National Nuclear Security Administration

Radiation Protection

Qualification Standard

Reference Guide

MARCH 2009

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ACRONYMS, ABBREVIATIONS and SYMBOLS	
A	mass number
AEA	Atomic Energy Act
ALARA	as low as reasonably achievable
AMAD	activity median aerodynamic diameter
amu	atomic mass unit
ANL	Argonne National Laboratory
ANSI	American National Standards Institute
ARAC	atmospheric release advisory capability
ARM	area radiation monitor
Bq(s)	becquerel(s)
CAM	continuous air monitor
CBA	cost-benefit analysis
CED	committed effective dose
CERCLA	Comprehensive Environmental Response, Compensation, and Liability Act
CNS	central nervous system
cpm	counts per minute
CSO	Cognizant Secretarial Officer
CSU	combined standard uncertainty
DAC(s)	derived air concentration(s)
DCG	derived concentration guide
DCGL	derived concentration guideline level
D&D	decontamination and decommissioning
dpm	disintegrations per minute
DSA	documented safety analysis
DTPA	diethylene triamine pentaacetic acid
E_{max}	maximum value of decay energy
EMC	elevated measurement comparison
EDTA	ethylenediaminetetraacetic acid
E.O.	Executive Order
ERPG	emergency response program guide
ES&H	environmental, safety and health
FET	field effect transistor
FGR	Federal guidance report (EPA)
FRMAC	Federal Radiological Monitoring and Assessment Center
FWHM	full-width at maximum height
Ge(Li)	germanium-lithium
GERT	general employee radiological training
GI	gastrointestinal
G-M	Geiger-Müller (Geiger-Mueller)
HLW	high-level waste
HVL	half-value layer
HTO	tritium oxide
IAEA	International Atomic Energy Agency
ICRP	International Commission on Radiation Protection

IDG	Internal Dosimetry Program Guide
IDLH	immediately dangerous to life or health
LET	linear energy transfer
LLD	lower limit of detection
LLW	low-level waste
MDA	minimum detectable activity
M&E	materials and equipment
mR or mrem	millirem ($1x10^{-3}$ rem)
n:p	neutron-to-proton (ratio)
NCRP	National Council on Radiation Protection
NCS	nuclear criticality safety
NDA	non-destructive assay
NEPA	National Environmental Protection Act
NIM	nuclear instrument module
NIOSH	National Institute of Occupational Safety and Health
NORM	naturally occurring radioactive material
ORR(s)	operational readiness review(s)
PAS	personal air sampler
QA	quality assurance
QAPP	quality assurance project plan
R	roentgen
RCTP	Radiological Control Technical Positions
RCRA	Resource Conservation and Recovery Act
RCS	Radiological Control Standard
RPP	radiation protection program
RSSI	radiological survey and site investigation
RWPs	radiological (radiation) work permit(s)
RWT	radiological worker training
S&M	surveillance and maintenance
SARs	safety analysis report(s)
SNF	spent nuclear fuel
SNM	special nuclear material
Sv	sievert
TLD	thermoluminescent dosimeter
TSCA	Toxic Substance Control Act
TSR	technical safety requirement
USQ	unreviewed safety question
Z	atomic number
α	alpha (particle)
β^+	beta plus (positron)
β^-	beta minus (negatron)
λ	radioactive decay constant
μ	linear attenuation coefficient

PURPOSE

The purpose of this reference guide is to provide a document that contains the information required for a Department of Energy (DOE)/National Nuclear Security Administration (NNSA) technical employee to successfully complete the Radiation Protection Functional Area Qualification Standard (FAQS). Information essential to meeting the qualification requirements is provided; however, some competency statements require extensive knowledge or skill development. Reproducing all the required information for those statements in this document is not practical. In those instances, references are included to guide the candidate to additional resources.

SCOPE

This reference guide has been developed to address the competency statements in the December 2003 edition of DOE-STD-1174-2003, *Radiation Protection Functional Area Qualification Standard*. The qualification standard for Radiation Protection contains 27 competency statements. However, this reference guide covers only the first 21 – the remainder relate to program and project management experience which should be covered through competency in the General Technical Base Qualification Standard. Therefore, the scope is limited to the scientific and technical topics in the Standard.

Please direct your questions or comments related to this document to the Learning and Career Development Department.

PREFACE

Competency statements and supporting knowledge and/or skill statements from the qualification standard are shown in contrasting bold type, while the corresponding information associated with each statement is provided below it.

A comprehensive list of acronyms, abbreviations, and symbols is provided at the beginning of this document. It is recommended that the candidate review the list prior to proceeding with the competencies, as the acronyms, abbreviations, and symbols may not be further defined within the text unless special emphasis is required.

The competencies and supporting knowledge, skill, and ability (KSA) statements are taken directly from the FAQS. Most corrections to spelling, punctuation, and grammar have been made without remark, and all document-related titles, which variously appear in roman or italic type or set within quotation marks, have been changed to plain text, also mostly without remark. Capitalized terms are found as such in the qualification standard and remain so in this reference guide. When they are needed for clarification, explanations are enclosed in brackets.

Every effort has been made to provide the most current information and references available as of March 2009. However, the candidate is advised to verify the applicability of the information provided. It is recognized that some personnel may oversee facilities that utilize predecessor documents to those identified. In those cases, such documents should be included in local qualification standards via the Technical Qualification Program.

In the cases where information about an FAQS topic in a competency or KSA statement is not available in the newest edition of a standard (consensus or industry), an older version is referenced. These references are noted in the text and in the bibliography.

Only significant corrections to errors in the technical content of the discussion text source material are identified. Editorial changes that do not affect the technical content (e.g., grammatical or spelling corrections, and changes to style) appear without remark.

TECHNICAL COMPETENCIES

1. **Radiation protection personnel shall demonstrate a working level knowledge of the various types of radiation and how they interact with matter.**

   a. **Describe each of the following forms of radiation in terms of structure, mass, origin, and electrostatic charge:**
      - **Alpha**
      - **Beta**
      - **Neutron**
      - **Gamma**
      - **X-ray**

The following is taken from DOE-HDBK-1122-99.

Alpha

Alpha radiation is particulate radiation emitted from the nucleus of an unstable atom. The alpha particle has a mass of 4 atomic mass units (amu) and consists of 2 protons and 2 neutrons. Since the alpha particle has 2 protons, it has a positive charge of + 2. With few exceptions, only relatively heavy radioactive nuclides with an atomic number (Z) less than 82 will decay by alpha emission. Alpha radiation is monoenergetic, meaning its emissions are at discrete energies. The symbol (α) is used to designate alpha particles.

A nucleus emitting an alpha particle decays to a daughter element, reduced in Z by 2 and reduced in mass number (A) by 4. The standard notation for alpha decay is:

$$^{A}_{Z}X \rightarrow ^{A-4}_{Z-2}Y + ^{4}_{2}\alpha$$

For example, ^{226}Ra decays by alpha emission to produce ^{222}Rn as follows:

$$^{226}_{88}Ra \rightarrow ^{222}_{86}Rn + ^{4}_{2}\alpha$$

Alpha particles are the least penetrating of the three types of particulate radiation (alpha, beta, and neutron). They can be absorbed or stopped by a few centimeters of air or a sheet of paper.

Beta

Beta radiation is particulate radiation in the form of an electron emitted from the nucleus of an unstable atom. The beta particle has a mass of 0.000548 amu and can have a negative charge of -1, a beta minus particle or negatron (β^-), or a positive charge of +1, a beta plus particle or positron (β^+).

Atoms that emit β^- radiation do so because the nucleus has an excess number of neutrons. An excess number of neutrons can occur when atoms are bombarded with neutrons (called activation), or when heavy atoms fission to produce neutron-rich fission fragments. In β^- emitters, the nucleus of the parent gives off a negatively charged particle, resulting in a daughter more positive by one unit of charge. It has been postulated that a β^- particle is

formed by the transformation of a neutron into a proton and electron. Because a neutron has been replaced by a proton, the Z number increases by one, but the A number is unchanged.

There is also the emission of a neutrino or an antineutrino (ΰ) which are neutral (uncharged) particles with negligible rest mass. They travel at the speed of light, and are very non-interacting. They account for the energy distribution among positrons and beta particles from given radionuclides in the positron and beta decay processes respectively. Because they are so non-interacting, no energy is deposited in tissue, and therefore, no dose results to personnel exposed to neutrinos.

The standard notation for β decay is:

$$^{A}_{Z}X \rightarrow ^{A}_{Z+1}Y + \beta^{-} + \acute{\upsilon}$$

For example, ^{210}Pb decays by beta-minus emission to produce ^{210}Bi as follows:

$$^{210}_{82}Pb \rightarrow ^{210}_{83}Bi + \beta^{-} + \acute{\upsilon}$$

Beta particles are emitted in a spectrum of kinetic energies ranging up to the maximum value of the decay energy (E_{max}). The average energy of beta particles is about $1/3E_{max}$. They travel several hundred times the distance of alpha particles in air (approximately 10 feet per MeV) and require a few millimeters of aluminum to stop them.

The following is taken from *Radiation Detection and Measurement*, by Glenn F. Knoll.

When the parent nucleus undergoes β^{+} decay, additional electromagnetic radiation is generated. The origin lies in the fate of the positrons emitted in the primary decay process. Because they generally travel only a few millimeters before losing their kinetic energy, the inherent encapsulation around the source is often sufficiently thick to fully stop the positrons. When their energy is very low, near the end of their range, they combine with normal negative electrons in the absorbing materials in the process of annihilation. The original positron and electron disappear and are replaced by two oppositely directed 0.511 MeV electromagnetic known as annihilation radiation. This radiation is then superimposed on whatever gamma radiation may be emitted in the subsequent of the daughter product.

The following is taken from DOE-HDBK-1019/1-93.

Neutron

Neutrons have no electrical charge. They have nearly the same mass as a proton (a hydrogen atom nucleus). A neutron has hundreds of times more mass than an electron, but 1/4 the mass of an alpha particle. The source of neutrons is primarily nuclear reactions, such as fission, but they may also be produced from the decay of radioactive nuclides. Because of its lack of charge, the neutron is difficult to stop and has a high penetrating power.

Gamma and X-ray

Gamma radiation is electromagnetic radiation. It is commonly referred to as a gamma ray and is very similar to an x-ray. The difference is that gamma rays are emitted from the nucleus of an atom, and x-rays are produced by orbiting electrons. The x-ray is produced when orbiting electrons move to a lower energy orbit or when fast-moving electrons approaching an atom are deflected and decelerated as they react with the atom's electrical field (called Bremsstrahlung). The gamma ray is produced by the decay of excited nuclei and by nuclear reactions. Because the gamma ray has no mass and no charge, it is difficult to stop and has a very high penetrating power. A small fraction of the original gamma stream will pass through several feet of concrete or several meters of water.

b. Describe the interactions of the following with matter:

- **Charged particle interactions**
  - **Alpha particle**
  - **Beta particle (Positron annihilation and Bremsstrahlung)**
- **Neutron interaction**
  - **Elastic scattering**
  - **Inelastic scattering**
  - **Fission**
  - **Capture, absorption, or activation**
- **Photon interactions**
  - **Photoelectric effect**
  - **Compton scattering**
  - **Pair production**

The following is taken from DOE-HDBK-1122-99.

Charged Particle Interactions

Charged particles do not require physical contact with atoms to interact with them. The "Coulomb force" (force from the electrical charge) will act over a distance to cause ionization and excitation in the absorber medium. Particles with charge (such as alpha and beta) that lose energy in this way are called directly ionizing radiation. The strength of this force depends on:

- Energy (speed) of the particle
- Charge of the particle
- Density and atomic number (number of protons) of the absorber.

The "Coulomb force" for even a singly charged particle (an electron) is significant over distances greater than atomic dimensions (remember this is the same force that holds the electrons in bound energy states about the nucleus). Therefore, for all but very low physical density materials, the loss of kinetic energy for even an electron is continuous because the "Coulomb force" is constantly "pushing" on electrons of at least one atom and possibly many atoms at the same time.

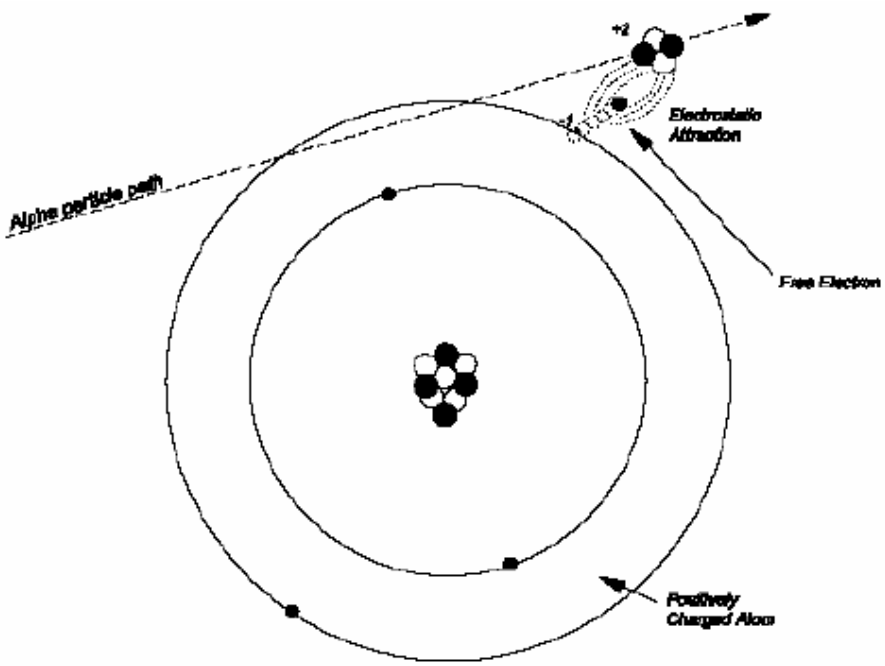
Alpha Particle

An alpha particle is made up of two protons (positively charged) and two neutrons, all strongly bound together by nuclear forces. If such a particle approaches an electron negatively charged, it experiences a strong electrostatic attraction, whereas if it approaches an atomic nucleus (also positively charged) it will tend to be repelled. Alpha particles have a

mass about 8,000 times that of an electron. They are ejected from the nuclei of radioactive atoms with velocities of the order of 1/20 the speed of light. All of these properties--its large mass, its charge, and its high velocity tend to make the alpha particle an efficient projectile when it encounters atoms of an absorbing material. In other words, it would have a high probability of interacting, or colliding, with orbital electrons, and also atomic nuclei.

When speaking of "collisions" between subatomic particles, it should be understood that the particles (for example an alpha and an electron) need approach each other only sufficiently close for Coulomb forces to interact. Such an interaction may then be referred to as a collision.

Figure 1 schematically shows such a collision, resulting in ionization. In this case, the kinetic energy of the alpha particle is decreased and shows up as a free electron with kinetic energy. The free electron's kinetic energy is less than the alpha energy loss by the amount of energy necessary to free the electron (its binding energy). Because the alpha particle is so much more massive than the electron, the alpha particle typically only loses a small fraction of its energy in any collision and travels in a relatively straight path through the material.



Source: DOE-HDBK-1122-99

Figure 1. Ionization by an alpha particle

Beta Particle

A beta particle is a free (unbound) electron with kinetic energy (e.g. a moving electron). Therefore, the rest mass and charge of a beta particle are the same as that of an orbital electron. The negatively charged electron has an anti-particle which has the same mass but a positive charge called a positron. Their masses are very much smaller than the mass of the

nuclei of the atoms making up the absorbing medium. An interaction between a positively charged beta particle or a negatively charged beta particle and an orbital electron is therefore an interaction between two charged particles of similar mass. Negatively charged beta particles and orbital electrons have like charges; therefore, they experience an electrostatic repulsion when in the vicinity of one another. Positively charged beta particles and orbital electrons have unlike charges, so they experience an electrostatic attraction when in the vicinity of one another.

Because the rest masses are equal, the interaction between either of these two beta particles and an orbital electron is similar to the collision between billiard balls.

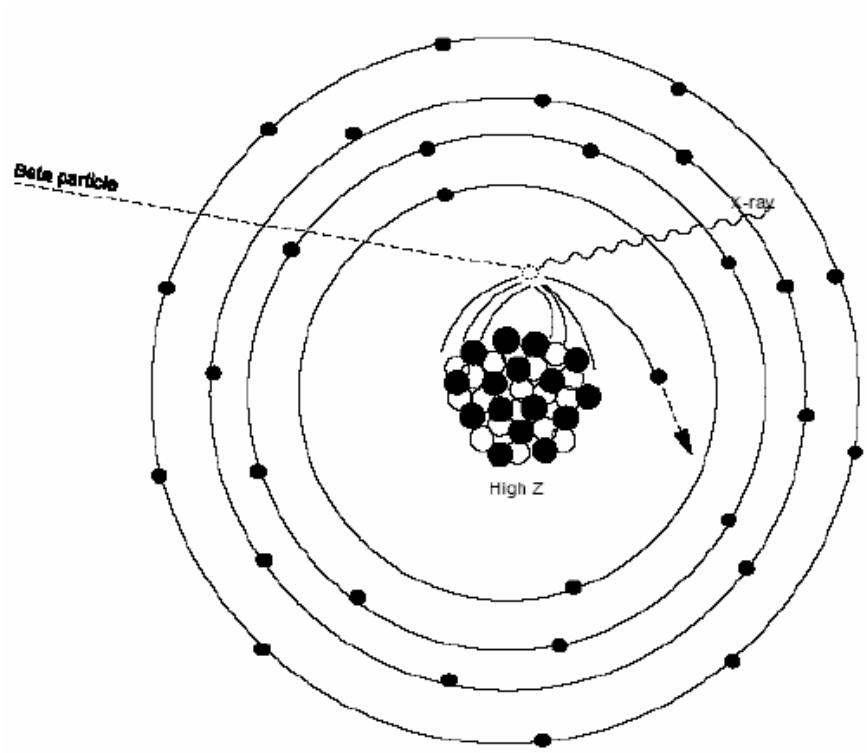
Therefore, a beta particle may lose all of its energy in a single collision. In such an interaction, the target electron acquires such high kinetic energy it becomes a particle similar to the beta particle.

Normally, however, a beta particle of either charge loses its energy in a large number of ionization and excitation events in a manner analogous to the alpha particle. Due to the smaller size and charge of the electron, however, there is a lower probability of beta radiation interacting in a given medium; consequently, the range of a beta particle is considerably greater than an alpha particle of comparable energy.

A negatively charged beta particle has a charge opposite to that of the atomic nucleus, therefore an electrostatic attraction will be experienced as the beta approaches the nucleus. A positively charged beta particle has a charge the same as that of the atomic nucleus, therefore an electrostatic repulsion will be experienced as the beta approaches the nucleus.

Since the mass of either particle is small compared with that of a nucleus, large deflections of the beta can occur in such collisions, particularly when electrons of low energies are scattered by high atomic number elements (high positive charge on the nucleus). As a result, a beta particle usually travels a tortuous, winding path in an absorbing medium.

Like an alpha particle, a beta particle may transfer energy through ionization and excitation. In addition, a beta may have a Bremsstrahlung interaction with an atom which results in the production of x-rays. Figure 2 schematically shows a Bremsstrahlung interaction. In this case, a high energy beta penetrates the electron cloud surrounding the nucleus of the atom, and experiences the strong electrostatic attractive force of the positively charged nucleus. This results in a change in velocity/kinetic energy of the particle and the emission of a Bremsstrahlung x-ray.



Source: DOE-HDBK-1122-99

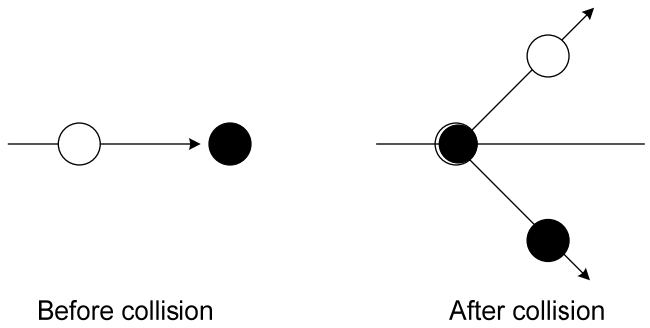
Figure 2. Bremsstrahlung radiation

Neutron Interaction

Elastic Scattering

Neutron scattering is a fourth type of interaction with the nucleus. This description is generally used when the original free neutron continues to be a free neutron following the interaction. Scattering is the dominant process for fast neutrons when the neutron is moving too fast to become a part of a nucleus. Multiple scattering by a neutron is the mechanism of slowing down or moderating fast neutrons to thermal energies.

This process is sometimes called thermalizing fast neutrons. Elastic scattering occurs when a neutron strikes a nucleus (typically of approximately the same mass as that of the neutron) as schematically shown in figure 3, on the following page. Depending on the size of the nucleus, the neutron can transfer much of its kinetic energy to that nucleus which recoils off with the energy lost by the neutron. Hydrogen causes the greatest energy loss to the neutron because the single proton in the nucleus is the same mass as the neutron. The process is analogous to the rapid dissipation of the energy of a cue ball when it hits other balls of equal mass on a billiard table. It is worth nothing that during elastic scattering reactions, no gamma radiation is given off by the nucleus. The recoil nucleus can be knocked away from its electrons and, being positively charged, can cause ionization and excitation.

