

DOSE LIMITS AND DOSIMETRY

Voss Associates

DOSE LIMITS

10CFR835

Radiological Workers

whole body (internal + external) (TEDE)	5 rem/year
lens of the eye (external)	15 rem/year
skin and extremities (external shallow dose)	50 rem/year
any organ or tissue (other than lens of eye) (internal + external)	50 rem/year
embryo/fetus (internal + external)	0.5 rem/ gestation period

DOSE LIMITS

10CFR20

Radiological Workers

whole body total effective dose equivalent	5 rem/year
lens of the eye	15 rem/year
skin and extremities (external shallow dose)	50 rem/year
any organ or tissue (other than lens of eye) (sume of DDE + CDE)	50 rem/year

DOSE LIMITS

10CFR835

Non-Radiological Worker

General employee: whole body (internal + external) (TEDE)	0.1 rem/year
Member of the public: whole body (internal + external) (TEDE)	0.1 rem/year

TYPES OF DOSIMETRY

Thermoluminescence (TL) is the ability of some materials to convert the energy from radiation to a radiation of a different wavelength, normally in the visible light range.

Ionizing radiation transfers energy to the electrons of the phosphor atoms in the TLD. These excited electrons detach from the phosphor atoms in the TLD and eventually become trapped. When the TLD is heated, these excited electrons escape from the traps and return to the ground state going from a higher energy state to a lower state. In going to a lower energy state, these excited electrons give up stored energy in the form of light photons.

TLD READER

Heating of the TL material causes the trapped electrons to return to the valence band. When this happens, energy is emitted in the form of visible light. The light output is detected and measured by a photomultiplier tube and a dose equivalent is then calculated. A typical basic TLD reader contains the following components:

1. Heater - raises the phosphor temperature
2. Photomultiplier Tube - measures the light output
3. Meter/Recorder - display and record data

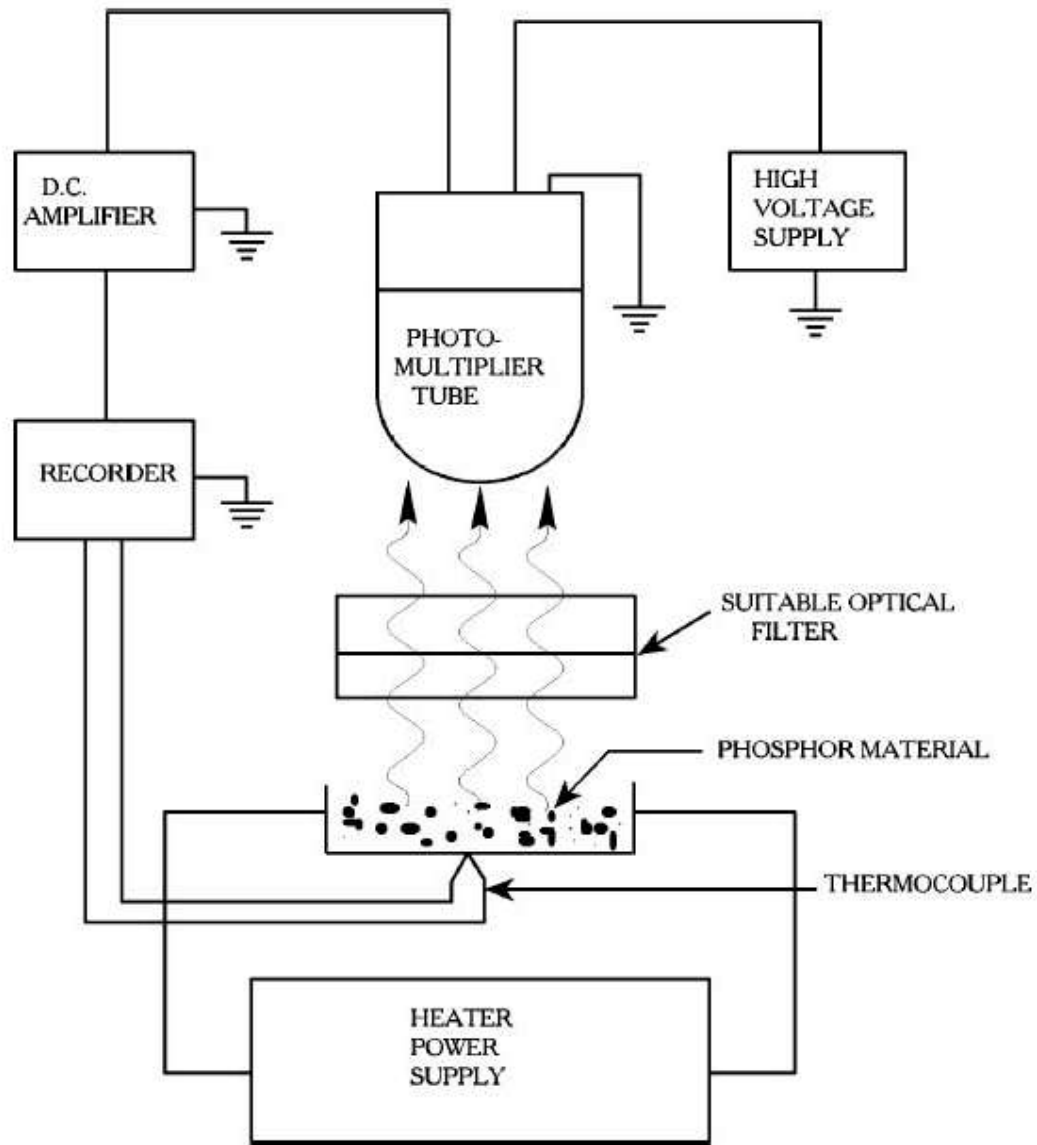


Figure 3 - TLD Reader

A glow curve can be obtained from the heating process.

The light output from TL material is not easily interpreted.

Multiple peaks result as the material is heated and electrons trapped in "shallow" traps are released.

This results in a peak as these traps are emptied.

As heating continues, the electrons in deeper traps are released.

This results in additional peaks.

The area under the curve represents the radiation energy deposited in the TLD.

