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BASIC COURSE FOR RADIATION SURVEYORS AND CONTAMINATION MONITORS OF THE RADIATION AND INDUSTRIAL SAFETY BRANCH

CRNL-281

by

J.H. FENN and W.R. BUSH

Chalk River, Ontario

September 1968

NOTE

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Radiation and Industrial Safety Branch

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ABSTRACT

The fundamentals of radioactivity and radiation protection are outlined. This course is given to all new Radiation Surveyors or Monitors of the Radiation and Industrial Safety Branch, at the Chalk River Nuclear Laboratories (CRNL).

The purpose of the course is to provide trainee Radiation Surveyors or Monitors with a basic knowledge of radioactivity and radiation protection prior to the commencement of on-the-job training.

Chalk River, Ontario
September, 1968

TABLE OF CONTENTS

Page

Page

APPENDICES

ILLUSTRATIONS

1. INTRODUCTION

1.1 Purpose

This course is designed to give new Radiation Surveyors and Monitors of the Radiation and Industrial Safety Branch, at the Chalk River Nuclear Laboratories (CRNL), a basic knowledge of the fundamentals of radioactivity and radiation safety. The information in this report is presented in approximately two weeks of lectures.

Following the completion of this course, trainees spend a year or more at on-the-job training with experienced Radiation Surveyors. During this time they become familiar with the various areas in which they may be required to work, and with the techniques they will apply as Radiation Surveyors or Monitors.

It is not possible to present a set of detailed instructions to cover all the situations that may confront a Radiation Surveyor or Monitor. However, a knowledge of the principles of radiation protection will

help him to make sound judgements when faced with special problems.

1.2 A Brief History of Radioactivity and Radiation Protection

1.2.1 Radioactivity

Some notable events in relation to ionizing radiation took place just before the turn of the last century. In 1895, Roentgen discovered X-rays; the following year Becquerel noted the natural radioactivity of uranium; and, in 1898, Pierre and Marie Curie found in uranium ore two new radioactive elements, radium and polonium.

In 1905, Einstein published his mass-energy equation which stated that if mass were converted to energy, the energy would be equal to the mass times the speed of light (186,300 miles per second) squared. Thus, the conversion of one pound of any material into energy would release 15 billion horsepower - hours.

In 1938, Hahn and Strassman accomplished the fission of uranium atoms by bombarding the metal with neutrons. This breaking of the atom resulted in the release of a tremendous amount of energy and also the

liberation of about 3 additional neutrons. These neutrons were ejected from the nucleus at high velocities and were available to produce further fissions, making possible a self-sustaining chain reaction. This reaction provided the means of obtaining power from the atom. The first chain reaction took place in the University of Chicago Pile on December 2, 1942. This feat led to the building of several reactors during World War II. In August, 1945, the first atomic bomb was detonated.

Following World War II, much research was devoted to the peaceful applications of atomic energy. New uses were found for radioisotopes in medicine and industry. In 1955, the United States launched the world's first atomic submarine, while in 1956 the first nuclear power station was operated at Calder Hall in Britain. Canada's first nuclear power station began producing electricity at Rolphton, Ontario on June 28, 1962.

1.2.2 Radiation Protection

The type of radiation associated with X-ray machines, radioactive materials, and accelerators is called ionizing radiation. Ionizing radiation consists of streams of fast flying particles or waves of energy, emitted by atoms. When radiation strikes matter it imparts energy to it. If the matter is living tissue biological changes may result.

The first reports of radiation injury date back to 1896. These were mainly skin burns on the hands of the early X-ray workers. Some of the injuries were serious, and in a few cases were fatal.

The next indication of danger came when Radiobiologists using X-rays to study the genetics of fruit flies found that many more mutations than normal were produced in flies exposed to excessive radiation. The implication of this discovery was that if X-rays could produce mutations in fruit flies, they could also produce mutations in man. For many years people involved with X-ray machines failed to regard this information seriously.

Another warning of the hazards of radiation was demonstrated by the watch dial painters. They had been in the habit of pointing their brushes with their lips, and continued to follow this practice after the introduction of radium into luminous paints. The radium from their lips was ingested, and some of it became fixed in their bones. Some years later, when a number of these workers had died, their deaths were attributed to radium-induced cancer of the bone.

Scientists realized that they must determine the amount of radiation that could be absorbed by the human body without causing any serious biological damage. In 1928, the "International X-ray and Radium Protection Commission" was established in Stockholm, Sweden. In 1950, the name was changed to the "International Commission on Radiological Protection" (ICRP), to more effectively cover the rapidly expanding field of radiation protection. Prior to World War II, the ICRP published recommendations at intervals of about three years. Since that time the ICRP have published recommendations frequently, at approximately two-year

intervals. These recommendations have been adopted by many countries as the basis of their legislation relating to radiation safety.

The Atomic Energy Control Act is the governing legislation in the field of atomic energy in Canada. The regulations thereunder are administered by the Atomic Energy Control Board. The Health and Safety Organization of Atomic Energy of Canada Limited is set forth in A.E.C.L. publication No. 1027 (fourth edition 1966):

> "Responsibility for the health and safety aspects of all operations of Atomic Energy of Canada Limited lies directly with management. Management is guided, however, by the advice of certain senior staff, divisions and committees, whose functions are to specify precautions to be taken."

Some of the advisory groups are:

- The CRNL Safety Committee
- The AECL Safety Committee
- The Review Committee on Reactor Safety
- The Criticality Panel
- The Waste Management Panel
- The Biology and Health Physics Division
- The Medical Division

The membership and functions of these groups are given in the appendices of AECL-1027.

The Radiation and Industrial Safety Branch of the General Services Division is charged with the administration of company policy with regard to the handling of radioactive materials at the Chalk River Nuclear Laboratories (CRNL). The various regulations that have been drawn up are contained in Standard Operating Policies and Procedures, and in the Radiation Safety Manual. These publications are available in branch offices at CRNL.

2. STRUCTURE OF THE ATOM

Atoms are too small to be seen and too light to be weighed individually. It would take 100 billion billion (10^{20}) atoms to cover the head of a common pin. Scientists use a unit called a MASS UNIT to compare atoms. This unit is 1/16 of the mass of an oxygen atom (it is 1.7×10^{-24} grams).

Atoms are composed of:

(a) A central portion called a nucleus, which is made up of two types of particle:

neutron (n) - no electrical charge

proton (p) - positive electrical charge

Each of these particles weighs 1 mass unit (approx.).

(b) An electron cloud surrounding the nucleus. An electron has one negative electrical charge and weighs only 1/1860 of a mass unit.

TABLE I

Name	Electrical Charge	Mass		Location
		Grams	Approximate Mass Units	
Electron	-1	9.1×10^{-28}	1/1860	orbits
Proton	+1	1.7×10^{-24}	1	nucleus
Neutron	0	1.7×10^{-24}	1	nucleus

2.1 Atomic Number and Atomic Mass

The number of protons in a nucleus determines the atomic number of an element. In a neutral atom, the atomic number also denotes the number of orbital electrons surrounding the nucleus. Elements are listed from 1 to 104 according to their atomic numbers (symbol Z).

The number of mass units in a nucleus determines the mass number of an element. This number is the sum of the protons and neutrons in a nucleus.

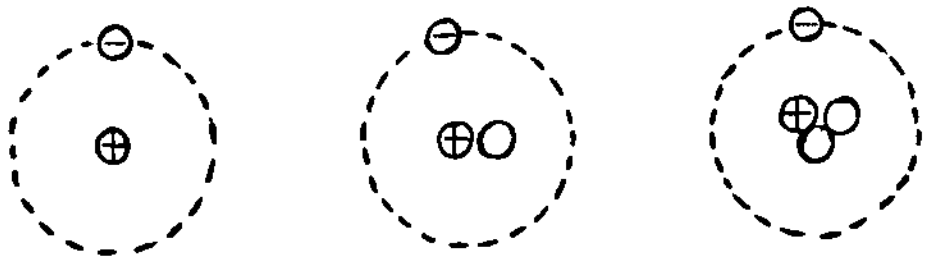
e.g. $^{238}_{92}U$ - contains 92 protons and 146 neutrons, so its atomic number is 92 and its mass number is 238.

$^{16}_{8}O$ - contains 8 protons and 8 neutrons, so its atomic number is 8 and its mass number is 16.

2.2 Isotopes

Elements with the same atomic number may have different mass numbers. The number of protons is the same but they have different numbers of neutrons in their nuclei. Elements of this type are called isotopes.

THE ISOTOPES OF HYDROGEN



Protium ($^{1}_{1}H$) Deuterium ($^{2}_{1}H$) Tritium ($^{3}_{1}H$)

FIGURE 1

Every element has three or more isotopes. Tin has 22 isotopes ranging from mass numbers 107 to 136.

2.3 Stable and Unstable Isotopes

Two forces are at work in the nucleus of an atom. A nuclear force tends to hold the neutrons and protons together. At the same time, electrical forces tend to force the protons apart. In some nuclei, the combination of neutrons and protons is such that these forces are well balanced. Isotopes with nuclei of this type are called stable.

In other nuclei, the combination of neutrons and protons may produce an imbalance of forces. These nuclei may undergo spontaneous rearrangement, ejecting charged particles to obtain a more stable combination

of protons and neutrons. Isotopes with nuclei of this type are called unstable. Their nuclei undergo radioactive decay. They are called radioactive isotopes or radioisotopes.

There are two types of radioisotopes:

(a) Natural - found in nature

(b) Artificial - produced in reactors, cyclotrons, betatrons, or other high-voltage accelerators.

Most radioisotopes, whether natural or artificial, decay by the emission of alpha or beta particles. A particular radioactive atom may emit either of these types of radiation, but cannot emit both types at the same time. However, different atoms of the same radioisotope may emit either alpha or beta particles. Gamma radiation may accompany either alpha or beta emission. Alpha, beta and gamma radiations are described in Section 3.

The energies of radiation are expressed in <u>electron volts</u>. An electron volt (eV) is the energy given to a single electron as it moves across a potential difference of one volt.

Energies of ionizing radiations such as alpha particles, beta particles and gamma photons may extend from a few thousand electron volts (keV) to several million electron volts (MeV). One million electron volts (1 MeV) is enough energy to lift a mass of one milligram one millionth of one centimeter, and is equal to 1.6×10^{-6} ergs.

3. RADIOACTIVITY AND NUCLEAR REACTIONS

3.1 Alpha Radiation (Symbol α)

An alpha particle is a high-energy charged particle consisting of 2 protons and 2 neutrons. It has a mass of 4 units and bears 2 positive electrical charges. Upon acquiring 2 electrons from its surroundings, it becomes a helium atom. Alpha particles may be symbolized as α, or ${}^{4}_{2}He$.

When an atom emits an alpha particle, 2 neutrons and 2 protons are lost from its nucleus. This reduces the atomic number by two and the mass by four. A new element called a daughter is formed.

e.g. $${}^{238}_{92}U \longrightarrow (\alpha \text{ out}) \longrightarrow {}^{234}_{90}Th$$

$${}^{216}_{84}Po \longrightarrow (\alpha \text{ out}) \longrightarrow {}^{212}_{82}Pb$$

The energy of alpha particles is high, ranging from 4 to 6 MeV. There is no external radiation hazard from alpha particles because their penetrating power is so low that clothing or the outer skin layer will stop them.

The hazard from these alpha emitters arises when they are deposited in the body. Most alpha emitters such as uranium, thorium, radium, plutonium and polonium are bone-seekers.

3.2 Beta Radiation (Symbol β)

A beta particle is produced when one of the neutrons in a nucleus changes into a proton and an electron. The proton remains in the nucleus, so there is no change in the mass number, but the electron, or beta particle, is emitted. Since the nucleus then has an additional proton, the atomic number has been increased by one.

e.g. $^{239}_{92}U \longrightarrow (\beta \text{ out}) \longrightarrow {}^{239}_{93}Np$

$^{239}_{93}Np \longrightarrow (\beta \text{ out}) \longrightarrow {}^{239}_{94}Pu$

During this process, another particle, called a neutrino is also emitted. It has negligible mass and no electrical charge. It accounts for the difference between the total available energy and the energy imparted to the beta particle.

The formation of a beta particle in this manner may be represented as follows:

$$ {}^{1}_{0}n \longrightarrow {}^{1}_{1}p + {}^{0}_{-1}\beta + {}^{0}_{0}\nu $$

Where ${}^{1}_{0}n$ is a neutron

${}^{1}_{1}p$ is a proton

${}^{0}_{-1}\beta$ is a beta particle

${}^{0}_{0}\nu$ is a neutrino

A nuclear proton may also be converted to a neutron. This results in the emission of a positron and a neutrino. This process may be represented as follows:

$$ {}^{1}_{1}p \longrightarrow {}^{1}_{0}n + {}^{0}_{+1}\beta + {}^{0}_{0}\nu $$

In this case, the mass remains the same, but 1 proton is lost, so the atomic number decreases by one. The energies of beta particles may range from a few KeV to several MeV. Most beta particles have sufficient energy to penetrate into the skin, and the lens of the human eye. High energy beta particles can penetrate an inch of lucite.

Although beta particles do not penetrate deeply into the human body, serious skin damage can result from contact with beta-emitting sources. Because beta particles can be shielded easily, and because of their relatively short range in air, they usually pose a greater problem internally than externally.

Natural radioisotopes are frequently alpha emitters, while artificial radioisotopes are usually beta emitters.

3.3 Gamma Radiation (Symbol γ)

Gamma radiation consists of high-energy photons, similar to light and radio waves, but of a much higher frequency. Gamma rays have no electrical charge, and are represented by the symbol γ.

Gamma rays possess extreme penetrating ability with ionization in tissue resulting from both the primary gamma and the secondary radiation caused by stopping the gammas.

Gammas will penetrate up to several feet of water or concrete. Heavy elements such as lead are used as shielding for gamma rays.

3.4 K-electron Capture

K-capture is the process whereby the nucleus absorbs an electron from one of its electron orbits, usually the inner or K-shell. The electron neutralizes one of the positive nuclear charges, and energy is given off by the emission of a gamma ray from the nucleus and an x-ray from one of the electron orbits.

In this case again, the mass remains the same, but the positive charge of the nucleus has been decreased by one.

Some radioactive materials have more than one decay scheme, and may decay by any one, or all of the above methods.

e.g.

$$^{65}_{30}Zn \longrightarrow \begin{cases} (\beta^{+}\ \text{out}) \longrightarrow {}^{65}_{29}Cu \\ (\beta^{-}\ \text{out}) \longrightarrow {}^{65}_{31}Ga \\ (\text{K-capture}) \longrightarrow {}^{65}_{29}Cu \end{cases}$$

3.5 Half-Life (T 1/2)

All radioisotopes decay at a rate which is determined by the atomic structure of the isotope. The decay rate cannot be changed by any known means. Decay rates are expressed according to a characteristic period known as the half-life, which is the time required for one-half of the atoms present to decay. This often provides the quickest way of identifying a radioisotope.

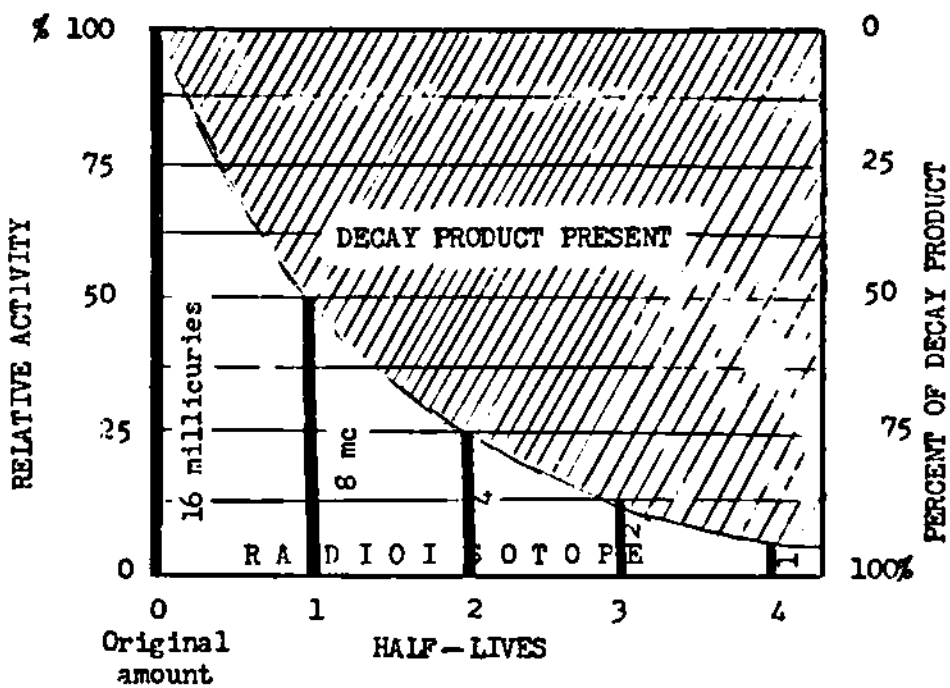


FIGURE 2

3.6 Neutrons

Neutrons may be produced by:

(a) Fission in a reactor.

(b) Van de Graaff accelerators, linear accelerators and cyclotrons.

(c) Alpha, neutron reactions (α, n): interaction of alpha particles from radium, plutonium, polonium or actinium with beryllium.

(d) Gamma, neutron or photoneutron reaction (γ, n): When a gamma ray of energy greater than 2.5 MeV is stopped by heavy water or beryllium, a neutron is emitted. This reaction contributes approximately 0.1% of the neutrons during the operation of heavy-water moderated reactors.

Shielding is accomplished by slowing down fast neutrons with hydrogenous materials such as paraffin wax, water, wood or Masonite. The slow neutrons can then be captured in materials such as cadmium or boron. Since gamma rays are produced when slow neutrons are absorbed in cadmium, gamma shielding may also be required. Neutrons are classified according

to their energies as follows:

thermal - around 0.025 eV;

epithermal - 0.1 eV to 100 eV;

slow - less than 100 eV;

intermediate - 10^2 to 10^5 eV;

fast - greater than 0.1 MeV.

3.6.1 Neutron Absorption

(a) Isotope Formation

When an element is placed in a neutron flux, the nuclei of some of its atoms may absorb a neutron. The mass number of these atoms is thereby increased by one and an isotope is formed (Section 2.2). If it is a radioisotope, it attempts to stabilize itself, usually by emitting a beta particle. The half-life for such beta emission varies with different isotopes from fractions of a second to billions of years.

e.g. $$^{238}_{92}U + ^{1}_{0}n \longrightarrow ^{239}_{92}U$$

$$^{239}_{92}U \longrightarrow (T_{1/2} = 23.5 \text{ min.}) \longrightarrow (\beta \text{ out})$$

$$\longrightarrow ^{239}_{93}Np$$

$$^{239}_{93}Np \longrightarrow (T_{1/2} = 2.35 \text{ days}) \longrightarrow (\beta \text{ out})$$

$$\longrightarrow {}^{239}_{94}Pu$$

(b) <u>Transmutation</u>

In some cases when a neutron is absorbed by a nucleus, a proton may be ejected. When this happens there is no change in mass, but one positive charge is lost from the nucleus. This results in a new element of atomic number one less than the original element.

e.g. $$^{14}_{7}N + {}^{1}_{0}n \longrightarrow {}^{14}_{6}C + {}^{1}_{1}H$$

In other cases an alpha particle may be ejected. This results in a reduction of the atomic number by 2 and of the mass by 4.

e.g. $$^{10}_{5}B + {}^{1}_{0}n \longrightarrow {}^{7}_{3}Li + {}^{4}_{2}He$$

In other cases, only γ radiation is released.

e.g. $$^{2}_{1}H + {}^{1}_{0}n \longrightarrow {}^{3}_{1}H + \gamma$$

4. NUCLEAR FISSION

4.1 History

Hahn and Strassman accomplished the fission of uranium in 1938 by bombarding uranium metal with neutrons from a Ra-Be source. Fermi had been fissioning uranium for some years but had not recognized the reaction that was taking place.

It was soon discovered that additional neutrons were released by fissioning uranium. It was thought if these neutrons could produce further fission, still more neutrons would be released and a self-sustaining chain reaction might be possible. It might also be possible to produce an enormously powerful fission bomb.

Great impetus was given to research by British and American scientists towards establishing the feasibility of a fission bomb because it was known that the Germans were actively studying the fission process.

Because of the war, British researchers moved to the United States where they were joined by other European scientists. A smaller project was started in Canada to study fission using heavy water as the moderator.

Success was achieved on December 2, 1942, when Enrico Fermi's graphite moderated reactor went critical at the University of Chicago. A controlled, self-sustained, fission chain reaction was demonstrated for the first time. Soon larger reactors were built to produce fissile plutonium, which could be separated from the uranium fuel. Other scientists solved the problem of separating fissile ^{235}U from natural uranium.

The first atomic bomb, using plutonium as the fissile material, was tested in the Nevada desert in July, 1945. This was followed on August 6, 1945, by a ^{235}U bomb which was detonated over Hiroshima, and on August 9, 1945, by a plutonium bomb over Nagasaki.

4.2 The Fission Process

When a neutron is captured by a nucleus of ^{235}U, ^{239}Pu, or other fissile material, the nucleus may split, or fission, into two new atoms called fission products. There are about thirty ways in which ^{235}U may split, yielding roughly sixty different fission products. These fission products range in atomic number from about 30 to 64 and in mass number

from 70 to 160. The greatest yield of fission products occurs in the vicinity of masses 95 and 140.

e.g. $$^{235}_{92}U + ^{1}_{0}n \longrightarrow ^{89}_{39}Sr + ^{144}_{54}Xe + 3^{1}_{0}n + 200 \text{ MeV}$$

Approximately 200 MeV of energy is released with each fission. The power production corresponding to the fission of one gram of uranium per day would be 0.96×10^3 kilowatts or roughly 1,000 kilowatts (1,000,000 watts or 1 megawatt). Uranium and other heavy metals may be fissioned by bombarding them with deuterons, protons, or alpha particles. The important distinction of fissioning ^{235}U with neutrons is that an average of 2.5 new neutrons are released during each fission, making a self-sustaining chain reaction possible.

Natural uranium is the fuel of choice in Canada. It consists of about 99.3% ^{238}U and 0.7% ^{235}U. (It also contains 0.0056% ^{234}U, which is generally ignored because of its insignificant concentration.) The probability of fissioning ^{238}U is practically negligible, which leaves the ^{235}U (only 0.7% of the total)

as essentially the only fissile component. When neutrons are slowed down (by a moderator) their probability of being captured in ^{235}U nuclei increases, and a self-sustaining chain reaction becomes possible. Two moderators commonly used, graphite and heavy water, can slow down neutrons without capturing too many and so permit a self-sustaining reaction with natural uranium.

4.3 Multiplication Factor and Criticality

Neutrons released in fission can cause new fissions, which release more neutrons which can cause more fissions, etc. However, not all the fission neutrons remain available to cause new fissions. Some of them escape from the edges of the fissile assembly and some are absorbed in non-fissile materials such as ^{238}U, steel, water or impurities that might be present. If a chain reaction is to be self-sustaining, (i.e. kept going solely by neutrons released in fission) at least one of the 2.5 (average) neutrons released in each fission must cause a new fission.

Some neutrons are always present in a fissile assembly, due to spontaneous fissions. These neutrons are multiplied by causing new fissions, which release neutrons and cause new fissions, etc. A "generation" is the time required for a neutron to be released in fission, slowed down, captured, and cause a new fission. That is, a generation is the average time between successive fissions in a chain reaction.

This process is depicted numerically in the following example:

Suppose that 1000 source neutrons enter a fissile assembly each generation. Source neutrons are those resulting from spontaneous fissions plus those released by the interaction of fission-product gamma rays with heavy water. Of these 1000 neutrons, suppose that 200 escape from the edges of the assembly, that 500 are absorbed in non-fissile reactions and that 300 cause fissions. An average of 2.5 neutrons are released per fission, therefore, of the 1000 neutrons present initially, 300 x 2.5 = 750 remain at the end of that generation.

The ratio of neutrons present at the end of a generation to those present at the beginning is called the multiplication factor, which is given the symbol k. In this example,

$$k = \frac{750}{1000} = 0.75$$

A fissile assembly having k less than one is not self-sustaining because the number of neutrons at the end of any generation is less than at the start of that generation. Such a system is said to be sub-critical. Nevertheless, multiplication of neutrons takes place, even in a sub-critical assembly, due to the continuous addition of source neutrons. At the start of the second generation, 1000 source neutrons are added to the 750 remaining from the first generation, for a total of 1750. If k remains at 0.75, the same fractions of these neutrons will escape or be otherwise lost as in the first generation, and 0.3 of them will cause new fissions:

No. of new fissions = 0.3 x 1750 = 525

No. of fission neutrons released = 2.5 x 525 = 1312

At the start of the third generation, 1000 source neutrons are added, for a total of 2312. Of these, 694 cause fissions which release 1735 neutrons, and so on. The results of continuing such calculations through several successive generations are tabulated in Table 2, following:

TABLE 2

Neutron Multiplication in a Sub-Critical Assembly (k = 0.75)

Generation	Neutrons per Generation			(d) No. of Fissions 0.3(c)	(e) No. of Fission Neutrons 2.5(d)
	(a) Source	(b) from Previous Generation	(c) Total (a) + (b)		
1st	1000	0	1000	300	750
2nd	1000	750	1750	525	1312
3rd	1000	1312	2312	694	1735
4th	1000	1735	2735	820	2050
5th	1000	2050	3050	915	2288
10th	1000	2775	3775	1132	2830
20th	1000	2987	3987	1196	2990
25th	1000	2997	3997	1199	2998
26th	1000	2998	3998	1199.4	2998.5
29th	1000	2999	3999	1199.75	2999.4
31st	1000	2999.5	3999.5	1199.85	2999.6

It is apparent from the above tabulation that the increase in fissions per generation decreases with each successive generation, until a constant number of neutrons per generation is approached. This constant value is equal to $\frac{s}{1 - k}$, where s is the number of source neutrons per generation. In the above example, $\frac{s}{1 - k} = \frac{1000}{1 - 0.75} = 4000$.

In other words, if k = 0.75, the sub-critical power is four times as high as the source power; that is, the fissile assembly has multiplied the source power by four.

If k is increased, by building a bigger fissile assembly (which results in relatively less leakage from the edges) or by adding a reflector (which scatters some of the escaping neutrons back into the assembly) or by decreasing the amount of non-fissile absorbing material (such as fuel cladding or control rods) the multiplication will increase according to $\frac{1}{1 - k}$. For example, if k = 0.9, the source power is multiplied by ten, and if k = 0.99, the source power is multiplied by one hundred.

The expression $\frac{s}{1 - k}$ can only be used when k is less than one, for if k equals one, $\frac{s}{1 - k}$ becomes infinite. In fact, when k = 1, the number of fissions in successive generations is sustained at a constant level by the fission chain itself, i.e., independently of the source neutrons. When such a self-sustaining chain reaction exists in a fissile assembly, the assembly is said to be critical.

If k is greater than one, the number of fissions increases in each successive generation and the fissile assembly is said to be supercritical.

When starting up a reactor, such as NRX or NRU, reactivity is increased by adding moderator (NRX) or by removing absorbers (NRU) until the reactor is supercritical. The power is then allowed to increase, at a predetermined, controlled rate, until it reaches the desired level, then reactivity is decreased until the reactor is just critical, and the power remains constant. Actually, if the power is to remain constant, the k factor for fission chain neutrons has to be held slightly below one, so that the source neutrons plus

neutrons from the fission process are just enough to sustain a chain reaction.

The core components of a reactor (fuel, moderator, control rods, experimental facilities, etc.) are rated according to their effect on the reactivity of that reactor. Ratings are given in milli-k units, where a milli-k is k divided by 1000. Reactivity changes may be positive, as in the case of a fuel rod, or negative, as in the case of a control rod or a neutron-absorbing sample rod. The milli-k rating is based on the effect in a particular reactor. It does not apply to the same element in a different reactor, or in a position of different neutron flux in the same reactor.

4.4 Fission Rate

Neutrons are released within 10^{-14} seconds of fission, making possible an extremely fast chain reaction, as in the atomic bomb. However, a bomb type reaction depends on certain minimum quantities of fission material such as U^{235} and Pu^{239} being brought together and the resulting reaction with fast neutrons releases energy at an explosive rate.

In reactors, the release of energy is controlled by spacing the fuel and using a moderator to slow down the fast neutrons. The moderator, if heavy water, serves as a coolant to remove the heat of fission from the fuel. With this arrangement, some of the neutrons are absorbed in the structural materials, the coolant, the ^{238}U component of the fuel and accumulated fission products. The remaining neutrons, in excess of those required to maintain power, are absorbed by control rods and not enough are left over to sustain a bomb like chain reaction with fast neutrons.

A chain reaction with slow neutrons is too slow to result in an atomic-bomb-type of explosion. It takes about 10^{-3} seconds to slow down (moderate) a fast neutron in a heavy water reactor. In other words, successive fissions in a given chain are at intervals of about 10^{-3} seconds, compared with 10^{-14} seconds in a fast-fission chain. Therefore, a chain reaction with slow neutrons will be 10^{11} times slower than is possible with fast neutrons. This would still be fast enough, under certain conditions, to produce

an explosion similar to an ordinary chemical explosion, but not fast enough to produce an explosion like an atomic bomb.

Although 10^{-3} seconds is an extremely long time compared to the generation time in a fast-fission chain, it is still short enough that extremely complex control devices would be required to keep a reactor under control. Fortunately, it happens that some of the short-lived fission fragments emit neutrons when they decay, an average of 13 seconds after fission. These delayed neutrons represent about 0.75% of the total neutrons produced in fission. Averaged with the 99.25% of the neutrons which have a 10^{-3} second life, they result in an overall average life of about 10^{-1} seconds. This is long enough to permit relatively easy control of the chain reaction.

If enough reactivity is added to a reactor to give a 0.75% increase in neutrons per generation, (i.e., 7.5 excess mK), the delayed neutrons will no longer be significant in the reaction. The effective life of the neutrons will then be 10^{-3} seconds, making the reaction impossible to control with a control

system designed for a 10^{-1} second neutron life. This condition is called "prompt critical", and would almost certainly result in melting of some reactor fuel. Once the fuel had melted, its geometry would no longer be favourable and the chain reaction would slow down or stop.

The effect of neutron lifetime on the rate of increase of power is illustrated in Table 3.

TABLE 3

Fissile Assembly	Duration of Average Generation Seconds	k	Excess Reactivity mk	Percentage Increase Per Generation	Time Rate of Increase
Reactor, supercritical but under control (supercritical on delayed neutrons)	10^{-1}	1.001	1	0.1	1% per sec.
Run-Away Reactor (prompt critical with slow neutrons)	10^{-3}	1.0075	7.5	0.75	1800 times per sec.
Bomb (prompt critical with fast neutrons)	10^{-14}	Assume 1.02	20	2	1 billion times in 10^{-12} sec.

5. RADIATION PROTECTION

5.1 Introduction

Before the first reactor was built (in 1942) the sources of harmful radiation were very few in number. X-ray machines had been in use for a few decades, and there were cyclotrons and Van de Graaff machines in some laboratories. In the whole world only about three pounds of radium had been refined.

Very few people were involved with radiation hazards and protection was left mainly in the hands of the individual scientist who was using radioactive materials or operating one of the radiation machines. Some individuals were injured seriously by radiation and a few lives were lost.

Now, atomic reactors produce tons or radioactive material every year. Radioisotopes are being used in ever increasing numbers in industry and in hospitals. New applications have been found for X-ray machines. The testing of nuclear weapons has given rise to radioactive fallout all over the globe and ionizing radiation has become of importance to every person on earth.

5.2 Biological Effects of Radiation

When ionizing radiation enters the cells of the human body, it causes a biological reaction which is always damaging to body cells or tissues. Many damaged cells will recover from radiation exposure, if the dose is not excessive. Damage to the genes, however, is permanent.

Those effects which damage an individual's body within his lifetime are called somatic effects, while those effects which are inherited as a result of damage to parental genetic material are called genetic effects.

5.2.1 Somatic Effects

Exposures to radiation may be of an acute or chronic nature. An acute exposure usually refers to a dose of radiation received within a 24-hour period. Some of the effects of acute exposure of the whole body to penetrating radiation are:

0 to 25 rem* - No detectable effects

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*rem - unit of biological dose (see Section 6.1)

25 to 50 rem - Possible blood changes, no serious injury

50 to 100 rem - Blood cell changes, some injury, no disability

100 to 200 rem - Injury, possible disability

200 to 400 rem - Injury and disability certain, death possible

400 to 600 rem - Fatal to 50%

600 rem or more - Fatal

Chronic exposure to high levels of radiation can result in ulcer-like skin burns, cloudiness of the eye (cataract), or in damage to the blood producing organs or intestines.

Another type of injury may develop years after a radiation exposure is received. Examples are cancer or leukemia. However, if radiation exposures are kept within permissible limits (Section 6.2.1), the probability of these effects occuring is low.

5.2.2 Genetic Effects

Genetic damage has no apparent effect on exposed persons, but is inheritable and may result in the children or later descendents of exposed persons being born with physical or mental defects. Genetic damage is permanent and is cumulative. That is, the greater the exposure accumulated by a person all through his life until the conception of children, the greater the probability of genetic damage. The radiation dose limits observed at CRNL are low enough that the amount of genetic damage we pass on to our descendents is negligible when compared with the benefits they will receive from such developments as nuclear power plants or radioisotopes for use in medicine and industry.

5.3 Radiation Exposures

5.3.1 Types of Exposures

Personnel who work around reactors or other areas where radioactive materials are present have to be protected against both external and internal exposures.

External exposure arises from a source located outside the body.