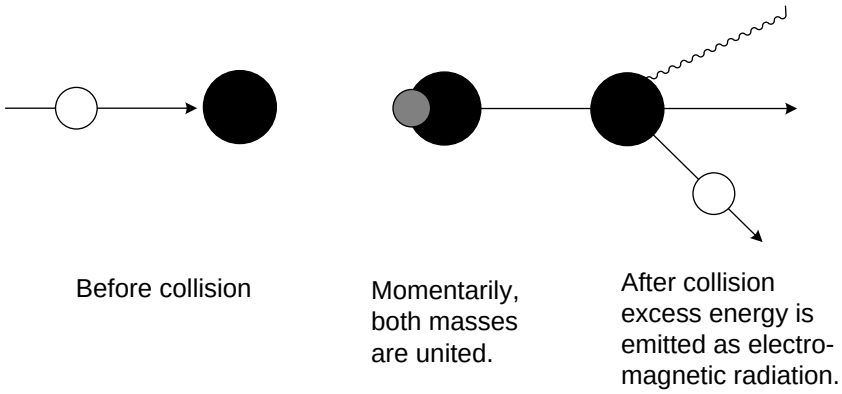


Source: DOE-HDBK-1122-99

Figure 3. Elastic scattering

Inelastic Scattering

Inelastic scattering occurs when a neutron strikes a large nucleus as schematically shown in figure 4. The neutron penetrates the nucleus for a short period of time, transfers energy to a nucleon inside, and then exits with a small decrease in energy. The nucleus is left in an excited state, emitting gamma radiation which can cause ionization and/or excitation.



Source: DOE-HDBK-1122-99

Figure 4. Inelastic scattering

Fission

A third type of neutron-induced nuclear reaction is fission. Typically, fission occurs following the absorption of a slow neutron by several of the very heavy elements. When ^{235}U nuclei undergo fission by neutrons, an average of 2 to 3 neutrons are expelled along with associated gamma radiation. The nucleus splits into two smaller nuclei which are called primary fission products or fission fragments. These products usually undergo radioactive decay to form secondary fission product nuclei. As an example, if one neutron fissions a ^{235}U nucleus, it could yield ^{95}Y, ^{139}I, two neutrons and fission energy. There are some 30 different ways that fission may take place with the production of about 60 primary fission fragments.

These fragments and the atoms which result from their decay are referred to as fission products, and they number between 400 and 600, according to the type and number of nucleons their nuclei possess.

Many fission products have found application in medicine, industry, and research. A well known example is ^{131}I which is used extensively in medicine as both a diagnostic and therapeutic agent. The fission process is the source of energy for nuclear reactors and some types of nuclear weapons. Also, neutrons generated from the fissioning of the fuel in a reactor are used to activate stable materials to a radioactive form as previously discussed. Many radioisotopes used in medicine are produced by neutron activation in this manner.

Capture, Absorption, or Activation

Radiative capture with gamma emission is the most common type of reaction for slow neutrons. This (n,Y) reaction often results in product nuclei which are radioactive. For example:

$$^{59}_{27}Co + ^{1}_{0}n \longrightarrow ^{60}_{27}Co + y$$

This process of converting a stable nucleus to its radioactive counterpart by neutron bombardment is called "neutron activation." Many radionuclides used in nuclear medicine are produced by this process.

A second type of general reaction for slow neutrons is that giving rise to charged particle emission. Typical examples include (n,p), (n,d), and (n, α) reactions, i.e., reaction in which a proton, a deuteron, or an alpha particle is ejected from the target nucleus.

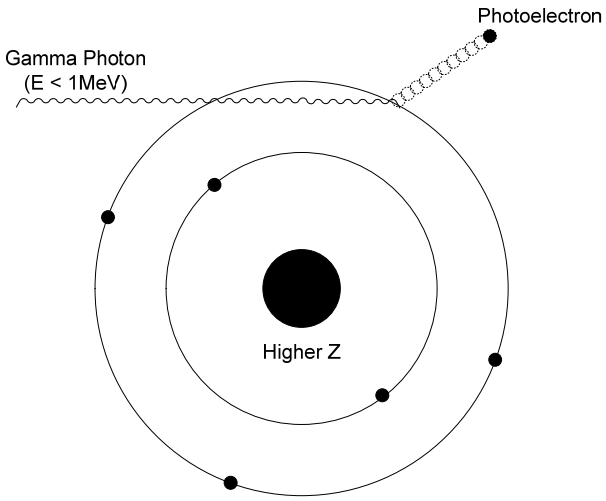
Photon Interactions

The Photoelectric Effect

In the photoelectric effect, the photon imparts all of its energy to an orbital electron of some atom. The photon, since it consisted only of energy in the first place, simply vanishes. Figure 5 schematically shows a photoelectric interaction. The energy is imparted to the orbital electron in the form of kinetic energy of motion, overcoming the attractive force of the nucleus for the electron (the binding energy) and usually causing the electron to fly from its orbit with considerable velocity. Thus, an ion pair results.

The high velocity electron, which is called a photoelectron, is a directly ionizing particle and typically has sufficient energy to knock other electrons from the orbits of other atoms. It goes on its way producing secondary ion pairs until all of its energy is expended. The probability of photoelectric effect at its maximum occurs when the energy of the photon is equal to the binding energy of the electron.

The tighter an electron is bound to the nucleus, the higher the probability of photoelectric effect, so most photoelectrons are inner-shell electrons. The photoelectric effect is seen primarily as an effect of low energy photons with energies near the electron binding energies of materials and high Z materials whose inner-shell electrons have high binding energies.

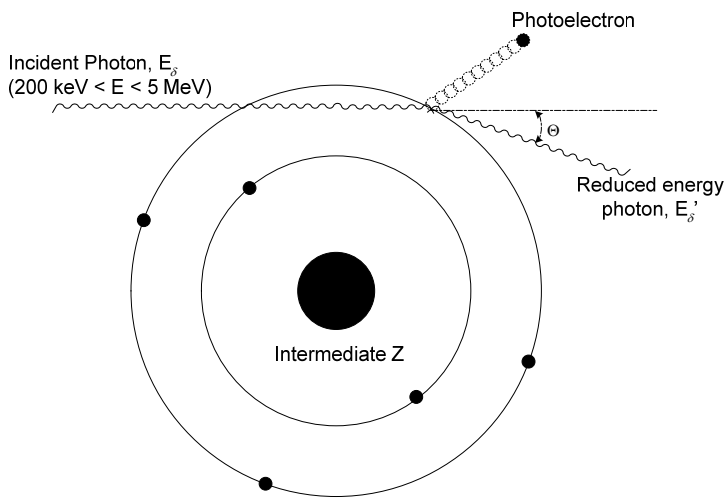


Source: DOE-HDBK-1122-99

Figure 5. Photoelectric effect

Compton Scattering

In Compton scattering there is a partial energy loss for the incoming photon. The photon interacts with an orbital electron of some atom, and only part of the photon energy is transferred to the electron. Figure 6 schematically shows a Compton scattering (also called Compton interaction).



Source: DOE-HDBK-1122-99

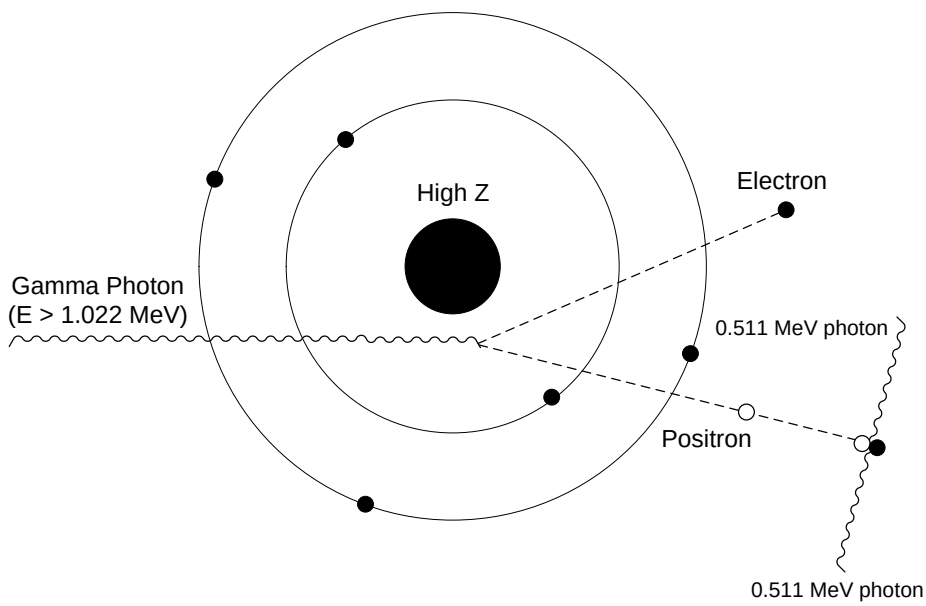
Figure 6. Compton scattering

After the collision, the photon is deflected in a different direction at a reduced energy. The recoil electron now referred to as a Compton electron, produces secondary ionization in the

same manner as does the photoelectron and the scattered photon continues on until it loses more energy in another photon interaction. By this mechanism of interaction, photons in a beam may be randomized in direction and energy so that scattered radiation may appear around corners and behind "shadow" type shields. The probability of a Compton interaction increases for loosely bound electrons and, therefore, increases proportionally to the Z of the material. Most Compton electrons are valence electrons. Compton scattering is primarily seen as an effect of medium energy photons and its probability decreases with increasing energy.

Pair Production

Pair production occurs when the photon is converted to mass. This conversion of energy to mass only occurs in the presence of a strong electric field, which can be viewed as a catalyst. Such strong electric fields are found near the nucleus of atoms and are stronger for high Z materials. Figure 7 schematically shows pair production and the fate of the positron when it combines with an electron (its anti-particle) at the end of its path.



Source: DOE-HDBK-1122-99

Figure 7. Pair production and annihilation

c. Discuss the shielding materials used for each of the above types of radiation and explain which are the best materials based on the interactions of radiation with matter.

The following is taken from DOE-HDBK-1017/2-93.

Alpha particles, being the largest particles of radiation and having a +2 charge, interact with matter more readily than other types of radiation. Each interaction results in a loss of energy.

This is why the alpha has the shortest range of all the types of radiation. Alpha particles generally are stopped by a thin sheet of paper. Because it deposits all of its energy in a very small area, the alpha particle travels only a short distance.

The beta particle is more penetrating than the alpha. However, because of the -1 charge, the beta particle interacts more readily than a non-charged particle. For this reason, it is less penetrating than uncharged types of radiation such as the gamma or neutron. The beta particle can generally be stopped by a sheet of aluminum. All materials described under neutron and gamma radiation are also effective at attenuating beta radiation.

Gamma radiation is the most difficult to shield against and, therefore, presents the biggest problem in the reactor plant. The penetrating power of the gamma is due, in part, to the fact that it has no charge or mass. Therefore, it does not interact as frequently as do the other types of radiation per given material. Gamma shielding is therefore more effectively performed by materials with high atomic mass number and high density. One such material is lead. Lead is dense and has about 82 electrons for each nucleus. Thus, a gamma would interact more times in passing through eight inches of lead then passing through the same thickness of a lighter material, such as water.

Materials containing hydrogen are known as hydrogenous material, and their value as a neutron shield is determined by their hydrogen content. Water ranks high and is probably the best neutron shield material with the advantage of low cost, although it is a poor absorber of gamma radiation.

d. Define "range" and describe the range energy relations of charged particles, including:
- **Factors that affect the range of charged particles**
- **Relative range of alpha and beta in air and tissue**

The following is taken from DOE-HDBK-1122-99.

Inversely related to the stopping power of the absorber is the range of the charged particle. The concept of range only has meaning for charged particles whose energy is kinetic energy which is lost continuously along their path. The range of a charged particle in an absorber is the average depth of penetration of the charged particle into the absorber before it loses all its kinetic energy and stops. If a particle has a high range, the absorber has a low stopping power. If the particle has a short range, the absorber has a high stopping power.

The factor that affects the range of a charged particle in any material is a unit called density-thickness. Density-thickness can be calculated by multiplying the density of a material in grams per cubic centimeter (g/cm^3) by the distance the particle traveled in that material in centimeters. The product is density-thickness in units of grams per square centimeter (g/cm^2). Density-thickness can be considered a cross-sectional target for a charged particle as it travels through the material.

The concept of density-thickness is important to discussions of beta radiation attenuation by human tissue, detector shielding/windows, and dosimetry filters. Although materials may have different densities and thicknesses, if their density-thickness values are the same, they will attenuate beta 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 beta radiation similar to the outer layer of dead skin of the human body which has a density-thickness of 7 mg/cm^2.

These values can be used to design radiation detection instrumentation such that detector windows and shields have the same, or similar, density-thickness values. For example, some instruments use a detector window of 7 mg/cm^2. Any beta radiation passing through the detector window would also pass through the outer layer of dead skin on the human body and deposit energy in living tissue of the skin. External dosimetry can be designed around these values such that dose equivalent is determined for the skin, lens of the eye, and the whole body. For example, a dosimeter filter may be designed as 1,000 mg/cm^2. Any radiation passing through this filter would deposit energy in deep tissue.

Beta particles have a range in air of about 10 feet per MeV of kinetic energy. Beta particles can penetrate the dead layer of skin if they possess more than 0.07 MeV of kinetic energy.

e. Describe the attenuation of gamma and neutron radiation in shielding materials, including:
- **Exponential attenuation**
- **Build-up**

The following is taken from DOE-HDBK-1122-99.

Exponential attenuation

When shielding against x-rays and gamma rays, it is important to realize that photons are removed from the incoming beam on the basis of the probability of an interaction (photoelectric, Compton, or pair production).

This process is called attenuation and can be described using the "linear attenuation coefficient," (μ), which is the probability of an interaction per path length (x) through a material. The linear attenuation coefficient varies with photon energy and type of material. Mathematically, the attenuation of a narrow beam of monoenergetic photons is given by:

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

where:

I(x) = radiation intensity exiting a material of thickness x
I = radiation intensity entering a material o
e = base of natural logarithms (2.714.....)
μ = linear attenuation coefficient
x = thickness of material

This equation shows that the intensity is reduced exponentially with thickness. I(x) never actually equals zero because x-rays and gamma rays interact based on probability, and there is a finite (albeit small) probability that a gamma could penetrate through a thick shield without interacting. Shielding for x-rays and gamma rays then becomes an "as low as is reasonably achievable" (ALARA) issue and not an issue of shielding to zero intensities.

The formula above is used to calculate the radiation intensity from a narrow beam behind a shield of thickness x, or to calculate the thickness of absorber necessary to reduce radiation intensity to a desired level. Tables and graphs are available that give values of μ determined experimentally for different radiation energies and many absorbing materials. The larger the value of μ, the greater the reduction in intensity for a given thickness of material. The fact that lead has a high μ for x-ray and gamma radiation is partially responsible for its wide use as a shielding material.

Buildup

Although attenuation of the initial beam of photons occurs by photoelectric, Compton, and pair production interactions, photons can be scattered by Compton interactions. If the beam is broad, generally considered to be a poor geometry condition, photons can be scattered into the area one is trying to shield. The scattered photons are accounted for by a build up factor (B) in the attenuation equation as follows:

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

where:

B = the buildup factor which is always greater than 1.

Tables of dose build-up factors (indicating that the increased radiation intensity is to be measured in terms of dose units) can be found in the Radiological Health Handbook, Revised Edition.

The buildup is mostly due to Compton scattering. Scattered radiation is present to some extent whenever an absorbing medium is in the path of radiation. Frequently, room walls, the floor, and other solid objects are near enough to a source of radiation to make scatter appreciable. When a point source is used under these conditions, the inverse square law is no longer completely valid for computing radiation intensity at a distance. Measurement of the radiation is then necessary to determine the potential exposure at any point.

Fast neutrons are poorly absorbed by most materials, and the neutrons merely scatter through the material. For efficient shielding of fast neutrons, one needs to slow them down and then provide a material that readily absorbs slow neutrons.

Since the greatest transfer of energy takes place in collisions between particles of equal mass, hydrogenous materials are most effective for slowing down fast neutrons. Water, paraffin, and concrete are all rich in hydrogen, and thus important in neutron shielding. Once the neutrons have been reduced in energy, typically either boron or cadmium is used to absorb the slowed neutrons.

Borated polyethylene is commonly available for shielding of fast neutrons. Polyethylene is rich in hydrogen, and boron is distributed more or less uniformly throughout the material to absorb the slowed neutrons that are available. When a boron atom captures a neutron, it emits an alpha particle, but because of the extremely short range of alpha particles, there is no additional hazard.

A shield using cadmium to absorb the slowed neutrons is usually built in a layered fashion. Neutron capture by cadmium results in the emission of gamma radiation. Lead or a similar gamma absorber must be used as a shield against these gamma rays. A complete shield for a capsule type neutron source may consist of, first, a thick layer of paraffin to slow down the neutrons, then a surrounding layer of cadmium to absorb the slow neutrons, and finally, an outer layer of lead to absorb both the gamma rays produced in the cadmium and those emanating from the capsule.

Rather than using linear or mass absorption coefficients, the microscopic cross section (σ) is used to describe the ability of a given absorber to remove neutrons from a beam, and is given in either cm^2 or in a unit called the barn (b). One barn equals 10^{-28} m^2. The cross section denotes the target size an atom has to a bombarding particle. It is not actually an area, but rather a probability that a reaction will occur. The macroscopic cross section (Σ) equals the number of absorber atoms per cm^3 (N) multiplied by the microscopic cross section (s) such that:

$$\Sigma = N\sigma$$

The attenuation of a narrow beam of monoenergetic neutrons passing through a material follows an exponential relationship given by the following equation:

$$\phi\ (x) = \phi_o e^{-\Sigma x}$$

where:

$\Phi(x)$ = neutron flux exiting a material of thickness x
Φ_o = neutron flux entering material
Σ = total microscopic cross section
x = thickness of material
e = 2.17875

f. Discuss radiation field characteristics for point, line, plane, and volume distributed sources.

The following is taken from DOE-HDBK-1122-99.

Point Source

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, especially when working with point sources, increasing the distance from the source is more effective than decreasing the time spent in the radiation field.

Theoretically, a point source is an imaginary point in space from which all the radiation is assumed to be emanating. While this kind of source is not real (all real sources have dimensions), any geometrically small source of radiation behaves as a point source when one is within three times the largest dimension of the source. Radiation from a point source is emitted equally in all directions. Thus, the photons spread out to cover a greater area as the

distance from the point source increases. The effect is analogous to the way light spreads out as we move away from a single source of light such as a light bulb.

The radiation intensity for a point source decreases according to the inverse square law, which states that as the distance from a point source decreases or increases, the dose rate increases or decreases by the square of the ratio of the distances from the source. The inverse square law becomes inaccurate close to the source (i.e., within three times the largest dimension of the source).

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

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

where:

I_1 = exposure rate at 1st distance (d_1)
I_2 = exposure rate at 2nd distance (d_2)
d_1 = 1st (known) distance
d_2 = 2nd (known) distance
This equation assumes the attenuation of the radiation in the intervening space is negligible, and the dimensions of the source and the detector are small compared with the distance between them.

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. Due to distance constraints, exposures at certain distances from some sources, such as for a pipe or tank, cannot be treated as point sources. In these situations, these sources must be treated as line sources or large surface sources.

Line Source

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:

$$l_1\ (d_1) = l_2\ (d_2)$$

i.e. 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.
- Valid to a point that is one half the distance of the longest dimension of the line source (L/2), beyond which the point source formula should be used.

Plane Source

Planar or surface sources of radiation can be the floor or wall of a room, a large cylindrical or rectangular tank or any other type of geometry where the width or diameter is not small compared to the length. Accurate calculations for these types of sources require the use of calculus; however, a relationship can be described for how exposure rate varies with distance from the source.

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 $1/d^2$.

The exposure rate versus distance calculations can be used to make an estimate of the radiation intensity at various distances. These estimates are valuable tools to estimate and verify the readings obtained from exposure rate meters.

Volume Distributed Source

The following is taken from NUREG-1575, Supp. 1.

Volumetric radioactive material is radioactive material that is distributed throughout or within the material or equipment being measured, as opposed to a surficial distribution. Volumetric radioactive material may be homogeneously (e.g., uniformly activated metal) or heterogeneously (e.g., activated reinforced concrete) distributed throughout the materials and equipment (M&E). Volumetric radioactive material may be distributed throughout the M&E being measured or distributed in layers. Layers of volumetric radioactive material may start at the surface (e.g., porous surfaces penetrated by radioactive material) or under a layer of other material (e.g., activated rebar inside a concrete wall). By definition all radioactive liquids and gases in containers and all bulk quantities of radioactive material when measured as a whole are volumetric radioactive material.

The concept of whether radioactivity is measurable is the major factor in demonstrating compliance with an action level. This document does not provide an exact definition for the transition between surficial and volumetric radioactive material. Rather, the assumptions used to quantify the radioactivity need to be clearly defined and identified so they can be compared to the data quality objectives and measurement quality objectives. Individual action levels may specify applicability to surficial or volumetric radioactivity. In these cases, the definition of surficial and volumetric radioactivity should be specified as part of the definition of the action level.

g. Describe the following particle ejection nuclear reactions and provide an example of each:
- **Alpha, n**
- **Gamma, n**
- **n, Alpha**

Alpha, n

The following is taken from the U.S. Environmental Protection Agency, Radiation Protection.

In an “Alpha, n” reaction, the nucleus is initially in an unstable energy state. An internal change takes place in the unstable nucleus and an alpha particle is ejected leaving a decay product. The atom has then lost two protons along with two neutrons.

Since the number of protons in the nucleus of an atom determines the element, the loss of an alpha particle actually changes the atom to a different element. For example, ^{210}Po is an alpha emitter. During radioactive decay, it loses two protons and becomes a ^{206}Pb atom, which is stable (i.e., nonradioactive).

Gamma, n

Gamma radiation emission occurs when the nucleus of a radioactive atom has too much energy. It often follows the emission of a beta particle.

n, Alpha

The following is taken from DOE-HDBK-1019/1-93.

