Course Title: **Radiological Control Technician**
Module Title: **Dosimetry**
Module Number: **2.04**

Objectives:

2.04.01 Identify the DOE external exposure limits for general employees.

2.04.02 Identify the DOE limits established for the embryo/fetus of a declared pregnant female general employee.

☞ 2.04.03 Identify the administrative exposure control guidelines at your site, including those for the:
a. General employee
b. Member of the public/minor
c. Incidents and emergencies
d. Embryo/fetus

☞ 2.04.04 Identify the requirements for a female general employee who has notified her employer in writing that she is pregnant.

2.04.05 Determine the theory of operation of a thermoluminescent dosimeter (TLD).

2.04.06 Determine how a TLD reader measures the radiation dose from a TLD.

2.04.07 Identify the advantages and disadvantages of a TLD compared to a film badge.

☞ 2.04.08 Identify the types of beta-gamma TLDs used at your site.

☞ 2.04.09 Identify the types of neutron TLDs used at your site.

☞ 2.04.10 Determine the requirements for use of TLDs used at your site.

☞ 2.04.11 Determine the principle of operation, and the types used, for the personnel neutron dosimeters used at your site.

☞ 2.04.12 Determine the principle of operation of self-reading dosimetry (SRD) used at your site.

☞ 2.04.13 Determine the principle of operation, and guidelines for use, for the alarming dosimeters used at your site.

☞ 2.04.14 List the types of bioassay monitoring methods at your site.

INTRODUCTION

Radiation dosimetry is the branch of science that attempts to quantitatively relate specific measures made in a radiation field to chemical and/or biological changes that the radiation would produce in a target. Dosimetry is essential for quantifying the incidence of various biological changes as a function of the amount of radiation received (dose-effect relationships), for comparing different experiments, for monitoring the radiation exposure of individuals, and for surveillance of the environment.

External dosimetry is the science dealing with the measurement of a radiation field incident to the body and the evaluation of the dose equivalent resulting from energy deposited within the body by radiation. External dose is usually a derived or inferred quantity since it is not possible to directly measure the exact dose to any organ or tissue. Any measurement must be compared to a known quantity to derive dose and dose equivalent. This process is called "calibration".

Internal dosimetry is the analysis and measurement of radionuclides in humans or bioassay samples and the evaluation of intakes and doses from those measurements. It involves evaluation of bioassay data, evaluation of the intake, distribution, retention, and elimination of radionuclides, and evaluation of various absorbed doses and dose equivalent quantities. Internal dosimetry is inherently indirect in nature. It is not possible to determine the exact organ absorbed dose, dose equivalent or effective dose equivalent in a living human being resulting from an intake of radioactive materials. Internal dose is usually a derived or inferred quantity, obtained by evaluation of indirect measurements and computational models. This is particularly true for alpha- and beta-emitting radionuclides in the body which have low photon emission abundances. Direct measurements of internalized photon-emitting radionuclides in organs also may be difficult because of attenuation and scattering by overlying tissues.

The capability to accurately measure and analyze radioactive materials and workplace conditions, and determine personnel radiation exposure is fundamental to the safe conduct of radiological operations. Accordingly, DOE shall ensure radiological measurements, analyses, worker monitoring results and estimates of public exposures are accurate and appropriately made. 10 CFR 835 prescribes the requirements for both external and internal dose monitoring.

It is the responsibility of all workers to wear personnel monitoring devices where required by Radiological Work Permits, signs, procedures or by radiological control personnel. They are also expected to report immediately the loss, damage or unexpected exposure of personnel monitoring devices or off-scale readings of self-reading dosimeters to the Radiological Control Organization (RCO). All employees are expected to keep track of their radiation exposure status and avoid exceeding radiological Administrative Control Levels. Additionally, all should notify the RCO of off-site occupational radiation exposures so that worker dosimetry records can be updated.

References:

1. "Basic Radiation Protection Technology," Gollnick, Daniel; Pacific Radiation Press; 1994.
2. ANL-88-26 (1988) "Operational Health Physics Training," Moe, Harold; Argonne National Laboratory, Chicago.
3. "Radiological Control Standard," DOE-STD-1098-99.
4. 10 CFR Part 835 (1998) "Occupational Radiation Protection."

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/cm^2).

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).

Weighting Factor (W_t):

The fraction of the overall health risk, resulting from uniform, whole body irradiation, attributable to specific tissue (T). The dose equivalent to the affected tissue 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
Breasts	0.15
Red bone marrow	0.12
Lungs	0.12
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.

Committed Effective Dose Equivalent (CEDE):

The sum of the Committed Dose Equivalents to various tissues, excluding the skin and lens of the eye, in the body, each multiplied by the appropriate weighting factor (W_t). Committed Effective Dose Equivalent is expressed in units of rem (or sievert).

Total Effective Dose Equivalent (TEDE):

The sum of the effective dose equivalent (for external exposures) and the Committed Effective Dose Equivalent (for internal exposures). Deep Dose Equivalent to the whole body may be used as effective dose equivalent for external 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):

For the radionuclides listed in Appendix A of 10 CFR 835, 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.

In Vivo:

A direct bioassay measurement of radioactivity in living tissue, for example, a whole body count or chest count.

In Vitro:

The bioassay measurement of radioactivity by means of internal representative sampling in order to estimate the radioactivity in tissue. Examples are analysis of urine and fecal collections.

Background:

Radiation from: naturally occurring radioactive materials which have not been technologically enhanced, cosmic sources, global fallout as it exists in the environment (such as from the testing of nuclear explosive devices), radon and its progeny in concentrations or levels existing in buildings or the environment which have not been elevated as a result of current or prior activities, and consumer products containing nominal amounts of radioactive material or producing nominal amounts of radiation.

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 exposure limits to the embryo/fetus as provided in 10 CFR 835.206. This declaration may be revoked, in writing, at any time by the declared pregnant worker.

DOE LIMITS

Limits are the legal maximum values stated in 10 CFR 835. To exceed these values is to violate the law. Programs must be in place to ensure that exposures to ionizing radiation are kept below these levels. To accomplish this, Administrative Control Levels are selected well

below the regulatory limits. These control levels are usually multi-tiered with increasing levels of authority required to approve higher Administrative Control Levels.