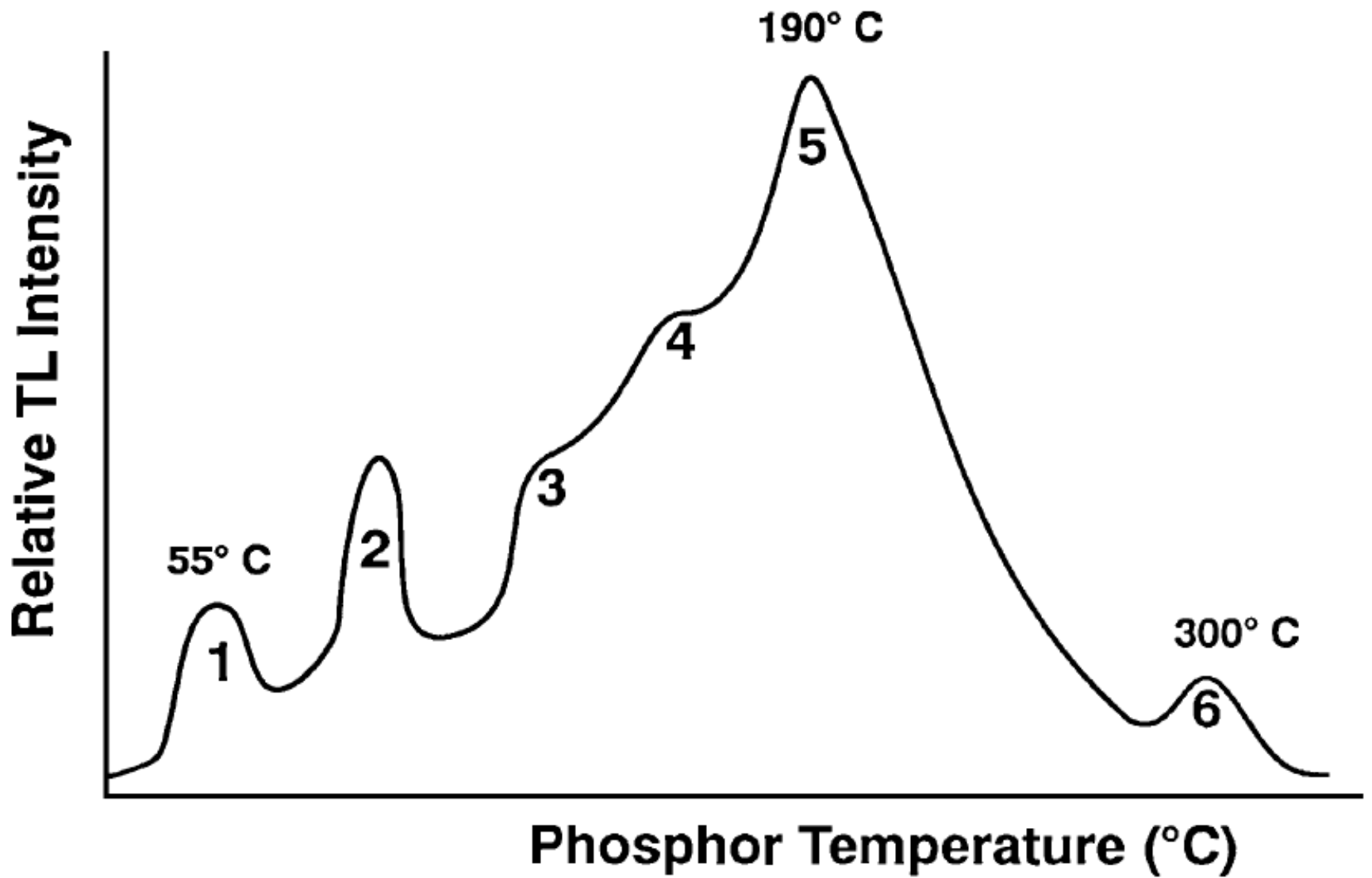


Figure 4 - Glow Curve

ADVANTAGES AND DISADVANTAGES OF TLDs

1. Advantages (as compared to film dosimeter badges) include:

- a. Able to measure a greater range of doses
- b. Doses may be easily obtained
- c. They can be read on site instead of being sent away for developing
- d. Quicker turnaround time for readout
- e. Reusable

2. Disadvantages

- a. Each dose cannot be read out more than once
- b. The readout process effectively "zeroes" the TLD

The TLD materials used in many dosimetry badges are lithium fluoride (LiF) and calcium fluoride (CaF).

Multiple chips are used to provide a separate measurements of neutron dose and photon deep dose and shallow dose.

^7LiF materials are sensitive to beta and photon radiation and ^6LiF is sensitive to neutron radiation.

CaF materials are sensitive to beta and photon radiation.

To measure the “shallow dose” a relatively thicker mylar window is placed on top of one of the CaF chips.

This reduces that chip’s response to shallow dose.

By comparing the two CaF chip responses a shallow dose can be determined.

Calcium Fluoride Dysprosium, Aluminum Oxide, Calcium Sulfate Dysprosium, and Calcium Fluoride Manganese are also used in TLD materials. Each has areas of usefulness.

Track-etch Dosimeter

Track-etch dosimeters are used to measure high energy neutrons (> 5 MeV).

High-energy neutrons knock protons out of hydrogen atoms or other light nuclei in the plastic material.

The recoil protons cause secondary ionization in the plastic material leaving a track of weakened material.

A chemical bath etches the track left by the recoil proton.

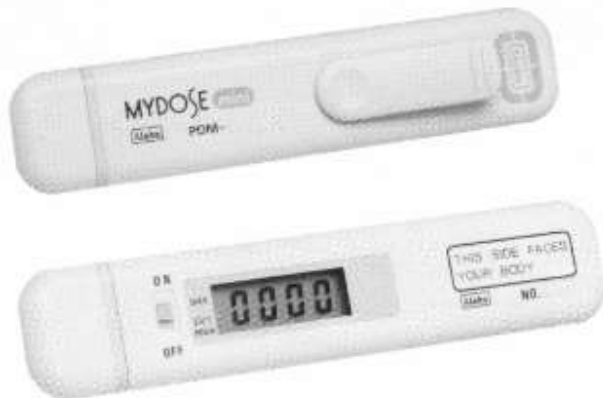
These etched tracks scatter light in the automatic reader.

The amount of scattered light is proportional to the number of tracks, which is proportional to the neutron dose.

Neutron and Photon Electronic Dosimeters

Electronic dosimeters are supplemental dosimeters that provide realtime indication of exposure to radiation and assist in maintaining personnel doses ALARA.

Aloka Dosimeter



Siemens Dosimeter



Internal Dosimetry

Internal dose evaluation programs (including routine bioassay programs) should be conducted for those who, under typical conditions, are likely to receive 0.1 rem (0.001 sievert) or more committed effective dose equivalent from all occupational radionuclide intakes in a year.

BIOASSAY ASSESSMENT METHODS

Bioassay assessment used to determine internal dose contributions relies on calculation of dose to affected portions of the body based on the quantities of radioactive materials in the body.

There are two types of bioassay measurements employed in nuclear industries: in vivo and in vitro.

In vivo bioassay involves counting the living tissue using a radiation detector.

This method is possible only for those radionuclides emitting penetrating radiation, e.g., Co-60 and Cs-137 or bremsstrahlung, e.g., P-32 and Sr-90.

Many radionuclides, Na-22, Fe-59, Co-60, Zn-65, Rb-86, Sr-85, Te-132, I-131, Cs-137, Ba-140, Ce-144, Au-198, U-235, Np-239, and Am-241 emit electromagnetic radiation of sufficient energy to be measured by external counting.

In vitro involves counting an excreted sample, such as urine.

The amount of material present in the body is estimated using the amount of materials present in excretions or secretions from the body.

Samples could include urine, feces, blood, sputum, saliva, hair, and nasal discharges.

Calculation of dose requires knowledge and use of metabolic models which allow sample activity to be related to activity present in the body.

Dosimetry Terms

Absorbed Dose (D): Energy absorbed by matter from ionizing radiation per unit mass of irradiated material at the place of interest in that material. The absorbed dose is expressed in units of rad (or gray) (1 rad = 0.01 gray).

Dose Equivalent (H): The product of the absorbed dose (D)(in rad or gray) in tissue, a quality factor (Q), and all other modifying factors (N). Dose equivalent is expressed in units of rem (or sievert) (1 rem = 0.01 sievert).

Deep Dose Equivalent (DDE): The dose equivalent derived from external radiation at a tissue depth of 1 cm in tissue (1000 mg/cm²).

Shallow Dose Equivalent (SDE): The dose equivalent derived from external radiation at a depth of 0.007 cm in tissue (7 mg/cm^2).

Whole Body: For the purposes of external exposure, head, trunk (including male gonads), arms above and including the elbow, or legs above and including the knee.

Extremity: Hands and arms below the elbow or feet and legs below the knee.

Committed Dose Equivalent (CDE): The dose equivalent calculated to be received by a tissue or organ over a 50-year period after the intake of a radionuclide into the body. It does not include contributions from radiation sources external to the body. Committed Dose Equivalent is expressed in units of rem (or sievert).

Committed effective dose equivalent ($H_{E,50}$)— The sum of the committed dose equivalents to various tissues in the body ($H_{T,50}$), each multiplied by the appropriate weighting factor (W_T) - that is $H_{E,50} = \sum W_T H_{T,50}$.

Committed effective dose equivalent is expressed in units of rem (or sievert).

Weighting factor (WT)—The fraction of the overall health risk, resulting from uniform, whole-body irradiation, attributable to specific tissue (T).

The dose equivalent to tissue (HT) is multiplied by the appropriate weighting factor to obtain the effective dose equivalent contribution from that tissue.

Table 1 - Weighting Factors For Various Tissues

Organs or tissues, T	Weighting factor, W_t
Gonads	0.25
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Thyroid	0.03
Bone surfaces	0.03
Remainder ¹	0.30
Whole body ²	1.00

¹ "Remainder" means the five other organs or tissues with the highest dose (e.g. liver, kidney, spleen, thymus, adrenal, pancreas, stomach, small intestine, and upper large intestine). The weighting factor for each remaining organ or tissue is 0.06.