Internal exposure arises from a source located within the body. Internal contamination can result from entry of radioactive materials into the body by:

(1) ingestion into the stomach, and subsequent absorption into the body;

(2) inhalation into the lungs, and subsequent absorption into the body;

(3) absorption through healthy skin;

(4) absorption through cuts, scrapes or abrasions of the skin.

5.4 Hazardous Areas at CRNL

The greatest concentration of radioactive materials is found within the NRX and NRU reactors, hence these are potentially the most hazardous areas. Sometimes fuel elements from the reactors, or other irradiated materials are transferred to the caves in Building 234, or Building 375 for examination, and, at times, these areas can also be quite hazardous.

Any laboratories working with plutonium (Pu) or strontium (Sr) or any other of the more dangerous radioisotopes will require special safety precautions.

5.5 External Radiation Hazards

5.5.1 Beams and Solid Sources

External radiation exposures at CRNL usually arise from sources such as:

(1) open experimental or fuel holes in reactors;

(2) anything that is removed from a reactor;

(3) fuel and other irradiated materials in fuel storage bays or hot cells (caves);

(4) equipment and waste from storage bays and hot cells;

(5) contamination around storage bays, hot cells and reactors;

(6) piping systems in reactor and chemical plant buildings;

(7) radioisotopes in laboratories;

(8) accelerator beams and targets.

Although the radiation hazards associated with reactors and accelerators are usually greater when these devices are operating, dangerous levels of

radiation can also be encountered when they are shut-down. Many areas which are inaccessible during operation may be opened for maintenance work during shut-down periods.

5.5.2 Radioactive Gases

External exposures may also be received from radioactive gases that occasionally escape from reactors or reactor fuels. The most common gases are argon-41 and the fission-product kryptons and xenons. Only traces of these gases are absorbed into the body, and since they are chemically inert, they are not retained. Therefore, they pose mainly an external radiation hazard. However, radionuclides that present an inhalation hazard, such as radioiodine, could be present in the air at the same time as these gases, therefore respirators are generally worn when workers are exposed to high concentrations of radioactive gases (Section 5.6.3.2).

The common fission-product gases are radioisotopes of krypton and xenon. They may be evolved from damaged fuel after it is removed from a reactor,

or they may accompany leaks of heavy water or helium from reactor systems. These short-lived fission-product gases are usually identified by the kryton daughter product rubidium-88 which has an 18-minute half-life.

Argon-41 is produced when air is irradiated in reactors, either in rod assemblies or in various air-cooled parts of reactors. Argon-41 has a half-life of 1.8 hours. It occasionally leaks from reactor systems and produces an external hazard which is usually not serious.

Under certain weather conditions, argon-41 descends from the main stack and blankets the entire CRNL area. Although the concentration is too low to pose a significant hazard, it is high enough to raise the background radiation level and so interfere with radiation measurements.

5.5.3 Precautions Against External Radiation Hazards

Three general types of precaution are taken in order that men may work safely in areas where alpha, beta, gamma or neutron sources are present:

(1) time - the longer the time spent near a radiation source, the greater will be the exposure. Therefore, if it is necessary to work where radiation is present, the simplest way to limit the exposure is to stay as short a time as possible.

(2) distance - exposure rate decreases rapidly with distance from a source. To reduce radiation exposures all work should be carried out as far as practicable from sources of radiation.

The intensity of gamma radiation decreases as the inverse square of the distance from a source. Doubling the distance from a gamma source decreases the dose rate to one quarter, while tripling the distance reduces the dose rate to one ninth, and so on. The intensity of beta radiation decreases even faster with distance because some of the beta particles are absorbed by the air.

(3) shielding - shielding can be used to reduce or eliminate radiation.

Alpha radiation is so easily shielded by skin, rubber gloves, paper or clothing that it presents no external hazard. (See Section 9.5.1).

Beta radiation is more difficult to shield than alpha, but may be shielded by plywood, asbestos, lucite, or aluminum. (See Section 9.5.2).

Gamma radiation is very penetrating, and requires thick shields of dense materials such as lead, steel, or concrete. Water is often a convenient shield, but the thickness required for a desired shielding is about ten times more than if lead is used. (See Section 9.5.4).

Neutrons are often shielded by mixtures of heavy and light materials, such as sandwich-type blocks made up of alternate layers of steel and masonite. The light material will slow down the neutrons and they are then captured in the steel. The steel will also shield secondary gamma radiation. Light materials such as parawax can be used alone if the neutron intensity is low. The hydrogen atom in the wax will slow down neutrons,

It also has an appreciable cross section to finally capture the neutrons. (See Section 9.5.3).

5.6 Internal Radiation Hazards and Contamination

5.6.1 Contamination

Contamination is uncontained radioactive material in a location where it is not wanted. It may be encountered in several forms:

(1) as visible or invisible dust on surfaces or in the air;

(2) as visible or invisible moisture on surfaces;

(3) as visible or invisible gas or vapour in air;

(4) as visible or invisible smoke or fumes.

Radioactive contamination can be encountered at concentrations high enough to pose an extremely dangerous external radiation hazard due to its beta gamma radiations. However, it is not hazardous at the levels usually encountered unless it enters the body. Once inside, it irradiates all or part of the body until it decays or is excreted. Such internal irradiation may continue for a long time (days or years), therefore amounts of contamination that are too small to present a significant external radiation hazard can be extremely

hazardous if they can enter the body. Furthermore, contamination that emits only alpha particles or low-energy beta particles is completely harmless until taken into the body; then it can be extremely hazardous.

5.6.2 Entry Into the Body

(1) Inhalation

(2) Ingestion

(3) Skin Absorption

Contamination may be inhaled into the lungs when we breathe air that contains radioactive dust, gas, vapour, fumes or smoke. Dangerous quantities could be inhaled even when the airborne contamination is completely invisible.

Some inhaled substances are completely absorbed through the lungs into the body; some are carried slowly out of the lungs by mucous and then swallowed and excreted; some are exhaled immediately after inhalation; and, some remain in the lungs.

Inhalation of radioactive contamination is prevented by wearing respirators in contaminated areas.

Contamination may be ingested (swallowed) if articles such as pipes, cigarettes or fingers become contaminated and then touch the lips or are put in the mouth. Large amounts of contamination could be ingested if radioactive liquids were pipetted using the mouth rather than a mechanical device for suction.

Some ingested materials are absorbed completely into the body via the digestive system, whereas others are partially absorbed and partially eliminated in the feces.

Ingestion is prevented by good habits such as not smoking in contaminated areas, not brushing contaminated hands across the face, and washing before eating.

Although the skin is an excellent barrier for most substances, some materials are absorbed through the skin into the body. The most notable example is tritium (Section 5.6.5). Contamination contained in solvents may also soak into the skin and be absorbed.

The skin can be damaged enough by harsh cleaning to permit absorption of contamination. Radioactive material can also enter the body through skin that is damaged by cuts, scratches or scrapes.

Absorption of contamination through the skin is prevented by wearing gloves and protective clothing to keep contamination off the skin. When the skin does become contaminated, it should be washed with soap and water.

5.6.3 Fission Product Contamination

Hundreds of different fission products are produced from uranium and plutonium in nuclear reactors. For simplicity, two of the more hazardous fission products, strontium-90 and iodine-131, are often the only ones considered. If these are controlled to safe levels, then the less hazardous fission products will also be under control.

5.6.3.1 Strontium-90

Strontium-90 is one of the most hazardous components of fission-product contamination. Strontium is chemically similar to calcium, which is one of the

major components of bones. Therefore, if strontium is taken into the body, by inhalation or ingestion, some of it finds its way into the bones. It remains with a half-life of about 16 years, and irradiates the bones all the while.

Strontium contamination, which is usually referred to as "fission-products" contamination, is found in fuel storage and hot cell areas. Inhalation of fission products is prevented by wearing respirators in contaminated areas, and ingestion is prevented by washing before eating and by keeping contaminated objects away from the mouth.

5.6.3.2 Radioiodine

Several isotopes of iodine are produced from uranium and plutonium in reactors. The most hazardous of these is usually iodine-131. Iodine may be released to the air from irradiated fuel if the fuel cladding is damaged. The iodine hazard is particularly severe with fuel that has been irradiated recently. Since iodine-131 decays with an 8-day half-life, the hazard decreases rather quickly with time after irradiation. For example, the amount of iodine-131 in fuel

40 days (five half-lives) after irradiation is only 1/32 of the amount present on the day of removal from a reactor. If iodine-contaminated air is breathed, iodine is absorbed into the body through the lungs. Some of it is then excreted with urine, and some of it is concentrated in the thyroid gland. It irradiates the thyroid gland during its decay period (8-day half-life).

Inhalation of iodine is prevented by wearing a respirator that contains charcoal, or an air-supplied respirator. The canisters of the army type and comfo type of filter respirators used at CRNL contain charcoal, and are therefore suitable for protecting against iodine.

5.6.4 Plutonium

Plutonium-239 is the most common isotope. It emits alpha particles (and very weak gamma rays) and does not constitute a significant external hazard. The principal hazard from plutonium arises when it enters the body, by inhalation or through a break in the skin. It then concentrates in bones, where it

remains virtually for life. It is possible to take in enough plutonium to result in bone fractures or cancers.

Plutonium is handled at CRNL in glove boxes, for example, in the plutonium laboratory in Building 375. It may be in the form of metal, oxide, or solution.

It can escape from glove boxes and become dispersed in the air if a glove is torn or punctured. Extremely dangerous amounts could be dispersed into the air in the event of a fire or explosion. In such cases, one should never approach the contaminated area without wearing a well fitted respirator.

5.6.5 Tritium

The most common internal contaminant at CRNL is tritium, which is a radioactive isotope of hydrogen. Tritium exists in only minute traces in nature, but large amounts are formed when heavy water is irradiated in the NRX and NRU reactors. Tritium

does not emit gamma radiation, and its beta particles do not have enough energy to penetrate the skin. Therefore, tritium is hazardous only when it is inside the body.

The water we drink is a compound of natural hydrogen and oxygen (H_2O). Heavy water is a compound of deuterium and oxygen (D_2O). When heavy water is irradiated by neutrons, some of the deuterium is converted to tritium, resulting in tritiated water or tritium oxide (DTO or T_2O). If heavy water is exposed to the atmosphere after irradiation, some of the tritiated water evaporates and disperses into the surrounding air as tritiated water vapour.

When a person is exposed in an atmosphere containing tritiated water vapour, there are two ways in which the radioactive vapour can enter his body. It may be inhaled and absorbed through the lungs, or it may be absorbed through the skin. The intake by these two methods is approximately equal.

Tritium is also absorbed rapidly when the skin is wet with tritiated heavy water. In addition, when tritiated heavy water is swallowed, it is absorbed completely into the body.

After being absorbed, through the skin, lungs or digestive system, tritiated water is distributed throughout the body. It reaches all parts of the body and does not concentrate in any particular organ. The biological half-life of tritiated water, which is the time taken by the body to eliminate half its content of tritium, is about ten days. This half-life can be reduced by increasing the intake of water or other fluids.

Potential tritium hazards exist at CRNL in all locations where heavy water is present. These are mainly in the NRU and NRX reactors, in Buildings 102 and 210 where heavy water is reconcentrated by electrolysis, and in Building 225 where heavy water is purified by ion exchange and

evaporation. A tritium hazard may be anticipated whenever opening tanks, drums, piping, or other system components that have contained heavy water. The helium systems associated with the NRU and NRX reactors also contain high concentrations of tritium.

Inhalation of tritium is prevented by wearing an air-supplied mask or hood. Absorption through the skin is prevented by wearing a plastic suit, hood, and gloves.

6. THE MEASUREMENT OF RADIATION EXPOSURES

6.1 Radiation Units

Several units are used for measuring radiation, including the Roentgen, rep, rad, and rem. The differences between these units are outlined below.

The Roentgen was first defined in 1928 with reference to the free-air chamber: A narrow, parallel (collimated) beam of X-rays was passed through the air between two large parallel metal plates. A narrow strip of one plate was insulated from the rest of the plate so that all the ions produced in a narrow rectangular volume of air between the plates could be collected by the insulated section and measured. The intersection of the X-ray beam with the collecting volume defined a volume of air in which ions were being produced by the X-rays. All the ions produced in this volume were collected and measured. The ion current produced by X-rays in a given volume of air is proportional to the amount of X-radiation. Thus, a quantity of X-radiation can be defined in terms of the volume of air and the ion current resulting from the action of X-rays on that air.

The Roentgen (R) was so defined. It is that quantity of x-radiation (or γ radiation) which will produce in 1 cm^3 of air (at S.T.P.) ionization equivalent to 1 e.s.u. (electrostatic unit) of charge of either sign. (One e.s.u. is the charge of about 2 billion electrons.) In other words, one R of X- or γ-rays will produce 2 billion ion pairs in every cm^3 of air through which it passes. In so doing, it delivers 83.8 ergs of energy to each gram of air. (One cm^3 of air at S.T.P. weighs 0.001293 grams.)

The radiologists were happy to have the Roentgen as a standard unit with which they could compare doses and results of various irradiations, but the physiologists were not content. They were beginning to explain various biological effects of radiation in terms of the amount of ionizing energy absorbed in tissue, whereas the Roentgen is a measure of the energy absorbed in air. One R imparts about 84 ergs to a gram of air, but 93 ergs to a gram of soft tissue, and even more to bone.

A new unit was needed. Therefore, the rep (Roentgen equivalent physical) was defined as that quantity of ionizing radiation which would impart 93 ergs to a gram of tissue. For various reasons, the definition of the rep was inaccurate. Alternative values were offered, but there was much disagreement as to which value was best. A better unit was obviously needed. "Ergs per gram" was suggested as a unit, however, it was desirable to adopt a unit which was similar in magnitude to the now familiar Roentgen and rep.

The solution came in 1953, when the rad (Roentgen absorbed dose) was defined. One rad is the amount of radiation that will impart 100 ergs to a gram of matter. The rad applies to any ionizing radiation and to any material, and is the presently accepted unit for measuring an absorbed dose of radiation.

The problem of units was still not completely solved for the biologists. Equal rads of different types of radiation, or radiation of different

energies, produces different biological effects. For example, 1 rad of the extremely low energy beta radiation from tritium is 1.7 times as damaging as 1 rad of gamma radiation. To provide a common basis for comparison, the RBE (relative biological effectiveness) was defined. It is the rads of X-radiation giving a certain effect divided by the rads of a different radiation required to give the same effect.

The product of RBE and rads defines a new unit, the rem (Roentgen equivalent man). Rems from different types of radiation can be added to give an estimate of the total biological effect of exposure to mixed radiation.

RBE is now reserved for radiobiology to describe a particular effect of radiation on some particular tissue. It has been replaced in radiation protection work by the quality factor (Q.F.). This is the linear-energy-transfer-dependent factor by which absorbed doses are to be multiplied to obtain a quantity that expresses on a common scale for all ionizing radiations the irradiation incurred by

exposed persons. A further modifying factor, the distribution factor (DF) is used in the case of non-uniformly distributed internally deposited radionuclides when calculating internal doses. The product of absorbed dose and modifying factors is called the dose equivalent (DE). e.g. (DE) is the product of absorbed dose, D, quality factor (QF), dose distribution factor (DF) and other necessary modifying factors.

$$(DE) = D(QF)\ (DF)....$$

The unit of dose equivalent is the "rem". The dose equivalent is numerically equal to the dose in rads multiplied by the appropriate modifying factors.

In practice the Q.F. for X and gamma rays is taken as unity, and for beta particles it is only greater than unity at very low energies. For fast neutrons having energies up to 10MeV the QF is taken as 10 for all body tissues except the lens of the eye. Here a Q.F. of 30 is used.

For internal exposure QF should be taken as 1 for β-, β+, gamma- and X-radiation and conversion electrons (except in the case of β-, β+, and e- radiations with maximum energy Em ≤ 0.03 MeV, for which the Q.F. is taken to be 1.7 (e.g. tritium), 10 for alpha particles and 20 for fission fragments and for nuclei recoiling during alpha emission.

(See the Recommendations of the International Commission on Radiological Protection - ICRP Publication 6 - revised 1962, for additional information regarding QF).

6.2 Maximum Permissible Doses (MPD)

Maximum permissible doses for both external and internal radiation are based on the assumption that a man will spend his working lifetime (50 years) exposed to radiation or radioactive materials. MPD's are set so that no apparent physical damage will result during the worker's lifetime, and so that genetic damage will be negligible.

Early MPD's were based on experience with X-rays and radium, and were set at a fraction of the measured doses which had resulted in barely observible damage.

Some of the MPD's were overly restrictive and have been lowered as more information has become available through experience gained in working with radioactivity. Our MPD's at CRNL are based on the Canadian Atomic Energy Regulations which in turn are based on the recommendations of the International Commission on Radiological Protection.

6.2.1 CRNL MPD's for External Radiation

(a) Total Body (blood-forming organs, gonads and lenses of the eyes)

0.6 rem (600 mrem) in 2 weeks

3 rem (3,000 mrem) in 14 consecutive weeks

5 rem (5,000 mrem) in 1 calendar year

(b) Skin

1.6 rem (1,600 mrem) in 2 weeks

8 rem (8,000 mrem) in 14 weeks

30 rem (30,000 mrem) in 1 year

(c) Extremities (hands, forearms, feet and ankles)

4 rem (4,000 mrem) in 2 weeks

20 rem (20,000 mrem) in 14 weeks

75 rem (75,000 mrem) in 1 year

The maximum permissible accumulated whole body occupational dose in rem at any age greater than 18 is calculated from the formula:

$$MPD = 5(N-18)$$

where N is the age in years.

6.2.2 CRNL MPD's for Internal Radiation

(a) Blood-Forming Organs and Gonads

The recommended permissible dose is the same as the recommended external dose for the total body (see Section 6.2.1 (a)).

(b) Thyroid Gland

The recommended permissible dose is the same as the recommended external dose for the skin (see Section 6.2.1 (b)).

(c) All Other Organs or Tissues

The recommended permissible dose for each organ singly is:

0.3 rem (300 mrem) in 1 week

4 rem (4,000 mrem) in 14 weeks

15 rem (15,000 mrem) in 1 year

Certain adjustments may be required to conform with lower permissible limits now recommended for some organs, and when isotopes are distributed throughout the whole body, or when several isotopes are present simultaneously each concentrating significantly in a different organ. Because of the absence of factual data concerning the effects of radioactivity on these organs, it is recommended that the previous maximum permissible limits for some organs (blood-forming organs, gonads, and lens of the eye) can be retained for the other organs also, that is, 0.3 rems/week and what may be regarded as a "whole body" exposure.

However, the limit is now expressed in terms of 13 consecutive weeks, which makes it 4 rems, in round figures, with an annual accumulated dose of 15 rems.

7. PERSONNEL MONITORING

Personnel monitoring is the measurement and recording of radiation exposures received by individual employees. Since some of the effects of radiation may not manifest themselves for several years after exposure, it is essential that an accurate record of all exposures to radiation should be maintained.

7.1 Measuring External Exposures

7.1.1 Film Dosimeter

One of the effects of ionizing radiation is that it will darken a photographic film. The degree of darkening depends on the intensity of the radiation and the total time of exposure. All CRNL employees wear a film, while at work, to measure their radiation exposures. The films are normally developed and read every two weeks, but they can be developed within a few hours on special request if a high exposure is suspected.

The CRNL film holder also contains other devices for measuring radiation exposures, as shown in Figure 3. The film package contains two films.

The more sensitive film can indicate from 20 mR to 20 R of gamma ray exposure and from 20 mrad to 20 rad of beta exposure. The less sensitive film can indicate from 5 R to 1000 R of gamma exposure and from 10 rad to 1000 rad of beta exposure.

The photobadge should be worn at all times with the picture side out, on the trunk of the body where the radiation dose rate is expected to be highest. In some situations, it is desirable to wear films in several positions on the body. Holders are available for films to be worn on the head, chest, waist, wrist or finger.

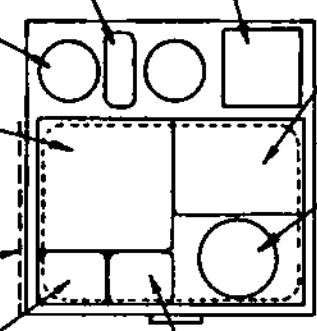
FIGURE 3

CRNL PHOTOBADGE

- Beta and gamma exposures are measured routinely once every two weeks.
- Fast and slow neutrons are measured when such exposures are suspected.
- Alpha radiation and the radiation from tritium are not detectable with the photobadge.

FILMS AND FILTERS

High and low range films in a single packet are sandwiched between similar sets of filters in the front and back of the film badge. Different areas of the film lie between filters of different metals. This results in different degrees of darkening of different areas of the film, which can be interpreted in terms of the amount of exposure from different types of radiation.



Phosphate glass (wrapped in Pb) for γ doses > 10 R

Indium foil, measures 1 to 3 MeV neutrons

Sulphur tablet

measures neutrons >2.5 MeV

Lead (0.025" Pb)

the gamma exposures reported routinely are measured on the part of the film between the lead filters.

Lead shields, in sides of badge cover, filter radiation from the sides.

Slow Neutrons (Cd and Sn)

Cadmium (0.040" Cd)

Slow neutrons will not register on the film, but they interact with Cd to produce γ rays, which do register on the film. Thus, the exposure from slow neutrons plus γ rays from other sources is registered on the film between the Cd filters.

Tin (0.040" Sn)

The film between the Sn filters gives the same response to γ as the film between the Cd, but does not respond to slow neutrons. This exposure, subtracted from the exposure between the Cd, leaves the exposure due to slow neutrons.

Aluminum (0.080" Al) - helps differentiate between low-energy X or γ radiation and β radiation.

Window ("beta-window" or "open-window") - low-energy X, γ and β radiation are registered on the film between the windows, but not on the film between the filters. (The front window is covered by the photograph.) Subtraction of the high-energy X or γ, measured between the Pb filters, leaves the dose received from low-energy X or γ, and from β radiation.

With the film in the correct position, the green tab can be seen through the beta-window hole in the rear cover.

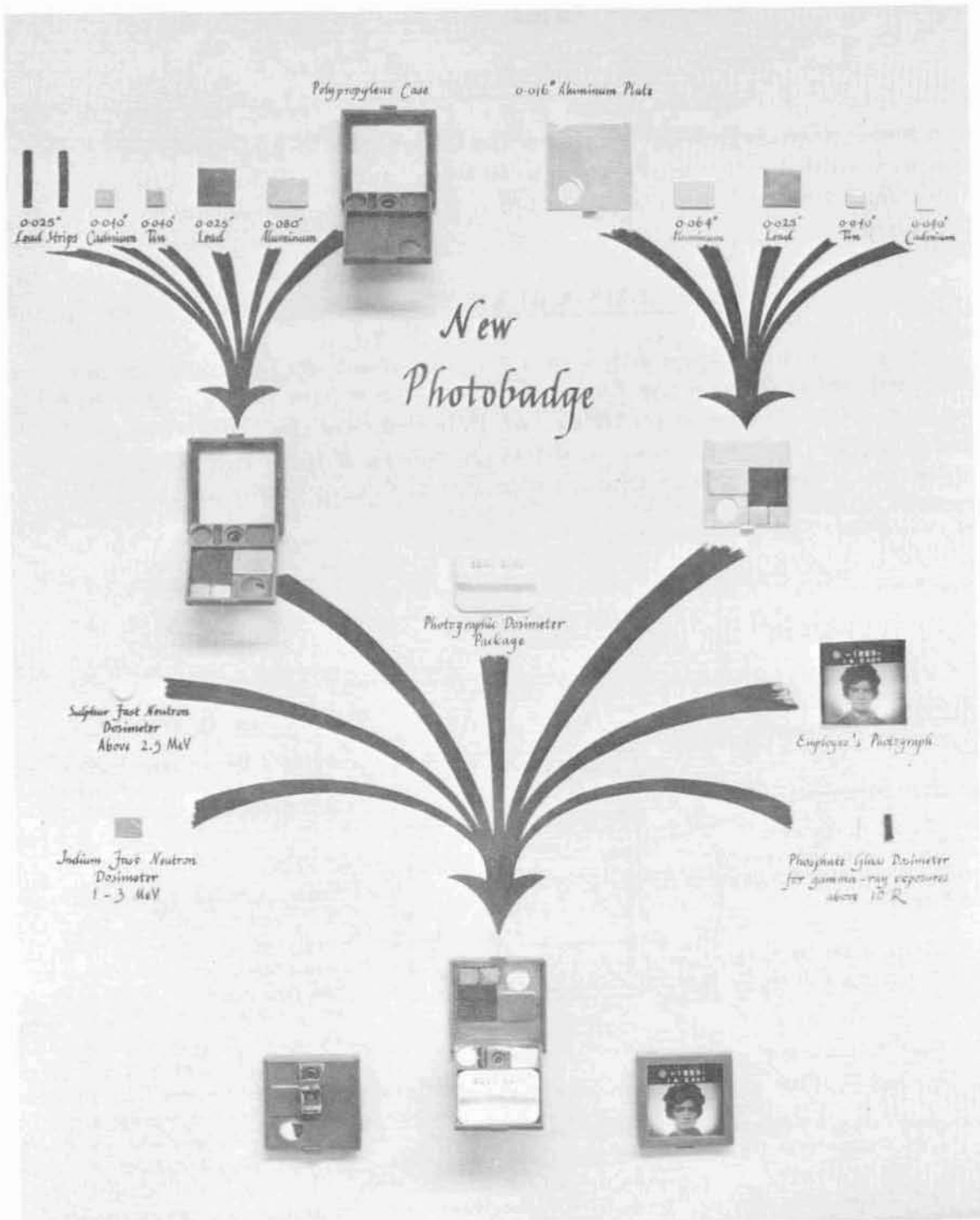


FIGURE 4
The CRNL Photobadge

7.1.2 Pocket Ion Chamber Dosimeter (Pencil Chamber) (Figure 5)

Pocket ion chamber dosimeters are frequently used to provide an instant indication of the gamma exposure. They should be worn in the same location as the film. Although the wearer may check his exposure at any time during the day, the dosimeter should be returned to the R & IS Branch Office at the end of the shift or work period so that the reading can be recorded. Over-exposures can sometimes be averted if a complete record of these readings is kept.

7.1.3 Miniature Warning Dosimeter (AEP 2165A) (Figure 6)

In areas where a high risk of radiation exposure exists, workers are provided with a miniature warning dosimeter. This device gives an audible clicking sound in a gamma radiation field. As the intensity of the field increases, the frequency of the clicks increases. When a dose of 200 mR has accumulated, a steady alarm signal is heard.

7.1.4 Thermo Luminescence Dosimeters (Figure 7)

Finger films enclosed in a plastic or rubber ring were used until recently to measure finger doses. They were awkward to wear, and the information obtained from them - a gamma reading only - was not too satisfactory. Developments in thermoluminescence dosimetry have provided an attractive alternative to the finger film for assessing extremity doses.

Dosimeters are available in the form of thin teflon discs (12.5 mm diameter and 0.4 mm thick) containing a uniform dispersion of finely-ground lithium fluoride (LiF) powder. The discs are enclosed in thin polyethylene satchets and can be attached to the skin by adhesive tapes provided with each dosimeter. The dosimeters are issued and worn in pairs and are numbered consecutively. A coded series of dots serves to identify each disc. The odd numbered disc should be worn on the left hand, and the even numbered disc on the right hand. In order to measure the maximum dose the dosimeters should be worn on the ball of each thumb unless circumstances dictate otherwise. Care

should be taken to avoid overlapping the dosimeter with adhesive tape.

When a thermoluminescent phosphor is exposed to ionizing radiation, free electrons are trapped in the crystal lattice. The excited electrons remain trapped for appreciable lengths of time if the phosphor is kept at room temperature. If the temperature is raised, the electrons leave their traps and recombine with positive holes, with the emission of light. The sum of light emitted serves as a measure of radiation dose or exposure.

Lithium fluoride (LiF) is relatively independent of the energy of incident gamma radiation so does not require the use of energy correction filters. The absence of filters makes LiF most suitable for personal dosimetry involving both beta and gamma radiation.

The dosimeters are capable of measuring a combined beta-gamma dose over a range from about 20 mrad to about 10^5 rad. Accuracy at a dose of 1 rad is better than ± 10%. Dosimeters should be stored in

a low background area if lower doses are expected. After dosimeters have been read, they can be annealed and re-used.

(See Appendix I, page 218, for an explanation of T.L.D. number coding.)

FIGURE 5
Pocket Ion Chamber Dosimeters

FIGURE 6
Miniature Warning Dosimeters

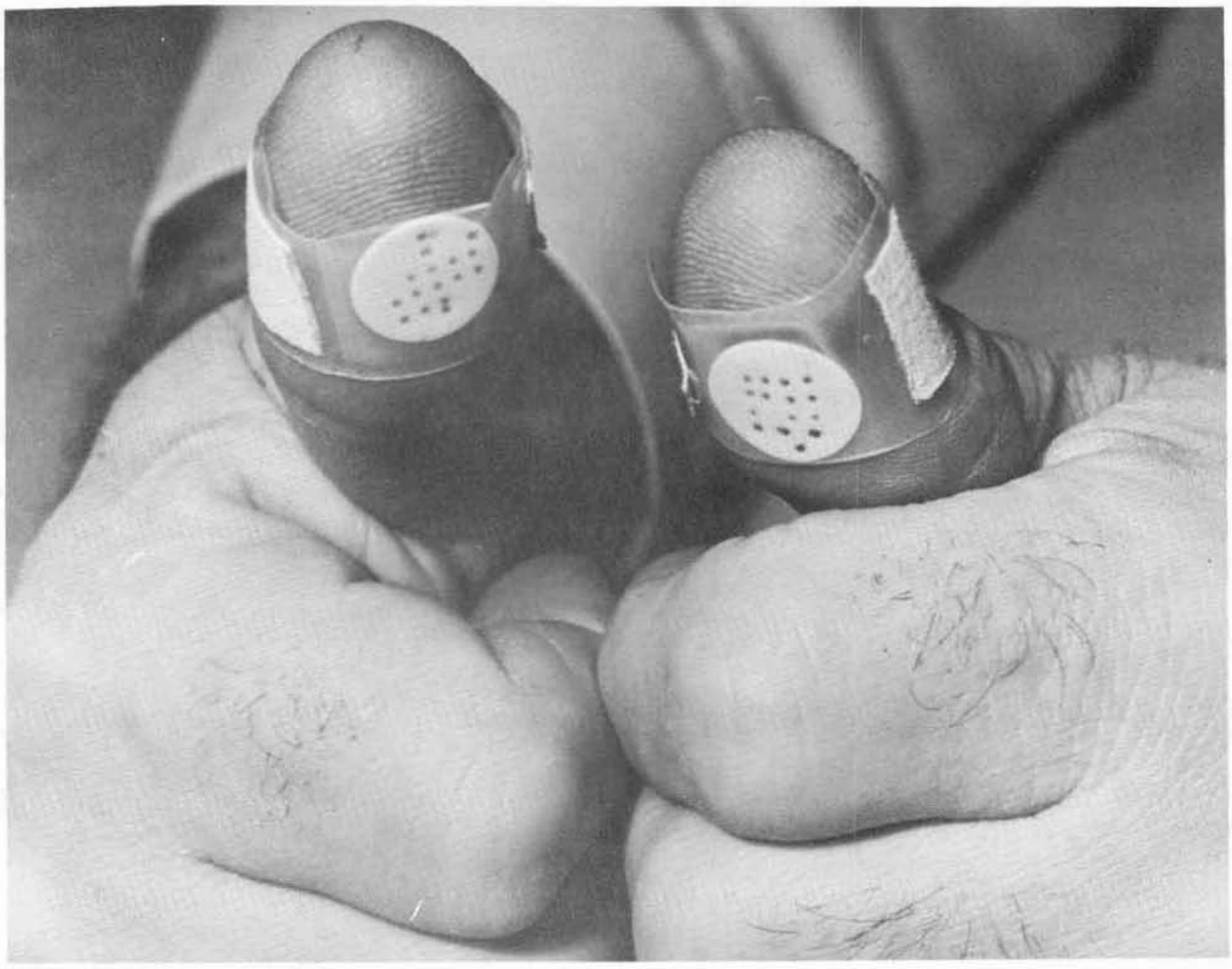


FIGURE 7

Thermo-Luminescence Dosimeters

7.2 Measuring Internal Exposures

7.2.1 Bioassay Samples

Internal contamination is eliminated from the body with the ordinary waste products: urine, sweat, and feces. By analyzing samples of these excreta, usually urine, the amount of radioactive material in the body can be estimated. From this is calculated the radiation dose delivered to the body by that material.

Urine samples are analyzed routinely for one or more of the fission products, including radio-iodine, or for tritium or plutonium. If a worker is exposed to other radio-elements, a special analysis will be done on request. An overnight sample is required for a fission-product analysis, whereas a small sample excreted at work (a "spot sample") is sufficient for tritium analysis. A fecal sample may be asked for in special cases.

7.2.2 Whole Body Counter

A whole body counter is situated at Deep River to secure a low counter background. If personnel

are internally contaminated with gamma-emitting radioisotopes, they may be counted periodically in the whole body counter. In this way an accurate record of the elimination or decay of their internal contamination may be established, or insoluble gamma-emitting isotopes may be detected.

7.2.3 Thyroid Counter

In some instances where a number of individuals have been exposed to a release of radioiodine to the atmosphere, an estimate of expected exposures can be obtained by using the thyroid counter on a representative group of the exposed persons.

The majority of exposures to radioiodine, however, are determined from bioassay samples.

8. INSTRUMENTS

8.1 General

The positive charge on the nucleus of an atom is balanced by the negative charges on the electrons in the orbits around the nucleus. When an outer orbital electron is removed from an atom, two charged particles are formed. The removed electron is negatively charged, and the remainder of the atom, since it now has insufficient electrons to balance its positive charges, is positive. The two particles are known as an Ion Pair.