Annual dose equivalent limits are based on a calendar year (January 1st through December 31st). For assigning internal dose equivalent received from intakes (CDE and CEDE), the total 50-year committed dose received is assigned to the time of the intake even though the actual dose is proportionally received over the 50-year period.

2.04.01 Identify the DOE external exposure limits for occupational workers.

General Employees

General employees are DOE employees or DOE contractors. A Radiological Worker is a general employee whose job assignment involves operation of radiation producing devices or working with radioactive materials, or who is likely to be routinely occupationally exposed above 0.1 rem (0.001 sievert) per year total effective dose equivalent.

Radiological workers from other DOE or DOE contractor facilities may receive occupational exposure to ionizing radiation as a radiological worker if they:

- Provide a record of current Radiological Worker I or II standardized core training,
- Receive site-specific Radiological Worker I or II training at the facility where they will be working, and
- Provide their radiation dose record or a written estimate for the current year.

Table 2 lists the various legal limits for exposure to ionizing radiation. There are four general categories listed: whole body, lens of the eyes, extremities and organ/tissue/skin. These limits are also covered in 10 CFR 835.208 and the Radiological Control Standard (RCS). Exposures should be well below the limits in this table and maintained as low as reasonably achievable.

Table 2 - Summary of Dose Limits

TYPE OF EXPOSURE	ANNUAL LIMIT
General Employees: Whole Body (internal + external)	5 rem (0.05 sievert)
Lens of Eye	15 rem (0.15 sievert)
Extremity (hands and arms below the elbow; feet and legs below the knees)	50 rem (0.5 sievert)
Any organ or tissue (other than lens of eye) and skin	50 rem (0.5 sievert)
Declared Pregnant Worker: Embryo/Fetus	0.5 rem (0.005 sievert) per gestation period
Minors (under age 18) and Students: Whole body (internal + external)	0.1 rem (0.001 sievert)
Extremity/Skin	5 rem (0.05 sievert)
Lens of Eye	1.5 rem (0.015 sievert)
Members of the public: Whole Body (internal + external)	0.1 rem (0.001 sievert)

Notes:

1. Internal dose to the whole body should be calculated as committed effective dose equivalent. The committed effective dose equivalent is the resulting dose committed to the whole body from internally deposited radionuclides over a 50-year period after intake.

2. The annual limit of exposure to "any organ or tissue" is based on the committed dose to that organ or tissue resulting from internally deposited radionuclides over a 50-year period after intake plus any external effective dose equivalent to that organ during the year.

3. Exposures due to background radiation, therapeutic and diagnostic medical procedures, and participation in medical research programs should not to be included in either personnel radiation dose records or assessment of dose against the limits in this table.

Minors//Public

Minors are individuals less than 18 years of age. The public are defined as individuals not occupationally exposed to radiation or radioactive materials. An individual is not a "member of the public" during any period in which the individual receives an occupational dose. Occupational dose is an individual's dose due to exposure to ionizing radiation (external and internal) as a result of that individual's work assignment. Occupational dose does not include exposure received as a medical patient, background radiation, or participation in medical research programs. The DOE limit for exposure to minors and the public is stated in 10 CFR 835.207 and 835.208 and are listed in Table 2.

2.04.02 Identify the DOE limits established for the embryo/fetus of a female occupational worker.

Embryo/Fetus of Declared Pregnant Workers

After a female general employee voluntarily notifies her supervisor in writing that she is pregnant, for the purposes of embryo/fetal dose protection, she should be considered a declared pregnant worker. The employer should provide the option of a mutually agreeable reassignment of work tasks, without loss of pay or promotional opportunity, such that further occupational radiation exposure is unlikely.

For a declared pregnant worker who chooses to continue radiological work:

- The dose limit for the embryo/fetus for the entire gestation period (from conception to birth) is 0.5 rem (0.005 sievert) {10 CFR 835.206}.

- Efforts should be made to avoid exceeding 0.05 rem (0.0005 sievert) per month to the pregnant worker {10 CFR 835.206}.

If the dose is likely to approach 0.05 rem/month (0.0005 sievert/month), additional dosimetry will be assigned to monitor the dose to the embryo/fetus.

If the dose to the embryo/fetus is determined to have already exceeded 0.5 rem (0.005 sievert) when a worker notifies her employer of her pregnancy, the worker should not be assigned to tasks where additional occupational radiation exposure is likely during the remainder of the gestation period.

Emergency Exposures

10 CFR PART 835.1302 Emergency Exposure Situations

For emergency situations, general employees could be allowed to exceed specified dose limits. The level of exposure permitted will depend upon the severity of the emergency situation. Exposures up to 2 times the annual dose limits could be permitted to protect against property loss. Higher exposures, up to 5 times the annual dose limits or greater, could be permitted to save lives and protect public health. The DOE requires that the details of any exposure in excess of the annual dose limits be documented in the occupational exposure record of the affected employee. In addition, the incident must be investigated and the results reported to DOE. Departmental requirements for occurrence reporting and processing provide a mechanism for such investigations and reports. The employee must not be allowed to receive further exposure until approval is first obtained from the contractor management and responsible DOE field organization. Also, the employee must receive counseling from the appropriate health experts regarding the consequences of receiving additional occupational exposure that year and the affected employee must agree, before returning to radiological work. The operation that caused the exposure must cease pending a finding by DOE that the conditions that caused the exposure had been eliminated.

Planned Special Exposures

A planned special exposure may be authorized for a radiological worker to receive doses in addition to and accounted for separately from the doses received under the normal occupational limits specified in Sec. 835.202(a) provided that each of the following conditions are satisfied:

1. The planned special exposure is considered only in an exceptional situation when alternatives that might prevent a radiological worker from exceeding the limits in 835.202(a) are unavailable or impractical;

2. The contractor management (and employer, if the employer is not the contractor) specifically requests the planned special exposure, in writing; and

3. Joint written approval is received from the appropriate DOE Headquarters program office and the Secretarial Officer responsible for environment, safety, and health matters.

