CHAPTER 3

RADIATION PROTECTION

PAGE

RADIATION PROTECTION

I. EXTERNAL RADIATION PROTECTION

The three basic methods used to reduce the external radiation hazard are time, distance, and shielding. Good radiation protection practices require optimization of these fundamental techniques.

A. Time

The amount of radiation an individual accumulates will depend on how long the individual stays in the radiation field, because:

Dose (mrem) = Dose Rate (mrem/hr) x Time (hr)

Therefore, to limit a person's dose, one can restrict the time spent in the area. How long a person can stay in an area without exceeding a prescribed limit is called the "stay time" and is calculated from the simple relationship:

$$\text{Stay Time} = \frac{\text{Limit (mrem)}}{\text{Dose Rate (mrem/hr)}}$$

Example: How long can a radiation worker stay in a 1.5 rem/hr radiation field if we wish to limit his dose to 100 mrem?

$$\text{Stay Time} = \frac{100\ \text{mrem}}{1500\ \text{mrem/hr}} = 0.067\ \text{hr} = 4\ \text{minutes}$$

B. Distance

The amount of radiation an individual receives will also depend on how close the person is to the source.

1. The Inverse Square Law - Point sources of x- and gamma radiation follow the inverse square law, which states that the intensity of the radiation (*I*) decreases in proportion to the inverse of the distance from the source (*d*) squared:

$$I \propto \frac{1}{d^2}$$

This can be rewritten:

$I = K\frac{1}{d^2}$ where *K* is a constant of unknown value.

So, for an intensity I_1 at distance d_1, and another intensity I_2 at distance d_2:

$$I_1 = K\frac{1}{d_1^{\,2}};\quad I_2 = K\frac{1}{d_2^{\,2}}$$

Now solve for the relationship by eliminating *K*:

$$\frac{I_1}{I_2} = \frac{K/d_1^2}{K/d_2^2}$$

$$\frac{I_1}{I_2} = \frac{d_2^2}{d_1^2}$$

OR

$$I_1 d_1^2 = I_2 d_2^2$$

Therefore, by knowing the intensity at one distance, one can find the intensity at any other distance.

Example: The exposure rate one foot from a source is 500 mR/hr. What would be the exposure rate three feet from the source?

I_1 = 500 mR/hr

d_1 = 1 foot

d_2 = 3 feet

$$d_2 = \frac{I_1 d_1^2}{d_2^2} = \frac{(500\ \text{mR/hr})(1\ \text{ft})^2}{(3\ \text{ft})^2} = \frac{500\ \text{mR/hr}}{9} = 55.6\ \text{mR/hr}$$

2. Gamma Constants

Gamma radiation levels (in R/hr) for one curie of many radionuclides at a distance of one meter have been measured. These gamma constants can be used to determine 1) the expected exposure rate at a given distance (using the inverse square law) for a known quantity of a radionuclide, or 2) the activity of a radionuclide from a measured exposure rate. Gamma constants (Γ) for selected radionuclides can be found on page 43. To determine the gamma radiation level in R/hr at one meter per curie, or equivalently, mR/hr at one meter per millicurie, you must divide the tabulated gamma constants (Γ) by 10.

Example No. 1: What is the radiation exposure rate one foot from a 100 mCi point source of Cs-137?

Γ= 3.3 R-cm^2/hr-mCi

OR

Γ/10 = 0.33 R/hr at 1 meter/Curie = 0.33 mR/hr at 1 meter/mCi

$I_1 d_1^2 = I_2 d_2^2$

I_1 = ?

I_2 = 0.33 mR/hr/mCi x 100 mCi = 33 mR/hr

d_2 = 1 meter

d_1 = 1 foot = 0.3048 m

$$I_1 = \frac{I_2 d_2^2}{d_1^2} = \frac{(22\,\text{mR/hr})(1\,\text{m})^2}{(0.3048\,\text{m})^2} = 355\ \text{mR/hr}$$

Example No. 2: If the exposure rate from Cs-137 at one meter is 250 mR/hr, how many Curies are present?

$$\frac{0.25\,\text{R/hr/m}}{0.33\,\text{R/hr/m/Ci}} = .076\ \text{Curies}$$

3. Gamma Exposure Rate Formula

The exposure rate from a gamma point source can be approximated from the following expression:

$$\text{mR/hr} = \frac{6CEf}{d^2}$$

Where: *C* is the activity of the gamma emitter, in millicuries

E is the gamma ray energy in MeV

f is the fraction of disintegrations yielding the gamma of energy E

d is the distance from the source in feet

If more than one gamma ray is emitted by the radionuclide of interest, then the contribution from each one must be calculated separately and summed. This expression is accurate to about 20% for gamma emitters with energies ranging from 0.07 MeV to 4 MeV.

Example: What would the exposure rate be one foot away from 100 mCi of I-131?

From the reference data for I-131 on page 44:

I-131 : γ_1 = 0.364 MeV, 81.2% γ/d

γ_2 = 0.636 MeV, 7.3% γ/d

R/hr at 1 foot = 6 (0.1Ci) [(0.364 x 0.812) + (0.636 x 0.073)]

= 0.21 R/hr at 1 foot

= 210 mR/hr at 1 foot

C. Shielding

When reducing the time or increasing the distance may not be possible, one can choose shielding material to reduce the external radiation hazard. The proper material to use depends on the type of radiation and its energy.

1. Alpha and Beta Radiation

Alpha particles are easily shielded. A thin piece of paper or several cm of air is usually sufficient to stop them. Thus, alpha particles present no external radiation hazard. Beta particles are more penetrating than alpha particles. Beta shields are usually made of aluminum, brass, plastic, or other materials of low atomic number to reduce the production of bremsstrahlung radiation. The range of beta radiation for various energies in air, plastic and various materials can be found on page 45.

2. X and Gamma Radiation

Monoenergetic x- or gamma rays collimated into a narrow beam are attenuated exponentially through a shield according to the following equation:

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

where I is the intensity outside of a shield of thickness x

I_o is the unshielded intensity

μ is the linear attenuation coefficient of the shielding material

x is the thickness of shielding material.

The linear attenuation coefficient is the sum of the probabilities of interaction per unit path length by each of the three scattering and absorption processes - photoelectric effect,

Compton effect, and pair production. Note that μ has dimensions of inverse length (1/cm). The reciprocal of μ is defined as the mean free path, which is the average distance the photon travels in an absorber before an interaction takes place.

Because linear attenuation coefficients are proportional to the absorber density, which usually does not have a unique value but depends somewhat on the physical state of the material, it is customary to use the mass attenuation coefficient, which removes density dependence:

Mass attenuation coefficient $\mu_m = \frac{\mu}{\rho}$

where ρ = density (g/cm^3)

For a given photon energy, μ_m does not change with the physical state of a given absorber. For example, it is the same for water whether present in liquid or vapor form. If the absorber thickness is in cm, then μ_m will have units of $\left[\frac{\text{cm}^{-1}}{\text{g/cm}^3}\right]$ = cm^2/g.

Values of the mass attenuation coefficient and densities for various shielding materials can be found on pages 46-49 and pages 50-51, respectively.

Using the mass attenuation coefficient instead of the linear attenuation coefficient, the attenuation equation can be rewritten:

$$I = I_o e^{-\mu_m \rho x}$$

NOTE: Lead is a common shielding material for x-rays and gamma radiation because it has a high density, is inexpensive, and is relatively easy to work with. When working with a radionuclide that emits multiple types of radiation such as beta particles and gamma radiation, it is sometimes necessary to shield with several materials. The less penetrating beta radiation can first be shielded with a layer of plastic or Plexiglas, thereby slowing or stopping the beta particles while reducing the production of bremsstrahlung. The more penetrable gamma radiation would require an additional layer of shielding. Types of shielding and amount of shielding vary depending on photon energy. A good rule of thumb: shield the less penetrable radiation type first then proceed to shield the more penetrable type. This usually decreases both scattering and the total amount of shielding material required.

Example: The intensity of an unshielded Cs-137 source is 1 R/hr. If the source is put into a lead shield two (2) inches thick, what would be the intensity on the outside of the shield? Assume your distance from the source has not changed. (Density of lead = 11.35 gm/cm^3).

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

I_o = 1 R/hr

$\mu = \mu_m \cdot \rho$ = (0.114cm^2/gm)(11.35gm/cm^3) = 1.29 cm^{-1}

x = 2 inches · 2.54 cm/inch = 5.08 cm

I = (1 R/hr) $e^{-[(1.29cm-1)(5.08cm)]}$ = 0.0014 R/hr = 1.4 mR/hr

3. Half Value Layer

The half value layer (HVL) is the thickness of a shielding material required to reduce the intensity of radiation at a point to one half of its original intensity. It can be calculated by setting $I = ½ I_0$ and solving the attenuation equation for x:

$$0.5 = e^{-\mu x_{1/2}}$$

$$x_{1/2} = -\frac{\ln(0.5)}{\mu}$$

$$x_{1/2} = \frac{0.693}{\mu} = \text{HVL}$$

Half value layers for various shielding materials and selected radionuclides can be found on page 52.

When the HVL is known rather than μ, the total attenuation from n half value layers can be calculated by using the following equation:

$$I = \frac{I_0}{2^n}$$

Example: How much lead shielding must be used to reduce the exposure rate from an I-131 source from 32 mR/hr to 2 mR/hr? HVL of lead for I-131 is 0.178 cm.

$$2^n = \frac{I_0}{I} = \frac{32\ \text{mR/hr}}{2\ \text{mR/hr}}$$

$$2^n = 16$$

$$n = 4$$

4 HVL x 0.178 cm/HVL = 0.71 cm

Several "Rules of Thumb" for external radiation protection may also be found on pages 21 and 22.

D. External Exposure Personnel Monitoring

External radiation exposure is measured by personnel monitoring devices. Personnel monitoring is required when it is likely that an individual will receive in 1 year a dose in excess of 10% of the following: a total effective dose equivalent (sum of internal and external) of 5 rem (0.05 Sv) to the whole body (head and trunk, upper arms, upper legs, active blood forming organs, gonads); 50 rem (0.5 mSv) to the skin of the whole body or any extremity; 15 rem (0.15 Sv) to the lens of the eye. Personnel monitoring provides a permanent, legal record of an individual's occupational exposure to radiation.

If the body is exposed fairly uniformly, the dosimeter should be worn on the trunk of the body. This will allow the dose to critical organs (gonads, red bone marrow) to be accurately estimated by the dosimeter. When wearing a protective lead apron, the film badge should be worn on the collar, outside of the apron. Ring badges or wrist badges should be worn when large amounts of radioactive materials are handled since the dose to the hands may be high. Ring badges should be worn on the inside of protective gloves to avoid contamination.

1. Types of Personnel Dosimeters

Four major types of monitoring devices in use today are the pocket dosimeter, the film badge, the thermoluminescent dosimeter (TLD), and the optically stimulated luminescent (OSL) dosimeter.

a) **Film Badge**

The film badge was once the most commonly used personnel monitoring device for x- and gamma radiation and charged particles, but in the United States it has been largely replaced by newer technologies. A film badge is composed of a piece of photographic film and a special film holder. The effect of radiation exposure is a darkening of the film, and the amount of darkening is proportional to the dose absorbed by the film. The film is placed inside a light tight packet which is placed in the film holder. The film holder contains various filters (e.g. lead, tin, aluminum, plastic). Radiation passing through the filters will produce a density distribution on the film from which the energy range and type of the radiation can be determined. See Figure 1 for illustration.

Photographic film has several disadvantages for its use as a personnel monitoring device:

1) Fogging may result from mechanical pressure, high temperatures or exposure to light before development.
2) Fading of the latent image can occur, which is dependent on the time interval between exposure and development.
3) Errors in the development process can affect the reading, and cannot usually be corrected.
4) Isotopes such as H-3, C-14 or S-35 have beta energies below the sensitivity of the film and cannot be detected.

NOTE: Many people still commonly refer to personnel dosimeters of any type as “film badges,” although most are actually TLD or OSL dosimeters.

b) **Thermoluminescent Dosimeters**

Thermoluminescent dosimeters, or "TLD's", are also used for monitoring beta, x-, and gamma radiations. Energy absorbed from the incident radiation excites and ionizes the molecules of the thermoluminescent material. Some of the energy is trapped by impurities or deformations in the material, and remains trapped until the material is heated to a high temperature. Once heated, the trapped energy is released as an emission of light. The amount of light emitted is proportional to the energy absorbed within the thermoluminescent material, which is proportional to the radiation dose absorbed. The emitted light is measured with a photomultiplier tube, the output of which is applied to a readout instrument.

The TLD is a better indicator of radiation exposure than film because it is composed of elements of low atomic number (human tissue also contains elements of low atomic number). TLD's are less affected by environmental conditions such as heat, light, and humidity than film, and are reusable. The TLD crystals are available in many sizes and shapes, such as rods, ribbons, pellets and single crystals.

TLD’s also suffer from several disadvantages:

1) The TLD can only be read once, since reading it erases it. Thus, there is no permanent record of the dose, and errors in the measurement process may lose the reading.
2) Fading of the stored signal can occur, which is dependent on the time interval between exposure and development.
3) The accuracy of the reading depends on the light sensitivity of the reader and heating rate of the TLD, but it can be difficult to maintain adequate reader constancy.
4) TLD’s do not give as much information about the energy of the incident radiation as do film and OSL dosimeters.

c) **OSL Dosimeters**

Optically stimulated luminescent (OSL) dosimeters are currently the most common type of personnel dosimeter used in the United States, and are marketed by Landauer, Inc. under the proprietary name “Luxel dosimeter.” The basic principle of operation is similar to that of the TLD, in that the energy absorbed from incident radiation becomes trapped in the material. However, green laser light, rather than heat, is used to stimulate release of the stored energy. The trapped energy is emitted as a blue light when it is released, so it can be collected and distinguished from the green incident light. As with the TLD, the amount of light emitted is proportional to the energy absorbed by the material, which is proportional to the radiation dose absorbed.

OSL dosimeters have several advantages over TLD's. They are more sensitive to a wider range of photon and beta particle energies, and provide more information about the energy of the incident radiation. This information is used to provide estimates of deep dose, dose to the lens of the eye, and shallow dose. The good resolution of the detector also allows analysis of whether the exposure was dynamic or static – i.e., whether the badge was exposed while moving or from many different angles, as would be expected if worn for an extended period of time, or exposed without being moved, such as in the case of an accidental exposure. Other major benefits of the OSL dosimeter are that it can be read multiple times without erasing all the information and that the read-out process is faster and more accurate than with TLD's. OSL dosimeters are also relatively unaffected by environmental conditions such as heat, light, and humidity.

d) **Pocket Dosimeters**

Pocket dosimeters are small ion chambers that are read on-site and are used when readings are desired quickly or on a frequent basis. They can be of two types, direct reading or indirect reading.

A direct reading dosimeter consists of a small capacitor in a pen-type housing. Before use, the dosimeter is charged using a dosimeter charger. Radiation exposure results in a loss of charge and a corresponding deflection of the fiber. This dosimeter also contains a lens and scale by which the amount of fiber deflection (dose) can easily be determined. (See Figure 2 for construction details.) The dosimeter is held up to a light and the dose read from the scale. Using this dosimeter, personal dose can be determined immediately, without interrupting the monitoring ability of the device. Direct reading dosimeters are usually calibrated to read exposure in milliroentgens, but some read in roentgens.

Indirect reading dosimeters are also shaped like a pen, but must be read using a charger-reader. The charger-reader is a voltmeter which is calibrated in roentgens.

Pocket ion chamber dosimeters are sensitive to moisture and rough handling, which will cause the charge to leak from the capacitor and give erroneous readings.

2. Use of Personnel Dosimeters

Personnel monitoring devices must be worn by personnel as specified below and/or in such instances as deemed necessary by the Radiation Control Department.

a) Whole Body Luxel/Film/TLD Badges

Whole body badges shall be worn when:

1) working with beta emitters where the energy is 300 keV or higher and the quantity greater than 1 millicurie (37 MBq) in any month.

2) working with any gamma emitters where the energy is 50 keV or higher and the quantity greater than 0.2 milliCuries (7.4 MBq) in any month.

3) working with neutron sources. Special neutron badges may be required in addition to other badges.

4) working with any apparatus capable of producing or emitting ionizing radiation as deemed necessary by the Radiation Control Department. For example, x-ray equipment, high power amplifying tubes, accelerators, etc.

5) specified by the Radiation Control Department, the Radiation Control Officer, and/or the Radiation Control Committee or the Human Use of Radionuclides and Radiation Committee.

Extremity badges shall be worn when working with beta emitters where the energy is 1 MeV or higher and the quantity greater than 1 millicurie (37 MBq) in any month, or by an individual who receives a dose of 40 mrem (400 uSv) or more on a whole body badge for two consecutive months.