Specific ionization is the number of ion pairs formed per centimetre of path in any given medium, for that particular ionizing radiation.

$$\text{Specific Ionization} = \frac{\text{number of ion pairs formed}}{\text{centimetres of path}}$$

Specific ionization varies directly as the magnitude of the mass and electrical charge and inversely as the speed of travel (i.e. energy). Thus, alpha particles with a double positive charge and a mass number of 4 produce a comparatively high number of ion pairs per

unit length of path. This is not a constant value for a particular alpha particle, but is a function of its velocity as well as the atomic number of the absorbing medium. The density of alpha particle ionization gradually increases as the particle velocity decreases, finally reaching a peak and dropping to zero when all the energy is lost. The increased density of ionization at lower velocities is due to increased time the alpha particle has to act on the orbital electrons of adjacent atoms. Specific ionization, therefore, is an average value for the total length of path. The range of specific ionization values in air for alpha particles extends from about 5000 to 80,000 ion pairs per centimetre of path.

Beta particles with a single electrical charge, and a mass approximately 1/8000 of the mass of an alpha particle have a much lower specific ionization than alpha particles. The specific ionization values in air for beta particles extend from approximately 50 to 500 ion pairs per centimetre of path.

Gamma rays which are a form of electromagnetic radiation produce no direct ionization by collision along their path because they have no mass. They are absorbed in three ways, known as the photoelectric effect, the Compton effect and pair production (Section 9.5.4.1, page 141). Each of these processes results in ionization of the atoms of the medium through which the gamma rays pass.

Total ionization is the total number of ion pairs produced by an ionizing particle regardless of the length of its track. The total number of ion pairs produced depends mainly upon the energy of the particle, since the particle loses an approximately constant amount of energy for each ion pair formed. The formation of a single ion pair by a moving charged particle results in a loss of energy of about 32.5 electron volts. Therefore total ionization can be calculated by dividing the energy of the ionizing particles by 32.5 electron volts for air. Thus an incident particle with energy of 1.5 MeV will produce in air a total of 46,000 ion pairs.

Because ions are charged particles, electronic devices can be constructed to collect and measure the number present in a detector of known volume. Since the Roentgen unit which is used to measure the intensity of X- or gamma radiation is defined as that quantity which will produce 2.083 x 10^9 ion pairs per cubic centimetre of standard air, instruments can be constructed to measure directly the intensities of the various types of ionizing radiation.

8.2 Ion Chambers

Ion chambers consist of an enclosed volume of air, or gas, with a central electrode in the chamber maintained at an electrical potential in relation to the chamber wall. The form of chamber most commonly used consists of a cylinder with the charged electrode located on the axis of the cylinder. Since the electrode is charged in relation to the chamber wall, ions formed within the chamber will be attracted either to the central electrode or to the chamber wall, due to the attraction of unlike electrical charges.

As the potential difference between the electrode and the chamber wall is increased, the degree of attraction to the ions increases. The flow of ions, for a given amount of ionization, will increase as the voltage of the chamber is increased until all the ions formed are collected. At this point further increases of voltage across the chamber have little effect on the output of the chamber. When this condition exists, it is said that the chamber is saturated. Ion chambers operate with applied voltages in this region.

An electroscope may be connected from the electrode to the chamber wall, and the voltage change caused by the ion current can be measured, or the ion current may be used to develop a voltage across a large resistor, and the voltage can be read on a meter.

If an ion chamber is filled with a special gas it can be made more sensitive to some types of radiation. Examples of this are hydrogen-filled chambers for fast neutrons, argon chambers for gamma measurements, and boron trifluoride chambers for slow neutrons.

Chambers may also be made more sensitive by increasing the pressure of the gas in the chamber.

8.3 Scintillation Counters

There are substances such as phosphors and some crystals which, when struck by radiation, produce a visible light. The light flashes produced were once counted visually by operators working in dark rooms. Today these flashes are usually fed to a light-sensitive photo-cell circuit, and converted either to a meter reading or to an audible signal. Scintillation counters are used mainly for particle detection as in alpha, beta, and neutron counting.

8.4 Geiger Counters

8.4.1 Mode of Operation

When a voltage is applied across a gas-filled tube, at a certain point the resistance of the gas will break down, allowing current to flow through it. Such a tube may be held on the verge of conduction indefinitely.

The arrival of a single ionizing event, such as a beta particle or a gamma ray, will, by the formation

of a few ions in the gas, initiate ionization. The primary ions are accelerated by the high voltage potential and this added energy enables them to produce secondary ion-pairs by collision with the gas molecules in the tube. Thus, many more ions are produced than were formed originally. The ratio of the total number of secondary ion-pairs to primary ion-pairs is called the Gas Amplification Factor of the tube, and may be as high as 10^9. In order that the geiger will be ready for the next ionizing event, this amplification must be controlled. It may be done by introducing a quenching gas.

When radiation enters the tube, ionization of the gas takes place and an avalanche of electrons flows towards the central collecting electrode. This avalanche sends a pulse to the indicating unit of the geiger counter. The quenching gas then stops the flood of electrons and the tube is ready for another ionizing event.

The amplification associated with a geiger tube makes it possible to detect a single beta particle or gamma photon.

8.4.2 Types

(a) Cylindrical Geigers

These counters consist of a cylindrical envelope with a thin wire on the central axis of the cylinder. The centre wire is usually tungsten and carries a potential of about 1,000 volts in relation to the cylinder, which may be made of brass, copper, aluminum, bismuth, or silver deposited on a glass envelope, as in the case of the Eck-Krebb tube.

(b) End-Window Geigers

Geiger tubes for beta counting are usually of the end-window type. They have a thin mica window, usually between 1.5 to 3 mg/cm^2. The central electrode extends to within a few millimeters of the mica window and the sensitive part of the tube is a shallow area around the end of the electrode and the mica window. Beta counting efficiencies of up to 30% can be obtained with these tubes. Those in use by R & IS at present, in swipe castles and beta counters are approximately 10% efficient.

(c) Comparison

	Eck & Krebb	End-Window
H.T. Voltage	900v - 1100v	1100v - 1300v
Window Thickness	30 mgm/cm^2	1.5 - 3 mgm/cm^2
Counter Efficiency	10% (max.)	30% (max.)
Counter Sensitive Area	40 cm^2	7 - 10 cm^2

8.4.3 Gas Mixture

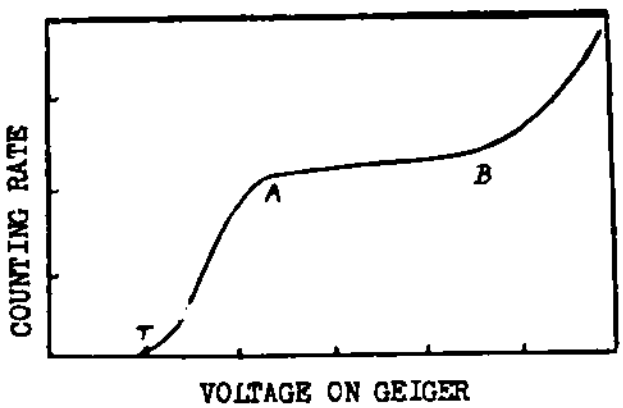
Geiger tubes are filled with gas under pressure of from 5 to 10 cm of mercury. The filling mixture used may be argon mixed with alcohol, in the ratio of eight parts of argon to one part of alcohol. Halogens, usually chlorine or bromine, are now being used as quenchers to provide a tube with a longer service life.

At each discharge in a tube containing an organic quencher such as alcohol, some of the vapour is dissociated, so its useful life is limited to approximately 10^8 counts.

Halogens used as quenchers recombine after dissociation, hence a tube using this type of quencher has an almost unlimited life.

8.4.4 The Voltage Plateau

If we use a constant source of radiation and a geiger counter, it will be noted in the following diagram that a certain starting voltage is required before any counts are obtained.



Where:

T = starting voltage
AB = working range (Plateau)

There is a region just above the starting potential where the number of counts increases as the voltage is increased.

Then a more or less flat region, called the Plateau is reached where there is very little change in the number of counts for increases in voltage. This plateau may be as broad as several hundred volts, or may be absent entirely. Its extent depends on the quenching resistance of the gas used in the tube. When the gas filling in the tube quenches large numbers of

counts, the voltage plateau of a tube becomes narrower and it is probably nearing the end of its useful life. After the plateau the number of counts increases rapidly as the applied voltage is increased.

Geiger counters should be operated with the applied voltage in the plateau region since small variations in voltage will not affect the number of counts.

8.4.5 Operating Geiger Counters

When changing the tube on a geiger monitor, always reduce the applied voltage below the expected start of the plateau. Even a few seconds of operation with the applied voltage too high will ruin a geiger tube. Increase the voltage gradually until the instrument counts background. The final voltage setting should be about 100 volts above the point where it starts counting. A source should be used to ensure that the geiger is operating properly.

The metal shield around an Eck-Krebb geiger tube is designed to eliminate betas from the counts recorded. Unless a specific need exists for a gamma

reading only, the counter should be used with this shield open, and the tube exposed, in order that low energy beta contamination will be detected.

Careful monitoring of materials cannot be rushed. Even relatively large counts of beta-gamma contamination will be overlooked unless the tube is moving slowly enough for the instrument to respond.

Most of the geiger counters we use are provided with a two-position time constant switch. The shorter time constant is used frequently when searching for possible contamination. This speeds up the response time of the instrument. However, the needle of the meter will be quite unstable especially at low counting rates. If contamination is detected, then the longer time constant is used to obtain greater meter stability and an accurate reading.

8.5 Portable Instruments for the Detection of Contamination

8.5.1 Battery Alpha Counter AEP 118A (a) (Figure 8)

This counter uses as a detector a thin layer of zinc sulphide powder deposited on the base

of a lucite cone. When an alpha particle strikes the zinc sulphide, a flash of light is transmitted by the lucite to the photo-cathode of a photomultiplier tube located at the apex of the lucite cone. To exclude light from the photomultiplier tube, the zinc sulphide screen is covered by a thin sheet of aluminized mylar. The mylar is protected by a coarse screen mesh.

This instrument has no meter. The signal is presented on a set of headphones. An estimate of the number of counts per minute is made by comparing the frequency of the clicks from an unknown source with those from a source of known strength.

The counter efficiency usually ranges from 15 to 20 per cent. Accurate calibration can be done by using an appropriate alpha source of known magnitude.

8.5.2 Transistorized Geiger Counter (Pogo Stick)

AEP 1910 (Bγ) (Figure 9)

This transistorized geiger counter uses a side-window halogen-quenched tube as the detector.

The signal is displayed on a logarithmic scale ranging from 0 to 20 K cpm. The efficiency for mixed fission product contamination is approximately 10%.

8.5.3 Contamination Ratemeter AEP 2160 ($\alpha\beta\gamma$) (Figure 10)

This is a transistorized ratemeter for the detection of alpha and beta-gamma contamination. The probe contains the photomultiplier, a dual phosphor scintillator, and a 2-inch indicating meter.

The photomultiplier has a 1-inch diameter end-window to which a 1/16 inch thick slice of plastic phosphor is cemented with a transparent adhesive. Over this a ZnS:Ag alpha detecting phosphor is laid. This is covered in turn by an alpha window of doubly aluminized mylar. A removeable end cap has a 0.001 inch thick aluminum window which is transparent to electrons with an energy of more than 20 KeV. A second removeable end cap with a 3/16 inch aluminum wall permits the entry of gamma rays but is opaque to 2.5 MeV electrons. This end cap fits over the first.

For the measurement of alpha activity both end caps should be removed and the function switch

set in the α position. To obtain the highest accuracy possible, the ratemeter should be calibrated with a known α source. The counting efficiency for α particles (5.4 MeV from 241 Am) is about 20 per cent.

To measure β activity, the β end cap should be in position and the function switch in the αβγ position. The discrimination against γ activity is high, but to correct for this, the same measurements should be repeated with the γ cap in place, and the reading so obtained subtracted from the first. The counter efficiency for β emitters does not vary critically with energy if the maximum β energy is greater than 0.5 MeV. Below this energy the efficiency falls to zero for beta particles with a maximum energy of 60 KeV or less. More accurate results can be obtained after calibration with a known β emitter. The efficiency for counting beta particles from a $^{90}Sr/^{90}Y$ source is about 20 per cent.

For the measurement of γ activity the β and γ end caps should be in position and the function switch in the αβγ position. Since the gamma efficiency is low and varies critically with energy, the

detector must be calibrated with a known gamma source of appropriate energy. The instrument may be operated with either a one inch or a five inch probe. Probes are not interchangeable since they require different operating voltages. Six ranges are available 1K, 3K, 10K, 30K, 100K, 300K cpm. A switch is provided for either short or long time constants, which are as shown below.

Range in cpm		1K	3K	10K	30K	100K	300K
Response Time	Short	14	5.8	1.4	1.4	1.4	1.4
in Seconds	Long	57	24	5.7	5.7	5.7	5.7

The AEP 2160 may be either battery or mains operated.

If only $\beta\gamma$ detection is desired, either side or end-window geigers may be used with the AEP 2160. No modifications to the internal circuitry are required. The function switch must be set in the $\alpha\beta\gamma$ position. With the protective cap off, efficiencies up to 26% are obtained for $^{90}Sr/^{90}Y\beta$ rays using the Type 18516 or Type 18536 end-window geigers. No side-window geigers are used with the AEP 2160 by R & IS Branch at CRNL.

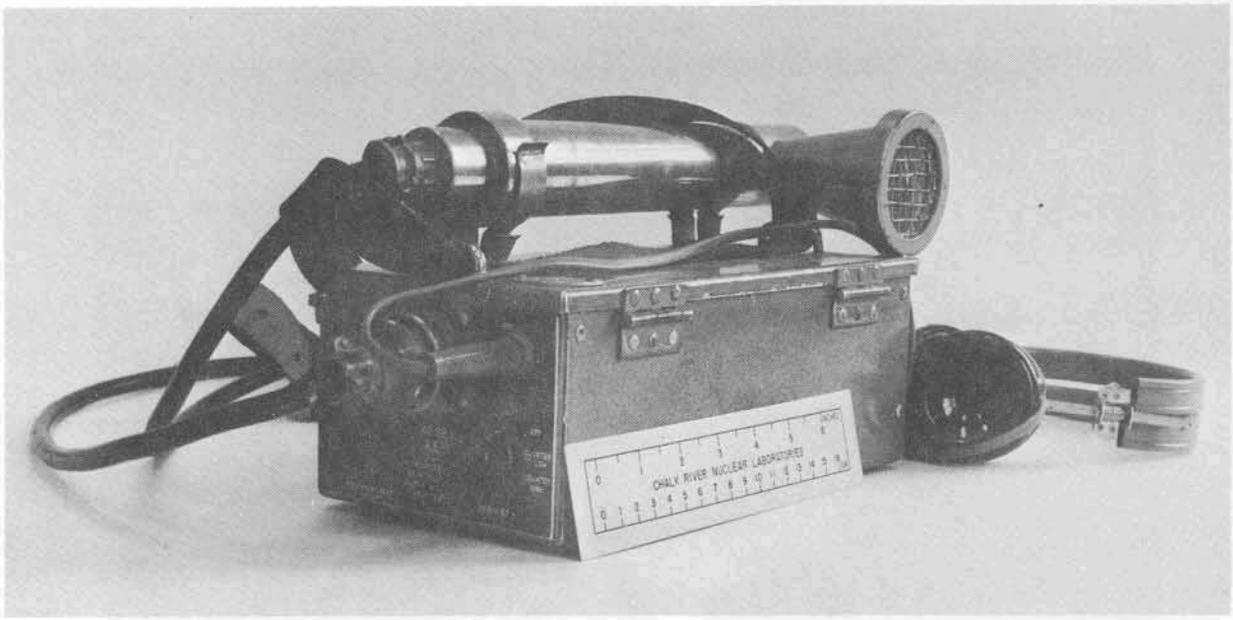


FIGURE 8
Battery Alpha Counter - AEP 118A

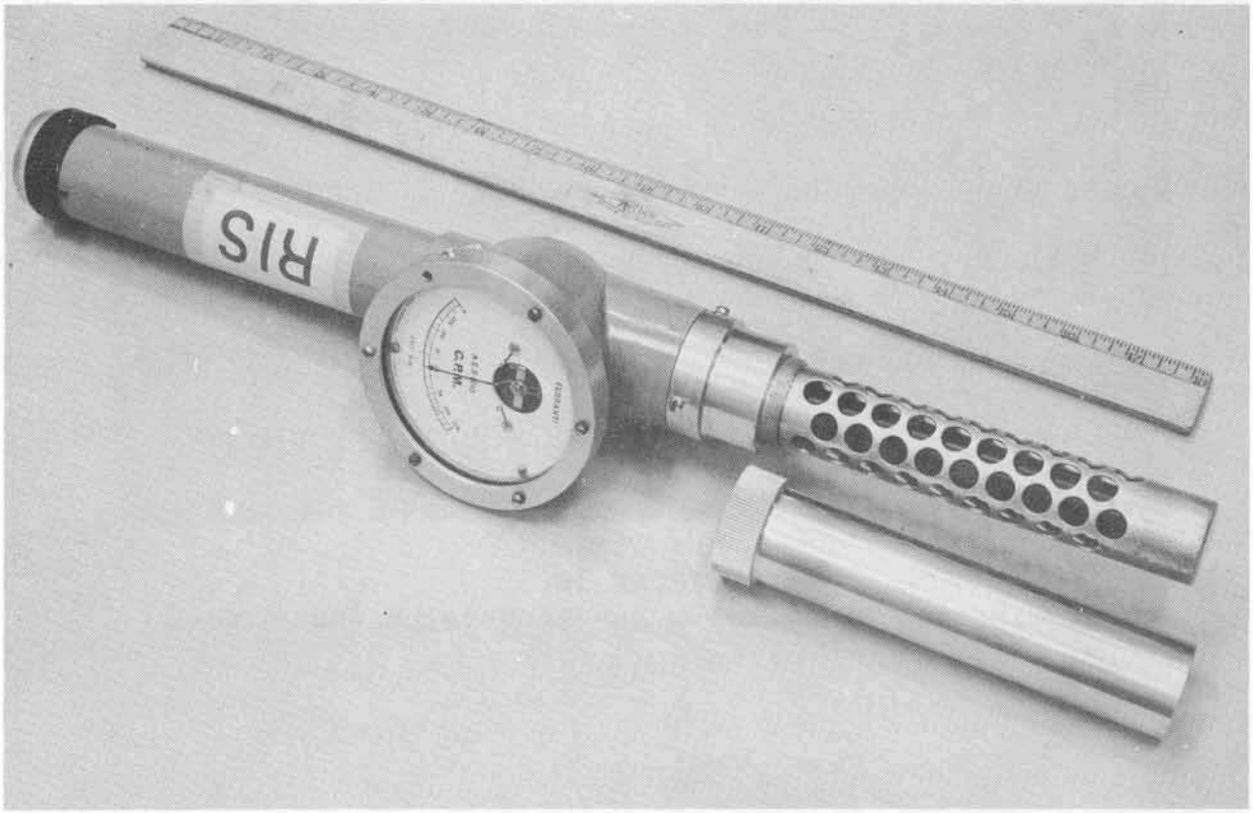


FIGURE 9
Transistorized Geiger Counter (Pogo Stick) AEP 1910 (Beta-Gamma)

FIGURE 10
Contamination Ratemeter - AEP 2160 (alpha-Beta-Gamma)

8.6 Non-Portable Instruments for the Detection of Contamination ($\alpha\beta\gamma$)

There are three instruments used in fixed locations for the detection of $\alpha\beta\gamma$ contamination, and one for $\beta\gamma$ detection only.

8.6.1 Geiger and Scintillation Probe Monitor - AEP 1903-S (Figure 11) ($\alpha\beta\gamma$)

8.6.2 Baird Atomic Model 410 - AEP 1901 (Figure 12) ($\alpha\beta\gamma$)

8.6.3 Tracerlab Laboratory Monitor - Model SU3D - AEP 1904 (Figure 13) ($\alpha\beta\gamma$)

8.6.4 Baird Atomic - Model 441A - AEP 4830 (Figure 14) ($\beta\gamma$)

All these instruments are mains operated, and all have outlets to which one may attach a geiger probe for $\beta\gamma$ detection. Three have outlets as well for an alpha scintillation probe AEP 117. The Baird Atomic 410 and Tracerlab SU3D are also used to operate the Large Area Filter Counters AEP 10057.

When used with end-window geigers as swipe or air sample counters the β efficiency is around 10 per cent. When used with the Large Area Filter Counters AEP 10057, efficiencies up to 50 per cent are

possible. Alpha counting efficiency with the AEP 117 probe is from 15-20 per cent.

The AEP 1903-S and AEP 1904 have ranges from 0 to 20K cpm; the AEP 4830 from 0-100K cpm; and the AEP 1901 from 0-200K cpm. Time constants available on these instruments are shown in the following table:

TABLE 4

Instrument	AEP No.	Normal Time Constants	Long Time Constants
Geiger and Scintillation Monitor	1903-S	0.8 and 8.0 sec.	70 - 80 sec.
Baird Atomics Model 410*	1901	5 sec. on 200 cpm scale 2 sec. on 2K cpm scale 0.5 sec. on 20K cpm scale 0.05 sec. on 200K cpm scale	60 sec.
Tracerlab Laboratory Monitor	1904	1 sec. and 11 sec.	60 sec.
Baird Atomic Model 441A	4830	1 sec. and 10 sec.	None
* No switching required			

FIGURE 11
Geiger and Scintillation Probe Monitor - AEP 1903-S (Alpha, Beta, Gamma)
(Alpha scintillation probe not shown)

FIGURE 12
Baird Atomic Model 410 - AEP 1901 (Alpha, Beta, Gamma)
(Alpha scintillation probe not shown)

FIGURE 13
Tracerlab Model SU3D - AEP 1904 (Alpha, Beta, Gamma)
(Alpha scintillation probe not shown)

FIGURE 14
Baird Atomic Model 441A - AEP 4830 (Beta, Gamma)

8.7 Portable Survey Meters for the Measurement of Radiation Fields

8.7.1 Multi-Purpose Survey Meter AEP 2153A (γ) (Figure 15)

The Multi-Purpose Survey Meter is used for the measurement of gamma radiation fields. It uses separate halogen quenched geigers as the detectors for its two ranges. When switched on the instrument registers on the lower range (0-100 mR/h). If a reading beyond this range is indicated a trigger on the front of the handle is depressed bringing the higher range (0-10 R/h) into operation. A moveable shield covers the low range geiger and may be opened to permit the entry of beta radiation.

An indicator on the top of the case should tilt when the instrument is switched on, otherwise the battery operating voltage is too low.

Calibration is normally done by the Electronics Branch although source castles have been provided to check the calibration of both ranges, in the field.

Response to gamma ray photons is reasonably energy independent from 75 KeV to 2 MeV.

One instrument is available with a 30 foot probe extension cord for remote monitoring.

8.7.2 High Range Survey Meter AEP 2163 (γ) (Figure 16)

The High Range Survey Meter is used for the measurement of gamma fields beyond the range of the Multi-Purpose Survey Meter. It uses a small halogen quenched geiger, situated about three inches from the end of a telescopic probe, as the detector. With the probe collapsed the detector is located about three feet from the indicating meter. With the probe extended it is about six feet away.

When the instrument is switched off, the meter indicates the condition of the battery. For satisfactory operation, the needle should register to the right of the red mark on the scale. When switched on the instrument will measure gamma radiation fields from 0 to 500 R/h. Readings are displayed on a logarithmic scale.

Response to gamma ray photons is reasonably energy-independent from 75 KeV to 2 MeV.

A ^{60}Co calibration source is located in the NRX reactor building. The AEP 2163 must read within -10 to +20% of the calibration dose rate.

8.7.3 Low Energy Survey Meter - Victoreen Model 440 RF AEP 4827 (X, γ) (Figure 17)

Our standard survey meters such as the AEP 2153A and AEP 2163 do not respond accurately to gamma photons with energies below about 75 KeV. The Victoreen Model 440 Survey Meter uses an air ionization chamber as the detector. The end-window is covered by a piece of ¼ mil mylar and over this is a thin magnesium foil. The accuracy for gamma photons from 10 KeV to .5 MeV is ± 15%.

Ranges available are 0-3, 0-10, 0-30, 0-100, 0-300 mR/h.

8.7.3 Low Energy Survey Meter - Nuclear Chicago - Model 2588 (X,γ) (Figure 18)

Another low energy survey meter, the Nuclear Chicago Model 2588 is also available for the

measurement of low energy gamma photons. This instrument also uses air ionization chambers as detectors. Two interchangeable chambers are supplied. Each chamber is provided with a two-position switch in order that the instrument may be used to measure either dose rates or accumulated doses.

The medium range Model 2526 ion chamber is 3 inches in diameter with a volume of 500 cc. The chamber wall is 1/16 inch bakelite while the end-window is a rubber hydrochloride membrane with an effective area of 30 cm^2 and a density less than 1 mg/cm^2. A 1 $gram/cm^2$ plastic beta shield is furnished for measuring gamma radiation. A cesium calibration source is stored within this shield. The Model 2526 chamber has three full-scale linear ranges of 25, 250, and 2500 mR/h for dose rate operation, and three full-scale linear ranges of 0.25, 2.5 and 25 mR for accumulated dose operation.

With the window exposed and facing the direction of the source the Model 2526 chamber is linear within ± 5% for X- or gamma rays of 10 KeV to 2 MeV.

The specifications for the Model 2520 high range ion chamber are identical to those for the 2526 chamber except as follows. The chamber is 3/4 inches in diameter, with a 5 cc sensitive volume. A white line around the chamber indicates the limit of the sensitive volume. The end-window has an effective area of 1.6 cm^2. Using the Model 2520 chamber there are three full-scale linear ranges of 2.5, 25 and 250 R/h for dose rate operation, and three full scale linear ranges of 0.25, 2.5 and 25R for accumulated dose operation.

With the window exposed and facing the direction of the source, the Model 2520 chamber is linear within ± 6% for X- or gamma-rays of 50 KeV to 2 MeV.

The response of both chambers to beta particles with energies above 40 KeV is approximately 80% of the actual rad value.

TABLE 5

Ion Chamber	Operating Mode	Desired Meter Range	Function Switch Position	Multiply Meter Reading in mR by
Model 2526 Medium Range Chamber	Rate	0-25 mR/h 0-250 mR/h 0-2500 mR/h	X 1 X 10 X 100	Read Directly 10 100
	Integrate	0-0.25 mR 0-2.5 mR 0-25 mR	X 1 X 10 X 100	0.01 0.1 Read Directly
Model 2520 High Range Chamber	Rate	0-2.5 R/h 0-25 R/h 0-250 R/h	X 1 X 10 X 100	100 1000 10000
	Integrate	0-0.25 R 0-2.5 R 0-25 R	X 1 X 10 X 100	10 100 1000

8.7.5 Beta Survey Meter AEP 2169 (β) (Figure 19)

The Beta Survey Meter is used to measure beta tissue dose rates. The detector employed is a thin anthracene scintillator. The window of the AEP 2169 has the same equivalent thickness as the outer dead layer of skin (7 mg/cm^2), therefore it will detect any beta radiation having sufficient energy to damage the skin.

Relative Response to Beta Energies
AEP 2169

End-Point Energy	Response
0.07 MeV	(lower limit)
0.3 MeV	0.6
0.8 MeV	0.8
1.7 MeV	0.9
2.3 MeV	1.0

To use the AEP 2169, take a reading with the end cap on, and another reading in the same spot with the cap off. Neither reading by itself is a reliable measure of the radiation field, but the uncapped reading minus the capped reading gives a measure of beta dose rate, with an accuracy of ± 15%.

The readings are displayed on a logarithmic scale from 0 to 100 rad/h.

Probe extension cords 5 or 24 feet in length are available for remote monitoring.

In order to determine β energies, an adapter may be fitted to the end of the 2169 probe. Ten aluminum absorbers covering the β energy range from .1 MeV to 3 MeV may be inserted into the adaptor singly, or in

combination.

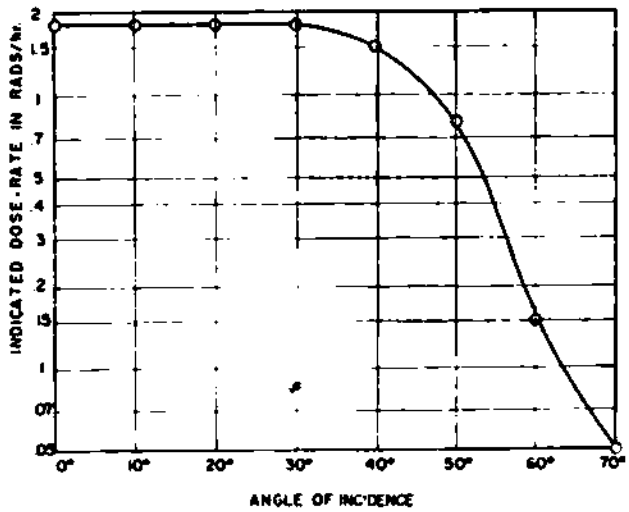
8.7.6 Beta Survey Meter - AEP 5210 (β) (Figure 20)

The AEP 5210 Beta Survey Meter is also used to measure beta tissue dose rates. The detector in this case is a silicon p-n junction, RCA type SJ 2421.

The energy response to beta particles is similar to that of the AEP 2169 (Section 8.7.5).

To obtain a beta dose rate reading, the procedure is similar to that used with the AEP 2169. One reading should be taken with cap covering the detector to determine the response of the instrument to the gamma background radiation. A second reading with the detector cap removed will give both the beta dose rate plus the response to gamma background radiation. By subtracting we obtain the beta dose rate.

The directional response of the AEP 5210 is shown in the graph on the next page.



It will be noted that for sources more than 60° away from normal incidence the instrument is relatively insensitive. Because of this directional characteristic it can be used for locating sources as well as measuring the dose rate from them.

8.7.7 <u>Slow Neutron Survey Meter Model 1399A - AEP 4820 (SN) (Figure 21)</u>

This instrument is designed to measure slow neutrons in terms of dose rate. It contains an ionization chamber lined with 1 mg/cm^2 of natural boron. The slow neutrons are detected by the reaction:

$$^{10}_{5}B + ^{1}_{0}n \longrightarrow ^{7}_{3}Li + ^{4}_{2}He + 2.5 \text{ MeV}$$

The 2.5 MeV reaction energy is carried away by the resultant lithium nucleus and alpha particle, and the ionization produced in their tracks is measured in the usual way. The discrimination against gamma dose-rate is about 90:1.

Two ranges are provided - 0 to 15 mrem/h and 0 to 150 mrem/h. Other settings of the function switch provide a means of monitoring the L.T. batteries and a means of checking the accuracy of calibration by an internal source of ^{90}Sr.

The input time constant is approximately 1 second.

8.7.8 Fast Neutron Survey Meter Model EIC - AEP 4824 (FN) (Figure 22)

The EIC Survey Meter is used to measure fast neutron radiation fields up to 500 mrep/h on 3 ranges. An additional range is provided which enables the measurement of up to 200 cumulative neutron counts. The radiation detection element is mounted in a probe connected to the meter by a flexible cable and neutron radiation levels may be measured at arm's length.

The counter has a response proportional to human tissue sensitivity to fast neutrons over the energy range 0.2 to 14 MeV. Each counter consists of a number of polyethylene-lined cells placed in line, with a common anode wire made from 0.001 inch diameter tungsten wire. The cells are formed from polyethylene discs and cylinders, which act as proton radiators in a fast neutron flux, and at any orientation to a neutron beam, present approximately the same area of radiation. This ensures that the response is independent of the direction of the beam. The gas filling is a mixture of methane and argon.

When the selector switch is on the "INT" position, the meter reads directly in total neutron counts. For readings lower than those obtainable on the 5 mrep/h range, or for the measurement of accumulated dose, the "INT" (integrate) range may be used. To measure the cumulative dose in mrep, divide the total count obtained in the "INT" position by 3275. The instrument may be rezeroed at any time during the integrating measurement without disturbing the measurement. To determine low dose rates divide the

cumulative dose in mrep by the time in hours.

Allow about 5 minutes for a reading on the 5 mrep/h range. Half a minute is sufficient for a reading on the 50 mrep/h range. Only a few seconds are required on the 500 mrep/h range.

8.7.9 Fast Neutron Survey Meter - Fairport Model 420 (FN) (Figure 23)

The Fairport Fast Neutron Survey Meter is used to measure fast neutron radiation fields up to 2500 mrem/h on 3 ranges. An additional range marked "REG" may be used for rates below 1 count per second. In this position, accumulated counts are recorded on a register. One count per second is equal to 8.33 mrem/h. The detector located within the instrument case consists of 3 cells formed from discs and cylinders which act as proton radiators in a fast neutron flux. At any orientation to a neutron beam, they present approximately the same area of radiator. The vacuum envelope is a stainless steel cylinder, filled with a mixture of methane and argon. Response to fast neutrons ranges from 0.1 MeV to 14 MeV.

This instrument is operated by rechargeable batteries. Battery condition may be checked by setting the selector switch to the "BAT" position. If the meter indicates in a green, the instrument should operate for approximately 20 hours. It requires about 70 hours to charge a fully discharged battery. The time constant for all ranges is 5 seconds.

8.7.10 Nemo Spherical Neutron Dosimeter System - Model 9140 (Figure 24)

The Nemo Neutron Dosimeter System consists of five units:

Model 9145 - Detector

Model 9146 - Ratemeter

Model 9147 - Recorder

Model 9148 - Scaler

Model 9149 - Carrier

The detecting element of the 9145 consists of a 4 x 8 mm ^{6}LiI(Eu) crystal surrounded by a 10" diameter solid polyethylene spherical moderator. A polished lucite light pipe transmits light pulses from the crystal to a magnetically shielded photomultiplier. The photomultiplier is followed by a

preamplifier and a discriminator.