Prior to requesting an individual to participate in an authorized planned special exposure, the individual's dose from all previous planned special exposures and all doses in excess of the occupational dose limits should be determined. An individual should not receive a planned special exposure that, in addition to these doses determined, would result in a dose exceeding:

1. In a year, the numerical value of the dose limits established in 835.202(a); or

2. Over the individual's lifetime, five times the numerical value of the dose limits provided in 835.202(a).

Prior to a planned special exposure, written consent should be obtained from each individual involved. Each individual consent should include:

1. The purpose of the planned operations and procedures to be used;

2. The estimated doses and associated potential risks and specific radiological conditions and other hazards which might be involved in performing the task: and

3. Instructions on the measures to be taken to keep the dose ALARA considering other risks that may be present.

Records of the conduct of a planned special exposure should be maintained and a written report submitted within 30 days after the planned special exposure to the approving organizations. The dose from these planned special exposures is not to be considered in controlling future occupational dose as part of the normal occupational dose of the individual under 835.202(a).

SITE ADMINISTRATIVE GUIDELINES

2.04.03 *Identify the administrative exposure control guidelines at your site, including those for the:*

a. Radiation worker
b. Non-radiation worker
c. Incidents and emergencies
d. Embryo/Fetus

Radiological Workers

(Insert site specific information here)

Non-radiation Worker

(Insert site specific information here)

Exposure from Incidents or Emergencies

(Insert site specific information here)

Embryo/Fetus of a Declared Pregnant Worker

(Insert site specific information here)

2.04.04 Identify the requirements for a female radiation worker who has notified her employer in writing that she is pregnant.

SITE EXPOSURE REQUIREMENTS FOR THE UNBORN CHILD

(Insert site specific information here)

TYPES OF DOSIMETRY

As a result of irradiation, some solid substances undergo changes in some of their physical properties. These changes amount to storage of the energy from the radiation. Since the energy is stored, these materials can be used for dosimeters. The features that have been studied include:

Optical density changes

Optical density changes involve a change in the color of some types of plastics and glass. In glass, the dose range is 10^3 to 10^6 rads (10 to 10^4 gray). The range for plastics is 10^6 to 10^9 rads (10^4 to 10^7 gray). Film badges, a type of optical density dosimetry, provides low range monitoring 10 mR to 10 R for personnel and high range monitoring 1 R to 1,000 R for accident readings.

Thermoluminescence

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. There are two categories of thermoluminescence.

Fluorescence This is emission of light during or immediately after irradiation (within fractions of a second) of the phosphor. This is not a particularly useful reaction for TLD use.

Phosphorescence This is the emission of light after the irradiation period. The delay time can be from a few seconds to weeks or months. This is the principle of operation used for thermoluminescent dosimeters.

The property of thermoluminescence of some materials is the main method used for personnel dosimeters at DOE facilities.

2.04.05 Determine the theory of operation of a thermoluminescent dosimeter (TLD).

TLD OPERATION

TLDs use phosphorescence as their means of detection of radiation.

Electrons in some solids can exist in two energy states, called the valence band and the conduction band. The difference between the two bands is called the band gap. Electrons in the conduction band or in the band gap have more energy than the valence band electrons. Normally in a solid, no electrons exist in energy states contained in the band gap. This is a "forbidden region."

In some materials, defects in the material exist or impurities are added that can trap electrons in the band gap and hold them there. These trapped electrons represent stored energy for the time that the electrons are held. (See figure 1) This energy is given up if the electron returns to the valence band.

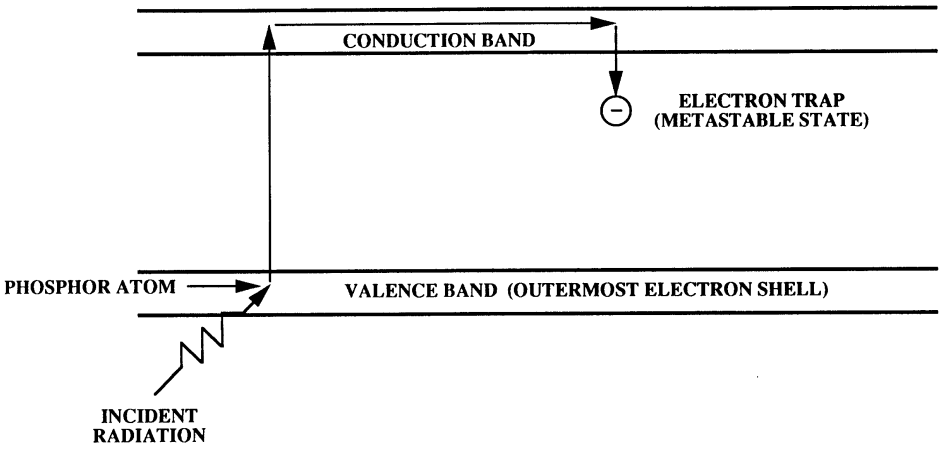

Figure 1 - Electron Entrapment

In most materials, this energy is given up as heat in the surrounding material, however, in some materials a portion of energy is emitted as light photons. This property is called luminescence. (See figure 2)

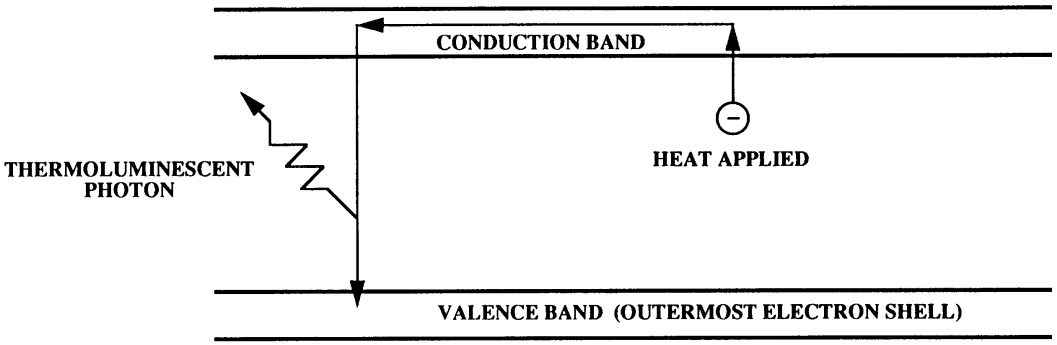

Figure 2 - Thermoluminescence

2.04.06 Determine how a TLD reader measures the radiation dose from a TLD.

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: (See figure 3)

