temperature. Near the bottom of the elevator, the fuel element is again immersed in water as it is deposited on to a conveyor belt which transports it to the irradiated fuel inspection bay. See Figure 3.

At Darlington and Bruce the fuelling machines are lowered on to a trolley which moves to ports connected to a 'transfer bay' from where the fuel is transferred directly into the fuel reception (or inspection) bay.

Once in the irradiated fuel inspection bay, fission products can leak out of damaged elements. This is controlled by placing damaged elements into a sealed container before they are put into storage. Normal elements are placed into a 'basket' and moved into the storage bay for long term storage.

Radiation Hazards in the FM Service Areas

At Pickering G.S. and in single unit stations, access to the FM Service Area is not allowed while the rolling shield is open, because the area will be subject to direct and scattered gamma and neutron radiations from the FM Vault. In addition there will be tritiated water vapour, noble gas, and associated particulate fission and activation products in the air. A similar situation arises in the Fuelling Machine Duct at Stations such as Bruce and Darlington. (As stated above, the shield between the FM Vault and the FM Duct is not normally in place.)

Access to the FM Service Area at single unit stations and Pickering units is allowed after the rolling shield has been placed over the opening to the FM Vault. Time for the radioactive decay of the short lived airborne activities, and for the ventilation system to reduce the long-lived airborne activity, is usually allowed before entry.

Radiation Hazards in this area will be:
External Gamma:

- The possibility exists of irradiated fuel being in the machines. Procedures and equipment must be in place to prevent entry under these conditions. Under normal circumstances there will be gamma fields arising from activation and fission products in the fuelling machine head.

External/beta gamma:

- There will be contamination on surfaces especially on internal surfaces of opened machines and machine components. Tritiated water vapour will be present in the atmosphere the level depending on the concentration of tritium in the heat transport water and there may be short and long lived particulates and radioiodines.

The FM Service Areas at Bruce and Darlington do not have the same problems of access because they are remote from the opening to the FM Vaults, and physically separated from the associated airborne contamination. The Bruce 'A'

Station however is noted for the relatively high surface contamination on the internal parts of the machines, and on surfaces in the FM Service Area. This is related to the design of the machines, which did not have 'flow through' heavy water to move the crud at the end of the reactor fuel channels away from the machine. This situation is being rectified - the machines are being modified. Also improved contamination control practices have been put in place.

All irradiated fuel handling rooms have high gamma field alarming systems installed to warn personnel of unusual gamma radiation fields.
(Fixed Area Alarming Gamma Monitors, F.A.A.G.M.)

Radiation Hazards in the Irradiated Fuel Transfer Room

At Pickering, and to a lesser extent at CANDU-600 stations, the fuel is not water cooled for the short periods of time that it is on the elevator which carries the fuel from the machine elevation down to the conveyor system (which takes the element to the inspection bay). This means that the fuel elements will heat-up due to the radioactive decay processes occurring in the element. Volatile and semi-volatile fission products are more likely to leak from defective fuel in these circumstances. When damaged fuel is being transferred, fuel transfer rooms and fuel bays have been known to have very high airborne concentrations of both short and long-lived fission products, (Using the old MPCa measures, 1×10^4 MPCa of Ce-144 and similar concurrent levels of Zr/Nb-95, Ba/La-140, Ru/Rh-106, have been measured).

Surface contamination levels in the fuel transfer room can be very high. Entry to these rooms should be delayed as long as possible following fuel transfer to allow the decay of the radioactive noble gases and the short lived fission products. The minimum protection required is an air supplied suit and hood. 'Double plastics' (i.e. a disposable plastic suit worn over the air supplied suit) may be desirable for contamination control purposes.

Radiation Hazards in Fuel Storage Bays

The major hazard is the possibility that one or more fuel elements could be raised high enough in the water to cause very high gamma radiation levels in the bay and surrounding areas. This eventuality is catered for by installed gamma alarming monitors (F.A.A.G.M.).

The more usual hazard is the presence of radioactive contamination on surfaces around the bay, and in the bay water, related to releases of fission product contamination into the pool water from damaged fuel elements. Radioactive noble gases and radioiodines will be released from the bay water to the air in the Irradiated fuel Bay. Radioiodine releases are considerably reduced by the addition of small amounts of Hydrazine to the bay water. Fuel bundles which are leaking badly are usually canned to prevent further releases. This event is catered for by the use of portable or fixed continuous particulate in air monitors.

Specific Radiation Hazards Associated with the Moderator

Introduction

The moderator is high purity heavy water. It fills the calandria and is subjected to intense neutron and gamma radiation fields. Moderator water is heated by conduction and radiation from hot fuel channels. Fuel channels are located in the centre of the calandria tubes by 'garter springs'. See Figures 4A and 4B. The garter spring supports the fuel channel inside the calandria tube. It also creates an annular air gap around the fuel channel which acts as an insulator between the hot fuel channel and the cool calandria tube.

Pumps circulate moderator water through a heat exchanger, filters and ion exchange purification columns. The water returns to the calandria via jets which help to circulate the water inside the tank and maintain a reasonably uniform temperature.

Figure 4A

Reactor Assembly

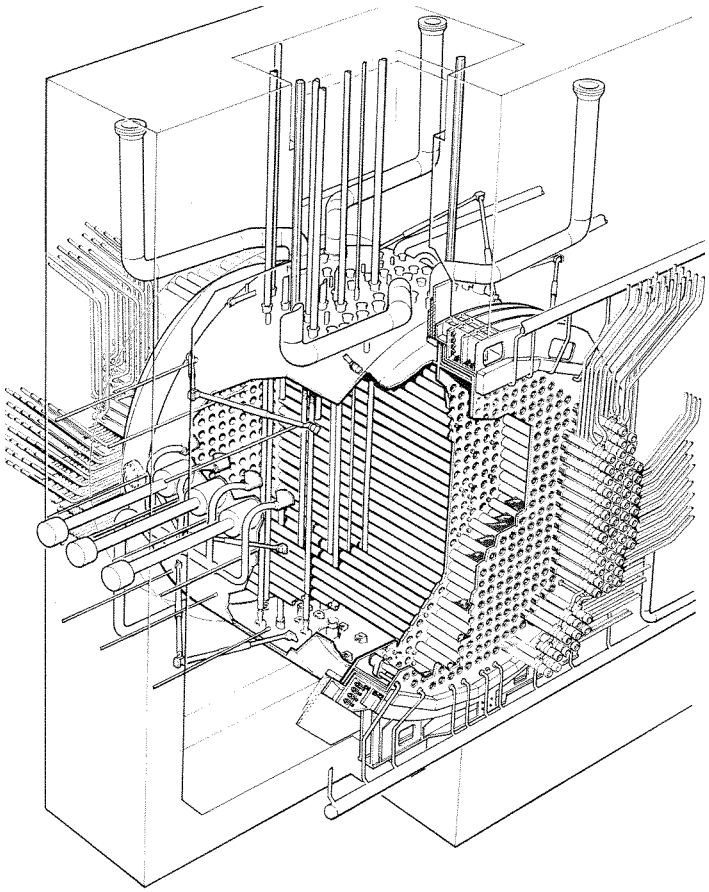
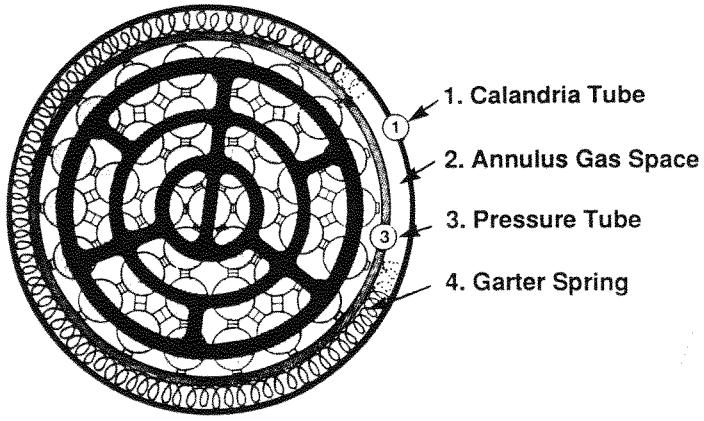


Figure 4B
End View of Fuel Bundle



Gadolinium (a neutron absorber) nitrate solution can be injected into the moderator tank via an array of jets to initiate prompt shut-down of the reactor. Gadolinium and Boron are both introduced into the moderator system for reactivity control.

Tritiated Water Handling and Airborne Tritium

Moderator water as previously described has a high tritium content. Figure 5 shows the number of DAC's of tritiated water vapour, in saturated air, versus the concentration of DTO in heavy water, at various temperatures. Using a concentration of 370 GBq/Kg (10 Ci/Kg) then it is possible to have airborne concentration of up to 4×10^4 DAC (Derived Air Concentrations).

Unprotected exposure for 1 hour at these levels will result in committed equivalent doses of the order of 0.5 or 1 Sv (50 to 100 rem). Entry to moderator areas requires 'double plastics' (a disposable plastic suit over an air supplied suit) to provide adequate protection in the event that moderator water is spilled onto a suit.

N-16 and O-19 fields are produced in moderator water and moderator equipment, such as heat exchangers,will be substantial gamma radiation sources during reactor operation. These fields will disappear when the reactor is shutdown.

Fission products are not found in the moderator system as it contains no fuel. Most of the activation products listed in Table 1 will be found. These are beta/gamma emitters. They tend to plate out on the internal surfaces of pipes, valves and pumps in the system, and will present significant radiation fields for persons working in the vicinity of these components.

Moderator equipment surfaces will be contaminated with tritium. This presents

special problems for decontamination as the tritium is absorbed in the material and decontamination is not always easy. Tritiated water vapour can emanate into the air from surfaces for many weeks after a spill. This is especially true for concrete surfaces which is porous and has many small cracks. Detection of tritium on surfaces is carried out by a special smearing and counting the smear by liquid scintillation.

Specific Radiation Hazards Associated with the Heat Transport System

The principal hazards associated with the heat transport (HT) and its auxiliary systems are:

1) External gamma radiation from O-19 and N-16 produced in heavy water being circulated through the reactor core. Boilers pumps and piping are all sources during operation but this source decays rapidly at shutdown.
2) Tritium in air concentrations in any room where there is heat transport equipment. Much of the heat transport system heavy water is at high temperature and pressure so that any leaks from the system will produce high airborne concentrations in the rooms containing them.
3) Photo-neutrons from high energy gamma interactions with deuterium nuclei. This reaction is primarily related to the presence of N-16 in the heat transport water. The field is relatively minor compared to the N-16 field and decays with the N-16 on shutdown.
4) Radiation fields on equipment and piping surfaces due to deposition of fission and activation products. The deposited materials produce gamma fields external to the equipment. When the system is opened up internal surfaces are a source of beta/gamma radiation and contamination.
5) Surfaces contaminated with fission and activation products when dried out may give rise to airborne contamination in rooms containing heat transport equipment. Procedures to control contamination spread and intake of airborne activity by inhalation are necessary.

Tritiated Water

If the HTO concentration in the HT system is of the order of 0.04 TBq/kg (1 Ci/kg), tritiated water vapour concentrations in air, from opened HT System components, may be around 1000 DAC, . This level is not as serious as the level associated with moderator water, but it is still high enough to demand great care to prevent and reduce uptake by inhalation, and by direct exposure to liquid water (i.e. wetting of the skin). Air supplied plastic suits are essential for proper protection when working on opened components of this system.

Other Airborne Contaminants

The most common airborne contaminants are the short lived noble gases and their particulate daughters. Opened system components which have heavily

contaminated surfaces may generate longer lived airborne contamination in air especially if they are disturbed by welding, hammering, drilling etc..

Surface Contamination

Opened HT system components may be contaminated with both fission and activation products. Procedures to control the spread of contamination from work sites are necessary - establishment of rubber areas, monitoring and identification of contamination levels on equipment and bagging of contaminated components. Gloves of appropriate thickness are used to reduce beta radiation dose to the skin of the hands.

Specific Radiation Hazards Associated with Miscellaneous Reactor Systems

Annulus Gas System

The annular space between the calandria tube and the pressure tube is a region of intense neutron flux (see Figures 4). This space must contain a gas other than air, because air is about 0.8% of Argon-40. This produces Argon-41 when irradiated. Argon-41 is a beta/gamma emitter with a half-life of about 2.8 h. It will cause a significant increase in the radiation field in the vault area if air leaks into the annulus gas space.

The annulus gas space of Pickering 'A' was filled with nitrogen, now CO_2. When nitrogen absorbs neutrons it produces carbon-14, both as a particulate, and as carbon dioxide.

$$^{14}N_7 + {}^{1}n_0 = {}^{14}C_6 + {}^{1}p_1$$

The radiation hazard associated with carbon dioxide is minimal, but the particulate has detection and measurement problems which make lung dose estimation difficult. For this reason respiratory protection is recommended for any exposure to particulate carbon-14 .

Most reactors use carbon dioxide as an annulus gas. Oxygen-17 (present in carbon dioxide as a naturally occurring isotope of oxygen) can also absorb a neutron to make carbon-14 (an n:γ reaction), but the quantity produced is very small relative to that produced by nitrogen-14.

$$^{17}O_8 + {}^{1}n_0 = {}^{14}C_6 + {}^{4}He_2$$

Exposure to C-14 has been high during retubing of the Pickering A units in the past few years. Extensive precautions to prevent any intake by inhalation and ingestion is necessary as intake and dose determination is difficult.

Moderator Cover Gas, Heat Transport De-Gassing Systems and Light Water Zone Control Absorbers

These systems are similar to the annulus gas system in that the introduction of air will give rise to high levels of argon-41. Very high radiation fields (>500 rem /hour) have been measured at the surface of the Moderator cover gas holding tank when air from gas cylinders was put into the cover gas instead of helium.

Ventilation Systems and Driers

Closed cycle ventilation systems are provided on a number of rooms. The intended design function of such systems is to remove heavy water from the vault atmosphere (to minimize heavy water losses) and to remove activation and fission product activity. Reactor vault ventilation systems are of this self-contained, re-circulating type.

The vault dryers in addition to removing and recovering heavy water keep airborne tritium levels in the vault atmosphere low. At times of high heavy water leakage from end fittings the tritium concentrations in the vault can be high. Vault dryers may also collect particulate activity from the air. When the drying agent is being replaced, precautions should be taken to prevent the dusty drying agent becoming airborne. Air purifying dust masks or airline supplied hoods must be worn to prevent uptake by inhalation.

The filtration systems on the reactor vault circuits remove any fission and activation product airborne contamination present in the atmosphere. The filters may accumulate significant amounts of beta/gamma emitting nuclides that could give rise to an external gamma radiation hazard in the vicinity of the filter housing. Inside the filter housing, the beta radiation dose rate may be between one and two orders of magnitude higher than the gamma dose rate. The radioactivity collected by the filter is loose so precautions must be taken to contain the contamination when the filters are to be replaced.

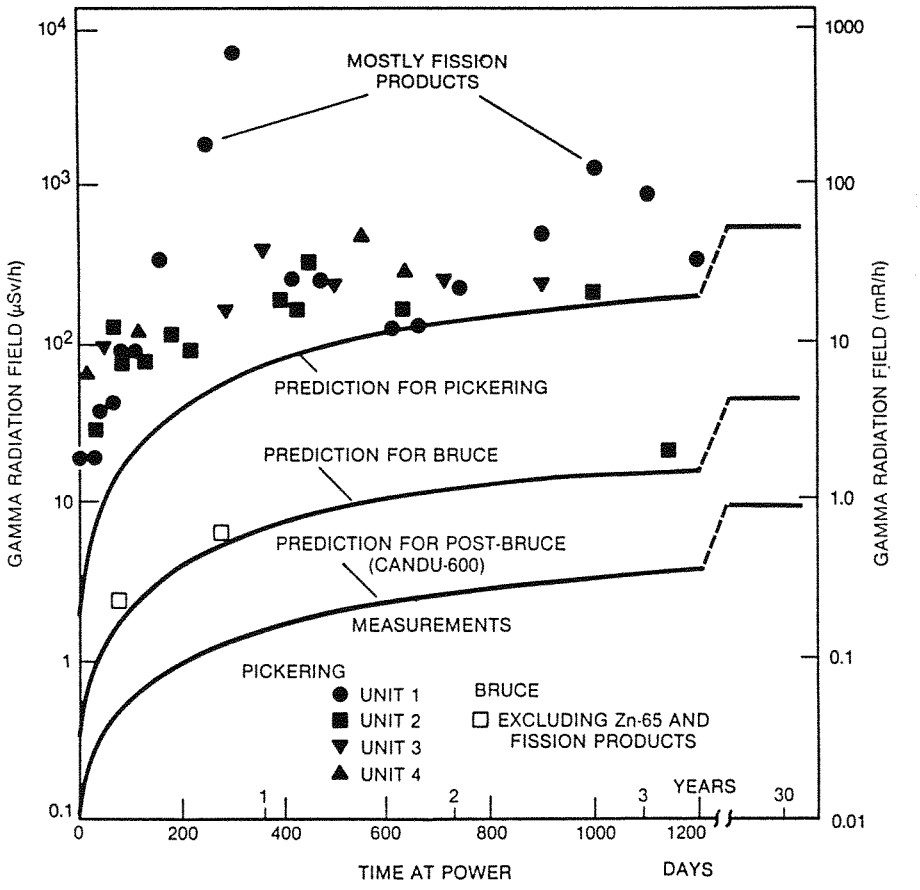
Application of the ALARA Principle

In the years since Douglas Point G.S. was commissioned (1968), considerable effort has been expended towards reducing the relatively high collective and individual doses received in the operation of the station. At Pickering G.S. after two years at power, the radiation fields in the vicinity of the steam generators, 24 hours after shut down, were about 1/10th of the levels experienced at Douglas Point after the same period at power. Figure 5 shows that the radiation levels predicted for post Bruce 'B' stations will be about 1/100th. of those Pickering levels.

These advances have been achieved by research and development in reactor water chemistry; by improvements in the performance of the fuel; by attention to the composition of reactor materials and by improvement in station equipment layout. Examples of what has been done are:

1. by maintaining a lithium concentration in the heat transport system water at about 1 mg/kg the movement of corrosion products around the heat transport system has been significantly reduced,
2. by maintaining a concentration of about 10 mL D_2/kg of D_2O the corrosion process has been greatly reduced,
3. removal of naturally occurring cobalt from reactor materials has eliminated a major source of gamma radiation. For example, the percentage of cobalt in reactor grade carbon steel is now less than 0.006.
4. the introduction of CAN-LUBE fuel (in which the interior surface of the Zircalloy fuel sheath is coated with graphite to enable the UO_2 fuel pellets to move with less friction) has resulted in a dramatic reduction in damaged fuel - from 1.23% in Pickering 'A' in 1972, to between 0.04 - 0.09% in current reactor practice. Improvements in the practices used for moving the fuel across the core to balance reactivity and achieve high 'burn-up' will further minimize these defects, and result in further reductions in the amount of fission products circulating in the heat transport system.

Figure 5
Steam Generator Radiation Field Predictions



Emission Sources and Derived Emission Limits for a CANDU Station

Table of Contents

Introduction

Although station design and operating practices are intended to contain radioactive materials within the station, small emissions do occur on a regular basis. A prime design and operating requirement is to ensure that in normal operation these emissions do not cause public radiation dose limits to be exceeded. To meet this requirement licensees must establish Derived Emission Limits (DEL). A DEL is "a limit for the emission of a single radionuclide, or radionuclide group, for airborne or liquid effluents, which if complied with will give a high degree of assurance that no member of the public will exceed the public dose limit due to the emission of that radionuclide". In establishing DEL the licensee must consider the pathways that any radioactive material could take to the population and establish emission limits based on the pathways which will give the highest dose to any segment of the population affected by the emission.