The energy response of the 9145 Detector Assembly is within ± 4 per cent of the R.B.E. dose at thermal energies, and within ± 15 per cent for fast neutrons up to about 7 MeV. Although the energy response falls off rather rapidly above 10 MeV, it has been found that little error is incurred in personnel areas in accelerator buildings producing neutrons up to 17 MeV (except where measurements are made in a direct beam). This is because the presence of scattering material quickly reduces the average neutron energy to within the accurate response range of the 9140.

The detector output may be fed either to the 9146 Ratemeter or to the 9148 Scaler. The Ratemeter may be operated by batteries or by a 110V ac supply. It has four switch-selected scales ranging from 1 mrem/h to 1000 mrem/h full scale. An additional feature is a six digit register which can scale individual events from the 9145 Detector, at low count rates. The 9145 ac operated Rustrak strip chart

recorder is mounted in the carrier beside the 9146 Ratemeter. It provides the operator with a continuous time-indexed recording of dose history. The standard chart speed is one inch per hour.

The dose sensitivity of the 9145 Detectors used on our instruments is recorded in the manual accompanying each instrument. Therefore, to convert Ratemeter or Scaler counts to m/rem divide the total counts registered by the number of counts/m rem as shown on the calibration chart for the detector in use.

FIGURE 15
Multi-Prupose Survey Meter - AEP 2153A (Gamma)

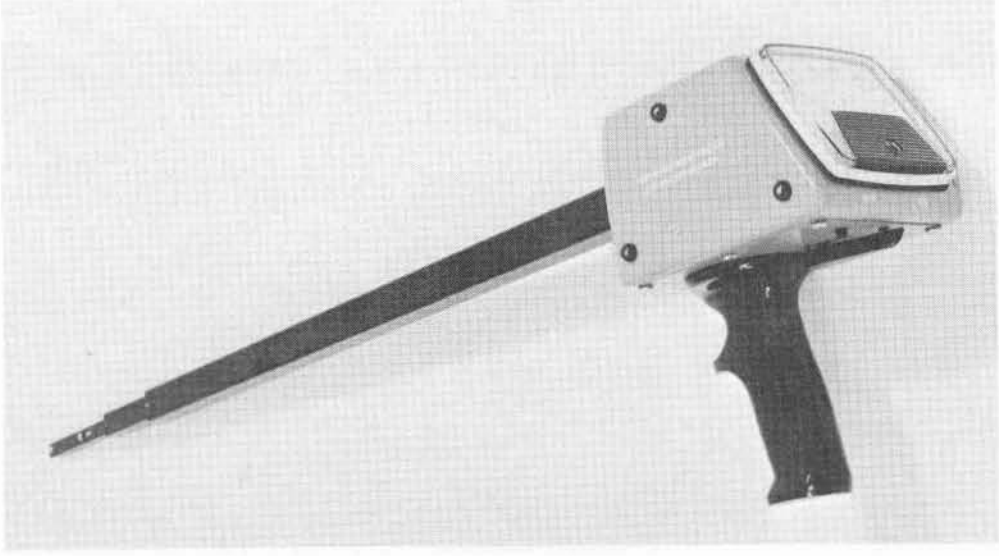


FIGURE 16
High Range Survey Meter - AEP 2163 (Gamma)

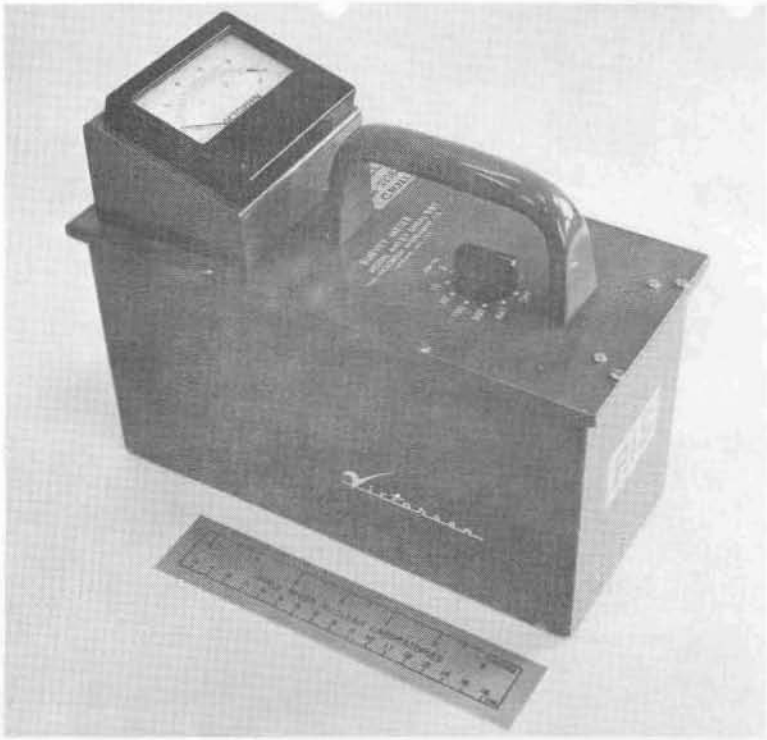


FIGURE 17
Low Energy Survey Meter - Victoreen Model 440 RF AEP 4826 (X-Gamma)

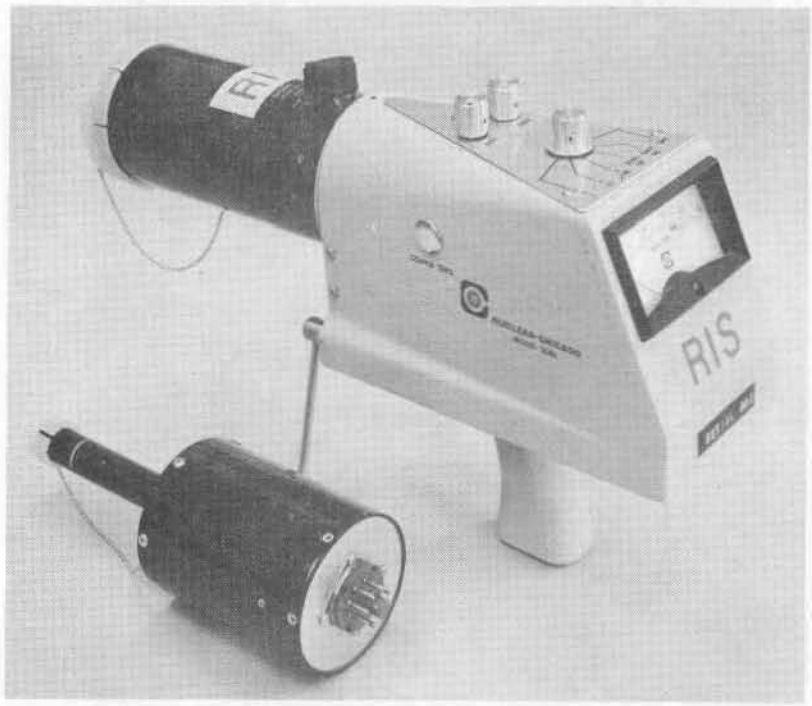


FIGURE 18
Low Energy Survey Meter - Nuclear Chicago Model 2588 (X, Gamma)

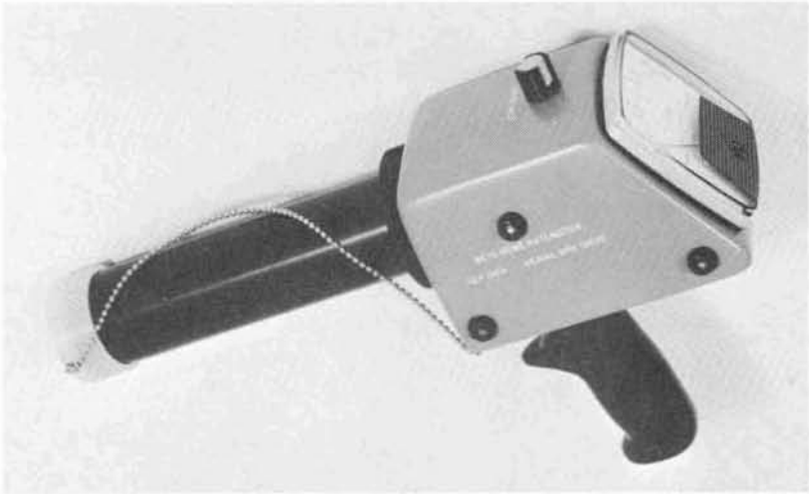


FIGURE 19
Beta Survey Meter AEP 2169

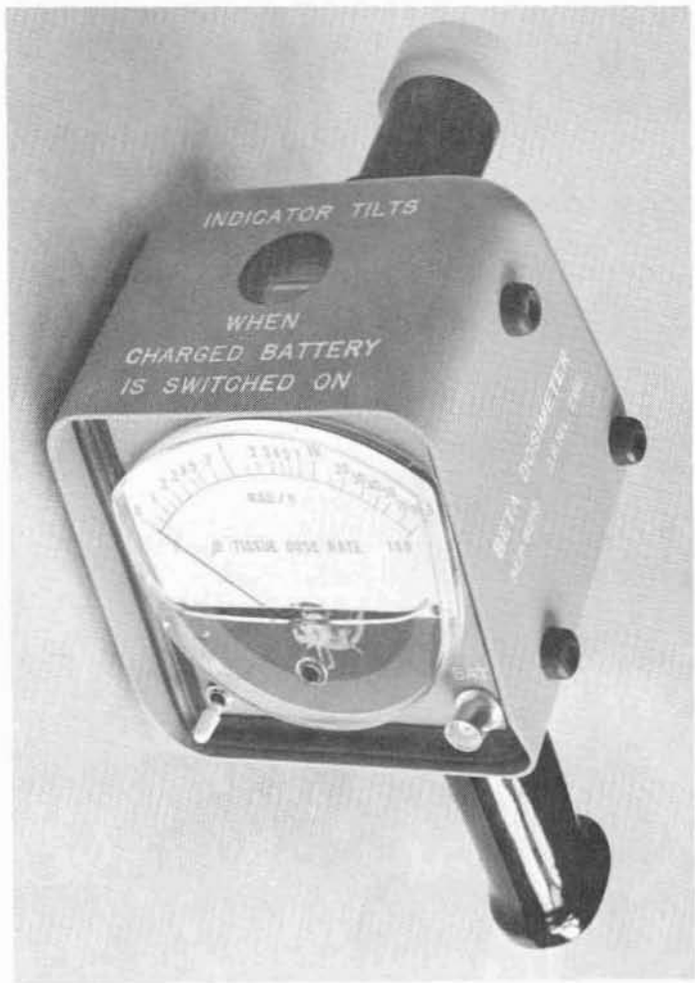


FIGURE 20
Beta Survey Meter - AEP 5210

FIGURE 21

Slow Neutron Survey Meter - AEP 4820

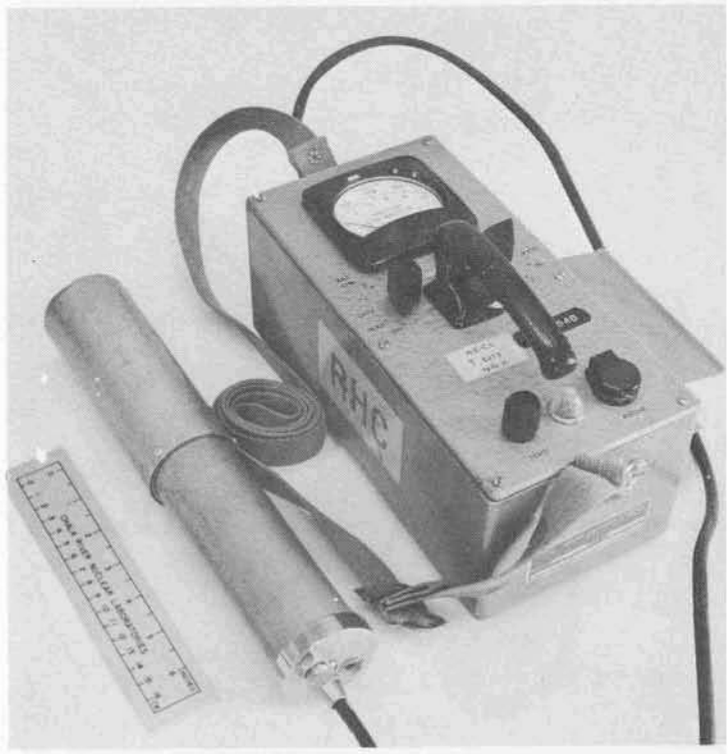


FIGURE 22

Fast Neutron Survey Meter - N.C.A. Model EIC - AEP 4824

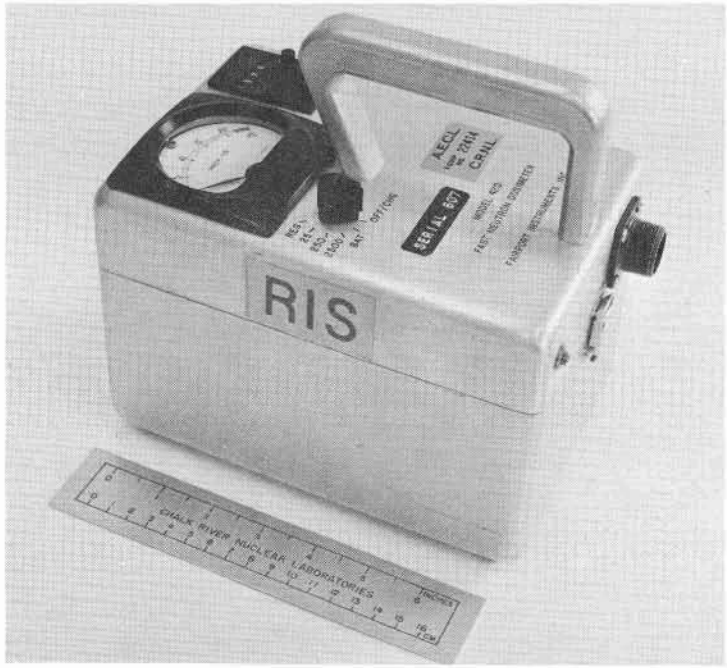


FIGURE 23
Fast Neutron Survey Meter - Fairport Model 420

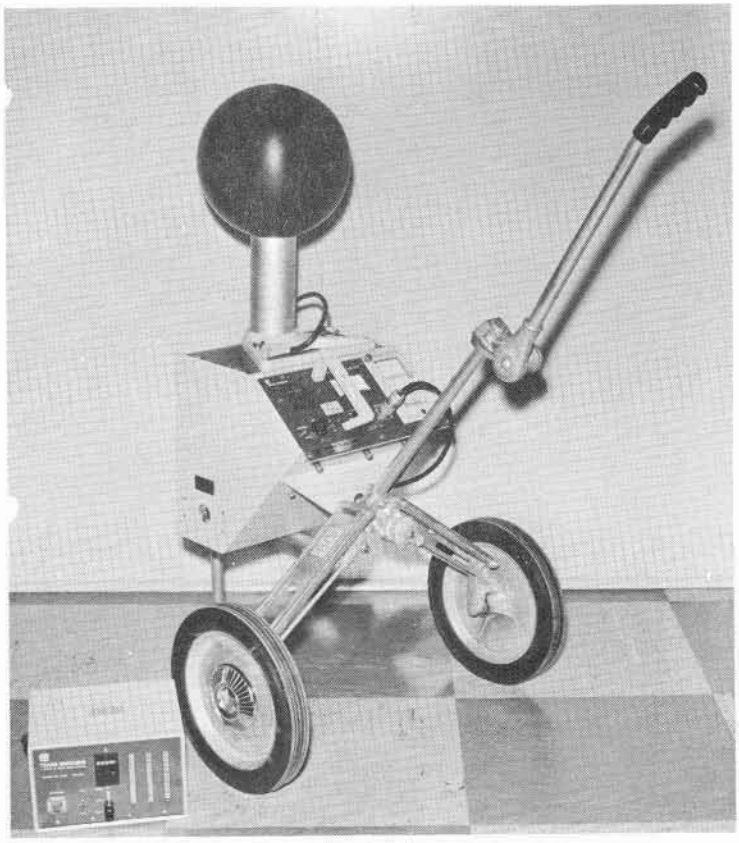


FIGURE 24
Nemo Neutron Dosimeter System

8.8 Mobile High-Range Survey Meter

8.8.1 Roentgen Ratemeter - Victoreen Model 510 (γ) (Figures 25, 26)

For the measurement of gamma radiation fields extending beyond the range of the AEP 2163, a remotely operated mobile unit has been developed at CRNL. The detection instrument used with this unit is a Victoreen Roentgen Ratemeter Model 510. A Telescoping boom to which the detection chamber is attached can be extended from six to eighteen feet, can be moved up and down and rotated in almost a complete circle, from a control panel located one hundred feet or more away.

The range and energy response of the instrument is governed by the probe selected. Those listed in Table 7 on the next page, are available at CRNL.

TABLE 7

Probe Model	Sensitivity Ranges (R/min)
612	Low Energy - 6 to 35 KeV (EFF.) 30, 100, 300, 1000
601	Medium Energy - 30 to 400 KeV (EFF) 3, 10, 30, 100
602	30, 100, 300, 1000
608	0.3, 1, 3, 10R/h
613	0.03, 0.1, 0.3, 1

FIGURE 25
Roentgen Ratemeter - Victoreen Model 510 (Gamma)

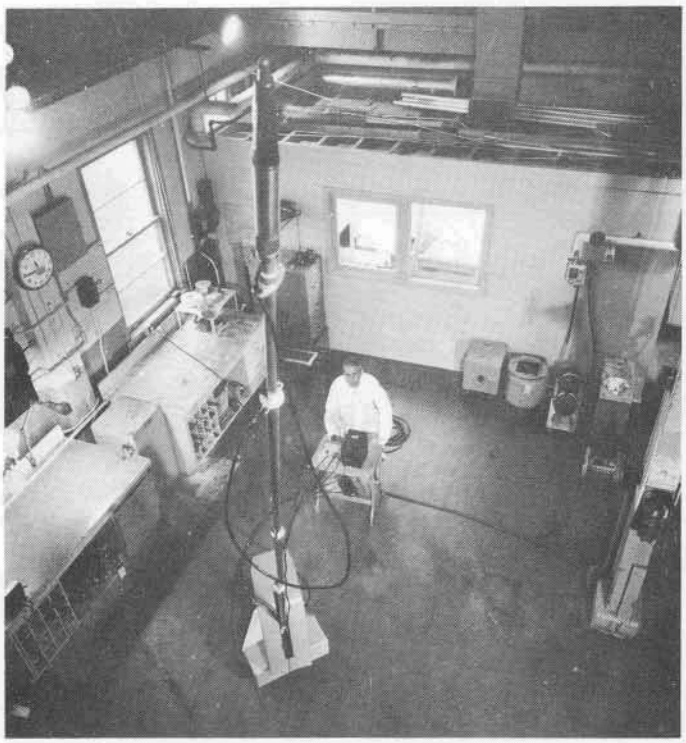


FIGURE 26
Remotely-Operated Monitor Using Roentgen Ratemeter Model 510 (Gamma)

8.9 Special Radiation Instruments (Available from the Radiation Dosimetry Branch)

8.9.1 Ultra High Range Gamma Meter (γ) Qty. 1

(a) Range - 0 to 5000 R/h.

(b) Detector - Halogen counter.

(c) Energy Dependence - ± 20% 100 KeV - 1.26 MeV.

(d) Portable - battery operated

(e) Cable length 50 feet to 100 feet between detector and monitor.

8.9.2 Road Surface Monitor - (γ) Qty. 1

(a) Range - 0 to 25 K cpm, 0 to 250 K cpm.

(b) Detector - sodium iodide crystal.

(c) Battery operated

(d) Designed for vehicle mounting, on front bumper

8.9.3 High Range Gamma Recording Monitor (γ) Qty. 3

(a) Range - 0 to 500 R/h.

(b) Detector - Halogen counter.

(c) Energy Dependence - ± 20% 80 KeV - 1.26 MeV.

(d) A.C. line operated with battery standby (1 week)

(e) Unit drives 1 ma Esterline-Angus Recorder and has alarm system

(f) Extension cable available (15-20 feet).

8.9.4 Aerial Survey Monitor AEP 2157 (γ) Qty. 2

(a) Range - 0 to 30 K cpm.

(b) Detector - sodium iodide crystal.

(c) Protable - battery operated with continuous recording.

(d) Designed for light aircraft.

8.9.5 Beta Dose Rate Meter - Modified AEP 2169 (β) Qty. 1

(a) Range - 0 to 50,000 rads/h.

(b) Portable - battery operated

8.9.6 E.I.L. Electrometer Model 37A (γ) Qty. 2

(a) Dose range - 0 to .45 mR minimum
0 to 150 R maximum — In 12 steps

Dose rate - .45 mR/sec. minimum
15 R/sec. maximum — In 11 steps

(b) Detector - ion chamber

(c) Portable - battery operated

(d) Energy response 20 KeV - 1.2 MeV

(e) Extension cable available (10-15 feet).

8.9.7 Victoreen Survey Meter Model 440 (X-γ) Qty. 1

(Section 8.7.3, page 101).

8.10 Special Safety Instruments

8.10.1 Mercury Vapour Meter

8.10.2 Oxygen Meter

8.10.3 Sound Level Meter

8.10.4 Combustible Gas Indicator

8.10.5 CO_2 Meter

8.10.6 Light Meter

8.10.7 RF Radiation Monitor

8.10.8 Laser Beam Detector

9. EXTERNAL EXPOSURE CONTROL

9.1 General

Procedures which have been used in the past at CRNL, and found to be effective, are outlined in this section. They are of a general nature, since each situation involving radiation or contamination will present special problems.

9.2 The Routine Survey

In order to ensure that no dangerous radiation fields exist in working areas of the plant, it is customary for Radiation Surveyors to carry out a routine survey at some time during each shift. Where no other warning is provided, workers in an area rely to a large extent on these surveys to warn them of unsafe working conditions.

When radiation fields are encountered which could cause an overexposure, the surveyor posts a "Caution" sign to warn anyone approaching the area that radiation levels are beyond acceptable limits. The sign states what type of radiation is present, where it originates, the radiation level on contact

with the source (if possible), and the radiation level at a safe working distance from the source. If potentially dangerous levels of radiation are encountered additional warning signs are displayed, and a barrier of red tape is erected to prevent entry into the dangerous area.

An effective routine survey requires a working knowledge of the operations being carried on in any area. This knowledge is necessary in order that hazards may be assessed properly, and so that the implications of any departure from routine operations may be appreciated. For example, a power increase in a reactor will cause a series of changes in normal radiation levels and will probably give rise to some entirely new radiation fields. These must be measured, evaluated and a report passed on to Supervision in the area. It is the responsibility of the surveyor to assess the importance of these readings, and keep Supervision informed of any important changes observed.

A daily log is maintained in each area, in which the results of routine surveys are summarized.

The value of a log entry is seldom appreciated until someone tries to recall the details of an operation which occurred some time ago. From time to time, a surveyor may be called upon as a witness during an investigation. Although it is not expected that a surveyor will remember all the details of some past event, he is expected to produce accurate evidence. He should be able to support this evidence by reference to a written account in an official log book.

9.3 The Work Permit (Appendix II, page 219)

A work permit is required before maintenance work is done in most of the buildings in the Active Area of the plant. The permit is authorized by Supervision of the building concerned, and is filled out in consultation with a Radiation Surveyor and a Maintenance Foreman. The permit is completed in triplicate - the white copy stays with building Supervision, the pink copy to the Maintenance Branch, and the blue copy to the R & IS Branch.

The surveyor is required to recommend the precautions that must be observed because of the

presence of radiation hazards. He must indicate on the work permit what these hazards are, he must set a working time such that no overexposure to external radiation will occur, he must recommend the appropriate clothing and respirator to avoid internal contamination, and must give advice regarding any final precautions, or movement of radioactive sources. It is essential that the surveyor signing a work permit should do a thorough radiation and contamination survey of the area where the work is to be done immediately prior to signing the permit. If there is any reason to feel that conditions may change during the progress of the work, the surveyor may deem it adviseable to remain in attendance, or to pay frequent visits to the area as the work proceeds.

The most important readings taken during the survey will be those recorded in the exact location where the men are expected to work. If higher levels of radiation exist in the vicinity, these areas should be brought to the attention of the workmen in case they should find it necessary to move from their original

working area. Close cooperation between the workmen and surveyor is essential at all times. The workmen can assist the surveyor greatly by telling him exactly what is involved in the work they are going to do, or by explaining briefly the construction or operation of equipment with which the surveyor is unfamiliar. The surveyor can assist the workmen by giving them all available information regarding the radiation hazards associated with the work in question.

9.4 Dose Estimation

In order to prevent the overexposure of personnel to external radiation, a surveyor must measure the various types of radiation existing and ensure Maximum Permissible Doses (MPD's) are not exceeded. (See Section 6.2.1). At CRNL the initial limiting figure is the dose permitted in a two-week period. The initial restriction in the Recommendations of the International Commission on Radiological Protection is the dose permitted in 13 weeks (14 weeks at CRNL). In some areas at CRNL, such as chemistry laboratories, workers may be exposed at a fairly uniform rate each

day of the week. To estimate a working time for such workers it would probably be convenient to set a daily limit. In other areas, such as a reactor, where the rate of exposure may vary considerably depending upon whether the reactor is operating or shut down, and also depending upon which day of the two-week film period has been reached, it might be necessary to allow a worker to receive the greater portion of his permissible dose for the two-week film period in one or two days.

9.4.1 <u>To Estimate the Working Time in a Gamma Radiation Field</u>

Measure the gamma dose rate with a Multi-Purpose Survey Meter - AEP 2153A

The MPD for the two-week film period is 600 mR.

∴ working time in hours

$$= \frac{600}{\text{Dose Rate Reading on AEP 2153A in mR/h}}$$

and working time in minutes

$$= \frac{600 \times 60}{\text{Dose Rate Reading on AEP 2153A in mR/h}}$$

If it is desirable to use a daily working time, divide the results obtained above by 10.

9.4.2 <u>To Estimate the Working Time in a Beta Radiation Field</u>

Measure the beta dose rate with a Beta Survey Meter - AEP 2169 or AEP 5210. Subtract the reading taken with the plastic end-cap on from the reading taken with the cap off.

The MPD for the skin of the whole body for the two-week film period is 1.6 rads (1600 mrads).

$$\therefore \text{ Working time in hours} = \frac{1600}{\text{Dose Rate Reading on AEP2169 in mrads/h}}$$

$$\text{and Working time in minutes} = \frac{1600 \times 60}{\text{Dose Rate Read. on AEP2169 in mrads/h}}$$

If it is desirable to use a daily working time, divide the results obtained above by 10.

9.4.3 <u>To Estimate Whole Body and Skin Dose Equivalents</u>

The meters of gamma survey instruments such as the Multi-Purpose Survey Meter - AEP 2153A, or High Range Probe - AEP 2163 are calibrated in mR/h or R/h units. Beta Survey instruments such as the AEP 2169, or AEP 5210 are calibrated in mrad/h or rad/h units.

Before gamma and beta doses can be compared, or added, they must be converted to a common unit, normally the mrem or rem. If we assume unity Quality Factor and a rad/R ratio of one, the gamma exposures in mR and the beta exposures in mrad are numberically equal to the corresponding Dose Equivalents in mrem. The Internal Whole Body Dose is calculated directly in mrem.

$$\text{Whole Body Dose Equivalent} = |(\text{Exposure, mR}) + (\text{Internal Dose, mrem})|$$

$$\text{Skin Dose Equivalent} = |(\text{Exposure, mR}) + (\text{Internal Dose, mrem}) + (\beta\text{-ray Skin Dose, mrad})|$$

At CRNL exposures to external radiation are measured and reported to individuals by the Health Physics Branch, on form AECL 1782. (See Appendix III, page 220).

Internal doses are calculated and recorded by the Medical Division, but at present tritium results only are added into the totals on the form AECL 1782.

9.4.4. Safety Factor

When an estimate has been made of the exposure likely to be received by an individual entering a radiation field, based upon information obtained from survey meter readings, the surveyor must ensure that the proper dosimeters are worn. The film dosimeter (Section 7.1.1.) will provide a permanent record of the exposure, while a pocket ion chamber dosimeter (Section 7.1.2.) will give an instant indication of the gamma exposure likely to be recorded on the film. However, since film dosimeter and pocket ion chamber dosimeter readings seldom agree exactly, it is customary to employ a safety factor, when estimating exposures. The safety factor used will depend upon experience gained with past exposures incurred under similar conditions, or if time permits by setting out test films and ion chamber dosimeters in order that a comparison of readings may be made before work in the area begins.

9.4.5 Controlling Beta Doses Using the Pocket Ion Chamber Dosimeter

The pocket ion chamber dosimeter used at CRNL for routine operations is a Stephens' 0.5R which is sensitive to gamma radiation only. When a worker enters a field where both beta and gamma radiation are present, an estimate of the gamma dose can be read directly from his pocket dosimeter.

In order to estimate what beta dose he is accumulating at the same time, a beta reading should be taken using a Beta Survey Meter AEP 2169 or AEP 5210. A gamma reading should be obtained using a Multi-Purpose Survey Meter AEP 2153A. Divide the beta dose rate by the gamma dose rate to get the β/γ ratio. If the accumulated gamma dose recorded on the pocket ion chamber dosimeter is multiplied by the β/γ ratio, it will give a reasonable estimate of the accumulated beta dose.

9.4.6 The Inverse Square Law

In some circumstances it is not convenient or practicable to obtain a reading close to, or on

contact with a large source, even though this information may be desirable. Since gamma radiation decreases in proportion to the square of distance from a source, the following formula may be used to calculate the intensity of gamma radiation fields at varying distances from a source:

Formula

$$\frac{I_{d_1}}{I_{d_2}} = \left(\frac{d_2}{d_1}\right)^2 \quad \text{or} \quad I_{d_1} = I_{d_2} \times \left(\frac{d_2}{d_1}\right)^2 \quad \text{or} \quad I_{d_2} = I_{d_1} \times \left(\frac{d_1}{d_2}\right)^2$$

Where: d_1 is the distance from the nearer point to the source.

d_2 is the distance from the more remote point to the source.

I_{d_1} is the intensity at distance d_1.

I_{d_2} is the intensity at distance d_2.

Examples:

(i) Find the intensity of the radiation from a point gamma source at 2 feet, if a reading taken at 10 feet is 200 mR/h.

$$I_{d_1} = I_{d_2} \times \left(\frac{d_2}{d_1}\right)^2$$

$$= 200 \times \left(\frac{10}{2}\right)^2 \text{ mR/h}$$

$$= 200 \times 25 \text{ mR/h}$$

$$= 5000 \text{ mR/h or } 5 \text{ R/h}$$

(ii) Find the intensity of radiation from a point gamma source at 20 feet, if a reading taken at 2 feet is 600 mR/h.

$$I_{d_2} = I_{d_1} \times \left(\frac{d_1}{d_2}\right)^2$$

$$= 600 \times \left(\frac{2}{20}\right)^2 \text{ mR/h}$$

$$= 600 \times \frac{1}{100} \text{ mR/h}$$

$$= 6 \text{ mR/h}$$

<u>Alternative Method</u>

If you know the radiation at a certain distance from a gamma source, the radiation at one unit of distance from the source will be the known radiation multiplied by the distance squared.

When you know the radiation at 1 unit of distance from the gamma source, the radiation at any other distance will be that radiation divided by the distance squared.

Example:

Find the intensity of radiation from a point gamma source at 2 feet, if a reading taken at 10 feet is 4 R/h.

Radiation at 1 foot $= 4 \times 10^2$ R/h

$= 400$ R/h

Radiation at 2 feet $= \frac{400}{2^2}$ R/h

$= 100$ R/h

9.5 Shielding

Shielding is one of the basic tools used to protect man from radiation. It decreases the intensity of radiation by absorbing it partially or completely. Many otherwise impossible operations can be performed in complete safety if a sufficient quantity of the proper type of shielding is used.

There are two main methods of shielding. One is the close method, where each radioactive source is shielded individually. This method is of greatest value with the less penetrating radiations such as alpha and beta. The other is a remote method, where the area around several sources of radiation is

surrounded with a shield. The remote method is used with highly penetrating gamma radiation and with neutrons. Regardless of the method of shielding, shields must be placed so as to protect anyone who might be above, below, or on any side of the source.

9.5.1 Alpha Shielding

Alpha particles can be shielded more easily than any of the other types of radiation. They can be stopped completely by about 1½ inches of air, a sheet of paper or a 1/64-inch thick aluminum sheet. Clothing or rubber gloves give adequate external protection against alpha radiation.

9.5.2 Beta Shielding

Beta particles are much more penetrating than alpha particles, therefore the materials which provide adequate protection against alpha radiation give almost no protection against beta radiation. Wool or cotton clothing will absorb only about 9% of beta particles with an energy of 4 MeV.

Beta particles lose their energy by interaction with the orbital electrons of the atoms of the

material through which they pass. Thus, it would seem that the most effective beta shielding materials would be dense, with closely packed orbital electrons. However, another result of the interaction of beta particles with orbital electrons is the production of X-rays called Bremsstrahlung. Since the quantity of X-radiation produced is directly proportional to the square of the atomic number of the absorber, it is preferable to use materials of low atomic number for beta shielding.

Some of the materials which are used effectively for shielding beta radiation are glass, plastics, plywood, masonite and aluminum. (Appendix IV, page 221). The air between the radiation source and the worker will act as beta shielding, to a certain degree. Beta particles of 1.0 MeV energy will travel about 12 feet in air, at 3.0 MeV about 23 feet, and at 5.0 MeV about 62 feet. The maximum range of beta particles most common at CRNL is about 10 feet in air.