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

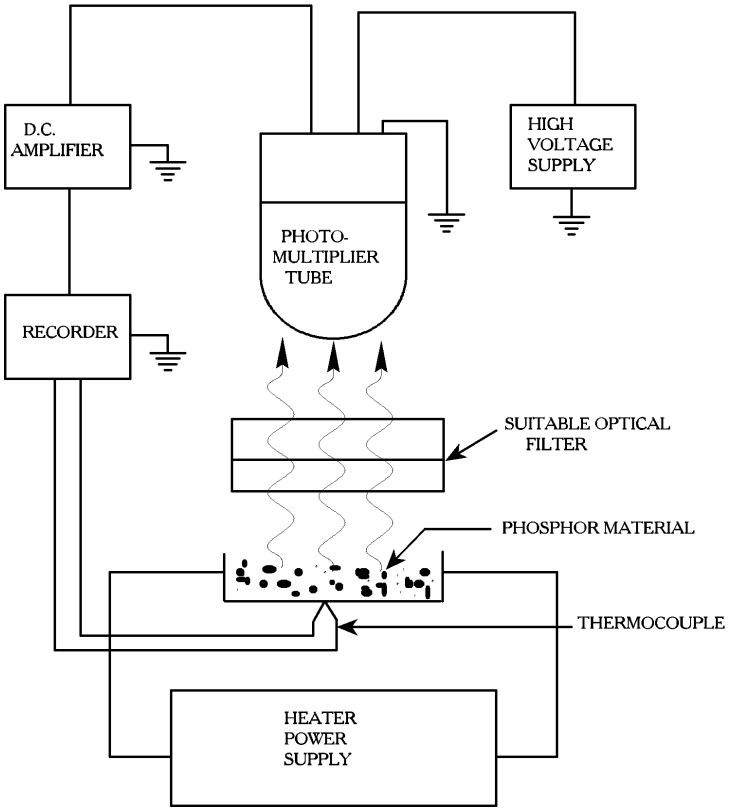

Figure 3 - TLD Reader

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. The light output drops off as these traps are depleted. As heating continues, the electrons in deeper traps are released. This results in additional peaks. Usually the highest peak is used to calculate the dose equivalent. The area under the curve represents the radiation energy deposited on the TLD. A simple glow curve is shown in figure 4.

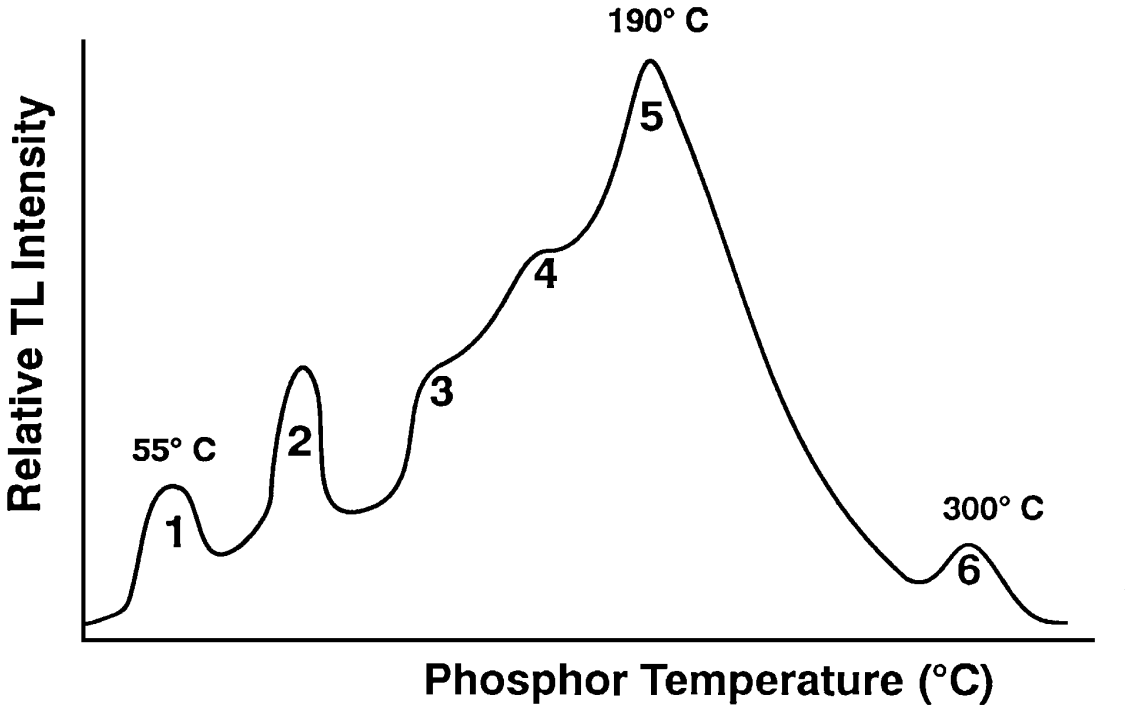

Figure 4 - Glow Curve

After the readout is complete, the TLD is annealed at a high temperature. This process essentially zeroes the TL material by releasing all trapped electrons. The TLD is then ready for reuse.