In Canada the method of establishing these DELs is currently based on a Canadian Standards Association document - Can.CSA.N288.1-M87, Guidelines for Calculating Derived Release Limits for Radioactive Materials in Airborne and Liquid Effluents for Normal Operation of Nuclear Facilities. This lesson is largely based on this document.

Public Dose Limits

Currently DEL are established by the utilities based on the existing AECB dose limits for the public. These are 5 mSv/year for whole body exposure with specific limits for organs such as the thyroid, lungs etc.. These regulations are currently under review to make them consistent with the most recent recommendations of the ICRP (ICRP 60). In future it is reasonably certain that DEL will be based on the limits recommended by the ICRP or limits very similar. This will not change the process of establishing DEL but it will of course affect the final value arrived at by the process.

For members of the public the ICRP recommended limits are:

- an effective dose of 1 mSv/year (100 mrem/year),
- a dose equivalent of 15 mSv/year (1.5 rem/year) to the lens of the eye,
- a dose equivalent to skin of 50 mSv/year (5 rem/year).

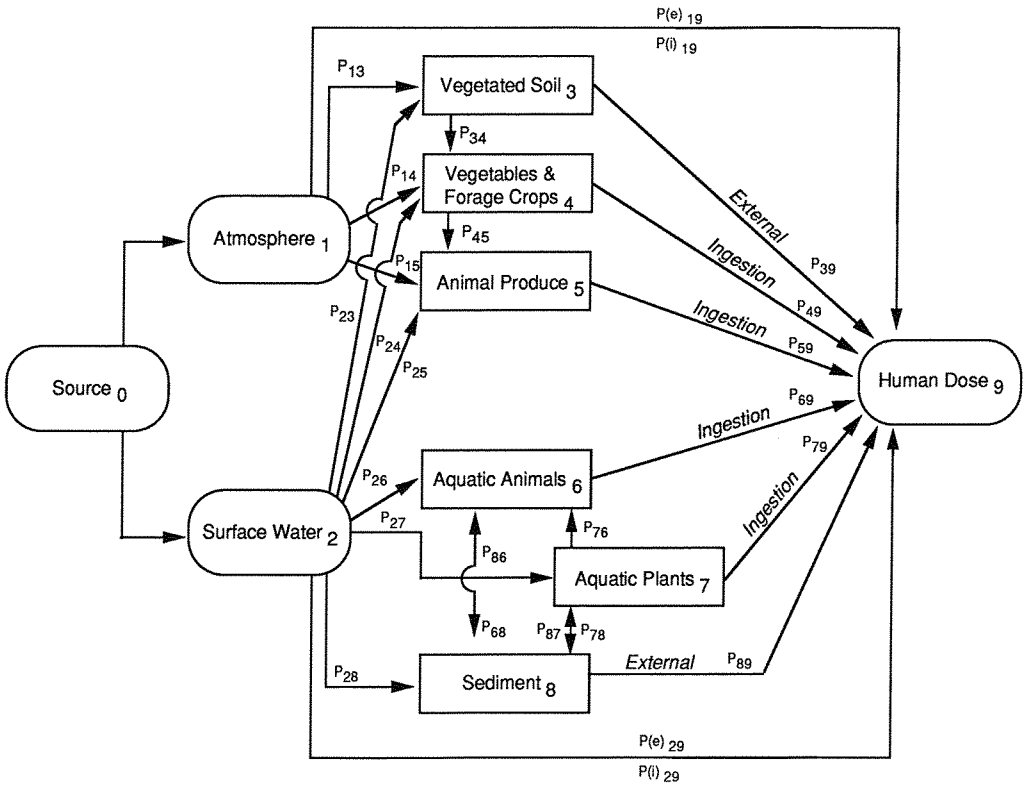
The effective dose must take into account contributions from external and internal exposure. Averaging is permitted over a 5 year period with a maximum in any year of 5 mSv. It is likely that an annual effective dose of 1 mSv/year will become the Canadian public dose limit.

Establishing a Derived Emissions Limit

The CSA standard provides a generic model of the process for establishing a DEL. This is shown in Figure 1. This model identifies all reasonable pathways for radioactive material released, to deliver a dose to population groups in the vicinity of a station. In the model the various compartments, and pathways between compartments, are identified. For example ($P_{1,9}$) is the path by which a member of the population is exposed to airborne tritium. The model identifies also External and Ingestion exposure paths. For example the ingestion of fish by the population is represented by ($P_{6,9}$). The paths $P(e)_{1,9}$ and $P(i)_{1,9}$ represent the external and internal exposure of members of the public resulting from radioactive materials in the atmosphere.

In establishing DEL the population is not considered to be uniform. The age, habits and activities of population groups have to be taken into account. For example in the United Kingdom when emission limits were being established for liquid discharges into the Irish Sea from the fuel reprocessing plant at Seascale, the 'critical group' was a small population in Wales who ate seaweed.

Figure 1:
General Model for Establishing DEL



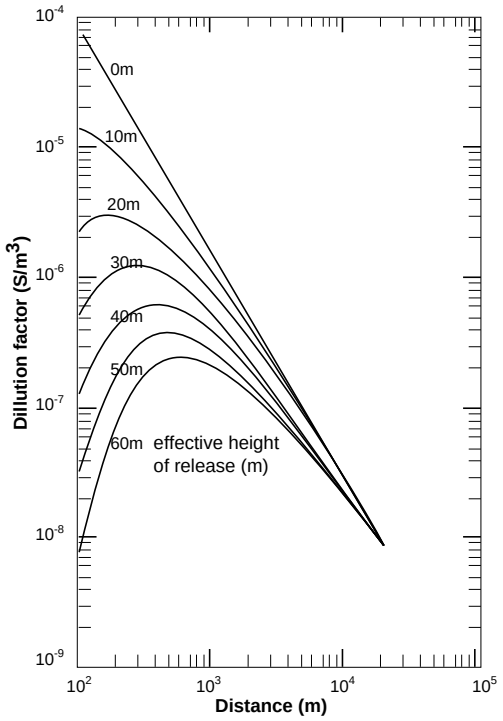
The general process for calculation of a DEL is as follows:

1. Identify the exposure pathways and critical organs,
2. Develop an expression for $X_9/X_{0(a)}$ and $X_9/X_{0(w}$), using appropriate transfer parameters (site specific values should be chosen over default values given in the standard if available),
   Where $X_{o(a})$ = Source Release Rate to atmosphere
   $X_{o(a)}$ = Source Release Rate to water
   $X_{o(a)}$ = Radiation Dose at point of release
3. Calculate the DEL based on 1 mSv/a, separately, for adults and children,
4. Calculate the DEL for radionuclides where the ratio of skin dose/effective dose > 50. (This applies to the immersion dose in a cloud of noble gases. If the skin dose is less than 50 times the effective dose then the effective dose is the limiting dose and is used as the basis of the DEL. If the skin dose is > 50 times the effective dose this is the limiting dose and is used as the basis of the DEL. An example where this would apply is for Kr-85 where this ratio is 140.

For airborne releases the path from source to atmosphere, the dispersal path, is of particular interest. There is a great deal of information about dispersal from a stack or ventilation outlet. The origins of this type of work extends to World War I when knowledge was needed about the dispersal of poison gas. The dispersal at any time is a function of the weather and may vary by several orders of magnitude. For DEL the interest is in an average long term dispersal factor K_a. Values of K_a are available and are provided in the CSA document.

Figure 2

Long Term Average Dilution Factors for Typical Canadian Weather and Uniform Wind Rose



Dispersal factor for various effective stack heights are shown in Figure 2 which is reproduced from the CSA standard. Typically for a nuclear station an effective stack height of 20 m and a distance from the emission point of 2 km would give a long term dispersal factor:

$K_a = 6 \times 10^{-7}$ s/m^3

If the average annual emission rate from the ventilation system or stack of the station, Q Bq/sec, is multiplied by Ka then the airborne concentration is obtained in Bq/m^3.

Principal Airborne Pathways - Noble Gases, Tritium, I-131

All reasonable pathways should be considered and theoretically the contributions from all pathways should be summed but in practice simpler models may be derived from this more complex generic model because the dose received via certain pathways swamps the contributions from all others.

Figure 3a
Model for Airborne Emission - Noble Gases

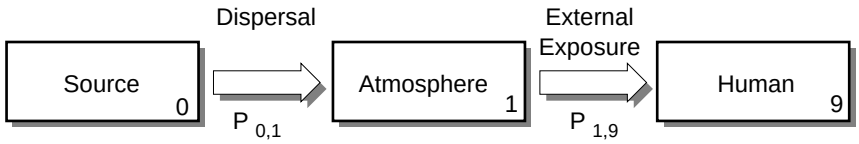
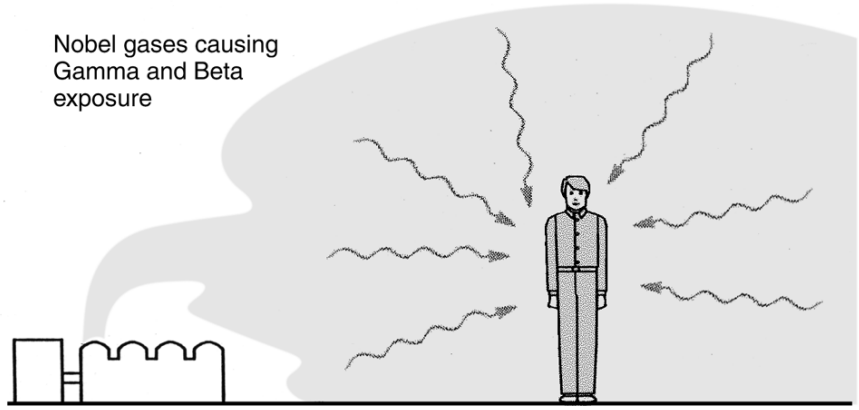


Figure 3b
Radioactive Noble Gases Exposure Route



DEL for Airborne Noble Gases

Figures 3A and 3B show the simplified model and a graphical representation of the model for the emission of noble gases from a station. The model illustrates a typical noble gas mixture being released from the station at a rate Q (Bq/s), being converted to a concentration of noble gas in the atmosphere (Bq/m^3) at the boundary of the station. This noble gas plume or cloud irradiates a person

standing in it and delivers an effective dose rate (mSv/year). The calculation is based on a person standing in a hemispherical cloud of large radius. Consideration has to be given for the time spent in doors as a house provides some shielding. It is possible that there may be a critical group who spend a large part of their time outside, so allowance would have to be made for this. Corrections may also have to be made for the fact that the plume is not a large volume hemisphere. Based on this model for noble gases the emission rate for an annual dose of 1 mSv may be derived. This is the DEL.

DEL for Airborne Tritium Oxide

The simplified model and graphical representation of the model for an airborne tritium oxide emission are shown in Figures 4A and 4B. Effective dose in this case is delivered by inhalation and skin absorption. For an annual effective dose of 1mSv the emission rate, that is the DEL, may be derived.

Figure 4a
Model for Airborne Emission Tritium (Oxide)

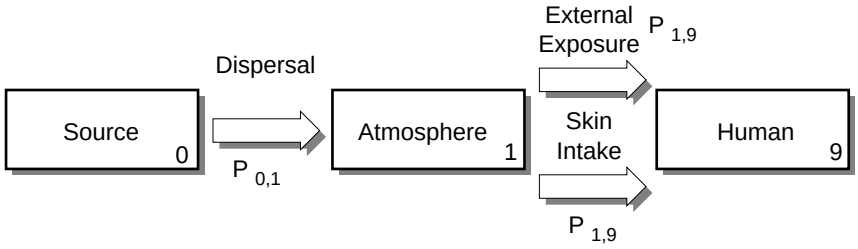
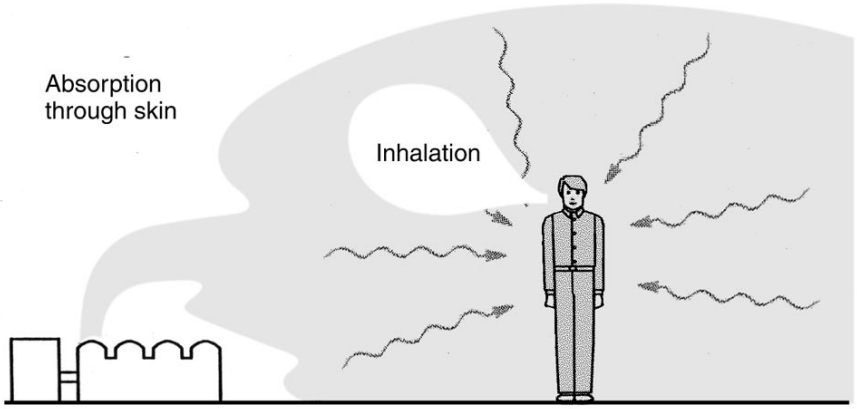


Figure 4b
Radioactive Tritium Exposure Route



DEL for Airborne I-131

The model and graphical representation of the model are shown in Figures 5A and 5B respectively. For this radionuclide 1 year old children consuming locally produced milk are the critical group. The pathway is form the source Q,(Bq/s) to an air concentration above the pasture land,(Bq/m^3). The airborne I-131 deposits or settles on the grass of the pasture at a known rate contaminating the grass,(Bq/m^2). The cow consumes the grass at a certain rate resulting in the

transfer of I-131 to milk, (Bq/m^3) and finally the child consumes a daily volume of milk and retains a quantity of I-131 in the thyroid. The emission rate which gives an annual dose equal to the public dose limit for the thyroid of a child is the DEL.

Figure 5a
Model for Airborne Emission - Radioiodine

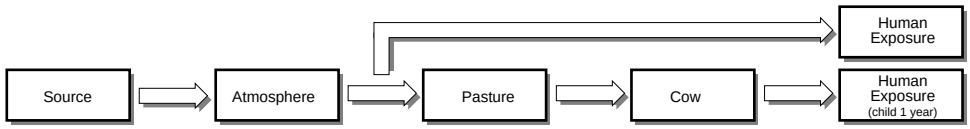
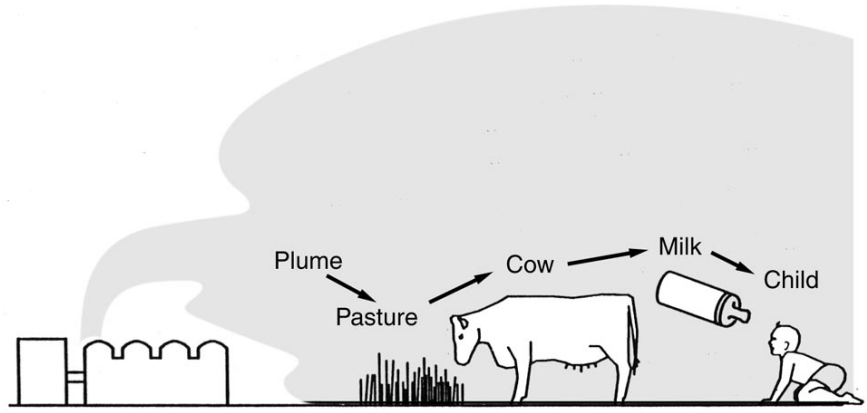


Figure 5b
Radioactive Iodine Exposure Route



DEL calculations should make conservative but not absurd assumptions. For example if dairy cattle are not on pasture for six months of the year then allowance should be made for this. Usually an airborne DEL is expressed as a limit per week to give some flexibility in making emissions but also to ensure that large emissions are not made when, because of the weather, poor dilution conditions exist.

These descriptions provide an understanding of the process of establishing an airborne DEL. A specific worked example of the establishment of a DEL for tritium is given in Appendix A.

Principal Aquatic Pathways

The principal aquatic pathways are shown in model and graphic form in figures 6A and 6B.

Figure 6a

Model for Aquatic Release E.G. Cs-137

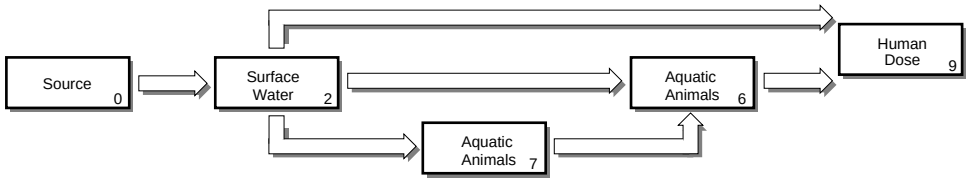


Figure 6b

Food Chain In Water

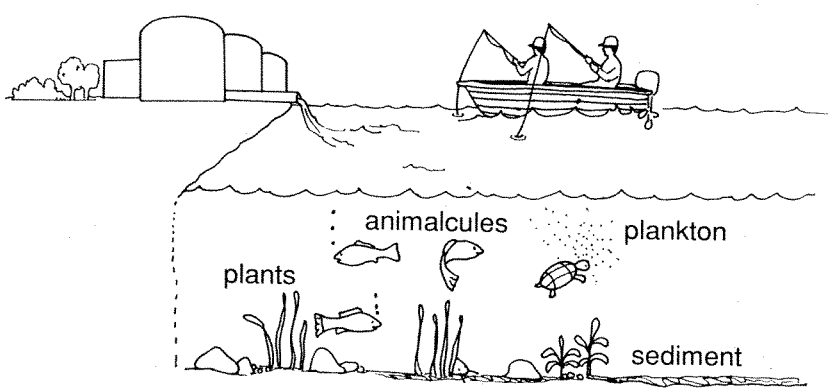
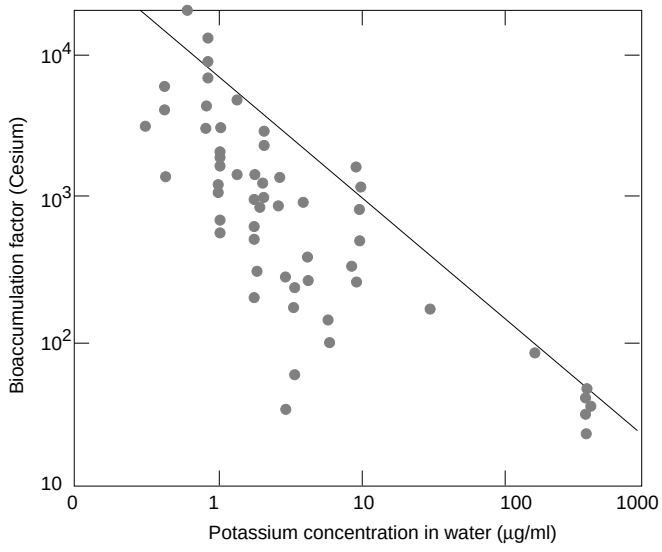


Figure 7

Bioaccumulation Factor for Cesium in Fresh Water Fish Flesh



Establishing DEL for aquatic pathways is a bit more complex than for airborne pathways. First there is no general model for dispersal of the effluent. Dispersion is quite site specific and depends on the body of water receiving the effluent. For rivers the dispersal is usually taken as the ratio of the flow of the river to the effluent flow. For station discharging into a lake by way of station condenser cooling water flow, use can be made of the dilution provided by this flow. Further dilution in the lake is small, typically less than 10. A site study is recommended to obtain dilution factors. A further complicating factor is that for some radionuclides for which the pathway to the critical group is via fish, the concentrating (bio-accumulation) in fish of a radionuclide such as cesium is dependent on the concentration of the natural potassium in water. If the potassium concentration is high the concentrating factor is low.

Operational Emission Targets

In addition to compliance with public dose limits by maintaining emissions within the DEL values, utilities have established operational targets. Operational targets apply to both public dose and station emissions.