All monitoring devices shall be obtained from the Radiation Control Office (see Appendix A for Personnel Monitoring Device Application form). Each dosimeter shall be assigned to and worn by only one individual. Dosimeters may be exchanged monthly, bi-monthly, or quarterly depending upon monitoring device wear location and expected radiation exposure. Delivery, exchange and pickup of badges shall be the responsibility of the Radiation Control Office; however, these functions are performed in cooperation with Film Badge Coordinators in some work areas. In the event that a dosimeter is damaged, lost, or accidentally exposed, it is the responsibility of the Principal Investigator to notify the Radiation Control Office immediately for replacement or processing. Permanent records of dosimeter readings are maintained by the Radiation Control Office. A copy of the monthly, bi-monthly, or quarterly readings is mailed to the Film Badge Coordinator in each work area.

Termination radiation exposure reports will also be provided to those badged individuals who terminate employment requiring personnel dosimetry. Forwarding addresses must be available to facilitate this mailing.

b) Pocket Dosimeters

Pocket ion chambers may be required to be worn in addition to the film badge if other types of monitors are inadequate in the judgment of the Radiation Control Officer or the Radiation Control Committee. This shall apply where the investigator is working with high level radioactive materials or other

ionizing radiations. When these devices are used, the Principal Investigator is responsible for maintaining daily pocket ion chamber records. Copies of these records shall be submitted monthly to the Radiation Control Office.

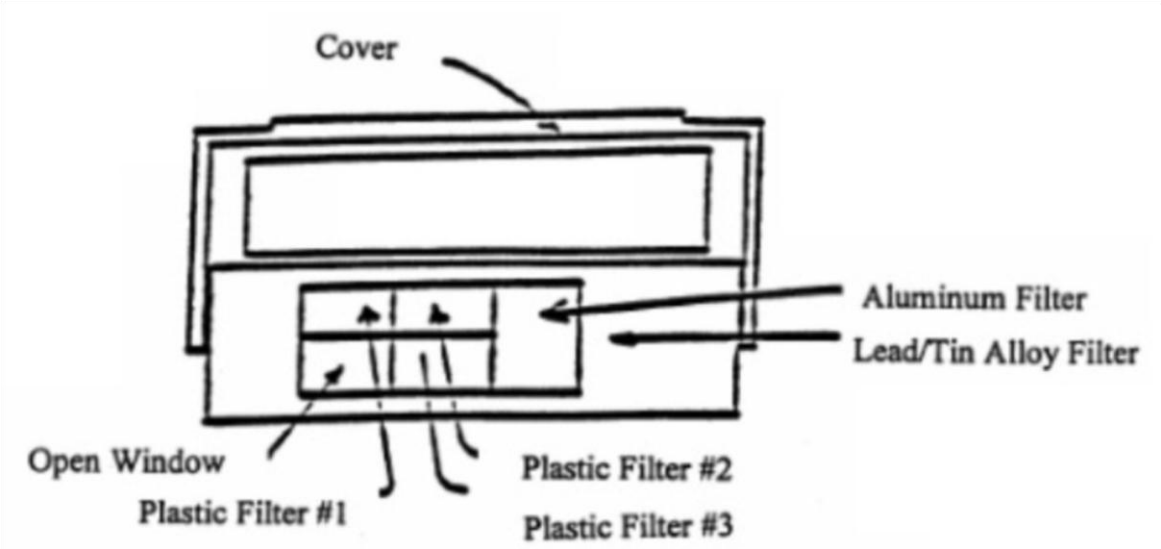


Figure 1. Typical Film Badge Construction (R.S. Landauer).

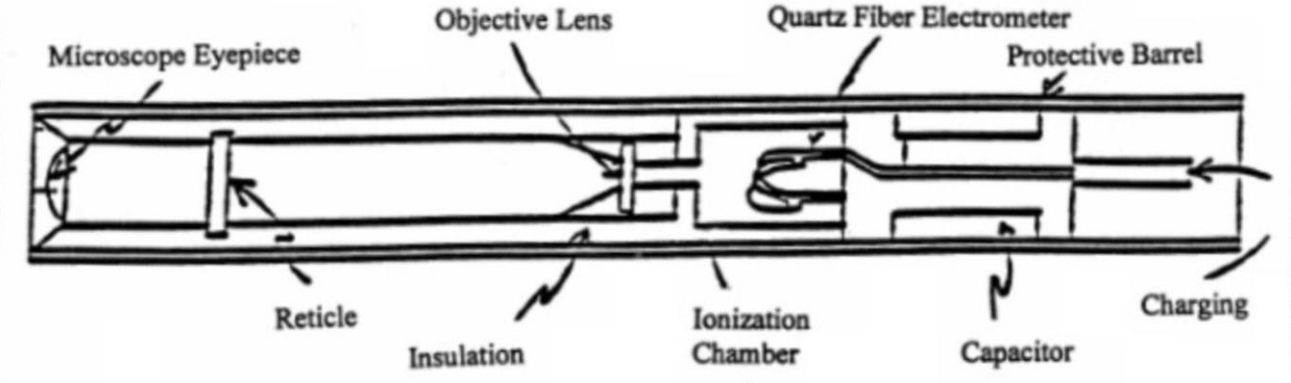


Figure 2. Typical Quartz Fiber Pocket Dosimeter.

3. Maximum Permissible Exposure Limits

Maximum Permissible Exposures are specified in the Code of Federal Regulations, Title 10, Part 20, "Standards for Protection Against Ionizing Radiation," and in the Florida Department of Health, Bureau of Radiation Control, Chapter 64E-5 (July, 1997), "Control of Radiation Hazard Regulations."

Since any radiation exposure is undesirable, it is important that all exposures be kept as low as reasonably achievable (ALARA). The maximum permissible ALARA exposures used at the University of Florida are more conservative than the State or Federal regulations. Specific approval to operate under the more liberal State or Federal regulations must be obtained for any such occasion from the Radiation Control Committee and/or Human Use of Radionuclides and Radiation Committee by submitting a written proposal through the Radiation Control Officer.

a) Occupational Dose Limits for Adults

Maximum Permissible Occupational Exposure to Adults or Restricted Area Exposure:

1) The total effective dose equivalent must not exceed 5 rem (0.05 Sv) per year.

2) The sum of the deep dose equivalent and the committed dose equivalent to an individual organ or tissue other than the lens of the eye must not exceed 50 rem (0.5 Sv) per year.

3) The dose equivalent to the lens of the eye must not exceed 15 rem (0.15 Sv) per year.

4) The shallow dose equivalent to the skin or to any extremities must not exceed 50 rem (0.5 Sv) per year.

b) Occupational Dose Limits for Minors

Occupational exposure to any individual who is under the age of 18 is permitted only if their exposure is limited to ten percent or less of the limits specified above for adult workers. For this reason, it is recommended that minors not be employed as full-time radiation workers.

c) Dose to an Embryo or Fetus for Women Who Have Declared Pregnancy

The dose to an embryo or fetus during the entire pregnancy from occupational exposure of a declared pregnant woman shall not exceed 0.5 rem (5 mSv). It is recommended that not more than 0.05 rem (0.5 mSv) be received by the embryo or fetus in any one month.

d) Dose Limits for Individual Members of the Public or Unrestricted Area Exposure

The total effective dose equivalent to individual members of the public shall not exceed 100 mrem (1 mSv) in a year. The dose in any unrestricted area shall not exceed 2 mrem (0.02 mSv) per hour.

E. Posting and Labeling of Radioactive Materials

1. Cautionary Signs

Cautionary signs are required to be posted under certain conditions as described below to warn other individuals in the area that radioactive material or radiation is present:

Caution - Radioactive Materials: In areas or on items where radioactive material is used or stored. Each label shall provide sufficient information to permit individuals handling or using containers or working in the general vicinity to take precautions to avoid or minimize exposures. Such information should include: 1) the type of radioactive material; 2) the estimated activity; 3) assay date; 4) the name of the individual responsible for the material.

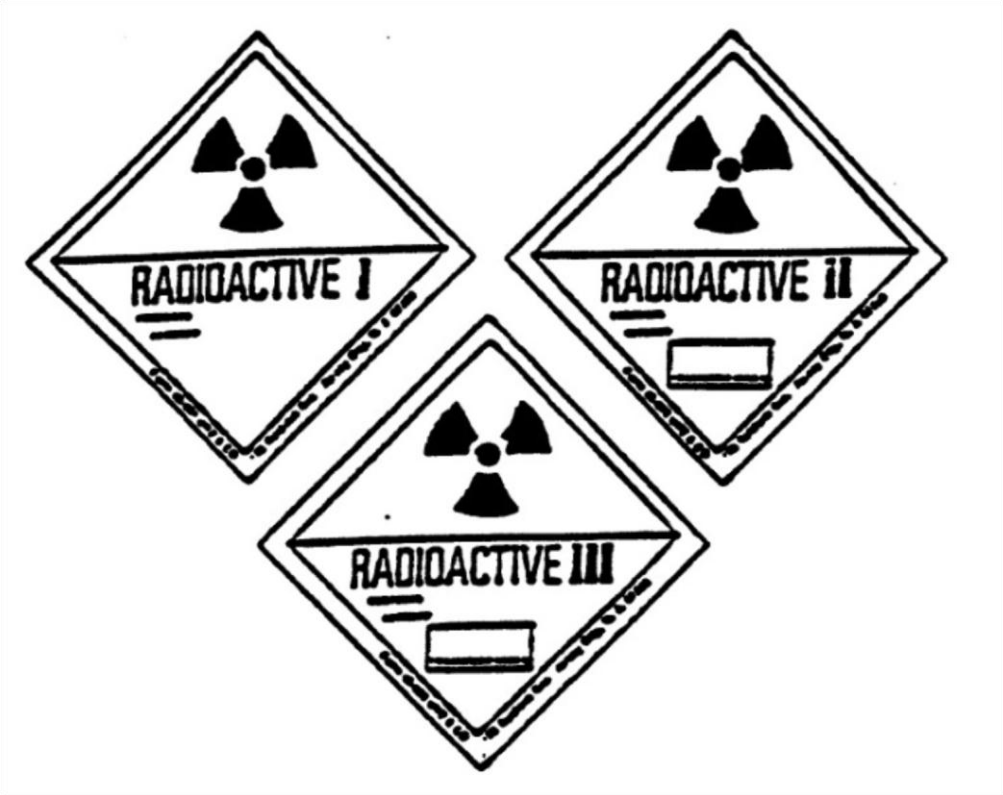
Caution - Radiation Area: In areas where the level of radiation could cause a major portion of an individual's body to receive an exposure from external radiation in any hour that exceeds 5 milliRems; or in a period of five (5) consecutive days that exceeds 100 milliRems. "Radiation Area" postings should be at the point at which an area measures 5 mR/hr (or 100 mR/hr per 40 hour work week).

Caution - High Radiation Area: In areas where the level of radiation could cause a major portion of an individual's body to receive an exposure from external radiation in any hour that exceeds 100 mRem. "High Radiation Area" postings should be at the point at which an area measures 100 mR/hr. High radiation areas also require strict access controls.

In addition, individuals posting radiation warning signs should provide information on the sign to aid others in minimizing their exposure. Information may include: 1) the source of radiation; 2) the exposure rate in mR/hr or R/hr on contact at the highest spot; 3) the name of the person posting the sign; and 4) the date the sign was posted.

2. Department of Transportation (DOT) Warning Labels

Each package of radioactive material offered for transportation, unless exempted, must be labeled on two sides with one of the three labels shown below:



DOT Warning Labels for Radioactive Materials Packages

The purpose of these labels is to alert individuals handling packages that special handling may be required. When the background color of the label is all white (Radioactive White-I), the external radiation level from the package is minimal and no special handling is necessary. If however, the background of the upper half of the label is yellow (Radioactive Yellow II or III), a radiation level may exist at the outside of the package, and precautions should be taken to minimize radiation exposures when handling the package. The radiation level in mR/hr at three feet from the external surface of the package is known as the Transport Index, and is written in the space provided on the warning label. Furthermore, if the package bears a Radioactive Yellow III, the rail or highway vehicle in which it is carried must be placarded. The table below defines the label criteria for radioactive materials packages:

Labels	Dose Rate Limits	
	At Any Point on Accessible Surface of Package	At Three Feet From External Surface of Package (Transport Index)
"RADIOACTIVE-WHITE I"	0.5 mR/hr	N/A
"RADIOACTIVE-YELLOW II"	50 mR/hr	1.0 mR/hr
"RADIOACTIVE-YELLOW III"	200 mR/hr	10 mR/hr

The cautionary signs and warning labels described in these sections must be removed or defaced when they are no longer serving their function.

II. INTERNAL RADIATION PROTECTION

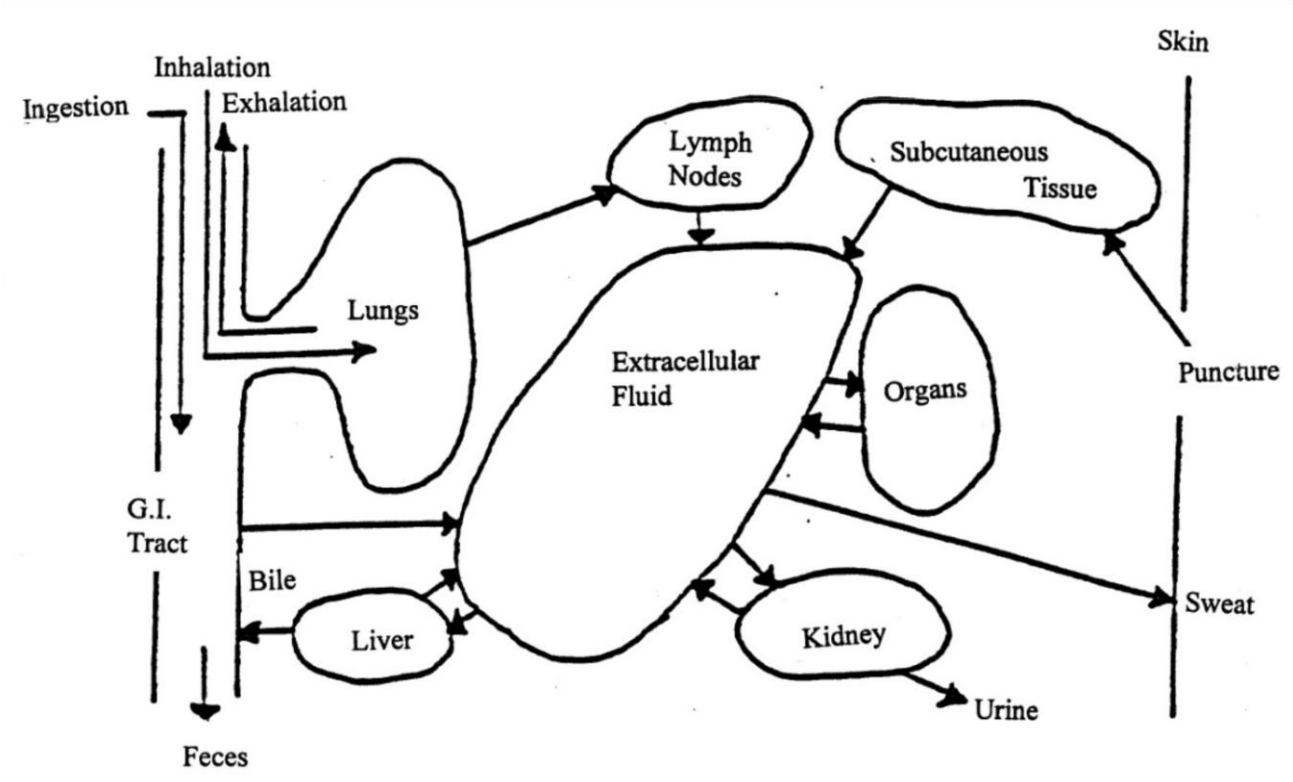
Internal radiation exposure results when the body is contaminated internally with a radionuclide. When radioactive materials enter into the body, they are metabolized and distributed to the tissues according to the chemical properties of the elements and compounds in which they are contained. For example, consider a complex molecule which can be equally satisfied with a C-12 (stable) atom or a C-14 (radioactive) atom at its regular carbon position. If the C-14 decays to nitrogen; the molecular structure is affected. If the molecule were DNA, this might be equivalent to gene mutation. Once radioactive material is in the body, little can be done to speed its removal. Thus, internal radiation protection is concerned with preventing or minimizing the deposition of radioactive substances in personnel.