In a [n, Alpha] particle ejection reaction the incident particle enters the target nucleus forming a compound nucleus. The newly formed compound nucleus has been excited to a high enough energy level to cause it to eject a new particle while the incident neutron remains in the nucleus. After the new particle is ejected, the remaining nucleus may or may not exist in an excited state depending upon the mass-energy balance of the reaction. An example of a particle ejection reaction is shown below.

$$ {}^{1}_{0}n + {}^{10}_{5}B \rightarrow \left({}^{11}_{5}B\right)^{*} \rightarrow {}^{7}_{3}Li + {}^{4}_{2}\alpha $$

2. Radiation protection personnel shall demonstrate a working level knowledge of radioactivity and transformation mechanisms.

a. Define the following terms:

- **Activity**
- **Radioactive decay constant**
- **Curie/becquerel**
- **Radioactive half-life**
- **Radioactive equilibrium**
- **Decay products**
- **Parent nuclide**
- **Activation**
- **Specific activity**
- **Naturally occurring radioactive material (NORM)**
- **Secular equilibrium**
- **Transient equilibrium**
- **Four factor formula**

The following is taken from DOE-HDBK-1019/1-93.

Activity

The activity of a sample is the rate of decay of that sample. This rate of decay is usually measured in the number of disintegrations that occur per second. For a sample containing millions of atoms, the activity is the product of the decay constant and the number of atoms present in the sample.

The relationship between the activity, number of atoms, and decay constant is shown in the following equation.

$$A = \lambda N$$

where:

A = activity in curies, becquerels, or disintegrations per second
λ = radioactive decay constant in reciprocal units of time (sec^{-1})
N = the number of atoms present in the sample

Since λ is a constant, the activity and the number of atoms are always proportional.

Radioactive Decay Constant

There is a certain probability that in a given time interval a certain fraction of the nuclei within a sample of a particular nuclide will decay. This probability per unit time that an atom of a nuclide will decay is known as the radioactive decay constant, λ.

Curie/Becquerel

Two common units to measure the activity of a substance are the curie (Ci) and becquerel (Bq). A curie is a unit of measure of the rate of radioactive decay equal to 3.7×10^{10} disintegrations per second. This is approximately equivalent to the number of disintegrations that one gram of ^{226}Ra will undergo in one second. A becquerel is a more fundamental unit of measure of radioactive decay that is equal to 1 disintegration per second.

Radioactive Half-Life

The radioactive half-life is defined as the amount of time required for the activity to decrease to one-half of its original value.

Radioactive Equilibrium

Radioactive equilibrium exists when a radioactive nuclide is decaying at the same rate at which it is being produced. Since the production rate and decay rate are equal, the number of atoms present remains constant over time.

Decay Products

According to DOE-HDBK-1122-99, the decay product, or daughter has become an atom of a new element with chemical properties entirely unlike the original parent atom. With each transmutation an emission from the nucleus occurs. There are several modes of decay and emissions associated with each mode.

Parent Nuclide

According to DOE-HDBK-1122-99, in a radioactive decay series, the decay of the parent nuclide produces a daughter product and radiation is emitted. The daughter nuclide also produces radioactivity when it decays, as does each successive daughter in the chain until stability is reached, resulting in total collective activity. The activity contributed from the parent versus the daughters will vary depending on the half-life of the parent and the half-lives of the daughters.

Activation

According to DOE-HDBK-1108-2002, activation is the process of producing a radioactive material by bombardment with neutrons, protons, or other nuclear particles.

Specific Activity

Specific activity is the activity per unit mass of a radionuclide. It is generally reported in units of curies per gram (Ci/g) or becquerels per kilogram (Bq/kg). Specific activity varies with the half-life and the gram atomic weight of the radionuclide.

Naturally Occurring Radioactive Material (NORM)

The following is taken from DOE M 435.1-1.

Naturally occurring radioactive materials are naturally occurring materials not regulated under the *Atomic Energy Act of 1954*, as amended, whose composition, radionuclide concentrations, availability, or proximity to man have been increased by or as a result of human practices. NORM does not include the natural radioactivity of rocks or soils, or background radiation.

Secular Equilibrium

According to DOE-HDBK-1122-99, secular equilibrium the half-life of the parent is very much longer than the half-life of the daughter. When in equilibrium, the activity of the daughter is equal to the activity of the parent. Initially, the majority of the activity will be contributed by the parent. As more and more of the parent nuclide decays, the amount of activity contributed by the daughter will increase.

Transient Equilibrium

According to DOE-HDBK-1122-99, in transient equilibrium, the half-life of the parent is longer than that of the daughter, but not very long. In a freshly purified parent fraction, the daughter activity builds up, and then decays with the same rate of decay as the parent.

Four Factor Formula

According to DOE-HDBK-1122-99, a group of fast neutrons produced by fission can enter into several reactions. Some of these reactions reduce the size of the neutron group while other reactions allow the group to increase in size or produce a second generation. There are four factors that are completely independent of the size and shape of the reactor that give the inherent multiplication ability of the fuel and moderator materials without regard to leakage. This four factor formula accurately represents the infinite multiplication factor as shown in the following equation

$$k\infty = \varepsilon\, p\, f\, \eta$$

where:

ε = fast fission factor
p = resonance escape probability
f = thermal utilization factor
η = reproduction factor

Each of these four factors represents a process that adds to or subtracts from the initial neutron group produced in a generation by fission.

b. Describe the following processes including any resulting product of decay:

- **Alpha decay**
- **Beta-minus decay**
- **Beta-plus decay**
- **Electron capture**
- **Isomeric transition**
- **Internal conversion**
- **X-ray generation**

The following is taken from DOE-HDBK-1122-99.

Alpha Decay

With a few exceptions, only relatively heavy radioactive nuclides decay by alpha emission. An alpha particle is essentially a helium nucleus. It consists of two protons and two neutrons, giving it a mass of 4 atomic mass units (amu). Because of the two protons, it has an electric charge of +2. The symbol (α) is used to designate alpha particles.

A nucleus emitting an alpha particle decays to a daughter element, reduced in Z by 2 and reduced in mass number (A) by 4. The standard notation for alpha decay is:

$$^{A}_{Z}X \rightarrow ^{A-4}_{Z-2}Y + ^{4}_{2}\alpha$$

For example, ^{226}Ra decays by alpha emission to produce ^{222}Rn as follows:

$$^{226}_{88}Ra \rightarrow ^{222}_{86}Rn + ^{4}_{2}\alpha$$

Alpha particles are the least penetrating of the three most common types of radiations. They can be absorbed, or stopped, by a few centimeters of air or a sheet of paper.

Beta-Minus Decay

A nuclide that has an excess number of neutrons (i.e., the neutron to proton, (n:p) ratio is high) will usually decay by beta-minus emission. The intranuclear effect would be the changing of a neutron into a proton, thereby decreasing the n:p ratio, resulting in the emission of a beta-minus particle. Beta-minus particles are negatively charged particles. They have the same mass as an electron (1/1836 of a proton or 5.49E-4 amu) as well as the

same charge (-1) and can be considered high speed electrons. Beta particles originate in the nucleus, in contrast with ordinary electrons, which exist in orbits around the nucleus. The symbol (β^-) is used to designate beta-minus particles.

In beta-minus emitters, the nucleus of the parent gives off a negatively charged particle due to a neutron changing into a proton. The atomic number increases by one, but the mass number is unchanged. There is also the emission of an antineutrino, symbolized by the Greek letter nu with a bar above it ($\bar{\upsilon}$).

The standard notation for beta-minus decay is:

$$^{A}_{Z}X \rightarrow ^{A}_{Z+1}Y + \beta^- + \bar{\upsilon}$$

For example, ^{214}Pb decays by beta-minus emission to produce ^{214}Bi as follows:

$$^{210}_{82}Pb \rightarrow ^{210}_{83}Bi + \beta^- + \bar{\upsilon}$$

Beta-minus particles are emitted with kinetic energies ranging up to the maximum value of the decay energy, E_{max}. The average energy of beta-minus particles is about $1/E_{max}$. They travel several hundred times the distance of alpha particles in air and require a few millimeters of aluminum to stop them.

Neutrinos (ν) and anti-neutrinos ($\bar{\upsilon}$) are neutral (uncharged) particles with negligible rest mass that travel at the speed of light and are very non-interacting. They account for the energy distribution among positrons and beta particles from given radionuclides in the positron- and beta-decay processes respectively. Since they are so non-interacting, no energy is deposited in tissue; therefore, no dose results to personnel exposed to neutrinos.

Beta-Plus Decay

A nuclide that has a low n:p ratio (too many protons) will tend to decay by positron emission. A positron is often mistakenly thought of as a positive electron. If positive electrons existed, when they encountered an ordinary negative electron, the Coulomb force would cause the two particles to accelerate toward each other. They would collide and the two equal but opposite charges would mutually cancel, which would leave two neutral electrons. Actually, a positron is the anti-particle of an electron. This means that it has the opposite charge (+1) of an electron (or beta particle). Thus, the positron is a positively charged, high-speed particle which originates in the nucleus. Because of its positive charge and rest mass equal to that of a beta particle, a positron is sometimes referred to as beta-plus. The symbol (β^+) is used to designate positrons. With positron emitters, the parent nucleus changes a proton into a neutron and gives off a positively charged particle. This results in a daughter less positive by one unit of charge. Because a proton has been replaced by a neutron, the atomic number decreases by one and the mass number remains unchanged. The emission of a neutrino also occurs in conjunction with the positron emission.

Electron Capture

For radionuclides having a low n:p ratio, another mode of decay known as orbital electron capture can occur. In this radioactive decay process the nucleus captures an electron from an orbital shell of the atom, usually the K-shell since the electrons in that shell are closest to the nucleus. The nucleus might conceivably capture an L-shell electron, but K-electron capture is much more probable. This mode of decay is frequently referred to as K-capture. The transmutation resembles that of positron emission, as follows:

$$^{A}_{Z}X + ^{0}_{-1}e \rightarrow ^{A}_{Z-1}Y + \nu$$

The electron combines with a proton to form a neutron, followed by the emission of a neutrino. Electrons from higher energy levels immediately move in to fill the vacancies left in the inner, lower-energy shells. The excess energy emitted in these moves results in a cascade of characteristic x-ray photons. Either positron emission or electron capture can be expected in nuclides with a low n:p ratio. The intranuclear effect of either mode of decay would be to change a proton into a neutron, thus increasing the n:p ratio.

Isomeric Transition

Isomeric transition commonly occurs immediately after particle emission; however, the nucleus may remain in an excited state for a measurable period of time before dropping to the ground state at its own characteristic rate. A nucleus that remains in such an excited state is known as an isomer because it is in a metastable state; that is, it differs in energy and behavior from other nuclei with the same atomic number and mass number. Generally, the isomer achieves ground state by emitting delayed (usually greater than 10^{-9} seconds) gamma radiation.

The metastable, or excited state, is usually represented by a small (m) following the mass number, A, in the standard nuclide notation. For example, ^{99m}Tc and ^{99}Tc are isomers.

Internal Conversion

The phenomena of internal conversion occurs when a gamma photon does not escape the electron cloud surrounding the nucleus, but transfers enough energy to one of the orbital electrons to eject it from the atom. The photon is said to have undergone internal conversion. The conversion electron is ejected from the atom with kinetic energy equal to the gamma energy minus the binding energy of the orbital electron. This process usually takes place in the K-shell. There will then follow emission of characteristic x-rays as with electron capture. In principle, it is similar to the photoelectric effect.

X-Ray Generation

X-rays are generated by both man-made and natural methods. In general, x-rays are generated by two processes: (1) the transition of orbital electrons from higher energy levels to lower energy levels, or (2) the change in energy or path of a high speed charged particle (i.e., bremsstrahlung).

Transitions of orbital electrons occur by adding energy to the atom. Energy is most commonly added to the atom by interactions with ionizing radiations. An example would be

gamma radiation causing a K-shell photoelectric effect and leaving a K-shell vacancy. The vacant spot will be filled by orbital electrons cascading down higher energy orbits. To conserve energy, characteristic x-rays are emitted from the electronic shell of the atom. The energy of the x-ray will be the difference between the electron shell energy levels. Charged particles interacting by ionizations and excitations can also result in x-ray emission. An electron can be raised to a higher energy level by a passing charged particle. After the charged particle passes, the electron will transition back to the ground state by emitting an x-ray.

Medical departments generate x-rays for diagnostic purposes by bombarding a high Z material, such as tungsten metal, with high speed electrons. The electrons come via thermionic emission by passing a current through a filament. The electrons are accelerated in a vacuum by a high voltage electric field into the tungsten target. When the electron passes near the proximity of the positively charged nucleus, it is deflected from its path and decelerated. Since the electron has undergone a loss of energy, an x-ray is given off equal to the energy lost by the electron. This type of x-ray generation is known as bremsstrahlung radiation, which is German for "braking radiation."

c. Given the "Chart of the Nuclides," trace the decay chain for a specified nuclide.
d. Given either the half-life or the radioactive decay constant, solve radioactive decay problems.
e. Using the specific activity or decay constant of an isotope, convert between mass quantities and curies.
f. Convert numerical amounts of radioactivity between curie, becquerel, and dpm.

Note: Competencies "c" through "f" are performance based. The Qualifying Official will evaluate their completion.

3. Radiation protection personnel shall demonstrate a working level knowledge of principles and concepts for internal and external dosimetry.

a. Define the following terms:
- **Dose equivalent**
- **Shallow dose equivalent**
- **Deep dose equivalent**
- **Effective dose equivalent**
- **Committed dose equivalent**
- **Committed effective dose equivalent**
- **Total effective dose equivalent**
- **Whole body**
- **Extremity**
- **Lens of the eye dose equivalent**
- **Derived air concentrations (DAC)**
- **Annual limit on intake (ALI)**
- **Quality factor**
- **Weighting factor**
- **Roentgen**
- **Rad**
- **Rem**

- **Sievert (SI)**
- **Gray (Gy)**
- **Stochastic effects**
- **Nonstochastic (deterministic) effects**

Note: Some of these terms have been modified and re-defined by 10 CFR 835.2, as of January 1, 2008. For purposes of clarity, these modifications will be identified in the definition. Other terms are defined in 10 CFR 835.2, 2007, and still others are taken from DOE-HDBK-1122-99.

Dose Equivalent

Dose equivalent, now known as equivalent dose (H_T), means the product of average absorbed dose ($D_{T,R}$) in rad (or gray) in a tissue or organ (T) and a radiation (R) weighting factor(w_R).

Shallow Dose Equivalent

Shallow dose equivalent is the dose equivalent deriving from external radiation at a depth of 0.007 cm in tissue.

Deep Dose Equivalent

Deep dose equivalent is the dose equivalent derived from external radiation at a tissue depth of 1 centimeter in tissue.

Effective Dose Equivalent

Effective dose equivalent, now known as effective dose, means the summation of the products of the equivalent dose received by specified tissues or organs of the body (H_T) and the appropriate tissue weighting factor (W_T)— that is, $E = \Sigma W_T H_T$.

Committed Dose Equivalent

Committed dose equivalent, now known as committed equivalent dose, means the equivalent dose calculated to be received by a tissue or organ over a 50-year period after the intake of a radionuclide into the body.

Committed Effective Dose Equivalent

Committed effective dose equivalent, now known as committed effective dose, (E_{50}) means the sum of the committed equivalent doses to various tissues or organs in the body ($H_{T,50}$), each multiplied by the appropriate tissue weighting factor (W_T)—that is, $E_{50} = \Sigma W_T H_{T,50} + W_{Remainder} H_{Remainder,50}$. Where $W_{Remainder}$ is the tissue weighting factor assigned to the remainder organs and tissues and $H_{Remainder,50}$ is the committed equivalent dose to the remainder organs and tissues. Committed effective dose is expressed in units of rem (or Sv).

Total Effective Dose Equivalent

Total effective dose equivalent, now known as total effective dose means the sum of the effective dose (for external exposures) and the committed effective dose.

Whole Body

Whole body is a term used to describe external exposure to the head, trunk (including male gonads), upper arms including the elbows, and upper legs including the knees.

Extremity

Extremity refers to the hands and arms below the elbows, or the feet and legs below the knees.

Lens of the Eye Dose Equivalent

The lens of the eye dose equivalent is the external exposure of the lens of the eye and is taken as the dose equivalent at a tissue depth of 0.3 cm.

Derived Air Concentration(s) (DAC)

For the radionuclides listed in appendix A of this part [10 CFR 835], the airborne concentration that equals the ALI divided by the volume of air breathed by an average worker for a working year of 2000 hours (assuming a breathing volume of 2400 m^3).

Annual Limit on Intake (ALI)

Annual limit on intake and the derived limit for the amount of radioactive material taken into the body of an adult worker by inhalation or ingestion in a year.

Quality Factor

Quality factor, now known as the radiation weighting factor, (W_R) means the modifying factor used to calculate the equivalent dose from the average tissue or organ absorbed dose; the absorbed dose (expressed in rad or gray) is multiplied by the appropriate radiation weighting factor.

Weighting Factor

Weighting factor, now known as the tissue weighting factor, (W_T) means the fraction of the overall health risk, resulting from uniform, whole body irradiation, attributable to specific tissue (T). The equivalent dose to tissue, (H_T), is multiplied by the appropriate tissue weighting factor to obtain the effective dose (E) contribution from that tissue.

Roentgen

Exposure is a measure of the ability of photons (X and gamma) to produce ionization in air. Traditionally, the unit of exposure is the roentgen (R). The unit is defined as the sum of charge per unit mass of air; that is:

1 roentgen = 2.58E-4 coulombs/kg of air

The roentgen was originally defined as the quantity of X or gamma radiation that will produce ions carrying 1.0 electrostatic unit of electrical charge in 1 cubic centimeter of dry air under standard conditions.

Rad

The old (CGS) unit of absorbed dose is the rad, which is an acronym for Radiation Absorbed Dose. The unit rad can be applied to all types of radiation and is defined as the deposition by any radiation of 100 ergs of energy in one gram of any material.

Note: For simplicity purposes, 1 rad of photons is usually considered to be equivalent to 1 R. The actual physical relationship is such that an exposure of 1 R would produce an absorbed dose of 0.87 air rads. This means that 1 R = 87 ergs/g.

Rem

The old unit of dose equivalent is the rem, which is an acronym for Roentgen Equivalent Man. The rem was the quantity of ionizing radiation whose biological effect (in man) is equal to that produced by 1 roentgen of x-rays or gamma radiation. The dose equivalent in rem is numerically equal to the absorbed dose in rad multiplied by the quality factor:

$$\text{rem} = \text{rad} \times Q$$

Sievert

The SI derived unit of dose equivalence is the sievert (Sv). The dose equivalent in sieverts is equal to the absorbed dose in grays multiplied by the quality factor:

$$\text{sievert} = \text{gray} \times Q$$

Since one gray is equal to 100 rad, it follows that:

$$1 \text{ Sv} = 100 \text{ rem}$$

It should be emphasized that the relative risk from one rem dose equivalent from neutrons is the same as the risk from one rem dose equivalent from gamma or any other radiation. Use of dose equivalent units for recording personnel radiation exposure permits us to add exposures from various types of radiation and get a total dose equivalent which is proportional to the risk.

Stochastic Effects

Stochastic effects means malignant and hereditary diseases for which the probability of an effect occurring, rather than its severity, is regarded as a function of dose without a threshold, for radiation protection purposes.

Non-stochastic (deterministic) Effects

Non-stochastic means effects due to radiation exposure for which the severity varies with the dose and for which a threshold normally exists (*e.g.*, radiation-induced opacities within the lens of the eye).

b. Describe the various types of bioassays and their applications and limitations.

The following is taken from DOE-HDBK-1122-99.

Internal contamination may be monitored in one of two ways. The first method includes whole body counts and specific organ counts (lungs, thyroid, etc.). This type of internal monitoring is called in vivo monitoring. The other type of internal contamination monitoring uses some sample from the person to determine the presence of contamination. Methods may include urinalysis, fecal analysis, blood sampling and others. These methods are called in vitro monitoring.

The following is taken from HPS N13.30-1996.

Direct Radiobioassay (In Vivo Counting) Measurements

The provisions of this standard are meant to apply to any direct radiobioassay system used to measure radionuclides that are distributed throughout the whole body or localized in individual organs such as thyroid or lungs, or both. Procedures shall be validated, written, reviewed, and approved in accordance with the service laboratory quality assurance plan.

The *in vivo* program (both the counter design and the operational protocol) shall be designed to minimize measurement uncertainties when used to measure actual depositions of analyte in individuals. A major source of uncertainty can occur because individuals and the distribution of radionuclides within those individuals may be different than those represented by phantoms used for counter calibration and performance testing. The design goal of the program should be to minimize these uncertainties such that significant levels of activity from the radionuclides reasonably expected to be present can be determined. For some situations (e.g., transuranics and uranium), the current technology may not allow these criteria to be met. In these cases, indirect radiobioassay should be considered to obtain estimates of deposition of radionuclides in the subject.

Indirect Radiobioassay (In Vitro) Measurements

Urine and fecal samples are frequently collected and analyzed to assist the health physicist in estimating the intake of radioactive material by the worker (person). On occasion, samples of breath, nasal fluid, blood, hair, finger nails, or other biological specimens are analyzed. The performance criteria of this standard, however, are directed primarily toward improving the accuracy and reliability of measurements of radioactivity in excreta samples.

The laboratory shall determine analytical results and propagated standard errors in units of activity (Bq) or mass as requested, and shall be capable of separating or resolving a mixture of radionuclides, where necessary or possible. (For example $^{239/240}$Pu can not be separated by current alpha spectroscopy methods.) The appropriate volume, recovery, and decay corrections shall be made. The standard error of each result shall be calculated and shall include propagation of the estimated measurement uncertainties. As a minimum, the uncertainties shall include those associated with calibration, counting, measurement of volume or weight, losses from chemical separations, transfer operations, and impurities.

c. Discuss the methods of reducing dose from internally deposited radionuclides.

The following is taken from DOE-HDBK-1122-99.

The measures used to minimize the concentration of airborne contaminants that exist 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.

Title 10 CFR 835 states that an Airborne Radioactivity Area is any area where the concentration of airborne radioactivity, above natural background, exceeds or is likely to exceed the DAC values listed in appendix A or appendix C of Part 835, or where an individual present in the area without respiratory protection could receive an intake exceeding 12 DAC-hours in a week.

Other protective actions include the following:

- Posting of airborne radioactivity areas controls access to minimize exposure
- Minimizing the stay time of workers in airborne areas to short periods of time
- Augmenting installed engineering controls with respiratory protection equipment to further reduce the concentration of contaminants in the air the workers are actually breathing.