Operating targets were established originally as a method of compliance with the ICRP recommendations to keep doses As Low As Reasonably Achievable (ALARA). At the time of their establishment in the early 1970s there was fierce controversy in North America about nuclear power and the effects of radioactive emissions. In Ontario Hydro the decision was made to keep public doses less than 1% of 5 mSv, the public dose limit at that time. Station design had been achieving these values and it was felt that any additional costs incurred by continuing to meet these low levels of dose would be worthwhile for the benefit derived in terms of public acceptance of nuclear power in Ontario. A rigorous optimization procedure was not applied to this decision, nevertheless it was an ALARA based decision. In practice, because annual doses of this level are difficult to measure and cannot be used as operational controls, targets were set for emissions for each radionuclide group for which a DEL had been established. These targets were that the station would operate at < 1% of the DEL. All utilities have established similar targets.

Despite aging and build-up of activities all Canadian nuclear stations have been able to achieve the DEL targets established. Figures 8, 9 and 10 show the annual emissions from Gentilly, Point Lepreau and Bruce 'A' respectively.

DEL have in the past been established on the basis of a public dose limit of 5 mSv/y. The public dose limit based on the proposals in Bill C-122 are likely to be 1 mSv/y. Utilities have not yet established new DEL based on this value. Theoretically this could be as simple a task as multiplying the current values by 0.2. However some of the assumptions made in DEL calculation may be

considered as conservative and it is possible, with the drastically lower public dose limits proposed, that the DEL may be recalculated.

Figure 8

Gentilly Annual Emissions

GENTILLY II EMISSION DATA, 1983 - 1991, % DEL

	AIRBORNE			LIQUID	
Year	**Tritium**	**Noble Gases**	**Iodine-131**	**Tritium**	**Total Beta**
1983	.001	.08	$<10^{-3}$	$<10^{-3}$	.006
1984	.014	.03	$<10^{-3}$	.031	.077
1985	.015	.15	$<4 \times 10^{-3}$	.017	.089
1986	.042	.06	$<10^{-3}$	.027	.028
1987	.036	.06	$<10^{-3}$	.050	.135
1988	.033	.10	$<10^{-3}$	.036	.033
1989	.040	.09	$<10^{-3}$	.030	.013
1990	.065	.07	$<10^{-3}$	.102	.048
1991	.095	.07	$<10^{-3}$	.156	.032

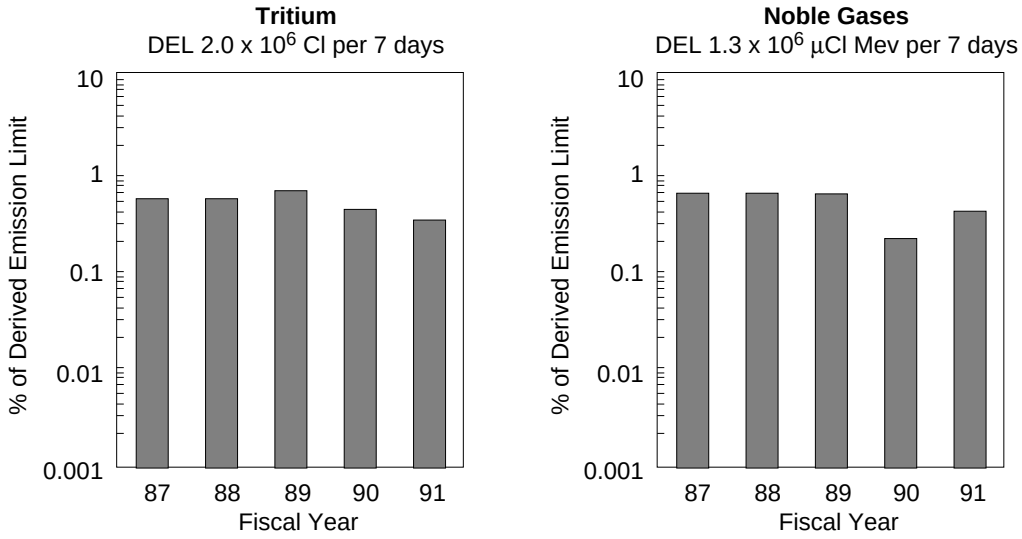
Figure 9

Point Lepreau Annual Emissions

EMISSION DATA FOR 1983 TO 1990

Year	**Gaseous Emissions**		**Liquid Emissions**		**Total Annual**
	% DEL	**uSv**	**% DEL**	**uSv**	**Dose, uSv**
1983	0.016	0.16	0.008	0.08	0.24
1984	0.022	0.22	0.003	0.03	0.25
1985	0.036	0.36	0.005	0.05	0.41
1986	0.075	0.75	0.003	0.03	0.78
1987	0.070	0.70	0.001	0.01	0.71
1988	0.080	0.80	0.005	0.05	0.85
1989	0.080	0.80	0.008	0.08	0.88
1990	0.095	0.95	0.017	0.17	1.12

Figure 10
Bruce NGS 'A Annual Emissions



Appendix A: Example Calculation of the DEL for Argon-41

Exposure Pathway

This simple model of Figure 3A shows the dose pathway.

Ar-41 is a noble gas so the first step is to determine if the skin/gamma dose rate is greater of less than 50 for Ar-41. Table D7 of the Canadian Standard 288.1 gives radionuclides with a high ratio of skin dose to effective dose. Ar-41 does not have a high ratio so the skin dose may be neglected in the calculation.

Dispersion Factor

Let Q Bq/sec be the emission rate which gives an annual dose equal to the limit of 0.1 mSv/a.

$P_{0,1}$ is atmospheric dispersal factor which converts the radioactive material emission rate to radioactive material airborne concentration. The default value for $P_{0,1}$ is obtained from Figure 2. Assuming a constant wind rose, a stack height of 40m and a distance of 1 km then

$P_{0,1} = 5 \times 10^{-7}$ sec/m^3 and

$C = Q \times 5 \times 10^{-7}$ Bq/m^3

where C is the concentration in Bq/m^3 which gives an annual dose of 0.1 mSv/a.

Dose Rate from an Airborne Concentration

From figure 4A, $P(e)_{1,9}$ is the transfer factor which converts the concentration of noble gas into dose rate.

$P(e)_{1,9} = [F_u + (1-F_u)S] \times DCF_a$

where F_u is the time spent outdoors (assume 30%),

S is a factor for building shielding when indoors
The default value of 0.9 for gamma radiation will be used here.

DCF_a is the conversion factor from airborne concentration to unshielded effective dose rate in a semi-infinite cloud of radioactive material. From Table 3 of N288.1 this has a value of 1.7×10^{-6} $Sv.a^{-1}.Bq^{-1}.m^3$

$P(e)_{1,9} = (0.3 + 0.7 \times 0.9) \times 1.7 \times 10^{-6}$ $Sv.a^{-1}.Bq^{-1}.m^3$
$= 1.58 \times 10^{-6}$ $Sv.a^{-1}.Bq^{-1}.m^3$

An emission rate of Q Bq/s will give a dose rate of D Sv/a given by:

$D = Q \times 5 \times 10^{-7} \times 1.58 \times 10^{-6}$ Sv/a
$= Q \times 7.9 \times 10^{-13}$ Sv/a

Weekly DEL

Now Q was originally selected to give the annual dose rate of 1 mSv/a (10^{-3} Sv/a). Therefore

$Q \times 7.9 \times 10^{-13} = 10^{-3}$
$Q = 1.27 \times 10^{9}$ Bq/s

Airborne DEL are usually expressed as weekly limits to provide flexibility in operation and to limit large emissions occurring when dispersal conditions are poor so:

Weekly DEL $= 1.27 \times 10^{9} \times 6.048 \times 10^{5}$ $Bq.s^{-1}.s.wk^{-1}$
$= 7.9 \times 10^{14}$ Bq/wk (2.1×10^{4} Ci/wk)

Potential Consequences of an Accident and Description of Important Reactor Accidents

Table of Contents

Introduction

Reactors, especially power reactors, have characteristics which make it necessary to give serious attention to the prevention of
malfunctions of the reactor systems and accidental releases of radioactivity. There are three principal factors associated with power reactors and their potential for serious accidents. It is interesting to compare these with power production by a coal fired station.

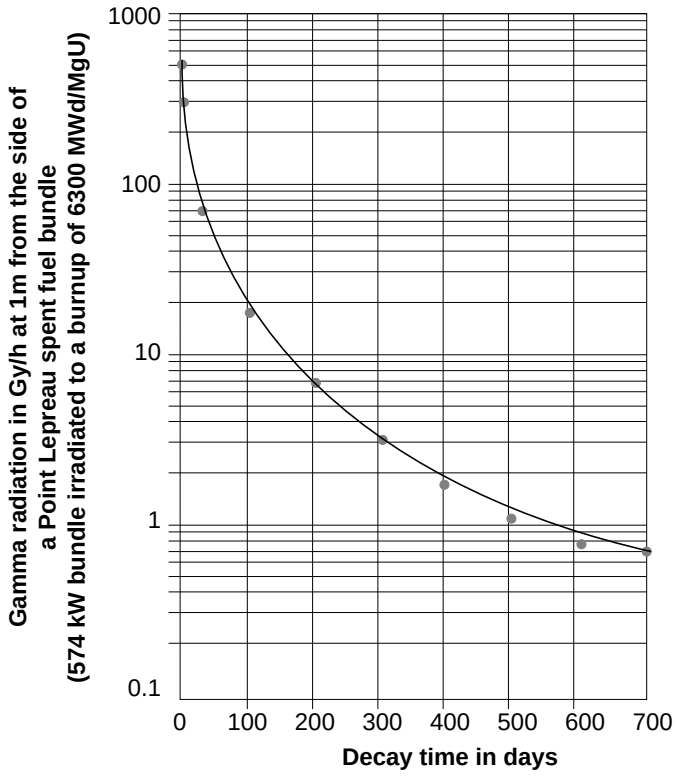
1. A reactor has the capability of going in a very short time to very high power levels - many times greater than the power at which it is normally expected to operate. In the Chernobyl reactor accident which we will be reviewing in detail the reactor went from about 20% full power to 100 times full power in 7 seconds. In a coal fired station the power rating is dependent on how fast coal can be fed into the furnace. This is clearly limited.

2. In a coal station when the combustion process is shutdown there is very little residual heat and no further heat production. In a reactor this is not the situation. Fission products produced in the fuel are radioactive. After reactor shutdown the fission power essentially becomes zero but the fission products decay and continue to produce heat. This may initially be as much as 7% of full power rating and will continue to be significant for a long time. Figure 1 shows how the gamma dose rate (approximately proportional to the heat production) from a Point Lepreau fuel bundle decreases with time. A bundle in a Darlington reactor at high power could be producing about 5 kW of heat just after reactor shutdown.

3. The waste products in a nuclear reactor are highly radioactive and if not contained may present a very serious hazard. Coal station waste does have some minor toxic concerns but fly-ash one of the wastes has been used for cement block and road construction.

These characteristics are the reason you have never heard about a major coal fired power station accident and why there are reactor accidents which have had wide publicity. These are also the reasons that there is such a powerful regulatory process in place to ensure that a satisfactory level of public and occupational safety in reactor operation is achieved.

Figure 1
The Decrease in Fission Product Activity with Time



Fission Product Inventory & Importance in Accidents

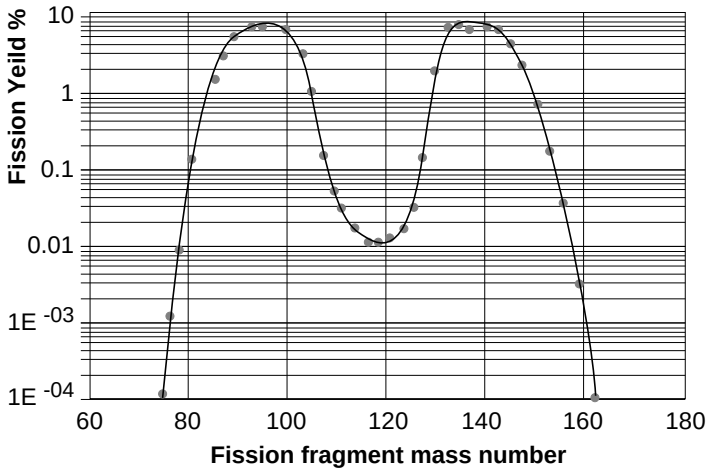
Fission Product Yield

As soon as a fuel element is placed into an operating reactor, fission of U-235 begins and the fuel element begins to accumulate fission products. Fig.2 shows Fission Product Yield -v- Mass Number. It shows that when the U-235 atom splits into fragments, the products fall into two main groups; a light group with mass number, in the range 88 - 100, and a heavy group with mass number in the range 132 - 144. The products in both these groups have 'fission yields' of the order of 5 to 6 percent. Nuclides with mass numbers on either side of these main groups have lower fission yields. There are approximately 200 known fission products.

In a reactor running at full power, the fission product content of a fuel element increases with time. Fission products which have a short half life quickly reach an equilibrium concentration where the production by fission equals the loss by radioactive decay. It takes about 5 half-lives for a fission product to reach its

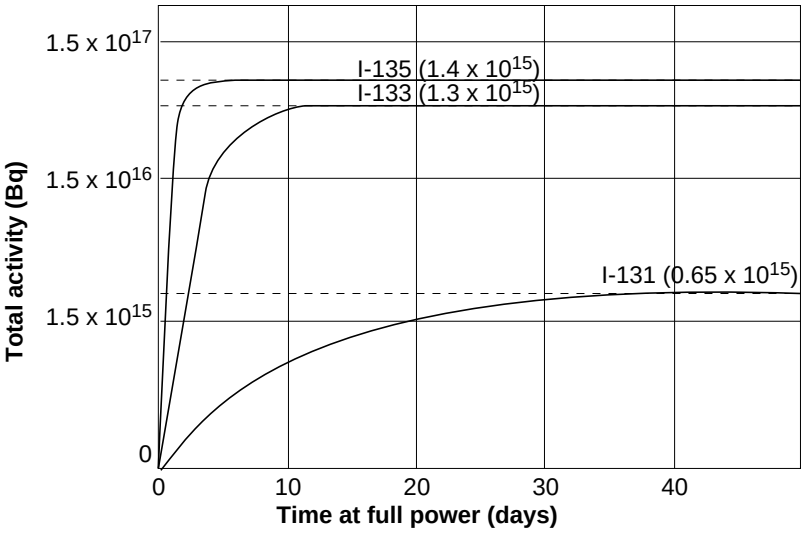
equilibrium concentration. Iodine-131 which has a half life of about 8 days will build up to it equilibrium concentration in 40 days. Fig. 3 shows the build-up of some iodine nuclides in a typical fuel element.

Figure 2
Fission Fragment Yield for U-235



Some radionuclides, such as strontium-90 and cesium-137, which have half-lives of the order of 30 years, will not reach an equilibrium level because the average fuel element stays in the reactor for only about 1 year. Both Sr-90 and Cs-137 have mass numbers which puts them in the vicinity of peak fission yields of about 6.0%.

Figure 3
Build Up of Radio-Iodines Activity Within a Pickering 'A' Fuel Bundle



Volatile Fission Product Inventory - Noble Gases

In an accident the fission products of greatest concern are those that are most readily released, namely the volatile fission products. The most volatile fission products (FP) are the Noble Gases (NG) and Iodines. Table 1 shows the common FP NG's that will be present in a fuel bundle and gives an estimate of the number of curies present immediately after shut-down.

The volatile FP's account for about 25% of the FP inventory in the fuel element. Depending upon the temperature of the fuel element, up to 100% of the FP inventory can be released.

Table 1

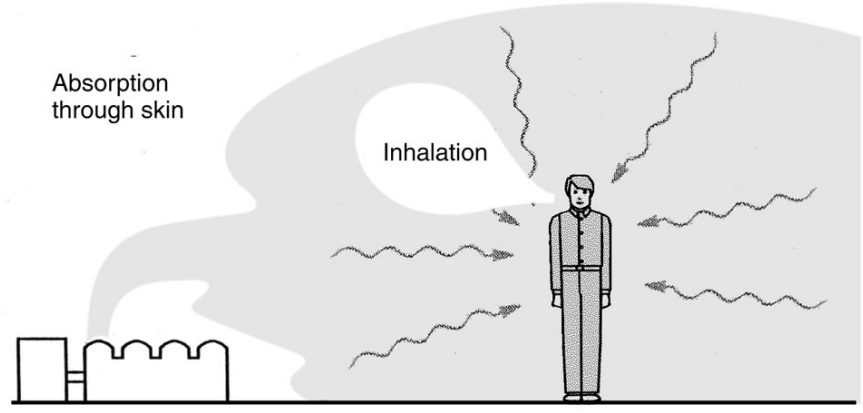
Fission Product Noble Gases in a Fuel Bundle (Pickering A)

Nuclide	Half-life	Equil.Activity TBq
Kr - 85m	4.5 h	1.9×10^2 (5.2×10^3 Ci)
Kr - 88	2.8 h	6.3×10^2 (1.7×10^3 Ci)
Xe - 133	2.3 d	1.3×10^3 (3.6×10^3 Ci)
Xe - 135	9.2 h	8.9×10^1 (2.4×10^3 Ci)
Xe - 138	17 min	1.4×10^3 (3.8×10^4 Ci)

Noble gases are chemically inert. They are released relatively quickly from a damaged fuel element, to form an intense beta-gamma radiation cloud in the immediate vicinity of the element. If all the noble gases in a fuel element were released shortly after shut down into a confined space such as the reactor vault the beta radiation dose to skin, could result in life threatening skin damage. Noble gases released from the station form a discharge plume which is a source of external whole body radiation dose primarily from the gamma emissions of the noble gases (Figure 4). As the cloud moves away from the station and disperses the dose rate will decrease. The decrease depends on the atmospheric dispersal conditions.

Figure 4

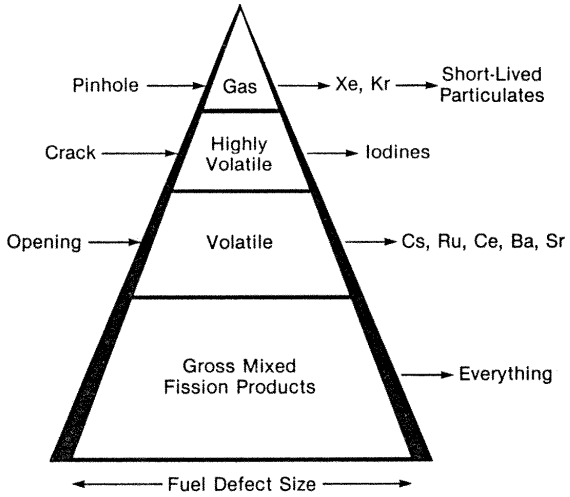
Noble Gases Exposure Pathway



Volatile Fission Products - Radio-Iodines

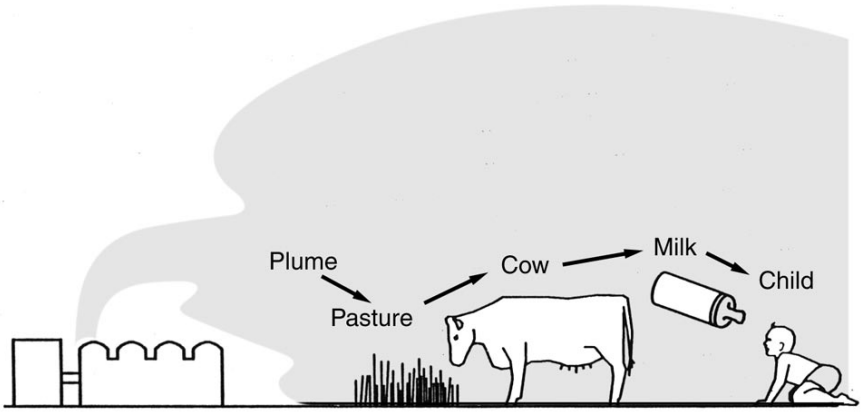
Radioiodines are volatile at fuel operating temperatures - but they are also very reactive. They will react with some components in the fuel element sheath, with metallic components in an over heated core, with aerosols present in the air, and with organic compounds present in the air. They also readily deposit on surfaces, including vegetation. The result is that the radioiodines are not released to the same extent as the NG's - perhaps up to 20% of the fuel element inventory may be expected to go to the environment in a serious accident. Radioiodines may also react with, or dissolve in water or steam present in the local environment, and this may further reduce the release.