² For the case of uniform external irradiation of the whole body, a weighting factor (w_T) equal to 1 may be used in determination of the effective dose equivalent.

Total Effective Dose Equivalent (TEDE) -- The sum of the effective dose equivalent (for external exposures) and the Committed Effective Dose Equivalent (for internal exposures).

Annual Limit on Intake (ALI) -- The limit for the amount of radioactive material taken into the body of an adult worker by inhalation or ingestion in a year. ALI is the smaller value of intake of a given radionuclide in a year by the reference man (ICRP Publication 23) that would result in a Committed Effective Dose Equivalent of 5 rems (0.05 sievert) or a Committed Dose Equivalent of 50 rems (0.5 sieverts) to any individual organ or tissue.

Derived Air Concentration (DAC) -- The airborne concentration that equals the ALI divided by the volume breathed by an average worker for a working year of 2000 hours (assuming a breathing volume of 2400m^3).

Bioassay -- The determination of kinds, quantities, or concentrations, and, in some cases, locations of radioactive material in the human body, whether by direct measurement or by analysis, and evaluation of radioactive materials excreted or removed from the human body.

Declared pregnant worker -- A woman who has voluntarily declared to her employer, in writing, her pregnancy for the purpose of being subject to the occupational dose limits to the embryo/fetus in accordance with 10CFR835.

Radiological Protection Standards

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History Of Standards

Because there are still unknowns, the setting of exposure limits involves judgments that cannot be wholly based upon the present body of scientific knowledge.

For this reason, the concept of an "acceptable risk" is used.

The benefits are weighed against the potential damage and then limits are set at some level at which the most benefit to mankind will accrue.

Since all exposure is assumed to involve risk to the individual, exposures should always be kept as low as practicable.

Erythema Dose

Early efforts at control were hampered by a lack of quantitative methods.

There were no units by which one could assess the amount of radiation.

As a result of the use of radiation by doctors in treating patients, a unit called the erythema dose came into use.

This was a highly qualitative unit; defined in terms of the amount of radiation that would produce a well defined reddening of the skin.

Erythema Dose

It varied not only with the type of radiation and the dose rate, but also with the response of different parts of the body.

This lack of a certain value for this unit made protection work more or less of a trial-and-error process.

Around 1914, radiation began to be used in industry -- radium dial-painting and x-rays used for showing flaws in materials.

ICRU, ICRP, AND NCRP

In 1925, at the First International Congress of Radiology, the International Commission on Radiological Units and Measurements (ICRU) was formed.

In 1928, this group adopted the definition of an international unit, the Roentgen.

For the first time measurements throughout the world could be made in terms of the same unit.

Over the years the ICRU has been the main force in defining and adopting units for use on an international basis.

ICRU, ICRP, AND NCRP

At the Second International Congress of Radiology in 1928, the first international body concerned with protection standards was formed.

At first known as the International X-ray and Radium Protection Commission, this group is now called the International Commission on Radiological Protection (ICRP).

This group discusses and reviews basic protection principles, and these recommendations then serve as a guide from which regulations can be drawn up by each country to suit its needs.

ICRU, ICRP, AND NCRP

In 1934, the ICRP made its first recommendation of a tolerance level of exposure: 0.2 R/day.

This limit remained in force until 1950.

Because of World War II, the ICRP did not meet between 1937 and 1950.

This left much of the study of protection standards during this time to the national committees.

ICRU, ICRP, AND NCRP

The National Committee on Radiation Protection and Measurements (NCRP) was formed in the United States in 1929.

The work of this body was coordinated by the National Bureau of Standards.

The early recommendations of the Committee appeared in the National Bureau of Standards Handbooks.

The NCRP recommendations as outlined in Handbooks 20 and 23, served as the basis for protection practices during the days of the Manhattan project.

ICRU, ICRP, AND NCRP

The development of the atomic bomb had a dramatic impact on radiation protection problems.

Before the war, most of the problems concerned rather low energy x-rays.

During WWII, not only were there these to treat, but also other types of radiation with a wide range of energies.

There was a large increase of workers in the radiation field.

ICRU, ICRP, AND NCRP

New units were needed to define the dose contributed by radiation other than x-rays.

Large amounts of waste were produced and methods of disposal were being developed.

With reactors in use, not only the workers, but also others not connected with the work, had to be considered.

ICRU, ICRP, AND NCRP

The NCRP met in 1946 to reorganize.

At this time a number of subcommittees were formed to deal with the new problems more effectively.

This resulted in the publication of a number of handbooks after the war that represented changes and additions to the old recommendations.

The Committee was replaced by a non-profit corporation chartered by Congress in 1964 and is now known as the National Council on Radiation Protection and Measurements.

ICRU, ICRP, AND NCRP

These various NCRP committees develop proposed recommendations on various aspects of radiation protection and radiation measurements, which when approved by the Council, are published as NCRP Reports.

The three organizations, ICRU, ICRP and NCRP, have figured prominently in the development of present day radiation protection practices.

Radiation Exposure Concerns

Initial concerns resulted from medical exposure to external radiation from the use of x-rays for diagnosis and therapy.

World War II introduced considerations about internal exposure and the dose to the general public.

Potential genetic effects of radiation and the impact of long-term exposure at low dose rates emerged.

Data from biological studies seemed to indicate that one could not assume that all effects had a threshold dose.

Radiation Exposure Concerns

Efforts have been directed toward quantifying the risk associated with a certain level of exposure.

The Non-Threshold relationship assumes that any dose carries some risk of producing damage therefore all exposure should be kept at the lowest practical levels.

Factors to be considered:

Information available for the quantification of risks is imperfect.

The assumptions of a risk by an individual presumes a willingness to chance the risk in exchange for some resultant benefit which justifies the risk.

The balancing of risk versus benefit in order to obtain a net benefit is not easily accomplished.

The approach adopted by both the ICRP and the NCRP is to keep exposures ALARA.

Radiation Exposure Concerns

The continuing reviews of biological data have revealed two types of radiation effects.

- Practical threshold dose -- nonstochastic effects.
- No threshold -- stochastic effects.

Nonstochastic effects -- prevented by limiting the dose to the individual to a value below the threshold dose for occurrence of the effects.

Stochastic effects -- limit the probability of occurrence to some level (deemed acceptable) by limiting the radiation exposure.

Federal Policy on Radiation Matters

In 1959, the Federal Radiation Council (FRC) was formed (Public Law 86-373).

- Advised the President concerning radiation matters
- Provided guidance for all Federal agencies in setting standards and in working with the States.
- The recommendations of the FRC were approved in 1960 and formed the basis of the Federal radiation protection guidance.
- The FRC was abolished by Reorganization Plan No. 3 in 1970.

Office of Radiation Programs (ORP) of the EPA took over the activities of the FRC.

In 1981 the EPA drafted proposed revised recommendations in the Federal Register regarding occupational exposure, and solicited comments.

The EPA believes that it is appropriate to adopt the general features of the ICRP approach in radiation protection guidance for use by Federal agencies for occupational exposure.

The revised EPA guidance was approved and issued in January 1987.

Regulating Agencies

In 1954 under the Atomic Energy Act the United States Atomic Energy commission (AEC) was given the responsibility of regulating the atomic energy industry.

- The Act authorized the AEC to set up a licensing program to be augmented by whatever rules or regulations were deemed appropriate.
- The bases for these rules are: to protect the public health and safety, and provide for national defense and security.

In 1974 the Energy Reorganization Act abolished the AEC and established two agencies to perform the functions of the AEC, Nuclear Regulatory Commission (NRC) and the Energy Research and Development Administration (ERDA).

Regulating Agencies

NRC

The NRC is charged with inspection and review in order to assure compliance.

- This function is carried out by NRC personnel (inspectors) at regular intervals.
- Their job is to make the inspections and report their findings.
- In the event that a failure to comply is noted, the licensee is required to correct this.

States have set up their own safety standards with the assistance of the NRC to assure that the state and Commission programs are compatible. These states are referred to as Agreement States.