A. Radioactive Materials in the Body

Radioactive substances, like other toxic agents, may gain entry into the body by four processes:

1. Inhalation - breathing radioactive aerosols or dust
2. Ingestion - drinking contaminated water, or transferring radioactivity to the mouth
3. Absorption – entry through intact skin
4. Injection - puncture of skin with an object bearing radioactive materials

The following diagram is a summary of radionuclide entry, transfer, and exit within the body:



How long a radioactive substance stays in the body is a combination of the radiological half-life of the radionuclide, as described in Chapter 2, and the biological half-life of the substance. The biological half-life is defined as the amount of time it takes for half of the substance to be eliminated from the body by biological means. It is completely independent from the radiological half-life, as it depends entirely on bodily processes such as metabolism, and is not specific to radioisotopes (e.g., radioactive C-14 has the same biological half-life as stable C-12). The combination of the two half-lives is called the effective half-life (T_{eff}), and is calculated from the following equation:

$$T_{eff} = \frac{T_R \, x \, T_B}{T_R + T_B}$$

where: T_R = the radiological half-life

T_B = the biological half-life

Note that the effective half-life is always shorter than either the radiological or biological half-life, since both processes act to eliminate the substance at the same time. Values of T_{eff} for selected radionuclides can be found in various publications, such as the "Radiological Health Handbook".

B. Guidelines

The basic methods to control and prevent radioactive contamination which can lead to internal radiation exposures are:

1. Isolate the contamination at the source: seal samples, use a fume hood, use catch trays lined with paper, change gloves often to avoid cross contamination.

2. Restrict contaminated personnel and articles: separate your radioactive and non-radioactive work areas.

3. Establish and maintain the contamination control zone.

4. Follow established laboratory procedures. Proper protective clothing, designated work areas, surface contamination monitoring, personnel monitoring, etc. are required in all laboratories that use radioactive material.

C. Limits

Limits pertaining to internal emitters are set up for particular radionuclides. These limits are called Annual Limits on Intake (ALI's). A permissible constant ALI of a radionuclide is a quantity (in μCi) which when present continuously in the body will deliver a dose rate not exceeding the maximum permissible dose rate. The constant ALI must not deliver a committed effective dose equivalent of more than 50 rem per year to any individual organ or tissue or more than 5 rem per year to the whole body. The concentrations of radionuclides in air required to yield an ALI are called Derived Air Concentrations (DAC's). A DAC is the concentration of a radionuclide (in microcuries per milliliter) which, when taken into the body on an occupational exposure basis, results in an organ burden which produces the maximum permissible

committed effective dose equivalent, e.g. one ALI to the organ of interest. Some of the factors which are considered in calculating ALI's and DAC's are:

1. the type and energy of the radiation emitted

2. its distribution in the body

3. the solubility/volatility of the compound containing the isotope

4. the effective half-life of the isotope

Reference Table 1 in Chapter 1 which lists values of ALI's and DAC's for selected radionuclides. These values are based on a 40 hour per week exposure, 50 weeks per year.

D. Internal Exposure Monitoring

Internally deposited radioactive material can be monitored by measuring the radiation emitted from the body or by measuring the amount of radioactive material contained in the urine or feces. Such monitoring techniques are called bioassays.

Bioassays are required whenever surveys or calculations indicate that an individual has been exposed to concentrations of radioactive material in excess of established limits or when required by State or Federal regulations.

BIOASSAY PROGRAM

1. Biological Samples

   Biological samples may be taken from all personnel working with heavy elements, millicurie quantities of tritium, or the training reactor, at intervals specified by the Radiation Control Officer. Biological samples will be taken from all personnel who have ingested or who are suspected to have ingested, radioactive material, including other occasions if deemed necessary. Requirements of the bioassay program for tritium are found in the Application of Bioassay for Tritium (Appendix 1).

2. Partial body/whole body counting

   Thyroid monitoring of individuals working with radioiodine is required as specified in the Application of Bioassay for I-125 and I-131 (Appendix 2).

3. Participation

   All personnel working with tritium and radioiodine will receive a questionnaire each month regarding their use of these radionuclides. If the amount of activity used does not meet the participation criteria this fact should be noted on the questionnaire. The questionnaire serves to remind the

individual of the program requirements and to verify participation of all individuals in the program.

Analysis for other radionuclides can be performed upon request.

III. GENERAL PRECAUTIONS AND RULES OF THUMB

General Precautions for Working with Radioactive Materials

1. Always keep radioactive and nonradioactive work separated as far as possible, preferably by maintaining rooms and/or areas used solely for radioactive work.

2. Always work over a spill tray and in a ventilated hood (except with compounds in a nonvolatile form).

3. Always use the minimum quantity of radioactivity compatible with the objectives of the experiment.

4. Always wear protective clothing, safety glasses and gloves when handling radioactivity.

5. Always wash your hands and monitor yourself before leaving a radioactive work area.

6. Always work carefully and monitor the working area regularly with both swipes and a survey meter to avoid ruining experiments by accidental contamination.

7. Always label containers of radioactive material clearly, indicating nuclide, total activity, compound, specific activity, date and the exposure rate at the surface of the container.

8. Never eat, drink, smoke or apply cosmetics in an area where unsealed radioactivity is handled.

9. Never use ordinary handkerchiefs; use paper tissues and dispose of them as radioactive waste as necessary.

10. Never work with cuts or breaks in the skin unprotected, particularly on the hands or forearms.

11. Never pipette radioactive solutions by mouth.

12. **In the event of a spill it is essential to minimize the spread of contamination:**
    a) Cordon off the suspected area of contamination.
    b) Ascertain, if possible, the type of contamination, i.e. the nuclide(s) involved (it may be necessary to use breathing apparatus, protective clothing or other equipment).
    c) Determine the area of contamination by monitoring with a survey meter after taking the necessary precautions.

d) Starting from the outer edge, decontaminate the area in convenient sectors by wiping or scrubbing, if necessary, as described in Chapter 6.
e) Before moving on, ensure that a sector is clean by monitoring with swipes.

13. Dispose of all radioactive waste according to statutory requirements. Short-lived radionuclides, for example, ^{32}P, may be stored with suitable shielding and left to decay. After 4 half-lives less than 10% of the original activity remains, after 7 half-lives <1%, and after 10 half-lives <0.1%. The rule of thumb is to store radioactive waste for 10 half-lives before disposal. For longer-lived radionuclides, for example, ^{3}H, this is impracticable and alternative disposal arrangements must be made.

14. Film badges should be worn for all radioactive work except with the low energy, β-emitting radionuclides ^{3}H, ^{14}C and ^{35}S.

15. Federal and State regulations require that occupational exposure should not exceed 5 rem (50 mSv) per year to the whole body. The exposure of the general public should not exceed 100 mrem (1 mSv) per year to the whole body.

16. To minimize the dose to the extremities, tongs or other remote handling equipment should be used where appropriate.

Rules of Thumb
(from NIH Publication 79-18. DHEW)

Beta Particles

1. Beta particles of at least 70 keV energy are required to penetrate the nominal protective layer of the skin (7mg/cm- or 0.07mm).

2. The average energy of a beta-ray spectrum is approximately one-third the maximum energy.

3. The range of beta particles in air is ~ 12 ft/MeV. (Maximum range of ^{32}P beta is 1.71 MeV x 12 ft/MeV $\simeq$ 20 ft).

4. The dose rate in rads per hour in a solution by a beta emitter is $1.12 EC/\rho$ where E is the average beta energy per disintegration in MeV, C is the concentration in microcuries per cubic centimeter, and ρ is the density of the medium in grams per cubic centimeter. The dose rate at the surface of the solution is one-half the value given by this relation. (For ^{32}P average energy of approximately 0.7 MeV, the dose rate from 1 $\mu Ci/cm^3$ (in water) is 1.48 rad/hr).

5. The surface dose rate through the nominal protective layer of skin (7 mg/cm^2) from a uniform thin deposition of 1 $\mu Ci/cm^2$ is about 9 rad/hour for energies above about 0.5 MeV. Note that in a thin layer, the beta dose rate exceeds the gamma dose rate, for equal energies released, by about a factor of 100.

6. For a point source of beta radiation (neglecting self and air absorption) of activity in millicuries, the dose rate in rad/hr at 1 cm is approximately equal to 200 x activity, and varies only slowly with beta energy. Dose rate for 1 mCi ^{32}P at 1 cm is approximately 200 rad/hour.

Gamma Rays

a. For a point source gamma emitter with energies between 0.07 and 4 MeV, the exposure rate (mR/hr) within approximately 20% at 1 foot = 6CEn, where C is the activity in millicuries; E, the energy in MeV; and n, the number of gammas per disintegration.

b. The dose rate to tissue in rads/hr in an infinite medium uniformly contaminated by a gamma emitter is 2.12 EC/ρ, where C is the number of microcuries per cubic centimeter, E is the average gamma energy per disintegration in MeV, and ρ is the density of the medium. At the surface of a large body, the dose rate is about half of this.

X-Ray

a. The exposure rate at 2 feet from diagnostic x-ray equipment operated at 100 kVp and 100 milliamperes is approximately 2.3 roentgen/second.

b. Exposure rate at the fluoroscopy table with tube potential at 80 kVp and tube current of 1 milliampere should not exceed 2.1 roentgen/minute.

c. Scattered radiation can be as penetrating as the primary beam.

X-Ray Diffraction

a. The x-ray beam intensities from the primary beam can be as much as 400,000 R/min.

b. Scattered radiation 10 cm from the points of scatter about the x-ray tube head has been measured to be on the order of 150 R/hr.

c. The threshold dose sufficient to produce skin erythema is 300 to 400 R.

d. The minimum cataractogenic single dose is 200 rads, while a dose of 750 rads exhibits a high incidence of cataract formation.

Miscellaneous

a. The activity of any radionuclide is reduced to less that 0.1% after 10 half-lives.

b. For material with a half-life greater than six days, the change in activity in 24 hours will be less than 10%.

IV. SPECIFIC HANDLING PRECAUTIONS FOR VARIOUS RADIONUCLIDES

Reference the following pages for guidelines on handling precautions for the following commonly used radioisotopes:

H-3
C-14
P-32
P-33
S-35
Cr-51
I-125

Also included is information on Radioiodination and Safety.

A. Tritium

^3H
12.28y
β-0.019
n$\sigma\gamma$
E 0.019

TRITIUM
Handling Precautions

DuPont has developed the following suggestions for handling tritium after years of experience working with this low energy beta emitter.

PHYSICAL DATA

Maximum Beta Energy: 0.019 MeV (100%)[1]
Maximum Range of Beta in Air: about 4.7 mm (0.19 inches)[2]

OCCUPATIONAL LIMITS[1]

Maximum Permissible Air Concentration (based on forty-hour working week) = 5×10^{-6} μCi/ml (190 kBq/m^3) [3]
Quarterly Inhalation Intake Limit = 6.3 mCi (230 MBq) [3]

DOSIMETRY

Millicurie quantities of tritium do not present an external exposure hazard because the low energy betas emitted cannot penetrate the outer dead layer of skin. The critical organ for tritium uptake is the whole body water. Three to four hours after intake, tritiated water is uniformly distributed in all body water.[5] On average, tritiated water is eliminated with a ten-day biological half-life. Elimination rates may be increased by increasing water intake.[5]

PRECAUTIONS

1. Designate area for handling ^3H and clearly label all containers.
2. Prohibit smoking, eating, and drinking in the room where ^3H is handled.
3. Confine contamination by using transfer pipets, spill trays, and absorbent coverings.
4. Handle potentially volatile compounds in ventilated enclosures.
5. If enhanced containment is necessary, handle volatile compounds in closed systems vented through suitable traps.
6. Sample exhausted effluent by drawing a known quantity through a membrane filter followed by a water impinger.
7. Wear disposable lab coat, gloves and wrist guards for secondary protection.
8. Select gloves appropriate for chemicals handled.
9. Maintain control by regular monitoring and prompt decontamination of gloves and surfaces.
10. Use open window ionization detector or liquid scintillation counter to detect ^3H.
11. Submit periodic urine samples for bioassay to determine uptake by personnel.
12. Isolate, label, and dispose wastes according to approved guidelines.
13. Establish air concentration, surface contamination and bioassay action levels below maximum permissible limits. Investigate any causes that threaten these levels to be exceeded.
14. On completing an operation, secure all ^3H, remove and dispose of protective clothing and coverings, monitor and decontaminate self and surfaces, wash hands and monitor hands again.

Many tritium compounds readily penetrate gloves and skin. Handle these compounds remotely, wear two pairs of gloves and change the outer layer at least every twenty minutes. Tritiated DNA precursers are considered more toxic than tritiated water.[6] However, they are generally less volatile and do not normally present a significantly greater hazard.

REFERENCES

(1) Kocher, David C., Radioactive Decay Data Tables. (Springfield: National Technical Information Service) 1981. DOE/TIC-11026.

(2) Kaplon, Irving, Nuclear Physics, New York: Addison-Wesley, 1964.

(3) 10 CFR 20 – Standards for Protection Against Radiation.

(4) Recommendations of the International Commission on Radiological Protection, ICRP Publication 2, Pergamon Press, London, 1959.

(5) Recommendations of the International Commission on Radiological Protection, ICRP Publication 10, Pergamon Press, London, 1968.

(6) Tritium and Other Radionuclide Labeled Organic Compounds Incorporated in Genetic Material, NCRP Report No. 63, 1979.

B. Carbon-14

^{14}C
5730y
$\beta-$ 0.156
nσγ
E. 0.156

CARBON-14
Handling Precautions

Du Pont has developed the following suggestions for handling carbon-14 after years of experience working with this low energy beta emitter.

PHYSICAL DATA

Maximum Beta Energy: 0.156 MeV (100%)[1]
Maximum Range of Beta in Air: about 22 cm (8.6 inches)[2]

OCCUPATIONAL LIMITS

Maximum Permissible Air Concentration (based on forty-hour working week) = 4×10^{-6} μCi/ml (150 kBq/m³)[3]
Quarterly Inhalation Intake Limit = 2.5 mCi (93 MBq) [3]

DOSIMETRY

Millicurie quantities of ^{14}C do not present a significant external exposure hazard because the low energy betas emitted barely penetrate the horny outer skin layer. The critical organ for uptake of many ^{14}C labeled carbonates is the bone.[5] The critical organ for uptake of many other ^{14}C labeled compounds is the fat.[4] Most ^{14}C labeled compounds are rapidly metabolized and the radionuclide is exhaled as $^{14}CO_2$. Some compounds and their metabolites are eliminated via the urine. Biological half-lives vary from a few minutes to 35 days – ten days being a conservative value for most compounds.[5]

PRECAUTIONS

1. Designate area for handling ^{14}C and clearly label all containers.
2. Prohibit smoking, eating and drinking in the room where ^{14}C is handled.
3. Use transfer pipets, spill trays, and absorbent coverings to confine contamination.
4. Handle potentially volatile compounds in ventilated enclosures.
5. If enhanced containment is necessary, handle volatile compounds in closed systems vented through suitable traps.
6. Sample exhausted effluent by drawing a known quantity through a membrane filter followed by a dilute NaOH impinger.
7. Wear disposable lab coat, gloves and wrist guards for secondary protection.
8. Select gloves appropriate for chemicals handled.
9. Regularly monitor and promptly decontaminate gloves and surfaces to maintain contamination control.
10. Use end window Geiger-Muller detectors or liquid scintillation counter to detect ^{14}C.
11. Submit periodic urine and breath samples (as appropriate) for bioassay to determine uptake by personnel.
12. Isolate, label, and dispose waste according to approved guidelines.
13. Establish air concentration, surface contamination and bioassay action levels below maximum permissible limits. Investigate any causes that threaten these levels to be exceeded.
14. On completing an operation, secure all ^{14}C, remove and dispose coverings, monitor and decontaminate self and surfaces, wash hands and monitor hands again.

Some ^{14}C labeled compounds may penetrate gloves and skin. Handle these compounds remotely; wear two pairs of gloves and change the outer layer frequently. Special caution should be taken when handling ^{14}C labeled halogenated acids. These compounds may be incorporated in the skin and give local dose commitments in the order of 10-100 rad per μCi deposited (3-30 Gy per MBq).

REFERENCES

[1] Kocher, David C., Radioactive Decay Data Tables. (Springfield: National Technical Information Service) 1981. DOE/TIC-11026.
[2] Kaplon, Irving, Nuclear Physics, New York: Addison-Wesley, 1964.
[3] 10 CFR 20 – Standards for Protection Against Radiation.
[4] Recommendations of the International Commission on Radiological Protection, ICRP Publication 2, Pergamon Press, London, 1959.
[5] Recommendations of the International Commission on Radiological Protection, ICRP Publication 10, Pergamon Press, London, 1968.