For detailed information on respirator use and regulations, see *Practices for Respiratory Protection*, American National Standards Institute (ANSI Z88.2-1992).

The following is taken from DOE-HDBK-1122-99.

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. A good example is increasing water intake following ^{3}H exposure. 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. A common form of this is the use of calcium to reduce deposition of strontium. 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 two 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. They are 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.

Diuretics

Diuretics increase urinary excretion of sodium and water. Diuretics are used to reduce internal exposure; however its use has been limited. Applications could include ^{3}H, ^{42}K, ^{38}Cl and others. Diuretics can lead to dehydration and other complications if not performed properly.

Expectorants and Inhalants

These are used to increase flow of respiratory tract excretions. Thus far this type of therapy has not been proven successful in removing radioactive particles from all areas of lungs.

Lung Lavage

This method 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.

d. Discuss the process used to evaluate dose based on bioassay results.

The following is taken from DOE-STD-1121-2003.

The derivation of intakes and retained quantities from bioassay data may be the critical step in the dose assessment process. Evaluations of exposure to internal radionuclides should account for all possible sites of retention and their associated retention times (if known) in the body.

Generalized biokinetic models, suitably modified to account for experience or studies at the facility, may provide a starting point for the initial assessment of an intake and for determining the specific needs for follow-up bioassay measurements. All organs contributing to the effective dose equivalent, calculated with the weighting factors given in 10 CFR 835, should be considered rather than only those organs I which the radionuclide can be readily measured.

When thorough bioassay histories are attainable, and good confidence can be placed on organ and whole body radionuclide content evaluations, it is possible to explicitly derive the retained quantity and retention history of an exposure without resorting to use of default parameters. In some cases the uncertainties associated with the bio-kinetics are much greater than the uncertainty in the direct assessments of intake and retained quantity.

A tritium exposure with sufficient urine assay data to document the biological excretion rate is an example of using excretion history. Due to uncertainty in the route of intake (e.g., skin absorption versus inhalation) and in the biological clearance rate (which depends on water consumption), the tritium excretion history provides the best assessment of the number of transitions and, thus, the dose equivalent. Similarly, a radioiodine exposure, well documented in time and monitored by in vivo thyroid counting, can be assessed directly from the bioassay result. In both cases, discrete or parameterized methods of summation of transitions in the well-known source organs will provide sufficient information for dose assessment.

Where direct uptake and retention history are used for dose assessment, the method for converting data to dose equivalent should be documented as part of the dose assessment. However, if the bioassay data are insufficient for a thorough assessment of retained quantities, or are of such poor quality that whole body or pertinent organ content cannot be directly derived, then biokinetic models should be used.

A biokinetic model is a time-dependent mathematical representation of the relationship between intake, uptake, retention, translocation, and excretion for radionuclides taken into

the body. Models differ in their scientific approach and mathematical formalism. Some models, such as systemic uptake excretion models, are empirically derived from studies of radionuclide behavior in humans or animals. Other models are derived from considerations of the fundamental physiological and biochemical processes of the body.

The following equation is taken from NUREG/CR-4884.

$$I = \frac{A(t)}{IRF(t)}$$

where:

I = estimate of intake
A(t) = the value of the bioassay measurement obtained at time t
IRF(t) = intake retention fraction corresponding to the type of measurement for time t after estimate time of intake

e. Describe the principle of operation, proper use, placement, function, and type of radiation detected by the following dose-measuring instruments:
- **Thermoluminescent dosimeter, including Albedo dosimeter**
- **Pocket dosimeter (quartz fiber and electronic)**
- **Film badge**
- **Personnel nuclear accident dosimeter**

The following is taken from DOE-HDBK-1122-99.

Thermoluminescent Dosimeter

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 and phosphorescence.

Fluorescence

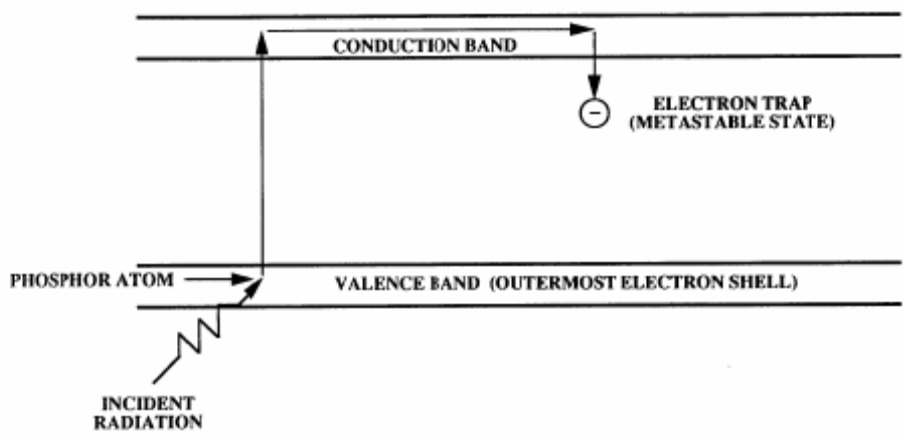
This is emission of light during or immediately after irradiation (within fractions of a second) of the phosphor. This is not a particularly useful reaction for thermoluminescent dosimetry (TLD) use.

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

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, as shown below in figure 8. This energy is given up if the electron returns to the valence band.



Source: DOE-HDBK-1122-99

Figure 8. 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. 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:

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

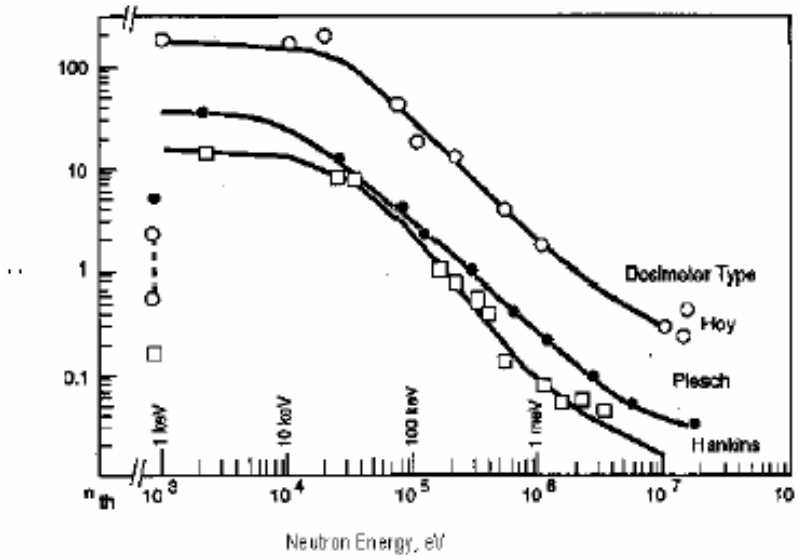
A glow curve can be obtained from the heating process. The light output from TL material is not easily interpreted. Multiple peaks result as the material is heated and electrons trapped in "shallow" traps are released. This results in a peak as these traps are emptied. The light output drops off as these 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.

Albedo Dosimeter

The following is taken from DOE-STD-1128-98.

Thermoluminescent dosimeters are the most widely used neutron dosimeters in plutonium facilities. The energy response of a typical TLD-albedo dosimeter is shown in the following figure. At neutron energies below about 20 keV, the energy response is almost constant. Above 20 keV, the response per unit dose equivalent drops dramatically by almost three

orders of magnitude at 10 MeV. Almost all neutrons emitted by plutonium have energies in the MeV range. However, about 50% of the neutrons striking a thick concrete wall or floor are reflected back into the room at lower energies, and neutrons typically are reflected two or three times before being absorbed. Thus, the low-energy scattered neutrons are often more important in determining TLD-albedo dosimeter response than the high-energy neutrons emitted by the plutonium source. The TLD-albedo dosimeters are often calibrated in specific facilities by exposing them on phantoms at locations where the dose equivalent has been carefully determined from dose and spectrometric measurements.



Source: DOE-STD-1128-98.

Figure 9. Energy dependence of various TLD-Albedo dosimeters

Pocket Dosimeter

The following is taken from DOE-HDBK-1122-99.

Pocket dosimeters are compact, easy-to-carry devices that indicate an individual's accumulated dose to radiation at anytime, thus eliminating the delay of film badge/TLD processing. However, because of the possibility of faulty readings due to rough treatment, the dosimeter reading does not constitute a permanent legal record of dose received. A pocket dosimeter can be self-reading or not. In the self-reading type, a small compound microscope is used to observe the response. The type which is not self-reading, called the pocket chamber, is similar in construction to the self-reading type, but another instrument called the charger reader must be used to read it. The self-reading type is normally preferred since it can be read anywhere and at any time.

A self-reading pocket dosimeter consists of a small air-filled chamber in which a quartz fiber electrometer, a small microscope and a graduated scope across which the shadow of the quartz fiber moves to indicate the applied dose, is suspended.

The design and operation of a self-reading pocket dosimeter utilizes the principle of discharging a pair of opposite charged surfaces when the air between them is exposed to ionizing radiation. The electric charge required to attract the ionized gas particles is impressed on the electrometer and the chamber wall by means of a suitable charging unit. Ionizing radiation penetrating the chamber forms positively and negatively charged gas particles. These charged particles are attracted to the oppositely charged surface; i.e., the negative particles are attracted to the electrometer and the positive particles are attracted to the chamber wall. The migration of the negative particles to the electrometer permit the fiber to move closer to the frame, which in turn causes the shadow of the fiber to move across the calibrated scale.

Pocket dosimeters are available in many ranges of gamma exposure from 0 through 200 milliroentgens to 0 through 1,000 R. The sensitivity of the instrument is determined at the time of manufacture by the selection of the electrical condenser incorporated in the unit.

The dosimeter charger is a small, portable, battery-operated power supply designed to impress a DC voltage on the charging electrode of a dosimeter. In use, the dosimeter is inserted into the charging well and pressed down firmly. Pressing the dosimeter into the well serves two purposes: (1) it actuates a switch that turns on a bulb so the fiber of the electrometer may be seen, and (2) it closes the charging switch in the end of the dosimeter so that an electrical connection is made between the charging circuit and the electrode. While holding the dosimeter firmly in the well, an adjusting knob is rotated until the hairline is on a mark just below zero. Frequently, a capacitance effect causes the hairline to shift slightly when releasing the contact with the charging well. This may necessitate several charging attempts, setting the hairline slightly above or below the mark, to accomplish a zero setting when the dosimeter is removed from the charging well. However, it is not necessary to have an exact zero setting since the exposure received is determined by subtracting the initial reading from the final reading.

Pocket dosimeters should be worn on the upper front part of the body close to the dosimeter of record (e.g., the TLD). Pocket dosimeters are quite rugged and durable; however, they are designed to be worn on a person's clothing and should not be subjected to any greater shock or abuse than one would expect during normal active work.

Film Badge

The following is taken from DOE-HDBK-1122-99.

All radiation measurement systems consist of a detector and some sort of a readout circuitry. A detector may be combined with appropriate circuitry to form an instrument, or the detector and the readout may be separate (TLD + film, for example).

In the detector, the incident radiation interacts with the detector material to produce an observable effect, be it a chemical change or creation of an electrical signal. With a few exceptions, the effect caused by radiation incident on a detector is not permanent. In these detectors the effect is observed as it occurs and yields a signal in terms of events per unit time.

These detectors are typically used in association with rate meters, instruments which read out in terms of count per minute (cpm), mR/hr, etc. The exceptions occur mostly in dosimetry instruments. In these detectors, the effects are accumulated for analysis at a later time. Thus, instead of events per unit time, the accumulated effect caused by all events is measured. These detectors are often classified as integrating detectors.

Detectors are characterized by the type of interaction which produces the effect and the way in which the detector is operated. In chemical detectors, the incident radiation causes ionization or excitation of the detector media thereby causing chemical changes which can be analyzed. Film badges are an example of a chemical detector.

Personnel Nuclear Accident Dosimeter

The following is taken from DOE-STD-1098-2008.

Facilities that possess fissile materials in sufficient quantities to create a critical mass such that the potential exists for excessive exposure of individuals in an accident shall provide nuclear accident dosimetry to affected individuals [see 835.1304(a)]. 2. The nuclear accident dosimetry system shall include the following:

- A method to conduct initial screening of potentially exposed individuals to identify those who have received significant doses [see 835.1304(b)(1)]
- Equipment and methods sufficient to analyze appropriate biological samples [see 835.1304(b)(2)] and dosimeters
- A system of fixed nuclear accident dosimeter units [see 835.1304(b)(3)] capable of measuring the estimated neutron dose and approximate neutron spectrum
- Personnel nuclear accident dosimeters [see 835.1304(b)(4)].

The fixed dosimeters discussed above should:

- Be capable of determining the neutron dose from 10 rads to approximately 10,000 rads with an accuracy of ± 25%
- Be capable of measuring fission gamma radiation from 10 rads to approximately 10,000 rads in the presence of neutron radiation with an accuracy of approximately ± 25%.

Personnel nuclear accident dosimeters should be capable of measuring an absorbed dose in or on a phantom from 10 rads to approximately 1,000 rads with an accuracy of ± 25%. An analysis of the fixed dosimetry system needs should be documented and should consider such factors as the nature of operations, structural design of the facility, area accessibility, number of dosimeters and their location, and the effect of intervening shielding. The analysis should be reevaluated as necessary to ensure facility modifications do not impair the capabilities of the fixed dosimetry system.

f. Discuss the concepts of ICRP Publications 26 and 30 as they relate to internal and external dosimetry.

The following is taken from International Council on Radiation Protection (ICRP) Publications 26 and 30.

Almost every exposure of the body involves the irradiation of more than one tissue, it is appropriate to recommend a dose equivalent limit based upon the total risk of all tissues

irradiated. This system incorporates the setting of a single dose equivalent limit for uniform irradiation of the whole body and implements a subsystem designed to ensure that the total risk from irradiations of parts of the body does not exceed the risk from uniform irradiation of the whole body.

For occupational exposures, it is appropriate to assess the levels of risk associated with the dose equivalent limits. A valid method for judging the acceptability of the level of risk in radiation work is to compare this risk with the risk of other occupations recognized as having high standards of safety. Safe industries are considered to have an accident rate of less than or equal to 10^{-4} fatalities per year.

One of the basic concepts is that the total risk of radiation exposure should include the risks from external and internal exposures. This is accomplished by developing a means to sum external and internal doses together and limit the total dose to 5 rem per year. The ICRP in report 26/30 developed the concept of total effective dose equivalent, or TEDE for short. The TEDE is calculated by adding the internal dose, expressed as the committed effective dose equivalent (CEDE), to the external dose, expressed as the deep dose equivalent (DDE) obtained from external dosimetry data at a measurement depth of 1 cm in tissue (a density thickness of 1,000 mg/cm^2).

External Dose Determination

External dose is determined by monitoring individuals with dosimetry devices that are worn at all times in radiation areas. External limits are specified based on the depth in tissue that the radiation is capable of penetrating. The following table lists the various tissue depths and limits.

Table 1. 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

Source: DOE-HDBK-1122-99.

Internal Dose Determination

Calculating internal dose is much harder than calculating external dose. Because of variations among individuals' size and metabolism, it has been found advantageous to define the physical and chemical properties of a representative individual. To provide a standard for evaluating the effects of radiation on the body, averages are determined for the physical and chemical makeup of the body. This average is called reference man. Two of the more

important models developed for reference man are the respiratory tract and the gastrointestinal tract.

The respiratory tract model is heavily dependent on the particle size of the radioactive material breathed in. The model assumes a particle size of 1 micron.

Using the reference man metabolic retention times and dimensions, calculations were made to determine the amount of radioactive material taken into the body by the specified route that satisfies the following dose limit conditions:

$$\left(\sum_T w_T H_T \leq 50\ mSv\ [5\ rem]\right)$$

Additionally, calculations were made to determine non-stochastic effects:

$$(H_T \leq 0.5\ \text{Sv}\ [50\ \text{rem}])$$

The amount of radioactive material taken into the body is called the ALI. The ALI is the quantity of radioactive material which, if inhaled or ingested in one year, would irradiate a person and result in a stochastic dose of 5 rem or a nonstochastic dose of 50 rem, whichever is less.

The DAC is calculated using the ALI and is the concentration of radioactive material in air that, when breathed by reference man for 2,000 hours in a year under conditions of light activity, would result in an intake equal to the ALI. The DAC is calculated from the following formulas, depending on whether the limit is stochastic or nonstochastic:

$$STOCHASTIC\ DAC = \frac{sALI}{2{,}400\ m^3}$$

$$NONSTOCHASTIC\ DAC = \frac{nALI}{2{,}400\ m^3}$$

g. Discuss the newer concepts of ICRP Publications, including 60, 66, 68, 71, and 72, and DOE Radiological Control Technical Positions (RCTP) papers, such as RCTP 2000-01, as they relate to DOE requirements for internal and external dosimetry.

ICRP Publication 60

The ICRP issued its last basic recommendations in 1977. The recommendations have been used widely throughout the world to limit exposure of radiation workers and members of the public to ionizing radiations. Supplementary statements to the 1977 recommendations were issued when necessary by the Commission, but developments in the last few years have made it necessary to issue a completely new set of recommendations, officially adopted in

November 1990. In publishing these recommendations, the Commission has had three aims in mind: to take account of new biological information and of trends in the setting of safety standards; to improve the presentation of the recommendations; and to maintain as much stability in the recommendations as is consistent with the new information. The recommendations are set out in the form of a main text supported by annexes. The main text contains all the recommendations, together with sufficient explanatory material to make clear the underlying reasoning for policy makers. The supporting annexes contain more detailed scientific information on specific points for specialists.

ICRP Publication 66

This report describes a revision of the model used in ICRP Publication 30 to calculate radiation doses to the respiratory tract of workers resulting from the intake of airborne radionuclides. This revision was motivated by the availability of increased knowledge of the anatomy and physiology of the respiratory tract and of the deposition, clearance, and biological effects of inhaled radioactive particles, and by greatly expanded dosimetry requirements. To meet the needs of radiation protection, a dosimetric model for the respiratory tract should:

- provide calculations of doses for individual members of the populations of all ethnic groups, in addition to workers;
- be useful for predictive and assessment purposes, as well as for deriving limits on intakes;
- account for the influence of smoking, air pollutants, and respiratory tract diseases;
- provide for estimates of respiratory tract tissue doses from bioassay data; and
- be equally applicable to radioactive gases and particles.

In addition, large differences in radiation sensitivity among the respiratory tract tissues and the doses they receive from inhaled radionuclides argue for calculating radiation doses to specific tissues of the respiratory tract. To use these tissue doses effectively for radiation protection purposes, they must be compatible with the ICRP dosimetric system. Addressing all of these requirements has resulted in a dosimetry model that is more complex than previous models. This complexity is reflected in the structure of this report, which includes chapters on respiratory tract morphometry, physiology, and radiation biology or health detriment. Other chapters deal with deposition and clearance of inhaled radioactive particles, inhalation of radioactive gases, and application of the model to estimate respiratory tract doses. The model provides most of the flexibility needed to calculate doses to the respiratory tract for a wide range of exposure conditions and for specific individuals. This flexibility also allows for revision of reference parameter values as new information becomes available without changing the model. The wide availability of personal computers allows for easy use of the model.

ICRP Publication 68

The Commission's 1990 recommendations on radiation protection standards in ICRP Publication 60 were developed to take into account new biological information related to the detriment associated with radiation exposures, and to supersede the earlier recommendations in ICRP Publication 26. To permit immediate application of these new recommendations, revised values of the ALIs, based on the methodology and biokinetic information and incorporating the new dose limits and tissue weighting factors (w_T), were issued as ICRP

Publication 61. Since issuing ICRP Publication 61, ICRP has published a revised kinetic and dosimetric model of the respiratory tract. The main aim of publication 68 is to give values of dose coefficients for workers using this new model.

ICRP Publication 71

Since issuing *ICRP Publication 56,* ICRP has issued in *ICRP Publication 66* a revised kinetic and dosimetric model of the respiratory tract, which is also age-specific. The main aim of the present report [Publication 71] is to give values of inhalation dose coefficients for members of the public using this new model, for selected radioisotopes of the 29 elements covered in *ICRP Publications 56, 67* and 69, and of calcium and curium, for which biokinetic models are given here.

ICRP Publication 72

This report is a compilation of age-dependent committed effective dose coefficients for members of the public from intakes by ingestion and inhalation of radioisotopes of the 31 elements covered by *ICRP Publications 56, 67, 79, and 71.* This report also gives committed effective dose coefficients for members of the public from radioisotopes of the additional 60 elements for which dose coefficients are given for workers in *ICRP Publication 68.* This report does not give committed equivalent doses to tissues and organs. In a few instances there are small changes in ingestion dose coefficients from those given in previous reports. In these cases, the values given here supersede the earlier values. These ICRP dose coefficients have been adopted in the International Basic Safety Standards and in the Euratom Directive. Values of dose coefficients for workers are also given in the International Basic Safety Standards and in the Euratom Directive. With the exception of inhalation dose coefficients for ^{226}RA, these values are the same as those given in *ICRP Publication 68.*

RCTP 2000-01

Title 10 Code of Federal Regulations, Part 835 (10 CFR 835), "Occupational Radiation Protection," establishes occupational radiation protection requirements for Department of Energy (DOE) activities and includes requirements for the monitoring and control of individual exposures to various airborne radionuclides. These requirements are based on Radiation Protection Guidance to Federal Agencies for Occupational Exposure, which was issued by the President in January 1987. The Presidential guidance recommended use of weighting factors which were, at the time, consistent with the most current international and national recommendations.

Title 10 CFR 835.2(b) defines weighting factors consistent with the Presidential guidance and requires their use for determining dose. More current national and international recommendations now include tissue weighting factors that differ from those in 10 CFR 835.