Figure 5
Release of Fission Products



If radioiodines escape from the reactor containment to the external environment they will be deposited on vegetation (deposition velocity = 2×10^{-2} $m.s^{-1}$). The critical route to human uptake is from forage, through cows to milk, and consumption by infants. Irradiation of the thyroid is the limiting factor

Figure 5a
Radio-Iodine Exposure Pathway



Other Fission Products of Importance in an Accident

Other radio-nuclides may be released in an accident, but this is very dependent on the nature of the accident. Only in the Windscale accident and Chernobyl have airborne radionuclides other than the noble gases been released in any significant quantities. In both these cases the reactor was a graphite moderator reactor, the fuel became overheated and a fire in the reactor core resulted. In such a situation semi-volatile and low volatility fission products may be released. This is partly due to their volatility and partly to physical processes related to the combustion. Table 2 shows examples of the volatile and semi-volatile fission products released in the Chernobyl accident

Table 2

Volatile and Semi-Volatile Radionuclides Released in the Chernobyl Accident

Radionuclide	Half-Life	Core Inventory (Bq)	% Released
Kr-85	3.9×10^{3}	3.3×10^{16}	100
Xe-133	5.3	1.7×10^{18}	100
I-131	8.1	1.3×10^{18}	20
Te-132	3.3	3.2×10^{17}	15
Cs-134	750.0	1.9×10^{17}	10
Cs-137	1.1×10^{4}	2.9×10^{17}	13
Ru-106	368.0	2.0×10^{18}	2.9
Ba-140	12.8	2.9×10^{18}	5.6
Ce-144	284.0	3.2×10^{18}	2.8
Sr-90	1.0×10^{4}	2.0×10^{14}	4.0
Pu-239	8.9×10^{6}	8.5×10^{14}	3.0

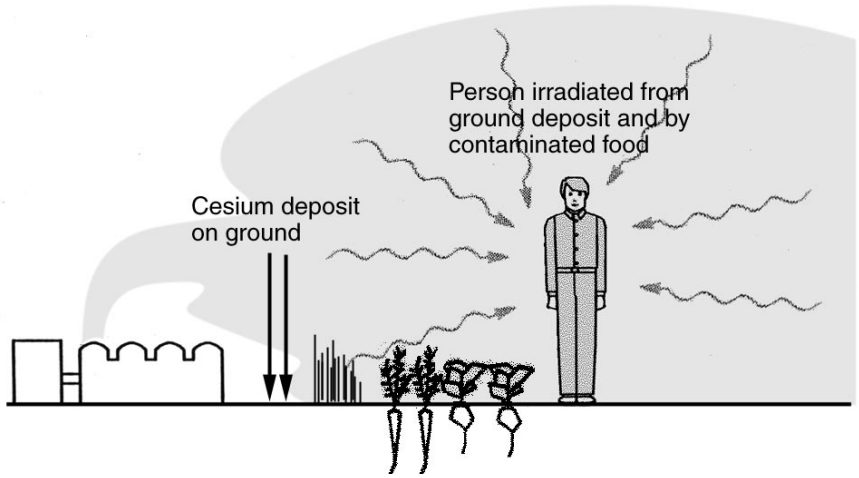


Figure 6

Cesium 137 Exposure Pathway

In a short time after an accident all the deposited radionuclides may contribute to individual and population dose because of their resultant external gamma

radiation field and also because they will enter the body when contaminated foodstuffs such as locally grown vegetables are consumed. Some may cause local problems, e.g. the plutonium isotopes are likely to be deposited within a short distance of the plant and may present an airborne inhalation hazard if resuspended. The plutonium isotopes have very low Annual Limits of Intake. Because of its half life, chemical properties and behaviour in the biosphere Cs-137 is the semi-volatile fission product that will be the source of the highest individual and population dose in the long term. It has a long half-life, and it is soluble. The fraction incorporated into body tissues (of that ingested) is close to 1. The chemistry of caesium is similar to that of potassium, so it is distributed into most body tissues. It is a whole body irradiator. When deposited on ground (vegetation) it is a source of external radiation. An extended plane source of 1 $kBq.m^{-2}$ will give a gamma radiation field of 1.6 $\mu Sv.h^{-1.}$ Contamination on vegetation eaten by animals is transferred to meat and dairy produce, so that many foodstuffs are sources of ingestion, (Figure 6).

Cesium on vegetation that is not consumed will ultimately go to soil. Uptake by vegetation from soil is affected by the mineral (e.g.potassium) content of the soil. The cesium will slowly migrate into deeper soil, with a corresponding decrease in the external radiation dose rate and the uptake by vegetation and animals. It will take several decades after its deposition for the total radiation dose to return to background levels.

Reactor Accidents

General Effects

Any abnormal incident which occurs in a reactor receives public scrutiny and there is intense public and media interest in reactor accidents. The effects of reactor accidents are wide ranging and may include:

1. Severe economic loss to the reactor owner or operator. The damage to the reactor, or the loss of the licence to operate the reactor pending correction of the initiating failures, results in heavy direct financial loss. If the reactor is significantly damaged the cost of clean-up and repair can run into billions of dollars.

2. Severe economic losses to the country or state in which the reactor is situated. In the case of severe accidents bordering and even distant states may also suffer. This is because of the disruption of normal everyday activities. Release of fission products to the environment may require evacuation and relocation of large populations. Agricultural land may become unusable. Foodstuffs may have to be destroyed. Water supplies may become contaminated requiring new uncontaminated sources to be found. Local industries may be lost. The cost to society can be billions of dollars.

3. In addition to any immediate health effects in the reactor work force, long term health effects are expected in both occupational and public populations from the exposure as a result of the accident. In an accident which results in a major release of radioactivity these long term health effects may include premature deaths from cancer and genetic defects in the offspring of exposed persons. The magnitude of this is not known precisely but based on the recent risk estimates of the ICRP there will be one death per 3×10^{-2} person-Sv of collective dose.

4. There is evidence that stress and psychological effects in populations affected by the accident are serious. These effects may arise from being evacuated, suffering from food shortages, lack of housing and fear of radiation.

Reactor Accidents of Interest

Reactor accidents (or incidents) have occurred at a low frequency since the first reactors were built in the late 40's and early 50's. In general they occur for reasons which were not considered (or even thought about) during either the initial concept development or the design and construction phases of the reactor. Lessons have been learned from all the reactor accidents which have occurred. Some of the more interesting or serious are discussed briefly below. The two accidents which caused serious international concern and alarm - the Three Mile Island reactor accident in the United States and the accident to unit 3 at the Chernobyl site in the former Soviet Union are discussed later in greater detail.

Windscale - United Kingdom

This reactor which was low power compared to today's power reactors was built to produce plutonium for the British weapons program. It was graphite moderated with uranium metal fuel. The accident was due to a peculiarity of graphite. Under irradiation, energy is stored in the graphite. This energy must be released by raising the temperature of the graphite. The amount of energy stored was underestimated and during a planned energy release (in which the temperature of the reactor was raised to trigger the energy release in a controlled manner) the graphite ignited. The fire in the graphite moderator spread to the fuel with resultant release of noble gases and iodines. The reactor was flooded with water to douse the fire. Damage to the reactor resulted in its total shutdown. This accident identified the importance of the radioiodines in reactor accidents as widespread contamination by I-131 of land in the North East of England resulted (See Figure 7) . Consumption of milk, from dairy herds was banned over a wide area.

NRX Chalk River - Canada

The NRX reactor is a 30 Mw heavy water moderated research reactor. The fuel is uranium metal clad in aluminum in vertical light water cooled tubes. The accident occurred in December 1952. The reactor was in the shutdown state and experiments to compare the reactivity of fresh fuel rods with rods of high burn-

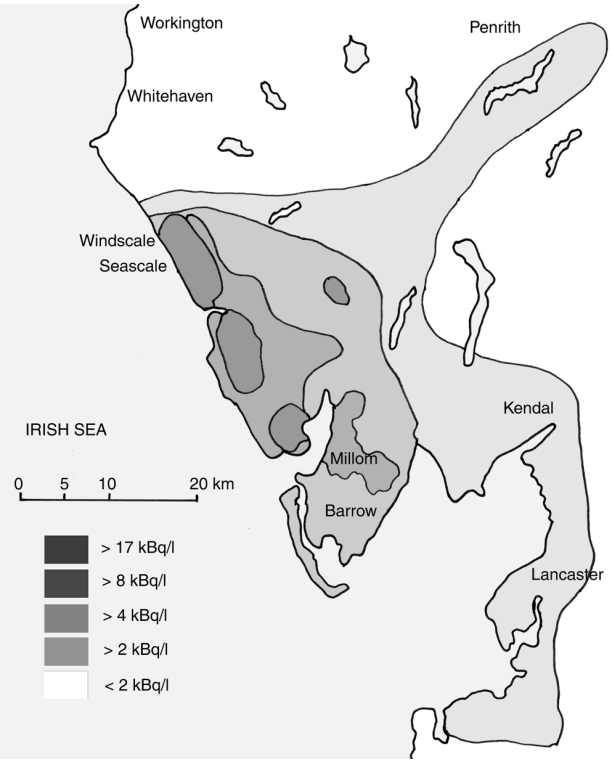


Figure 7
Geographical Distribution of Different Iodine-131 Contents in Milk after the Windscale Accident (1957)

up were underway when a power excursion occurred because of a complex occurrence of mechanical defects in the shut-off rod system and poor operating practices. An operator in the basement opened by mistake valves which caused three or more shut-off rods to withdraw from the reactor. The shift supervisor went to the basement, recognized the operator's mistake and rectified the valving. Unfortunately, some of the shut-off rods did not drop back into the core because of some unexplained mechanical fault. The shift supervisor then gave orders by telephone to the control room to continue their program and remove shut-off rods in a controlled manner. His instructions on rod bank removal were wrong and power increased rapidly reaching 90 Mw in about 50 seconds. The power excursion caused melting of some of the uranium fuel rods and rupturing of cooling water tubes. The excursion was terminated by dumping of the moderator and ingress of light water. The final outcome was that the reactor calandria and core were severely damaged and a large volume of water containing 3.7×10^{14} Bq (10^4 Ci) of long lived fission products was released to the basement. A minimum flow of cooling water was maintained because of concern that the high burn-up uranium in the basement needed cooling. Eventually 10^6 gallons of water containing the released fission products was pumped to storage tanks and then to a lagoon constructed for the purpose. This activity eventually drained into the ground.

Canadian reactor designers and operators learned many important lessons from this accident and established important reactor safety principles in effect today.

Fermi 1 - Sodium Cooled Fast Reactor - United States

This reactor was an experimental liquid sodium cooled fast breeder reactor. The accident resulted when a thin stainless steel plate, part of the thermal shield, came loose and was picked up by the sodium coolant. Subsequently it lodged over the inlet to several fuel channels. Loss of coolant flow through these channels resulted in partial core meltdown with heavy irreparable damage to the reactor.

Browns Ferry - United States

In this accident which occurred at the Tennessee Valley Authority plant at Brown's Ferry, technicians were looking for air flow in a duct carrying reactor control cables using the flame of a candle to detect the air flow. The open flame ignited organic insulating material inside the duct. It proved to be difficult to extinguish the flames which caused serious damage to the cables and some difficulty in shutting the reactor down. The resultant inquiry and repairs, plus redesign of the cabling system, resulted in several years downtime before the reactor was re-licensed for power production. An important lesson from Browns Ferry was the need for careful consideration to be given to fireproofing and distribution of cabling so that the probability of a fire disabling safety systems totally is low.

There have been a number of other accidents. Most of these have had similar, unexpected initiation sequences - a series of one or more events which were unusual, and totally unconsidered in the accident analysis.

The Three Mile Island Accident

Accident Description

The Three mile Island Generating Station has two Babcock and Wilcox Pressurised Water Reactors (PWR's). These reactors are similar to the design shown in Fig.8.

They consist of a pressure vessel with 8 inch thick steel walls. It contains the reactor core which consists of many hundreds of individual fuel elements, suspended in a support lattice. The lattice controls the flow of cooling water (ordinary light water) through the individual fuel bundles in order to ensure that each element gets its proper share of cooling. The fuel elements contain enriched (about 2% U-235) uranium dioxide pellets .The moderator is the same water as used for cooling - ordinary light water. The entire pressure vessel is filled with water with the core being submerged under about 8 feet.

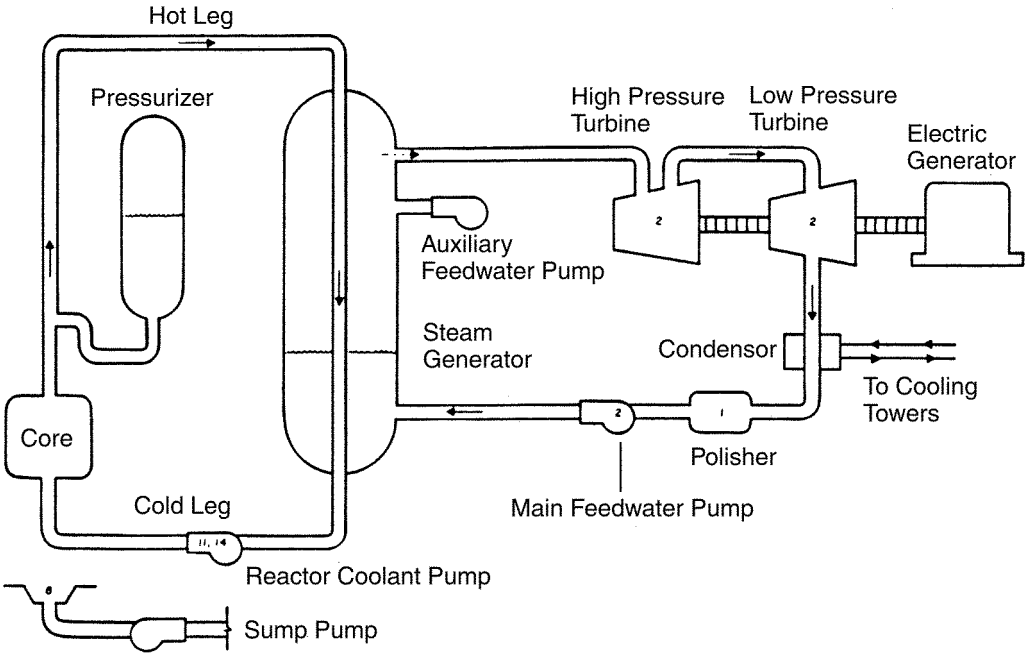


Figure 8
Schematic Diagram of the Reactor

The steam cycle is a dual system - a primary system in which water at 2150 psig. and 570ºC is pumped through the reactor (to remove the heat from the core), then to a steam drum where the heat is given up to a secondary steam circuit which generates the steam to drive the turbine. Water in the primary circuit is completely separate from that in the secondary circuit.

The accident scenario is complex. It began two days prior to the accident, during a routine shutdown. Valves, known as the twelves, were closed to allow a routine test of the auxiliary steam feed water pumps (i.e.reserve pumps in the secondary steam system). These valves were not re-opened. One of the lights on the control panel which showed that these valves were closed was obscured by a label relating to activity being carried out on another system.

The next initiating event was a leakage of water from a secondary steam system feedwater IX column, into the instrument compressed air system, after the compressed air was used to release a resin blockage in the IX column. About 1 hour later, at 04.00 h on March 28th, 1979, the secondary system feedwater pumps stopped (i.e.automatically shut down due to water in the instrument air control mechanism). Some few seconds later the auxiliary pumps were switched on automatically. They ran, but could not pump water because the 'twelves' were closed. The control room operators did not notice the indicators which showed the closed valves.

Now, with no heat removal available to the primary water cooling system, the temperature and pressure in the reactor began to increase. This triggered the opening of the 'electromatic' valve on the top of the primary system pressuriser, which released steam and water into a holding tank in the reactor building. The solenoid which opened the valve moved to the closed position when the pressure had fallen back to normal (and this closure showed on the operators control panel), but the valve itself was actually stuck in the open position. Steam and water escaped from the pressurizer, allowing the pressure to fall further, but the temperature of the reactor core kept increasing. Also the level of water in the pressurizer was indicating much higher than normal and to the operators appeared in danger of going 'solid' (i.e. filling to the top). At this time the reactor tripped and shut down. The operators now had several problems with little or no understanding of the events leading to the situation. Also there was nothing in their emergency operating instructions which gave guidance on what to do to cope with the symptoms displayed on the control panel.
At 8 minutes after the initial trip of the feed water pumps one of the operators noticed that the 'twelves' were closed and immediately opened them. One of the steam drums was damaged and had to be closed off, (they boiled dry 90 secs. after the feedwater pump was shut off) leaving only one drum to remove the heat from the reactor. The operators now had a mechanism for removing decay heat from the core. Unfortunately they did not know that they were loosing water and steam from the primary system. They thought that the pressuriser was about to go solid, so they turned off two of the high pressure injection pumps to reduce the pressure in the pressurizer. These pumps had turned on automatically to provide a cooling water flow to the core. Turning off the pumps further reduced core cooling.

The low pressure resulting from the high pressure injection pumps being shut down allowed the formation of steam bubbles in the water being circulated by the primary system pumps, with the result that vibration of the main heat transport pumps became excessive. The operators shut down two of these pumps to prevent damage to them. Shortly after, the remaining two pumps were automatically shut down and could not be restarted. There was now no direct mechanism for removing decay heat from the core.

Over the next two hours or so the level of water in the reactor fell because it to was being lost as through the open electromatic valve. The operators were confused and still had no clear idea what was wrong. They had called senior staff to come in, and asked for assistance from unit 1 operators. By this time the holding tank for steam and water from the pressuriser had blown its bursting disc and the excess water was flooding onto the containment building floor. This water was being collected in drainage sumps and automatically pumped to holding tanks, outside containment, in the Auxiliaries Building. (Noble gases in this water was the source of most of the Noble gases released during the accident.)

At 6.20 am a senior engineer asked an assistant operator if the Blocking Valve was closed (the 'blocking valve' is in the line to the pressuriser overflow tank; its function is to close off (i.e. stop the flow from) the electromatic valve, if it sticks in the open position). That person went to see, and came back with the answer 'yes'. (The computer recorded the fact that the blocking valve was closed at 6.22 am). This contributed to a continuing lack of understanding of the real problem with the reactor, and the course of events which were unfolding.

At about this time station personnel became aware that there were high radiation levels in the Auxiliaries Building (up to 10 mGy/h (1000 mrem/h)). Many radiation warning and alarm signals were now appearing on the Control panels. Radiation levels in the reactor containment building were in the range 0.1 to 60 Gy/h. It was now apparent that a major release of activity had occurred and a Site Emergency, later up-graded to a General Emergency, was declared. Appropriate authorities were told of 'minor problems; no public safety concerns; and minor fuel damage'. It was to take several more hours for station personnel to obtain a better understanding of the critical damage that had occurred to the reactor. Several days were to elapse. including days of intense stress to station personnel, regulatory authorities, local, state, and federal authorities, and to members of the public, before the situation could be said to be under reasonable control. For some days it was thought that there was the possibility of a hydrogen explosion in the core, and the possibility of a complete melt-down could not be excluded. These possibilities resulted in the evacuation of women and children from the immediate area considered at risk.

Information and mis-information; theory and counter-theory played their parts in imparting the image of a totally confused 'Authority'. In the end the public could trust no-one.