9.5.3 Neutron Shielding

Neutrons are generated during the fissioning of uranium and plutonium in reactors and are produced by various reactions in accelerators. Neutrons are released from beryllium when it is irradiated with alpha particles. Portable neutron sources of this type are found in many laboratories at CRNL. They are made up of mixtures of beryllium (Be) with one of the alpha emitters: radium (Ra), polonium (Po), actinium (Ac), or plutonium (Pu).

Neutrons released during fission in a reactor have energies as high as 10 MeV. They are slowed down by collision with the nuclei of the atoms of the materials through which they pass, and are finally absorbed by radioactive capture. Neutron capture is usually accompanied by a release of gamma radiation as in the fission process itself. Thus, shielding against fission neutrons also involves shielding against gamma radiation. The shielding material used must counteract both types of radiation.

Materials with a high content of hydrogen are effective for neutron shielding. One of the cheapest hydrogenous materials is concrete. To improve its shielding characteristics, ordinary concrete is usually reinforced with iron ore, iron punchings, or iron shot. This "heavy concrete" is used as bulk shielding around reactor cores. If a portable shield is required lighter materials must be used. The best light-weight moderators are beryllium and carbon. Lithium and boron have high capture cross-sections for neutrons. Effective mobile shields may be constructed by laminating the above materials. At CRNL boxes of paraffin wax and laminated blocks of steel and masonite are used to shield radiation areas around experimental equipment at the face of the NRX and NRU reactors.

The effectiveness of materials in slowing down and capturing neutrons varies widely from one material to another and even from one isotope to another of the same substance. It also depends strongly on neutron energy. Hence, it is difficult

to specify the range of neutrons. When fast neutrons travel through ordinary water, 90% of them are slowed and captured after travelling about one foot.

9.5.4 Gamma Shielding

Gamma rays are electro-magnetic waves, similar to light, but are extremely penetrating. The distance gamma rays can travel through a material depends on their energy, but even weak gamma rays can be very penetrating. A gamma ray of 1 MeV energy is reduced to one-half its original intensity after passing through about 1/8 inch of lead. The effect of clothing on gamma radiation is negligible. Over 300 layers of wool or cotton would be required to reduce the intensity of a 1 MeV gamma ray by one-half. Materials such as lead, steel, concrete and water are normally used to shield gamma radiation.

9.5.4.1 Absorption of Gamma Rays

The absorption of gamma rays varies with the energy of the incident gamma ray, and with the Atomic Number of the shielding material. Three effects have been noted at different energy levels.

(a) Photo-Electric Effect

The gamma ray acts on the atom as a whole. By adding energy to the whole atom, it raises the atom to an excited state of higher energy than its normal or ground state. When the atom returns to its ground state by the emission of one of its orbital electrons, it is said to have emitted a photo-electron. That is to say, an electron is caused to be emitted by a photon.

Note:

(i) This effect occurs when gamma radiation of 1 MeV or lower strikes shielding material containing heavier elements.

(ii) It results in the final disappearance of the gamma ray.

(b) Compton Effect

A gamma ray may produce a shaking effect on an electron, imparting kinetic energy to it. This effect is known as the Compton Effect. The gamma ray acts on an electron only, as if it were a free electron, not on the atom as a whole (as in the photo-electric

effect), nor on the nucleus, (as in Pair Production).

By this process, the gamma ray transfers some, but not all of its energy into the kinetic energy of the electron with which it collided. Therefore, the gamma ray does not disappear completely, but continues to exist as a gamma ray of lower energy. The gamma ray will also suffer a change in direction depending upon the amount of energy lost.

Note:

(i) This effect occurs when gamma radiation with energies up to 5 MeV strikes any shielding material; the density of the material is not a factor.

(ii) It results in an electron being set in motion, and in loss of energy by the gamma ray, but not in the disappearance of the gamma ray.

(c) Pair Production

A gamma ray of sufficiently high energy may, under the influence of a nucleus near which it passes, be transformed into a pair of electrons, one positive and one negative. This is a complete transformation, according to Einstein's Law ($E = mc^2$), of the energy

of the gamma ray, into the masses and energies of the two particles. The gamma ray disappears. This reaction cannot take place unless the energy of the gamma ray is at least 1.02 MeV. The negative electron gives up its energy by ionization, in a relatively short distance, and may be forgotten. The positron, after it has lost some of its energy by ionizing, will recombine with an electron. This is the reverse of the process of pair production, and results in the liberation of two gamma rays (called annihilation radiation) each of 0.51 MeV, in opposite directions.

Note:

(i) This effect is of most importance at the higher end of the gamma ray energy spectrum. It occurs most frequently at energies greater than 4.8 MeV.

(ii) It does not occur for gamma rays whose energy is below 1.02 MeV.

(iii) It is followed by the emission of two 0.51 MeV annihilation gamma rays.

(iv) The two 0.51 MeV gamma rays are more easily absorbed by the photo-electric and Compton effects than the

1.02 MeV gamma rays. Since the two annihilation gamma rays are emitted in opposite directions, an average of half the energy of the original ray has been turned back into the shield.

9.5.4.2 Half-Value Shielding Calculations

The thickness of a shielding material required to reduce the intensity of a gamma ray to one-half of its original value is called a half-value layer (H.V.L.). This information for materials used frequently for shielding gamma radiation is available in various reference books. (See table on page 146.

In doing half-value shielding calculations, the following formula may be used:

$$A_p = \frac{A_o}{2^n}$$

where: A_p is the shielded intensity

A_o is the original intensity

n is the number of H.V.L.

(See also Section 13.7.)

9.5.4.3 Tenth-Value Shielding Calculations

The thickness of a shielding material required to reduce the intensity of a gamma ray to one-tenth of its original value is called a tenth-value layer. Information regarding tenth-value layers (T.V.L.) is also available in various reference books. (See table below).

Gamma ray shielding calculations may be done using tenth-value layers rather than half-value layers, or by using a combination of both tenth- and half-value layers (Section 13.8).

9.5.4.4 TABLE 8

Approximate Tenth- and Half-Value Thicknesses for Shielding Radioactive Sources

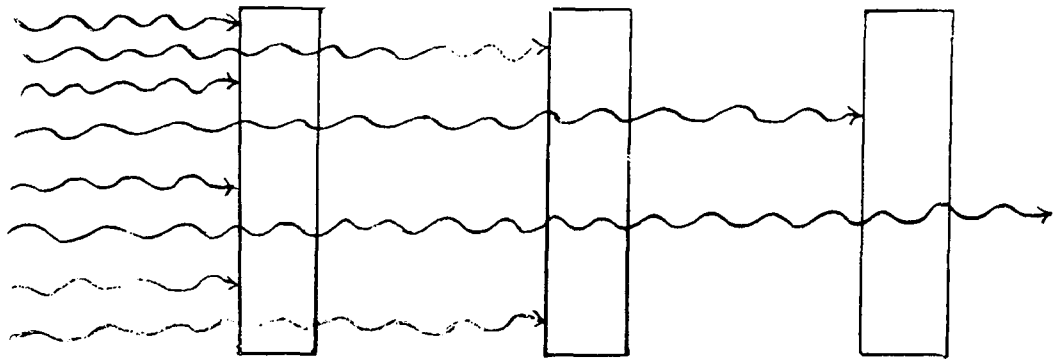
SOURCE	LEAD		IRON		CONCRETE	
	1/10	1/2	1/10	1/2	1/10	1/2
Co^{60}	1.62	0.49	2.90	0.87	9.0	2.7
Ra^{226}	1.85	0.56	3.03	0.91	9.6	2.9
Cs^{137}	0.84	0.25	2.25	0.68	7.1	2.1
Ir^{192}	0.64	0.19	-	-	6.2	1.9

Note: Tenth- and half-value thicknesses given above are in inches.

(See also Appendix V, page 222).

9.5.4.5

Exponential Absorption of Gamma Rays



If a given thickness of shielding material will reduce a gamma ray to one-half of its original intensity, a similar piece of shielding material twice as thick will reduce it to ½ of ½, or ¼, of its original intensity; and a piece three times as thick will reduce it to ½ of ¼ or 1/8 of its original intensity. Seven such half-layers will reduce the gamma ray to 1/128 or less than 1% of its original strength. Radiation absorption of this type is called exponential absorption, and would follow an exponential curve if plotted on a graph (Section 3.5).

9.5.4.6 Linear Absorption Coefficient

The linear absorption coefficient is a factor expressing the fraction of a beam of X- or gamma radiation absorbed in unit thickness of material.

The following formula may be used with Linear absorption coefficients to calculate shielding thicknesses for point gamma sources.

$$I = I_o e^{-\mu t}$$

where: I_o is the original intensity of the radiation.

I is intensity of the shielded radiation.

e is the base of the natural logarithms (2.718).

μ is the linear absorption coefficient.

t is the thickness of shielding in centimeters.

(See Appendix VI, page 223, and Section 13.9.)

10. INTERNAL EXPOSURE CONTROL

10.1 General

When radioactive contamination enters the body it irradiates all or part of the body until it decays or is excreted. Such internal irradiation may continue for days, weeks, or years. Thus, amounts of contamination which present no external radiation hazard can be extremely dangerous if they enter the body. Also, contamination that emits only alpha particles, or low energy beta particles is harmless until taken into the body; there it can be extremely hazardous.

There are three ways in which contamination can enter the body:

(a) Inhalation - Radioactive dust, gas, vapour, fumes or smoke can be inhaled into the lungs.

(b) Ingestion - Contamination can enter the digestive system if contaminated objects are placed in the mouth or allowed to touch the lips.

(c) Absorption - Radioactive contamination in the form of a vapour such as tritium can be absorbed through the skin. Contamination contained in solvents may also soak into the skin and be absorbed. Skin damaged by cuts, scratches or scrapes may provide an entry point.

Thus, to control internal exposures, precautions must be taken to prevent radioactive contamination from entering the body, in any of the ways mentioned above.

10.2 The Control of a Contaminated Area

When radioactive contamination is discovered, prompt action taken by a surveyor or a monitor can save a great deal of time and money. Also, the protective measures advised for those who must remain in the area, or the prompt removal and decontamination of others, can minimize radiation exposures or internal contamination of personnel.

As soon as possible, barriers should be erected to prevent the spread of contamination. Such barriers should enclose an area beyond the suspected area. Once the barrier has been established, a contamination check should be done around the perimeter, to ensure that all the contamination has been contained. Traffic in the area should be kept to a minimum.

If the incident was of a nature which could cause airborne contamination, protective masks should be worn until an air sample can be taken. Personnel should be monitored for contamination, and removed from the area as soon as possible. Contaminated clothing should be removed at the barrier. Replacement clothing should be made available as soon as the barrier is set up.

10.3 Air Monitoring

In order to prevent the inhalation of contaminated air, we must be able to determine the concentration of radioactivity in the air, in working areas, at all times. This is accomplished by air sampling. Then, guided by the recommendations of the International Commission on Radiological Protection (ICRP) regarding maximum permissible concentrations of radionuclides in air $(MPC)_a$, we are able to advise what precautions are necessary.

The $(MPC)_a$ of a particular radioactive material is that concentration in air to which a man could be exposed for 40h/wk, week after week, without absorbing sufficient radioactive material to result in an internal radiation dose greater than the maximum permissible dose, which is 5 rem/yr. for the whole body, 30 rem/yr. for the thyroid gland or bone, and 15 rem/yr. for other organs. Each hour of exposure to 1 $(MPC)_a$ may be considered as equivalent to receiving a radiation dose of 2.5 mrem for materials that irradiate the whole body, 15 mrem for those which are concentrated mainly in the thyroid gland or bone, and 7.5 mrem for those which are concentrated mainly in other single organs.

It is difficult to measure the concentration of radioactive material in air quickly and accurately, therefore, we must be very careful when using the $(MPC)_a$ values. In practice this means that respirators are usually worn whenever

a concentration greater than one $(MPC)_a$ is measured, or whenever a radiation surveyor deems it advisable on the basis of his experience. Respirators are worn during exposure to some airborne contaminants at a CRNL control level that is somewhat different than the $(MPC)_a$ value, as indicated in section 10.3.1.

10.3.1

Table 9

$(MPC)_a$'s and CRNL Control Limits

	$(MPC)_a$ dis/Min.m^3)	CRNL Control Limit dis/(min.m^3)
Strontium-90 (Sr^{90})	2,000	1,000
Iodine-131 (I^{131})	20,000	7,000
Short-lived beta emitters (half-life about 18 min.)	-	100,000
Tritium oxide (HTO or DTO)	11,000,000 (5 μCi/m^3)	same
Natural uranium (U^{nat})	150	same
Plutonium-239 (Pu^{239})	4	same
Alpha activity, if not known to be uranium	-	4

10.3.2 $(MPC)_a$ Hours

In order to assess the hazards of an exposure to airborne activity one should know not only the concentration

of activity in the air, but also the duration of the exposure. Thus, to obtain the number of $(MPC)_a$ hours of exposure, we multiply the number of $(MPC)_a$'s by the duration of the exposure in hours. This information can then be used to assess the hazards associated with a past exposure, or in deciding what precautionary measures should be taken in connection with an anticipated exposure.

For example, in the NRU reactor where workers are occasionally exposed to tritium hazards, if the anticipated exposure is less than 30 $(MPC)_a$ hours, an air supplied mask is worn with cotton clothing, but if the anticipated exposure is greater than 30 $(MPC)_a$ hours or if there is a chance of being wetted with tritiated heavy water, then an air-supplied mask is worn with a plastic suit. Air-supplied masks are worn whenever working with exposed heavy water regardless of the measured tritium concentration.

10.3.3 Long- and Short-Lived Radioactivities

Generally, long-lived radioactivities are more hazardous internally than short-lived radio activities, not only because of their slow rates of decay, but also because their biological elimination rates are usually relatively slow as well. The total dose to the affected organ varies with the energy of the radiation and the effective half-life of the isotope. The effective half-life is the time required for a radioactive element in the body to be reduced by one-

half, as a result of the combined action of radioactive decay, and biological elimination,

$$\text{Effective half-life} = \frac{\text{biological half-life} \times \text{radioactive half-life}}{\text{biological half-life} + \text{radioactive half-life}}$$

Thus, when sampling air, it is usually the radioisotopes with long half-lives that we are interested in measuring. However, there are some short-lived activities such as the daughter products of radon and thoron gases which exist in nature and are found in practically any location. Each consists of a group of several emitters of either alpha or beta particles.

The radon and thoron are produced by small concentrations of radium and thoron which are present in most soil and rocks. Therefore, the concentration of radon and thoron and their daughters is greater in air in areas where there is a dirt floor, or unpainted concrete.

Radon products have an alpha half-life of about 48 minutes. Thus, the contribution of the radon decay-products to an alpha air sample may be determined by recounting the sample a little less than an hour after the first count.

Thoron products show a distinct increase in alpha counts for several hours after sampling. They then assume a half-life of 10 to 11 hours. The initial build-up of activity is usually sufficient to identify thoron products.

The beta component of radon or thoron products in an average air sample is usually so small, in relation to the permissible levels $(MPC)_a$, that it can be ignored.

Another type of short-lived activity is encountered in the NRX and NRU reactors at CRNL, when a fuel element sheath ruptures. When newly-fissioned uranium is exposed, large quantities of krypton-88 (^{88}Kr) with a beta half-life of 2.8 hours may be released to the atmosphere. This activity, along with its daughter-product rubidium-88 (^{88}Rb) with an 18-minute beta half-life is not very hazardous. The $(MPC)_a$ has been set at 100,000 dpm/m^3, or 100 times the $(MPC)_a$ for mixed fission products. (See Section 10.3.1)

Although not too hazardous by themselves, these short-lived activities, if they are present in significant quantities, can mask the presence of the longer-lived, more hazardous fission products. When an air sample does not clearly show the 18-minute half-life, it should be considered to contain long-lived fission products, and the appropriate $(MPC)_a$ (1000 dpm/m^3) should be used.

10.3.4 Air Monitoring Equipment

Several different types of equipment must be employed to collect the various kinds of activity which may contaminate the air. The purpose, and operation of this equipment is explained below.

10.3.4.1 Continuous Air Monitors

(a) Moving-Tape Air Monitors (Figure 27)

This type is used to detect both alpha- and beta-gamma emitting particles in air. A vacuum pump pulls air

through a slowly moving tape of filter paper. Dust particles from the air are deposited on the tape. In the case of a beta-gamma sampler, the tape is monitored by a shielded end-window geiger. In the alpha sampler it is monitored by two alpha scintillation probes. The air activity is recorded on chart recorders.

(b) Single-Disc Beta-Gamma Air Monitors (Figure 28)

A vacuum pump pulls air through a 1 1/8 inch filter. The filter used for particulate activity is normally fibreglass (GF/A), but in areas where airborne radioiodine (^{131}I) is suspected, charcoal impregnated filters (ACG/B) are used. The airflow through these filters with a Gast Vacuum Pump, type 0210 or 0211, is approximately 2 m^3/h.

The filter holder is inserted into a two-piece lead shielding castle by operating a small crank at the rear of the mechanism. When the two pieces of the castle are in the closed position, the filter is held in a fixed position near the open face of a shielded end-window geiger. Air reaches the filter through an opening in the shielding near the face of the geiger.

On top of the shielding castle is a 1/6 rpm motor fitted with a cam. The cam operates a micro-switch which energizes a solenoid. When the solenoid is not energized, a .05 inch steel shutter drops down between the filter and the end-window geiger for 2 minutes. When the solenoid is energized, the steel shutter is withdrawn for 4 minutes.

This cycle is repeated as long as the power switch on the side of unit is switched on.

The measured activity is indicated on a rate-meter and a chart recorder. With the shutter between the geiger and the filter (2 min.), the reading indicates gamma background radiation. With the shutter withdrawn the reading is a combination of the gamma background reading and any beta activity collected on the filter disc.

Filters are changed every 24 hours during normal operations, or more frequently if an unusual concentration of activity is detected. The radioactivity on the filter may be examined further with other radiation measuring instruments.

(c) Personal Air Sampler (Figure 29)

The personal air sampler is a small portable unit consisting of a battery operated diaphragm pump connected by a piece of plastic tubing to a filter head. The filter head, which uses a 1 inch GF/A fibreglass filter, is pinned to the worker's lapel, to provide a "breathing zone" sample. The unit can be operated continuously. Batteries are rechargeable. The operating time is recorded on a register. Normal airflow using the GF/A fibreglass filter is 0.14 m^3/h.

(d) The Sequential Air Sampler (Figure 30)

A sequential sampler unit will take up to six consecutive air samples, without attention, over a maximum

period of 12 hours. The duration of each sample can be pre-set from 10 to 120 minutes. The total sampling time can be extended by the addition of extra units operating in series.

If samples are not required to be taken continuously, a delay can be programmed between samples. The sampler may be started manually, or automatically by a signal from a monitoring instrument. To collect beta-gamma samples, in-line filter holders containing a GF/A fibreglass 1 1/8 inch diameter filter are used.

To collect iodine samples either ACG/B charcoal-impregnated filter discs alone, or the iodine sample pack containing activated copper screens, ACG/B filters, and an activated charcoal cartridge are used (Figures 31, 32).

(e) Tritium Monitors (Figures 33, 34)

Ion-chamber type tritium monitors are used at CRNL. A vacuum pump pulls air through a filter and ion trap to remove radioactive dust and ions from the air, and thence through the outer of two concentric ion chambers of equal volume. The inner, sealed ion chamber responds to any gamma radiation that may be present, whereas the outer chamber responds to both gamma radiation and to tritium oxide (and to other gaseous activity that is drawn into the chamber). The difference in response between the two chambers is due to the tritium oxide (or other radioactive gases).

The responses of the two chambers are subtracted electronically, and the difference is registered on a meter calibrated in multiples of the $(MPC)_a$ for tritium oxide.

The AEP 10101 tritium monitor (Fig. 33) is mobile but not portable. Originally, it had 2 ranges - 5 $(MPC)_a$ full-scale on the low range; 50 $(MPC)_a$ on the high range. New amplifiers have now been fitted to several of these instruments to provide 10 ranges from 1.5 to 50,000 $(MPC)_a$ full-scale. The AEP 1498 tritium monitor (Fig. 34) is portable and provides ranges from 10 to 1000 $(MPC)_a$ full-scale.

Both these tritium monitors will respond to radioactive gases other than tritium. To determine if an indication is caused by tritium or by some other radioactive gas, air is drawn through a tube of silica gel before entering the tritium monitor. Any tritium oxide in the air is removed by the silica gel, whereas other radioactive gases pass through. Therefore, if the tritium monitor indication falls when the air is passed through silica gel, the indication is due to tritium oxide; if not, the indication is due to some other radioactive gas.

The tritium monitors indicate the tritium concentration in air at any time, but if one wishes to measure the average concentration over a long period, such as an hour or a day, one draws the air at a known rate through

ordinary distilled water in a glass bubbler. The tritium is washed out of the air as it bubbles through the water. After the desired sampling time, the water is analysed for tritium and the average concentration in air is calculated. (Figure 35)

10.3.4.2 Air Monitors for Spot Samples

If a continuous air monitor indicates an abnormal concentration of radioactivity, a spot sample may be taken to give a more accurate measurement.

(a) Beta-Gamma Activity in Air (Figure 36)

A Staplex air sampler with a 4.2 inch fiberglass filter is used to collect samples of airborne particulate radioactivity. The filter discs are then counted in a scintillation counter and the results are calculated in disintegrations per minute per cubic metre of air, dis/(min. m^3).

(b) Alpha Activity in Air (Figure 37)

The α-emitting decay products of radon (which is always in the air) are often present in a concentration high enough to completely mask the presence of airborne radionuclides such as plutonium, uranium or other long-lived alpha emitters. This difficulty is overcome by sampling the air with a Staplex air sampler fitted with an annular impactor. Ninety per cent of radon-decay-product activity is carried on dust particles less than 0.035 microns in diameter, and these particles are too small to be collected by

the impactor. On the other hand, when alpha-emitting materials become airborne by mechanical disruption from solid or liquid surfaces, most of the resulting particles are larger than one micron in diameter. The annular impactor collects particles greater than 0.5 microns in diameter. Thus, samples of long-lived radionuclides are collected while the short-lived radon-decay-products are by-passed.

The sample is deposited on a metal disc by the impactor and the number of disintegrations per minute is measured with an alpha scintillation counter. Knowing the volume of air that passed through the impactor, one can calculate the airborne activity in dis/min/m^3.

(c) Iodine

For a rapid spot check of iodine-131 in air, a Staplex sampler with a charcoal-impregnated ACG/B filter is often used. For a more accurate estimate of radio-iodine contamination in the air, a sample is taken for about 40 minutes using a vacuum pump with an iodine sample pack. (See Figures 31, 32.) The filters and cartridge are then counted using a gamma-ray spectrometer which is pre-set to measure the energies associated with iodine-131.

10.3.5 The Interpretation of Air Sample Results

Conditions during airborne contamination incidents and the effect on personnel in an area can vary widely.

Therefore, the recognition of all the implications is of utmost importance. Although the measured concentration of activity is important, it is often secondary in assessing the overall hazard. Some of the important factors to be considered are listed below.

10.3.5.1 Origin of the Airborne Contamination

The origin of the airborne contamination should be located as soon as possible, and its release stopped. Location of the origin of the release may give a good indication of the type of radioactivity present, and of the length of time the condition may have existed before detection. The length of time the condition existed before detection will indicate which personnel may have been exposed, and who should, therefore, be asked to submit urine samples.

10.3.5.2 Duration of Exposure to the Airborne Contamination

In assessing the hazard to exposed personnel, it is necessary to consider not only the concentration of the radioactivity, but also the duration of the exposure. It is customary to measure concentrations of airborne contamination in disintegrations per minute per cubic meter of air (dpm/m^3). If we divide this figure by the Maximum Permissible Concentration in Air $(MPC)_a$ for the particular isotope(s) present, then we will know the concentration in $(MPC)_a$'s. However, this result does not really tell us the true extent of the hazard until we multiply it by the number of hours of

exposure to obtain an answer in $(MPC)_a$-hours.

10.3.5.3 The Possibility of Recurrence

When a positive air sample result has been obtained, and the cause determined, the possibility of radioactivity being released again should be considered. If such a possibility exists, a careful watch should be maintained, either by the use of a continuous air monitor with a recorder, or by a series of spot samples.

10.4 Respiratory Protection

When concentrations of airborne contamination exceed permissible levels (See Section 10.3.1) it is necessary for anyone who may have to enter such areas to wear respiratory protection. Respirators are also worn in areas where it is anticipated that concentrations of airborne contamination beyond permissible levels will result from operations to be performed.

10.4.1 Types of Respirators

(a) Filter Respirators (Figures 38, 39)

Two CRNL respirators, the full-face and the Comfo are fitted with paper and charcoal filters which purify contaminated air. The full-face respirator is preferred in higher levels of air contamination, or where beta radiation is a hazard to the eyes. The filters of these respirators are effective against dust contamination, radioiodine, and most other vapours, but they are not recommended for protection against tritium.

(b) Air-Supplied Respirators (Figures 40, 41, 42)

When tritium vapour is present in the air, protection is provided by air-supplied masks or hoods. The Willson mask and the CRNL hood are supplied with air from building air lines, while the M.S.A. mask is supplied from a cylinder of compressed air carried on the user's back. The Willson mask is being replaced by the CRNL hood for many routine operations, while the M.S.A. mask is normally reserved for emergency use. Both masks and the hood provide excellent protection against all types of air contamination.

10.4.2 Respirator Fitting and Testing

The respirator laboratory of R & IS Branch is responsible for fitting and testing respirators at CRNL. Anyone who may be required to wear a respirator in the course of his duties is fitted with both a full-face and a Comfo respirator. The fit is checked on the respirator test apparatus shown in Figures 43, 44.

This apparatus injects a salt aerosol into a plastic hood. The person being fitted wears a respirator while standing with his head and shoulders in the hood. He is asked to move his head about, and do exercises to simulate normal working movements. The exhaust from the respirator is fed to a flame photometer. If there is any leakage due to a faulty respirator, or the fitting, this

will be indicated on a meter on the front of the test apparatus.

Employees are then informed what type of respirator they should wear. Upon the request of a Branch Head, a personal respirator is issued to an employee who may be required to remain on duty during an emergency. The condition of personal respirators is checked by the respirator laboratory staff at frequent intervals.

10.5 Protective Clothing

When workers must enter a contaminated area, they are provided with protective clothing. Cotton clothing is worn in areas where slight contamination may be encountered. Plastic clothing (Figures 45-49) is worn in areas that are highly contaminated with radioactive dust or tritium, or where radioactive liquids or contaminated ordinary or heavy water may spill on a worker.

Decontamination operators usually help workers remove contaminated clothing. The most highly contaminated outer clothing is removed first and bagged. Respirators must be worn until this is done. Any remaining clothing and the skin should then be monitored for contamination.

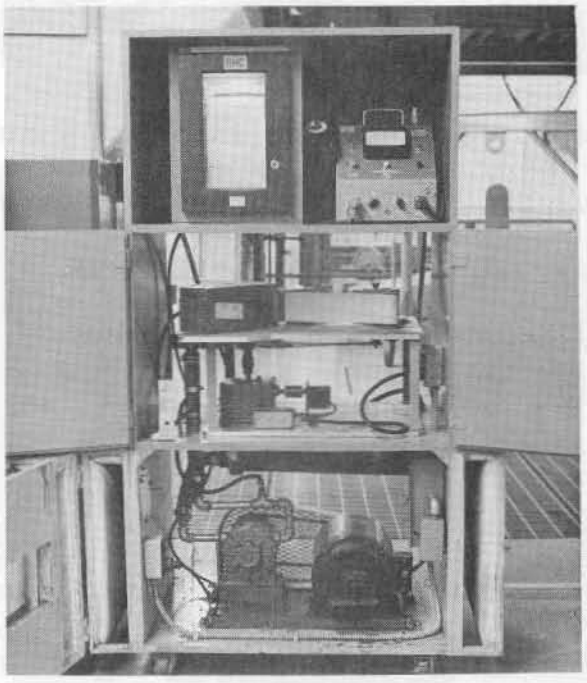


FIGURE 27
Moving Tape Air Monitor (Beta-Gamma)

FIGURE 28
Single-Disc Air Monitor (Beta-Gamma)