2.04.07 Identify the advantages and disadvantages of a TLD compared to a film badge.

ADVANTAGES AND DISADVANTAGES OF TLDs

Advantages (as compared to film dosimeter badges) includes:

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

Disadvantages

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

2.04.08 Identify the types of beta-gamma TLDs used at your site.

SITE BETA/GAMMA TLDs

(Insert site specific material here)

2.04.09 Identify the types of neutron TLDs used at your site.

SITE NEUTRON TLDs

(Insert site specific material here)

DOE EXTERNAL DOSIMETRY REQUIREMENTS

Personnel dosimetry should be provided to and used by individuals as follows:

1. General employees who are expected to receive a deep dose equivalent to any portion of the whole body of 0.1 rem (0.001 sievert) or more in a year or a dose equivalent to the extremities, or organs and other tissues (including lens of the eye and skin) of 10 percent or more of the corresponding limits *[835.402(a)(1)]*;

2. Declared pregnant workers who are expected to receive from external sources a dose equivalent of 0.05 rem (0.0005 sievert) or more to the embryo/fetus during the gestation period *[835.402(a)(2)]*;

3. Occupationally exposed minors likely to receive from external sources an exposure in excess of 50% of the limits *[835.402(a)(3)]*;

4. Members of the public who enter the controlled area and are likely to receive an external deep dose equivalent of 0.05 rem (0.0005 sievert) or more in a year *[835.402(a)(4)]*; and

5. Individuals entering a high or very high radiation area radiation area *[835.402(a)(5)]*.

Neutron dosimetry shall be provided when an individual is likely to exceed the applicable threshold provided above due to neutron radiation *[835.402(b)]*.

Dosimeters should be issued only to individuals knowledgeable of their proper use and worn only by those to whom the dosimeters were issued.

To minimize the number of individuals in the dosimetry program, the issuance of dosimeters is discouraged to other than individuals entering radiological areas where there is a likelihood of external exposure in excess of the monitoring thresholds established in Article 511.1 of the Radiological Control Standard. Although issuing dosimeters to individuals who are not occupationally exposed to radiation can appear as a conservative practice, it creates the impression that the wearers are occupationally exposed to radiation. Implementation of an unnecessarily broad dosimetry program is not an acceptable substitute for development of a comprehensive workplace monitoring program.

Individuals should return dosimeters for processing as scheduled or upon request, and should be restricted by line management from continued radiological work until dosimeters are returned.

Individuals should wear their primary dosimeters on the chest area, on or between the waist and the neck, or in the manner prescribed by radiological control procedures or work authorizations.

Film dosimeters should not be worn or taken off-site unless specifically authorized by the Radiological Control Manager or designee.

The practice at some facilities of taking thermoluminescent dosimeters (TLDs) off-site is discouraged and should not be implemented where not in place.

Individuals should not wear dosimeters issued by their resident facilities while being monitored by a dosimeter at another facility unless authorized by the Radiological Control Manager or designee. Individuals should not expose their dosimeters to security X-ray devices, excessive heat, or medical sources of radiation.

An individual whose dosimeter is lost, damaged, or contaminated should place work in a safe condition, immediately exit the area, and report the occurrence to the Radiological Control Organization. Reentry of the individual into radiological areas should not be made until a review has been conducted and management has approved reentry.

2.04.10 Determine the requirements for use of TLDs used at your site.

SITE REQUIREMENTS FOR USE OF TLDs

(Insert site specific material here)

2.04.11 Determine the principle of operation, and the types used, for the personnel neutron dosimeters used at your site.

SITE PERSONNEL NEUTRON DOSIMETERS

(Insert site specific material here)

POCKET AND ELECTRONIC DOSIMETERS

Pocket and electronic dosimeters are supplemental dosimeters that provide real-time indication of exposure to radiation and assist in maintaining personnel doses less than Administrative Control Levels.

Supplemental dosimeters shall be issued to personnel prior to entry into a High or Very High Radiation Area *[835.502(a)(2)]*. Supplemental dosimeters should also be issued when planned activities could cause an individual to exceed 50 millirem or 10 percent of a facility Administrative Control Level from external radiation in 1 work day, whichever is greater or when required by a Radiological Work Permit. Pocket dosimeters should be selected with the lowest range applicable (typically 0-200 mR) for anticipated personnel exposures.

Supplemental dosimeters should be worn simultaneously with the primary dosimeter and located on the chest area, on or between the waist and the neck.

Supplemental dosimeters should be read periodically while in use and should not be allowed to exceed 75 percent of full scale.

Work authorized by written authorization should be stopped when supplemental dosimeter readings indicate total exposure or rate of exposure substantially greater than planned. The Radiological Control Organization should be consulted prior to continuation of work.

The energy dependence of supplemental dosimeters, particularly to low-energy beta radiation, should be considered in determining their applicability. For example, the SRPD (shown in figure 5) has a thick case that effectively shields most betas.

Use of electronic dosimeters is encouraged for entry into High Radiation Areas or when planned doses greater than 0.1 rem (0.001 sievert) in 1 work day are expected. An electronic dosimeter provides an early warning of elevated exposure through the use of alarm set points at specified dose rates or integrated doses.

When the dose results from the pocket or electronic dosimeters differ by more than 50 percent from the primary dosimeter result and the primary dosimeter result is greater than 0.1 rem (0.001 sievert), an investigation should be initiated to explain the difference.

2.04.12 Determine the principle of operation of self-reading dosimetry (SRD) used at your site.

SITE SELF-READING DOSIMETERS

(Insert site specific material here)

Self Reading Pocket Dosimeters (SRPD)

The direct reading pocket dosimeter consists of an ionization chamber sensitive to a desired radiation; a quartz fiber electrometer to measure the charge; and a microscope to read the fiber image off a scale (reticle). (See figure 5)

The electrometer embodies two electrodes, one of which is a moveable quartz fiber and the other a metal frame. When the electrometer is charged to a predetermined voltage, the electrodes assume a calibrated separation.