C. Phosphorus-32

32P
14.29 d
β– 1.71
no γ
E 1.71

PHOSPHORUS-32
Handling Precautions

Du Pont has developed the following suggestions for handling phosphorus-32 after years of experience working with this high energy beta emitter.

PHYSICAL DATA

Maximum Beta Energy:
1.71 MeV (100%)[1]

Maximum Range of Beta in Air
about 6 m (20 feet)[2]

Maximum Range of Beta in Water
about 8 mm (0.3 inch)[2]

Phosphorus-32 Decay Table
Physical Half-Life: 14.29 Days

DAYS	0.0	0.5	1.0	1.5	2.0	2.5	3.0	3.5	4.0	4.5
0	1.000	0.976	0.953	0.930	0.908	0.866	0.865	0.844	0.824	0.804
5	0.785	0.766	0.748	0.730	0.712	0.695	0.678	0.662	0.646	0.631
10	0.616	0.601	0.587	0.573	0.559	0.545	0.532	0.520	0.507	0.495
15	0.483	0.472	0.460	0.449	0.438	0.428	0.418	0.408	0.398	0.388
20	0.379	0.370	0.361	0.353	0.344	0.336	0.328	0.320	0.312	0.305
DAYS 25	0.297	0.290	0.283	0.277	0.270	0.264	0.257	0.251	0.245	0.239
30	0.233	0.228	0.222	0.217	0.212	0.207	0.202	0.197	0.192	0.188
35	0.183	0.179	0.174	0.170	0.166	0.162	0.158	0.155	0.151	0.147
40	0.144	0.140	0.137	0.134	0.130	0.127	0.124	0.121	0.118	0.116
45	0.113	0.110	0.107	0.105	0.102	0.100	0.098	0.095	0.093	0.091
50	0.088	0.086	0.084	0.082	0.080	0.078	0.077	0.075	0.073	0.071
55	0.069	0.068	0.066	0.065	0.063	0.062	0.060	0.059	0.057	0.056
60	0.054	0.053	0.052	0.051	0.049	0.048	0.047	0.046	0.045	0.040

OCCUPATIONAL LIMITS

Maximum Permissible Air Concentration
(based on 40 hour working week) =
7×10^{-8} μCi/ml (2.6 kBq/m³)[3]

Quarterly Inhalation Intake Limits:
44 μCi (1.6 MBq)[3]

DOSIMETRY

The bone is the critical organ for intake of transportable compounds of ^{32}P.[4] Phosphorus metabolism is complex; 30% is rapidly eliminated from the body, 40% possesses a 19-day biological half-life, and the remaining 30% is reduced by radioactive decay.[5] The lung and lower large intestine are the critical organs for inhalation and ingestion, respectively, of non-transportable ^{32}P compounds.

The high energy beta emissions can present a substantial skin dose hazard. Multi 100-millicurie (3.7 GBq) quantities of ^{32}P can produce significant secondary radiation presenting an external exposure hazard.

PRECAUTIONS

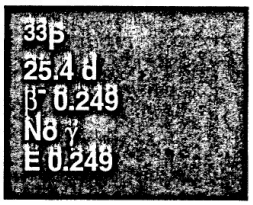
1. Designate area for handling ^{32}P and clearly label all containers.
2. Store ^{32}P behind lead shielding.
3. Wear extremity and whole body dosimeters while handling mCi (37 MBq) quantities.
4. Handle millicurie (37 MBq) quantities of ^{32}P behind 1.3 cm (1/2) inch thick Lucite shielding. Where necessary, increase shielding by attaching 3 to 6 mm (1/8 to 1/4 inch) thick lead sheets to the outside of the Lucite to reduce secondary radiation.
5. Do not work over open containers.
6. Practice routine operations to improve dexterity and speed before using ^{32}P.
7. Avoid skin exposure by using tools to indirectly handle unshielded sources and potentially contaminated vessels.
8. Prohibit smoking, eating, drinking and mouth pipetting in room where ^{32}P is handled.
9. Use transfer pipets, spill trays and absorbent coverings to confine contamination.
10. Handle potentially volatile chemical forms in ventilated enclosures.
11. Sample exhausted effluent and room air by continuously drawing a known quantity through membrane filters.
12. Use lab coat, wrist guards and disposble gloves for secondary protection.
13. Regularly monitor and promptly decontaminate gloves and surfaces to maintain contamination and exposure control.
14. Use end window Geiger-Mueller detector or liquid scintillation counter to detect ^{32}P.
15. Submit urine samples for bioassay from two hours to seven days after handling ^{32}P to indicate uptake by personnel.
16. Isolate waste in clearly labeled shielded container and hold for decay.
17. Establish surface contamination, air concentration, and urinalysis action levels below maximum permissible limits and investigate any causes which threaten these levels to be exceeded.
18. On completing an operation, secure all ^{32}P, remove protective clothing, dispose protective coverings, monitor and decontaminate self and surfaces, wash hands and monitor hands again.

The dose rate at the mouth of an open combi-vial containing 1 mCi (37 MBq) of ^{32}P in 1 ml of liquid is roughly 26 rem/hour (260 mSv/hour).[6] Since this dose rate will not be attenuated significantly by air, shielding materials should be placed between the source and personnel to absorb most of the radiation. The best shield for a ^{32}P source is a material like Lucite 1.3 cm (½ inch) thick or other plastic, which will absorb the beta particles while generating little secondary radiation. For mCi (37 MBq) amounts of ^{32}P, thin, high density shielding such as lead 3 to 6 mm (⅛–¼ inch) thick should be added to the exterior of the Lucite shield to absorb the higher intensity secondary radiation.

A high local dose can be received if the radioactive material is touched and allowed to remain on the skin or gloves. Both the hands and the face can receive a considerable dose of radiation near an open container of ^{32}P, particularly if the radioactivity is in a concentrated form. *Therefore, never work over an open container of ^{32}P.*

REFERENCES
(1) Kocher, David C., Radioactive Decay Data Tables, Springfield: National Technical Information Service, 1981 DOE/TIC-11026.
(2) Kaplan, Irving, Nuclear Physics, New York: Addison Wesley 1964.
(3) 10 CFR 20 — Standards for Protection Against Radiation.
(4) Recommendations of the International Commission on Radiological Protection, ICRP Publication 2, Pergamon Press, London, 1959.
(5) Recommendations of the International Commission on Radiological Protection, ICRP Publication 10, Pergamon Press, London, 1968.
(6) Measurements made by Safety and Environmental Affairs Department using Harshaw Chemical Company TLD 100 chips.

NEN® Phosphorus-33 Technical and Handling Information



Physical Data

Maximum Beta Energy = 0.249 MeV (100%)[1]

Maximum Range of Beta in Air = 46 cm (18 in.)[2]

Occupational Limits

Annual Limit on Intake = 3 mCi (110 MBq)[3]

Derived Air Concentration = 1×10^{-6} µCi/mL (37 kBq/m³)[3]

Dosimetry

Millicurie (37 MBq) quantities of ^{33}P do not present a significant external exposure hazard because the low-energy betas emitted barely penetrate gloves and the outer dead layer of skin. The bone is the critical organ for intake of transportable compounds of ^{33}P.[4] Phosphorus metabolism is complex; 30% is rapidly eliminated from the body, 40% possesses a 19-day biological half life and the remaining 30% is reduced by radioactive decay.[5] The lung and the lower large intestine are the critical organs for inhalation and ingestion, respectively, of non-transportable ^{33}P compounds.

General Handling Precautions For Phosphorus-33

Du Pont has developed the following suggestions for handling phosphorus-33 after years of experience working with this low-energy beta emitter.

1. Designate area for handling ^{33}P and clearly label all containers.
2. Prohibit eating, drinking, smoking and mouth pipetting in room where ^{33}P is handled.
3. Use transfer pipettes, spill trays and absorbent coverings to confine contamination.
4. Use lab coat, wrist guards and disposable gloves for secondary protection.
5. Select gloves appropriate for chemicals handled.
6. Regularly monitor and promptly decontaminate gloves and surfaces to maintain contamination control.
7. Use end-window Geiger-Mueller detector or liquid scintillation counter to detect ^{33}P.

Phosphorus-33 Decay Data Table

Physical Half Life: 25.4 Days[1]

Days	0	1	2	3	4	5	6	7	8	9
0	1.000	0.973	0.947	0.921	0.897	0.872	0.849	0.826	0.804	0.782
10	0.761	0.741	0.721	0.701	0.683	0.664	0.646	0.629	0.612	0.595
20	0.579	0.564	0.549	0.534	0.520	0.506	0.492	0.479	0.466	0.453
30	0.441	0.429	0.418	0.406	0.395	0.385	0.374	0.364	0.355	0.345
40	0.336	0.327	0.318	0.309	0.301	0.293	0.285	0.277	0.270	0.263
50	0.256	0.249	0.242	0.236	0.229	0.223	0.217	0.211	0.205	0.200
60	0.195	0.189	0.184	0.179	0.174	0.170	0.165	0.161	0.156	0.152
70	0.148	0.144	0.140	0.136	0.133	0.129	0.126	0.122	0.119	0.116
80	0.113	0.110	0.107	0.104	0.101	0.098	0.096	0.093	0.091	0.088
90	0.086	0.084	0.081	0.079	0.077	0.075	0.073	0.071	0.069	0.067
100	0.065	0.064	0.062	0.060	0.059	0.057	0.055	0.054	0.053	0.051
110	0.050	0.048	0.047	0.046	0.045	0.043	0.042	0.041	0.040	0.039
120	0.038	0.037	0.036	0.035	0.034	0.033	0.032	0.031	0.030	0.030

8. Handle ^{33}P compounds that are potentially volatile or in powder form in ventilated enclosures.
9. Sample exhausted effluent and room air by continuously drawing a known quantity through membrane filters.
10. Submit urine samples for bioassay from two hours to seven days after handling ^{33}P to indicate uptake by personnel.
11. Isolate waste in clearly labeled shielded container and hold for decay.
12. Establish surface contamination, air concentration and urinalysis action levels below regulatory limits. Investigate and correct any causes that may threaten these levels to be exceeded.
13. On completing an operation, secure all ^{33}P; remove protective clothing; dispose of protective coverings; monitor and decontaminate self and surfaces; wash hands and monitor hands again. ■

References

1. Kocher, David C., Radioactive Decay Data Tables, Springfield: National Technical Information Service, 1981 DOE/TIC-11026.
2. Kaplan, Irving, Nuclear Physics, New York: Addison-Wesley, 1964.
3. Revised 10 CFR 20 – Standards for Protection Against Radiation, 1991.
4. Limits for Intakes of Radionuclides by Workers, ICRP Publication 30, Pergamon Press, Oxford, 1980.
5. Recommendations of the International Commission on Radiological Protection, ICRP Publication 10, Pergamon Press, London, 1968.

^{35}S
87.4 d
β- 0.167
n$\sigma\gamma$
E 0.167

SULFUR-35
Handling Precautions

Du Pont has developed the following suggestions for handling sulfur-35 after years of experience working with this low energy beta emitter.

PHYSICAL DATA

Maximum Beta Energy: 0.167 MeV (100%)[1]
Maximum Range of Beta in Air: about 24 cm (9.6 inches)[2]

OCCUPATIONAL LIMITS

Maximum Permissible Air Concentration (Based on a forty-hour working week) = 3×10^{-7} μCi/ml (11 kBq/m^3)[3]
Quarterly Inhalation Intake Limit = 190 μCi (7.0 MBq) [3]

DOSIMETRY

Millicurie quantities of ^{35}S do not present a significant external exposure hazard since the low energy emissions barely penetrate the horny outer layer of skin. The critical organ for ^{35}S is the whole body.[5] The elimination rate of ^{35}S depends on the chemical form. Most ^{35}S labeled compounds are eliminated via the urine. Ninety days is a conservative biological half-life.[4]

Sulfur-35 Decay Table
Physical Half-Life: 87.4 Days

DAYS	0	3	6	9	12	15	18	21	24	27
0	1.000	.976	.954	.931	.090	.888	.867	.847	.827	.807
30	.788	.770	.752	.734	.717	.700	.683	.667	.652	.636
60	.621	.607	.593	.579	.565	.552	.539	.526	.514	.502
90	.490	.478	.467	.456	.445	.435	.425	.415	.405	.395
120	.386	.377	.368	.360	.351	.343	.335	.327	.319	.312
150	.304	.297	.290	.283	.277	.270	.264	.258	.252	.246
180	.240	.234	.229	.223	.218	.213	.208	.203	.198	.194
210	.189	.185	.180	.176	.172	.168	.164	.160	.156	.153
240	.149	.146	.142	.139	.136	.132	.129	.126	.123	.120
270	.118	.115	.112	1.09	.107	.104	.102	.100	.097	.095
300	.093	.090	.088	.086	.084	.082	.080	.078	.077	.075
330	.073	.071	.070	.068	.066	.065	.063	.062	.060	.059
360	.058	.056	.055	.054	.052	.051	.050	.049	.048	.046

PRECAUTIONS

1. Designate area for handling ^{35}S and clearly label all containers.
2. Prohibit smoking, eating and drinking in room where ^{35}S is handled.
3. Use transfer pipets, spill trays and absorbent coverings to confine contamination.
4. Handle potentially volatile compounds in ventilated enclosures.
5. If enhanced containment is necessary, handle volatile compounds in closed systems vented through suitable traps.
6. Sample exhausted effluent by drawing a known quantity through a membrane filter followed by an impinger containing dilute NaOH.
7. Wear disposable lab coat, gloves and wrist guards for secondary protection.
8. Select appropriate gloves for chemicals handled.
9. Regularly monitor and promptly decontaminate gloves and surfaces to maintain contamination control.
10. Use end window Geiger-Muller detector or liquid scintillation counter to detect ^{35}S.
11. Submit periodic urine samples for bioassay to determine uptake by personnel.
12. Isolate, label and dispose waste according to approved guidelines.
13. Establish air concentration, surface contamination and bioassay action levels below maximum permissible limits. Investigate any causes that threaten these levels to be exceeded.
14. On completing an operation, secure all ^{35}S, remove and dispose protective clothing and coverings, monitor and decontaminate self and surfaces, wash hands and monitor hands again.

^{35}S may be difficult to distinguish from ^{14}C because the beta emissions are of similar energy. If both radionuclides are being used in the same area, establish controls which are conservative for both.

REFERENCES
[1] Kocher, David C. Radioactive Decay Data Tables. (Springfield: National Technical Information Service) 1981. DOE/TLC-11026.
[2] Kaplon, Irving, Nuclear Physics, New York: Addison-Wesley, 1964.
[3] 10 CFR 20 – Standards for Protection Against Radiation.
[4] Recommendations of the International Commission on Radiological Protection, ICRP Publication 2, Pergamon Press, London, 1959.
[5] Recommendations of the International Commission of Radiological Protection, ICRP Publication 10, Pergamon Press, London, 1968.

F. Chromium-51

PHYSICAL DATA[1]

Principal Radiation Emissions

Gamma	0.320 MeV	(9.8%)
X-ray	0.005 MeV	(22.3%)
Auger Electrons	0.004 MeV	(66.9%)

Half Value Layer for Lead Shielding = 1.7 mm[2] (0.067 inches)
Unshielded Exposure Rate from 1 mCi
Point Source at 1 cm = 0.18 $R \cdot h^{-1}$ [2]
(Unshielded Exposure Rate from 1 MBq
Point Source at 1 m = 0.13 $nC \cdot kg^{-1} \cdot h^{-1}$

OCCUPATIONAL LIMITS

Maximum Permissible Air Concentration based on a
40 hour working week = $1 \times 10^{-5} \mu Ci/ml$ (370 kBq/m^3)
Quarterly Inhalation Intake Limit = 6.3 mCi[3] (230 MBq)

CHROMIUM 51 DECAY

The table below gives the decay of Chromium 51 (Half Life = 27.7 Days

DAYS	0	1	2	3	4	5	6	7	8	9
0	1.000	.975	.951	.928	.905	.882	.861	.839	.819	.798
10	.779	.759	.741	.722	.705	.687	.670	.654	.637	.622
20	.606	.591	.577	.562	.549	.535	.522	.509	.496	.484
30	.472	.460	.449	.438	.427	.417	.406	.396	.386	.377
40	.368	.359	.350	.341	.333	.324	.316	.309	.301	.293
50	.286	.279	.272	.266	.259	.253	.246	.240	.234	.229
60	.223	.217	.212	.207	.202	.197	.192	.187	.182	.178
70	.174	.169	.165	.161	.157	.153	.149	.146	.142	.139
80	.135	.132	.129	.125	.122	.119	.116	.113	.111	.108
90	.105	.103	.100	.098	.095	.093	.091	.088	.086	.084
100	.082	.080	.078	.076	.074	.072	.071	.069	.067	.065
110	.064	.062	.061	.059	.058	.056	.055	.054	.052	.051
120	.050	.048	.047	.046	.045	.044	.043	.042	.041	.040

DOSIMETRY

The lower large intestine is the critical organ for intake of soluble ^{51}Cr compounds and ingestion of insoluble compounds.[4] The lung is the critical organ for inhalation of insoluble compounds.[4] An uptake of chromium is slowly eliminated from the body equally via urine and feces with a biological half life of 616 days.[4] The dose committed is reduced by the short physical half life of ^{51}Cr.