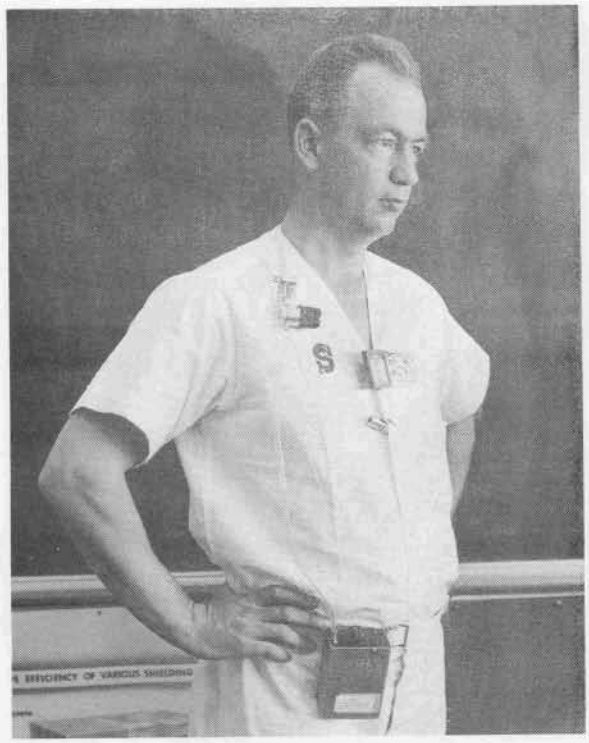


FIGURE 29
Personal Air Sampler

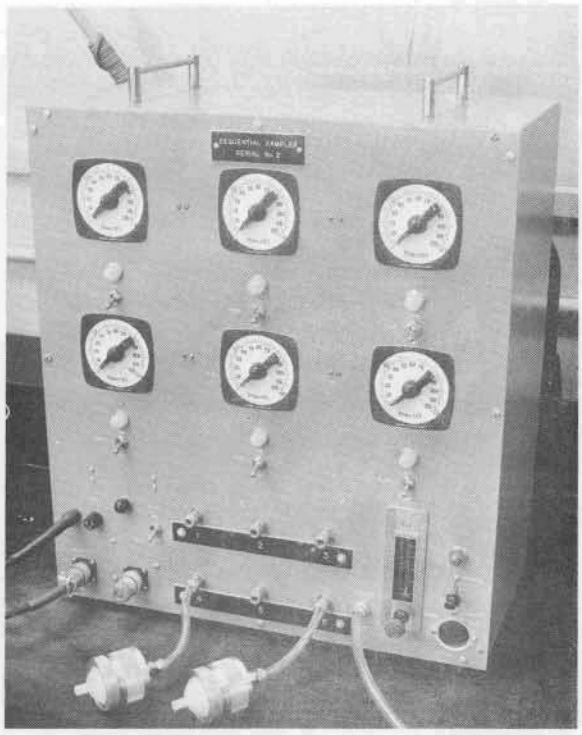


FIGURE 30
Sequential Air Sampler - May-Pack Iodine Filters

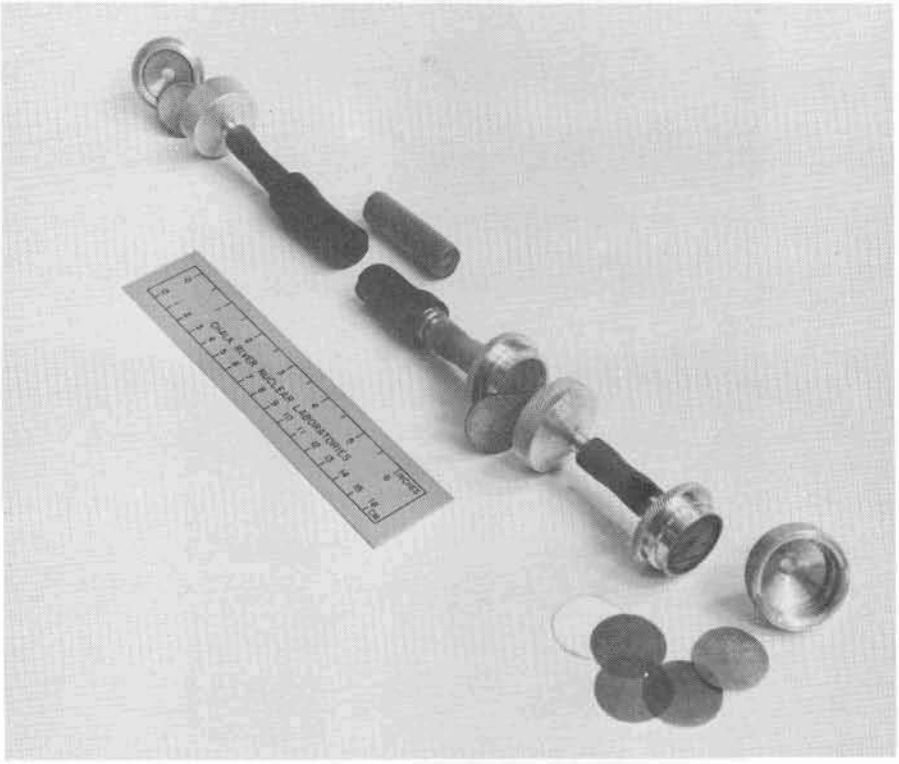


FIGURE 31
Iodine Sample Pack

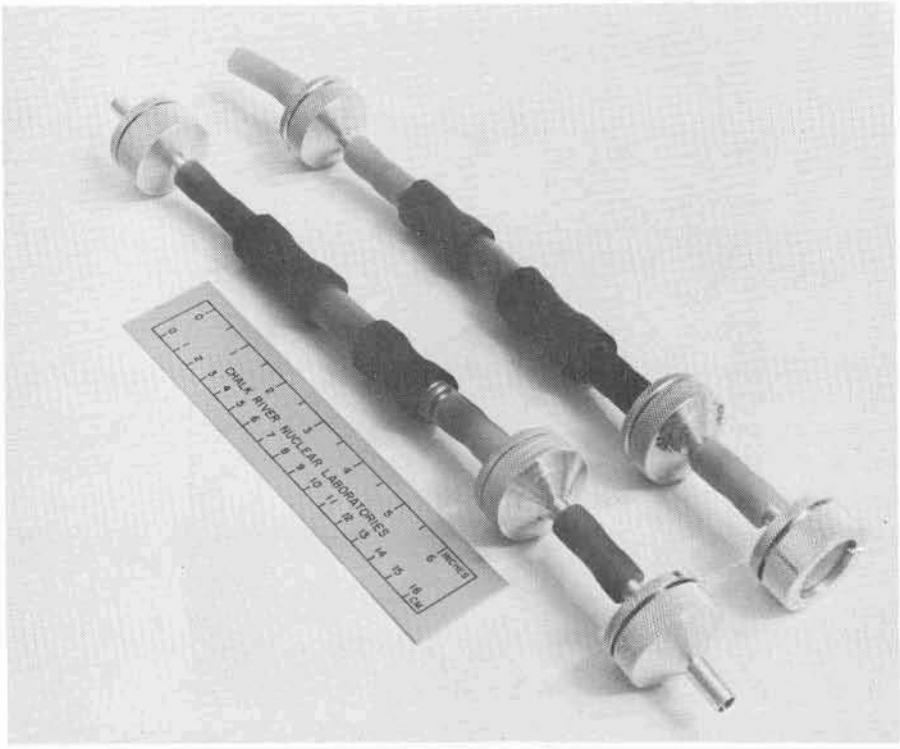


FIGURE 32
Assembled Iodine Sample Pack

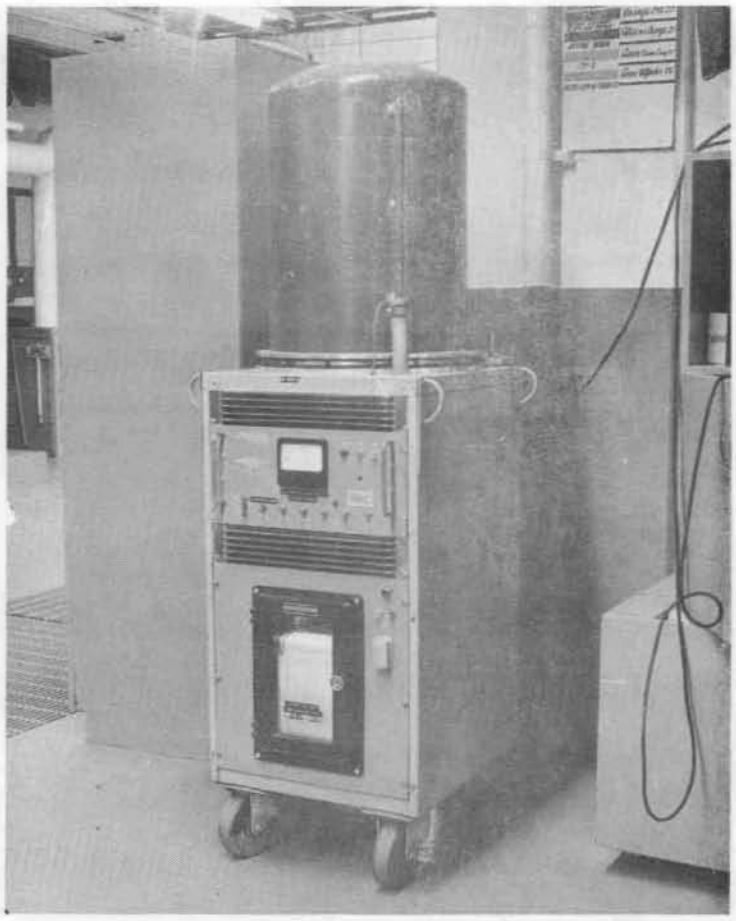


FIGURE 33
Tritium Monitor AEP 10101

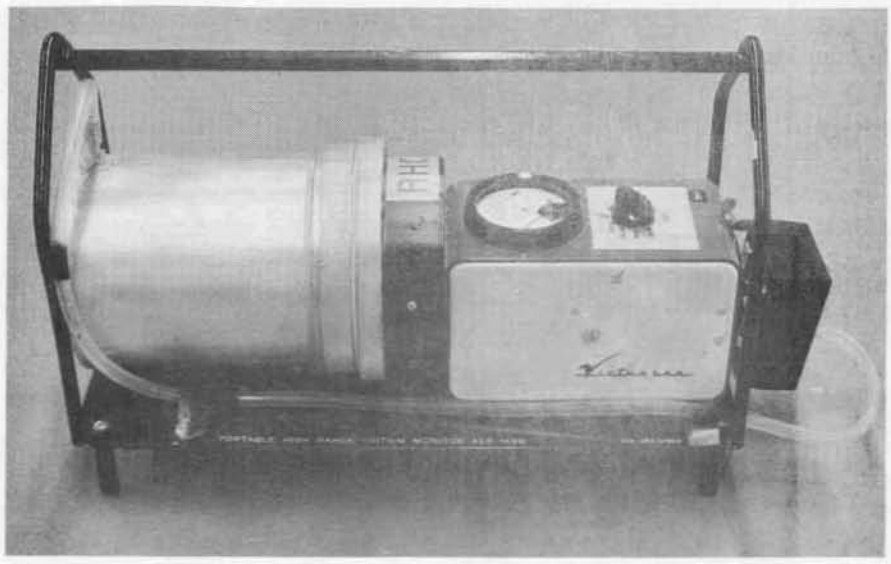


FIGURE 34
Portable Tritium Monitor AEP 1498

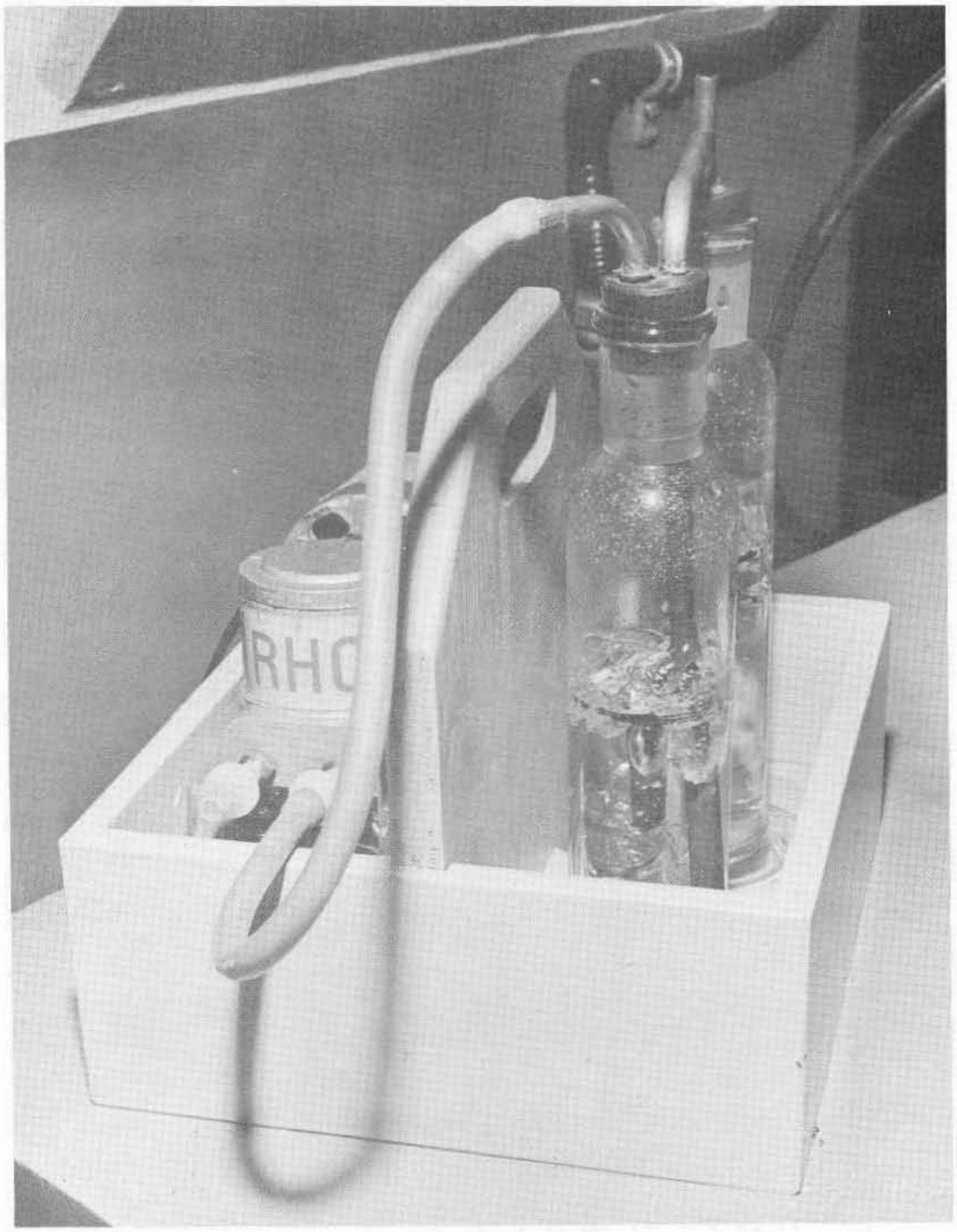


FIGURE 35

Tritium Bubbler

FIGURE 36
Staplex Sampler - GF/A Filter (Beta-Gamma)

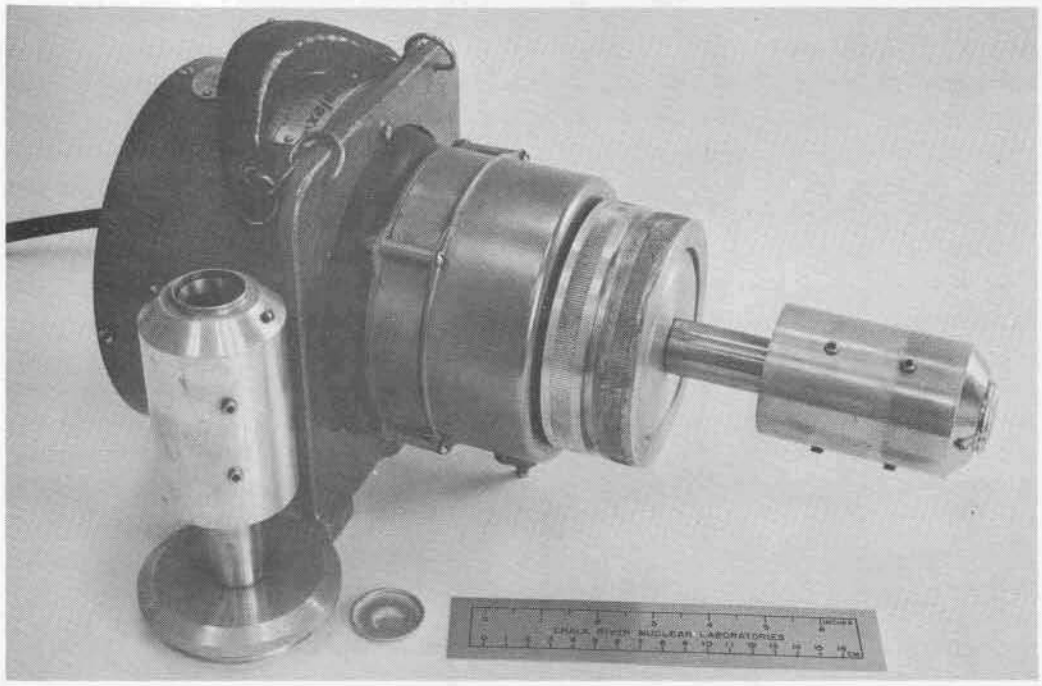


FIGURE 37
Staplex Sampler - Annular Impactor (Alpha)

FIGURE 38
Full-Face Respirator

FIGURE 39
Modified Comfo Respirator

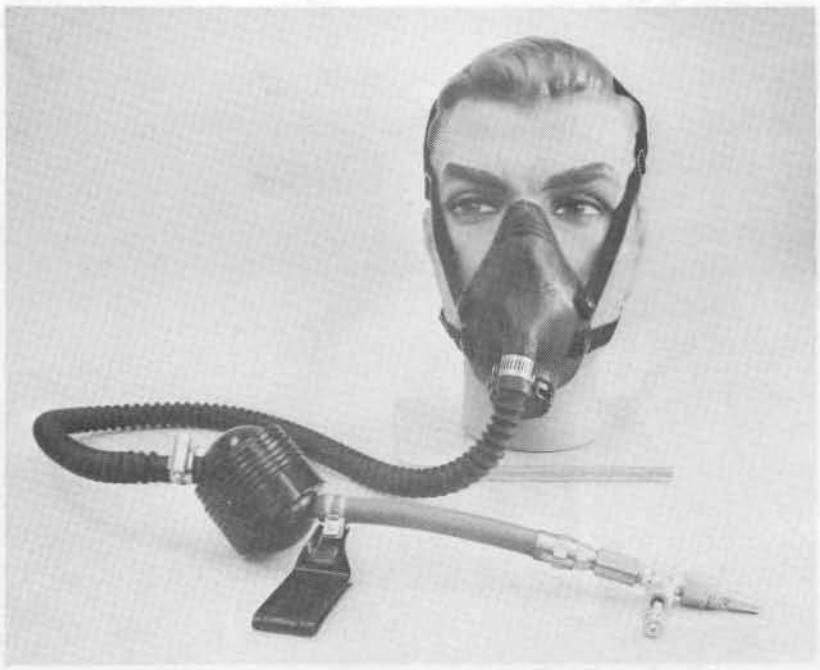


FIGURE 40
Willson Air Mask

FIGURE 41
Air-Supplied Hood

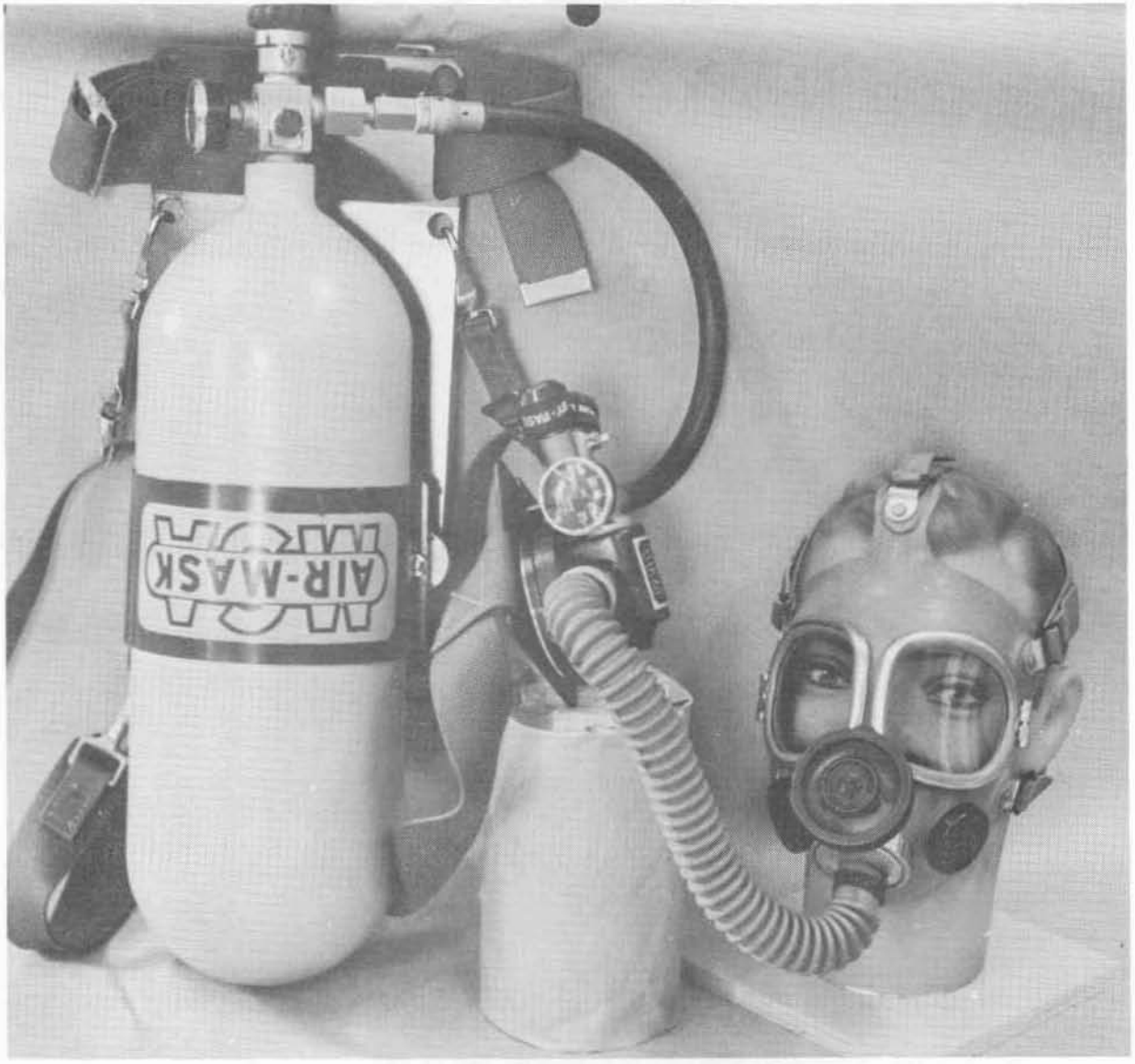


FIGURE 42

M.S.A. Demand Air Mask

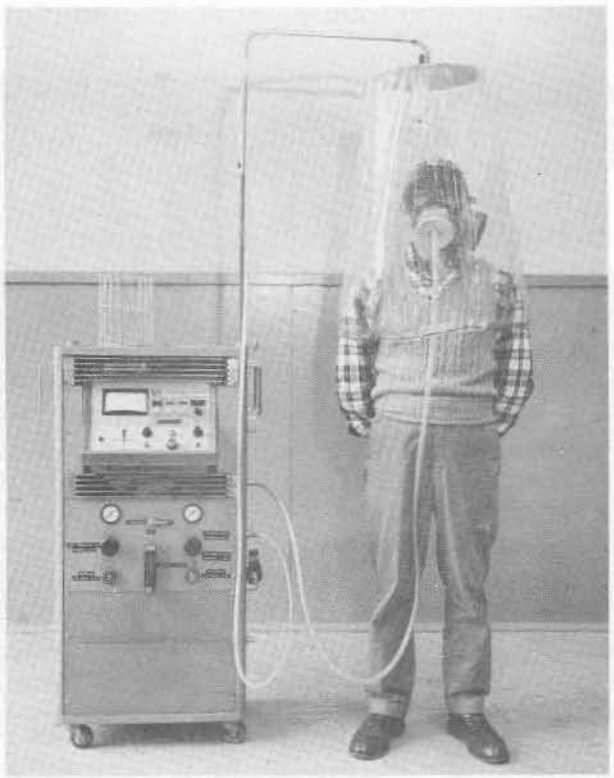


FIGURE 43
Respirator Test Apparatus Fron

FIGURE 44
Respirator Test Apparatus (Rear)

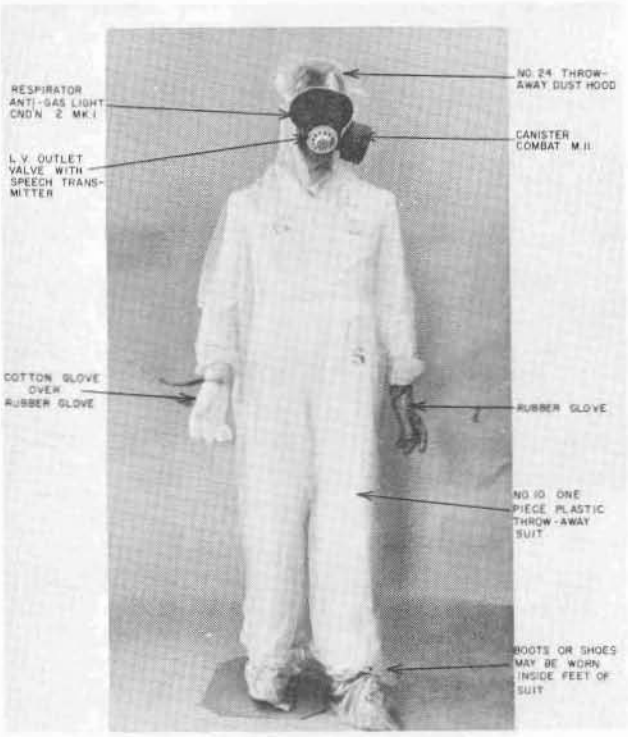


FIGURE 45
Disposable Plastic Suit

FIGURE 46
Waterproof Plastic Suit - Supplied-Air Hood

FIGURE 47
Air-Cooled Plastic Suit

FIGURE 48
Air Harness for Cooling Plastic Suit (Figure 47)

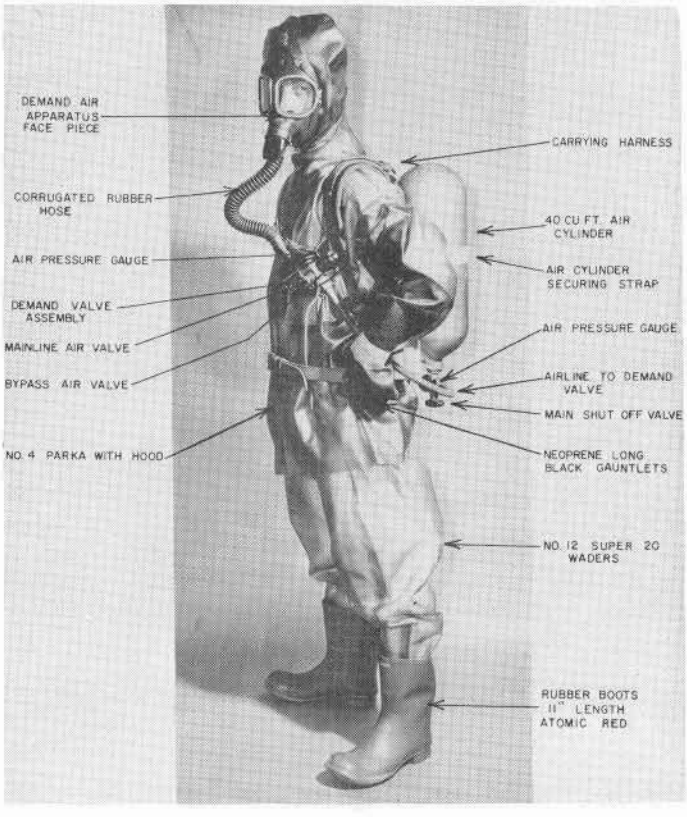


FIGURE 49
Waterproof Plastic Suit - M.S.A. Demand Air Apparatus

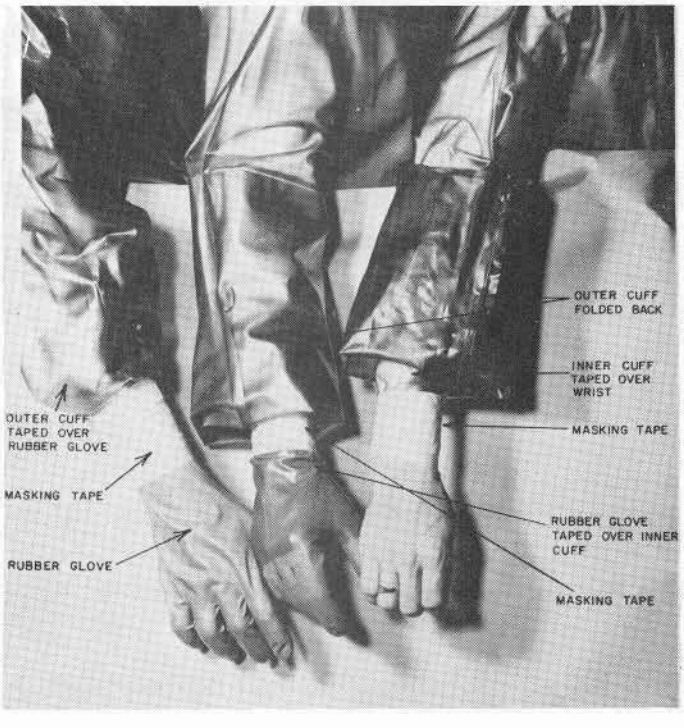


FIGURE 50
Fitting Rubber Gloves With Waterproof Plastic Suit

11. DECONTAMINATION

Decontamination is the removal of radioactive material from surfaces. Much valuable equipment is salvaged for reuse by decontaminating it.

Decontamination personnel are employed in many Active Area buildings to help control contamination. They conduct swipe checks for contamination throughout working areas on a routine basis. If contamination is encountered, they clean the area immediately, or rope it off until cleaning can be accomplished.

If radioactive material spills or leaks out of a system, high levels of radiation will result. In many cases, rooms or equipment must be decontaminated before maintenance groups are able to undertake repairs.

Large items of equipment that become contaminated are usually decontaminated in place.

11.1 The Decontamination Centre

Items of equipment that become contaminated, if they can be wrapped and transported readily, are

sent to the decontamination centre (Building 507) for cleaning. Larger items such as heavy water drums, NRU rod plug-ends and lock-sections are decontaminated in Building 468, the Vehicle Decontamination building, adjacent to Building 507.

A radiation surveyor or contamination monitor should check all items before they are sent to the decontamination centre. He will advise how the article should be wrapped for shipment, and will attach to it a decontamination tag listing the building of origin, the material, the radiation and contamination levels, and any special precautions to be taken during cleaning. The tag is signed by the owner and by the surveyor or monitor. (See Appendix VIII, page 227).

Deliveries to and from the decontamination centre are made routinely at a frequency determined by the needs of each building. The driver who transports the article to the decontamination centre returns the bottom portion of the decontamination tag as a receipt. He takes the receipt back to the decontamination centre

on his next visit, and if the article has been decontaminated, he returns it to its owner.

11.2 Decontamination Processes

If an article is lightly contaminated, it can often be decontaminated by scrubbing with hot water and detergent.

If an article is heavily contaminated, it is sometimes flushed in a large washer before regular decontamination is attempted. Steam and detergent cleaning has proven effective in many cases. In other cases, ultra-sonic cleaning, which employs high frequency vibrations passing through a tank filled with hot water and detergent, has produced good results. Rubbers and plastic clothing are washed in large commercial washing machines.

The decontamination processes are not always completely effective, so articles are checked for both loose and fixed contamination. A decontamination process tag is attached to indicate the condition of the article after it has been "decontaminated".

A pink tag indicates that all loose contamination has been removed. If fixed activity remains, the type and amount are shown on the tag. (See Appendix VIII, page 227).

A red tag indicates the presence of loose contamination. The location, type, and amount are shown on the tag. If the contamination is on the outer surface, the article will be wrapped before being returned to its owner. (See Appendix VIII, page 227).

11.3 Decontamination of Personnel

When skin contamination is encountered in any area, an attempt should be made to remove it, as soon as possible, by washing with soap and water. A soft nail brush may be used, but care should be taken to avoid damage to the outer layer of skin. Abrasive or harsh cleansers should not be used.

If several washings with soap fail to remove the contamination a further attempt may be made by using Glucona Delta Lactone. This cleanser should be used liberally under running water.

If the contamination still persists, or if there is a cut, a sore or an abrasion in the contaminated skin area, the worker should visit the plant hospital for assistance as soon as possible.

Any person with a significant level of skin contamination should be asked to submit a urine sample for bioassay.

11.4 Aids to Special Decontamination Problems

The R&IS Branch Development Section maintains an adequate supply of equipment to assist decontamination personnel with any special problems which may arise in any area. Listed below are the items which are available:

(a) Mobile Ventilation Unit (Figure 52)

(b) Mobile Air Sampler (Figure 53)

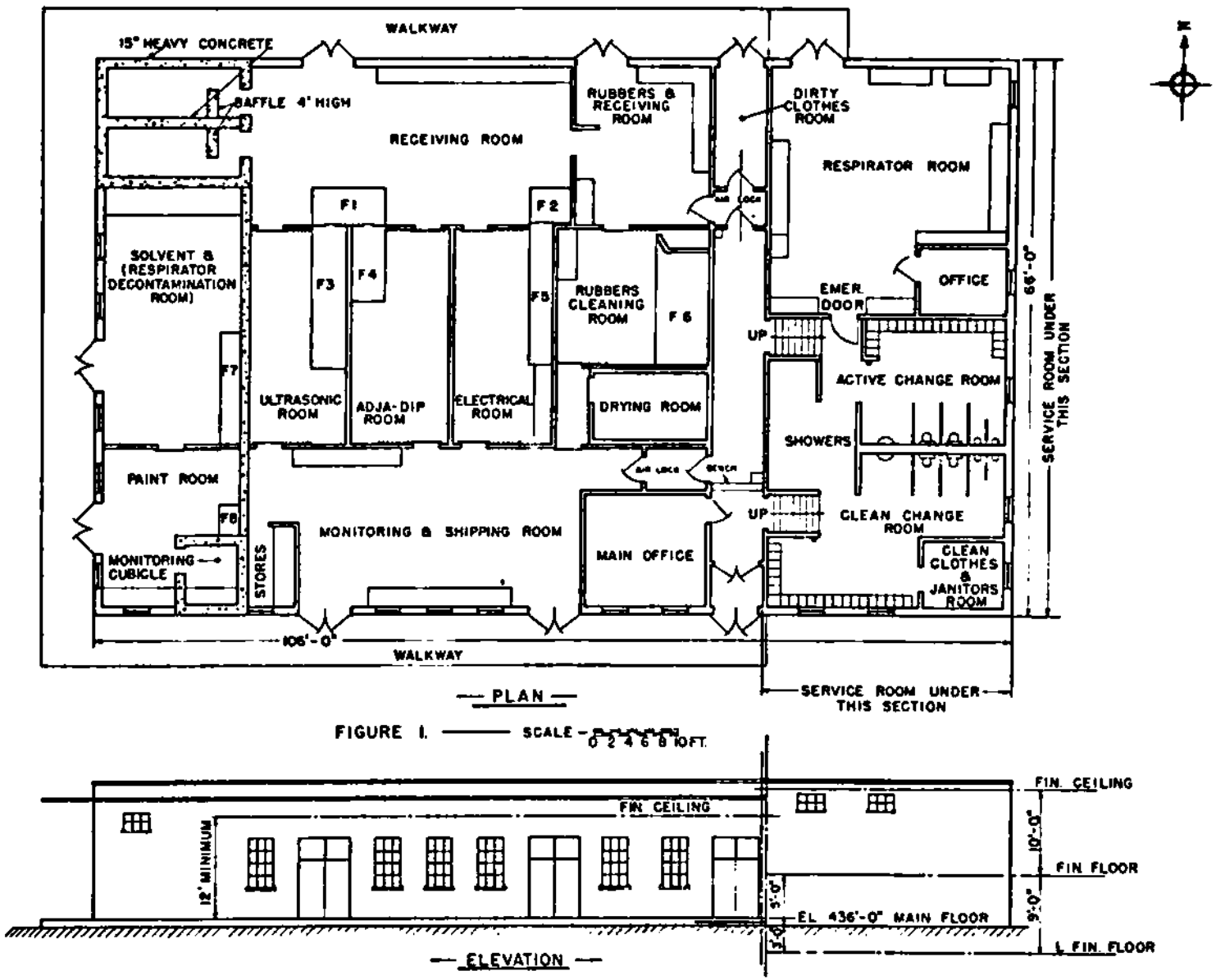
(c) Mobile Change Room (Figure 54)

(d) Vacuum Pick-Up and In-Line Filter Boxes (Figure 55)

(e) Portable Steam-Detergent Cleaner (Figures 56, 57)

(f) Remotely-Controlled Radiation Monitor (Figure 58)

(g) Filtered-Exhaust Vacuum Cleaners (Figures 59,60)



LAYOUT OF DECONTAMINATION CENTRE.