Not everything about the accident was bad. In spite of a hydrogen explosion inside the containment building which caused a pressure spike of 28 psig, the containment was good. Very little noble gas fission product escaped from containment - radiation doses to the public were very small. Radioiodine releases were minor. The most likely explanation for the low radioiodine release is that it had reacted with the silver/boron control rods which had oxidised and melted in the very high temperatures inside the core.

The accident as is true for most accidents had no single causal factor. Deficiencies existed in design, operating and maintenance procedures, operator training and follow up of prior experience. Emergency systems did what they were supposed to do but were interfered with and were permitted from doing their job during the critical period of the accident.

Effects of the Accident

Economic Effects

The accident resulted in a large economic loss to the Company. Both reactors were shut down; unit 1 for over five years until the safety of its operation could be guaranteed and it was re-licensed to operate. An initial estimate of the cost was made by the Kemeny Commission lying between approximately 1 billion and 2 billion dollar. This assumed that unit 2 could be refurbished which has not occurred. The full cost is not known but is likely to be much more than the upper figure given by the Commission. There was also a major loss to the Nuclear Power Industry as a whole. All construction of nuclear power stations in the USA stopped for a time. Eleven nuclear power stations were cancelled. The industry has still not completely recovered.

Radiological Effects

Exposures of plant personnel were not high. In the period from March 28th to June 30th, three TMI workers received radiation doses between 30 and 40 mSv (3 and 4 rem). The U.S regulatory dose limit of 30 mSv per quarter year was exceeded.

The containment system worked well and most of the non-gaseous fission products were retained in either the core, the reactor systems or auxiliary systems and buildings (See Figure 9). Radioiodine which is volatile at low relatively low temperatures was essentially totally contained by the containment or auxiliary buildings. Of a core inventory of I-131 estimated at 2.5×10^{18} Bq (6.7×10^{7} Ci), no more than 5.6×10^{11} Bq (15 Ci) were released to atmosphere and 4.1×10^{9} Bq (0.11 Ci) via liquid effluents. The release of radioactive noble gases was higher - a total noble gas release to atmosphere of about 5×10^{17} Bq (1.3×10^{7} Ci). The total population dose to the public in the vicinity of the station from the gaseous releases is estimated at 33 person-Sv (3,300 person-rem)

Health Effects

The estimated health effects are low. Based on the risk per unit dose in use at the time Arthur Upton of the Institute of Environmental Medicine calculated the values below for fatal and non-fatal cancers in the population within 50 miles radius of the plant. These are given in Table 3.

Table 3

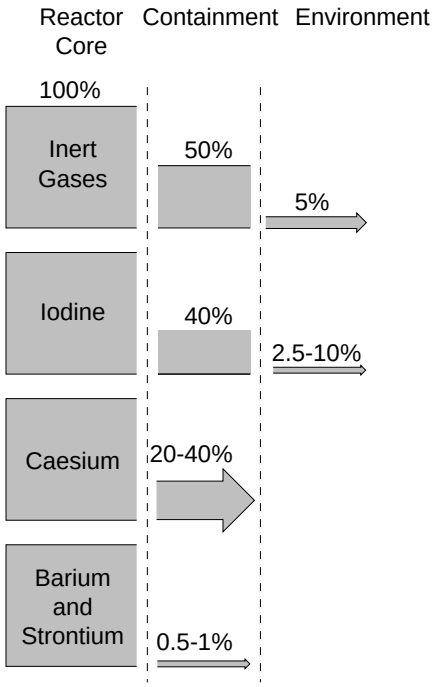
Estimated Health Effects from TMI Accident

	Estimated Numbers Attributable to TMI	**Naturally Occurring Numbers of Cancers**
Fatal Cancers	0.7 (0.15 - 2.4)	325,000
Non-Fatal Cancers	0.7 (0.15 - 2.4)	216,000

The genetic impact is estimated to be of approximately the same order.
The radiological and social impact on the public in the vicinity of the accident was minimal. A few people were evacuated and some restrictions were placed on local produce. There have been however significant stress and psychological problems in the population in the wake of the accident. There is a chronic fear of radiation and its effects upon health. Many people refuse to believe that the average radiation dose received by members of the population was only a fraction of that received in a year from background radiation - often citing examples of public and animal somatic ill-health and genetic effects to support their claims. Stress and psychological problems are the most significant health problem resulting from the TMI accident.

Figure 9

Diagram illustrating the leakage of radioactive substances from the reactor core at Three Mile Island through the first barriers (fuel and primary system) and also the reactor containment



The Chernobyl Reactor Accident Description

The accident at the Chernobyl Generating Station in Russia is the worst ever nuclear reactor accident.

'Chernobyl' is the name given to a nuclear generating station which has four reactors, each capable of producing about 1000 Mw(e). (3200 Mw thermal). By any standards these are large reactors. They are of the type known as RBMK i.e. graphite moderated, water cooled, using a 2% U-235 enrichment in the uranium dioxide fuel. Figure 10 shows a schematic of the reactor construction and layout

and Figures 11A and 11B show the location of the generating station relative to nearby towns,

Figure 10
Schematic Diagram of the RBMK 1000

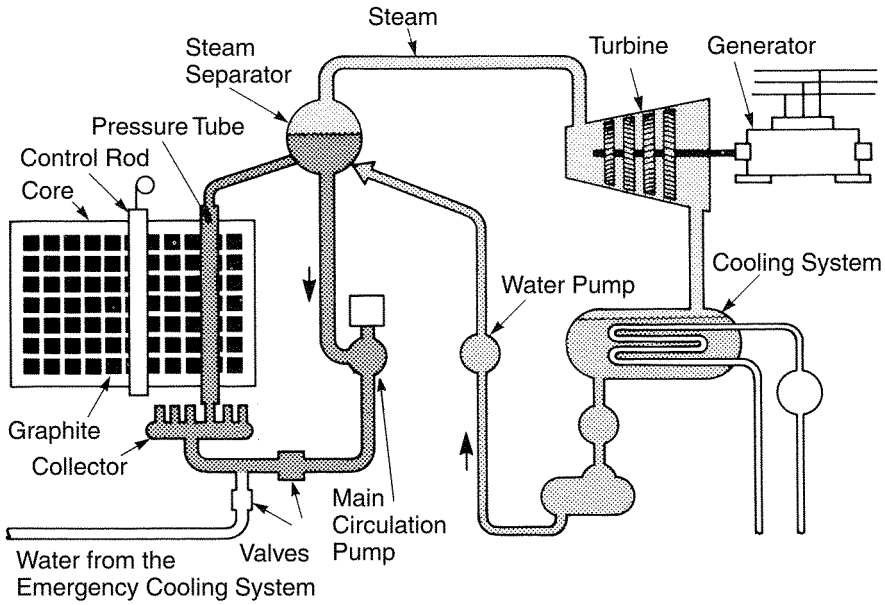
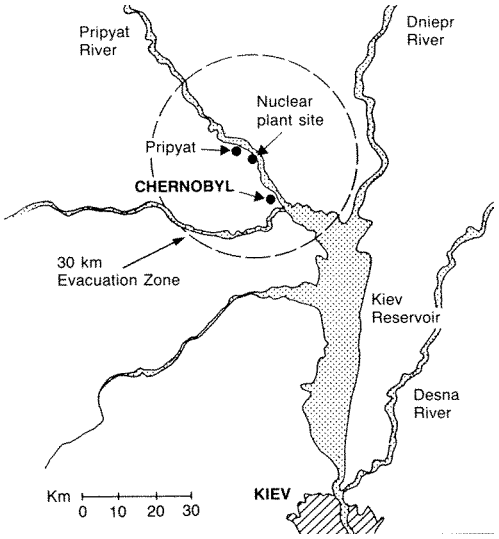


Figure 11b
Chernobyl Reactor Location



The core of the RMBK reactor consists of graphite blocks assembled into a vertical right cylinder. The blocks are pierced by 1660 vertical channels, in which 1600 pressure tubes and 60 control rods are fitted. The control rods are the only shut-down system. Water is pumped from the bottom of the pressure tubes, over the fuel to remove fission heat, then it turns to steam to drive the turbine. Water is also pumped over the control rods to keep them cool. Condensed steam is returned to the coolant system. The graphite core operates at about 700°C (faintly red hot). Heat from the core is removed by the water flowing through the pressure tubes (i.e.there is no direct moderator cooling system). In order to prevent oxidation of the graphite, the core is enclosed in a metal container through which flows a mixture of helium and nitrogen. The metal container is strong enough to contain the steam if a pressure tube or feeder tube should break.

The core is shielded by concrete, sand and water at the sides, with concrete top and bottom shields. All the pressure tubes and control rods are attached to, and supported by the top shield. The bottom of the reactor is above a bubbler pond' intended to capture and store FP's in the event of an accident. This section is a containment structure. The top of the reactor is enclosed in a normal industrial building, probably because the refuelling machine is very tall and the cost of a containment building around it would be very high. The Chernobyl unit 4 reactor had shutdown and emergency core cooling systems but as just described only partial containment.

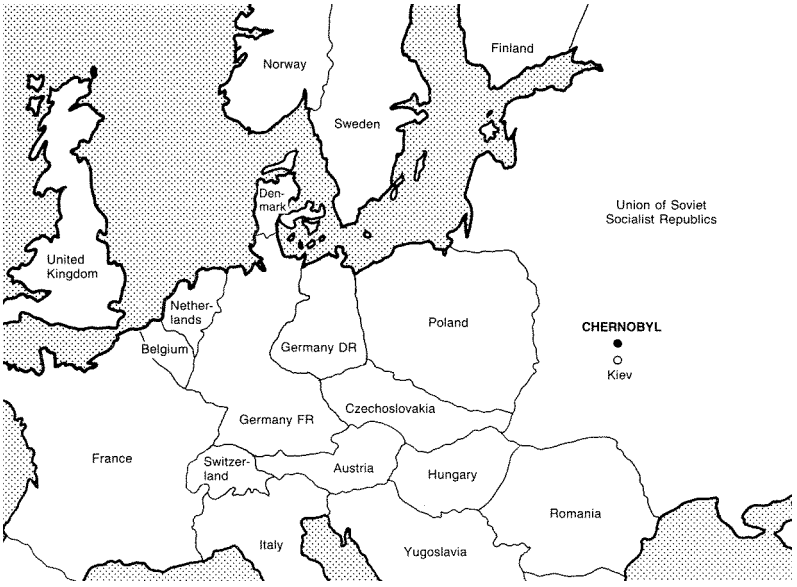


Figure 11B
Chernobyl Reactor Location

The accident scenario was set by an experiment to find out how long a spinning turbine could be used to generate power for plant systems, such as the primary coolant pumps, after the reactor had been shut down. The objective was to see if

the turbine could provide power while the emergency diesel generators were being started up. The opportunity to do the test was provided by an imminent shut-down. Failure to complete the test on this occasion would mean a delay of about one year before a similar opportunity occurred again. Reactor operators were therefore under pressure to see that the test was completed on schedule.

The reactor operators considered themselves an elite crew (i.e. they may have been over confident). They also saw this test as an electrical test, and did not consider the possible effects on reactor performance. In addition the test was being supervised by the turbine manufacturer, rather than the station engineers.

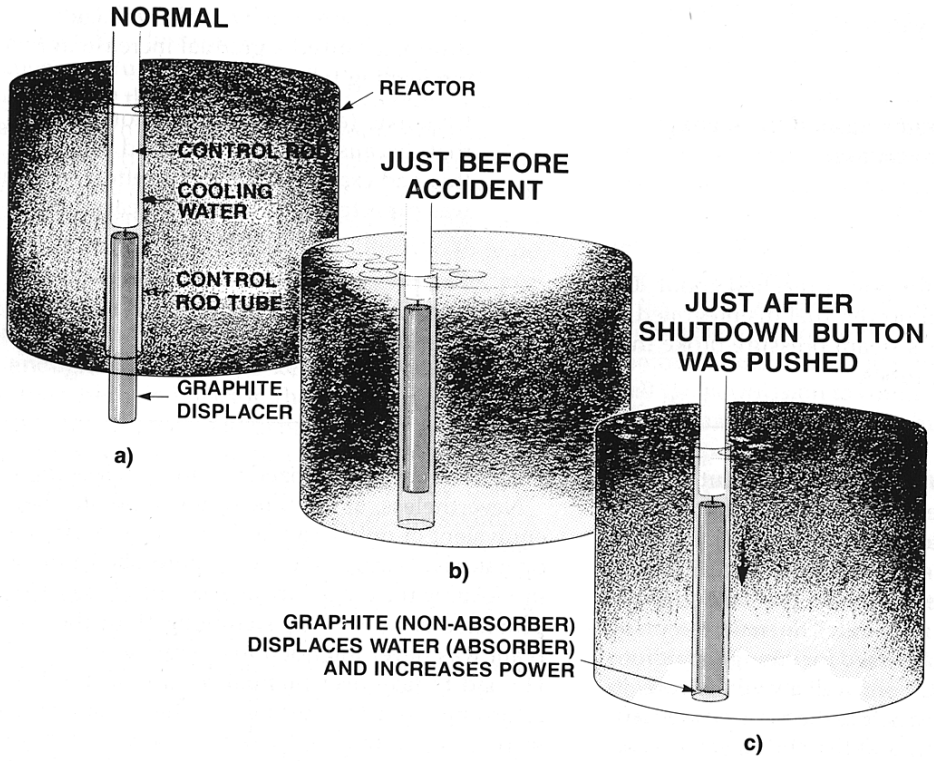
At 13.00 h on 25th.April 1986 the reactor was slowly taken from full power, down to 50%, and all steam was switched to one turbine. An unexpected electrical demand delayed the test for a further 9 hours, the reactor remaining at 50% power. At 00.28 h on 26th April the power was again reduced in order to proceed with the test, but on this occasion the operator made an error which reduced power to about 1% instead of the required 30%. An unavoidable loss of reactivity due to Xenon-133 build-up occurred (reactor poisoning), and the water in the pressure tubes cooled so that there were no steam bubbles in the core. This had the effect of increasing neutron absorption, tending to further decrease the available power. To offset these two effects, the operator pulled out almost all of the control rods (only 6 were left in the core, contrary to standing operating instructions which required a minimum of 32). Power increased to about 7%. The operator spent some time trying to stabilize the reactor manually, being ready to proceed at about 01.20 h.

The reactor was now in an unstable condition because of the combined effect of the position of the control rods and the fact that there were no steam bubbles in the pressure tubes. To further complicate the situation at 01.20 h the operator blocked automatic shut-down so that he could change conditions to maintain the reactor in the operating mode. Finally, at 01.23.04 he tripped the remaining turbine which was generating the power for the feed water pumps. As the turbine slowed, so the pump slowed, and the feed water (moving more slowly over the fuel) began to heat up, generating steam bubbles again. This made more neutrons available for fission, and the reactor power began to rise.

At 01.23.40 the operator pushed the manual shutdown button. As the control rods began to move into the reactor, the graphite displacers (see Figure 12), which do not absorb neutrons, displaced the water (which does absorb neutrons), in the tubes beneath the end of the graphite displacer rod, making more neutrons available for fission. This was the reverse of the intended effect. Within 4 seconds the reactor power had increased to perhaps 100 times full power. The sudden increase in temperature and steam pressure ruptured the pressure tubes. Steam pressure lifted the 200 ton top shield, snapping the 1600 attached pressure tubes and lifting this shield so that it was almost vertical on

one edge supported by the building structure. The top of the building structure was destroyed and blown away by the power surge. White hot fuel elements were blown out of the pressure tubes like rockets, some landing on the roof of the turbine building and the adjacent Unit 3 reactor causing fires.

Figure 12
Role of shutdown rods



By 05.00 h in the morning local fire fighters had extinguished the fires. Many of these men had received excessive radiation doses and were among the 29 people to die from radiation exposure. Two other persons died as a result of the explosion'

The next few days were spent trying to cool the damaged core. Initially they tried to flood the core with water, but this did not work. By the second day the moderator graphite had begun to burn in some places. An attempt was made to smother the core by dropping a mixture of materials onto it from helicopters, primarily to try to stop the release of airborne particulate FP's. These materials did shield the core, and trap the heat, causing an increase in the temperature of the core (decay heat again!) and an increase in the rate of release of airborne activity. The core was finally cooled by flooding it with nitrogen. The release of FP's to the environment virtually ceased. The damaged reactor was entombed in a 'sarcophagus' in the months following the accident. A cooling air flow passes through the structure and is filtered on leaving

Effects of the Chernobyl Accident

Economic Effects

The economic costs of the accident have not been determined, but they are staggering, and are still mounting. Consider that the accident resulted in:

1. Destruction of a 1000 MWe power station.
2. Loss of power from two adjacent units for an extended period.
3. Evacuation relocation and rehousing of several hundred thousand people.
4. Creation of a limited access zone of 30 km radius. This includes the abandonment of several towns and villages.
5. Deployment of large emergency forces to implement protective measures.
6. Construction of containment for the destroyed reactor and dams to control water contamination.
7. Compensation for victims and families of victims.
8. Hospitalization and treatment of injured persons and loss of productivity of these persons.
9. Follow-up health studies on large numbers of people.
10. Loss of food production from contaminated land both in the USSR and much of Western Europe.

The TMI accident cost at least two billion dollars (US). The Chernobyl accident is likely to cost between ten and 100 times this amount.

Radiological Effects

The amount of published information on the radiological effects is enormous and at some times conflicting. The following is a distillation of that information attempting to give an overall broad view of the radiological consequences of the accident.

Radiological Effects - On Site

The radiation fields following the accident were extremely high. Instruments to measure dose rate were unsuitable and read off scale. It is estimated that the radiation fields in the vicinity of unit 4 were greater than 100 Gy/h.
The sources of radiation were:

- short term external fields from beta and gamma radiation from the cloud of gaseous volatile and suspended fission products being emitted from the overheated core,
- short term radiation fields from the burning fuel and core fragments that were dispersed prior to and during their clean up,
- skin contamination from fallout and dispersed fission products,
- inhalation of gases and dusts.

The first three of these were the main contributors to the immediate effects experienced by the emergency response team - particularly the fire fighters and

unit three shift personnel. The distribution of dose among the 444 people on site is estimated to be as follows:

- 22 received doses in the range 6000 to 16000 mSv (all died, one was severely burned)
- 23 received doses in the range 4000 to 6000 mSv (7 died)
- 185 received doses in the range 1000 to 4000 mSv (1 died)
- One person was never found following the accident and the remainder of the people on site had doses less than 1000 mSv.

There were 300 persons admitted to hospital shortly after the accident suffering from the effects of high exposure, 203 of these were diagnosed as suffering from acute radiation syndrome. Another 200 were subsequently admitted to hospital for observation and treatment.

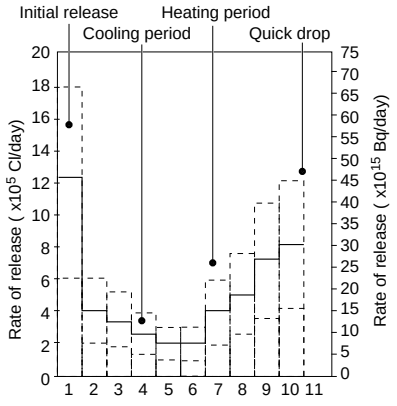
Following the initial phase of the accident when the immediate attempts to deal with the fires and distributed radioactive materials had been carried out the authorities in the USSR established annual occupational dose limits of 250 mSv for personnel at the site and engaged in restorative work.

Radiological Effects - Releases & Plume Spread

The rate of release of radioactive material in the days following the accident was affected by the actions being taken to bring the release under control. The initial release rate dropped as the reactor cooled down partly due to cooling by air passing through it as a result of the moderator graphite burning. Then the action of covering the reactor with 5000 tons of a mixture of lead, boron, sand, clay and limestone to smother the burning parts of the core and stop the airborne release occurring caused the reactor to heat up again because cooling air flow was stopped. This resulted in an increased release of volatile fission products. The burning core was eventually extinguished and releases terminated by pumping nitrogen into the bottom of the core. The release of radioactive material from the damaged reactor extended over a ten day period and is shown in Figure 13. The radionuclides released are shown in Table 4.