PRECAUTIONS

1. Designate area for handling ^{51}Cr and clearly label all containers.
2. Store ^{51}Cr behind lead shielding.
3. Wear extremity and whole body dosimeters while handling mCi quantities.
4. Use shielding to minimize exposure while handling ^{51}Cr.
5. Use tools to indirectly handle unshielded sources and potentially contaminated vessels.
6. Prohibit eating, drinking and smoking in room where ^{51}Cr is handled.
7. Use transfer pipets, spill trays and absorbent coverings to confine contamination.
8. Handle potentially volatile chemical forms and powders in ventilated enclosures.
9. Sample exhausted effluent and room air by continuously drawing a known volume through membrane filters.
10. Use lab coat, wrist guards and disposable gloves for secondary protection.
11. Regularly monitor and promptly decontaminate gloves and surfaces to maintain contamination and exposure control.
12. Use end window Geiger-Mueller detectors. NaI (Tl) detector or liquid scintillation counter to detect ^{51}Cr.
13. Submit urine samples for bioassay at least 4 hours after handling ^{51}Cr to indicate uptake by personnel.
14. Isolate waste in clearly labeled shielded containers and dispose according to approved guidelines.
15. Establish surface contamination, air concentration and urinalysis action levels below maximum permissible levels and investigate and correct any causes which threaten these levels to be exceeded.
16. On completing an operation, secure all ^{51}Cr, remove and dispose protective clothing and coverings, monitor and decontaminate self and surfaces, wash hands and monitor them again.

^{51}Cr is slowly eliminated from the body. Whole body counting provides a more sensitive method than urinalysis for determining ^{51}Cr body burdens. Whole body counting may be used occasionally to verify the bioassay results.

REFERENCES

[1]*Kocher, David C., Radioactive Decay Tables, Springfield: National Technical Information Service, 1981 DOE/TIC-11026.*
[2]*Calculated with Computer Code 'Gamma' Utilizing Decay Scheme Data from Kocher and Mass Attenuation Coefficients for Lead and Mass Energy Absorption Coefficients for Air from the Radiological Health Handbook, Washington: Bureau of Radiological Health, 1970. The HVL reported here is the initial HVL for narrow beam geometry.*
[3]*10 CFR 20 — Standards for Protection Against Radiation.*
[4]*Recommendations of the International Commission on Radiological Protection, ICRP Publication 2, Permagon Press, London, 1959.*

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G. Iodine-125

125I
60 d
EC
γ 0.035
E 0.177

IODINE-125
Handling Precautions

Du Pont has developed the following suggestions for handling iodine-125 after years of experience working with this low energy gamma and x-ray emitter.

PHYSICAL DATA

Principle Radiation Emissions[1]

Gamma	0.035 MeV (6.5%)
Kα X-rays	0.027 MeV (112.0%)
Kβ X-rays	0.031 MeV (25.4%)

Unshielded Exposure Rate at 1 cm from 1 mCi Point Source = 1.4 $R.h^{-1}$[2]

(Unshielded Exposure Rate at 1 m from a 1 MBq Point Source = 0.98 nC $kg^{-1}h^{-1}$)

Half Value Layer for Lead Shielding = 0.02 mm (0.001 inches)[2]

Iodine-125 Decay Table
Physical Half-Life: 60 Days

DAYS	0	2	4	6	8	10	12	14	16	18
0	1.000	0.977	0.955	0.933	0.912	0.891	0.871	0.851	0.831	0.812
20	0.794	0.776	0.758	0.741	0.724	0.707	0.691	0.675	0.660	0.645
40	0.630	0.616	0.602	0.588	0.574	0.561	0.548	0.536	0.524	0.512
60	0.500	0.489	0.477	0.467	0.456	0.445	0.435	0.425	0.416	0.406
80	0.397	0.388	0.379	0.370	0.362	0.354	0.345	0.338	0.330	0.322
100	0.315	0.308	0.301	0.294	0.287	0.281	0.274	0.268	0.262	0.256
120	0.250	0.244	0.239	0.233	0.228	0.223	0.218	0.213	0.208	0.203
140	0.198	0.194	0.189	0.185	0.181	0.177	0.173	0.169	0.165	0.161
160	0.157	0.154	0.150	0.147	0.144	0.140	0.137	0.134	0.131	0.128
180	0.125	0.122	0.119	0.117	0.114	0.111	0.109	0.106	0.104	0.102
200	0.099	0.097	0.095	0.093	0.090	0.088	0.086	0.084	0.082	0.081
220	0.079	0.077	0.075	0.073	0.072	0.070	0.069	0.067	0.065	0.064
240	0.063	0.061	0.060	0.058	0.057	0.056	0.054	0.053	0.052	0.051

OCCUPATIONAL LIMITS

Maximum Permissible Air Concentration (based on 40 hour working week) = 5×10^{-9} μCi/ml (190 Bq/m^3)[3].

Quarterly Inhalation Intake Limit: 3.2 μCi (120 kBq).[3]

Maximum Permissible Thyroid Burden = 1.2 μCi (44 kBq).[5]

DOSIMETRY

The thyroid is the critical organ for 125I uptake. Individual uptake and metabolism vary over a wide range. The thyroid may be assumed to accumulate 30% of soluble radioiodine uptake to the body and retain iodine with a 138 day biological half-life.[4] All radioiodine in the body can be assumed to be eliminated via the urine.[6]

PRECAUTIONS

1. Designate area for handling 125I and clearly label all containers.
2. Store mCi (37 MBq) quantities of 125I in containers surrounded by 3 mm (1/8″) thick lead.
3. Use tools to prevent direct handling of potentially contaminated vessels and unshielded multi mCi (37 MBq) sources.
4. Prohibit smoking, eating, drinking and mouth pipetting in the laboratory where 125I is handled.
5. Use transfer pipets, spill trays and absorbent coverings to confine contamination.
6. Handle quantities greater than 10 μCi (370 kBq) in a ventilated enclosure.
7. Handle mCi (37 MBq) quantities in closed systems vented through activated charcoal traps.
8. Sample exhausted effluent by continuously drawing a known quantity of air through cartridges containing activated charcoal.
9. Use disposable lab coat, gloves and wrist guards for secondary protection.
10. Select gloves appropriate for chemicals handled.
11. Regularly monitor and promptly decontaminate gloves and surfaces to maintain contamination control.
12. Use NaI (Tl) detector or liquid scintillation counter to detect 125I.
13. Submit urine samples for bioassay from 2 to 12 hours after handling 125I to indicate uptake by personnel.
14. Conduct periodic thyroid counts to determine dose.
15. Isolate waste in sealed labeled containers.
16. Establish surface contamination, air concentration, urinalysis, and thyroid burden action levels below maximum permissible limits and investigate any causes which threaten these levels to be exceeded.

Store Na 125I solutions at room temperature because freezing results in subsequent volatilization of radioiodine. Avoid acidic solutions to minimize volatilization. Some radioiodine compounds may penetrate gloves and skin. When handling these compounds use remote tools, wear two pairs of gloves and change the outer pair frequently or whenever suspected to be contaminated.

REFERENCES

(1) Kocher, David C., Radioactive Decay Data Tables, Springfield: National Technical Information Service, 1981 DOE/TIC-11026.

(2) Calculated with Computer Code 'Gamma' Utilizing Decay Scheme Data from Kocher and Mass Attenuation Coefficients for Lead and Mass Energy Absorption Coefficients for Air from the Radiological Health Handbook, Washington: Bureau of Radiological Health, 1970. The HVL reported here is the initial HVL for narrow beam geometry.

(3) 10 CFR 20 — Standards for Protection Against Radiation.

(4) Recommendations of the International Commission on Radiological Protection, ICRP Publication 2, Pergamon Press, London, 1959.

(5) Calculated using ICRP2 methods with a quality factor of 1.

(6) Evaluation of Radiation Doses to Body Tissues from Internal Contamination Due to Occupational Exposure, ICRP Publication 10, Pergamon Press, 1968.

H. Radioiodination And Safety

Introduction

Publications of the International Commission for Radiation Protection (ICRP) deal with the basic principles of radiation protection, leaving the various national protection committees (NCRP) the right and responsibility for introducing the detailed technical regulations, recommendations or codes of practice best suited to the needs of their individual countries. Because ICRP publications are circulated internationally, it is not possible to discuss the regulations and requirements of any country in detail. It is possible, however, to give general recommendations and guidelines. For further details the reader should consult his national (NCRP) and state regulatory authority.

There are two distinct levels of hazard in laboratories where radioactive iodine is handled. In the first case, where purified iodine-labeled proteins or other antigens, or commercially available radioimmunoassay kits are handled, the activity is used at the μCi (tens of kBq) level, and there is little radiation hazard **provided that** surfaces in the laboratory are regularly monitored (swiped) and that normal good handling practice is observed.

However, in areas where iodine labeling reactions are carried out, solutions of sodium iodide at the mCi (tens of MBq) level may be handled, frequently at high radioactive concentration, and this may present significant hazards both of radiation exposure rate and of internal and external contamination. Various generally accepted limits are given in Table 1. The critical organ for iodine is the thyroid, which may accumulate 30% or more of the total iodine ingested.

Table 1

Nuclide	Half-life days	Annual limit on intake* μCi	Derived air concentration μCi/ml
^{125}I	60	60	3×10^{-8}
^{131}I	8	50	2×10^{-8}

*Annual limit on intake figures are for inhalation. Figures taken from 10 CFR Part 20, Appendix B.

Radiation exposure rate

There are two problems involved in working with most radionuclides, namely exposure rate and contamination levels.

The exposure rate associated with Iodine-125 may be high. For example, the exposure rate on the outside of a glass vial containing a few mCi (a few tens of MBq), of Iodine-125 in a concentrated solution is of the order of R/hr. Shielding for Iodine-125 is relatively easy, however, (see Table 2) but during work with concentrated solutions the hands and body should be kept at the maximum practicable **distance** by the use of tongs or other remote handling devices, the iodinations should be carried out **rapidly** and the high concentration

solution **shielded** immediately after use. For example, columns used for the purification of labeled protein preparations can be effectively shielded by inserting them into copper piping.

For Iodine-131 much more shielding is required, and local screening is often needed when performing the actual radioiodination. In addition, the exposure rate within the working environment should be monitored with a suitable survey meter. In general the exposure rate should be kept as low as reasonably practicable and normally less than 2.0 mR/hr.

Table 2

Nuclide	Principal Photon Emissions (MeV)	Dose rate from 1mCi, (37Mbq) in mR/h		
		at surface	at 10cm	at 10cm through 6mm lead
^{125}I	0.027, 0.031	1400	14	0.05
^{131}I	0.36, 0.64	----	210	55†

*Values measured at Amersham in a standard P10 vial containing 0.1 ml solution using thermoluminescent dosimeters.

†Values calculated at Amersham

It should also be noted that solutions of iodine or iodine compounds are supplied by manufacturers packaged in a form which meets transport regulations. This does not imply that the packaging as supplied is suitable for storage, since the requirements for transport packaging are based on the fact that the material is transient, and is not in a position to give a dose to any person for extended periods of time. **For storage** the requirements are quite different: the material may be stored in the same corner of the same laboratory for long periods of time, and **may well need more shielding than that provided by the manufacturer for transport**.

Facilities

For work involving more than 1 mCi (37MBq) of radioiodine, the laboratory should meet the standards as described in the University of Florida's "Radiation Control Guide", "Laboratory Safety Manual" and "Biological Safety Manual". In particular all radioiodination operations must be performed in a ventilated, stainless steel lined non-manifolded hood and not on the open bench. The linear flow rate through any working aperture of a ventilated enclosure should be 150 lfm as verified by the EH&S Occupational Safety and Research Department. Consideration should be given to filtration of the radioactive material from such hoods.

Operations with radioiodine at high radioactive concentration are particularly hazardous because the various manipulations may result in minute droplets, commonly representing up to 2×10^{-3} μCi (~75 Bq) becoming airborne. Steps should be taken to avoid these: for example, uncapped vials should never be vortex-mixed. However, such droplets can never be totally eliminated, for the simple action of removing the vial cap can produce them. A purpose-built decapping tool should be used whenever possible (these are commercially available). Operators carrying out radioiodinations should always wear surgical type gloves, which should be monitored frequently. They should be changed if contamination levels exceed 100 dpm, (1.7Bq). An additional potential risk is that radioiodine in some forms can penetrate gloves, thereby contaminating the skin. An added precaution can be to wear

polyethylene gloves over surgical gloves or double glove. In view of the contamination problem with volatile iodine, adequate space must be available within the hood for disposal of waste and to enable the work to be laid out logically in trays. Routine contamination surveys of the working areas to ensure that levels are kept below 100dpm/100cm^2 (1.7Bq/100cm^2), are essential.

Where iodine is being used regularly, monitoring by air samples can be performed to ensure that the concentration of iodine in laboratory air is measured reliably and action taken to ensure that it is as low as reasonably practicable and is in any case less than the derived air concentration shown in Table 1. Personnel working with radioiodine should be monitored by a thyroid uptake bioassay using a detector containing a sodium iodide crystal.

Where an incident involving inhalation or ingestion of iodine is suspected, external monitoring over the thyroid within a few hours of the event provides an indication of the seriousness of the uptake.

Emergency protective action

1. Thyroidial uptake of radioiodine can be blocked by the oral administration of an excess of stable iodine. A physician should be consulted with a view to maintaining a supply of stable iodine for administration in the event of a suspected radioiodine intake.

   A suitable blocking dose is 100mg of iodine. It should be taken as soon as possible after the incident and under medical supervision. The iodine can be conveniently administered in the form of sodium or potassium iodide or iodate tablets. Since these materials may cause mild discomfort in some cases, they should be given only after suspected accidental intakes.

2. When a spillage involving radioiodine occurs, the spilled material should be treated with an excess of sodium thiosulphate solution to render it chemically stable prior to beginning the decontamination operation.

Practical aspects

The following is a summary of practical hints for use during radioiodination reactions.

1. A sensitive survey meter suitable for the radionuclide being used must be at hand and switched on throughout the operation.

2. Clothing, especially sleeves, cuffs and gloves, should be monitored frequently. Gloves should be replaced when contaminated (>100dpm/100cm^2 (1.7Bq/100cm^2)).

3. A 'recipe' or protocol should be pinned up on the fume hood. Apart from the convenience, it can prove very helpful in an emergency.

4. A practice or 'dummy' run should be carried out. Every movement and operation should be carefully considered and rehearsed.

5. An assistant who is familiar with the process should be available to help if anything unforeseen occurs.

6. The steps in the actual radioiodination should be carried out in a systematic and unhurried manner.

7. Solutions of sodium thiosulphate and carrier iodine for use in dealing with a spill or other unplanned release should be readily available.[1] The fume hood should be supplied with the following:

   - A good supply of disposable rubber gloves.
   - A good supply of tissues.
   - A designated radioactive waste can with lid for highly contaminated solid waste, contaminated gloves, tissues, etc.
   - Local shielding, if appropriate.
   - A large tray which could contain at least ten times the volume of any foreseeable leakage.
   - A small tray for iodination pipettes, etc. which must be reserved for iodinations and not taken from the hood.
   - A decapper for removing the cap from the vial of iodine.
   - The ventilation conditions of any hood used should be checked to ensure that they are satisfactory.

[1] A suitable solution is prepared from sodium thiosulphate (25g) and sodium iodine (2g) in 1 liter of N-sodium hydroxide. (**Note:** Care should be taken to prevent sodium hydroxide splashing into the eyes).

APPENDIX 1

APPLICATION OF BIOASSAY FOR TRITIUM

1. CONDITIONS UNDER WHICH BIOASSAY IS NECESSARY

a. Routine bioassay is necessary when quantities processed by an individual at any one time or the total amount processed per month exceed those for the forms of tritium shown in Table 1.

TABLE 1: ACTIVITY LEVELS ABOVE WHICH TRITIUM BIOASSAY IS REQUIRED

Types of Operation	HTO and Other Tritiated Compounds (Including Nucleotide Precursors)
Processes in open room or bench with possible escape of tritium from process vessels	25 mCi (925 Mbq)
Processes with possible escape of tritium carried out within a fume hood of adequate design, face velocity and performance reliability	25 mCi (925 MBq)
Processes carried out within gloveboxes that are ordinarily closed but with possible release of tritium from process vessels and occasional exposure to contaminated box and box leakage	250 mCi (9250 MBq)

b. Bioassay is not required, but recommended, when process quantities handled by a worker are less than those in Table 1.