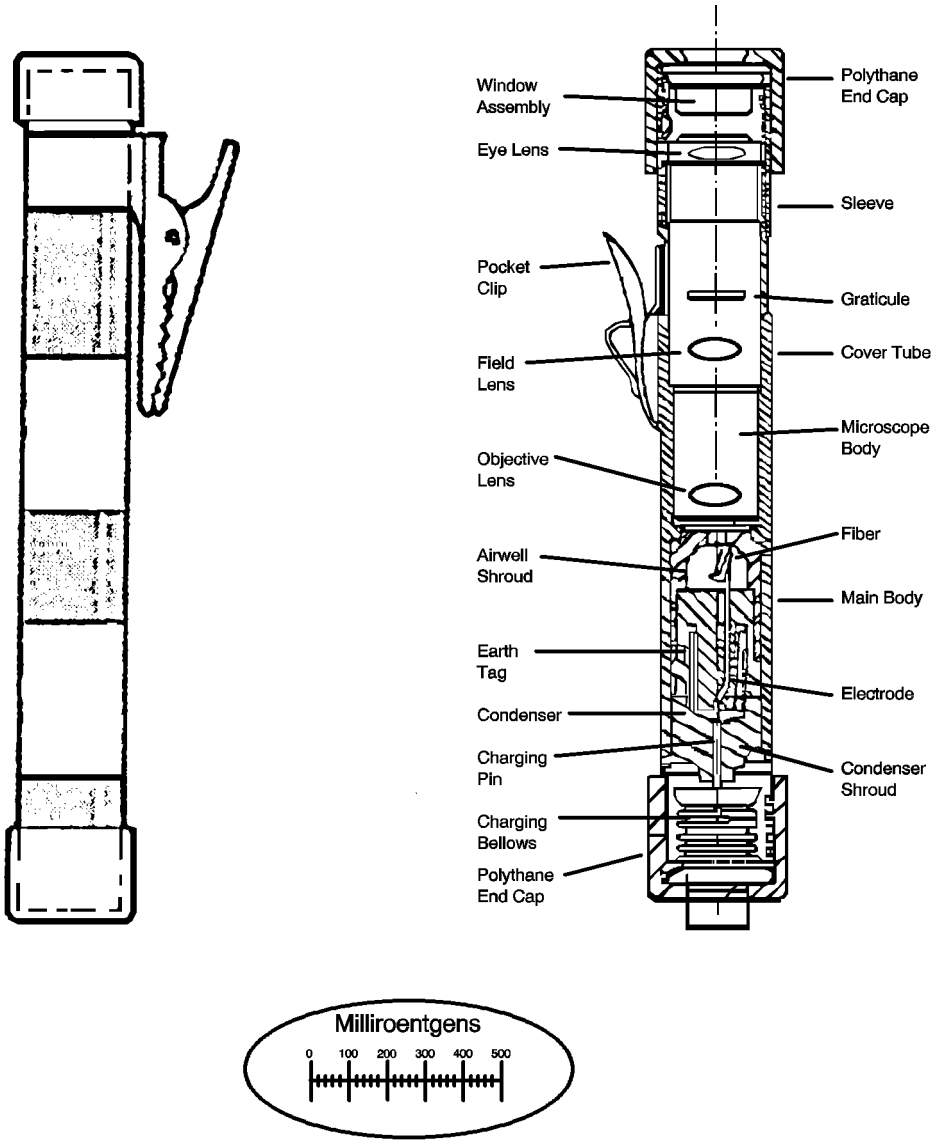

Figure 5 - SRPD

As the dosimeter is exposed to radiation, ionization occurs in the surrounding chamber decreasing the charge on the electrode in proportion to the exposure. The deflection of the moveable quartz fiber electrode is projected by a light source through an objective lens to a calibrated scale and read through a microscope eyepiece. (See Figure 6)

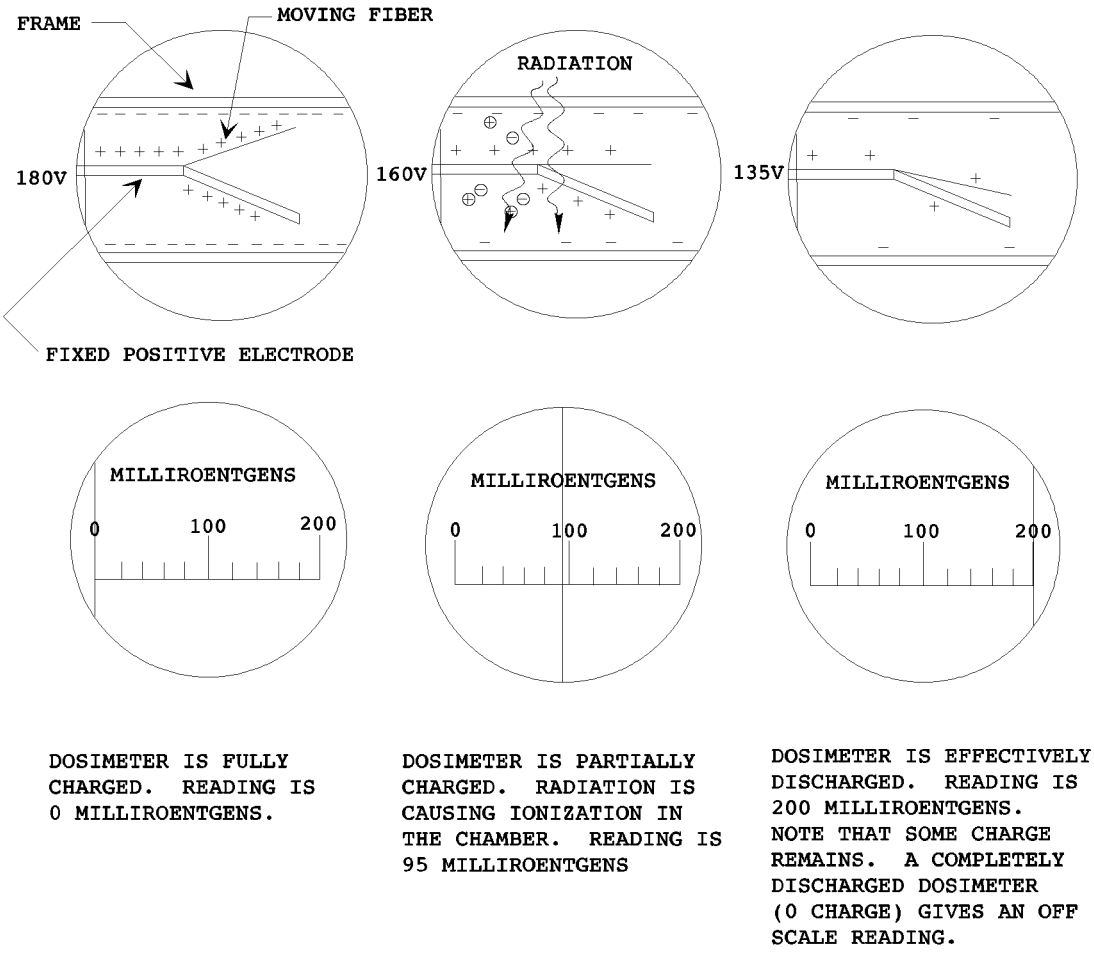

Figure 6 - SRPD Reading

Illumination for the optical system is obtained by pointing the dosimeter at any convenient light source. Light passes through the clear glass bottom seal to illuminate the scale.