Figure 13

Daily radioactive release



Daily amounts of radioactivity released to the atmosphere during the Chernobyl accident (inert gases not included)

Table 4

Core Inventories and Total Releases

Element	Half-Life (d)	Inventory* (Bq)	Percentage released
Kr-85	3,930	3.3×10^{16}	~100
Xe-133	5.27	1.7×10^{18}	~100
I-131	8.05	1.3×10^{18}	20
Te-132	3.25	3.2×10^{17}	15
Cs-134	750	1.9×10^{17}	10
Cs-137	1.1×10^{4}	2.9×10^{17}	13
Mo-99	2.8	4.8×10^{18}	2.3
Zr-95	65.5	4.4×10^{18}	3.2
Ru-103	39.5	4.1×10^{18}	2.9
Ru-106	368	2.0×10^{18}	2.9
Ba-140	12.8	2.9×10^{18}	5.6
Ce-141	32.5	4.4×10^{18}	2.3
Ce-144	284	3.2×10^{18}	2.8
Sr-89	53	2.0×10^{18}	4
Sr-90	1.02×101^{4}	2.0×10^{17}	4
Np-239	2.35	1.4×10^{17}	3
Pu-238	3.15×10^{4}	1.0×10^{15}	3
Pu-239	8.9×10^{6}	8.5×10^{14}	3
Pu-240	2.4×10^{6}	1.2×10^{15}	3
Pu-241	4,800	1.7×10^{17}	3
Cm-242	164	2.6×10^{16}	3

** Decay corrected to 6 May 1986 and calculated as prescribed by Soviet experts.*
(Source: IAEA, 1986)

The first indications that anything had gone wrong at a Soviet nuclear installation was high readings obtained from radiation monitoring systems in Sweden and Denmark. The plume from the plant initially was blown in a NNW direction but later was blown West and in the latter phase of the release contaminated land to the South and South West of the plant. The plume distribution was widespread and extended over all of Western Europe. (Figures 14A, 14B, 14C, 14D)

Radiological Effects - Immediate Vicinity

Dose rates in the vicinity of the plant were high. Figure 15 shows the gamma dose rate as measured by Soviet authorities on the 29th May. On Saturday April 26th the dose rate in Pripyat the nearest town of any size was not considered too alarming with gamma dose rate readings between 10 and 100 micro-Sv/h (1-10 mrem/h). Late that evening a decision was made to evacuate the town. Forty five thousand people of the town were evacuated over about three hours to villages in the surrounding regions. Over the next few days a further 90,000 people were evacuated from within a radius of about 30 km. The evacuated people did not carry radiation dosimeters so that the radiation dose they received is an estimate from dose rate measurements. The estimated average dose was about 30 mSv (3 rem) and the estimated highest dose was 100 mSv (10 rem) .

Figure 14A
April 28, 1986

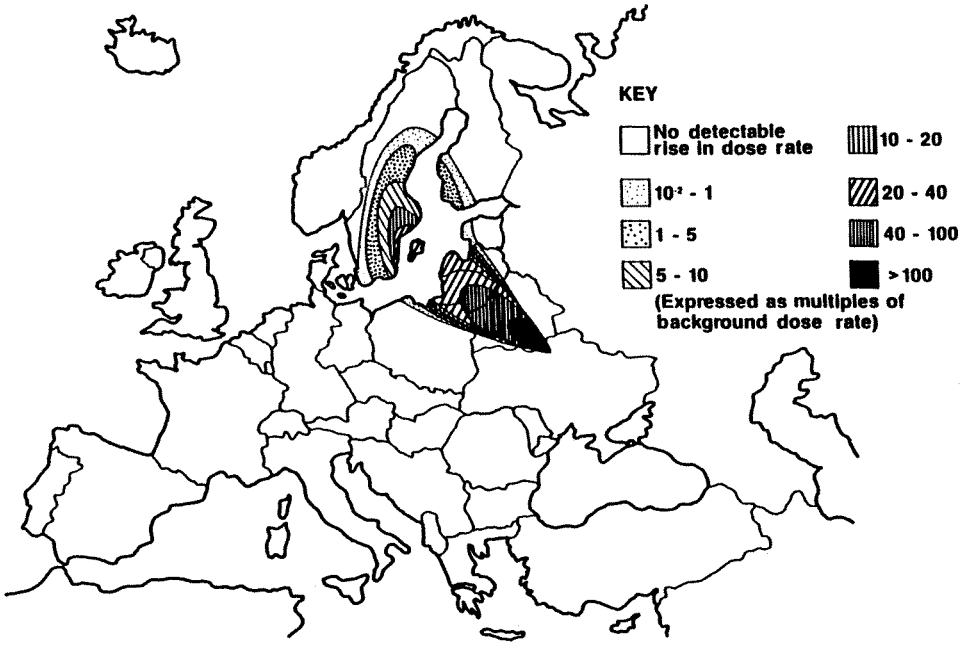


Figure 14B
April 30, 1986

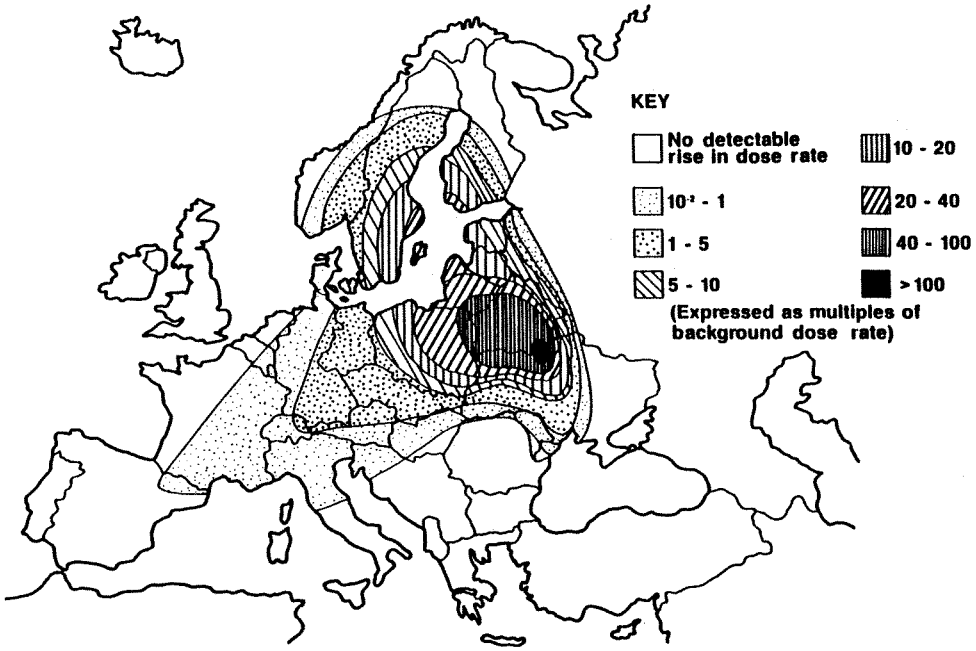


Figure 14C
May 1, 1986

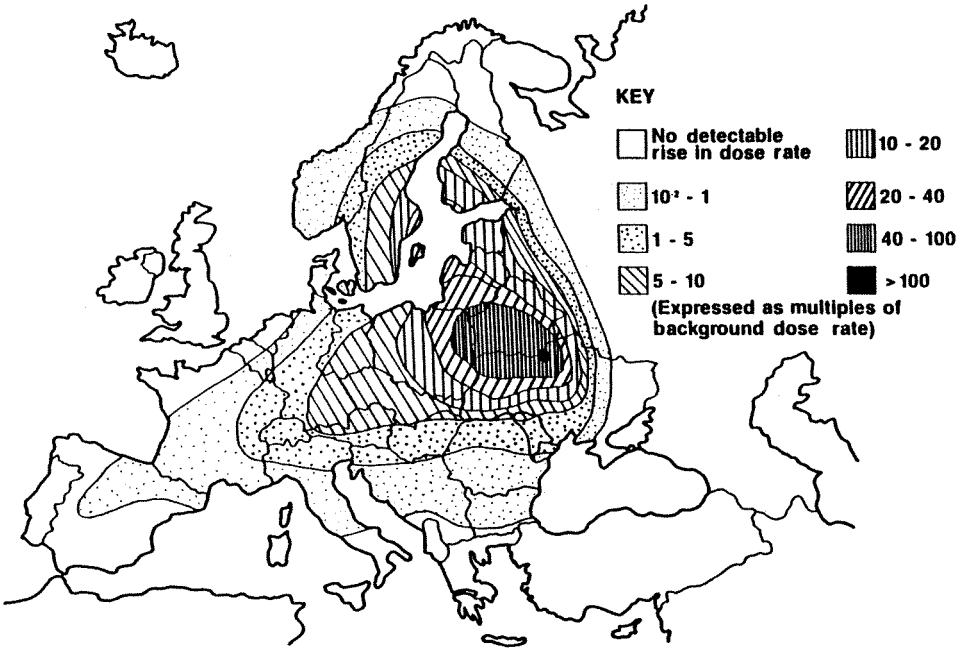


Figure 14D
May 3, 1986

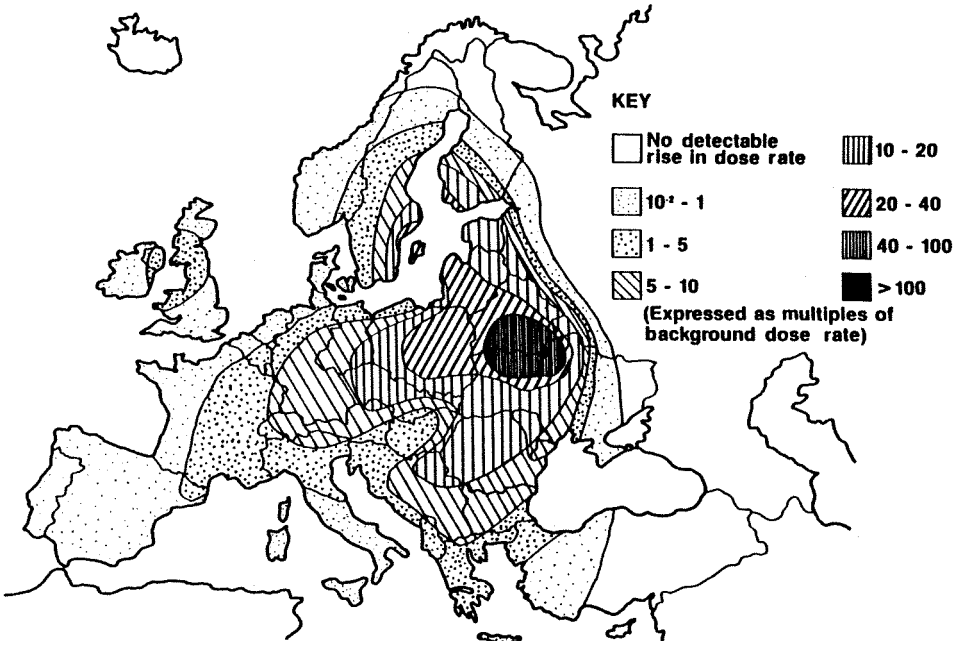


Figure 15

Radiation levels resulting from Gamma radiation from the ground around Chernobyl on 29th May. Taken from the Soviet report that was submitted to IAEA. Items marked 1, 10 and 100 mR/h correspond to 0.15, 1.5 and 15 mSv per day

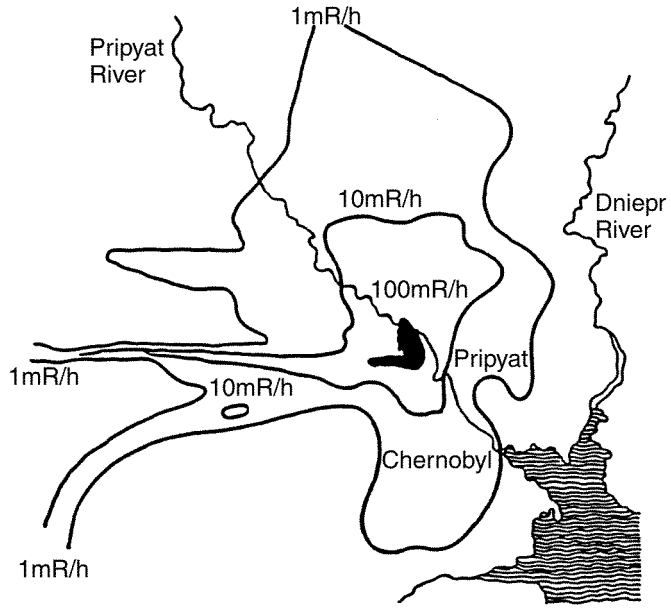


Figure 16

General pattern of deposition of iodine-131, Bq m^2. Values are rounded to the nearest order of magnitude

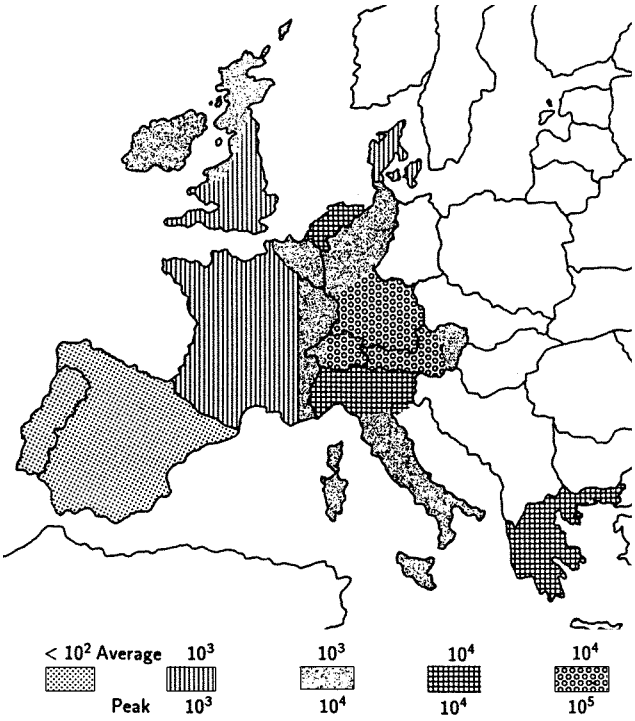


Table 5

First year thyroid dose equivalent in countries and subregions (mSv)

Region	Infants	Adults
Northern Europe (Denmark, Finland, Norway, Sweden)	0.15-1.8	0.06-1.2
Central Europe (Austria, Czechoslovakia, FRG, GDR, Hungary, Poland, Romania, Switzerland)	1.7-18	0.4-2.8
Western Europe (Belgium, France, Ireland, Luxembourg, Netherlands, UK)	0.7-2.7	0.1-0.6
Southern Europe (Bulgaria, Greece, Italy, Portugal, Spain, Yugoslavia)	<0.01-25	<0.01-5.5

(Source: UNSCEAR, 1988.)

The population in the immediate vicinity of the plant, about 24,200, people who were evacuated late and were not instructed in simple protection methods e.g sheltering, received an estimated average dose of 450 mSv (45 rem). In 1987 the Soviet authorities said that these early dose estimates might have been too high but did not give corrected values.(possibly by a factor of 2 to 3)

The Soviet authorities have established three categories of contamination zone:

1. 400-550 kBq/m^2 (1.1-1.5 micro-Ci/m^2): under sporadic control, no special measures adopted.
2. 550-1500 kBq/m^2 (1.5-4 micro-Ci/m^2): under constant control, decontamination measures implemented, food restrictions.
3. >1500 kBq/m^2 (>4 micro-Ci/m^2): under strict control. Unfit for human habitation. 200,000 people have been evacuated from these zones.

The basis established by the Soviet authorities in conjunction with the World Health Association in June 1989 is that a lifetime dose should not exceed 350 mSv average.

The authorities distributed stable iodide pills as a preventative measure. It is reported that over 5 million people received these pills. In the Ukraine despite the distribution of stable iodide pills it is reported that 200,000 children received and average thyroid dose of 1000 mSv and 500 are reported to have received a thyroid dose in excess of 10,000 mSv.

Radiological Effects - Continental

As seen from figures 14A to 14D the plume from Chernobyl spread over much of Western Europe as well as the eastern part of the European Soviet Union. In the countries of Western Europe the concerns were for I-131 and Cs-137 . The concern was for the dose delivered through the food chain in the case of I-131 and through the food-chain and from external radiation in the case of Cs-137.

The general deposition pattern of I-131 in Western Europe is shown in Figure 16

Table 6

Continental Radiological Effects

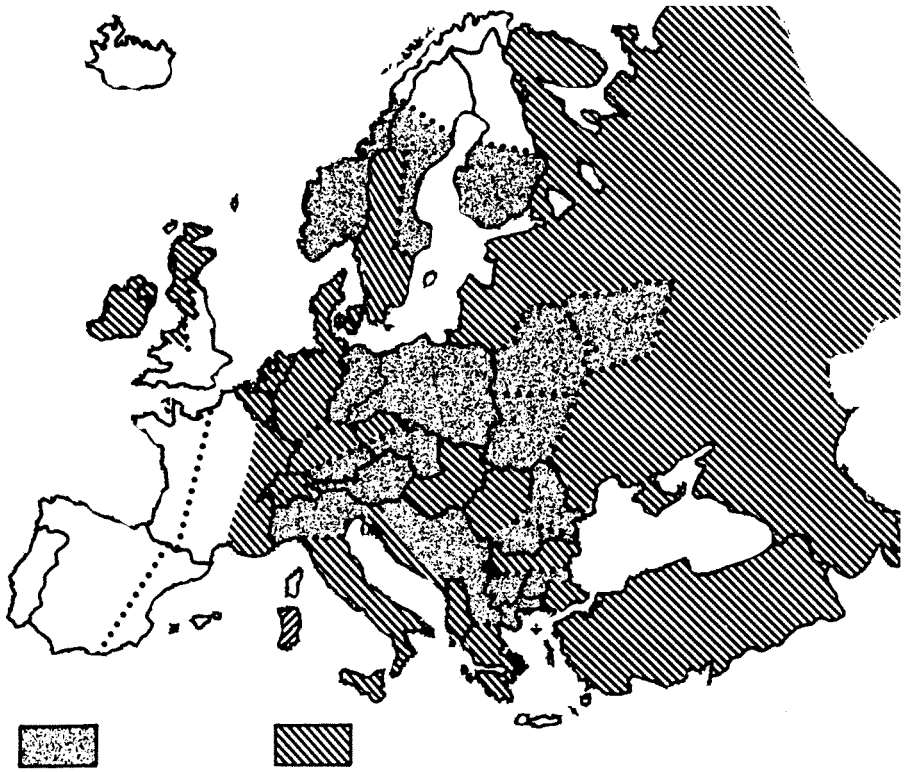
Country	Dose range (_Sv)
Austria, Bulgaria, Greece, Romania	500-750
Czechoslovakia, Finland, Italy, Poland, Switzerland, Yugoslavia	250-500
Federal Republic of Germany, German Democratic Republic, Hungary, Ireland, Norway, Sweden	100-250
Belgium, Denmark, France, Luxembourg, Netherlands, Portugal, Spain, UK	<100
Overall average	200

(Source: UNSCEAR, 1988.)

Average first year effective dose equivalent in the European countries.
The total collective effective dose equivalent commitment for the population of the European continent for the accident estimated by UNSCEAR is about 556,000 person-Sv of which 226,000 person-Sv is incurred by the population of the former Soviet Union and 330,000 person-Sv by the population of other European countries.

Figure 17

Average Cs-137 deposition density in countries or large subregions in Europe



J.H Gittus of the UKAEA has reported on the individual and population doses for various European countries. Table 7 gives the estimated population dose by pathway and Table 8 gives individual and collective dose data for selected countries.