2. PARTICIPATION

All workers involved in the processing of tritium under conditions specified in Section 1 or in the environs of the process should participate in the bioassay program.

3. TYPES OF BIOASSAY THAT SHOULD BE PERFORMED

3.1 Baseline (pre-employment or pre-operational)

A baseline bioassay should be conducted not more than one month prior to beginning work with tritium in amounts that would require participation in the bioassay program.

3.2 Routine Urinalysis

Regular bioassays should be conducted to monitor routine operations at frequencies specified in Section 4.

3.3 Emergency

If the initial sample or other data indicates a possible exposure high enough to warrant immediate medical attention, a complete and immediate follow-up should be conducted as described in Item b of Section 5.1.

3.4 Post-Operational and Termination of Usage

A bioassay should be performed within one month of the last possible exposure to tritium such as when operations are being discontinued, or when the worker is terminating activities with potential exposure.

3.5 Diagnostic

Follow-up bioassay should be performed as soon as possible but within one week of any sample exceeding levels given as action points in Section 5, in order to confirm the initial results and in the case of a single intake, to allow an estimate of the effective half-life of the tritium in the body.

4. FREQUENCY

Initial Routine

A bioassay sample of at least 50 ml of urine should be taken within 72 hours following entry of an individual into an area where operations require bioassay according to Section 1 and then every month or more frequently thereafter, as long as the individual is working with tritium. When work with tritium is on an infrequent basis (less frequently than every month), bioassay should be performed within 10 days of the end of the work period during which tritium was handled.

5. ACTION POINTS AND CORRESPONDING ACTIONS

5.1 Monthly and Other Sampling

a. If urinary excretion rates exceed 5 μCi/ml (0.18 MBq/ml), but are less than 50 μCi/ml (1.8 MBq/ml), the following course of action should be taken:

(1) A survey of the operations involved, including air and surface contamination monitoring, should be carried out to determine the causes of the exposure and evaluate the potential for further exposures or for the possible involvement of other employees.

(2) Any reasonable corrective actions that the survey indicates may lower the potential for further exposures should be implemented.

(3) A repeat urine sample should be taken within one week of the previous sample and should be evaluated within a week after collection. Internal dose commitments should be estimated using at least these two urine sample evaluations and other survey data, including the probable times of the intake of tritium.

(4) Any evidence indicating that further work in the area might result in an employee receiving a dose commitment in excess of the limits established in 64E-5.304, should serve as cause to remove the employee from work in this operation until the sources of exposure is discovered and corrected.

(5) Reports or notification must be provided as required by 64E-5.344 and 64E-5.345 of Chapter 64E-5 or as required by conditions of the license.

b. If urinary excretion rates exceed 50 μCi/ml (1.8 MBq/ml), the following course of action should be taken:

(1) Carry out all steps in Item a of this regulatory position.

(2) If the projected dose commitment exceeds levels for whole body as provided in 64E-5.344 and 345 of Chapter 64E-5, provide appropriate notification to DOH.

(3) Refer the case to appropriate medical/health physics consultation for recommendations regarding immediate therapeutic procedures that may be carried out to accelerate removal of tritium from the body and reduce the dose to as low as is reasonably achievable.

(4) Carry out repeated sampling (urine collections of at least 100 ml each) at approximately one week intervals at least until samples show an excretion rate less than 5 μCi/ml (0.18 MBq/ml). If there is a possibility of long term organic compartments of tritium that require evaluation (reference NUREG-0938), continue sampling as long as necessary to ensure that appreciable exposures to these other compartments do not go undetected and to provide estimates of total dose commitments.

APPENDIX 2

APPLICATIONS OF BIOASSAY FOR I-125 AND I-131

1. CONDITIONS UNDER WHICH BIOASSAY IS NECESSARY

a. Routine bioassay is necessary when an individual handles in open form, unsealed quantities of radioactive iodine that exceed those shown in Table 1 (page 41). The quantities shown in Table 1 apply to both the quantity handled at any one time or integrated as the total amount of activity introduced into a process by an employee over a three month period.

b. Bioassay is not required, but recommended, when process quantities handled by a worker are less than those in Table 1.

2. PARTICIPATION

All workers handling radioactive iodine or sufficiently close to the process so that intake is possible (i.e., within a few meters and in the same room as the worker handling the material), shall participate in bioassay programs.

3. TYPES OF BIOASSAY THAT SHOULD BE PERFORMED

3.1 Baseline

A baseline bioassay should be conducted prior to beginning work with radioactive iodine in amounts that would require participation in the bioassay program.

3.2 Routine

Regular bioassay should be conducted to monitor routine operations at the frequency specified in Section 4.

3.3 Emergency

A bioassay should be performed as soon as possible after any incident that might cause thyroid uptakes to exceed burdens given in Section 5.1 so that actions recommended in Item b Section 5.1 can be most effective.

3.4 Post-Operational and Termination of Usage

A bioassay should be performed within 2 weeks of the last possible exposure to I-125 or I-131 when operations are being discontinued or when the worker is terminating activities with potential exposure to these radionuclides.

3.5 Diagnostic

Follow-up bioassay should be performed within 2 weeks of any measurements exceeding levels given as action points in Section 5. in order to confirm the initial results and in the case of a single intake, to allow an estimate of the effective half-life of radioiodine in the thyroid.

4. FREQUENCY

4.1 Initial Routine

A bioassay sample or measurement should be obtained within 72 hours following entry of an individual into an area where bioassay is performed in accordance with Sections 1. and 2. and every 4 weeks or more frequently thereafter, as long as the conditions described in Sections 1. and 2. exist. When work with radioactive iodine is on an infrequent basis, (less frequently than every 4 weeks), bioassay should be performed within 10 days of the end of the work period during which radioactive iodine was handled, unless emergency action is appropriate.

5. ACTION POINTS AND CORRESPONDING ACTIONS

5.1 Monthly and Other Measurements

a. Whenever the thyroid burden at the time of measurement exceeds 0.12 μCi (4.4 kBq) of I-125 or 0.04 μCi (1.5 kBq) of I-131, the following actions will be taken:

(1) An investigation of the operations involved, including air and other in-house surveys, will be carried out to determine the causes of exposure and to evaluate the potential for further exposures or for the possible involvement of other employees.

(2) Any evidence indicating that further work in the area might result in an employee receiving a dose commitment in excess of the limits established in 64E-5.304, should serve as cause to remove the employee from work in this operation until the sources of exposure is discovered and corrected.

(3) Reports or notification must be provided as required by 64E-5.344 and 64E-5.345 of Chapter 64E-5 or as required by conditions of the license.

(4) Corrective actions that will eliminate or lower the potential for further exposures should be implemented.

(5) A repeat bioassay should be taken within 2 weeks of the previous measurement and should be evaluated within 24 hours after measurement in order to confirm the presence of internal radioiodine

and to obtain an estimate of effective half-life for use in estimating dose commitment.

b. If the thyroid burden at any time exceeds 0.5 μCi (18.5 kBq) of I-125 or 0.14 μCi (5.2 kBq) of I-131, the following actions will be taken:

(1) Carry out all steps in Item a of this regulatory position.

(2) If the projected dose commitment exceeds levels for whole body as provided in 64E-5.344 and 345 of Chapter 64E-5, provide appropriate notification to DOH.

(3) As soon as possible, the case will be referred to appropriate medical consultation for recommendations regarding therapeutic procedures that may be carried out to accelerate removal of radioiodine from the body. This should be done within 2-3 hours after exposure when the time of exposure is known so that any prescribed thyroid blocking agent would be effective.

(4) Carry out repeated measurements at approximately 1 week intervals at least until the thyroid burden is less than 0.12 μCi (4.4 kBq) of I-125 or 0.04 μCi (1.5 kBq) of I-131.

TABLE 1: ACTIVITY LEVELS ABOVE WHICH BIOASSAY FOR I-125 OR I-131 IS REQUIRED

Types of Operation	Activity Handled in Unsealed Form Making Bioassay Necessary *	
	Volatile or Dispersible *	Bound to Nonvolatile Agent *
Processes in open room or bench, with possible escape of iodine from process vessels	0.1 mCi (3.7 MBq)	1.0 mCi (37 MBq)
Processes with possible escape of iodine carried out within a fume hood of adequate design, face velocity, and performance reliability	1.0 mCi (37 MBq)	10.0 mCi (370 MBq)
Processes carried out within gloveboxes, ordinarily closed, but with possible release of iodine from process and occasional exposure to contaminated box and box leakage	10.0 mCi (370 MBq)	100.0 mCi (3700 MBq)

*Quantities may be considered the cumulative amount in process handled by a worker during a 3-month period; e.g., the total quantity introduced into a chemical or physical process over a 3 month period, or on one or more occasions in that period, by opening stock reagent containers from which radioactive iodine may escape. Quantities in the right-hand column may be used when it can be shown that activity in process is always chemically bound and processed in such a manner that I-125 and I-131 will remain in nonvolatile form and diluted to concentrations less than 0.1 mCi/mg (3.7 MBq/mg) of nonvolatile agent. Capsules (such as gelatin capsules given to patients for diagnostic test), may be considered to contain the radioiodine in non-free form, and bioassay would not be necessary unless a capsule were inadvertently opened (e.g., dropped and crushed). However, certain compounds where radioiodine is normally bound are known to release radioiodine when the material is in process, and the left-hand column may then be applicable. In those laboratories working only with I-125 in radioimmunassay (RIA) kits, the quantities of I-125 are very small and in less volatile forms; thus, bioassay requirements may be judged from the right-hand column. In field operations, where reagent containers are opened outdoors for simple operations such as pouring liquid solutions, the above table does not apply; bioassay should be performed whenever an individual employee handles in open form (e.g., an open bottle or container) more than 50 mCi (1850 MBq) at any one time. Operations involving the routine use of I-125 or I-131 in an open room or bench are discouraged. Whenever practicable, sealed bottles or containers holding more than 0.1 mCi (3.7 MBq) of I-125 or I-131 should be opened within hoods having adequate face velocities of 150 lf/min.

APPENDIX 3

SELECTED CONSTANTS AND DATA

Included in this appendix is the following information:

1. Gamma radiation levels for selected radionuclides
2. Reference data for selected radioisotopes
3. Penetration ability of beta radiation
4. Mass attenuation coefficients
5. Density of elements and common materials
6. Half-value layer for gamma and x-ray radiations at varying energies for various materials

GAMMA RADIATION LEVELS FOR ONE CURIE OF SOME RADIONUCLIDES

Nuclide	Γ†
Actinium-277	~2.2
Antimony-122	2.4
Antimony-124	9.8
Antimony-125	~2.7
Arsenic-72	10.1
Arsenic-74	4.4
Arsenic-76	2.4
Barium-131	~3.0
Barium-133	~2.4
Barium-140	12.4
Beryllium-7	~0.3
Bromine-82	14.6
Cadmium-115m	~0.2
Calcium-47	5.7
Carbon-11	5.9
Cerium-141	0.35
Cerium-144	~0.4
Cesium-134	8.7
Cesium-137	3.3
Chlorine-38	8.8
Chromium-51	0.16
Cobalt-56	17.6
Cobalt-57	0.9
Cobalt-58	5.5
Cobalt-60	13.2
Copper-64	1.2
Europium-152	5.8
Europium-154	~6.2
Europium-155	~0.3
Gallium-67	~1.1
Gallium-72	11.6

Nuclide	Γ†
Gold-198	2.3
Gold-199	~0.9
Hafnium-175	~2.1
Hafnium-181	~3.1
Idium-144m	~0.2
Iodine-124	7.2
Iodine-125	~0.7
Iodine-126	2.5
Iodine-130	12.2
Iodine-131	2.2
Iodine-132	11.8
Iridium-192	4.8
Iridium-194	1.5
Iron-59	6.4
Krypton-85	~0.04
Lanthanum-140	11.3
Lutecium-177	0.09
Magnesium-28	15.7
Maganese-52	18.6
Maganese-54	4.7
Maganese-56	8.3
Mercury-197	~0.4
Mercury-203	1.3
Molybdenum-99	~1.8
Neodymium-147	0.8
Nickel-65	~3.1
Niobium-95	4.2
Osmium-191	~0.6
Palladium-109	0.03
Platinum-197	~0.5
Potassium-42	1.4

Nuclide	Γ†
Potassium-43	5.6
Radium-226	8.25
Radium-228	~5.1
Rhenium-186	~0.2
Rubidium-86	0.5
Ruthenium-106	1.7
Scandium-46	10.9
Scandium-47	0.56
Selenium-75	2.0
Silver-110m	14.3
Silver-111	~0.2
Sodium-22	12.0
Sodium-24	18.4
Strontium-85	3.0
Tantalum-182	6.8
Tellurium-121	3.30
Tellurium-132	2.2
Thulium-170	0.025
Tin-113	~1.7
Tungsten-185	~0.5
Tungsten-187	3.0
Uranium-234	~0.1
Vanadium-48	15.6
Xenon-133	0.1
Ytterbium-175	0.4
Yttrium-88	14.1
Yttrium-91	0.01
Zine-65	2.7
Zirconium-95	4.1

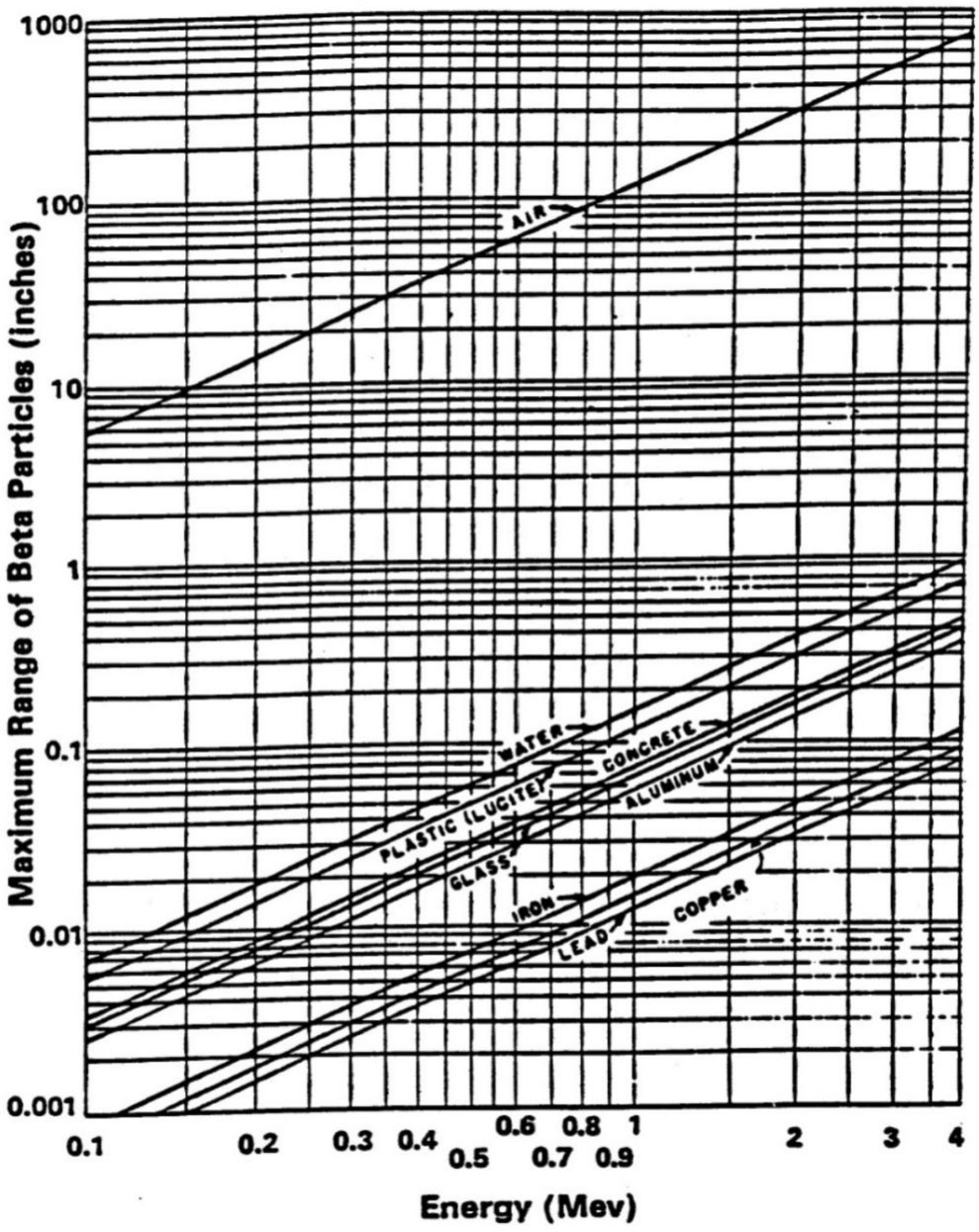
† Γ = R-cm²/hr-mCi or Γ/10 = R/hr at 1 m/Ci = mR/hr at 1 m/mCi