FIGURE 51

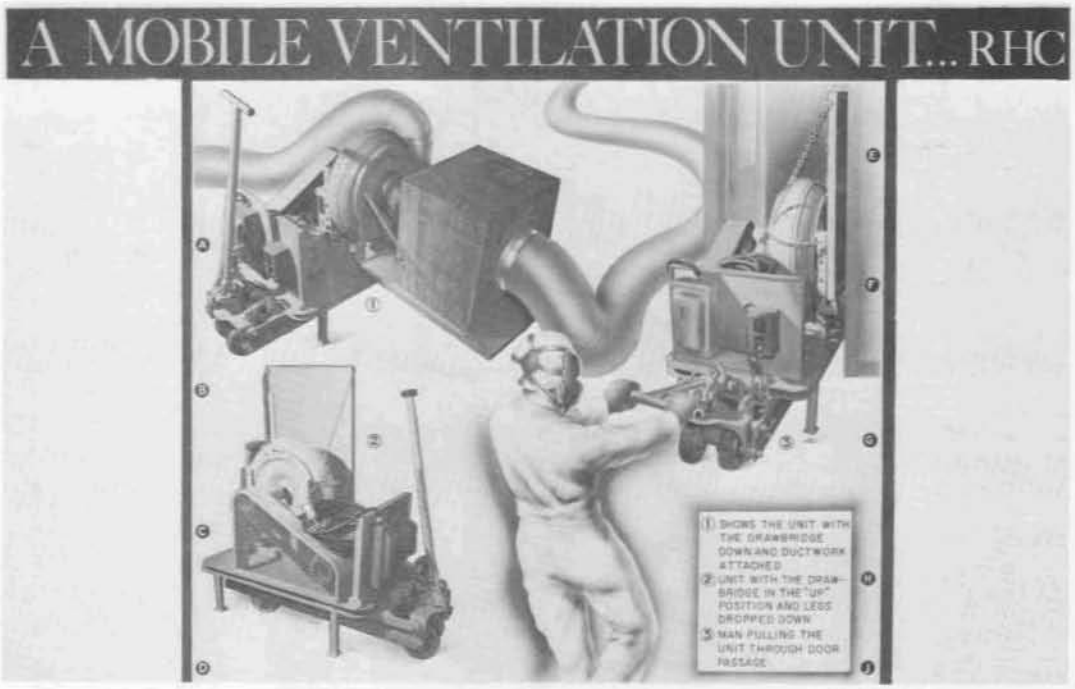


FIGURE 52

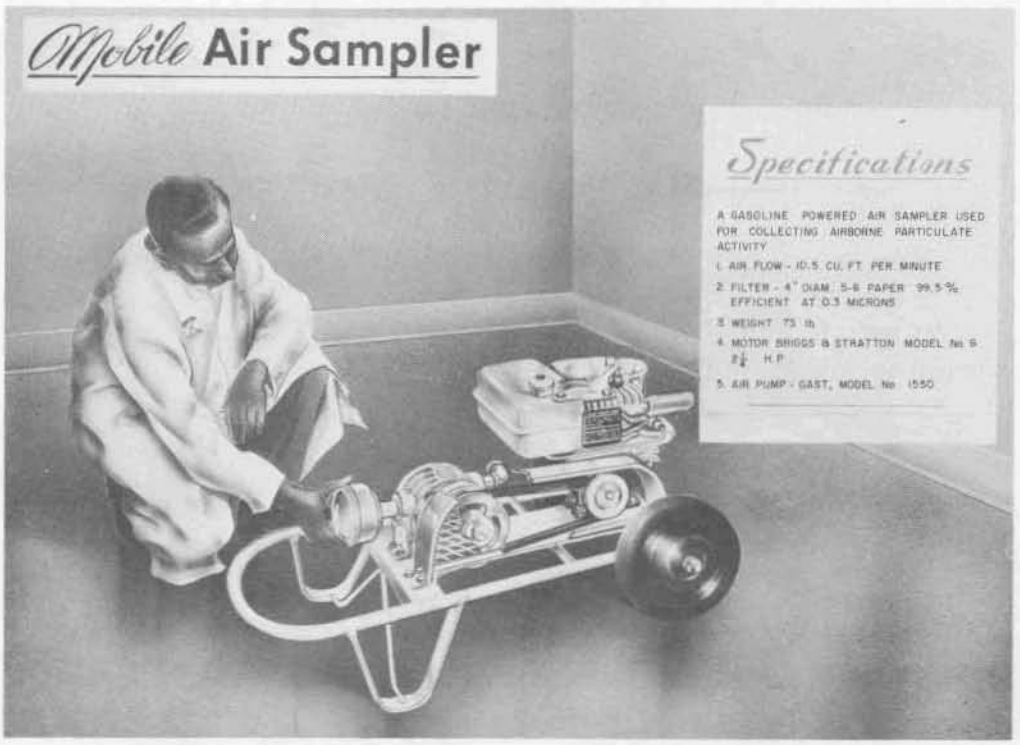


FIGURE 53

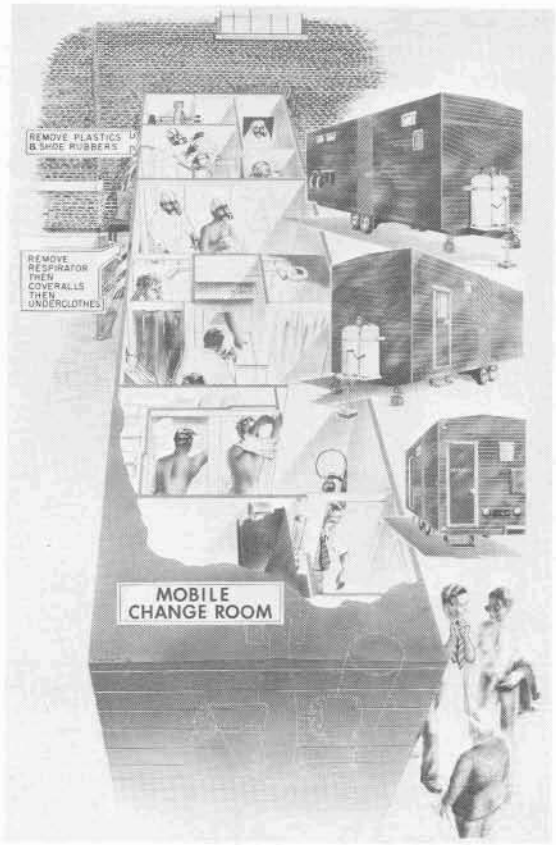


FIGURE 54

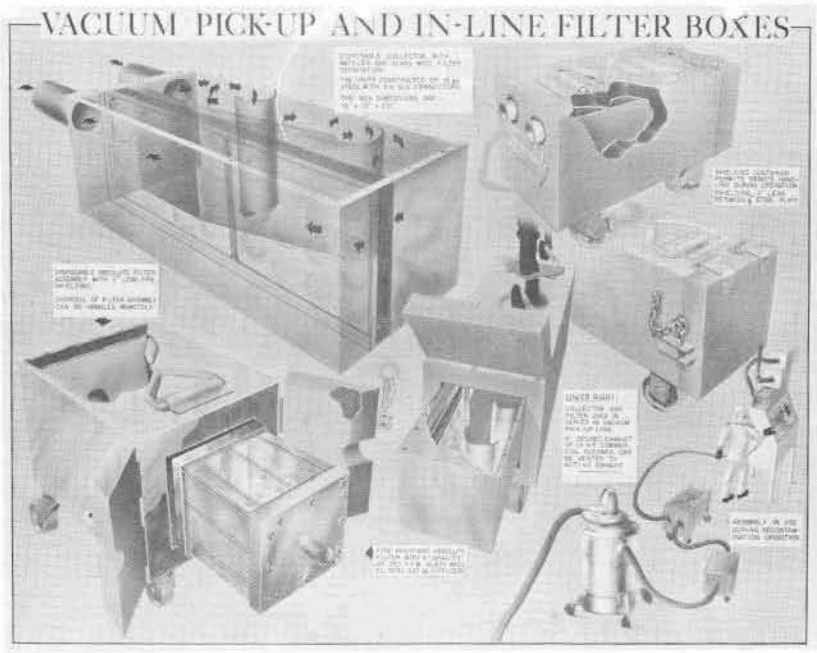


FIGURE 55

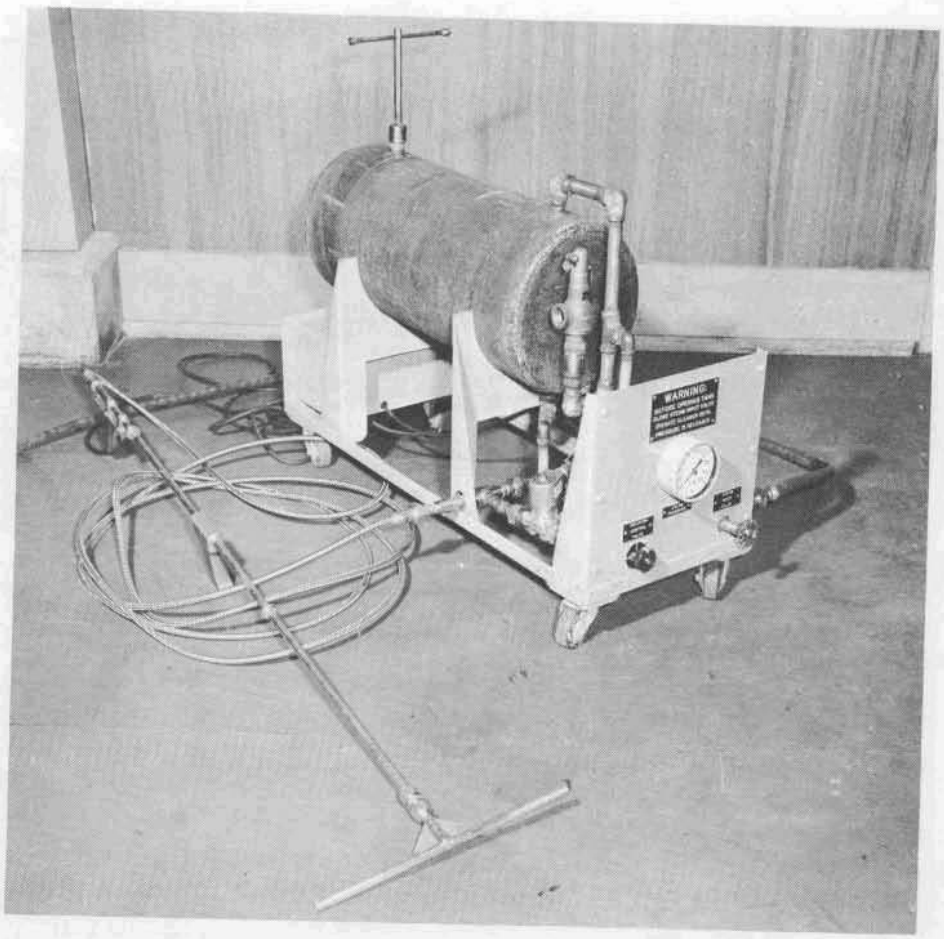


FIGURE 56
Portable Steam-Detergent Cleaner

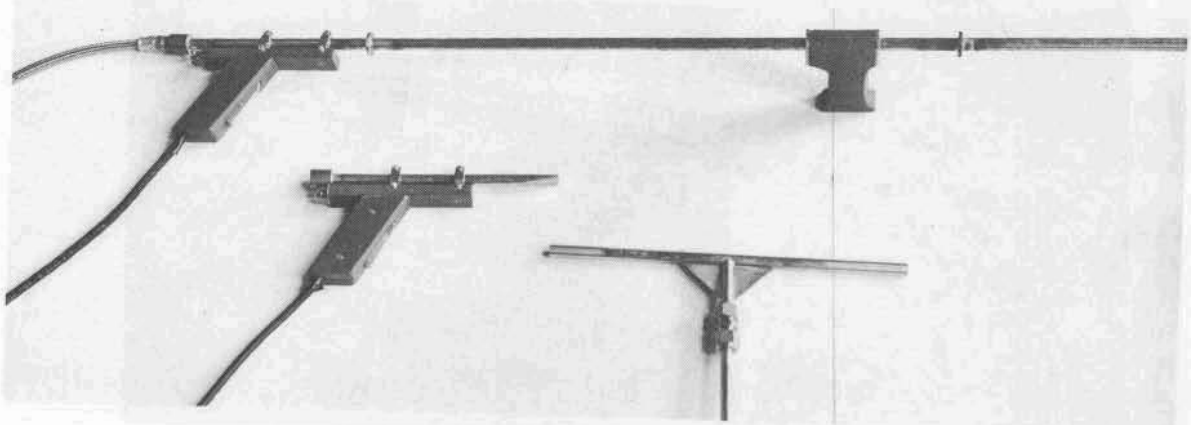


FIGURE 57
Nozzles for Portable Steam-Detergent Cleaner

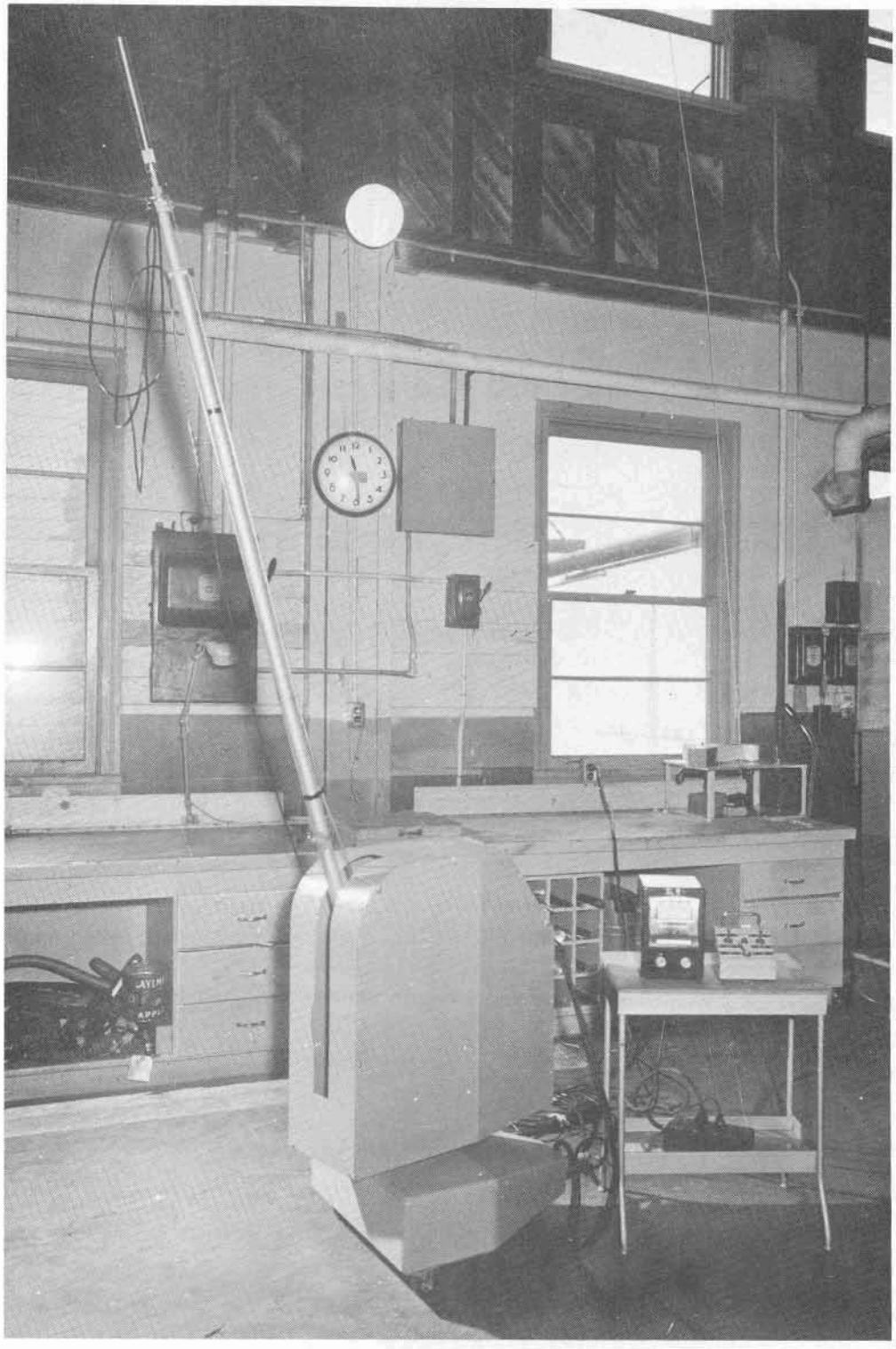


FIGURE 58
Mobile Remotely-Controlled Radiation Monitor (Gamma)

FIGURE 59
Filtered Exhaust Vacuum Cleaner

FIGURE 60
Filter for Vacuum Cleaner (Figure 59)

12. DISPOSAL OF RADIOACTIVE WASTES

A major problem created by the operation of an atomic energy plant is the safe disposal of radioactive wastes. They must be stored or buried in such a manner that radioactivity will not escape in quantities large enough to be a hazard to the public.

Solid wastes are classified as active or inactive at the various buildings. Active wastes are suitably packaged and are placed on separate stands, for collection and disposal by the Mechanical Services Branch. The R & IS Branch supervises the collection and disposal to assure that radiation hazards are properly controlled.

12.1 Disposal Areas at CRNL

There are three main disposal areas at CRNL. All of these are located near the main road between Chalk River and the Plant, and are fenced to exclude unauthorized personnel.

The "A" disposal area is situated about ½ mile from the Plant, and was used exclusively for the disposal of solid radioactive wastes until late 1952. Several thousand gallons of water, highly contaminated with fission product activity was pumped into a seepage pit in the "A" disposal grounds at that time. Since then, the area has been used mainly for the storage of large items of contaminated or obsolete equipment. The reactor vessel, removed

from NRX in 1953, is buried there. New seepage pits have been constructed nearby (Figure 67)

In 1954, the "B" disposal area was opened at a site about a mile beyond the "A" area near the road to Chalk River. Since that time it has served as the main area for the disposal of solid, and small quantities of liquid radioactive wastes. In 1962 a pilot plant incinerator was erected near the entrance to the "B" area, and was operated successfully for several months. When the equipment had been proven, the incinerator was shut down because of the cost of operation. (Figure 61)

In 1962 the "C" area was opened for the disposal of low-activity wastes, by burial in the soil. It is located near the main road to Chalk River, about 5 miles from the Plant.

12.2 Solid Wastes

12.2.1 Inactive Solids

Most solid wastes from the Inner Area are inactive. Many wastes from the Active Area are also inactive. For example, waste paper from offices, packing containers from new equipment, and other materials which have not entered the active sections of the buildings concerned.

Wastes that are known to be free of detectable contamination are placed on inactive disposal stands. They are then collected and burned.

12.2.2 Collection of Active Wastes and Preparation for Disposal

Laboratories and other buildings in the Active Area, and in some cases in the Inner Area, are supplied with active waste cans. These are usually ordinary garbage cans with an orange band painted around the middle to signify that they are used for active wastes. A double paper bag with a waxed inner liner is provided to prevent contamination of the can. (Figure 62)

These cans are used for dry, low-activity solid wastes, including materials which have seen service in the active section of Active Area buildings, or in some locations in the Inner Area. When the bags are full, they are sealed with masking tape by decontamination operators, placed on the "active" disposal stands, and are taken to the Disposal Area for burial. (Figure 63) Some laboratories are provided with cans with foot-operated lids so that wastes may be deposited in them without touching any part of the can with hands or gloves. (Figure 64)

Items for disposal must be packaged in such a manner that they can be handled safely by disposal personnel. There must be no loose contamination on the outer surface of the containers. Extensive use is made of sheet polyethylene or heavy re-inforced paper as a final wrapping.

Radiation levels must be stated on the forms which accompany each disposal. (Appendix IX, page 228).

12.2.3 Segregation of Active Wastes

Active wastes are segregated according to the radiation level on the waste packages.

If the radiation field measured with a Multi-Purpose Survey Meter (AEP-2153A), with the window open, is less than 100 mR/h at one foot from the waste container, the waste is buried in a sand trench in the Disposal Area. Most of the waste packages contain so little radioactivity that no reading can be obtained on the survey meter. Sand-trench disposals are classified as "routine". (Figure 65)

About 10% of the waste packages contain sufficient radioactivity that it is undesirable to bury them in the sand-trenches. If the radiation reading one foot from the package is greater than 100 mR/h, the package is normally placed in a concrete-lined trench. Disposals of this type are referred to as "special". The trenches are capped with concrete when they are full. (Figure 66)

Some intensely radioactive fuel wastes and other materials such as cobalt-60 are placed in special storage wells in the disposal area.

12.2.4 Precautions in Handling Wastes

Only dry wastes should be placed in disposal cans, since the paper liners will not contain liquids.

The activity level of such material should be low, otherwise workers in the vicinity might be exposed unnecessarily to radiation.

When removing dry wastes from a fume hood or glove box, first put them in a clean container (ice cream carton, polyethylene or paper bag) in the fume hood or glove box. Bring the disposal can as near as possible to the material that is to be thrown away. Do not carry active wastes from a fume hood on one side of the room to a disposal can on the other side of the room. The chances of spreading contamination are too great.

Disposal cans should not be handled while wearing contaminated gloves. Remove the gloves or cover the handles with kleenex or some other suitable material before touching the disposal can. (Figure 62)

12.3 Liquid Wastes

12.3.1 Inactive Liquids

Storm sewers in the Inner and Active Areas carry surface drainage directly to the Ottawa River.

Sanitary sewers in the Inner and Active Areas carry inactive wastes from toilets and other building drains to the sewage treatment plant, and thence to the river.

No radioactivity may be disposed of in these systems without authorization by the Waste Management Panel.

12.3.2 Low Activity Liquids

12.3.2.1 Disposal to the River (Figure 67)

Operation of the NRX reactor requires approximately 3200 gallons of water per minute to cool the fuel rods. The cooling water is slightly radioactive after passing through the reactor, due to neutron activation of oxygen and impurities. The cooling water flows through two delay tanks to permit the decay of short-lived products before it is discharged to the Ottawa River.

Wastes from the laundry, and condensate from some evaporators, are passed directly to the river via the process sewers.

Wastes from the decontamination centre are collected in a holding tank. They may be discharged to the river or pumped to a seepage pit in the disposal area, depending on the level of activity.

12.3.2.2 Disposal to Seepage Pits (Figure 68)

The waste farm disposal system consists of collection tanks (Buildings 242, 243) for wastes from active drains, a pumping station (Building 240), two underground disposal lines and seepage pits near the "A" Disposal Area.

The average volume of wastes handled from the

NRX and NRU reactors is around 600,000 gallons per month. The average volume from the chemical and research areas is around 200,000 gallons per month.

12.3.2.3 Emergency Disposal Pit (Figure 69)

An emergency storage pit is available in the Outer Area for use in the event of an incident necessitating the storage of a large volume of contaminated water. This pit is dug out of the sand and is lined with sheet polyethylene. It will hold 1,000,000 gallons of water, which can be pumped in by a pipeline from the liquid waste disposal system.

12.3.3 High Activity Liquids

12.3.3.1 Storage in Tanks

Several thousand gallons of high activity liquid wastes accumulated from the operation of the Chemical Extraction Plant. This plant recovered plutonium from the NRX fuel rods, but has not been operated for this purpose for some years. The liquid wastes are now stored in underground stainless steel tanks. Some wastes have been evaporated to reduce their volume.

An ion exchange plant is operated in Building 200 to deionize water from the NRX and NRU fuel storage bays. Approximately 100,000 gallons are processed per day and returned to the bays. The ion exchange columns are regenerated periodically. The contaminated regenerants are evaporated and stored in tanks.

Moderately active wastes from some of the storage tanks have been mixed with concrete, placed in steel drums, then sealed in a concrete monolith in the disposal area. (Figure 70)

12.3.3.2 Storage in Bottles

Small volumes of wastes from laboratory analyses are collected in shielded plastic bottles in fume hoods or glove boxes. The bottles are sealed in plastic bags and are placed in stainless steel cans on the disposal stands. A statement of the amount of activity in each bottle is required by the R & IS Branch. These wastes are buried in an asphalt-lined pit at the disposal area. (Figure 71)

12.4 Gaseous Wastes

The cooling air from the NRX and NRU reactors contains a substantial concentration of argon-41 whenever the reactors are operating. Fission product gases are occasionally present in the cooling air too, even if the reactors are shut down (See Section 3.2). Since these gases are short-lived, and are only mildly hazardous compared with some other radioisotopes, they can be discharged safely to the atmosphere. However, they must be well-diluted with atmospheric air to prevent hazardous concentrations being encountered by people in the vicinity of the discharge. To accomplish the necessary dilution, the

gases are discharged from a high stack, ½ mile from occupied areas of CRNL. (Figure 72)

Reactor cooling air is passed through high efficiency filters, before being discharged up the stack, to remove any radioactive particles that might be released from the reactors.

Several smaller, lower stacks are used to discharge small quantities of radioactive gases from various CRNL laboratories. Those which could release dangerous radioactive particles are equipped with high efficiency filters.

FIGURE 61
Incinerator - "B" Disposal Area

FIGURE 62
Active Waste Disposal Can

FIGURE 63
Active Waste Disposal Trailer

FIGURE 64
Foot-Operated Dry Waste Disposal Can

FIGURE 65
Routine Disposals - Sand Trench

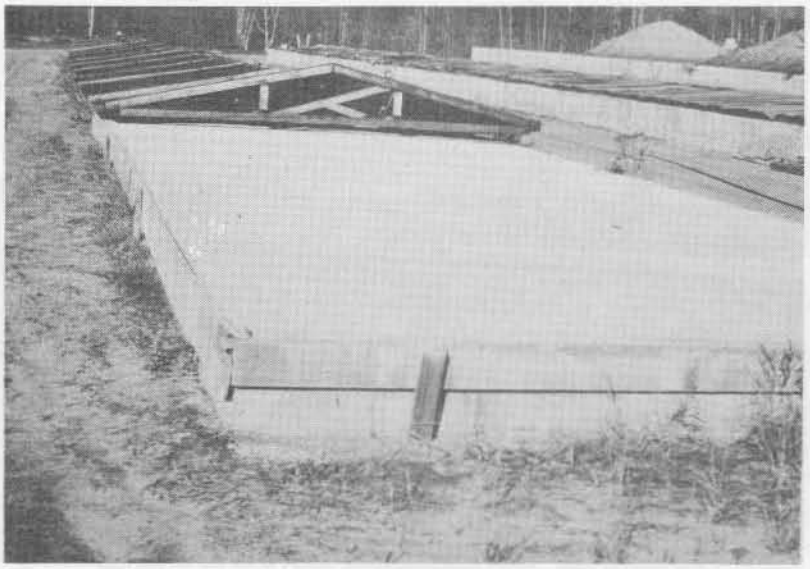


FIGURE 66
Special Disposals - Concrete Trench

FIGURE 67
CRNL on the Ottawa River

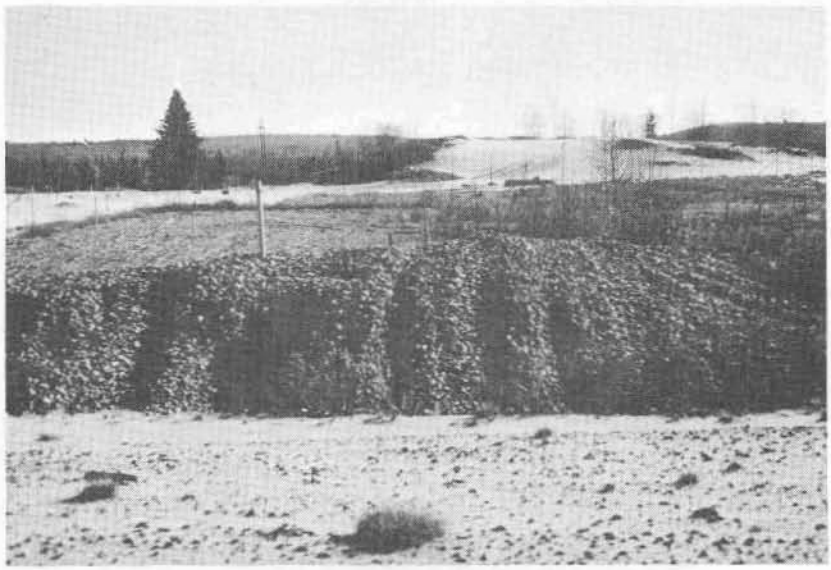


FIGURE 68
Active Liquid Seepage Pit

FIGURE 69
Emergency Active Liquid Disposal Pit

FIGURE 70
Concrete Monolith

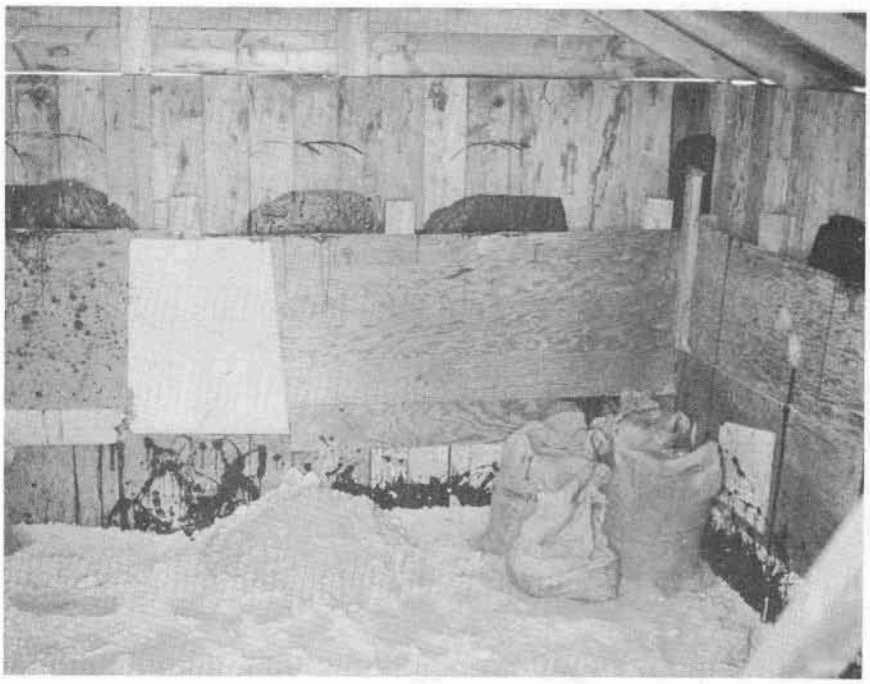


FIGURE 71
Asphalt Pit

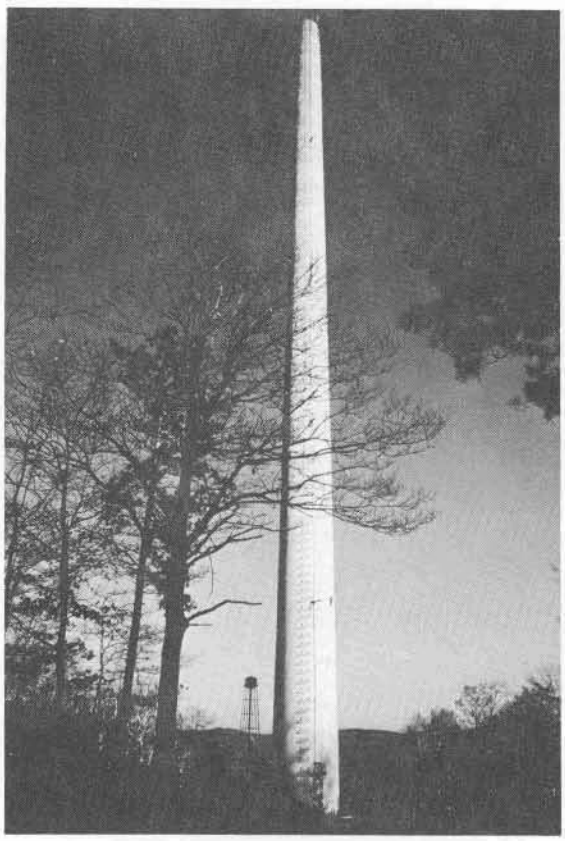


FIGURE 72
Main Stack - CRNL

13. SURVEY ARITHMETIC

13.1 To Work Out an Air Sample Result

(In disintegrations per minute per cubic metre of air - dpm/m^3)

When an air sample is taken using a Single-Disc Continuous Sampler (Figure 28) in a Staplex Sampler with GF/A filter paper (Figure 36) or an annular impactor (Figure 37), it is customary to work out the result in dpm/m^3.

The information we must have to do this is:

(i) the total number of counts per minute on the collected sample;

(ii) the counter background;

(iii) the efficiency of the counter;

(iv) the air flow through the sampler during collection of the sample;

(v) the duration of the sample.

Example 1:

A beta air sample is taken for 24 hours with a Single-Disc Continuous Sampler using GF/A filter paper. Find the result in dpm/m^3.

Method:

(i) The filter is counted with a shielded end-window geiger in conjunction with a Marconi Scaler AEP 908 to obtain the total number of counts on the sample.

(ii) The background of the counter is established by operating the counter with no sample inserted.

(iii) The efficiency of the counter is found by counting a standard source.

(iv) The airflow using a type 0210, 0211, or 1030 Gast vacuum pump with a 1-1/18 inch GF/A filter is approximately 2 m^3/h.

(v) The total airflow for 24 hours would be 48 m^3.

Total count = 5840 cpm

Counter background = 20 cpm

True count = 5820 cpm

Counter efficiency = 10%

$\therefore$ Total activity $= 5820 \times \frac{100}{10}$

$= 58,200$ dpm

Total airflow $= 48\ m^3$

$\therefore$ Result $= \frac{58200}{48}$

$= 1212\ dpm/m^3$

Example 2:

A 3-minute beta air sample was taken with a Staplex Sampler using a 4¼" GF/A filter. The indicated airflow was 1 m^3/min. The total count was 5640 cpm. The scintillation counter assembly AEP A-10057 had an efficiency of 50% and a background of 200 cpm. Find the result in dpm/m^3.

Total count = 5640 cpm

Counter background = 200 cpm

True count = 5440 cpm

Counter efficiency = 50%

$\therefore$ Total activity $= 5440 \times \frac{100}{50}$

$= 10{,}880$ dpm

Total airflow $= 3\ m^3$

$\therefore$ Result $= \frac{10{,}880}{3}$

$= 3626$ dpm/m^3

Example 3:

A 5-minute alpha air sample was taken with a Staplex Sampler using an annular impactor. The indicated airflow was 1 m^3/min. The total count was 176 cpm. The counter efficiency was 33% and the background was 2 cpm. Find the result in dpm/m^3.

Total count = 176 cpm

Counter background = 2 cpm

True count = 174 cpm

Counter efficiency = 33%

$\therefore$ Total activity $= 174 \times \frac{100}{33}$

$= 522$ dpm

Total airflow $= 5\ m^3$

$\therefore$ Result $= \frac{522}{5}$

$= 104\ dpm/m^3$

13.2 <u>To Find the Exposure Rate at 1 Foot From a Gamma Source</u> (Approx.)