Table 7

Dosimetric Assessment for Western Europe Contribution by Pathway

Pathway	Collective Dose (Person -Sv)	Percent
Inhalation	3,600	5
Ingestion		
Milk	11,000	14
Veg	15,200	20
Meat	12,500	17
External	33,300	44

Table 8

Dosimetric Assessments For Western European Countries

Country	Mean Individual Dose (Micro-Sv)	Collective Dose (Person-Sv)
France	46	2,500
Italy	500	28,600
Sweden	770	6,400
United Kingdom	50	2,800
West Germany	70	15,400

Health Effects

An estimate of health effects may be made using the values from the 1990 recommendations of the ICRP on radiation detriment. The value given for detriment coefficients per Sv for the general public is 7.3×10^{-2} per Sv. The component of this value for fatal cancers is 5×10^{-2}. Using this latter value and the value for population dose of 555,600 person-Sv quoted earlier in this section the number of fatal cancers that may result in Western Europe and the eastern European section of the former Soviet Union is 28,800. This number will be difficult to detect in the background of natural cancers.

When any large release of radioactive material occurs there are anecdotal reports of cancer cases and severe genetic abnormalities, e.g. calf born in the Ukraine plays the guitar and looks like Elvis Presley. Scientifically based health studies are the only way in which valid information may be obtained. Health studies are being carried out under the International Chernobyl Project.

The findings of the International Chernobyl Project were released in May 1991. The findings were:

1. The actions of the authorities (where these could be properly assessed) were

reasonable and consistent with internationally established guidelines established at the time.

2. The protective measures taken, generally exceed what would have been strictly necessary from a radiological protection viewpoint. Relocation and foodstuff restrictions should have been less extensive.
3. Dose estimates were lower than officially reported estimates by factors of two or three - but this is reasonable agreement.
4. The project team could see no health disorders that could be directly imputed to radiation exposures, other than psychological, which was due to anxiety and stress from continuing high levels of uncertainty. The vast majority of adults examined (in both contaminated and control areas) thought that they had an illness due to radiation exposure.
5. The difference between the dose limits for planned exposures and those for intervention was not understood by the authorities. The dose averted should be the relevant quantity for judging the radiological benefits of relocation. However, of those surveyed (and living in a contaminated area), 75% wanted to relocate.

In the years following the accident the inhabitants of the area have been subjected to conflicting statements as to the health risks which they must live with (perhaps as much from western sources as from Russian). They are now confused, and do not trust any 'authority'.

Radiation Emergency Response

Table of Contents

Introduction

One of the principal objectives of power reactor designers is to produce a safe reactor with an exceedingly low risk of an accident leading to a release of radioactive material to the environment. Despite these intentions, there have been several reactor accidents and incidents in the history of nuclear reactor operation.

It is prudent therefore, to plan for emergencies involving the reactor. The objectives of this emergency planning are to lessen the effects of an accident and protect both the workers at the site, and the public living and working in the areas surrounding the reactor. Emergency planning should be considered as an extension of the defence in depth philosophy that characterizes reactor safety. It is the final barrier.

Radionuclides of Importance and Principal Exposure Pathways in an Emergency

It is useful to consider exposure pathways as **Short Term** and **Long Term** Short term exposure pathways in this context are measured in weeks or months from the time of an accidental release of radioactive material. Long term follows and may extend for years after the release.

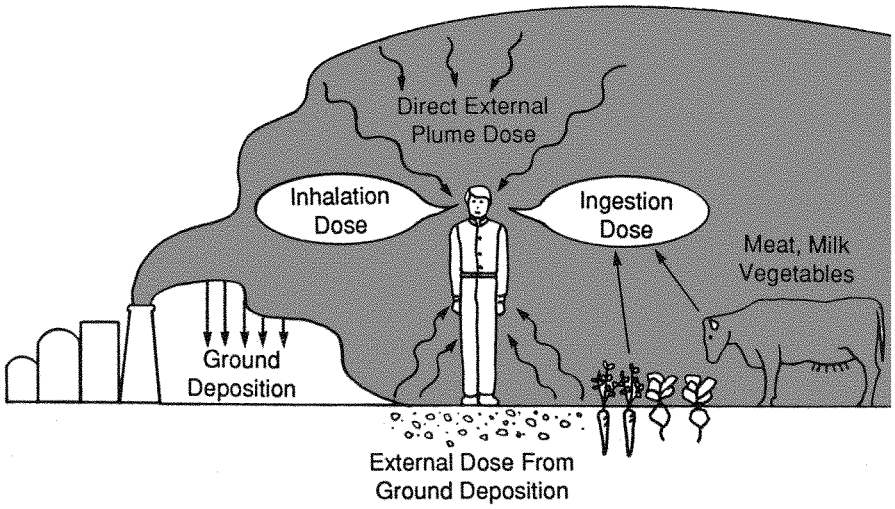
Short Term Pathways

Short term pathways of importance are:

- External beta/gamma dose from the radioactive plume resulting from the incident. All radionuclides emitted may contribute to this 'cloud dose' but the radioactive noble gases such as K-88 and X-133 are likely to be the major dose contributors.
- Internal dose from inhalation of radionuclides in the plume. The noble gases do not contribute significantly to this dose as they are not taken into the body and the lung dose is only that due to the lung gas content. Important contributors are likely to be the radioiodine, I-135, I-133 and I-131, and cesium-134 and cesium-137 and possibly other fission products which are semi-volatile. In CANDU reactors, tritium may also be important but the dose this may contribute is quite limited by the inventory of tritium in the reactor.
- External beta/gamma dose from ground surfaces contaminated by fallout from the plume. Radioiodine, cesium 134 and cesium-137 and other particulate activity in the plume are the principal contributors to this pathway. Noble gases do not contribute as they disperse in the atmosphere.
- Internal dose from ingestion of foodstuffs such as vegetables which have had radioactive material deposited from the plume onto their surfaces. Radioiodine, cesium-134 and cesium-137 and other particulate in the plume are important.

- Internal dose to the thyroid, particularly of children, from ingestion of milk from cows grazing contaminated pastures is an extremely important pathway. I-131 because of its eight-day half life is of principal concern.

Figure 1
Principal Exposure Pathways Following an Accidental Radioactive Release from a Reactor



Longer Term Pathways

In the longer term the important exposure pathways are defined by long lived fission products especially those that are volatile. By far the most important are cesium-134 and cesium-137, although in incidents where non-volatile have been dispersed by physical forces Sr-90 and transuranic radionuclides may be significant.

Long term pathways of importance are:

- External gamma dose from radionuclides retained in the soil. The radiocesiums are of greatest importance.
- Internal dose from ingestion of flesh from animals and fish which have fed on vegetation, etc. grown in contaminated soil or water.
- Internal dose from drinking water from contaminated sources.
- Internal dose from airborne radioactivity re-suspended from ground fallout.

Intervention Strategies

The short term and longer term exposure pathways determine the useful intervention strategy, that is action, which may be taken to reduce the exposure of the public.

Intervention actions that may be taken to reduce dose resulting from the short term exposure pathways are:

Staying Indoors

this reduces cloud beta/gamma dose and the inhalation dose if windows and doors are tightly closed.

Temporary Evacuation

this draconian measure reduces exposure from all pathways.

KI Pill Distribution

taking a 100 mg KI pill, blocks uptake of iodine by the thyroid; this reduces the dose from exposure to radioiodine either by inhalation in the cloud or by drinking milk containing I-131.

Intervention actions that are useful in reducing dose in the longer term are:

Food Controls

this reduces ingestion dose; this may be initiated early to reduce dose from surface contamination on food; later, food restrictions reduce dose on radionuclides incorporated into meats and root crops etc.

Changed Agricultural Techniques

these reduce the uptake of radionuclides by plants and animals, e.g deep ploughing. This takes deposited radionuclides deep into the soil providing shielding and making the radionuclides unavailable to growing crops.

Decontamination

this may be initiated early to reduce radiation doses from radionuclides on roofs and streets etc.

Spraying Roads Etc. With Oil

this keeps down dust and reduces airborne exposure and inhalation dose from re-suspended radionuclides.

Relocation

this is a measure that will reduce all forms of exposure; it is only necessary in extreme accidents.

Regulatory Basis for Nuclear Emergency Planning in Canada

Three levels of government and a provincial agency are involved in nuclear emergency planning for reactors. These are:

- The Federal Government
- The Provincial Government
- Local Municipalities
- The utility or other owner of the reactor

The regulatory responsibility for nuclear emergency planning is reasonably clear but there are some grey areas. Responsibility for emergency planning is as follows:

1. The provinces are responsible for public health and so are responsible for off-site nuclear emergency planning.
2. The owner of a reactor is responsible for the preparation of an on-site emergency plan. This must be approved by the AECB.
3. At the federal government level there are two agencies with important responsibilities - Health and Welfare Canada and the AECB. Health and Welfare has been designated as the coordinating agency in the Federal Nuclear Emergency Plan (FNERP). This plan identifies roles for government departments which are likely to be involved.
   The Atomic Energy Control Board is responsible for licensing nuclear stations and as part of this process approves station emergency plans. The AECB does not approve off-site plans but requires that these be in place before a licence being granted.
4. Municipalities are responsible for directing the principal off-site forces that would be involved, e.g., police and firefighter. They therefore have a responsibility to prepare plans for the involvement of their staff and to conform with provincial plans if necessary.

Utility and Station Emergency Planning

Corporate Planning

Planning in a utility takes place at both the corporate level and the station. Corporate planning has several major activities:

1. Providing accurate and timely information about the station and reactor conditions;
2. Providing human and technical resources needed by the station from both inside and outside the company;
3. Providing an objective and broader perspective on the accident than is likely to be developed by those who are in the thick of the emergency at the station;
4. Keeping senior corporate executives informed of the status of the accident and emergency actions being taken.
5. Communicating with senior provincial officials and the AECB.

Planning usually requires that there be a senior utility group established. Accommodation for this group to meet, hold conferences and communicate with the media has to be prearranged. A major accident will attract attention from international news media. If reporters are not provided with satisfactory and timely information they will manufacture news and could interfere with staff engaged in emergency activities. It is important that there be only one spokesperson providing accurate timely official information about station conditions. The conference facility requires reliable communications to the station and other locations such as the provincial emergency centre, some reference documentation - maps etc..and information on resource sources. The purpose of the corporate activities is to provide assistance to station staff and thereby relieve the staff of all but essential emergency recovery and response work.

The corporation will also provide assistance to the provincial authorities during the emergency. For utilities with a single station this assistance may be provided by personnel from the station involved in the incident. In an accident at an Ontario Hydro station, other stations and parts of the organization will provide personnel. This assistance will include the provision of radiological protection services to emergency forces and assistance at reception centres for people evacuated from contaminated areas.

Station Emergency Planning

All Nuclear Generating Stations are required to have an on-site emergency plan in place and approved by the AECB before a licence to operate the station is given by the AECB. A plan should include an appropriate emergency station organization to:

a) direct the actions to put the reactor involved in a safe stable state;
b) manage the remaining units on the site in a multi unit station;
c) carry out planned emergency activities;
d) obtain needed technical advice.

The shift supervisor (or superintendent) is in charge of the station until relieved by station management. To ensure that the shift supervisor has appropriate technical advice available an advisory group is usually established for this purpose.

Figure 2
Stages of Response

The Stages of Response

Category I	Category II	Category III			
Province notified	Province continues to monitor Municipalities notified Emergency organizations placed on standby	Province and municipalities notified Provincial and municipal plans activated	Public notification begins Provincial and municipal plans continue	Province assumes control Provincial and municipal plans continue	Protective actions begin • Traffic control • School, business recreation activities suspended • Other protective actions as required
Ontario Hydro emergency response centres activated	Ontario Hydro monitors and takes corrective action	Ontario Hydro continues to monitor and take corrective action	Ontario Hydro continues to monitor and take corrective action	Ontario Hydro continues to monitor and take corrective action	Ontario Hydro continues to monitor and take corrective action
Routine news release issued	Routine news release issued	Routine news release issues	News releases continue	News releases continue	
Time: Varies	Time: Varies	Time: .5 hrs	Time: 1.0 hrs	Time: 3.0 hrs	Time: Varies

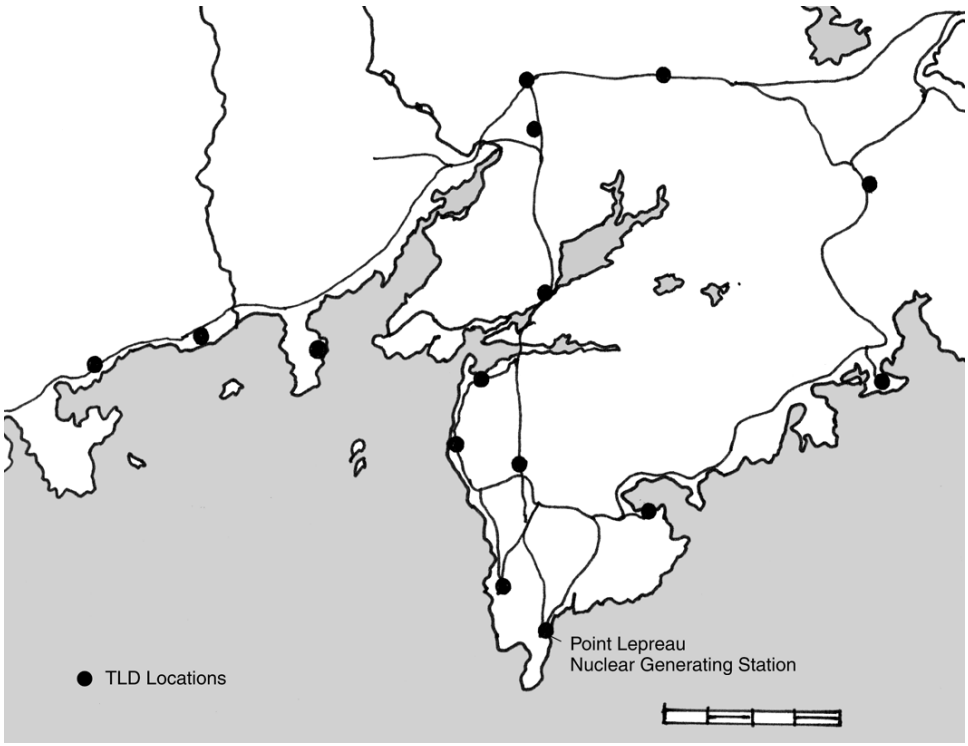
Planned Station Emergency Actions

Emergency plans should be fully documented to avoid the possibility that some actions may be overlooked in the intense activity that usually accompanies an accident. Typically planned emergency actions include:

- Measures to assure the safety of plant personnel and visitors, such as gathering at identified assembly locations, taking roll-calls and carrying out search and rescue operations. Activities such as the latter can require radiation dose and dose rate meters suitable for high fields.
- notification of provincial emergency authorities, the AECB, company officials etc.. A preplanned communication list is established which operates on a

fanout basis but with a call-back confirmation requirement. The stages of a typical notification and response are shown in figure 2. Included in the notification of provincial emergency authorities may be a requirement for an assessment of the magnitude of the accident.

Figure 3A
Emergency TLD Array Around Point Lepreau Nuclear Station



In the plans of Ontario Hydro stations there is a definition of categories of emergencies for this purpose. These categories agreed upon by the Province of Ontario and Ontario Hydro are as follows:

Category I

an abnormal condition that will not result in a hazard to the public. No action outside the station boundaries, but Authorities informed.

Category II

an abnormal condition combined with problems with station safety systems. No abnormal release of radiation. Authorities informed and Emergency Organizations on standby.

Category III

a serious incident with potential for the release of radiation: Emergency Organizations activated. The public would be notified and kept informed;

- arrangements completed with local hospitals for treatment of contaminated and exposed casualties; additional arrangements may have been made with central hospitals or special groups for treatment of highly exposed workers;

- arrangements with local fire departments for assistance with fires. This and the arrangements with local hospitals require that personnel from these groups receive training in radiation protection.
- completion of extensive off-site radiation surveys and dose measurements. The surveys concentrate on the down wind area around the plant. The purpose of this action is to assist in establishing a measure of the magnitude of the accident and to be able to advise local police forces on any urgent actions needed, such as issuing stay-in warnings, traffic control, distribution of iodide pills or evacuation. Radiation surveys require meters and techniques which can measure slight increases in background radiation levels. In addition, distributed around stations there is usually a grid of passive sensitive dosimeters, which can be collected to give the integrated dose at off-site locations. Figure 3A shows the distribution of environmental TLD's around the New Brunswick Power's Point Lepreau station, Figure 3B around Quebec Hydro's Gentilly station and Figures 3C and 3D around Ontario Hydro's Pickering Stations.

Figure 3C
Near Boundary Emergency TLD Array Around Pickering Nuclear Stations

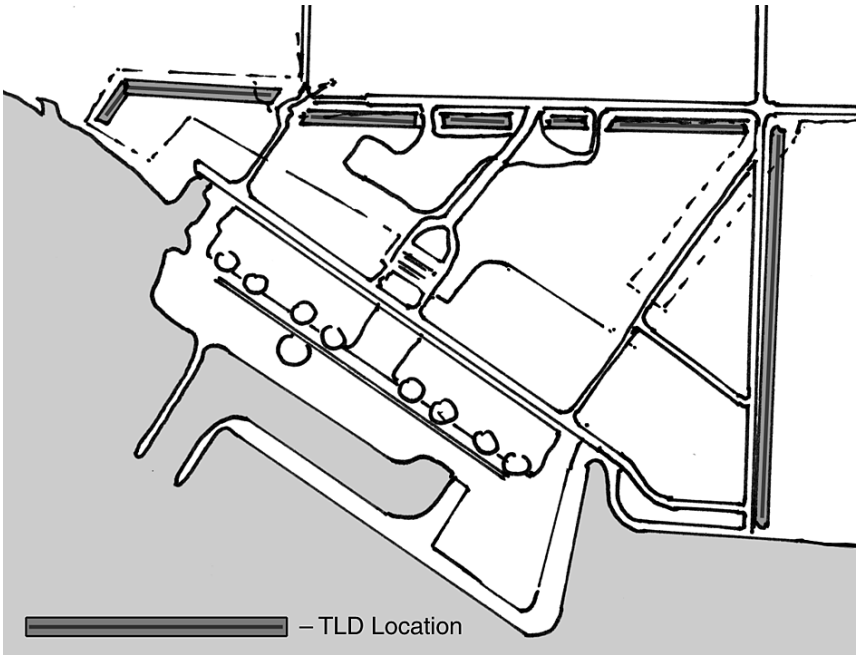
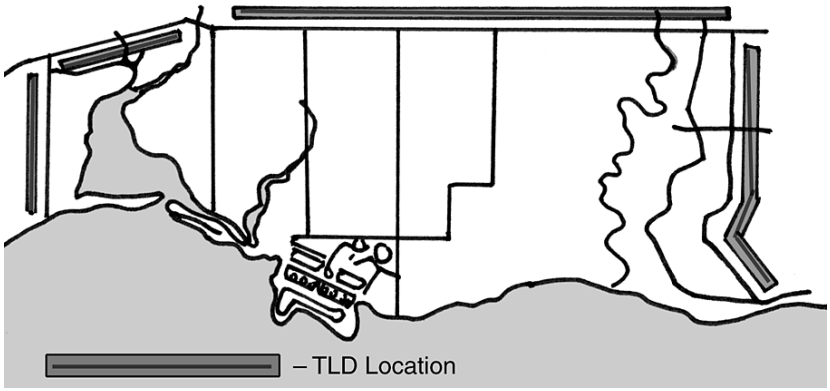


Figure 3B
Emergency TLD Array Around Gentilly Nuclear Station

Figure 3D
Far Boundary Emergency TLD Array Around Pickering Nuclear Stations



An important concern during an emergency is the timing of planned releases. Although containment systems will delay releases, and contain the bulk of any radioactivity escaping from systems, it may be necessary to release some activity in a planned way. Provincial emergency authorities and the AECB would participate in discussions on the timing of such actions.