Appendix A of 10 CFR 835, Derived Air Concentrations (DAC) for Controlling Radiation Exposure to Workers at DOE Facilities provides DAC values for numerous radionuclides that are used to control individual internal doses, identify the need for air monitoring, and identify the need for posting. Use of more current national and international recommended internal dosimetry methodologies would result in much different DAC values. Questions have been raised concerning DOE's position on the use of more current national and international scientific consensus recommendations as a source for tissue weighting factors

and allowance to use these revised tissue weighting factors and other factors to revise the 10 CFR 835 appendix A DAC values.

Title 10 CFR 835 requires the use of the 10 CFR 835 defined tissue weighting factors in assessing dose. DOE recognizes the limitation this requirement imposes on adopting the latest international and national recommendations for assessing internal dose and is working on developing a DOE complex-wide resolution. Excluding revisions to tissue weighting factors, DOE does allow individual sites to evaluate and use site specific particle size corrections and newer internal dosimetry models to best assess dose.

DAC values from 10 CFR 835 are required to be used for controlling individual internal doses, identifying the need for air monitoring, and identifying and posting airborne radioactivity areas. 10 CFR 835 does not require that these DAC values be used to determine individual internal doses. The use of a newer internal dosimetry methodology could result in situations with significantly different DAC values. While 10 CFR 835 only allows appropriate particle size corrections to be made to the 10 CFR 835 DAC values, the regulations do not prevent individual sites from implementing more restrictive internal exposure controls than those specified in 10 CFR 835. As necessary, sites may implement radiation protection program elements and controls that are more restrictive than those required by 10 CFR 835.

h. Discuss various methods used to estimate worker exposure in the absence of individual monitoring results.

The following is taken from DOE-STD-1121-2003.

DOE's occupational radiation protection system is dose-based. 10 CFR 835.209(b) is the only requirement that addresses methods of internal dose assessment: The estimation of internal dose shall 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.

Title 10 CFR 835.209(b) does not require sites to use air monitoring data for internal dose assessment, but permits sites to use air monitoring data under certain conditions. "Inadequate bioassay," for compliance with 10 CFR 835.209(c), may be taken to pertain to radionuclides with effective half-lives too short to be feasible for routine or special bioassay. Such radionuclides include radon and thoron and their short-lived progeny, as well as radionuclides such as ^{227}Th, ^{223}Ra, ^{225}Ra, and ^{225}Ac when separated from their long-lived parents. It may also include routine bioassay for isotopes with high dose per intake coefficients, such as many of the transuranics.

As defined in the Internal Dosimetry Program Guide (IDG), a technology shortfall, such as for routine bioassay monitoring for Pu, does not preclude the use of routine bioassay monitoring nor force the use of air sample data for dose calculations. Rather, the IDG suggests that the capabilities of the bioassay program be stretched as far as reasonable, that workplace monitoring be enhanced, and that state-of-the-art techniques be used in general.

Reliance should be placed on prompt detection of possible intakes in the workplace, and that special bioassay should be promptly initiated (usually the same day) when intakes are suspected. In vivo count times should be as long as reasonable, and minimum detectable amounts (MDAs) should be as low as reasonably achievable, with an emphasis in both cases on "reasonable" as explained in the IDG. Air sample data may be used for initiating special bioassay without being used for dose assessment.

i. Given airborne radioactivity concentration, DAC value, and worker occupancy time, evaluate resulting worker dose.

The following is taken from DOE-STD-1121-2003.

Note: Refer to appendix A of 10 CFR 835 for current DAC values of various nuclides.

The DAC is the quotient of the annual limit on intake (ALI); not tabulated in 10 CFR 835 by the volume of air that Reference Man breathes in 1 working year (40 hr wk^{-1} × 50 wk yr^{-1} × 1.2 m^3 hr^{-1} = 2400 m^3 yr^{-1} or 2.4 × 10^9 mL yr^{-1}). The DACs are expressed in µCi mL^{-1} or Bq·m^{-3}, or, for radon and thoron progeny, in working levels (WL). For a stochastic ALI (denoted SALI), breathing air at one DAC_s (stochastic DAC) for 2000 hours results in a committed effective dose equivalent ($H_{E,50}$) of 5 rem to Reference Man. For a nonstochastic or deterministic ALI (NALI), breathing air at one DAC_n (nonstochastic DAC) for 2000 hours results in a 50-year tissue committed dose equivalent to tissue T ($H_{T,50}$) of 50 rem..

Note that the DACs listed in appendix A to 10 CFR 835 may be either stochastic (denoted as "St" in the right-hand column) or nonstochastic (denoted by "BS," "K," "L," "SW," and "T" [bone surfaces, kidneys, liver, stomach wall, and thyroid, respectively] in the right-hand column), so that reference to other documents may be needed for dose assessment.

4. Radiation protection personnel shall demonstrate a working level knowledge of the biological effects of radiation.

a. Describe the effects of radiation exposure on the cellular level, including:
- **Direct effects**
- **Indirect effects**

The following is taken from USNRC Technical Training Center, Reactor Concepts Manual, Biological Effects of Radiation.

Even though all subsequent biological effects can be traced back to the interaction of radiation with atoms, there are two mechanisms by which radiation ultimately affects cells. These two mechanisms are commonly called direct and indirect effects.

Direct Effect

If radiation interacts with the atoms of the DNA molecule, or some other cellular component critical to the survival of the cell, it is referred to as a direct effect. Such an interaction may affect the ability of the cell to reproduce and, thus, survive. If enough atoms are affected such that the chromosomes do not replicate properly, or if there is significant alteration in the

information carried by the DNA molecule, then the cell may be destroyed by "direct" interference with its life-sustaining system.

Indirect Effect

If a cell is exposed to radiation, the probability of the radiation interacting with the DNA molecule is very small since these critical components make up such a small part of the cell. However, each cell, just as is the case for the human body, is mostly water. Therefore, there is a much higher probability of radiation interacting with the water that makes up most of the cell's volume. When radiation interacts with water, it may break the bonds that hold the water molecule together, producing fragments such as hydrogen (H) and hydroxyls (OH). These fragments may recombine or may interact with other fragments or ions to form compounds, such as water, which would not harm the cell. However, they could combine to form toxic substances, such as hydrogen peroxide (H_2O_2), which can contribute to the destruction of the cell.

b. Describe the factors affecting radiation sensitivity of cells (i.e., the law of Bergonie and Tribondeau).

The following is taken from DOE-HDBK-1122-99.

As early as 1906 an attempt was made to correlate the differences in sensitivity of various cells with differences in cellular physiology. These differences in sensitivity are stated in the Law of Bergonie and Tribondeau: "The radiosensitivity of a tissue is directly proportional to its reproductive capacity and inversely proportional to its degree of differentiation." In other words, cells most active in reproducing themselves and cells not fully mature will be most harmed by radiation. This law is considered to be a rule-of-thumb, with some cells and issues showing exceptions. Since the time that the law of Bergonie and Tribondeau was formulated, it is generally accepted that cells tend to be radiosensitive if they

- have a high division rate (i.e., cell cycle time, or time between divisions)
- have a high metabolic rate
- are of a non-specialized type (i.e., a cell that is capable of specialization into an adult cell type, such as a fertilized ovum)
- are well nourished

Used generally, tissues that are young and rapidly growing are most likely radiosensitive. The law can be used to classify the following tissues as radiosensitive:

- Germinal (reproductive) cells of the ovary and testis, i.e., spermatogonia
- Hematopoietic (blood-forming) tissues, i.e., red bone marrow, spleen, lymph nodes, thymus
- Epithelium of the skin
- Epithelium of the gastrointestinal tract (interstitial crypt cells)

The law can be used to classify the following tissues as radio-resistant:

- Bone
- Liver
- Kidney
- Cartilage
- Muscle

- Nervous tissue

c. Describe the acute effects and corresponding doses associated with the following:
- **Blood changes**
- **Hemopoietic syndrome**
- **Gastrointestinal syndrome**
- **Central nervous system syndrome**

The following is taken from DOE-HDBK-1122-99.

Acute effects are classified as effects that occur within 1-2 months of the exposure. This definition is somewhat arbitrary in view of the various factors that can affect the length of time between the exposure and the effect. Normally, acute effects are only observed if the dose is greater than 10 rads (0.1 gray) and delivered over a short time (acutely). Generally, the larger the dose, the shorter the time to produce an acute effect. For example, in the absence of medical treatment, the time between an acute dose and death is typically 1-2 months for 500 rad, compared with 1-2 days for 5000 rad.

At high dose rates the body repair mechanisms become less effective and the differences in the biological damage from low linear energy transfer (LET) radiation disappear. As such, the concept of the dose equivalent does not apply to acute exposures greater that 15 rem (0.15 sievert). Above this, the exposure should only be expressed as a dose in rads. For example, the dose from gammas that would kill 50% of an exposed human population is estimated to be from 350 to 450 rads (3.5-4.5 gray), one to two times the required neutron dose. At low doses however, the dose from gammas typically needs to be 5 to 10 times the neutron dose to produce comparable effects.

Blood Changes

The following is taken from the State of Washington, Department of Health, FactSheet 11.

The average person in the U.S. is exposed to background radiation levels that would result in an annual dose of approximately 360 mrem. Higher and more short-term doses, although unlikely, are termed acute exposures. Whole-body doses of radiation (of the type in x-ray or gamma radiation) in significant doses of 35 rad can cause nausea, weakness and appetite loss within a few hours following an acute exposure. These symptoms will disappear within a few hours of the exposure. After doses of 50 rad or above from acute radiation exposures, many immune competent cells are used up defending the body from infection, and others are prevented from performing their duty. Virtually no new replacement cells are produced because of the extensive damage to stem cells in bone marrow.

At doses between 125 – 300 rad, there is increasing likelihood of severity of nausea, vomiting and weakness with symptoms persisting for up to two days. There is 50% mortality from acute exposures greater than 350 rad without medical treatment. Infection is the main cause of death after an acute exposure to radiation. However, cells differ in their sensitivity to ionizing radiation damage. The loss of a specific component could lead to an overall failure in the defense mechanism. Table 2 provides radiation dose effects on humans who have experienced short-term radiation exposure.

Table 2. Biological effects of short term radiation on humans

Biological Effects of Short Term Radiation on Humans	
Dose (Rad)	**Effect**
0-20	No detectable effects
20-100	Measurable transient blood changes. Temporary decrease in white blood cell count.
100-200	Acute radiation sickness - nausea, vomiting, longer-term decrease in white blood cells.
200-300	Vomiting, diarrhea, loss of appetite, listlessness, death in some cases.
300-600	Vomiting, diarrhea, hemorrhaging, deaths occurring in 50% of cases at 350 rad or above without medical treatment.
Above 600	Eventual death in almost all cases

Source: NCRP Report 138

The following is taken from DOE-HDBK-1122-99.

Hemopoietic Syndrome

Hemopoietic Syndrome (or Hematopoietic Syndrome) is also called "therapeutic range" because treatment can play a large role. The dose level is between 200 to 1,000 rads (2 -10 gray) though some blood changes can be seen at lower doses). Critical organs affected are the blood forming organs. It affects the production of white blood cells, as well as leukopenia- a decreased ability to fight infection. It also causes lowered platelet count, which in turn causes hemorrhaging and slowing of the healing process.

Symptoms include nausea and vomiting and epilation. Treatment is usually through antibiotics to fight infection and bone marrow transplants to replace damaged cells, though it is uncertain if this works. If death does occur it will be due to infection and hemorrhaging.

Gastrointestinal Syndrome

The gastrointestinal tract (GI) is covered with small finger-like projections called villi. Villi add to the effective surface area of the lining and thereby increase the capacity of the body to absorb nutrients. The cells on the surface of the villi are constantly migrating towards the tip of the projections where they are sloughed off. Mitotically-active cells (crypt cells) at the base of the villi replace those that are lost. The turnover rate of these epithelial cells is high, and they have an average life span from one to three days.

Sufficiently large acute exposures lead to the reproductive death of the rapidly dividing crypt cells. The cells covering the villi continue to be sloughed off, but are no longer replaced. This deterioration of the lining of the GI tract then leads to a loss of body fluid, inadequate absorption of nutrients, and infection from the intestinal area.

At doses less than 1,000 rad, the crypt cells can recover in about a week. For acute whole-body exposures greater than 1,000 rad, the crypt cells will not survive. Surgical replacement of the small intestine is not possible, and therefore, survival is impossible. Death occurs in 1 to 2 weeks from the damage to the lining of the GI tract (resulting in circulatory collapse) and damage to the hematopoietic system. A dose of 1,000 rad represents the maximum survival whole-body dose, provided that extensive medical care is given.

Central Nervous System (CNS) Syndrome

The CNS syndrome is produced by acute whole body exposures above 5,000 rads (50 gray); exposure of the head alone may have similar effects. Survival is impossible. Death results from respiratory failure and/or brain edema caused from direct or indirect effects on the CNS.

Although the CNS syndrome is not well understood, it most likely involves a combination of cellular and vascular damage. In other words, there may be direct damage to the brain cells by the radiation and indirect damage mediated by effects on the blood vessels of the brain. The latter are known to be damaged by such doses of radiation. Fluid from the blood is lost through the damaged vessel walls into the skull cavity so the pressure inside the skull builds up. Perhaps pressure on certain areas of the brain, i.e., the respiratory center, may be most important, or it may be the change in the blood supply to the brain.

At these high doses, the individual stages of the central nervous system syndrome become so short that they cannot be distinguished. Following such exposures the individual may function coherently for a short while or immediately go into shock.

Within hours the symptoms become very severe. Symptoms include vomiting, diarrhea, apathy, disorientation, and tremors. The victim is also likely to fall into a coma. Death will be due to respiratory failure and/or brain edema and occurs within 30 hours.

d. Discuss delayed effects of radiation exposure, including:

- **Cancer induction**
- **Genetic effects**
- **Prenatal developmental effects**
- **Cataracts**

The following is taken from DOE-HDBK-1122-99.

Cancer Induction

With proper selection of animal species, strains, and dose, ionizing radiation may be shown to exert an almost universal carcinogenic action resulting in tumors in a great variety of organs and tissues. There is human evidence that radiation may contribute to the induction of various kinds of neoplastic diseases. Human evidence includes radium dial painters, radiologists and dentists, uranium miners, and atomic bomb survivors. The main sites of solid tumors are the breasts in women, the thyroid, the lungs, and some digestive organs. These

tumors have long latent periods (approximately 10 to greater than 30 years) and occur in larger numbers than leukemia. Leukemia (abnormal increase in white blood cells) has a much shorter latent period. The incidence peaks within a few years of exposure and returns to normal levels after about 25 years.

Genetic Effects

The following is taken from USNRC Technical Training Center, Reactor Concepts Manual, Biological Effects of Radiation.

The "Genetic Effect" involves the mutation of very specific cells, namely the sperm or egg cells. Mutations of these reproductive cells are passed to the offspring of the individual exposed.

Radiation is an example of a physical mutagenic agent. There are also many chemical agents as well as biological agents (such as viruses) that cause mutations. One very important fact to remember is that radiation increases the spontaneous mutation rate, but does not produce any new mutations. Therefore, despite all of the hideous creatures supposedly produced by radiation in the science fiction literature and cinema, no such transformations have been observed in humans. One possible reason why genetic effects from low dose exposures have not been observed in human studies is that mutations in the reproductive cells may produce such significant changes in the fertilized egg that the result is a nonviable organism which is spontaneously resorbed or aborted during the earliest stages of fertilization.

Although not all mutations would be lethal or even harmful, it is prudent to assume that all mutations are bad, and thus, by USNRC regulation (10 CFR 20), radiation exposure shall be held to the absolute minimum or as low as reasonably achievable (ALARA). This is particularly important since it is believed that risk is directly proportional to dose, without any threshold.

Prenatal Developmental Effects

The following is taken from US N.R.C. Regulatory Guide 8.13.

As discussed in Regulatory Guide 8.29, exposure to any level of radiation is assumed to carry with it a certain amount of risk. In the absence of scientific certainty regarding the relationship between low dose exposure and health effects, and as a conservative assumption for radiation protection purposes, the scientific community generally assumes that any exposure to ionizing radiation may cause undesirable biological effects and that the likelihood of these effects increases as the dose increases. At the occupational dose limit for the whole body of 5 rem (50 mSv) per year, the risk is believed to be very low.

The magnitude of risk of childhood cancer following in utero exposure is uncertain in that both negative and positive studies have been reported. The data from these studies "are consistent with a lifetime cancer risk resulting from exposure during gestation which is two to three times that for the adult" (NCRP Report No. 116, Ref. 2). The NRC has reviewed the available scientific literature and has concluded that the 0.5 rem (5 mSv) limit specified in 10 CFR 20.1208 provides an adequate margin of protection for the embryo/fetus. This dose limit reflects the desire to limit the total lifetime risk of leukemia and other cancers associated with radiation exposure during pregnancy.

Cataracts

The lens of the eye is highly susceptible to irreversible damage by radiation. When the cells of the lens become damaged, they lose their transparency and a cataract is formed. Exposures around 200 rad may produce a cataract, but the symptoms and signs may not be apparent for years after the exposure. The damaging effects of penetrating radiation to the lens of the eye may be cumulative, and repeated small doses may result in cataract formation. Radiation induced cataracts are produced primarily by neutron and gamma radiation. Experiments with animals and human case histories indicate that neutron radiation constitutes the greatest danger, with gamma radiation of slightly less importance. Radiation-induced cataracts differ from naturally occurring cataracts. Radiation induced cataracts form on a different position on the lens of the eye. Susceptibility to radiation induced cataract formation seems to be somewhat dependent on age. Radiation is more likely to produce cataracts in younger persons, because of continuous growth of the lens (growing tissues are more radiosensitive). Extensive irradiation of the eye may result in inflammation of the cornea or in an increase in tension within, and hardening of, the eyeball. These conditions usually become manifest several weeks after the exposure and may terminate in loss of vision.

e. Discuss how the Linear Non-threshold Theory is used in developing risk estimates and dose limits associated with exposure to radiation. (The use of International Commission on Radiological Protection (ICRP) Publications 26 and 60, or National Council on Radiation Protection and Measurements (NCRP) Report No. 116, may be helpful).

The following is taken from DOE-HDBK-1110-2008.

The linear non-threshold hypothesis (theory) assumes the risk of detriment from radiation is directly proportional to the dose and no threshold exists below which there is no detriment (damage). This theory is controversial because it is derived from extrapolation of low dose and low dose rate effects from high dose and high dose rate data. To ensure adequate protection, national and international groups have recommended, and DOE has adopted, a system of regulatory limits and an emphasis on ALARA to keep exposures as far below the limits as is reasonable.

5. Radiation Protection personnel shall demonstrate a working level knowledge of the principles and use of radiological instrumentation and radiological monitoring/survey practices.

a. Describe the principle of operation of gas-filled detectors.

The following is taken from DOE-HDBK-1122-99.

For a gas filled ionization detector to be of value for radiological control purposes, the manner in which the response varies as a function of the energy, quantity, and type of radiation must be known. Factors such as the size and shape of the detector, the pressure and composition of the gas, the size of the voltage potential across the electrodes, the material of construction, the type of radiation, the quantity of radiation, and the energy of the radiation can all affect the response of the detector. Detectors for a special purpose are designed to incorporate the optimum characteristics necessary to obtain the desired response.

Each type of radiation has a specific probability of interaction with the detector media. This probability varies with the energy of the incident radiation and the characteristics of the detector gas. The probability of interaction is expressed in terms of specific ionization with units of ion pairs per centimeter. A radiation with a high specific ionization, such as alpha, will produce more ion pairs in each centimeter that it travels than will a radiation with a low specific ionization such as gamma. In Table 1of DOE-HDBK-1122-99, module 1.13, note the magnitude of the difference between the specific ionization for the three types of radiation.

Generally, the probability of interaction between the incident particle radiation and the detector gas (and therefore the production of ions) decreases with increasing radiation energy. In photon interactions, the overall probability of interaction increases because of the increasing contribution of the pair production reactions. As the energy of the particle radiation decreases, the probability of interaction increases, not only in the gas, but also in the materials of construction. Low energy radiations may be attenuated by the walls of the detector and not reach the gas volume. Obviously, this must be accounted for in the design of the detector. As the number of radiation events striking a detector increases, the overall probability of an interaction occurring with the formation of an ion pair increases. In addition, the number of ion pairs created increases and therefore detector response increases.

The probability of an interaction occurring between the incident radiation and a gas atom increases as the number of atoms present increases. A larger detector volume offers more "targets" for the incident radiation, resulting in a larger number of ion pairs. Since, each radiation has a specific ionization in terms of ion pairs per centimeter, increasing the detector size also increases the length of the path that the radiation traverses through the detector. The longer the path, the larger the number of ion pairs.

The amount of energy expended in the creation of an ion pair is a function of the type of radiation, the energy of the radiation, and the characteristics of the absorber (in this case, the gas). This energy is referred to as the ionization potential, or W-Value, and is expressed in units of electron volts (eV) per ion pair. Typical gases have W-Values of 25-50 eV, with an average of about 34 eV per ion pair.

In the section on detector size, it was shown the probability of interaction increases with detector size. In many cases, there is a practical limit to detector size. Instead of increasing detector size to increase the number of "target" atoms, increasing the pressure of the gas will accomplish the same goal. Gas under pressure has a higher density (more atoms per cm^3) than a gas not under pressure, and therefore offers more targets, a higher probability of interaction, and greater ion pair production. For example, increasing the pressure of a typical gas to 100 psig increases the density by about 7 times.

Once the ion pair is created, it must be collected in order to produce an output pulse or current flow from the detector. If left undisturbed, the ion pairs will recombine, and not be collected. If a voltage potential is applied across the electrodes, a field is created in the detectors, and the ion pairs will be accelerated towards the electrodes. The stronger the field, the stronger the acceleration. As the velocity of the electron increases, the electron may cause one or more ionizations on its own. This process is known as secondary ionization. The secondary ion pairs are accelerated towards the electrode and collected, resulting in a stronger pulse than would have been created by the ions from primary ionization.