The bottom is sealed by a bellows containing an insulated charging pin. When charging, the charging pin moves up to contact the electrometer closing the circuit. Sufficient voltage is applied to recharge the system. The entire dosimeter system is hermetically sealed in a protective barrel.

SITE ALARMING DOSIMETRY

2.04.13 *Determine the principle of operation, and guidelines for use, for the alarming dosimeters used at your site.*

(Insert site specific material here)

INTERNAL DOSIMETRY REQUIREMENTS

Per 10 CFR 835: for the purpose of monitoring individual exposures to internal radiation, internal dose evaluation programs (including routine bioassay programs) shall be conducted for:

1. General employees 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;

2. Declared pregnant workers likely to receive an intake resulting in a dose equivalent to the embryo/fetus in excess of 10 percent of the limit (or 0.05 rem [0.0005 sievert]);

3. Occupationally exposed minors who are likely to receive a committed effective dose equivalent in excess of 50 percent of the applicable limit (or 0.05 rem [0.0005 sievert]) from all radionuclide intakes in a year;

4. Members of the public entering a controlled area likely to receive a dose equivalent in excess of 50 percent of the limit (or 0.05 rem [0.0005 sievert]) from all radionuclide intakes in a year.

The estimation of internal dose should be based on bioassay data rather than air concentration values unless bioassay data are unavailable, inadequate, or internal dose estimates based on air concentration values are demonstrated to be as or more accurate.

Personnel should participate in follow-up bioassay monitoring when their routine bioassay results indicate an intake in the current year with a committed effective dose equivalent of 0.1 rem (0.001 sievert) or more.

Personnel whose routine duties may involve exposure to surface or airborne contamination or to radionuclides readily absorbed through the skin, such as tritium, should be considered for participation in the bioassay program.

Personnel should submit bioassay samples, such as urine or fecal samples, and participate in bioassay monitoring, such as whole body or lung counting, at the frequency required by the bioassay program.

Personnel should be notified promptly of positive bioassay results and the results of dose assessments and subsequent refinements. Dose assessment results should be provided in terms of rem or mrem.

BIOASSAY ASSESSMENT METHODS

Today's technology has not produced a device that allows accurate determination of internal exposure following the entry of radioactive materials into the body.

The method that is 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. Thus, the real problem becomes one of quantifying the amount of material present.

Bioassay is the term that is used to describe the assessment of the quantity of radioactive material present in the body. There are currently two types of bioassay measurements employed in nuclear industries: in vivo and in vitro. In vivo bioassay involves counting the living tissue, as described below. In vitro involves counting an excreted sample, such as urine.

Bioassay programs are designed to fulfill two needs:

1) Evaluate effectiveness of contamination control practices

    - Routine bioassay programs utilize submission and analysis of samples from workers in facilities where the likelihood of intake exists

    - Primarily limited to urinalysis due to ease of sample collection

    - Also includes initial, routine, and termination whole body counts

2) Evaluate potential consequences of accidental inhalation or ingestion of large quantities of radioactive materials

    - Can involve all types of bioassay measurements with collection and analysis of nasal, urine, and fecal samples.

    - Whole body counts provide immediate indications for given radionuclides if individual(s) involved are free of external contamination.

Quantification of materials actually in the body can be affected by the availability of measurements taken early after the incident. The elimination rate of some materials from the body falls off as the concentration in the body falls off, or with time. Accurate quantification of initial quantities, present, thus accurate dose assessment, can be dependent on availability of early data.

Identification of the proper bioassay technique to use is aided by a knowledge of the types of contamination present in a particular work area. For example, if you know that the contamination in a facility typically includes radionuclides that cannot be detected with in vivo measurements, then it would be obvious that collection and measurement of urine or other samples is necessary.

If the presence of gamma emitting nuclides is identified, consider the possibility of the presence of materials that do not decay with gamma emission. Periodic radionuclide assessment of contamination in facilities will provide information on relative radionuclide concentrations. Caution must be exercised in using information of this nature. Cycles of contamination should be used as an indicator only. Remember, fresh coolant does not have the same isotopic makeup of coolant that has decayed.

Contamination control measures cannot be too stringent during collection, handling, and analysis of bioassay samples. Cross-contamination can cause erroneous assumptions and inaccurate dose assessments. If procedural guidance is not sufficient to determine required actions, consult supervision.

In Vivo Measurements

In vivo techniques consist of direct measurements of gamma or X-radiation emanating from the body. This method is very useful for any radionuclide which emits (or has daughters which emit) photons of sufficient energy to escape the body. The photon flux density must be large enough for measurement in a reasonable time period, even though the quantity of material in the organ is very small.

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. If the counter has been calibrated previously, one may rapidly determine the identity and amount of any of these radionuclides. Such measurements are more acceptable to the subject than the provision of samples of excreta, although they do require him to be absent from work during the period of measurement. Direct counting of the individual without preparation beforehand (changing into clean clothes and external decontamination) may give misleading results, since this method measures all gamma emitting radionuclides in or on a subject; therefore, sensitive counts (lung) should be done immediately after the subject washes and changes into clean clothing. Radon daughters that cling to body hair due to their electrostatic charge are the chief source of bad lung counts.

When this method errs, it usually does so by being too high, so that a negative result is likely to be a reliable indication that there is no internal contamination with gamma emitters.

In external counting, the requirement for sensitivity and energy discrimination determines the complexity of the measuring equipment. Estimations of very small quantities require elaborate shielding of both the sensing element and the subject, sensitive detectors, and the best discrimination between gamma ray energies. However, a single moderately large, well-shielded sodium iodide crystal coupled with a multichannel analyzer can usually meet the need. This system in conjunction with a shielded chair or moving bed, is capable of determining:

- I-131 in the thyroid gland.
- Insoluble radionuclides in the chest.
- Insoluble radionuclides in the intestine.
- Insoluble radionuclides in wounds.