Emergency plans and response actions must be exercised regularly. All stations in all three nuclear utilities do this. The frequency of the drills depends on the scope of the exercise. Typically a station will exercise parts of their plan e.g., roll call and rescue operations several times a year, and carry out a full scale on-site exercise every year or two years. Exercises in conjunction with an off-site provincial exercise are also held.

Provincial Emergency Planning

Introduction

The responsibility for off-site nuclear emergency planning and response rests with the provinces. In New Brunswick that responsibility has been assigned under the Emergency Measures Act to the Emergency Measures Organization. In Quebec the responsibility rests with Sécurité Civile du Québec and in Ontario the Ministry of the Solicitor General has the responsibility under the Emergency Plans Act of 1983. The plans of all the provinces with nuclear power programs are similar but not identical. Typically the plans consider the exposure pathways and intervention strategies previously described in section 2.

Assembly

Provincial emergency forces will be alerted and assembled depending on the information supplied by the utility. At the first level of alert, for an emergency with no off-site involvement beyond normal operation, the director of the plan will simply maintain communication with the station and prepare necessary public communications. At higher levels of emergencies, the off-site forces will assemble and direct off-site operations through provincial, municipal and utility forces.

Typical Emergency Organizations

Emergency organizations that have been established typically consist of:

- an executive group,
- an operations group,
- a technical advisory group and,
- an information group.

The Executive Group establishes communications with the highest level of provincial government and directs the off-site activities. The Operations Group is composed of people who are familiar with the provincial, municipal and utility forces and transmit or carry out the directions given by the executive group. Technical advice on radiological conditions, plume path and dispersal, environmental releases and projected doses and reactor status are provided by the Technical Group. Responsibility for collecting information and preparing official information for release to other involved agencies and the public is assigned to the Information Group.

In an emergency there is a clear division of responsibility between the nuclear station involved and the provincial emergency organization. The stations are responsible for on-site actions to evaluate the seriousness or otherwise of the emergency, to put the station in a safe state and establish control of any radioactive releases occurring. In addition by agreement with the province until the provincial emergency group assembles, stations will recommend to municipal forces any urgent initial off-site actions that are necessary. This is to

ensure there is no period without some competent persons to provide advice. After the provincial emergency group has assembled and taken command off-site emergency actions are under their overall control. Nuclear stations assist by providing trained staff and resources to man reception centres, issue dosimeters to police, etc..

The emergency planning of the Province of New Brunswick identifies a significant role for Wardens in the Point Lepreau area. Wardens are persons identified in the general provincial emergency plan. They are local people and are used to carry out some emergency actions under the various emergency plans in the province.

Off-Site Emergency Actions

Three phases for emergency actions are identified in the plans of Quebec and New Brunswick. Ontario identifies only two; the second phase is a combined ingestion control and restoration phase. There are therefore in place in Canadian provincial plans three phases:

- **The Plume Or Cloud Dosage Phase**
- **The Ingestion Phase**
- **The Restoration Phase**

In the **Plume Phase** the emergency actions and responses planned include establishing traffic control, distributing iodide pills and issuing stay-in and evacuation directives. These actions will be implemented in around the plant depending on the plume path and predicted plume dose. Sectors are often predefined for this purpose. The Technical Group will carry out plume path plotting and plume dose calculations to help the executive group in this decision making. The Technical Group will also provide advice on radiological surveys and measurements that should be carried out.

In the **Ingestion Phase**, the Executive Group will obtain information on radiological conditions and advice on these conditions from the Technical Group. They will then issue directives to place restrictions on the consumption of food which may be contaminated, most likely with I-131 or Cesium-134 or 137.

All of the provinces have included in their plans a range of levels at which the actions just described will be implemented. These planned intervention or action levels along with U.S. levels and international recommendations are given in Table 1. The differences in these planning values from province to province and the United States could be the source of public concern in an emergency involving a Canadian reactor. Control measures will be implemented through the appropriate ministries of the provincial government.

Table 1

Protective Measures (1)	International Agencies (2)		ACRP (5)		Ontario (6)		Quebec (7,8,9)		New-Brunswick (10)	United States (11,12)	
	Whole Body (3)	Single Organs (4)	Whole Body	Thyroid	Whole Body	Thyroid	Whole Body	Single Organs		Whole Body	Thyroid
Suspension of recreational activities (3,10)		0.05-5	0.5-50							5 mSv/y	
Use of protective equipment (6)											
Control of Access								1µSv/hr		10 µGy/hr	
Sheltering	5-50	50-500	0.5-5	5-50	1-10	3-30	5-50	50-500		10-<50	50-<250
Evacuation	50-500	500-5000	50-500	10^3-10^4	10-100	30-300	50-500	500-5000	50 mSv whole body dose equivalent save (1-5 mGy/h)	≥50	≥250
Administration of Stable Iodine	-	50-500	-	50-1000	-	30-300		50-500	500 mSv infant thyroid dose equivalent saved (50µGy/y)		
Relocation	50-500	-								≥20	
Decontamination of Persons											
Decontamination of Land and Property									≥20		
Water Controls	5-50	50-500	0.05-5	0.5-50	0.5-5	1.5-1.5	5-50	50-500	5 mSv whole body ALI (13) Short term release = 100 x weekly DEL (14) (gaseous)		
Food Controls	5-50	50-500	0.05-5	0.5-50	0.5-5	1.5-15			5 mSv whole body ALI	50	150
Seafood Controls (10)									Short term release = 10 x monthl DEL (liquid)		
Use of Stored Animal Food										5	15
Emergency Planning Zone Radius (km): - Plume Exposure - Ingestion					10 km 50 km		16 km 80 km		20 km 80 km	16 km 80 km	

Footnotes to Table 1

1. *IA86c*
2. *NE89a*
3. *Whole body effective dose equivalent.*
4. *Lung, thyroid or any single organ preferentially irradiated. In the event of high dose alpha irradiation of the lung, the numerical values apply to the product of the relative biological effectiveness and the absorbed dose in milligrays. (IA86)*
5. *AC83*
6. *ON86*
7. *HQ81*
8. *QU88*
9. *HQ90*
10. *NB90b*
11. *US90b*
12. *NR78*
13. *ALI - Allowable Limit on Intake*
14. *DEL - Derived Emission Limit*

All provincial plans are weak on the **Restoration** phase and do not include discussion of the decontamination of buildings or land, identification of disposal sites for radioactive wastes and many other actions that would be necessary then. If evacuation of the population in any sector in an emergency were to occur, important decisions will need to be made regarding return of those evacuated and the lifetime dose they could be permitted. Consideration and discussion of this type of problem are probably best done before any need for the action.

Municipal Emergency Plans

All provinces carry out emergency exercises regularly. Some exercises test parts of the plan but full scale exercises are also carried out. The Province of Ontario has recently started annual audits of its emergency planning.
Municipalities around existing Canadian nuclear stations vary in their size, population and complexity of their organizations. Sometimes the municipalities are rural, e.g., Ville de Bécancour in the case of Gentilly, and some are semi-urban such as Durham region which contains both the Pickering and Darlington sites. The rural municipalities rely heavily on the utility and the province for resources and expertise and in the preparation of their plan. Semi-urban municipalities have large competent firefighting, police, ambulance and social service groups and these municipalities prepare their own emergency plans which of course must mesh with provincial and utility plans. The plans mainly describe how the municipality will carry out the directions and advice given by provincial emergency organization. This includes informing of stay-in requirements, traffic control, evacuation, reception decontamination and housing of evacuated people.

Federal Nuclear Emergency Planning

There are many agencies at the federal government level involved in nuclear emergency planning. If the emergency arises out of military action then the Department of National Defence (DND) is the agency responsible for emergency planning and response. For nuclear emergencies associated with nuclear power stations there are two departments with very important roles and others with significant roles.

Role of the AECB

Power reactors have been regulated by the AECB since the 1950s. The role of the AECB has taken in that time has been to approve on-site plans and to require that there be off-site plans. An interesting situation could arise if the province was satisfied that a satisfactory off-site plan was in place and the AECB was not, eg. the AECB could refuse to issue an operating licence. This is one of the grey areas which arises in a federal-provincial system where health and safety, a

provincial responsibility overlaps the control of radioactive materials and nuclear processes, the responsibility of the AECB. The AECB has not in the time it has been regulation nuclear reactors produced any documents providing guidance in either on-site or off-site planning for reactor accidents which could have provided a basis for consistent plans. This is in contrast to what has taken place in the United States, where the Nuclear Regulatory Commission has produced several regulatory guides, which have been the basis of utility and state emergency response planning.

Federal Nuclear Emergency Response Plan (FNERP)

Following the Three Mile Island accident in March 1979 the Province of Ontario formally requested the federal government to take steps to provide a national focal point for nuclear emergencies. This resulted in a draft working plan being prepared in 1982. In 1984, Pierre Trudeau the prime minister of the time, designated the Minister of Health as the lead minister. The first Federal Nuclear Emergency Plan was published then. A revised plan was issued in October 1991 and distributed in February 1992.

The plan identifies responsibilities (called capabilities in the most recent plan) for the following fifteen federal agencies.

- Agriculture Canada
- Atomic Energy Control Board
- Atomic Energy of Canada Limited
- Communications Canada
- Emergency Preparedness Canada
- Employment and Immigration Canada
- Energy, Mines and Resources
- Environment Canada
- External Affairs
- Fisheries and Oceans
- National Defence
- Health and Welfare Canada
- Revenue Canada
- Solicitor General
- Transport Canada

Health and Welfare Canada is the coordinating agency for emergencies that are not related to nuclear weapons, The Department of Defence is responsible for response to emergencies involving nuclear weapons material. Responsibilities assigned to other agencies identified in the plan are similar to those that they normally hold.

In the FNERP, in addition to the responsibilities of federal agencies being outlined, the responsibility for off-site nuclear emergency planning is clearly stated as a provincial responsibility. However the plan also says that under

circumstances where coordination between provinces is required coordination of operations will be the responsibility of the Federal Nuclear Emergency Response Centre. This is a facility specified in FNERP.

In the FNERP the Minister of National Health and Welfare is identified as having the responsibility for coordinating federal response to a nuclear emergency and for fulfilling requests for provincial assistance. The Director Bureau of Radiation and Medical Devices is the Federal Coordinator. This is the person who effects the coordination responsibility. The plan also specifies 'Nuclear Emergencies Requiring Federal Action' These are

1. Federal assistance is requested by the designated official in a province.
2. More than one province may be affected.
3. The emergency originates inside Canada and may affect other countries.
4. The emergency originates outside Canada and may affect Canada.
5. An incident involving nuclear sources belonging to another country occurs on Canadian territory.
6. Nuclear weapons tests which may affect Canada.

Under the plan a Coordinating Group, an Operations Group, a Technical Advisory Group and a Public Information Group are established with functions and representation. These groups will provide the advice and expertise suggested by their names. Guidelines for the vitally important task of public communication are provided and a strategy for this is presented. Designated spokespersons are named and three **'Levels of Crisis**' are defined which range from no immediate risk (**Level One**) to one which has an immediate acute threat to health.

Communication with the Public

Communication with the public in a nuclear emergency is such a critical part of any planning that it requires discussion in more detail. If the accident leading to emergency action is serious and if there is a significant release of radioactive material to the environment then the communications media will be intensely interested in the incident and the actions being taken to protect the public. Plans should include:

- the provision of up to date accurate information on the incident,
- designation of official spokespersons,
- briefing of media personnel on the nature of the incident and status of the emergency,
- provision of premises for briefing including supplies of technical material on the unit involved,
- communication facilities for media personnel.

International Nuclear Event Scale

An important aspect of this communication process is bringing some perspective to the accident. Technical material should be prepared with this objective in mind. Conveying to lay people the extent or magnitude of an accident is one of the more difficult problems in this area. Recently the IAEA has developed a nuclear event scale, Figure 4. This is a scale going from 1 representing a slight anomaly to 7 representing a major accident. The underlying logic of the scale is shown in Table 2, and the application of the scale to past accidents is shown in Table 3. The scale is currently being applied internationally for a trial period of about a year. Adoption of the scale should simplify communication with the media and the public on this difficult topic.

Figure 4
International Nuclear Event Scale

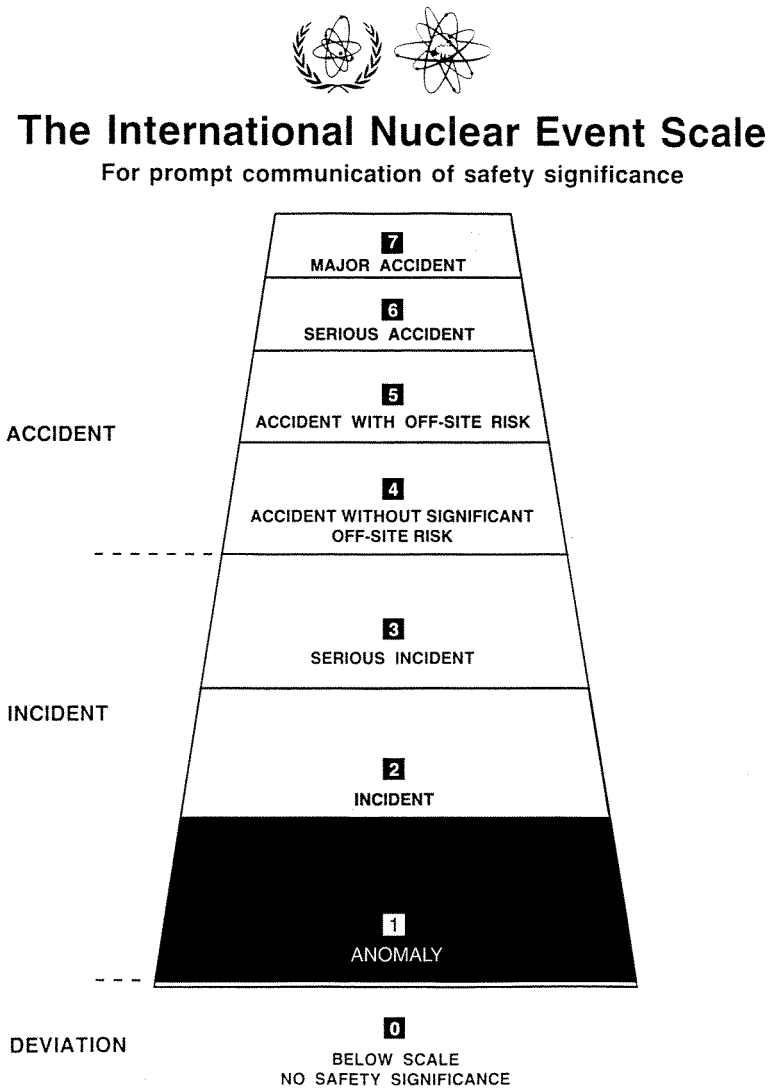


Table 2

Underlying Logic Of The Scale

BASIC STRUCTURE OF THE SCALE
(Criteria given in matrix are broad indicators only)
Detailed definitions are provided in the INES users manual

	CRITERIA OR SAFETY ATTRIBUTES		
	OFF-SITE IMPACT	**ON-SITE IMPACT**	**DEFENCE IN DEPTH DEGRADATION**
7 Major accident	Major release: widespread health and environmental effects		
6 Serious accident	Significant release: likely to require full implementation of planned countermeasures		
5 Accident with off-site risk	Limited release: likely to require partial implementation of planned countermeasures	Severe damage to reactor core/ radiological barriers	
4 Accident without significant off-site risk	Minor release: public exposure of the order of prescribed limits	Significant damage to reactor core/radiological barriers/fatal exposure of a worker	
3 Serious incident	Very small release: public exposure at a fraction of prescribed limits	Severe spread of contamination/acute health effects to a worker	Near accident- no safety layers remaining
2 Incident		Significant spread of contamination/ overexposure of a worker	Incidents with significant failures in safety provisions
1 Anomaly			Anomaly beyond the authorised operating regime
0 Below scale event deviation	No safety significance		
Out of scale event	No safety relevance		

Table 3

Past Applications Of Scale

LEVEL	DESCRIPTOR	CRITERIA	EXAMPLES
ACCIDENTS 7	**MAJOR ACCIDENT**	• External release of a large fraction of the radioactive material in a large facility (e.g. the core of a power reactor). This would typically involve a mixture of short and long-lived radioactive fission products (in quantities radiologically equivalent to more than tens of thousands terabecquerels of iodine-131). Such a release would result in the possibility of acute health effects; delayed health effects over a wide area, possibly involving more than one country; long-term environmental consequences.	Chernobyl NPP, USSR (now in Ukraine), 1986
6	**SERIOUS ACCIDENT**	• External release of radioactive material (in quantities radiologically equivalent to the order of thousands to tens of thousands of terabecquerels of iodine-131). Such a release would be likely to result in full implementation of countermeasures covered by local emergency plans to limit serious health effects.	Kyshtym Reprocessing Plant, USSR (now in Russia), 1957
5	**ACCIDENT WITH OFF-SITE RISK**	• External release of radioactive material (in quantities radiologically equivalent to the order of hundreds to thousands of terabecquerels of iodine-131). Such a release would be likely to result in partial implementation of countermeasures covered by emergency plans to lessen the likelihood of health effects.	Windscale Pile, UK, 1957
		• Severe damage to the nuclear facility. This may involve severe damage to a large fraction of the core of a power reactor, a major criticality accident or a major fire or explosion releasing large quantities or radioactivity within the installation.	Three Mile Island USA, 1979
4	**ACCIDENT WITHOUT SIGNIFICANT OFF-SITE RISK**	• External release of radioactivity resulting in a dose to the most exposed individual off-site of the order of a few millisieverts.* With such a release the need for off-site protective actions would be generally unlikely except possibly for local food control.	Windscale Reprocessing Plant, UK, 1973
		• Significant damage to the nuclear facility. Such an accident might include damage to nuclear plant leading to major on-site recovery problems such as partial core melt in a power reactor and comparable events at non-reactor installations.	Saint-Laurent NPP, France 1980
		• Irradiation of one or more workers which result in an overexposure where a high probability of early death occurs.	Buenos Aires Critical Assembly Argentina, 1983
INCIDENTS 3	**SERIOUS INCIDENT**	• External release of radioactivity above authorised limits, resulting in a dose to the most exposed individual off-site of the order of tenths of millisievert.* With such a release, off-site protective measures may not be needed.	
		• On-site events resulting in doses to workers sufficient to cause acute health effects and/or an event resulting in a severe spread of contamination for example a few thousand terabecquerels of activity released in a secondary containment where the material can be returned to a satisfactory storage area.	
		• Incidents in which a further failure of safety systems could lead to accident conditions, or a situation in which safety systems would be unable to prevent an accident if certain initiators were to occur.	Vandellos NPP Spain, 1989
2	**INCIDENT**	• Incidents with significant failure in safety provisions but with sufficient defence in depth remaining to cope with additional failures.	
		• An event resulting in a dose to a worker exceeding a statutory annual dose limit and/or an event which leads to the presence of significant quantities of radioactivity in the installation in areas not expected by design and which require corrective action.	
1	**ANOMALY**	• Anomaly beyond the authorised operating regime. This may be due to equipment failure, human error or procedural inadequacies. (Such anomalies should be distinguished from situations where operational limits and conditions are not exceeded and which are properly managed in accordance with adequate procedures. These are typically 'below scale'.	
BELOW SCALE/ ZERO	**DEVIATION**	No safety significance	

* The doses are expressed in terms of effective dose equivalent (whole body dose). Those criteria where appropriate can also be expressed in terms of corresponding
annual effluent discharge limits authorised by National authorities.