If the applied voltage potential is varied from 0 to a high value, and the pulse size recorded, a response curve will be observed. For the purposes of discussion, this curve is broken into six regions. The ion chamber region, the proportional region, and the Geiger-Müller region are useful for detector designs used in radiological control. Other regions are not useful. In the recombination region, the applied voltage is insufficient to collect all of the ion pairs before some of them recombine.

As the voltage to the detector is increased, a point is reached at which essentially all of the ions are collected before they can recombine. No secondary ionization or gas amplification occurs. At this point, the output current of the detector will be at a maximum for a given radiation intensity and will be proportional to that incident radiation intensity. Also, the output current will be relatively independent of small fluctuations in the power supply. The output of a gas-filled detector when 100% of the primary ion pairs are collected is called the saturation current.

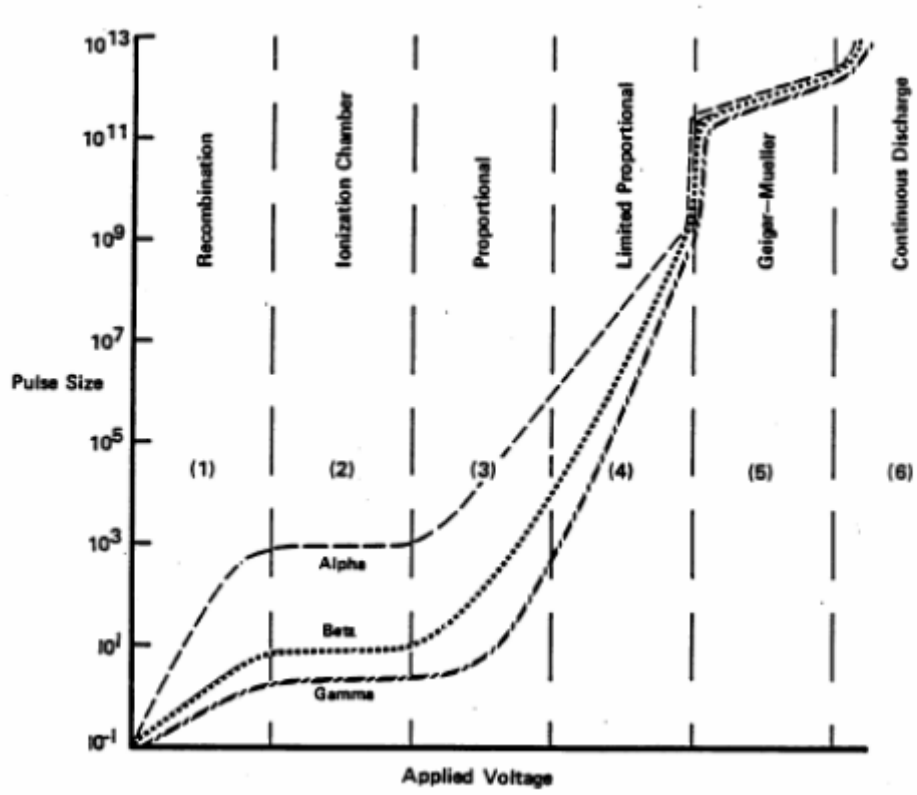
b. Discuss the following for gas-filled detectors:

- **Voltage-response curve (i.e., six region curve)**
- **The three regions useful for radiation detection and measurement**
- **The sequence of events that occur following an initial ionizing event in an Ionization Chamber, a Proportional Counter and a Geiger-Müeller Detector**

The following is taken from DOE-HDBK-1122-99.

Voltage-Response Curve (i.e., Six Region Curve)

Figure 10 is a graphical representation of the six region curve.



Source: DOE-HDBK-1122-99.

Figure 10. Voltage-response curve

If the applied voltage potential is varied from 0 to a high value, and the pulse size recorded, a response curve will be observed. For the purposes of discussion, this curve is broken into six regions.

The Three Regions Useful for Radiation Detection and Measurement

The ion chamber region, the proportional region, and the Geiger-Müller region are useful for detector designs used in radiological control. Other regions are not useful. In the recombination region, the applied voltage is insufficient to collect all of the ion pairs before some of them recombine. In the limited proportional region, neither the output current nor the number of output pulses are proportional to the radiation level. Calibration is impossible. In the continuous discharge region, the voltage is sufficient to cause arcing and breakdown of the detector gas.

The Sequence of Events that Occur Following an Initial Ionizing Event in an Ionization Chamber, a Proportional Counter and a Geiger-Müller Detector

Ionization Chamber

As the voltage to the detector is increased, a point is reached at which essentially all of the ions are collected before they can recombine. No secondary ionization or gas amplification occurs. At this point, the output current of the detector will be at a maximum for a given

radiation intensity and will be proportional to that incident radiation intensity. Also, the output current will be relatively independent of small fluctuations in the power supply.

Proportional Counter

As the voltage on the detector is increased beyond the ion chamber region, the ions created by primary ionization are accelerated by the electric field towards the electrode. Unlike the ion chamber region, however, the primary ions gain enough energy in the acceleration to produce secondary ionization pairs. These newly formed secondary ions are also accelerated, causing additional ionizations. The large number of events, known as an avalanche, creates a single, large electrical pulse.

Geiger-Müller Detector

As the voltage on the detector is increased beyond the proportional region, the detector enters the limited proportional region. As mentioned before, this region is unusable for radiological control purposes. In this region the small individual avalanches which occur within the tube start to interfere with each other. This interference is unpredictable and reduces the overall output signal.

c. Describe the principles of operation of scintillation and solid state detectors.

The following is taken from DOE-HDBK-1122-99.

Scintillation Detectors

Scintillation detectors measure radiation by analyzing the effects of the excitation of the detector material by the incident radiation. Scintillation is the process by which a material emits light when excited. In a scintillation detector, this emitted light is collected and measured to provide an indication of the amount of incident radiation. Numerous materials scintillate - liquids, solids, and gases. A common example is a television picture material which scintillates is commonly called a phosphor or a fluor. The scintillations are commonly detected by a photomultiplier tube.

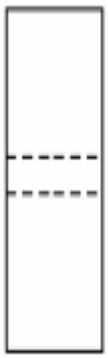
Solid State Detectors

Note: Solid-state detectors are more commonly referred to as semiconductor detectors (for example, germanium – a common semiconductor used in radiation detection). This discussion will focus on that premise.

Conduction Band

Forbidden Band

Valence Band



Conductor
(narrow gap)

Figure 11.
Conductor

In a crystal, the atoms are packed so tightly together that the energy states of individual atoms are modified. This modification splits the states into a number of closely spaced energy levels or bands. The top most band (called the conduction band) has unfilled energy levels. In a conducting solid, the group of "filled bands" are in direct contact with the group of "unfilled bands," so electrons are easily moved into the conduction band.

Conduction Band

Forbidden Band

Valence Band

Insulator
(wide gap)

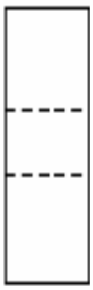
Figure 12.
Insulator

In a good insulator, there is a large enough gap between the group of filled bands and the group of unfilled bands so that a large amount of energy is required to move an electron to the conduction band.

Conduction Band

Forbidden Band

Valence Band



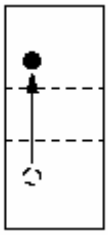
Semiconductor
(smaller gap)

Figure 13.
Semiconductor

A semiconductor has a smaller gap between the two groups of bands so that under certain conditions, electrons can be moved to the conduction band. (For example, heating the material will move at least some electrons to the conduction band.)

e^-

hole



Electron Hole

Figure 14.
Electron hole

When an electron is moved to a higher band, that is, from valence to conduction, a vacancy occurs in the band which it left. This vacancy is called a hole.

Source for figures 12-14: DOE-HDBK-1122-99.

If a strong electric field is applied to the crystal, the electron in the conduction band moves in accordance with the applied field. Similarly, in the group of filled bands, an electron from a lower energy band moves up to fill the hole (vacancy) in the valence band. The hole it leaves behind is filled by an electron from yet a lower energy band. This process continues, so the net effect is that the hole appears to move down through the energy bands in the filled group. Thus, the electron moves in one direction in the unfilled group of bands, while the hole moves in the opposite direction in the filled group of bands. This can be likened to a line of cars awaiting a toll booth, the toll booth being the forbidden band. As a car leaves the "filled valence band" for the unfilled conductance band, a hole is formed. The next car in line fills this hole, and creates a hole, and so on. Consequently, the hole appears to move back through the line of cars.

Any impurities in the crystalline structure can affect the conducting ability of the crystalline solid. There are always some impurities in a semiconductor, no matter how “pure” it is. However, in the fabrication of semiconductors, impurities are intentionally added under controlled conditions. If the impurity added has an excess of outer electrons, it is known as a donor impurity, because the “extra” electron can easily be raided or donated to the conduction band. In effect the presence of this donor impurity decreases the “gap” between the group of filled bands and the group of unfilled bands. Since conduction occurs by the movement of a negative charge, the substance is known as an n-type material. Similarly, if the impurity does not contain enough outer electrons, a vacancy or hole exists. This hole can easily accept electrons from other energy levels in the group of filled bands, and is called an acceptor substance. Although electrons move to fill holes, as described above, the appearance is that the holes move in the opposite direction. Since this impurity gives the appearance of positive holes moving, it is known as a p-type material.

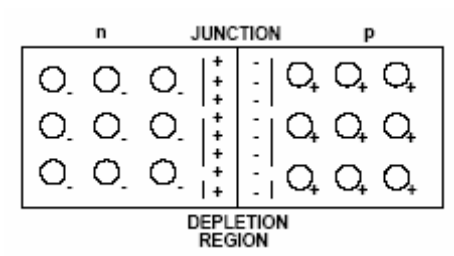
Since any crystalline material has some impurities in it, a given semiconductor will be an n-type or a p-type depending on which concentration of impurity is higher. If the number of n-type impurities is exactly equal to the number of p-type impurities, the crystalline material is referred to as an “intrinsic semiconductor.”

A semiconductor that has been “doped” with the proper amount of the correct type of impurity to make the energy gap between the two groups of bands just right, makes a good radiation detector. A charged particle loses energy by creating electron-hole pairs.

If the semiconductor is connected to an external electrical field, the collection of electron-hole pairs can lead to an induced charge in the external circuit much as the collect of electron-positive atom pairs (ion pairs) is used to measure radiation in an ion chamber. Therefore, the semi-conductor detector relies on the collection of electron-hole pairs to produce a usable electrical signal.

One disadvantage of the semiconductor "detector" is that the impurities, in addition to controlling the size of the energy gap also act as traps. As electrons (or holes) move through the crystalline material, they are attracted to the impurity areas or centers because these impurity centers usually have a net charge. The carrier (electron or hole) may be trapped for awhile at the impurity center and then released. As it begins to move again, it may be trapped at another impurity center and then released again. If the electron or hole is delayed long enough during transit through the crystal, it may not add to the electrical output.

Thus, although the carrier is not actually lost, the net effect on readout is that it is lost. Another disadvantage of the semiconductor detector is that the presence of impurities in the crystal is hard to control to keep the energy gap where it is desired. A newer technique, the junction counter, has been developed to overcome these disadvantages.



n-p junction

In a semiconductor junction counter, an n-type substance is united with a p-type substance. When the two are diffused together to make a diffused junction, a depletion layer is created between the two materials. (This depletion layer is formed by the diffusion of electrons from the n-type material into the p-type material and the diffusion of holes from the p-type material into the n-type material.)

Source: DOE-HDBK-1122-99.

Figure 15. n-p junction

This results in a narrow region which is depleted of carriers and which behaves like an insulator bounded by conducting electrodes. That is, a net charge on each side of the depletion region impedes the further transfer of charge. This charge is positive in the n-region and negative in the p-region. This barrier can be broken if we apply an external voltage to the system and apply it with the proper bias. A "forward bias" is applied when we connect the positive electrode to the p-region. In this case, the barrier breaks down and electrons flow across the junction. However, if we apply a "reverse bias" (negative electrode connected to the p-region), the barrier height is increased and the depleted region is extended.

A further advancement in junction counters is the p-n type. This counter has an intrinsic region between the n and p surface layers. (An intrinsic semiconductor was discussed earlier and is effectively a pure semiconductor.) The presence of an intrinsic region effectively

creates a thicker depletion area. A germanium-lithium Ge(Li) detector is an example of this type of detector.

Lithium (an n-type material) is diffused into p-type germanium. The n-p junction that results is put under reverse bias, and the temperature of the material is raised. Under these conditions, the lithium ions drift through the germanium, balancing n and p material and forming an intrinsic region.

The heat and bias are removed and the crystal cooled quickly to liquid nitrogen temperatures. This intrinsic region serves as the region in which interactions can take place. The intrinsic region can be thought of as a built-in depletion region.

Due to the large size of the depletion region and the reduced mobility of the electrons and holes due to the depressed temperature, a high charge is necessary to cause conduction. The charge is chosen high enough to collect ion pairs, but low enough to prevent noise.

Due to the increased stopping power of germanium over air at -321°F the energy required to create an ion pair is only 2.96 eV compared to 33.7 eV for air. This means that by theory, a germanium detector will respond to any radiation that will create ion pairs. In actuality, however, the response to radiations other than gamma is limited by the materials surrounding the detector, material necessary to maintain temperature. Another consideration limiting response is the geometry of the crystal. The most efficient response occurs when the interaction takes place in the center of the intrinsic region, this can only occur for gamma.

Radiation interacts with atoms in the intrinsic region to produce electron hole pairs. The presence of ion pairs in the depletion region causes current flow. This is similar to a transistor, in that instead of inducing charges in the center section (the base in a transistor) by a battery or another source, the charge is induced by the creation of ion pairs. Since it is not necessary for the ion produced to reach the p and n region to be collected, as in a gas filled chamber, the response is faster.

Since the number of ion pairs produced is a function of the incident energy, and the resulting current is a function of the amount of ion pairs, Ge(Li) response is in terms of energy.

d. Describe the principle of operation and application of nuclear spectroscopy.

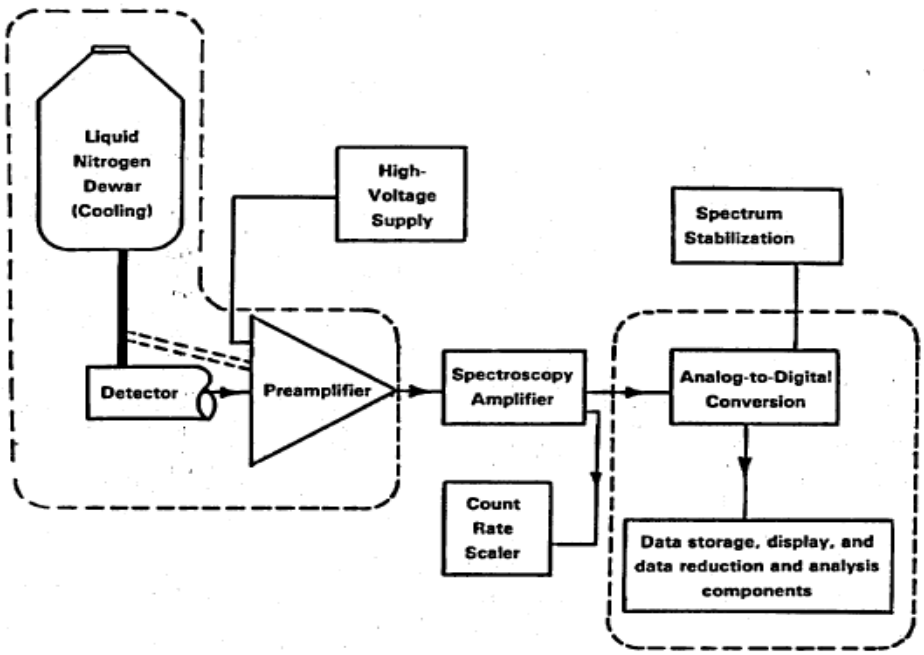
The following is taken from NRC Regulatory Guide 5.9.

Gamma ray spectroscopy systems are used for non-destructive assay (NDA) of various special nuclear material forms encountered in the nuclear fuel cycle, both for quantitative determination of the special nuclear material (SNM) content and for the determination of radionuclide abundances.

Applications of high-resolution gamma ray spectroscopy have multiplied greatly in recent years. The samples encountered range from fresh fuel rods and reprocessing solutions to boxes and cans of uncharacterized waste material. Measurement conditions also vary widely from controlled laboratory environments to the unpredictable plant environment that can be hostile to the measurement equipment and can often contribute serious background interferences to the spectral data. As a result, there is no single gamma ray assay system that

can be effective in all cases. The system chosen for a particular NDA task must therefore be determined from careful consideration of all factors that may affect the measurement and of the requirements for the precision and accuracy of the assay.

A block diagram of a typical high-resolution gamma ray spectroscopy system is shown in the figure below. In such a system, the solid state Ge(Li) or high purity germanium detector converts some or all of the incident gamma ray energy into a proportional amount of electric charge, which can be analyzed by the subsequent electronics. The detector output is converted into an analog voltage signal by the preamplifier, which is an integral part of the detector package. The preamplifier signal is further amplified and shaped and is then converted into digital information that can be stored, displayed, and otherwise processed by the data reduction and analytical components of the system.



Source: US NRC Regulatory Guide 5.9.

Figure 16. High-resolution gamma ray spectroscopy system

Germanium detectors may be represented by full width of the gamma ray photopeak at half of its maximum height (FWHM) energy resolutions of approximately 1.7 keV at 1332 keV (^{60}Co) and approximately 0.7 keV at 122 keV (^{5}Co) for detectors with efficiencies up to 20 percent.5 The full width at 0.1 maximum for such detectors is typically up to 1.9 times the FWHM. For these higher efficiency detectors, “peak-to-Compton ratios” are usually quoted in the range of 25 to 40. These ratios are strong functions of resolution, efficiency, and exact detector crystal geometry, and no typical values can be given without knowledge of all of these parameters. Coaxial detectors with this kind of resolution will usually have cooled field-effect transistor (FET) preamplifiers and an energy-rate capability of approximately 50,000 MeV/sec. Room temperature preamplifiers have somewhat worse resolution but have

rate capabilities on the order of 150,000 MeV/sec. The resolution of planar detectors is a stronger function of the crystal size and shape than that of coaxial detectors, so representative resolutions cannot be given over a range of sizes. As an example from the middle of the range of sizes usually offered, an excellent 2 cm^3 planar detector (le., 2 cm^2 front face area x 1 cm thick) would have a resolution of approximately 0.5 keV at 122 keV (^{57}Co) and 0.21 keV at 5.9 keV (Mn x-ray from ^{55}Fe decay). Planar detectors will always have liquid nitrogen-cooled FET preamplifiers in order to achieve the excellent resolution of these systems. The preamplifier feedback loop may be either pulsed optical or resistive, and the system will have fairly modest rate capabilities in the range of 5000 MeV/sec. It is important to decouple the detector from noisy mechanical environments to avoid microphonic pickup.

For ease of use, maintenance, and replacement of the components in a high-resolution gamma ray spectroscopy system, the electronic components should be standard nuclear instrument modules, with the possible exception of the pulse-height analysis (Le., multichannel analyzer) components. Pulse signals should be transmitted from module to module in shielded coaxial cable to minimize the effects of possible electronic noise from nearby machinery at the measurement site. The cables should have a characteristic impedance that matches the terminations used in the nuclear instrument modules (NIM) modules (generally 93 ohms). The system power supplies (detector high voltage, preamplifier, and NIM bin) should be capable of operating the system within the operating specifications when supplied with 115 volts (±10 percent) at 50 to 65 hertz (at constant room temperature). The power supplied for the detection system should be stabilized against voltage shifts in order to maintain resolution. The output voltage of the detector bias supply is determined by the detector requirements; 5 kilovolts is sufficient for most applications.

The main amplifier, commonly referred to as the spectroscopy amplifier, should have variable gain and pulse-shaping controls for maximum setup flexibility. Most high-quality amplifiers are equipped with baseline restoration and pole-zero cancellation circuits (Ref. 2), which greatly improve the resolution that can be achieved on a routine basis. Baseline restoration is essential for assay situations in which count rates in excess of several kilohertz are anticipated. Pulse pileup suppression is also a useful feature, if available; it may be found in some spectroscopy amplifiers and even in separate NIM modules designed for that purpose.

Electronic components should be obtained with state-of-the-art linearity and temperature sensitivity. Maintenance of long-term gain stability may require the use of a spectrum stabilizer. Centroid variations of a stabilization peak of less than one channel in a 4096-channel spectrum are achievable with commercially available stabilizer modules. Stabilization peaks can be provided either by a pulser or by a radioactive source. Generally, a radioactive source is preferred because it contributes less distortion to the gamma ray spectrum and has a stable (although decaying) emission rate. Furthermore, stabilization peaks from natural sources may be obtained from existing peaks in the assay spectrum itself, which simplifies the assay setup. Dead-time and pileup corrections may also be performed using a pulser or a separate radioactive source fixed to the detector. The latter method is preferred for the reasons stated above.

e. Discuss the purpose, principles of detection and operation, and field application of the following:
- **Continuous air monitors (CAMs)**
- **Airborne radioactivity samplers**
- **Area radiation monitors (ARMs)**
- **Criticality detection/alarm systems**
- **Personnel contamination monitors**
- **Process radiation monitors**

The following is taken from DOE-HDBK-1122-99.

CAMs

Portable continuous air monitors (CAMs) provide an estimate of airborne radioactivity concentrations averaged over time at a particular location, and provide immediate readout and alarm capabilities for preset concentrations. These air monitors are portable low flow rate (20 liters per minute) sampling systems, containing the necessary sampling devices and built-in detection systems to monitor the activity on the filters, cartridges, planchettes and/or chambers in the system. The system may provide a visual readout device for each type of sample medium, a recording system for data, and computer functions such as data trending, preset audible and visual alarms/warning levels and alerts for system malfunctions. Typical CAMs provide information on alpha and/or beta/gamma particulates (filtration), radioiodine activity (adsorption) and noble gas activity (volumetric chamber or in-line detector).