Regulating Agencies

NRC

The NRC has established radiation protection standards that pertain to many areas of activities.

10CFR20

10CFR30

10CFR34

10CFR35

10CFR70

10CFR71

are the regulations of most interest for NRC-regulated activities.

Regulating Agencies

DOE

In 1977 the U.S. Department of Energy (DOE) replaced ERDA.

The DOE activities relate to energy research and development.

The DOE has issued occupational radiation protection standards that pertain to its own activities as well as to those of its contractors.

- These standards appear in 10 CFR Part 835.
- These standards are based upon the recommendations of the ICRP, NCRP and the guidance of the EPA.

Regulating Agencies

10 CFR 835

The scope of 10CFR835 establishes radiation protection standards, limits and program requirements for protecting individuals from radiation resulting from the conduct of DOE activities.

- Implemented by the Price-Anderson Amendments Act
- Civil penalties may be assessed
- Requires DOE activities be conducted with a written radiation protection program.

Regulating Agencies

Department of Transportation (DOT)

- Safety in the shipment of radioactive substances
- Title 49 Transportation of the CFRs deals with hazardous shipments including radioactive materials.

Other Agencies

- Interstate Commerce Commission,
- Coast Guard,
- Federal Aviation Agency,
- Postal Service,
- International Atomic Energy Agency

CONTAMINATION CONTROL

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Contamination is defined as radioactive material in an unwanted location, e.g., personnel, work areas, etc.

Two types of contamination are possible, fixed and removable (loose).

Fixed contamination is radioactive surface contamination that is not easily transferred to other personnel or equipment through normal contact.

Removable contamination is radioactive surface contamination that is easily transferred to other personnel or equipment through normal contact.

Fixed contamination is measured by use of a direct survey technique using a portable radiation survey instrument.

This technique, commonly referred to as "frisking" or "scanning", indicates the total contamination on a surface apparent to the detector from both fixed and removable contamination.

When non-removable levels are to be recorded, the removable level must be subtracted from the total.

Removable contamination is measured by a transfer test using a suitable sampling material. Common materials used for the monitoring are the standard paper disk smear or cloth smear. The standard technique involves wiping approximately 100 cm² of the surface of interest using moderate pressure. A common sampling practice used to ensure a 100 cm² sample is to wipe a 16 square inch "S" shape on the surface (i.e., 4 inches by 4 inches).

Qualitative, large area wipe surveys may be taken using other materials, such as Masslinn cloth or Kimwipe, to indicate the presence of removable contamination. Large area swipes are commonly used when exact levels of contamination are not required.

Constant (Continuous) Monitoring

There are various types of constant monitoring instruments throughout the facilities to warn personnel of radiation and contamination hazards.

Some instruments are permanently installed, and some instruments are portable to allow movement from place to place as deemed appropriate by the radiological control staff.

Continuous Air Monitor (CAM)

These instruments continuously sample the air for radioactive contamination in specific locations.

The air being sampled is typically drawn through a moving particulate filter which is then monitored by a detector system or through an internal detector to directly identify radioactive materials present.

A CAM can give both a visual and audible alarm to warn personnel of the presence of airborne contamination.

Process Monitoring Systems

Process monitoring systems monitor certain operations in various facilities to alert operators of abnormal conditions which might lead to the release of excessive amounts of radioactivity to the facility or environment.

Process monitors include temperature, pressure, and flow sensors.

Area and Equipment Surveys

Area and equipment surveys are conducted routinely throughout the facilities to locate sources of radiation and contamination and to detect potential changes in radiological conditions.

Pre-job surveys are performed prior to work in radiological areas in order to evaluate the hazards and determine work limitations and physical safeguards.

Direct surveys with portable radiation detection instruments and removable contamination surveys with smears or swipes are used.

External Personnel Surveys

Personnel surveys are either performed by the individual (self-monitoring) using handheld or automated instruments or by a radiological control technician.

Self-monitoring is typically performed upon exiting a contaminated area at established boundary points.

Personnel monitoring by a RCT is usually conducted whenever contamination of the body or clothing is suspected, or as required by exit monitoring when self-monitoring is not feasible (remote location) or not allowed.

Personnel Internal Monitoring

A routine program of internal contamination monitoring is conducted as a final check on contamination control procedures.

The program consists of external whole/partial body counting and/or urinalysis.

Once the presence of radioactive material has been located, the basic goal underlying any effective contamination control program is to minimize contaminated areas and maintain contamination levels as low as reasonably achievable.

In some situations, this is not always possible due to:

- Economical conditions: Cost of time and labor to decontaminate a location(s) outweighs the hazards of the contamination present.
- Radiological conditions: Radiation dose rates or other radiological conditions present hazards which far exceed the benefits of decontamination.
- Operating conditions: Some areas, e.g., hot cells, will be contaminated due to normal operations.

Other means of control must be initiated when decontamination is not possible.

Engineering control (ventilation and containment), administrative procedures (RWPs), and personnel protective equipment are alternatives for the control of contamination.

In Fixed Contamination Areas the contamination may be covered by paint, floor tiles, etc. when decontamination is not possible.

CONTAMINATION CONTROL MEASURES

- Access/Administrative Controls
- Engineering Controls
- Personnel Protective Measures
- Decontamination
- Preventive Methods

Access/Administrative Controls

Once contamination has been located and quantified and radiological areas have been determined, access control to these areas must be adequately established.

Work Authorization, Radiological Posting, and Radiological Work Permits are the primary administrative controls.

ENGINEERING CONTROLS

Ventilation and Containment (Confinement) are the main engineering controls.

Ventilation is a process of providing adequate air flow to keep the radiological area well ventilated and to keep the potential airborne radioactivity from migrating to other non-radiological areas. This is done thru negative pressure, much like some businesses use to keep their shops temperature and humidity controlled without trying to air condition the outside environment.

Containment (or confinement) is simply keeping radioactive materials in enclosures adequate to prevent the materials from getting outside the container.

PERSONNEL PROTECTIVE MEASURES

In vitro involves counting an excreted sample, such as urine.

The amount of material present in the body is estimated using the amount of materials present in excretions or secretions from the body.

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Table 1 - Weighting Factors For Various Tissues

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External Exposure Control

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Basic Methods for Exposure Reduction

The radiological control organization shall make whatever reasonable efforts it can to reduce exposure to the lowest levels.

There are four basic methods available to reduce external exposure to personnel:

- Reduce the amount of source material (or reduce emission rate for electronically-generated radiation).
- Reduce the amount of time of exposure to the source of radiation.
- Increase the distance from the source of radiation.
- Reduce the radiation intensity by using shielding between the source and personnel.

Basic Methods for Exposure Reduction

In order to use the basic methods for controlling exposure, the worker must be able to determine the intensity of the radiation fields. The following equations are used to make this determination.

A "rule-of-thumb" method to determine the radiation field intensity for simple sources of radioactive material is the "curie/meter/rem" rule. (Co-60)

$$1 \text{ Ci @ 1 meter} = 1 \text{ R/hr}$$

Basic Methods for Exposure Reduction

To determine the gamma radiation field intensity for a radioactive point source

$$I_{1\text{ft}} = 6CEN$$

where:

$I_{1\text{ft}}$ = Exposure rate in R/hr at 1 ft.

C = Activity of the source in Ci

E = The gamma energy in MeV

N = The number of gammas per disintegration

Basic Methods for Exposure Reduction

- This equation is accurate to within +20% for gamma energies between 0.05 MeV and 3 MeV.
- If N is not given, assume 100% photon yield (1.00 photons/disintegration).
- If more than one photon energy is given, take the sum of each photon multiplied by its percentage, i.e.:

$$[(\gamma_1)(\%1) + (\gamma_2)(\%2) + \cdots + (\gamma_n)(\%n)]$$

Basic Methods for Exposure Reduction

For distances in meters:

$$I_{1m} = 0.5CEN$$

For short distances greater than 1 foot from the source, the inverse square law can be applied with reference to the dose rate at 1 foot, resulting in the following equation:

$$I = \frac{6CEN \times 1^2}{d^2}$$

where: d = distance in feet

Example

Determine the exposure rate at 10 ft for a 8 Ci point source of Co-60 that emits a 1.173 and 1.332 MeV gamma, both at 100% of the disintegrations.

$$I \text{ at } 10\text{ft} = \frac{[(6)(8)(\Sigma 1.173(1) + 1.332(1))][1^2]}{(10 \text{ ft})^2} = 1.2 \text{ R/hr}$$

Source Reduction

The first method that should be employed to reduce personnel external exposure is source reduction. If a source can be eliminated or if its hazard potential can be significantly reduced, then other engineering means may not be necessary.