REFERENCE DATA FOR SELECTED RADIOISOTOPES

NUCLIDE	HALF LIVE	BETA MAX ENERGY MeV	BETA I %	BETA RANGE IN INCHES AIR	BETA RANGE IN INCHES PLASTIC	GAMMA ENERGY MeV	GAMMA I %	Γ	HVL (cm Pb)	ATTN.COEF. cm²/gm Pb	CRITICAL ORGAN (μCi) ALI-oral ingestion	EFFECTIVE HALF-LIFE (d)
Calcium-45	163 d	0.257	(100)	20	.02	—		—	—	—	Bone' 2.000	17
Carbon-14	5730 y	0.156	(100)	10	.01	—		—	—	—	Whole Body 2,000	10
Cesium-137	30.17 y	1.173	(5.4)	150	.15	0.6616	(89.9)	0.33	0.536	0.114	Whp;e Body 100	11308
Chromium-51	2707 d	—		—	—	0.3201 1.17	(9.8) (99.9)	0.016	0.165	0.369	Lower Large Intestine 40,000	26.6
Cobalt-60	5.27y	0.318	(99.9)	25	.03	1.33	(99.98)	1.32	1.035	0.059	Whole Body 500	9.5
Copper-64	12.71 h	0.578	(37.2)	60	.06	1.346	(0.49)	0.12	1.11	0.055	Whole Body 10,000	0.529
Hydrogen-3	12.33 y	0.0186	(100)	0.5	0.00	—		—	—	—	Whole Body 80,000	10
Iodine-125	60.14 d	—		—	—	0.0355	(6.67)	0.07	0.0029	21.0	Thyroid 40	42
Iodine-131	80.4 d	0.606	(89.4)	60	.06	0.364 0.636	(81.2) (7.27)	0.22	0.178	0.342	Thyroid 30	7.6
Potassium-42	12.36 h	3.521 1.996	(81.2) (17.5)	600 300	0.6 0.3	1.524	(17.9)	0.14	1.174	0.052	Whole Body 5,000	*
Phosphorous-32	14.28 d	1.71	(100)	250	.25	—	—		—	—	Bone 600	13.5
Sodium-22	2.60 y	$0.548^{\beta+}$	(89.8)	55	0.05	1.274	(99.9)	1.2	1.00	0.061	Whole Body 400	11
Sulfur-35	87.4 d	0.1675	(100)	11	.01	—	—	—	—	—	Whole Body 10,000	44.3
Zinc-65	243.9 d	$0.329^{\beta+}$	(1.5)	30	.03	1.115	(50.8)	0.27	0.925	0.066	Whole Body 400	193.2

I = Intensity D = Days h = hours y = years Γ = Roentgens per hour at one meter per Curie * Data not available

PENETRATION ABILITY OF BETA RADIATION



Maximum Range of Beta Particles vs. Energy.
(The maximum range of beta particles as a function of energy in the various materials indicated. From FRI Report No. 361, *The Industrial Uses of Radioactive Fission Products*, with permission Of the Stanford Research Institute and the U.S. Atomic Energy Commission.)

Mass Attenuation Coefficients (mc^2/g)[a]
(Calculated Using XCOM: Berger and Hubbell 1987)

Photon Energy (MeV)	H	Be	B	C	N	O	Na	Mg	Al	Si
1.0E-03	6.87	604	1230	2210	3310	4590	652	920	1180	1570
1.5E-03	1.85	179	376	699	1080	1550	3190	4000	400	533
2.0E-03	0.812	74.3	159	302	476	694	1520	1930	2260	2770
3.0E-03	0.396	21	46.3	89.7	145	216	506	657	787	977
4.0E-03	0.342	8.45	18.9	37.3	61	92.4	225	296	359	451
5.0E-03	0.339	4.18	9.43	18.8	31	47.3	118	157	192	244
6.0E-03	0.344	2.36	5.33	10.7	17.7	27.2	69.5	92.9	114	146
8.0E-03	0.355	0.998	2.19	4.37	7.29	11.3	29.6	39.9	49.6	63.9
1.0E-02	0.361	0.549	1.13	2.21	3.68	5.7	15.1	20.5	25.7	33.3
1.5E-02	0.365	0.251	0.409	0.71	1.12	1.69	4.45	6.07	7.64	9.98
2.0E-02	0.363	0.19	0.254	0.377	0.537	0.766	1.9	2.58	3.24	4.23
3.0E-02	0.354	0.162	0.182	0.223	0.264	0.325	0.634	0.83	1.02	1.31
3.0E-02	0.344	0.153	0.165	0.187	0.203	0.226	0.344	0.425	0.5	0.622
4.0E-02	0.334	0.148	0.157	0.173	0.181	0.192	0.244	0.287	0.321	0.385
6.0E-02	0.325	0.144	0.151	0.166	0.169	0.175	0.201	0.226	0.244	0.281
8.0E-02	0.309	0.137	0.143	0.155	0.157	0.159	0.164	0.177	0.182	0.2
1.0E-01	0.294	0.131	0.136	0.148	0.148	0.149	0.148	0.157	0.157	0.168
1.5E-01	0.265	0.118	0.123	0.133	0.133	0.133	0.129	0.134	0.132	0.138
2.0E-01	0.243	0.108	0.113	0.122	0.122	0.122	0.117	0.121	0.119	0.123
3.0E-01	0.211	0.0944	0.0983	0.106	0.106	0.106	0.102	0.105	0.103	0.106
4.0E-01	0.189	0.0846	0.0882	0.0952	0.0953	0.0953	0.0912	0.0941	0.0919	0.0951
5.0E-01	0.173	0.0773	0.0806	0.087	0.087	0.087	0.0833	0.086	0.0839	0.0868
6.0E-01	0.16	0.0715	0.0745	0.0805	0.0805	0.0805	0.0771	0.075	0.0776	0.0803
8.0E-01	0.14	0.0628	0.0655	0.0707	0.0707	0.0708	0.0677	0.0699	0.0682	0.0706
1.0E+00	0.126	0.0565	0.0589	0.0636	0.0636	0.0637	0.0609	0.0628	0.0613	0.0634
1.5E+00	0.103	0.046	0.0479	0.0518	0.0518	0.0518	0.0496	0.0212	0.05	0.0518
2.0E+00	0.0877	0.0394	0.0411	0.0444	0.0455	0.0446	0.0428	0.0442	0.0432	0.0448
3.0E+00	0.0692	0.0314	0.0328	0.0356	0.0358	0.036	0.0349	0.0361	0.0354	0.0368
4.0E+00	0.0581	0.0266	0.028	0.0305	0.0307	0.031	0.0304	0.0316	0.0311	0.0324
5.0E+00	0.0505	0.0235	0.0248	0.0271	0.0274	0.0278	0.0275	0.0287	0.0284	0.0297
6.0E+00	0.045	0.0212	0.0225	00247	0.0251	0.0255	0.0256	0.0268	0.0265	0.0279
8.0E+00	0.0375	0.0182	0.0194	0.0215	0.0221	0.0226	0.0232	0.0244	0.0244	0.0257
1.0E+01	0.0325	0.0163	0.0176	0.0196	0.0202	0.0209	0.0218	0.0231	0.0232	0.0246
2.0E+01	0.0215	0.0123	0.0137	0.0158	0.0167	0.0177	0.0197	0.0213	0.0217	0.0234
3.0E+01	0.0175	0.011	0.0125	0.0147	0.0159	0.0171	0.0196	0.0214	0.022	0.0238
4.0E+01	0.0154	0.0105	0.0121	0.0144	0.0157	0.017	0.0199	0.0218	0.0225	0.0245
5.0E+01	0.0142	0.0102	0.0119	0.0143	0.0157	0.0171	0.0203	0.0233	0.0231	0.0252
6.0E+01	0.0134	0.01	0.0118	0.0143	0.0157	0.0172	0.0207	0.0227	0.0236	0.0258
8.0E+01	0.0125	0.00994	0.0118	0.0144	0.016	0.0176	0.0213	0.0236	0.0245	0.0269
1.0E+02	0.0119	0.00994	0.0119	0.0146	0.0163	0.0179	0.0219	0.0242	0.0252	0.0276
1.0E+03	0.0116	0.0112	0.0137	0.017	0.0192	0.0213	0.0262	0.029	0.0303	0.0334
1.0E+04	0.0122	0.0117	0.0144	0.0179	0.0201	0.0222	0.0273	0.0303	0.0317	0.0349
1.0E+05	0.0124	0.0118	0.0146	0.018	0.0202	0.0224	0.0275	0.0305	0.0319	0.0352

[a] Without coherent scattering. Used in gamma ray transport calculations (for discussion, see Jaeger et all. 1967, p. 197).

Mass Attenuation Coefficients (mc^2/g)[a]
(Calculated Using XCOM: Berger and Hubbell 1987)

Photon Energy (MeV)	P	S	Ar	K	Ca	Fe	Cu	Ge	Mo	Sn
1.0E-03	1910	2430	3180	4050	4860	9080	10600	1890	49404	8150
1.5E-03	652	832	1100	1420	1710	3400	4410	5470	1920	3290
2.0E-03	300	383	509	656	797	1620	2150	2710	953	1660
3.0E-03	1120	1340	168	217	265	554	745	957	2010	608
4.0E-03	523	632	755	924	120	254	344	446	966	934
5.0E-03	285	347	421	517	601	137	187	244	541	842
6.0E-03	172	210	258	319	372	82.8	113	148	334	525
8.0E-03	75.8	93.7	117	146	172	304	50.7	66.8	154	247
1.0E-02	39.7	49.4	62.4	78.2	92.5	169	215	35.8	83.5	136
1.5E-02	12	15.1	19.4	24.5	29.2	56.3	73.2	90.5	27.1	44.9
2.0E-02	5.1	6.43	8.33	10.6	12.7	25.2	33.2	41.5	78.6	20.2
3.0E-02	1.57	1.96	2.53	3.22	3.87	7.89	10.6	13.5	27.5	40.5
3.0E-02	0.725	0.892	1.13	1.42	1.7	3.45	4.65	5.97	12.6	19
4.0E-02	0.434	0.519	0.63	0.786	0.929	1.83	2.47	3.17	6.79	10.4
6.0E-02	0.307	0.357	0.415	0.508	0.591	1.11	1.48	1.9	4.09	6.33
8.0E-02	0.207	0.23	0.245	0.289	0.325	0.539	0.697	0.875	1.85	2.88
1.0E-01	0.17	0.183	0.184	0.21	0.23	0.334	0.414	0.505	1.02	1.58
1.5E-01	0.136	0.142	0.133	0.147	0.155	0.179	0.201	0.225	0.383	0.561
2.0E-01	0.121	0.125	0.115	0.125	0.13	0.136	0.144	0.152	0.22	0.298
3.0E-01	0.103	0.107	0.097	0.105	0.108	0.105	0.106	0.107	0.128	0.151
4.0E-01	0.0925	0.0954	0.0863	0.0933	0.096	0.0913	0.0909	0.0896	0.0989	0.108
5.0E-01	0.0844	0.087	0.0787	0.0849	0.0873	0.0824	0.0816	0.0797	0.0847	0.0888
6.0E-01	0.078	0.0804	0.0727	0.0785	0.0806	0.0758	0.0748	0.0729	0.0759	0.0777
8.0E-01	0.0686	0.0707	0.0638	0.0689	0.0707	0.0663	0.0652	0.0633	0.0647	0.0647
1.0E+00	0.0616	0.0635	0.0574	0.0619	0.0636	0.0595	0.0585	0.0567	0.0574	0.0567
1.5E+00	0.0503	0.0518	0.0468	0.0506	0.0519	0.0486	0.0478	0.0463	0.0467	0.0458
2.0E+00	0.0435	0.0449	0.0407	0.0439	0.0452	0.0425	0.0419	0.0407	0.0414	0.0408
3.0E+00	0.0359	0.0371	0.0338	0.0366	0.0378	0.0362	0.0359	0.0352	0.0366	0.0367
4.0E+00	0.0317	0.0329	0.0302	0.0328	0.0339	0.0331	0.0331	0.0327	0.0349	0.0355
5.0E+00	0.0291	0.0304	0.028	0.0305	0.0317	0.0314	0.0317	0.0316	0.0344	0.0354
6.0E+00	0.0275	0.0287	0.0267	0.0291	0.0303	0.0306	0.0311	0.0311	0.0344	0.0358
8.0E+00	0.0255	0.0268	0.0252	0.0277	0.0289	0.0299	0.0307	0.031	0.0352	0.0372
1.0E+01	0.0245	0.0259	0.0245	0.027	0.0284	0.0299	0.031	0.0316	0.0365	0.0389
2.0E+01	0.0236	0.0253	0.0245	0.0274	0.029	0.0322	0.0341	0.0353	0.0426	0.0466
3.0E+01	0.0243	0.0261	0.0256	0.0287	0.0306	0.0347	0.0369	0.0385	0.0472	0.052
4.0E+01	0.0251	0.0271	0.0267	0.03	0032	0.0367	0.0392	0.041	0.0506	0.056
5.0E+01	0.0258	0.0279	0.0276	0.0311	0.0332	0.0383	0.041	0.043	0.0533	0.059
6.0E+01	0.0265	0.0287	0.0284	0.032	0.0342	0.0396	0.0425	0.0446	0.0554	0.0614
8.0E+01	0.0276	0.0299	0.0297	00.0335	0.0359	0.0417	0.0449	0.0471	0.0586	0.0651
1.0E+02	0.0284	0.0308	0.0307	0.0346	0.0371	0.0433	0.0466	0.0489	0.061	0.0678
1.0E+03	0.0345	0.0375	0.0375	0.0424	0.0456	0.0533	0.0572	0.06	0.0751	0.0835
1.0E+04	0.036	0.0392	0.0391	0.0443	0.0477	0.0555	0.0595	0.0624	0.0781	0.0868
1.0E+05	0.0362	0.0395	0.0394	0.0446	0.048	0.05590	0.0598	0.0628	0.0786	0.0874

[a] Without coherent scattering. Used in gamma ray transport calculations (for discussion, see Jaeger et all. 1967, p. 197).