Portable CAMs can be utilized as:
- Low volume general area samplers
- Monitors with alarm capabilities for areas where airborne radioactivity conditions may quickly degrade
- Trending devices in selected areas
- Devices to locate system leaks, if used with the appropriate length hose or tubing.

Installed continuous air monitoring systems (CAMs) provide an estimate of airborne radioactivity concentrations averaged over time at a fixed, designated location, and provide immediate local and remote readout and alarm capabilities for preset concentrations. These air monitors are fixed low flow rate sampling systems, and contain the necessary sampling devices and built-in detection systems to monitor the activity of selected areas or airstreams. The system may provide a local and remote visual readout device, a recording system for data, and computer functions such as data trending, preset audible and visual alarms/warning levels and alerts for system malfunctions. Installed CAM applications include:
- Fixed installations capable of sampling several locations through valved sample lines
- Stack monitors
- Duct monitors

Airborne Radioactivity Samplers

The five primary types of airborne radioactivity samplers/monitors are:
- Personal air samplers (breathing zone)
- High volume/flow rate air samplers
- Low volume/flow rate air samplers
- Portable continuous air monitors (CAMs)
- Installed continuous air monitoring systems

Personal air samplers (PAS) provide an estimate of the airborne radioactivity concentration in the air the worker is breathing during the sampling period. PAS may also be used to determine if the protection factor for respiratory equipment is exceeded, to compare with other workplace air samples, and to verify the effectiveness of engineered and administrative controls.

PAS are small, portable battery-powered devices which sample the air in the breathing zone of the worker's environment, making allowances to eliminate interferences the samplers themselves may have on a worker's activities. Some characteristics are:

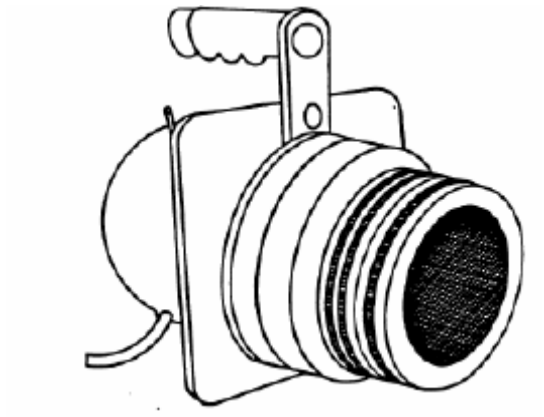
- The device contains a small battery-powered pump that is calibrated to a flow rate approximately 1/10 (2 liters per minute) the breathing rate of a worker performing light activity.
- The sampling line terminates in a filter cassette which contains the filtration medium for the radioactive particulate contaminants. The sample filter cassette is attached close to the nose and mouth of the individual.

High volume/flow rate samplers provide an estimate of the airborne radioactivity concentration at a particular location in a short period of time. Portable high flow rate samplers are used to collect airborne aerosols on a filter paper (filtration) or on a greased planchette (impaction). Portable high flow rate samplers can also be used to collect radioiodine samples using activated charcoal cartridges (adsorption) as long as the maximum flow rate of the cartridge is not exceeded or a correction factor is used. These samplers do not have installed detectors and the sample must be removed from the sampler and analyzed on separate analysis equipment. The high volume/flow rate samplers may be used to:

- Provide a routine "slice of time" estimate of the general area airborne radioactivity
- Verify boundaries of areas posted for airborne radioactivity Monitor the airborne radioactivity related to a specific work activity

High volume samplers typically use flow rates of at least 10 cubic feet per minute (cfm). Although these samplers are noisy and not intended for continuous duty, the shorter sample times allow for greater sensitivity.

Figure 17 illustrates a typical high volume sampler.



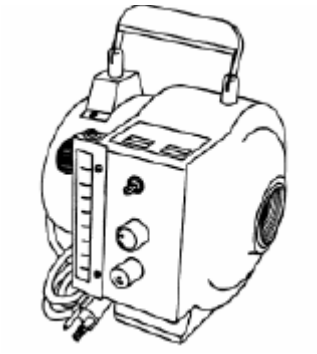
Source: DOE-HDBK-1122-99.

Figure 17. High-volume sampler

Low volume/flow rate samplers provide an estimate of airborne radioactivity concentrations averaged over a longer period of time at a particular location. Portable low volume/flow rate samplers are used to collect samples for aerosols on filter paper (filtration) and radioiodine on an adsorption medium, such as an activated charcoal cartridge. Low volume/flow rate samplers may be used to provide average airborne radioactivity estimates over a period of time for:

- Commonly traversed areas that normally have a low probability of airborne radioactivity problems
- Areas not commonly traversed with a higher probability of airborne radioactivity problems
- Backup samples in areas where airborne radioactivity problems are discovered by other means
- Work maintenance activities normally characterized by low airborne radioactivity concentrations.
- Low volume samplers generally have flow rates set at approximately 20 liters per minute, the breathing rate of a worker performing light activity. Although these samplers must run longer for reasonable sensitivity, they are generally quiet and can be used for continuous duty.

Figure 18 illustrates a typical low volume sampler.



Source: DOE-HDBK-1122-99.

Figure 18. Low-volume sampler

Note: Portable and installed continuous air monitors are discussed above.

ARMs

The following is taken from DOE-STD-1098-2008.

In addition to the requirements and recommendations of Article 551, area radiation monitors (not to include area monitoring dosimeters discussed in Article 514) should be installed in frequently occupied locations with the potential for unexpected increases in dose rates and in remote locations where there is a need for local indication of dose rates prior to personnel entry.

Area radiation monitors should not be substituted for radiation exposure surveys in characterizing a workplace. The need for and placement of area radiation monitors should be documented and assessed when changes to facilities, systems, or equipment occur. In addition to the requirements of Article 562, area radiation monitors should be tested periodically (e.g., quarterly) to verify audible alarm system operability and audibility under ambient working conditions and operability of visual alarms when so equipped. If installed instrumentation is removed from service for maintenance or calibration, a radiation monitoring program providing similar detection capability should be provided, consistent with the potential for unexpected increases in radiation dose rates.

Where an area radiation monitor is incorporated into a safety interlock system, the circuitry should be such that a failure of the monitor either prevents entry into the area or prevents operation of the radiation producing device. If the circuitry is required to ensure compliance with the high radiation area access control requirements of 10 CFR 835.502, then the circuitry shall be fail-safe [see 825.502(b)].

Criticality Detection/Alarm Systems

The following is taken from DOE-STD-1128-98.

As specified in ANSI/ANS-8.3, *Criticality Accident Alarm System,* the need for criticality alarm systems shall be evaluated for all activities in which the inventory of fissionable material in individual unrelated work areas exceeds 700 g of ^{235}U, 520 g of ^{233}U, 450 g of ^{239}Pu, or 450 g of any combination of these three isotopes.

If the fissionable material mass exceeds the ANSI/ANS-8.3 limits and the probability of criticality is greater than 10^{-6} per year, a criticality alarm system shall (DOE O 420.1B, section 4.3.3) be provided to cover occupied areas in which the expected dose exceeds 12 rad in free air. Nuclear accident dosimetry shall also be provided, as required by 10 CFR 835.1304. The criticality alarm system should include a criticality detection device and a personnel evacuation alarm.

Note: In what follows, 10^{-6} per year is used as a measure of credibility, and does not mean that a probabilistic risk assessment has to be performed. Reasonable grounds for incredibility may be presented on the basis of commonly accepted engineering judgment.

- If the fissionable material mass exceeds the ANSI/ANS-8.3 limits and the probability of criticality is greater than 10^{-6} per year, but there are no occupied areas in which the expected dose exceeds 12 rad in free air, then only a criticality detector system (i.e., nuclear accident dosimetry) is needed (DOE O 420.1B, section 4.3.3).
- If the fissionable material mass exceeds the ANSI/ANS-8.3 limits, but a criticality accident is determined to be impossible or less than 10^{-6} per year (per a safety analysis report documentation), then neither a criticality alarm system nor nuclear accident dosimetry is needed (DOE O 420.1B, section 4.3.3).

ANSI/ANS-8.3 provides several additional requirements regarding criticality alarm systems. The alarm signal shall be for immediate evacuation purposes only and of sufficient volume and coverage to be heard in all areas that are to be evacuated. Information on sound levels of the alarm can be found in ANSI/ANS-8.3. The alarm trip point shall be set low enough to detect the minimum accident of concern.

The minimum accident of concern may be assumed to deliver the equivalent of an absorbed dose in free air of 20 rad at a distance of 2 m from the reacting material within 60 sec. The alarm signal shall activate promptly (i.e., within 0.5 sec) when the dose rate at the detectors equals or exceeds a value equivalent to 20 rad/min at 2 m from the reacting material. A visible or audible warning signal shall be provided at a normally occupied location to indicate system malfunction or loss of primary power. Each alarm system should be tested at least once every three months. An evacuation drill shall be conducted at least annually.

Personnel Contamination Monitors

The following is taken from DOE-STD-1128-98.

The facility design should provide for location of personnel monitoring devices in the vicinity of the workplace. To minimize the potential spread of radioactive contamination, personnel survey instruments should be available at suitable locations within the process area, such as for personnel exiting from glove boxes, at bag-out stations, and at exits from compartmentalized facilities. Survey instruments or monitoring instruments should be available at contamination-control change rooms and at exits from controlled areas.

Process Radiation Monitors

The following is taken from DOE-HDBK-1122-99.

Portable survey instruments used for measuring dose rates are typically ion chamber instruments. Ion chambers may also be used in several installed monitor systems such as the area radiation monitor systems (ARMS) and the various process radiation monitors (PRMs).

As the voltage to the detector is increased, a point is reached at which essentially all of the ions are collected before they can recombine. No secondary ionization or gas amplification occurs. At this point, the output current of the detector will be at a maximum for a given radiation intensity and will be proportional to that incident radiation intensity. Also, the output current will be relatively independent of small fluctuations in the power supply.

f. Discuss the basic elements and applicable standards of a radiological instrument calibration program, including the following:
- **Calibration source selection and traceability**
- **Source check and calibration frequency**
- **Instrument energy dependence**

The following is taken from DOE-STD-1128-98.

Calibration Source Selection and Traceability

Radiation doses and energies in the work areas should be well characterized. Calibration of instruments should be conducted where possible under conditions and with radiation energies similar to those encountered at the work stations. Knowledge of the work area radiation spectra and instrument energy response should permit the application of correction factors when it is not possible to calibrate with a source that has the same energy spectrum. All calibration sources should be traceable to recognized national standards. Neutron energy spectral information is considered particularly important because neutron instruments and dosimetry are highly energy-dependent.

Source Check and Calibration Frequency

ANSI N323 (ANSI, 1997b), *Radiation Protection Instrumentation and Calibration,* provides requirements on the calibration of portable instruments. The reproducibility of the instrument readings should be known prior to making calibration adjustments. This is particularly important if the instrument has failed to pass a periodic performance test (i.e., the instrument response varies by more than ±20% from a set of reference readings using a check source) or if the instrument has been repaired. The effect of energy dependence, temperature, humidity, ambient pressure, and source-to-detector geometry should be known when performing the primary calibration. Primary calibration should be performed at least annually.

Instrument Energy Dependence

The calibration of alpha-detection instruments normally should be performed with ^{239}Pu, ^{241}Am, or ^{230}Th sources. Several sources of different activities should be used to calibrate different ranges.

Whenever possible, beta detectors should be calibrated to the beta energies of interest in the workplace. A natural or depleted uranium slab source can be used for calibration of beta detectors when beta radiations in the workplace have energies similar to the uranium. International Organization for Standardization beta sources should be used for all other

purposes: the energy dependence of beta detectors can be tested using the calibration sources listed in the ISO Publication 1980 (1984); these include ^{90}Sr, ^{90}Y, ^{204}Tl, and ^{147}Pm.

The calibration of photon monitoring instruments over the energy range from a few keV to 300 keV is best accomplished with an x-ray machine and appropriate filters that provide known x-ray spectra from a few kiloelectron volts to approximately 300 keV. Radionuclide sources should be used for higher energies. Most ion chambers used to measure photon radiations have a relatively flat energy response above 80 to 100 keV; ^{137}Cs or ^{60}Co are typically used to calibrate these instruments. These sources also may be used to calibrate Geiger-Müller (GM) type detectors used for dose rate measurements. It should be noted that some GM detectors (e.g., those with no energy compensation) can show a large energy dependence, especially below approximately 200 keV. GM detectors should not be used if not energy compensated.

The multisphere or Bonner sphere spectrometer (Bramblett et al., 1960) is the neutron spectrometer system most often used by health physicists for neutron energy spectrum measurements, perhaps because it is simple to operate. Multisphere spectrometers are typically used for measuring neutron energy spectra over a wide energy range from thermal energies to over 20 MeV although detailed energy spectra are not obtained. With the use of an appropriate spectrum unfolding code, the multisphere system will determine the average neutron energy, dose equivalent rate, total flux, kerma, and graphical plots of differential flux versus energy and dose equivalent distribution versus energy.

g. Discuss the following concepts as they relate to radiological counting measurements:
- **Background**
- **Lower limit of detection (LLD)**
- **Minimum detectable activity (MDA)**
- **Counting efficiency**
- **Counting uncertainties**

The following is taken from 10 CFR 835.2.

Background (radiation)

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.

Lower Limit of Detection (LLD)

Note: According to DOE-STD-1121-2003 and HPS N13.30, LLD is synonymous with minimum detectable activity (MDA), discussed below.

Minimum Detectable Amount (MDA)

The following is taken from HPS N13.30.

Minimum detectable amount (MDA) is the smallest amount (activity or mass) of an analyte in a sample that will be detected with a probability ß of non-detection (Type II error) while accepting a probability α of erroneously deciding that a positive (non-zero) quantity of analyte is present in an appropriate blank sample (Type I error).

The value of the MDA indicates the ability of the service laboratory to detect an analyte in a sample or person. The L_C provides a way for distinguishing the difference between the count rate from the analyte under analysis and the count rate from the appropriate blank. For *in vivo* measurements, the sample matrix (i.e., the person) of the analyte is a variable; therefore, the MDA is person-dependent. To lead to consistency, the MDA calculated for a given sample represented by a uniform source distribution, either in a person or in a phantom, shall be used to characterize the detection capability of the service laboratory. The service laboratory shall determine and document the MDA for each analyte for which a service is provided.

Counting Efficiency

The following is taken from DOE-HDBK-1122-99.

A detector intercepts and registers only a fraction of the total number of radiations emitted by a radioactive source. The major factors determining the fraction of radiations emitted by a source that are detected include:

- The fraction of radiations emitted by the source which travel in the direction of the detector window
- The fraction emitted in the direction of the detector window which actually reach the window
- The fraction of radiations incident on the window which actually pass through the window and produce an ionization
- The fraction scattered into the detector window

All radiation detectors will, in principle, produce an output pulse for each particle or photon which interacts within its active volume. The detector then would be said to be 100 percent "efficient," because 100 percent of the activity was detected and reported. Therefore, there is only a certain fraction of the disintegrations occurring that results in counts reported by the detector. Using a calibrated source with a known activity, a precise figure can be determined for this fraction. This value can then be used as a ratio in order to relate the number of pulses counted to the number of particles and/or photons incident on the detector. This ratio is called the efficiency. It can also be referred to as the detector yield, since the detector yields a certain percentage of the actual counts.

The detector efficiency gives us the fraction of counts detected per disintegration, or c/d. Since activity is the number of disintegrations per unit time, and the number of counts are detected in a finite time, the two rates can be used to determine the efficiency if both rates are in the same units of time. Cpm and disintegrations per minute (dpm) are the most common. Thus, the efficiency, E, can be determined as shown in the following equation. Used in this manner the time units will cancel, resulting in counts/disintegration (c/d).

In practice, because of the factors outlined above, the actual (or total) activity emitted from the source is not detected.

$$E = \frac{cpm}{dpm} = \frac{c}{d}$$

The efficiency obtained in the formula above will be in fractional or decimal form. To calculate the percent efficiency, the fraction can be multiplied by 100. For example, an efficiency of 0.25 would mean 0.25 × 100, or 25%.

Counting Uncertainties

Note: The usual eight steps for evaluating and reporting the uncertainty of a measurement are summarized in NUREG 1575, Supp. 1. The publication also includes additional discussion and equations for clarification. The steps are summarized here.

1. Identify the measurand, Y, and all the input quantities, X_i, for the mathematical model.
2. Determine an estimate, x_i, of the value of each input quantity, X_i.
3. Evaluate the standard uncertainty, u(xi), for each input estimate, xi, using a Type A method, a Type B method, or a combination of both.
4. Evaluate the covariances, $u(x_i,x_j)$, for all pairs of input estimates with potentially significant correlations.
5. Calculate the estimate, y, of the measurand from the relationship $y = f(X_1,X_2,\ldots,X_N)$.
6. Determine the combined standard uncertainty, (CSU) $u_c(y)$, of the estimate, y.
7. Optionally multiply $u_c(y)$ by a coverage factor k to obtain the expanded uncertainty U.
8. Report the result as y ± U with the unit of measurement.

h. Describe various radiological situations and the use of appropriate radiological surveys, including: radiation, contamination, and airborne radioactivity surveys.

The following is taken from DOE-HDBK-1122-99.

Radiological surveys are conducted to monitor radioactive contamination. General monitoring requirements include:

- Ambient air in the immediate vicinity of active and inactive sites.
- Surface water (rivers, estuaries, lakes and oceans) and sediments are monitored for constituents indicating the status of operational practices and control.
- Soil and vegetation are monitored to detect possible contamination from fallout and uptake.
- Ground water wells are surveyed to ensure their physical integrity.
- Background dose rates are monitored near facilities that may have elevated dose rates.
- Radiation surveys are performed to detect contamination spread.

Survey frequencies for particular sites are to be determined by the technical judgment of environmental protection and/or radiological control and may depend on the site history,

radiological status, use and general conditions. Appropriate documentation must be completed for each environmental survey.

6. Radiation protection personnel shall demonstrate a working level knowledge of internal and external radiation protection principles and control techniques.

a. Discuss the implication of the following on the identification of hazards associated with radiological work activities and how it might affect the controls specified on a Radiation Work Permit (RWP):
- **Location of the work (i.e., in a radiation, contaminated, or airborne area)**
- **System being worked on (i.e., fluid under pressure, hazardous or radioactive)**
- **Nature of the work activity (inspection, opening system, etc.)**

The following is taken from DOE-STD-1098-2008.

The RWP is an administrative mechanism used to establish radiological controls for intended work activities. The RWP informs workers of area radiological conditions and entry requirements and provides a mechanism to relate worker exposure to specific work activities.

The RWP should be integrated with other work authorizations that address safety and health issues, such as those for industrial safety and hygiene, welding, or confined space entry. An alternative formal mechanism, such as written procedures or experiment authorizations, may be used in lieu of an RWP as the administrative control over radiological work activities. If an alternative mechanism is used, it should meet the standards established in this Article and Articles 322 and 323.The RWP should include the following information, unless the information is contained in other related work-control documents:
- Description of work
- Work area radiological conditions
- Dosimetry requirements, including any bioassay requirements
- Pre-job briefing requirements, as applicable
- Training requirements for entry
- Protective clothing and respiratory protection requirements
- Radiological control coverage requirements and stay time controls, as applicable
- Limiting radiological conditions that may void the RWP
- Special dose or contamination reduction considerations
- Special personnel frisking considerations
- Technical work document number, as applicable
- Unique identifying number
- Date of issue and expiration
- Authorizing signatures

b. Discuss special exposure control, survey, and personnel monitoring techniques associated with work in the following areas or situations:
- **Non-uniform-radiation fields**
- **High-radiation areas**
- **Contact work with radioactive materials/sources**

The following is taken from DOE-STD-1098-2008.

Non-Uniform-Radiation Fields

Multiple dosimeters should be issued to individuals to assess effective dose in non-uniform radiation fields. Non-uniform radiation fields exist when the dose to a portion of the whole body will exceed the dose to the primary dosimeter by more than 50 percent and the anticipated whole body dose is greater than100 millirem. When the radiation field is well characterized and the worker's orientation is known, relocation of the primary dosimeter is permitted in lieu of issuance of multiple dosimeters. Under such conditions, the individual's dosimeter should be relocated to the portion of the whole body likely to receive the highest dose. Dosimeter relocation should be conducted in conformance with facility procedures or specific work authorizations, such as RWPs. The technical basis document should describe the methodology used in determining the dose of record when multiple dosimeters are used and when dosimeters are relocated.

High-Radiation Areas

Individuals entering a high radiation or very high radiation area shall be monitored by a supplemental dosimeter or other means of determining the individual's effective dose during the entry (see Article 334 for entry requirements) [see 10 CFR 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 gamma 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.

Contact Work with Radioactive Materials/Sources

The following is taken from DOE-STD-1128-98.

Most plutonium operations involve contact work, so increasing the distance may not always be practical. But significant reductions in doses can be achieved by reducing plutonium inventories in glove boxes. It is good practice not to store plutonium samples in glove boxes, but to remove them to storage vaults or other shielded locations. In many cases, the plutonium samples can be stored in the glove box in "wells" or specially shielded areas at some distance from the work areas where the plutonium technicians spend most of their time. The best method of reducing neutron dose is simply to remove the plutonium from the glove box and minimize inventories in the glove box.

Area monitoring in the workplace shall be routinely performed, as necessary, to identify and control potential sources of personnel exposure (10 CFR 835.401(a) (6)). This monitoring should include surveys in areas that are not ordinarily expected to be contaminated. The program should define minimum requirements, survey types, and frequencies. Surveys should be performed at frequencies adequate to identify changes in posting required or an activity buildup, and to ensure that current radiological controls are appropriate. The surveys suggested by this section are minimum recommendations; additional surveys should be conducted, recorded, and reviewed as necessary to ensure full protection of personnel.

c. Discuss the hierarchy of controls used to prevent uptakes of radioactive material by personnel, and potential worker hazards associated with implementation of these controls.

The following is taken from DOE-STD-1098-2008.