Various techniques are employed to accomplish external exposure reduction using source reduction.

Allow natural decay to reduce source strength

- If the radioisotopes involved are short-lived, then waiting to perform the task may significantly reduce the hazard.
- By waiting for natural decay to reduce the source strength, a considerable savings in external exposure can be achieved.

Source Reduction

Move the source material to another location

- Decontaminate the equipment or material through mechanical or chemical means to remove the source material prior to working in the area or on the equipment.
- Reduce the source material in the system by flushing equipment with hot water or chemical solutions and collect it in a less frequently occupied area.
- Discharge or remove the resin or filtering media prior to working in the area or on the system.
- Move the radioactive source (e.g., a drum, barrel or calibration source) to another location prior to starting work.

Time Savings

Personnel working in radiation fields must limit their exposure time so that they do not exceed their established permissible dose limits and are able to keep exposures ALARA.

The longer the time spent in the radiation field, the greater the exposure to the individual; therefore, the amount of time spent in radiation fields should be reduced.

The Radiological Control Technician needs to be aware that radiation exposures are directly proportional to the time spent in the field. If the amount of time is doubled, then the amount of exposure received is doubled.

Time Savings

Analyze and train using mock-ups of the work site

- A particular task can be analyzed on a mock-up of the system to determine the quickest and most efficient method to perform the task.
- The team of workers assigned to the task can rehearse, without radioactive materials, so that problems can be worked out and the efficiency of the team increased prior to any exposure.
- By determining the most efficient method and rehearsing the task, the amount of time, and therefore the exposure, can be reduced.

Time Savings

Pre-job briefings are an important part of any good ALARA program

- Discussions at the pre-job briefing with the individuals assigned to the task can identify any potential problems not previously identified.
- Identifying personnel responsibilities and the points at which various individuals are required to be present can reduce the overall time required to perform the job.

Review job history files -- Review the files from previously completed tasks of the same nature to identify previous problems and spots where time could be saved.

Time Savings

Pre-stage all tools and equipment -- All tools should be staged prior to entry to prevent the worker from waiting in a radiation field for a tool to be brought.

Pre-assemble equipment and tools outside the area

- Equipment that can be preassembled should be preassembled prior to any entry into the radiation field.
- Tools that require assembly, pre-testing, and/or calibration should be performed outside the radiation field.

Time Savings

Use time limiting devices -- Time limitations for workers can be monitored and limited using various devices such as stopwatches, alarming dosimeters, or radio-transmitting dosimeters.

- Use communication devices such as walkie-talkies
- Poor communication can lead to incorrect or poor quality work and prolonged waiting in the radiation field while supervisors or experts are contacted.
 - Communication devices such as walkie-talkies or radio headsets can alleviate these problems and reduce the amount of time that is spent in the radiation field.

Time Savings

Use a team of workers instead of allowing one individual to receive all of the exposure

- Even if the task requires a minimum amount of time, if it causes one individual to receive an exposure greater than allowable, a team of workers should be used to reduce the individual exposures.
- If a team of workers is used, good communications are necessary to ensure the total exposure for the job does not increase significantly.

Time Savings

Use experienced personnel

- The total time required to perform a job is reduced if experts are used instead of inexperienced personnel.
- Inexperienced personnel should not be trained in significant radiation fields

Exposure Calculation

The exposure received by personnel will increase as the time spent in the radiation field increases.

$$X = \left(\frac{R}{t}\right)T$$

where:

$$\frac{R}{t} = \text{dose rate}$$

$$T = \text{length of time exposed}$$

Stay Time Calculation

When the time allowed in a radiation field is calculated to prevent a worker from exceeding an allowable dose equivalent, it is called "stay time."

Stay time is calculated as follows:

$$\textit{Stay Time} = \frac{H_{\textit{allowable}} - H_{\textit{received}}}{\textit{dose equivalent rate}}$$

where:

$$\begin{aligned} H_{\textit{allowable}} &= \textit{maximum allowable dose equivalent} \\ H_{\textit{received}} &= \textit{dose equivalent already accumulated} \end{aligned}$$

Distance

The intensity of the radiation field decreases as the distance from the source increases.

Therefore, increasing the distance will reduce the amount of exposure received.

In many cases, increasing the distance from the source is more effective than decreasing the time spent in the radiation field.

Distance

The radiation intensity for a point source decreases according to the Inverse Square Law which states that as the distance from a point source changes the dose rate decreases or increases by the square of the ratio of the distances from the source.

The inverse square law becomes inaccurate close to the source (i.e., about 10 times the diameter of the source).

For a point source, if the distance is doubled, the radiation intensity will be reduced by a factor of $(2)^2$ or 4.

Distance

Remote handling tools/remote control devices

- Tools, such as tongs or long-handled tools, are an effective means of increasing the distance from a point source to a worker.
- For very high radiation fields, remote control devices may be appropriate, especially if the task is performed frequently.

Remote observation by cameras or indicators

- Gauges or meters can be moved to a location remote from the source of radiation.
- Closed-circuit television and video cameras can be used to allow observation of work activities or system operations from a location remote to the source of radiation.

Distance

Move work to another location

- If the source of radiation can not be reduced, then possibly the work can be moved to a low exposure area.
- For example, if a pump or valve needs reworking, then an exposure savings could be achieved by removing the component from the system and performing all repair work in a lower exposure area.

Distance

Posting of areas -- Posting of radiological areas based on radiation level is a method for increasing the distance between the workers and the radiation source.

Extendable Instruments -- Extendable radiation survey instruments, such as the Eberline Teletector or RO-7, can reduce the exposure to the surveyor by increasing the distance.

Distance

- For workers or inspectors not actively engaged in the work activity in the radiation field, moving to a lower exposure rate "waiting" area can be effective.
- Identifying "low dose rate waiting areas" can notify workers of the location of the lowest exposure rate in an area or room.
- Be aware of the location of radiation sources at the worksite and locate the worker at a point farthest from the source.
- Work at arm's length and do not lie on or hug radioactive components.

Inverse Square Law

The exposure rate is inversely proportional to the square of the distance from the source. The mathematical equation is:

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

where:

I_1 = Exposure rate at distance (d_1)

I_2 = Exposure rate at distance (d_2)

d_1 = First distance from the source

d_2 = Second distance source

Inverse Square Law

A somewhat useful variation on the inverse square law is the use of a quadratic equation when the actual distance to the source is not known.

How does the inverse square law apply to alpha, beta, and neutron radiation ?

Inverse Square Law

The inverse square law holds true only for point sources; however, it gives a good approximation when the source dimensions are smaller than the distance from the source to the exposure point.

Some sources, such as a pipe or tank, cannot be treated as a point source unless the distance to the source is greater than the length of the source.

Line Source Calculations

The actual calculations for a line source involve calculus; however, the mathematics can be simplified if the line source is treated as a series of point sources placed side by side along the length of the source.

If the line source is treated in this manner, the relationship between distance and exposure rate can be written mathematically as:

$$I_1 d_1 = I_2 d_2$$

Line Source Calculations

The exposure rate is inversely proportional to the distance from the source

Assuming the source material is distributed evenly along the line

Assuming the point at which the exposure rate is calculated is on a line perpendicular to the center of the line source

Assuming the width or diameter of the line is small compared to the length

Planar Or Surface Sources

Planar or surface sources of radiation can be any type of geometry where the width or diameter is not small compared to the length.

When the distance to the plane source is small compared to the longest dimension, then the exposure rate falls off a little slower than $1/d$ (i.e. not as quickly as a line source).