These need not emit highly penetrating radiation, since much of the material may be on or near the surface, i.e., for wounds.

Because large sodium iodide crystals do not have good collimation capabilities, it is usually not possible to measure specific organ contents directly. In some cases, solid state detector (GeLi) can be used for specific organ determination. However, the decreased sensitivity of this method limits the usefulness of these measurements. Small sodium iodide detectors are used for determining thyroid dose.

2.04.14 List the types of bioassay monitoring methods at your site.

Site In Vivo Methods

(Insert site specific material here)

Advantages of In Vivo Measurements

- No sample required
- Results obtained quickly
- Some equipment design allows field use
- Time and manpower requirements minimized.

Disadvantages of In Vivo Measurements

- Limited to detection and measurement of gamma emitters
- Individual must be free of external contamination

- Long count times for identification
- Effects of background
- Complex calibration procedure and calibration equipment
- Expense
- Quantification error due to differences in tissue structure from one person to another as compared to calibration phantom.

In Vitro Measurements

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

Resulting dose calculations to quantify committed and effective dose equivalents are estimates. This is due partly to use of default values for measurements that cannot be readily made such as mass of particular organs, volumes of particular fluids, etc., in lieu of actual values for individual involved. Remember that reference man is an average. Another contributing factor is the difference in metabolism from one individual to another.

Urinalysis

Indicates effectively that soluble radioactive material has been deposited in the blood for transport to various organs. A fraction of the material is normally removed from the blood by the kidneys and excreted. Later, material absorbed by various organs may be released to the blood through biological exchange processes, and then may be excreted in the urine.

Certain compounds are determined to be insoluble because they are avidly retained in the lung. However, they also eventually appear in the urine. Particles are removed to the pharynx by the ciliary-mucus transport mechanism where they are swallowed, dissipated and partially absorbed in the gastrointestinal tract for transport to the blood. Other particles are removed by transport to the lymphatic system for subsequent release to the blood. Other particles slowly enter into a physical or chemical state which allows direct transport from the pulmonary region of the lung to the blood. All three cases lead to urinary excretion of the material.

Taking samples of urine involves two special difficulties. One is the possibility of contamination if the sample is taken at work. The other is the problem of collecting a sample from which can be calculated the total excretion of radionuclide per unit time, usually per day. It is ordinarily not convenient to collect a full 24-hr sample of urine, so it is frequently necessary to estimate the fraction this is of the relatively constant daily urine excretion.

One of the advantages of measuring the radionuclide content of urine is that if a radionuclide is found in a carefully collected sample of urine, there can be no doubt that it was in extracellular body fluids. Furthermore, under the most favorable conditions, the amount of daily urinary excretion of radionuclide may be used directly to calculate total body content.

One of the simplest examples of practical importance is tritium oxide which is present in the same concentration in urine as in extracellular fluids of the body.

Almost all employees are willing to provide a limited number of urine samples; however, prolonged urine sampling involving samples taken both at home and at work will often meet with increasing employee resistance.

Fecal Analysis

An appreciable fraction of the particles entering the gastrointestinal tract may not be absorbed; these appear in the feces within twenty-four hours. Thus, fecal analysis is an excellent and relatively rapid indicator that an exposure has occurred. Fecal analysis is particularly useful for inhaled, insoluble materials that do not appear in the urine for weeks. For many highly insoluble materials, particles remaining in the pulmonary system continue to reach the mucus blanket, although at a greatly reduced rate. These particles are then transported by ciliary action to the gastrointestinal tract. Thus, fecal analysis can also contribute to the estimate of the lung burden.

Two drawbacks to fecal analysis are: (1) there is considerable employee resistance to provide fecal samples and (2) there is very little correlation between fecal content and organ depositions. Thus, fecal analysis is primarily a qualitative method used only for detecting the intake of insoluble materials and providing indication of clearance of such materials from the lungs. Fecal sampling is normally done immediately following an incident because correlation is best when intake times are known.

Sputum

When obtainable, sputum may contain insoluble material initially deposited in the lung and later eliminated by ciliary action. However, clearance time for sputum is very rapid and samples must be taken immediately after an incident.

Saliva

May be analyzed to detect internal contamination, but the only practical case in which saliva can be used to estimate body content is that of tritium oxide, for which urine is the usual method.

Nasal Discharge

The presence of radionuclides in nasal discharge and nasal swabs generally gives an indication of the deposition of the coarsest inhaled particles in the nose. Measurement of the amounts present cannot always be used for quantitative estimation of the amount in the body, but it can be useful in detecting significant exposures and identifying the radionuclide involved in an accident.

2.04.14 List the types of bioassay monitoring methods at your facility.

Site In Vitro Methods

(Insert site specific material here)

Advantages of In Vitro Measurements

- Can be used for estimation of neutron doses using activation product concentration in hair and blood (P^{32} and Na^{24})
- Can be used to quantify presence of materials which decay by alpha and beta emission to allow detection and measurement with external detector systems.

Disadvantages of In Vitro Measurements

- Requires sample submission and analysis
- Time and manpower requirements

BIOASSAY SCHEDULING PROGRAM

Contamination found in a given facility will depend on the materials that are used and produced in the facility. Thus, the materials that internal dosimetrists are primarily concerned with will change from one site to another as well.

Baseline/Routine/Exit Evaluations

(Insert site specific material here)

Special Evaluations

(Insert site specific material here)

Investigation Levels

(Insert site specific material here)

Medical Uses

(Insert site specific material here)

SUMMARY

The method of operation of dosimeters is a vital knowledge for RCT. RC personnel are the first line of defense against abuse of these instruments and must ensure the proper wearing and use of them.

Internal exposure involves a source (contaminant) inside the body. It is more difficult to measure; sophisticated whole body counters or indirect measurements of excreta samples are required to obtain an estimate. The exposure from the contaminant does not stop when the person leaves the radiation field and the contaminant continues to irradiate tissue all day and all night. If necessary, medical treatment is required to enhance the removal of the source material from the body. Alpha radiation poses the biggest problem.