<u>Formula</u>:

R/h @ 1 ft $\cong$ 6 CiE

where Ci = number of curies

E = gamma ray energy (MeV)

<u>Examples</u>:

(a) What is the exposure rate at 1 foot from a 250 mCi source of ^{137}Cs?

$6\ CiE = 6 \times \frac{250}{1000} \times 0.662$

$= .993$ R/h

$= 993$ mR/h

(b) What is the exposure rate at 1 foot from a 200 mCi source of ^{65}Zn?

$$6\ CiE = 6 \times \frac{200}{1000} \times 1.12$$

$$= 1.344\ R/h$$

$$= 1344\ mR/h$$

13.3 <u>To Find the Exposure Rate at Any Distance From a Gamma Source</u>

<u>Formula</u>:

$$mR/h = \frac{n \times I^{\gamma}}{S^2}$$

where n = number of millicuries (mCi)

I^{γ} = mR/h at 1 metre per mCi

S = distance in metres

<u>Examples</u>:

(a) If we have a 5 mCi ^{60}Co source, at what distance will the radiation field be 120 mR/h?

$$mR/h = \frac{n \times I^{\gamma}}{S^2}$$

$$\therefore S^2 = \frac{n \times I^{\gamma}}{mR/h}$$

$$= \frac{5 \times 1.32}{120}$$

$$= 0.055\ metres$$

$S = \sqrt{0.055}$ metres

$= 0.234$ metres

$= 23.4$ cm

(b) If we have a 10 mCi ^{226}Ra source, at what distance will the radiation field be 12 mR/h?

$$mR/h = \frac{n \times I^{\gamma}}{S^2}$$

$$\therefore S^2 = \frac{n \times I^{\gamma}}{mR/h}$$

$$= \frac{10 \times .84}{12}$$

$= 0.70$ metres

$S = \sqrt{0.70}$ metres

$= 0.836$ metres

$= 83.6$ cm

13.4 <u>To Convert $\mu Ci/cm^3$ to dpm/m^3</u>

1 curie (Ci) = 3.7×10^{10} dis/sec.

1 millicurie (mCi) = 3.7×10^{7} dis/sec.

1 microcurie (μCi) = 3.7×10^{4} dis/sec.

Formula: $dpm/m^3 = \mu Ci/cm^3 \times 3.7 \times 10^4 \times 10^6 \times 60$

<u>Examples</u>:

(a) What is the $(MPC)_a$ for ^{90}Sr for a 40-hour week?

(Use the value shown in the Recommendations of the International Commission on Radiological Protection - ICRP Publication 6.)

$dpm/m^3 = 10^{-9} \times 3.7 \times 10^4 \times 10^6 \times 60$

$= 2220$

(b) What is the $(MPC)_a$ for U-nat. for a 40-hour week?

$dpm/m^3 = 7 \times 10^{-11} \times 3.7 \times 10^4 \times 10^6 \times 60$

$= 155$

13.5 To Set a Working Time in a Mixed Radiation Field

Before different units of radiation can be added together, they must be converted to a common term. The term used to describe a body exposure to radiation is the "roentgen equivalent man" (rem).

Example:

A mixed radiation field is measured with the following results:

gamma - 100 mR/h

slow neutrons - 20 mrem/h

fast neutrons - 3 mrep/h

The maximum permissible dose for total body external radiation (MPD) is 600 mrem/2 weeks.

Use the relationship Quality Factor (QF) = $\frac{\text{rem}}{\text{rep. (rad)}}$

The QF for gamma rays is 1; for fast neutrons 10.

Therefore,	gamma 100 mR/h	= 100 mrem/h
	slow neutrons 20 mrem/h	= 20 mrem/h
	fast neutrons 3 mrep/h	= 30 mrem/h
	Total	= 150 mrem/h

$\therefore$ the working time for the 2-week period $= \frac{600}{150}$

$= 4$ hours

13.6 Half-Life Calculations

Formula: $A_p = \frac{A_o}{2^n}$

where A_p is the activity remaining after n half-lives

A_o is the original activity

n is the number of half-lives

Example:

The dose rate from a ^{60}Co source is 10 R/h at 1 foot.

What dose rate can be expected at the same distance from this source in 3 years?

Half-life of ^{60}Co is 5.2 years.

$\therefore n = \frac{3}{5.2}$

$= 0.58$ half-lives

$$A_p = \frac{A_o}{2^n}$$

$$= \frac{10}{2^{.58}}$$

$$= \frac{10}{1.5}$$

$$= 6.6 \text{ R/h}$$

In finding $2^{.58}$, use logarithms

$$\log 2 = 0.3010$$

$$\therefore \log 2^{.58} = 0.3010 \times 0.58$$

$$= 0.17458$$

Antilog 0.17458 = 1.5

13.7 <u>Shielding Calculations</u> (H.V.L.)

Formula: $A_p = \frac{A_o}{2^n}$

where A_p is the dose rate shielded

A_o is the dose rate unshielded

n is the number of half-value layers (H.V.L.)

<u>Example</u>:

What thickness of concrete would be required to reduce the radiation from a ^{60}Co source which reads 6 R/h on contact to 200 mR/h at the surface of the shielding?

$$A_p = \frac{A_o}{2^n} \text{ or } 2^n = \frac{A_o}{A_p}$$

$$= \frac{6000}{200}$$

$$= 30$$

$$\log 2^n = \log 30$$

$$0.3010 \times n = 1.47712$$

$$n = 4.9 \text{ half-value layers (H.V.L.)}$$

1 H.V.L. of concrete for ^{60}Co is 2.7 inches (Table 8, page 146).

$\therefore$ 4.9 H.V.L. = 4.9 x 2.7

= 13.23 inches of concrete

13.8 <u>Shielding Calculations</u> (H.V.L. and T.V.L.)

In simple shielding problems it is usually possible to calculate the thickness of shielding required by inspection of the original and desired intensity of radiation.

<u>Example</u>:

A reading of 12.8 R/h is obtained on contact with a ^{137}Cs source. What thickness of lead is required to reduce this reading to 2 mR/h at the surface of the shielding?

12.8 R/h = 12,800 mR/h

∴ 2 tenth-value layers will reduce the radiation intensity from 12,800 mR/h to 128 mR/h and 6 half-value layers will reduce the radiation intensity from 128 mR/h to 2 mR/h.

∴ the thickness of lead required is:

2 x .84 = 1.68 in.

6 x .25 = 1.50 in.

3.18 in.

(See Table 8, page 146).

13.9 Shielding Calculations Using the Linear Absorption Coefficient

Formula: $I = I_o e^{-\mu t}$

where I_o is the original intensity

I is the shielded intensity

e is the base of the natural logarithms (2.718)

μ is the linear absorption coefficient

t is the thickness of absorber in centimetres

Example:

A gamma source of 1 MeV energy gives a reading of 100 R/h at 1 foot. What reading will be obtained if a lead shield 10 centimetres thick is placed around the source?

$I = I_o e^{-\mu t}$

$\mu = 0.8$

$t = 10$

$-\mu t = -8$

From the tables on page 71 Radiological Handbook (1957)

$e^{-8} = 0.00034$

$\therefore I = I_o \times 0.00034$

$= 100 \times 0.00034$

$= 0.034$ R/h

$= 34$ mR/h

13.10 Shielding Calculations Using the Reduction Factor

Formula: $\text{Reduction Factor} = \dfrac{\text{Dose Rate Without Shielding}}{\text{Dose Rate With Shielding}}$

Example:

What thickness of (a) lead (b) iron (c) concrete will be required to shield a ^{60}Co source which gives a

reading of 6 R/h on contact, to a value of 6 mR/h at the surface of the container?

Reduction Factor = $\frac{6000}{6}$ = 1000

From the graphs in Appendix VII, the shielding required is:

(a) lead 4.9 inches (Graph 1);

(b) iron 8.6 inches (Graph 2);

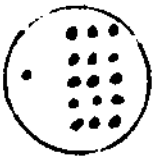
(c) concrete 27 inches (Graph 3).

APPENDIX I

THERMO LUMINESCENCE DOSIMETER CODING

Disc Number Coding

A template is used to mark the dots on the discs, and if dots were made through all the holes in the template a disc would look like this -



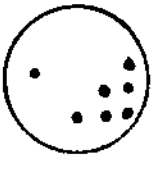
To read the number, the reference dot on the edge should be in the 9 o'clock position so that three vertical rows of dots can be seen. These three rows of dots represent the disc number - in this case 555.

For any figure from 1 to 5 the dots are marked in a vertical row starting at the top of the template pattern and working down.

Example, disc 123:

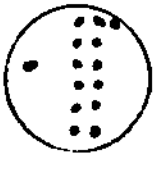
For any figure from 6 to 9 the number of dots less five are placed in the vertical rows starting at the bottom of the template pattern.

Example, disc 678:



To make a nought or zero, an extra dot is made at the bottom of a vertical row of 5 dots. Thus 6 dots in a vertical row represents zero.

Example, disc 001:



Each disc bears three figures.

APPENDIX II

AECL-121 REV. F-69

ATOMIC ENERGY OF CANADA LIMITED

NO. _____ **WORK PERMIT** DATE _____

PERMIT EXPIRES AT END OF:- 1- ☐ 2- ☐ 3- ☐ or D- ☐ SHIFT

BLDG. NO. | AREA | ROOM

WORK:-

PRECAUTIONS

	YES	NO
RADIATION SURVEY REQ'D	☐	☐
HOLD TAGS REQ'D	☐	☐
WHITE TAGS INSTALLED	☐	☐
CONSULT SURVEYOR IN AREA	☐	

CHECK EXPOSURE AFTER

VALVES OR SWITCHES TO BE TAGGED

INDUSTRIAL HAZARDS

FIRE ☐ CHEMICAL ☐ TEMPERATURE ☐ PRESSURE ☐

OPERATIONAL, SERVICING, OR OTHER PRECAUTIONS:

RADIATION HAZARDS

EXTERNAL RADIATION	GAMMA ☐	BETA ☐	OTHER ☐
AIR CONTAMINATION	BETA ☐	ALPHA ☐	TRITIUM ☐
SURFACE CONTAMINATION	BETA ☐	ALPHA ☐	TRITIUM ☐

EQUIPMENT REQUIRED

CLOTHING

☐ LAB COAT	☐ COTTON GLOVES
☐ COVERALLS OR WHITES	☐ RUBBER GLOVES
☐ DOUBLE COVERALLS	☐ SAFETY SHOES
☐ UNDERWEAR	☐ SAFETY GLASSES
☐ PLASTIC HOOD	☐ FACE SHIELD
☐ PLASTIC SUIT	☐ HARD HAT
☐ RUBBER BOOTS	☐ _____

RESPIRATOR	DOSIMETERS
☐ COMFO	☐ PENCIL CHAMB.
☐ FULL MASK	☐ SPECIAL FILMS
☐ AIR MASK	☐ ALARMING DOSIMETER
☐ AIR HOOD	☐ _____

FINAL PRECAUTIONS

☐ MONITOR TOOLS ☐ SHOWER

☐ MONITOR CLOTHES URINE SAMPLE

☐ MONITOR SKIN & HAIR ☐ SPOT ☐ OVERNIGHT

APPROVAL TO START WORK

Issuer _____

Officer in Charge of Area _____

Surveyor _____

Receiver _____

WORK COMPLETED YES ☐ NO ☐

SIGNED _____

Distribution — White copy to Issuer, Pink copy to receiver, Blue copy to R & IS Br.

APPENDIX III

ATOMIC ENERGY OF CANADA LIMITED

INDIVIDUAL MONITORING EXPOSURES

NAME: BADGE: BRANCH:

PLEASE NOTE THE FOLLOWING RADIATION EXPOSURE(S):

If two or more films are worn simultaneously, only the highest exposure is noted.

PERIOD		EXTERNAL γ-RAY EXPOSURE, *mR*			INTERNAL WHOLE BODY DOSE *mrem*	BETA-RAY SKIN DOSE, *mrad*		
FROM	TO	BADGE	BODY	HEAD		BADGE	BODY	HEAD

WHOLE BODY DOSE EQUIVALENT:

14 WEEK TOTAL ..... *mrem* YEAR TOTAL ..... *mrem*

SKIN DOSE EQUIVALENT:

14 WEEK TOTAL ..... *mrem* YEAR TOTAL ..... *mrem*

A CHRONIC WHOLE BODY DOSE RATE OF 200 MREM, OR MORE, PER 2 WEEKS, AND/OR A SKIN DOSE RATE OF 1200 MREM, OR MORE, PER 2 WEEKS, WILL RESULT IN OVER EXPOSURES FOR THE YEAR. YOU ARE BEING ADVISED SO THAT YOU CAN TAKE STEPS TO AVOID THIS SITUATION.

AECL - 1782 (5-64) ▲ PTO

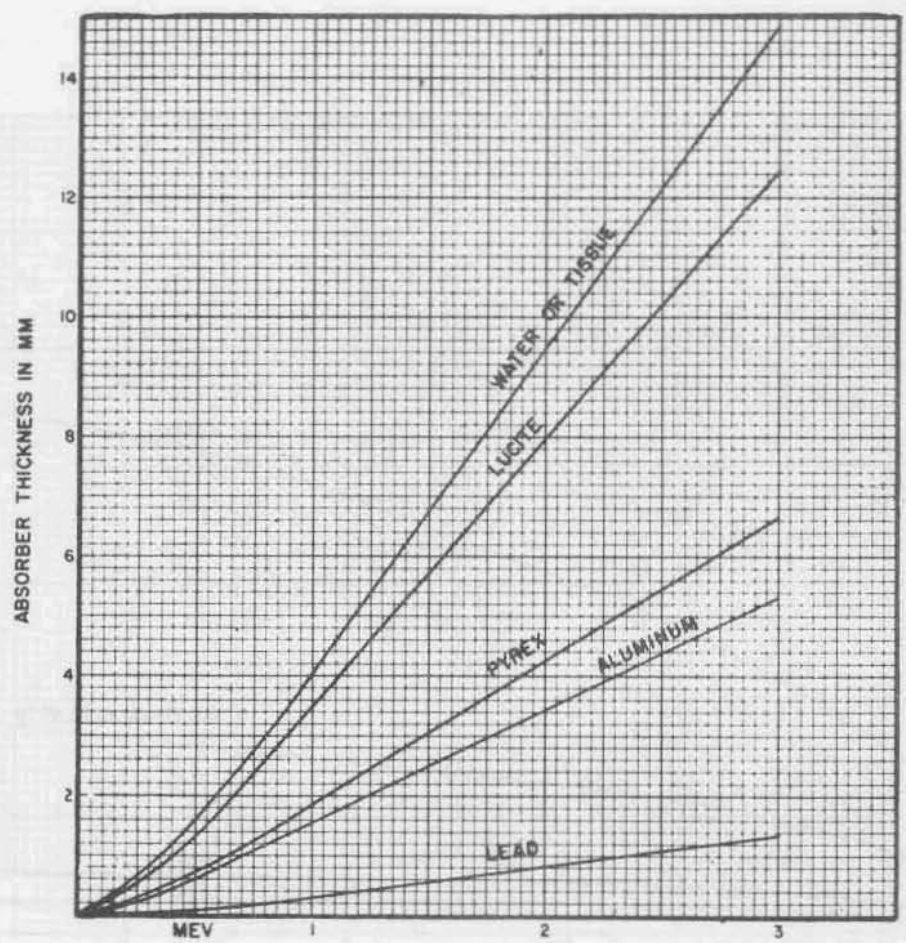
Film dosimeter exposures for external gamma-radiation are stated in mR, and beta-ray skin doses are stated in mrad. Assuming unity Quality Factor and a rad/R ratio of one, both the Exposure in mR, and the Skin Dose in mrad are numerically equal to the corresponding Dose Equivalents in mrem. The Internal Whole Body Dose is calculated directly in mrem.

Whole Body Dose Equivalent, mrem = |*(Exposure, mR) + (Internal Dose, mrem)*|

Skin Dose Equivalent, mrem = |*(Exposure, mR) + (Internal Dose, mrem) + (β-ray Skin Dose, mrad)*|

APPENDIX IV

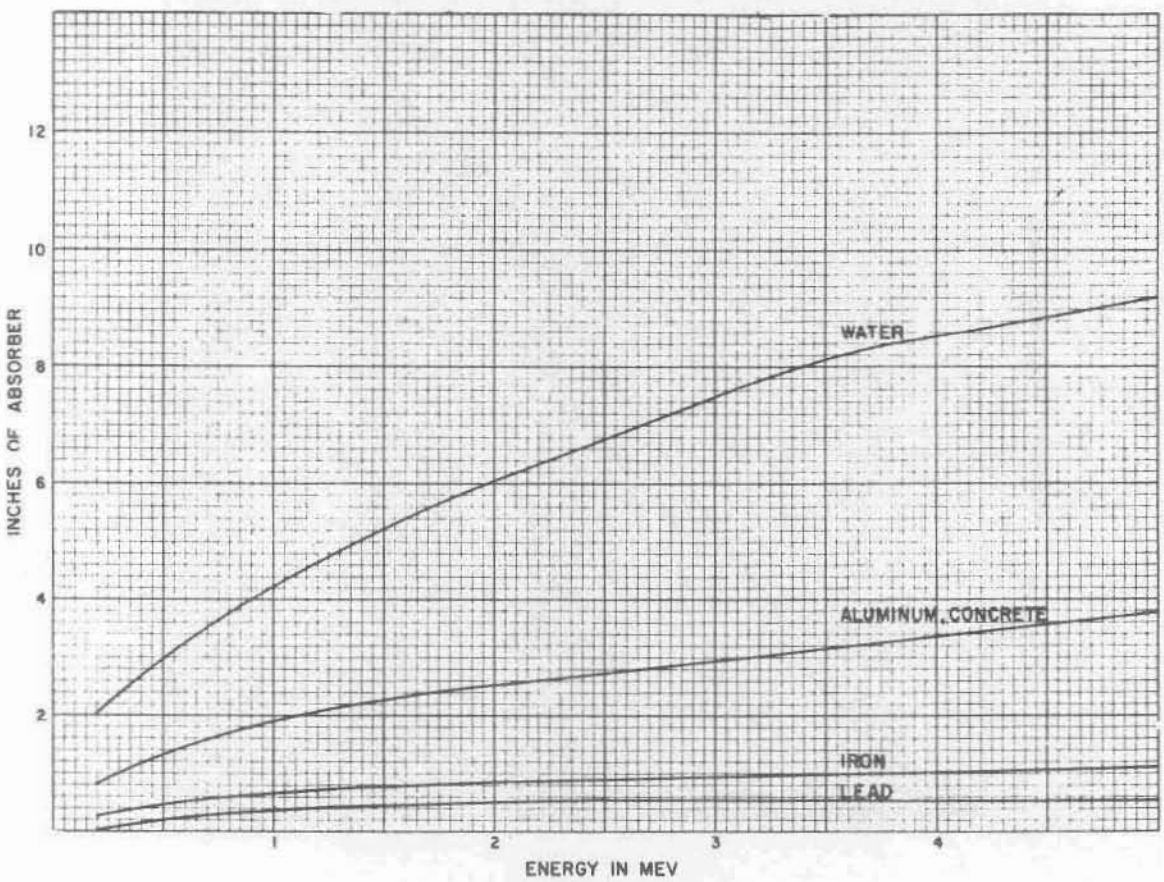
BETA ABSORPTION IN VARIOUS MATERIALS



Beta Radiation Energy Level

APPENDIX V

HALF-VALUE LAYERS FOR GAMMA RADIATION



APPENDIX VI

LINEAR ABSORPTION COEFFICIENTS
(Per centimetre)

Gamma-Ray Energy (MEV)	Carbon	Water	Aluminum	Copper	Lead
0.25	0.26	0.124	0.29	0.91	
0.50	0.20	0.095	0.22	0.70	1.7
0.75	0.17		0.19	0.58	
1.00	0.15	0.69	0.16	0.50	0.80
1.25	0.13		0.146	0.45	
1.50	0.12		0.132	0.41	
1.75	0.114		0.122	0.38	
2.00	0.106		0.115	0.35	
2.50	0.087	0.043	0.105	0.33	0.475
3.00	0.083		0.100	0.32	
3.50	0.078		0.095	0.31	
4.00	0.069		0.086	0.30	
4.50			0.078	0.28	
5.00		0.030	0.075	0.27	0.480
5.50			0.073	0.28	
6.00			0.071	0.28	
7.00			0.068	0.30	
8.00			0.065	0.30	
9.00			0.063	0.31	0.61
10.00		0.022	0.061	0.31	
15.00			0.061	0.32	
20.00		0.017	0.054	0.32	
30.00			0.058	0.34	
50.00		0.015	0.061	0.38	1.02

APPENDIX VII

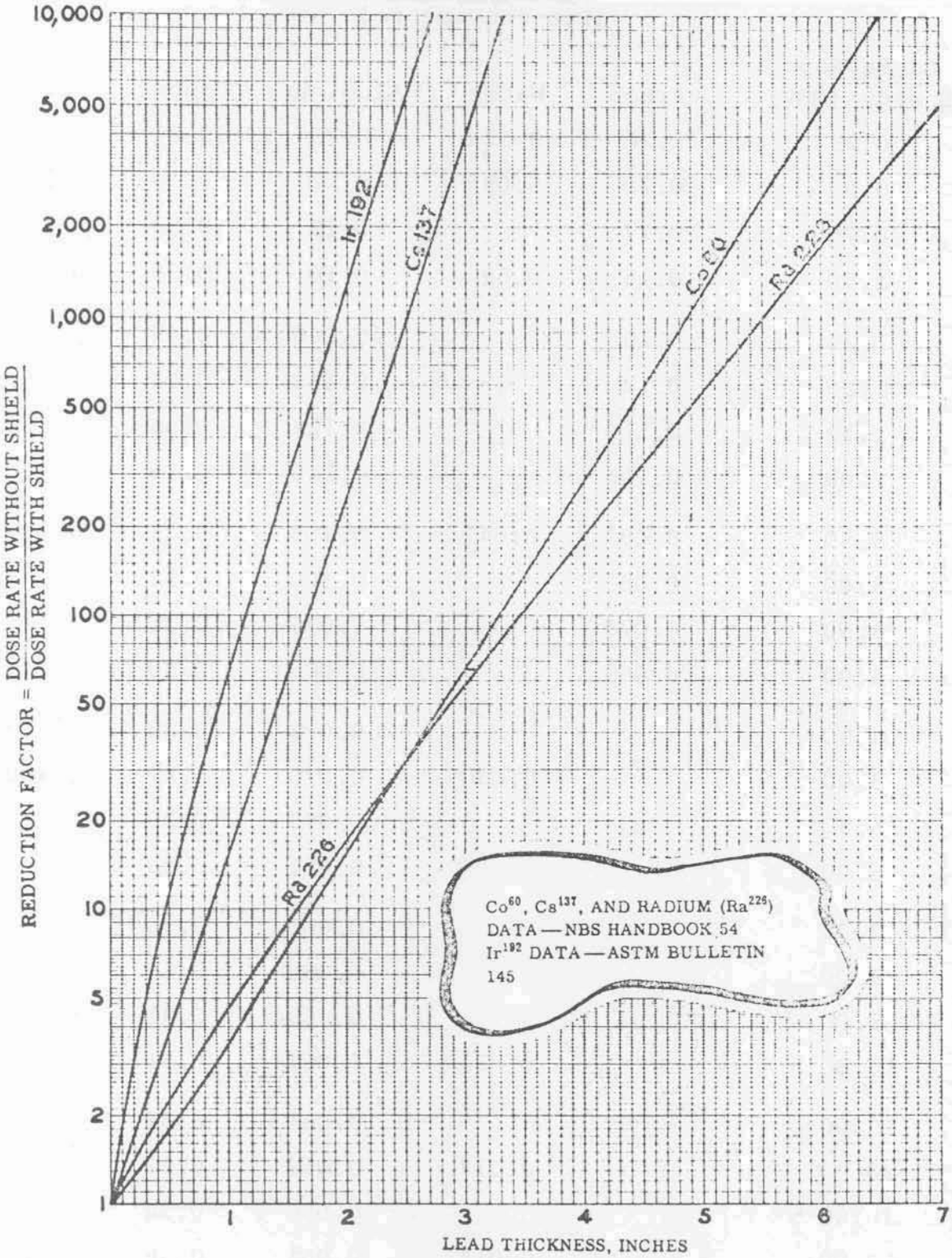
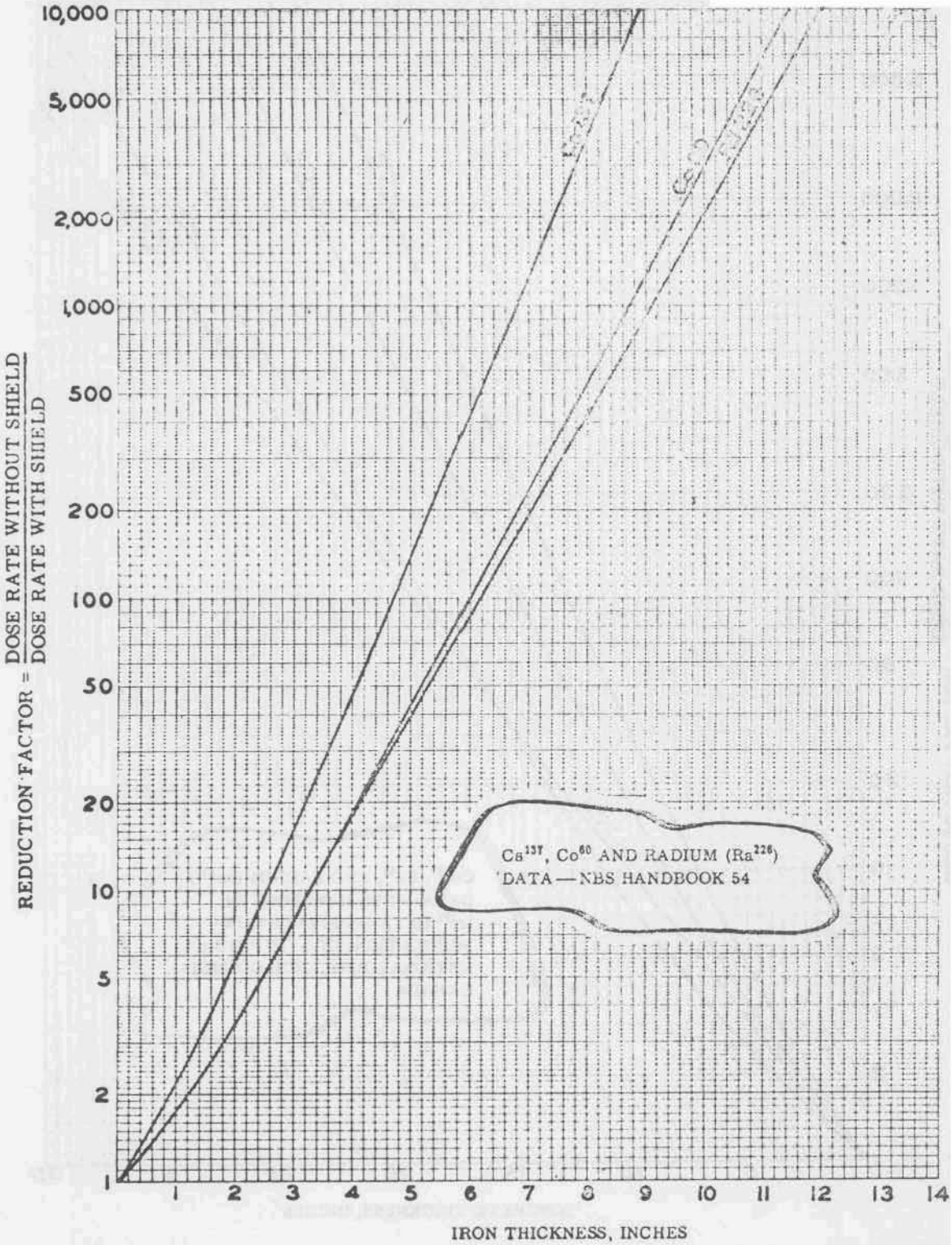


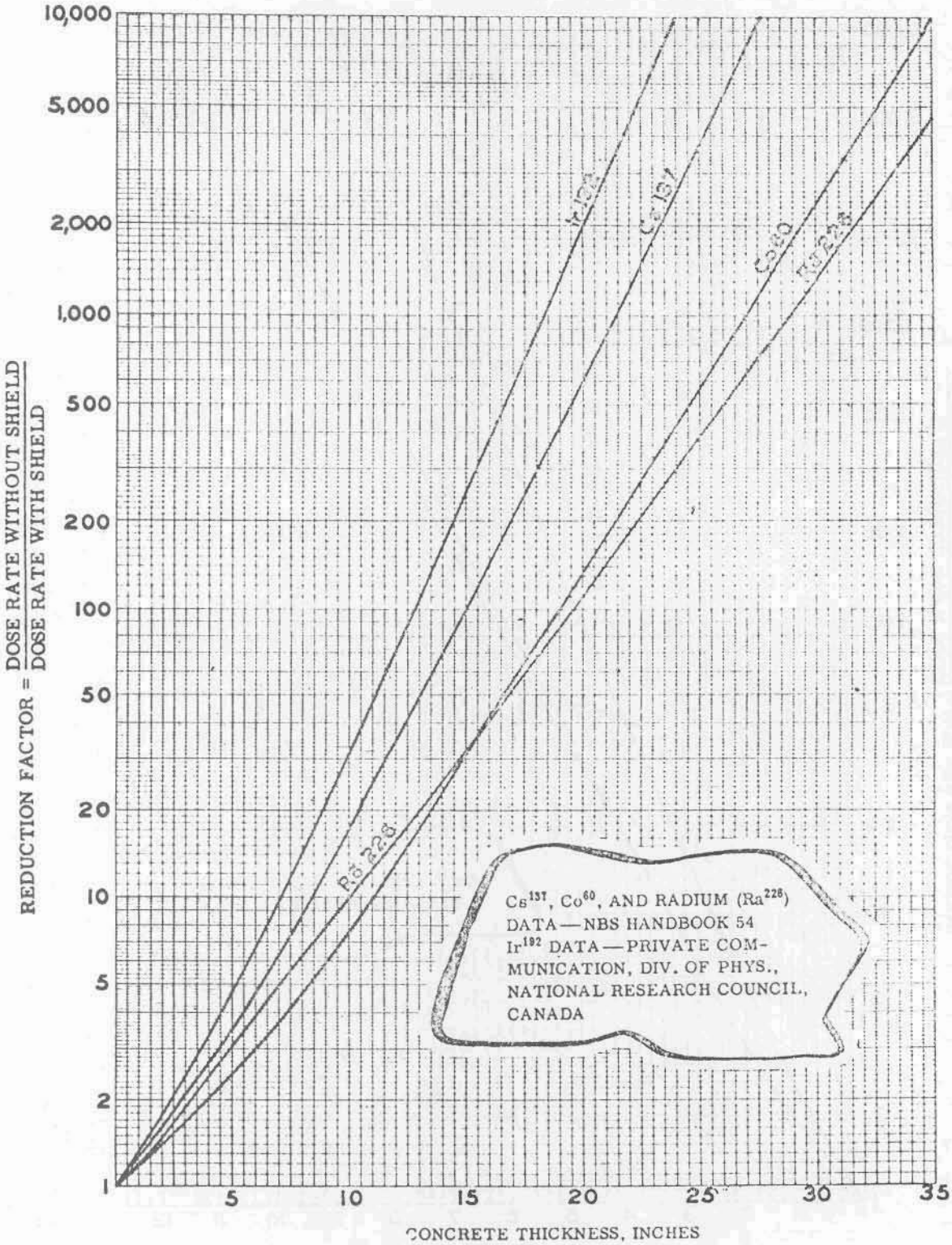
Figure 2 — Broadbeam shielding for absorption of Ir^{192}, Cs^{137}, Co^{60}, and Ra^{226} gamma rays in lead.

APPENDIX VII (Cont'd)



— Broadbeam shielding for absorption of Cs^{137}, Co^{60} and Ra^{226}, gamma rays in iron.

APPENDIX VII (Cont'd)



[redacted]—Broadbeam shielding for absorption of Ir^{192}, Cs^{137}, Co^{60}, and Ra^{226} gamma rays in concrete.

APPENDIX VIII

DECONTAMINATION TAGS

3630 FOR DECONTAMINATION

Date.....
From Bldg.....
Material.....
.....
.....
Type of Activity.....
Activity.....
Special Precautions.....
.....
.....
Requested By.....
Surveyed By.....
(OVER)

3630 Date..... Received At Decontamination Centre

Material.....
.....
.....
Received by.....
(OVER)

FOR INSTRUCTIONS SEE REVERSE SIDE

DECONTAMINATION PROCESS TAG

THIS ARTICLE HAS PASSED THROUGH
A DECONTAMINATION PROCESS

AMOUNT OF ACTIVITY FIXED ..... C.P.M.

..... C.P.M.

SURVEYED BY .....

CAUTION

THIS ARTICLE HAS NOT RECEIVED COMPLETE DECONTAMINATION

NOTE.....

.....

.....

SURVEYED BY:

APPENDIX IX

DISPOSAL FORM

DISTRIBUTION
White Copy – Environmental Res. Br.
Yellow – M & S Br.
Pink – Shipper

DISPOSAL OF ACTIVE WASTE

ATOMIC ENERGY OF CANADA LIMITED
ENVIRONMENTAL RESEARCH BRANCH

NAME OF BRANCH			DATE
MATERIAL	RADIONUCLIDE	AMOUNT (CURIES)	RADIATION LEVELS
			CONTACT: AT 1 FOOT:
SPECIAL INSTRUCTIONS			
AUTHORIZED SIGNATURE		BRANCH SURVEYOR	

(FOR USE OF ENVIRONMENTAL RESEARCH BRANCH)

LOCATION IN DISPOSAL AREA

ENVIRONMENTAL RESEARCH

AECL-169 (REVISED 6-64) A

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