As the distance from the plane source increases, then the exposure rate drops off at a rate approaching $1/d^2$

Mass Attenuation Coefficient

The linear attenuation coefficient (μ) is dependent on:

- photon energy
- chemical composition of the absorber (Z)
- the physical density of the absorber

The linear attenuation is the probability of a photon interaction per path length and has units of $(\text{length})^{-1}$ (typically cm^{-1}).

The linear attenuation coefficient will change depending on the physical density of the absorber.

Mass Attenuation Coefficient

Mathematically:

$$\mu_m = \mu_l / \rho$$

where:

μ_m = mass attenuation coefficient

μ_l = linear attenuation coefficient

ρ = physical density

When the units of the linear attenuation coefficient are cm^{-1} and the units of physical density are mg/cm^3 then the units of mass attenuation coefficient become cm^2/mg .

Density-Thickness

The mass attenuation coefficient μ_m is used in the attenuation equation

$$I = I_0 e^{-\mu_m x}$$

where:

I = shielded (attenuated) radiation intensity

I_0 = unshielded radiation intensity

μ_m = mass attenuation coefficient

x = density-thickness value

Density-Thickness

When μm is used in the attenuation equation, x must have units of mg/cm^2 to cancel out the units of μm . With these units, x is called *density-thickness*.

Density-thickness is a value equal to the product of the density of the absorbing material and its thickness.

This value is given in units of mg/cm^2 .

The density of any material is a measure of its mass per unit volume, as compared to the density of water. Water has a density of $1 \text{ g}/\text{cm}^3$, or $1000 \text{ mg}/\text{cm}^3$.

Density-Thickness

According to ICRP 15, the density of soft human tissue is equal to 1000 mg/cm^3 . Using this value we can calculate density-thickness values for various depths that radiation may penetrate into the human body and cause damage.

For purposes of reporting radiation dose the tissue depths of concern are the skin (shallow dose), the lens of the eye, and the whole body (deep dose).

The concept of "density-thickness" is important to discussions of radiation attenuation by human tissue, as well as detector shielding and windows, and dosimetry filters.

Density-Thickness

Although materials may have different densities and thicknesses, if their density-thickness values are the same, they will attenuate radiation in a similar manner.

For example, a piece of mylar used as a detector window with a density of 7 mg/cm^2 will attenuate radiation similar to the skin of the human body.

These values can be used to design radiation detection instrumentation such that detector windows and shields have the same or similar density-thickness values.

Density-Thickness

Any radiation passing the detector window would also penetrate to the basal layer of the skin on the human body and deposit energy in living tissue.

External dosimetry can be designed around these values such that dose equivalent is determined for the skin of the whole body, lens of the eye, and whole body.

For example, a dosimeter filter may be designed as 1000 mg/cm². Any radiation passing this filter would also pass through the skin of the whole body and deposit energy in vital human organs.

Density-thickness Values for Human Body

Tissue	Depth	Density-thickness
Skin (shallow dose)	0.007 cm	7 mg/cm ²
Lens of the eye	0.3 cm	300 mg/cm ²
Whole body (deep dose)	1.0 cm	1000 mg/cm ²

Shielding Calculations

The simplest method for determining the effectiveness of the shielding material is using the concepts of half-value layers (HVL) and tenth-value layers (TVL).

One **half-value layer** is defined as the amount of shielding material required to reduce the radiation intensity to **one-half** of the unshielded value.

$$HVL = \frac{\ln 2}{\mu} = \frac{0.693}{\mu}$$

The symbol μ is known as the *linear attenuation coefficient* and is obtained from standard tables for various shielding materials.

Shielding Calculations

One **tenth-value layer** is defined as the amount of shielding material required to reduce the radiation intensity to **one-tenth** of the unshielded value.

$$TVL = \frac{\ln(10)}{\mu} = \frac{2.3026}{\mu}$$

Both of these concepts are dependent on the energy of the photon radiation and a chart can be constructed to show the HVL and TVL values for photon energies.

Half-Value Layers

Photon Energy (keV)	<u>HVL (cm)</u>			
	Lead (11.35 g/cm ³)	Iron (7.86 g/cm ³)	Concrete (2.4 g/cm ³)	Water (1.0 g/cm ³)
500	0.38	1.0	3.3	7.2
1000	0.86	1.5	4.5	9.8
1500	1.2	1.8	5.6	12.0
2000	1.3	2.1	6.4	14.0
3000	1.5	2.4	7.9	17.5

HVL Equation

The basic equation for using the HVL concept is:

$$I = I_0 \left(\frac{1}{2}\right)^n$$

Where:

$I =$ shielded exposure rate

$I_0 =$ unshielded exposure rate

$n =$ no. of HVLs $= \frac{\text{shield thickness (cm)}}{\text{HVL thickness (cm)}}$

TVL Equation

The basic equation for using the TVL concept is:

$$I = I_0 \left(\frac{1}{10}\right)^n$$

Where

$I =$ shielded exposure rate

$I_0 =$ unshielded exposure rate

$n =$ no. of TLVs $= \frac{\text{shield thickness (cm)}}{\text{TVL thickness (cm)}}$

Internal Exposure Control

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Entry Of Radioactive Materials Into The Body

Modes of Entry

Inhalation: Materials enter the body in the air that is breathed.

Ingestion: Materials enter the body through the mouth.

Absorption: Material enters the body through intact skin.

Entry through wounds:

- *Penetration:* Materials enter (passively) through existing wounds which were not adequately covered.
- *Injection:* Materials enter (forcefully) through wounds incurred on the job.

Preventive Measures

Inhalation -- assessment of conditions, use of engineering controls, respiratory protection equipment

Ingestion -- proper radiological controls and work practices

Absorption -- assessment of conditions and protective clothing

Entry through wounds -- not allowing contamination near a wound by work restriction or proper radiological controls if an injury occurs in a contaminated area.

Preventive Measures

Note that the preventive measures are designed to do one of two things:

- Minimize the amount of radioactive materials present which are available to enter the body, or
- Block the pathway from the source of radioactive materials into the body.

Annual Limit On Intake Derived Air Concentration

Assimilation of radioactive materials in the workplace occurs most often as a result of inhalation of airborne radioactive contaminants. With some nuclides, specifically tritium, absorption through the skin is also a major concern.

Two limiting values have been calculated and are available for use in limiting the inhalation of radioactive materials.

These limiting values are:

- Annual Limit on Intake (ALI)
- Derived Air Concentration (DAC)

Annual Limit On Intake Derived Air Concentration

Annual Limit on Intake is the quantity of a single radionuclide which, if inhaled or ingested in one year, would irradiate a person, represented by reference man (ICRP Publication 23), to the limiting value for control of the workplace.

ICRP 68 methodology is being applied to 10CFR835 DAC factors but have not been adopted by 10CFR20.

Derived Air Concentration is the quantity obtained by dividing the ALI for any given radionuclide by the volume of air breathed by an average worker during a working year ($2.4E3m^3$).

Annual Limit On Intake Derived Air Concentration

According to ICRP 23, reference man breathes at an average rate of 20 liters per minute, or 0.02 m³/min. In the course of one working year, the total volume breathed would be:

$$\frac{0.02 \text{ m}^3}{\text{min}} \times \frac{60 \text{ min}}{\text{hr}} \times \frac{8 \text{ hrs.}}{\text{day}} \times \frac{5 \text{ days}}{\text{week}} \times \frac{50 \text{ weeks}}{\text{year}} = 2400 \text{ m}^3$$

Annual Limit On Intake Derived Air Concentration

The DAC is equal to the ALI divided by the volume of air breathed by the average worker during a working year:

$$DAC = \frac{ALI}{2400 m^3}$$

Annual Limit On Intake Derived Air Concentration

During routine operations, the combination of physical design features and administrative controls shall provide that:

- a) The anticipated occupational dose to general employees shall not exceed the limits.
- b) The ALARA process is utilized for personnel exposures to ionizing radiation.

Annual Limit On Intake Derived Air Concentration

Monitoring of airborne radioactivity shall be performed:

- 1) Where an individual is likely to receive an exposure of 40 or more DAC-hours in a year; or
- 2) As necessary to characterize the airborne radioactivity hazard where respiratory protective devices for protection against airborne radionuclides have been prescribed.