Mass Attenuation Coefficients (mc^2/g)[a]
(Calculated Using XCOM: Berger and Hubbell 1987)

Photon Energy (MeV)	I	W	Pb	Air	Bakelite	Boone	Concrete	H_2O	Lucite
1.0E-03	9090	3670	5200	3610	2470	3440	3440	4080	2790
1.5E-03	3910	1630	2340	1190	795	1160	1230	1370	914
2.0E-03	1990	3910	1270	527	347	524	1450	616	403
3.0E-03	736	1890	1950	162	105	239	497	192	123
4.0E-03	355	948	1240	77.3	43.9	106	241	82.1	51.9
5.0E-03	838	546	722	39.8	22.2	135	173	42	26.4
6.0E-03	613	345	460	23	12.7	81.9	104	24.2	15.1
8.0E-03	289	165	223	9.64	5.23	36.8	49.5	10.1	6.25
1.0E-02	160	92.5	126	4.91	2.65	19.6	26.1	5.1	3.18
1.5E-02	53.3	136	108	1.49	0.842	6.12	8.02	1.54	0.994
2.0E-02	24.1	63.7	84	0.692	0.436	2.7	3.47	0.721	0.5
3.0E-02	7.82	21.5	28.9	0.308	0.244	0.91	1.12	0.329	0.266
3.0E-02	21.6	9.87	13.4	0.22	0.201	0.476	0.552	0.24	0.212
4.0E-02	12	5.39	7.39	0.189	0.184	0.322	0.353	0.208	0.192
6.0E-02	7.32	3.29	4.53	0.174	0.175	0.253	0.266	0.192	0.181
8.0E-02	3.35	7.54	2.11	0.158	0.163	0.195	0.195	0.175	0.169
1.0E-01	1.83	4.26	5.34	0.149	0.155	0.171	0.167	0.165	0.16
1.5E-01	0.646	1.49	1.91	0.133	0.139	0.145	0.138	0.148	0.144
2.0E-01	0.335	0.732	0.936	0.122	0.128	0.131	0.124	0.136	0.132
3.0E-01	0.163	0.299	0.373	0.106	0.111	0.113	0.107	0.118	0.115
4.0E-01	0.113	0.178	0.215	0.0951	0.0997	0.101	0.0955	0.106	0.103
5.0E-01	0.0916	0.128	0.15	0.0869	0.0911	0.0924	0.0872	0.0966	0.0939
6.0E-01	0.0794	0.103	0.117	0.0804	0.0843	0.0854	0.0806	0.0894	0.0869
8.0E-01	0.0653	0.0768	0.0841	0.0706	0.0741	0.075	0.0708	0.0786	0.0763
1.0E+00	0.057	0.0637	0.068	0.0635	0.0666	0.0675	0.0637	0.0707	0.0687
1.5E+00	0.0459	0.0489	0.0509	0.0517	0.0542	0.055	0.0519	0.0575	0.0559
2.0E+00	0.0409	0.0437	0.0453	0.0445	0.0465	0.0473	0.0448	0.0494	0.048
3.0E+00	0.037	0.0405	0.042	0.0358	0.0373	0.0383	0.0365	0.0397	0.0384
4.0E+00	0.036	0.0402	0.0418	0.0308	0.0319	0.0331	0.0319	0.034	0.0329
5.0E+00	0.036	0.0409	0.0426	0.0275	0.0283	0.0297	0.0289	0.0303	0.0292
6.0E+00	0.0365	0.042	0.0438	0.0252	0.0258	0.0274	0.027	0.0277	0.0266
8.0E+00	0.0381	0.0447	0.0467	0.0223	0.0225	0.0244	0.0245	0.0243	0.0232
1.0E+01	0.04	0.0474	0.0497	0.0204	0.0204	0.0226	0.0231	0.0222	0.0211
2.0E+01	0.0482	0.0589	0.062	0.0171	0.0163	0.0195	0.021	0.0181	0.0168
3.0E+01	0.054	0.0665	0.0702	0.0163	0.0152	0.0189	0.021	0.0171	0.0157
4.0E+01	0.0581	0.072	0.0761	0.0161	0.0148	0.0189	0.0214	0.0168	0.0153
5.0E+01	0.0613	0.0762	0.0806	0.0161	0.0147	0.0191	0.0218	0.0167	0.0151
6.0E+01	0.0638	0.0795	0.0841	0.0163	0.0147	0.0193	0.0223	0.0168	0.0151
8.0E+01	0.0676	0.0844	0.0893	0.0165	0.0148	0.0197	0.023	0.017	0.0153
1.0E+02	0.0704	0.088	0.0931	0.0168	0.015	0.0202	0.0236	0.0173	0.0154
1.0E+03	0.0869	0.108	0.115	0.0199	0.0174	0.024	0.0284	0.0202	0.0179
1.0E+04	0.0904	0.112	0.119	0.0208	0.0183	0.0251	0.0297	0.0211	0.0188
1.0E+05	0.0909	0.113	0.12	0.021	0.0184	0.0253	0.0299	0.0213	0.019

[a] Without coherent scattering. Used in gamma ray transport calculations (for discussion, see Jaeger et all. 1967, p. 197).

Mass Attenuation Coefficients (mc^2/g)[a]
(Calculated Using XCOM: Berger and Hubbell 1987)

Photon Energy (MeV)	Marble ($CaCO_3$)	Muscle	NaI	Nylon	Poly-ethylene	Poly-Styrene	Pyrex	SiO_2	Cellulos (Wood)
1.0E-03	4410	3760	7790	2470	1890	2640	3160	3180	3250
1.5E-03	1510	1270	3800	798	599	857	1150	1070	1070
2.0E-03	688	566	1920	349	258	376	1500	1670	476
3.0E-03	221	184	700	106	76.9	114	511	572	146
4.0E-03	96.7	81.3	335	44.4	32	48.1	234	260	62.2
5.0E-03	266	41.9	728	22.5	16.1	24.4	125	139	31.7
6.0E-03	163	24.3	530	12.9	9.17	13.9	74.3	82.7	18.2
8.0E-03	74.6	10.2	249	5.31	3.79	5.76	32.2	35.9	7.53
1.0E-02	40	5.18	138	2.7	1.95	2.92	16.6	18.6	3.82
1.5E-02	12.6	1.57	45.8	0.864	0.66	0.919	4.99	5.56	1.17
2.0E-02	5.49	0.738	20.7	0.451	0.375	0.468	2.14	2.39	0.568
3.0E-02	1.73	0.334	6.71	0.255	0.241	0.254	0.718	0.786	0.281
3.0E-02	0.812	0.241	18.3	0.21	0.21	0.206	0.384	0.411	0.216
4.0E-02	0.485	0.208	10.2	0.193	0.197	0.187	0.268	0.282	0.192
6.0E-02	0.341	0.191	6.23	0.183	0.188	0.177	0.217	0.225	0.18
8.0E-02	0.225	0.174	2.86	0.171	0.177	0.165	0.174	0.178	0.167
1.0E-01	0.181	0.164	1.58	0.162	0.169	0.157	0.156	0.158	0.157
1.5E-01	0.142	0.147	0.566	0.146	0.152	0.141	0.134	0.135	0.141
2.0E-01	0.125	0.134	0.302	0.134	0.139	0.129	0.122	0.123	0.13
3.0E-01	0.107	0.117	0.153	0.117	0.121	0.112	0.106	0.106	0.113
4.0E-01	0.0955	0.105	0.11	0.104	0.109	0.101	0.0948	0.0952	0.101
5.0E-01	0.0871	0.0958	0.0904	0.0954	0.0993	0.0921	0.0865	0.0869	0.0924
6.0E-01	0.0806	0.0886	0.079	0.0883	0.0919	0.0852	0.0801	0.0804	0.0854
8.0E-01	0.0708	0.0779	0.0657	0.0775	0.0807	0.0748	0.0704	0.0707	0.0751
1.0E+00	0.0636	0.07	0.0576	0.0697	0.0726	0.0673	0.0633	0.0636	0.0675
1.5E+00	0.0519	0.057	0.0464	0.0568	0.0591	0.0548	0.0516	0.0518	0.055
2.0E+00	0.0448	0.049	0.0412	0.0487	0.0506	0.047	0.0444	0.0447	0.0472
3.0E+00	0.0366	0.0393	0.0367	0.039	0.0404	0.0377	0.0361	0.0363	0.0379
4.0E+00	0.0321	0.0337	0.0351	0.0333	0.0344	0.0322	0.0314	0.0316	0.0324
5.0E+00	0.0293	0.03	0.0347	0.0295	0.0304	0.0286	0.0284	0.0287	0.0289
6.0E+00	0.0274	0.0274	0.0348	0.0268	0.0276	0.0261	0.0263	0.0266	0.0264
8.0E+00	0.025	0.024	0.0358	0.0233	0.0238	0.0227	0.0237	0.0241	0.0231
1.0E+01	0.0237	0.0219	0.0372	0.0211	0.0215	0.0207	0.0222	0.0226	0.021
2.0E+01	0.022	0.0179	0.0439	0.0167	0.0166	0.0166	0.0198	0.0204	0.0171
3.0E+01	0.0222	0.0169	0.0487	0.0155	0.0151	0.0154	0.0196	0.0202	0.016
4.0E+01	0.0227	0.0165	0.0522	0.015	0.0145	0.015	0.0198	0.0205	0.0157
5.0E+01	0.0232	0.0165	0.055	0.0148	0.0142	0.0149	0.0201	0.0209	0.0156
6.0E+01	0.0237	0.0165	0.0572	0.0148	0.0141	0.0149	0.0205	0.0212	0.0157
8.0E+01	0.0245	0.0168	0.0605	0.0148	0.0141	0.015	0.0211	0.0219	0.0158
1.0E+02	0.0252	0.017	0.063	0.015	0.0142	0.0152	0.0216	0.0225	0.0161
1.0E+03	0.0305	0.0199	0.0776	0.0174	0.0162	0.0177	0.0259	0.027	0.0188
1.0E+04	0.0319	0.0208	0.0807	0.0182	0.017	0.0186	0.027	0.0282	0.0197
1.0E+05	0.0321	0.0209	0.0812	0.0183	0.0172	0.0187	0.0272	0.0284	0.0198

[a] Without coherent scattering. Used in gamma ray transport calculations (for discussion, see Jaeger et all. 1967, p. 197).

DENSITY OF ELEMENTS AND COMMON MATERIALS

Element	Atomic #	Atomic Weight	MIP*	Density	Element	Atomic #	Atomic Weight	MIP*	Density
H	1	1.00797	18.0	0.0586	I	53	126.9044		4.93
Hc	2	4.0026	40.0	0.126	Xe	54	131.30	757.52	3.52
Li	3	6.939	39.032	0.534	Cs	55	132.905		1.873
Bc	4	9.0122	56.0	1.8	Ba	56	137.34		3.5
B	5	10.811		2.34	La	57	138.91		6.155
C	6	12.01115	79.0	2.25	Ce	58	140.12		3.92
N	7	14.0067	92.0	0.808	Pr	59	140.907		6.5
O	8	15.9994	105.0	1.14	Nd	60	144.24		6.95
F	9	18.9984		1.11	Pm	61	147		
Ne	10	20.183	130.016	1.2	Sm	62	150.35		7.8
Na	11	22.9898		0.971	Eu	63	151.96		5.24
Mg	12	24.312	156.4	1.74	Gd	64	157.25		
Al	13	26.9815	163	2.699	Tb	65	158.924		
Sf	14	28.086		2.42	Dy	66	162.50		8.56
P	15	30.9738		1.82	Ho	67	164.930		
S	16	32.064		2.07	Er	68	167.26		4.77
Cl	17	35.453		1.56	Tm	69	168.934		
Ar	18	39.948	240	1.40	Yb	70	173.04		
K	19	39.102		0.87	Lu	71	174.97		
Ca	20	40.08	200	1.55	Hf	72	178.49		13.3
Sc	21	44.956		3.02	Ta	73	180.948	720	16.6
Ti	22	47.90	225	4.5	W	74	183.85	740	19.3
V	23	50.942	254	5.96	Re	75	186.2		20.53
Cr	24	51.996		7.1	Os	76	190.2		22.48
Mn	25	54.9380		7.2	Ir	77	192.2	760	22.42
Fe	26	55.847	273	7.86	Pt	78	195.09	777	21.37
Co	27	58.9332	298	8.9	Au	79	196.967	786	19.32
Ni	28	58.71	312	8.9	Ug	80	200.59		13.546
Cu	29	63.54	322	8.94	Tl	81	204.37		11.85
Zn	30	65.37	331	7.14	Pb	82	207.19	818	11.35
Ga	31	69.72		5.91	Br	83	209.980	826	9.747
Ge	32	72.59		5.36	Po	84	210		
As	33	74.9216		5.73	At	85	210		
Se	34	78.96		4.8	Rn	86	222		9.73
Br	35	79.909		3.12	Fr	87	223		
Kr	36	83.80	493.68	2.6	Ra	88	226		
Rb	37	85.47		1.53	Ac	89	227		
Sr	38	87.62		2.54	Th	90	232.038		11.3
Y	39	88.905		5.51	Pa	91	231		
Zr	40	91.22		6.4	U	92	238.03	908	18.68
Nb	41	92.906	410	8.4	Np	93			
Mo	42	95.94	420	10.2	Pu	94			
Tc	43	99			Am	95			
Ru	44	101.07		12.2	Cm	96			
Rh	45	102.905	450	12.5	Bk	97			
Pd	46	106.4	460	12.10	Cf	98			
Ag	47	107.870	485	10.50	Rs	99			
Cd	48	112.40	468	8.65	Fm	100			
In	49	114.82	490	7.28	Md	101			
Sn	50	118.69	500	7.31	No	102			
Sb	51	121.75		6.691	Lw	103			
Te	52	127.6		6.24	Ku	104			

* Mean Ionization Potential

Density of Common Materials

(After Handbook of Chemistry and Physics, 1991: Trout et al. 1961: IAEA Tech. Rept Series 188, 1997; and Hubbell, NBS 29, 1969)

Material	Density Range (g/cm^3)	Normal Density (g/cm^3)
Air	- -	0.0012929
Aluminum	- -	2.70
Asbestos	2.0 - 2.8	- -
Asphalt	1.1 - 1.5	- -
Bone	1.7 - 2.0	- -
Bakelite (typical)	1.20 - 1.70	- -
Brick (soft)	1.4 - 1.9	1.65
Brick (hard)	1.8 - 2.3	2.05
Cement	2.7 - 3.0	- -
Clay	1.8 - 2.6	- -
Concretes:		
Ordinary (silicacious)	2.2 - 2.4	2.35
Barite (barytes, nat. $BaSo_4$)	3.0 - 3.8	- -
Limonite (Goethite, hyd. Fe_2O_3)	2.6 - 3.7	- -
Ilmenite (nat. $FeTiO_3$)	2.9 - 3.9	- -
Magnetite (nat. Fe_3O_4)	2.9 - 4.0	- -
Iron (shot, punchings, etc)	4.0 - 6.0	- -
Copper	- -	8.96
Earth	1.5 - 1.9	1.7
Ebonite	- -	1.15
Gelatin	- -	1.27
Glass (common)	2.4 - 2.8	- -
Glass (flint)	2.9 - 5.9	- -
Granitie	2.60 - 2.76	2.65
Graphite	2.30 - 2.72	- -
Gypsum	2.31 - 2.33	- -
Lead	- -	11.35
Limestone	1.87 - 2.76	2.46
Linoleum	- -	1.18
Lucite	- -	1.19
Marble	2.47 - 2.86	2.7
Paraffin	0.87 - 0.91	- -
Plaster, sand	- -	1.54
Polyethylene	- -	0.92
Polystyrene	1.05 - 1.07	- -
Pressed wood: Pulp board	- -	0.19
Pyrex glass (Corning No. 7740)	- -	2.23
Sandstone	- -	1.90
Slate	2.6 - 3.3	- -
Steel	- -	7.8
Tile	1.6 - 2.5	- -
Water	- -	1.000
Water (heavy)	- -	1.105
Wood:		
Oak	0.60 - 0.90	- -
White Pine	0.35 - 0.50	- -
Yellow Pine	0.37 - 0.60	-

Half-Value layer (in cm) for gamma and X-ray radiations at varying energies for various materials

Energy	Lead Density 11.35 g/cm^3		Iron Density 7.86 g/cm^3		Aluminum Denisty 2.82 g/cm^3		Water Density 1 g/cm		Air Density 0.0012929 g/cm^3		Stone Concrete Density 2.35 g/cm^3	
MeV	μ	HVL	μ	HVL	μ	HVL	μ	HVL	μ	HVL	μ	HVL
0.3	4.32	0.16	0.864	0.845	0.282	2.457	0.119	5.823	$.135 \times 10^{-3}$	5.133×10^{3}	0.251	2.76
0.5	1.75	0.396	0.652	1.062	0.236	2.936	0.092	7.532	$.111 \times 10^{-3}$	6.243×10^{3}	0.204	3.39
1	0.85	0.816	0.471	1.471	0.164	4.225	0.071	9.76	$.082 \times 10^{-3}$	8.451×10^{3}	0.149	4.65
1.5	0.59	1.174	0.378	1.833	0.137	5.058	0.057	12.157	$.067 \times 10^{-3}$	10.434×10^{3}	0.121	5.72
2	0.51	1.358	0.334	2.074	0.112	6.187	0.05	13.86	$.056 \times 10^{-3}$	12.375×10^{3}	0.104	6.66
2.5	0.48	1.443	0.302	2.294	0.102	6.794	0.044	15.75	$.050 \times 10^{-3}$	13.860×10^{3}		
3	0.47	1.474	0.279	2.843	0.094	7.372	0.039	17.769	$.046 \times 10^{-3}$	15.065×10^{3}	0.085	8.15
3.5	0.46	1.506[a]	0.268	2.585	0.087	7.965	0.036	19.25	$.042 \times 10^{-3}$	16.500×10^{3}		
4			0.251	2.76	0.083	8.349	0.034	20.382	$.039 \times 10^{-3}$	17.769×10^{3}	0.074	9.36
5			0.243	2.851	0.072	9.625	0.03	23.1	$.034 \times 10^{-3}$	20.382×10^{3}	0.067	10.34
10			0.235	2.948[a]	0.059	11.745	0.023	30.13	$.028 \times 10^{-3}$	24.750×10^{3}	0.05	13.86
20					0.057	12.157[a]	0.018	38.5	$.021 \times 10^{-3}$	33.000×10^{3}	0.049	14.14[a]
30							0.017	40.764	$.020 \times 10^{-3}$	33.640×10^{3}		
40							0.016	43.312[a]	$.020 \times 10^{-3}$	34.650×10^{3}		

[a] Value beyond which the half value layer will not increase regardless of increase in energy.