Real-time air monitoring shall be performed as necessary to detect and provide warning of airborne radioactivity concentrations that warrant immediate action to terminate inhalation of airborne radioactive material.

Annual Limit On Intake Derived Air Concentration

Measures used to minimize the concentration of airborne contaminants remain the primary means of minimizing potential exposure.

Minimizing the concentrations to below DAC values helps insure that workers could not exceed the ALI even if they were in the area continuously for long durations and breathing air at those concentrations.

An Airborne Radioactivity Area is any area where the concentration of airborne radioactivity exceeds or is likely to exceed the DAC value or where an individual present in the area without respiratory protection could receive an intake exceeding 12 DAC-hours in a week.

Annual Limit On Intake Derived Air Concentration

Posting of airborne radioactivity areas controls access to minimize exposure.

Minimize the stay time of workers in airborne areas to short periods of time.

Augment installed engineering controls with respiratory protection equipment to further reduce the concentration of contaminants in the air the workers are actually breathing.

Annual Limit On Intake Derived Air Concentration

The limitations imposed in terms of dosage to exposed workers are expressed as an annual limit.

Concentrations of contaminants in the air are monitored by continuous monitoring equipment and are supplemented by grab sampling as required.

Engineering controls are augmented with respiratory protection equipment when airborne contaminants exceed or potentially exceed DAC values.

Movement of Radioactive Materials Through the Body

There is no simple device that can be placed on or in the body to determine the quantities of radioactive materials in the body or the dose received by the individual as a result of irradiation of body tissues by these materials.

When radioactive material enters the body, the assessment methods must be based on what happens to the materials, or what the body does with them.

The body does not possess the ability to differentiate between a nonradioactive atom and a radioactive atom of the same element. In terms of metabolic processes, the material is handled the same way.

Movement of Radioactive Materials Through the Body

Once the material is in the body, then its behavior is governed by the chemical form, its location in the body, and the body's need for that material.

- Chemical form - solubility

- Location - pathways

- Body's need - intake and incorporation vs. elimination

Intake and Uptake

Intake: the amount of radionuclide taken into the body

Uptake: the amount of radionuclide deposited in the body which makes its way into the body fluids or systemic system (i.e., blood)

Inhaled Radioactive Materials

General Pathways

- Deposition in lungs with eventual transfer to GI tract or retention
- Transport to body fluids
- Transfer to lymph nodes with eventual movement to body fluids
- Retention in lymph nodes

Inhaled Radioactive Materials

Once in the bodily fluids, possibilities include:

- Transfer to specific organ
- Filtration and elimination by kidneys
- Transport and removal from body fluids through circulatory systems (perspiration)

Inhaled Radioactive Materials

Insoluble particulates

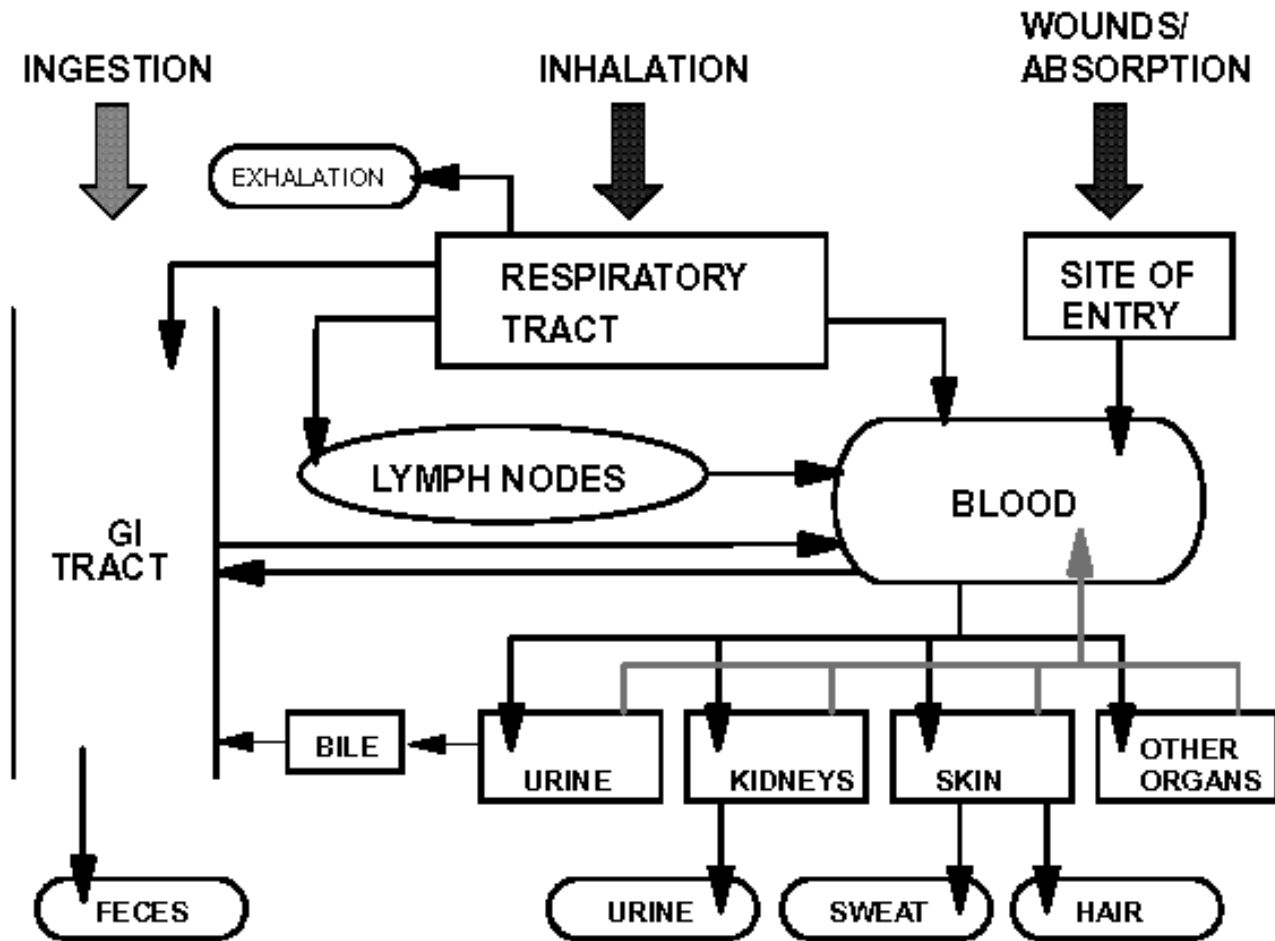
- Lung retention time based on particle size and density
- Removal in mucous to digestive tract
- Elimination in fecal waste

Inhaled Radioactive Materials

Soluble particulate materials

- Retention in lungs based on size and density – some exhalation
- Some removed to GI tract for elimination or to body fluids
- Transfer to body fluids via lymph nodes or directly from lungs
- Some retention in lymph nodes
- Body fluids to tissue or organ of interest
- Excretion

Metabolic Pathways



Ingested Radioactive Materials

For elements not used by the body, absorption by ingestion is poor, and most materials will pass straight through the body.

Materials pass through stomach to small intestine where transport of soluble materials to body fluids will occur.

From body fluids, materials go to the organs and/or are removed through normal biological elimination processes.

Ingested Radioactive Materials

Soluble materials

- Transfer to body fluids in intestines
- Circulation, absorption, incorporation in tissues and organs
- Elimination in urine

Insoluble materials

- Passes straight through
- Elimination in feces

Absorbed Radioactive Materials

Many radioactive nuclides are absorbable through the skin. These nuclides include tritium, iodine, and some of the transuranics in an acidic form.

Most of these do not pose any considerable concern because of the relative percentages absorbed as opposed to entry through inhalation.

The most important of these is tritium as water vapor.

Once absorbed into the body, tritium exchanges freely with hydrogen, disperses throughout the body almost immediately, and irradiates tissues throughout the body.

Target Organs

Some elements are collected in target organs. As an example, iodine is collected by the thyroid gland.

Major dose to the thyroid could be expected as a result of gamma and beta interactions emitted by iodine collected in the thyroid gland.

The radiation emitted from iodine in the thyroid also can irradiate other nearby parts of the body. Gamma radiation can penetrate tissue very easily and cause interactions in parts of the body in which no iodine is located.

Target Organs

Other elements are processed differently.

- Some are distributed freely throughout body fluids.
- Some are collected in specific organs such as the kidneys, spleen or bone.
- Sr, U, Pu are concentrated in the bone

Some materials which enter as particulate materials may spend the majority of their stay in the body in the lungs.

Elimination Processes

Normal Biological Elimination

Radioactive materials incorporated into body tissues and organs are eliminated from the body as are their non-radioactive counterparts.

Eliminated through exhalation, perspiration, urination, and defecation.

Each element has a measurable biological half-life - the time required to reduce the amount of material in the body to one-half of its original value.

The biological half-life is independent of the physical or radiological half-life.

Elimination Processes

Radioactive Decay

Each radioactive nuclide has a distinctive decay rate that is not influenced by any physical process, including biological functions.

The amount of time required for one half of the material in the body to decay is called the radiological half-life.

Radioactive decay will result in reduction of the quantity of the original nuclides deposited in the body.

It is important to remember that the progeny of these nuclides may also be radioactive.

It is possible that the progeny could introduce completely different concerns for internal dose assessments.

Effective Half-life

The combined processes of biological elimination and physical decay result in the removal of radioactive materials at a faster rate than the individual reduction rate produced by either method.

This means that:

$$T_e < T_b, T_p$$

Effective Half-life

The removal rate as a result of the combined processes is measured as an effective half-life and is calculated using the following formula:

$$T_e = \frac{T_b \times T_p}{T_b + T_p}$$

where:

T_e	= effective half-life
T_b	= biological half-life
T_p	= physical half-life

Effective Removal Constant

Another way that this is expressed is the effective removal constant, λ_e , which is the composite of the physical decay constant λ_p and the biological elimination constant λ_b .

$$\lambda_e = \lambda_b + \lambda_p$$

Example Calculations

Determine the effective half-life of tritium if the biological half-life is 10 days and the physical half-life is 12.3 years.

Determine the effective half-life of ^{59}Fe if the biological half-life is 2000 days and the physical half-life is 44.56 days.

Blocking Agents

A blocking agent saturates the metabolic processes in a specific tissue with the stable element and reduces uptake of the radioactive forms of the element.

As a rule, these must be administered prior to or almost immediately after the intake for maximum effectiveness and must be in a form that is readily absorbed.

The most well known example of this is stable iodine, as potassium iodide, which is used to saturate the thyroid gland, thus preventing uptake of radioactive iodine in the thyroid.

Diluting Agents

A diluting agent is a compound which includes a stable form of the nuclide of concern. By introducing a large number of stable atoms, the statistical probability of the body incorporating radioactive atoms is reduced.

Diluting agents can also involve the use of different elements which the body processes in the same way. This type of treatment is called displacement therapy.

The compound used must be as readily absorbed and metabolized as the compound that contains the radioisotope.

Mobilizing Agents

A mobilizing agent is a compound that increases the natural turnover process, thus releasing some forms of radioisotopes from body tissues.

Usually most effective within 2 weeks after exposure; however, use for extended periods may produce less dramatic reductions.

Chelating Agents

A chelating agent is a compound which acts on insoluble compounds to form a soluble complex ion which can then be removed through the kidneys.

Commonly used to enhance elimination of transuranics and other metals.

Therapy is most effective when begun immediately after exposure if metallic ions are still in circulation and is less effective once metallic ions are incorporated into cells or deposited in tissue such as bone.

Chelating Agents

Common chelating agents include EDTA and DTPA

–CaNa₂ EDTA - commonly used in cases of lead poisoning. It is also effective against zinc, copper, cadmium, chromium, manganese, and nickel.

–CaNa₃ DTPA - transuranics such as plutonium and americium. If the chelating agent is administered within a few hours of the intake the residual transuranics in the body can be reduced by a factor of 10.

Diuretics

Diuretics increase urinary excretion of sodium and water.

Used to reduce internal exposure, however its use has been limited. Applications include 3-H, 42-K, 38-Cl and others.

Can lead to dehydration and other complications if not performed properly.

Expectorants and Inhalants

Used to increase flow of respiratory tract excretions.

Inhalants have been used in removing radioactive particles from all areas of lungs.

Lung Lavage

Involves multiple flushing of lungs with appropriate fluid to remove radioactive materials in the lungs.

Usually limited to applications where resulting exposures would result in appearance of acute or subacute radiation effects.

The procedure has been proven effective in cases of “black lung” disease exhibited by coal miners where the procedure is used to remove some of the small coal particles that were deposited at the alveolar region of the lungs.

RADIOACTIVE SOURCE CONTROL

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A radioactive source is material used for its emitted radiation. Sources are constructed as sealed or unsealed and are classified as accountable or non-accountable . Radioactive sources are used for response checks in the field, functional checks, calibration of instruments and monitors to traceable standards. To ensure the safety and welfare of all personnel it is important to maintain control of radioactive sources.

Radioactive sources are controlled to minimize the potential for:

Spread of contamination

Unnecessary exposure to personnel

Loss or theft

Improper disposal

In accordance with 10 CFR 835, Subpart M, the following provisions apply to sealed sources:

A. Sealed radioactive sources shall be used, handled, and stored in a manner commensurate with the hazards associated with operations involving the sources.

B. Each accountable sealed radioactive source shall be inventoried at intervals not to exceed six months. This inventory shall:

1. Establish the physical location of each accountable sealed radioactive source;
2. Verify the presence and adequacy of associated postings and labels; and
3. Establish the adequacy of storage locations, containers, and devices.

C. Except for sealed sources consisting solely of gaseous radioactive material or tritium, each accountable sealed radioactive source shall be subject to a source leak test upon receipt, when damage is suspected, and at intervals not to exceed six months. Source leak tests shall be capable of detecting radioactive material leakage equal to or exceeding 0.005 μCi .

D. An accountable sealed radioactive source is not subject to periodic source leak testing if that source has been removed from service. Such sources shall be stored in a controlled location, subject to periodic inventory, and subject to source leak testing prior to being returned to service.

E. An accountable sealed radioactive source is not subject to periodic inventory and source leak testing if that source is located in an area that is unsafe for human entry or otherwise inaccessible.

F. An accountable sealed radioactive source found to be leaking radioactive material shall be controlled in a manner that minimizes the spread of radioactive contamination.

Sources are controlled using the following precautions:

1. Each source is to be inspected before each use.
2. Remove damaged sources from service.
3. Fingers, whether gloved or not, or other objects should never be allowed to touch the active surface of unsealed sources.
4. Protect the source from being contaminated when used in a surface contamination area.

Sealed radioactive sources not in storage containers or devices and not labeled by the manufacturer must be clearly marked with a radiation symbol and have a durable label/ tag containing the following information:

- a. Radionuclide
- b. Amount of activity
- c. Name of manufacturer
- d. Date of assay
- e. Model and serial numbers (where available)

1. Accountable Sealed Radioactive Source means a **sealed radioactive** source having a half-life equal to or greater than 30 days and an isotopic activity equal to or greater than the corresponding value provided in Appendix E of 10CFR 835.

2. Radioactive Material Area means **any area within a controlled area**, accessible to individuals, in which items or containers of radioactive material exist and the total activity of radioactive material exceeds the applicable values provided in Appendix E to 10 CFR 835.

3. Sealed Radioactive Source means a radioactive source manufactured, obtained, or retained for the purpose of utilizing the emitted radiation. The sealed radioactive source consists of a known or estimated quantity of radioactive material contained within a sealed capsule, sealed between layer(s) of nonradioactive material, or firmly fixed to a non-radioactive surface by electroplating or other means intended to prevent leakage or escape of the radioactive material.

4. Source Leak Test means a test to determine if a sealed radioactive source is leaking radioactive material.