The primary methods used to maintain individual internal doses ALARA shall be physical design features, such as confinement, ventilation, and remote handling [see 10 CFR 835.1001(a)]. The design objective shall be, under normal conditions, to avoid releases of radioactive material to the workplace atmosphere. The objective, under all conditions, shall be to control inhalation of radioactive material to levels that are ALARA [see 10 CFR 835.1002(c)].

Administrative controls, including access restrictions and the use of specific work practices designed to control airborne radioactivity, shall be used as the secondary method to maintain internal doses ALARA [see 10 CFR 835.1001(b)].

When engineering and administrative controls have been applied and the potential for airborne radioactivity still exists, respiratory protection should be used to limit internal exposures. Use of respiratory protection should be considered under the following conditions:

- Entry into airborne radioactivity areas
- During breach of contaminated systems or components
- During work in areas or on equipment with removable contamination levels greater than 100 times the values in table 2-2 of DOE-STD-1098-2008
- During work on contaminated or activated surfaces with the potential to generate airborne radioactivity.

The selection of respiratory protection equipment should include consideration of worker safety, comfort, and efficiency. The use of positive pressure respiratory protection devices is recommended wherever practicable to alleviate fatigue and increase comfort. See chapter 5, part 3, for additional guidance on respiratory protection.

In specific situations, the use of respiratory protection may be inadvisable due to physical limitations or the potential for significantly increased external exposure. In such situations, and if the anticipated internal dose is likely to exceed 0.1 rem, a formal radiological review should be conducted to ensure measures are implemented to assess available options, monitor and reduce worker exposure, and provide for follow-up monitoring, as required. The rationale for not requiring respiratory protection**,**, including a description of measures taken to mitigate the airborne radioactivity, should be documented as part of the review process.

The following controls are applicable to activities authorized in accordance with the above:

- Stay time controls to limit intake should be established for the entry
- Evaluation of workplace airborne radioactivity levels should be provided through the use of continuous air monitors or air samplers with expedited assessment and analysis of results.

When notified that an individual with an open wound wishes to enter an area where contact with radioactive contamination is possible, a representative of the radiological control

organization or medical services should examine the wound and require appropriate measures to prevent the entry of radioactive contamination. These measures may be range from requiring an appropriate bandage or other covering up to prohibiting access to affected areas until the wound has healed. If other (non-radiological) hazards are present in the area to be entered, the individual should be directed to contact the applicable safety personnel.

d. For a radiological incident (i.e., spill, loss of containment), discuss the potential and magnitude of the following:
- **Loose surface contamination levels**
- **Airborne radioactivity levels**

The following is taken from DOE-HDBK-1122-99.

A radiological incident is an unplanned event involving radiation or radioactive materials (part of an emergency). The response taken to an incident is usually governed by normal procedures. Emergencies are classified as either an alert, site area emergency, or general emergency, in order of increasing severity, when events occur that represent a specific threat to workers and the public due to the release or potential release of significant quantities of radiological and non-radiological hazardous materials. Classification aids in the rapid communication of critical information and the initiation of appropriate time-urgent emergency response actions.

Loose Surface Contamination Levels

The following actions to be taken by a Radiological Control Technician are taken from DOE-HDBK-1122-99.

- Stop the spill. Take appropriate precautions that are dependent on the situation.
- All spills are different. Correct the situation immediately if possible without taking undue risks.
- Warn other personnel. Let people around know what is going on. If the situation warrants, evacuate the area. Notify your supervisor, facility management, and emergency response network if appropriate. As before, whether or not to activate a site emergency response program (such as dialing 911) is determined by the nature of the incident. Activation usually automatically fulfills this requirement. When a situation is confusing, not fully understood, or may not be controllable; over reacting is better than under reacting.
- Isolate the area. Establish boundaries and post the area around the spill area for exposure and contamination control.
- Minimize exposure to yourself as well as others. Practice ALARA principles and use all protective gear available.
- Secure ventilation by controlling HVAC (heating, ventilation, air conditioning). Unless one is certain that ventilation is contributing to the incident, this may involve no more than just ensuring that conditions are correct for normal designed ventilation.
- Follow through as necessary by starting and collecting air samples as may be indicated, surveying for contamination, and decontaminating. The cleanup of major spills may very likely involve many people and require radiation work permits and ALARA reviews of activities. Do not try to clean up a major spill by yourself, just keep it contained and isolated until the entire clean up operation is formulated. Complete all documentation of surveys and logs.

If you are unsure if you can contain the spill, or if you do not know the nature of the spill, use the "WIN" process:

- Warn others
- Isolate the area, keep personnel out
- Notify authorities

Airborne Radioactivity Levels

Initial Response

- Instruct affected worker to remain in area (standfast).
- Report to the scene with at least portable instruments for direct surveys and smear media.
- Perform whole body surveys (frisk) for the appropriate type of radiation (alpha
- and/or beta-gamma).
- Take actions to minimize cross-contamination, such as covering or placing a glove over a contaminated hand.

Supplemental Actions

- Survey affected area to characterize the extent of contamination.
- Suspect an up-take if contamination is verified and survey facial area for contamination, taking nasal smears or nose blows. If positive, contact RCT supervision and refer to your facility specific procedures.
- If contaminated, follow-up actions include saving any radioactive material pertaining to the contamination event, as this may help characterize the event at a later time.
- Refer to facility specific procedures if contamination persists.
- Document all surveys and estimate skin dose on proper forms. Do not unduly delay any decontamination efforts by taking too long in documenting contamination for skin dose estimates. Remember that dose is being incurred all the time that the skin is contaminated. Think ALARA especially in the case of high energy beta emitters.
- Report all confirmed skin contaminations to RCT supervision and refer to your facility specific procedures if transporting to a medical facility.
- Gather appropriate information for follow-up surveys.

e. Discuss appropriate personal protective equipment, including respiratory protection for subsequent entry into and decontamination of the above area.

The following is taken from DOE-STD-1098-2008.

Individuals shall wear protective clothing during work in contamination and high contamination areas [see 10 CFR 835.1102(e)] and should wear protective clothing during the following activities:

- Handling of contaminated materials with removable contamination in excess of Table 2-2 of DOE-STD-1098-2008 levels
- Work in airborne radioactivity areas
- As directed by the radiological control organization or as required by the RWP or other work authorization.

Protective clothing and shoes designated for radiological control should be:

- Distinctive

- Used only for its intended purposes.

Workers should proceed directly to the designated area after donning personal protective equipment and clothing.

General guidelines for protective clothing selection and use are provided in appendix 3C and in Table 3-1 of DOE-STD-1098-2008.

The use of labcoats as radiological protective clothing is appropriate for limited applications, such as those discussed in appendix 3C where the potential for personal contamination is limited to the hands, arms, and upper front portion of the body. Labcoats should not be used as protective clothing for performing physical work activities in contamination, high contamination, or airborne radioactivity areas where contamination of the lower legs is likely.

As appropriate for the work conditions, the radiological control organization should consider posting instructions for donning and removing protective clothing at the dress-out areas and step-off pad(s) for the affected work areas.

The use of personal protective equipment or clothing (including respiratory protection) beyond that authorized by the radiological control organization or other cognizant safety authorities detracts from work performance and is contrary to ALARA principles and waste minimization practices. Such use should not be authorized.

For radiological control purposes, company-issued clothing that is not specifically intended to protect individuals from contamination hazards should be considered the same as personal clothing.

The following is from DOE-STD-1128-98.

Respiratory protection should be readily available. Respiratory protective equipment should be used for all bag-out operations, bag and glove changes, and any situation involving a potential or actual breach of confinement. Alternatively, the operation could be performed in a glovebag to maintain confinement. In any case, protection, in the form of air-purifying or atmosphere-supplying respirators, should be considered whenever concentrations of radionuclides in the air are likely to exceed 30% of the DAC. For good performance, the respirator must fit closely on the facial contours and make an impenetrable seal so that all air enters through the filter or is supplied by the breathing-air supply. ANSI Z88.2-1992 (ANSI, 1980b), *Practices for Respiratory Protection,* describes qualitative and quantitative tests that should be used to ensure that the respirator fits the individual; only the quantitative test should be used for verification of respirator fit at plutonium facilities. Respirator fit tests should be performed annually.

The respiratory protective device selected should provide a protection factor appropriate for the air concentration anticipated. ANSI Z88.2 provides protection factors guidance. Air-supplied hoods are becoming more popular because a fitting is not required and facial hair does not prohibit their use. Protection factors greater than 1000 have been determined with air-supplied hoods. All respirators, including air-supplied hoods, require approval. While the

National Institute of Occupational Safety and Health (NIOSH) approves most respirators, some respirator types in use at DOE facilities are not part of the NIOSH testing program.

f. Using reference material and given the activity, calculate radiation levels from a point, line, and plane source.

This activity is performance based. The Qualifying Official will evaluate its completion. The following is taken from DOE-HDBK-1122-99, and may be helpful.

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. The decrease of exposure rate with distance is very dependent on the type of geometry. There are three types of geometries commonly encountered in the field: point sources, line sources, and plane sources. Each of these geometries uses a different formula to calculate exposure rate.

Point Source

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, especially when working with point sources, increasing the distance from the source is more effective than decreasing the time spent in the radiation field.

Theoretically, a point source is an imaginary point in space from which all the radiation is assumed to be emanating. While this kind of source is not real (all real sources have dimensions), any geometrically small source of radiation behaves as a point source when one is within three times the largest dimension of the source. Radiation from a source is emitted equally in all directions. Thus, the photons spread out to cover a greater area as the distance from the point source increases. The effect is analogous to the way light spreads out as we move away from a single source of light, such as a light bulb.

The radiation intensity for a point source decreases according to the inverse square law, which states that as the distance from a point source decreases, or increases, the dose rate increases, or decreases by the square of the ratio of the distances from the source. The inverse square law becomes inaccurate close to the source (i.e., within three times the largest dimension of the source).

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

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

where:
I_1 = exposure rate at 1st distance (d_1)
I_2 = exposure rate at 2nd distance (d_2)
d_1 = 1st (known) distance
d_2 = 2nd (known) distance

The assumptions are that the attenuation of the radiation in the intervening space is negligible, and the dimensions of the source and the detector are small compared with the distance between them.

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. Due to distance constraints, exposures at certain distances from some sources, such as for a pipe or tank, cannot be treated as point sources. In these situations, these sources must be treated as line sources or large surface sources.

Line Source

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:

$$l_1 (d_1) = l_2 (d_2)$$

i.e. 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.
- Valid to a point that is one half the distance of the longest dimension of the line source (L/2), beyond which the point source formula should be used.

Plane Source

Planar, or surface, sources of radiation can be the floor or wall of a room, a large cylindrical or rectangular tank, or any other type of geometry where the width or diameter is not small compared to the length. Accurate calculations for these types of sources require the use of calculus; however, a relationship can be described for how exposure rate varies with distance from the source.

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/d2.

The exposure rate versus distance calculations can be used to make an estimate of the radiation intensity at various distances. These estimates are valuable tools to estimate and verify the readings obtained from exposure rate meters.

g. Given buildup factors and half value layers, perform shielding calculations.

The following is taken from DOE-HDBK-1122-99.

When shielding against x-rays and gamma rays, it is important to realize that photons are removed from the incoming beam on the basis of the probability of an interaction such as

photoelectric effect, Compton scatter, or pair production. This process is called attenuation and can be described using the "linear attenuation coefficient", μ, which is the probability of an interaction per path length x through a material (typical units are 1/cm or cm^{-1}). The linear attenuation coefficient varies with photon energy, type of material, and physical density of material. Mathematically the attenuation of photons is given by:

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

where:

I(x) = Radiation intensity exiting x thickness of material
I_o = Radiation intensity entering material
e = Base of natural logarithms (2.714.....)
μ = Linear attenuation coefficient
x = Thickness of material.

This equation shows that the intensity is reduced exponentially with thickness and only approaches zero for large thicknesses. I(x) never actually equals zero. This is consistent with the notion that because x-rays and gamma rays interact based on probability, there is a finite (albeit small) probability that a gamma could penetrate through a thick shield without interacting. Shielding for x-rays and gamma rays then becomes an ALARA issue and not an issue of shielding to zero intensities.

The formula above is used to calculate the radiation intensity from a narrow beam behind a shield of thickness x, or to calculate the thickness of absorber necessary to reduce radiation intensity to a desired level. Tables and graphs are available which give values of μ determined experimentally for all radiation energies and many absorbing materials. The larger the value of μ the greater the reduction in intensity for a given thickness of material. The fact that lead has a high μ for X- and gamma radiation is partially why it is widely used as a shielding material.

Although attenuation of the initial beam of photons occurs by photoelectric, Compton, and pair production interactions, additional photons can be produced by subsequent interactions (immediately in the case of Compton). If the beam is narrow, these additional photons are "randomized" and are no longer part of the narrow beam of radiation. If the beam is broad, photons can be "randomized" and scattered into the area one is trying to shield. The secondary photons are accounted for by a build up factor, B, in the attenuation equation as follows:

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

where B is the buildup factor. Tables of dose build-up factors (indicating that the increased radiation intensity is to be measured in terms of dose units) can be found in the Radiological Health Handbook.

The buildup is mostly due to scatter. Scattered radiation is present to some extent whenever an absorbing medium is in the path of radiation. The absorber then acts as a new source of radiation. Frequently, room walls, the floor, and other solid objects are near enough to a source of radiation to make scatter appreciable. When a point source is used under these conditions, the inverse square law is no longer completely valid for computing radiation intensity at a distance. Measurement of the radiation is then necessary to determine the potential exposure at any point.

The simplest method for determining the effectiveness of the shielding material is using the half-value layer (HVL) concept. One HVL is defined as the amount of shielding material required to reduce the radiation intensity to one-half of the unshielded value.

$$HVL = \ln 2/\mu = 0.693/\mu$$

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

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.

$$TVL = \ln(10)/\mu = 2.3026/\mu$$

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.

h. Using reference material and given a scenario including bioassay results, isotopic and chemical form etc., calculate the internal dose to be assigned to an individual.

Note: This activity is performance based. The Qualifying Official will evaluate its completion.

7. Radiation protection personnel shall demonstrate a working level knowledge of ALARA principles, and their application to radiological work activities.

a. Describe the various components of an effective ALARA program, including operations, engineering, and management controls.

The following is taken from 10 CFR 835.2.

ALARA means as low as is reasonably achievable, which is the approach in radiation protection to manage and control exposures (both individual and collective) to the workforce and to the general public to as low as is reasonable, taking into account social, technical, economic, practical, and public policy considerations. ALARA is not a dose limit, but a process which has the objective of keeping doses as far below the applicable limits as is reasonably achievable.

The following is taken from DOE-HDBK-1122-99.

The DOE and other regulating agencies have through their Orders and CFRs mandated that there should not be any occupational exposure of workers to ionizing radiation without the expectation of an overall benefit from the activity causing the exposure. All personal radiation exposure shall be maintained As-Low-As-Reasonably Achievable (ALARA).

Every facility and the DOE Complex, as a whole, will strive to keep radiation exposure to the work force and public well below regulatory limits and ensure that there is no radiation exposure without commensurate benefit. For a facility to meet its ALARA objectives it should first establish a program to maintain exposures ALARA. This program should be updated throughout the evolution of the facility. The program should consider:

- Design and modification of the facility and selected equipment and components to integrate the ALARA concepts.
- Updating the ALARA procedures and plans to reflect the current need of the facility.
- The availability of equipment, instrumentation and facilities necessary for ALARA program.
- Training facility workers and management as well as radiological control personnel in
- ALARA programs and reduction techniques.
- Overall cost or job completion benefit versus the risk involved in receiving the exposure.

Implementation of ALARA concepts should be carried out through all phases of a facilities lifetime. ALARA Program concerns include:

Operations

- Inspection tour - access, mirrors, visibility
- In-service Inspections - use of remote control equipment, TV, snap on insulation, platforms, etc.
- Remote readout instrumentation
- Remote valve/equipment operators
- Sampling stations, piping, valving, hoods, sinks

Engineering

- The discharge of radioactive liquid to the environment
- Control of contamination
- Efficiency of maintenance
- Ease of decontamination and operations
- Selection of components to minimize the buildup of radioactivity
- Provision of support facilities for donning and removing protective clothing and for personnel monitoring
- Shielding requirements
- Ergonomics considerations
- Access control designed for hazard level
- Surfaces that can be decontaminated or removed
- Equipment that can be decontaminated

Management

Each individual involved in radiological work must demonstrate responsibility and accountability through an informed, disciplined and cautious attitude toward radiation and radioactivity. Every one has responsibilities in this area.

- Design and implement ALARA program
- Provide resources such as tools, equipment, adequate personnel
- Create and support ALARA Review Committee
- Approve ALARA goals
- Design and implement worker training

b. Describe how optimization techniques, including cost-benefit analysis, are used in the ALARA process.

The following is taken from ICRP Publication 37, *Cost-Benefit Analysis in the Optimization of Radiation Protection.*

This report is concerned primarily with the optimization of radiation protection, and with the rationale and techniques to establish what is reasonably achievable in the control of radiation exposures.

A wide range of techniques is available to optimize radiation protection. Some of these techniques are drawn from operational research, some from economics, and some from engineering. The use of a given technique implies, explicitly or implicitly, value judgments about the possible objectives of optimization.

Techniques for use in the optimization of radiation protection include, but are not confined to, the procedures based on cost-benefit analysis and it is these procedures that are discussed in detail in this report. It is important to recognize that other techniques, some quantitative, some more qualitative, may also be used in the optimization of radiation protection.

Optimization of radiation protection applies to all situations where radiation exposures can be controlled by protection measures. It is an important consideration in the selection and design of protection systems for installations and equipment, and in the performance of operations. Although optimization, as a component of the system of dose limitation, applies strictly to situations where the radiation source is under control, similar considerations can be used for planning some of the protective actions to be taken in case of an accident.

c. Discuss the essential elements of the job planning process and the post-job ALARA review for work performed in a radiation or radioactive contamination area.

The following is taken from DOE-HDBK-1122-99.

To ensure the ALARA concepts are incorporated into radiological work, it is often necessary to conduct ALARA reviews. These reviews can be conducted in many different ways and each facility may have its own preferences. Regardless of what each facility calls them, they are an important part of maintaining exposures at a reasonable level. These reviews are called by various names; the following is an attempt to outline what they are and how they are used.

Jobs determined by procedure and by unusual circumstances may require a post-job ALARA review. This will ensure the overall effectiveness of job planning and implementation. These reviews are handled usually by the ALARA group. The RCT will participate by giving survey data and their opinions on how the job was performed and ways it may be performed better in the future. It is important to note that the good portions of the job must be stressed. This process is not just for the things that went wrong.

Unusual exposure events are investigated in this process to determine the root cause of things that did go wrong. Recommendations are made and corrective actions are then taken to prevent future reoccurrences of these events.

d. Describe the various radiological performance indicators that are applicable to the ALARA process.

The following is taken from DOE-STD-1098-2008.

The radiological control manager or designee should provide a periodic summary report to the contractor senior management for sites which exceed an annual collective dose of one person-rem. This report should include feedback on the radiological control goals established in accordance with Article 131. Examples of performance indicators that provide a more detailed analysis of performance are identified in following table. The periodic report should provide current performance indicators, as well as tracking and trending for the prior twelve-month period.

The radiological control manager should provide appropriate performance indicator information to supervisors and managers on a frequent enough basis to permit management of radiological control performance. The frequency should be consistent with the nature of the workload and the potential for not achieving the established goals.

To promote worker awareness of radiological control performance, supervisors should consider placing in the workplace selected indicators related to their work group should be posted in the workplace.

Table 3. Suggested radiological control performance indicators

Exposure control a. Collective dose in person-rem b. Average worker dose in rem c. Maximum dose to a worker in rem d. Number of unplanned exposures resulting in doses greater than the facility administrative control level e. Number of dose assessments for lost or damaged dosimeters
Personnel contamination a. Number of skin and personal clothing contaminations b. Number of contaminated wounds c. Number of facial contaminations
Control of internal exposure a. Number of unplanned intakes b. Number of airborne events c. Number of alarms on airborne monitors (actual and false) d. Number of airborne radioactivity areas e. Area of airborne radioactivity areas in square feet
Control of contaminated areas in operational areas a. Number of contamination and high contamination areas b. Area of contamination areas in square feet c. Area of high contamination areas in square feet d. Number of spills requiring posting of an area
Minimization of radioactive waste a. Volume and activity of radioactive waste in cubic feet and curies, respectively b. Number of cubic feet not subject to volume reduction by incineration, compaction, or other means
Control of radioactive discharges a. Activity of liquid radioactivity discharges in curies b. Activity of airborne radioactivity discharges in curies

Source: DOE-STD-1098-2008

e. Calculate person-rem estimates and use the results in ALARA cost-benefit analysis.

Note: This activity is performance based. The Qualifying Official will evaluate the analysis. The following is taken from DOE-HDBK-1110-2008 and may be helpful.

In a cost-benefit analysis (CBA), all of the items to be considered must be expressed in the same units, usually dollars. Thus, to convert doses in a CBA to dollars, we must have a conversion factor. Such a conversion factor is called the (dollar) value of a person-rem. There

may be actually several different values for different purposes. DOE has specified that this value(s) is to be set on a site-by-site basis; at (your site), this value(s) is $ per person-rem.

For comparison, note that NUREG-1530 states that the NRC now uses a value of $2,000 per person-rem, makes future value adjustments, and restricts the basis of this figure to health effects only.

The point of optimization is to maximize the net benefit of an activity or design or of a feature of an activity or design. Thus, we have the following general equation:

$$B = V - P - X - Y$$

where B is the net benefit, V is the gross benefit, P is the costs of production not including radiation protection costs, X is the radiation protection costs, and Y is the dose cost. As can be seen, we can maximize B by maximizing V and minimizing P, X, and Y.

In practice, it is sometimes difficult to place a value on benefits and costs; for example, what is the value of a research activity that does not produce a marketable "product" but does produce useful information? For this reason, we try to look at only those components of B (i.e., V, P, X, and Y) that change and compare those. For example, if the only question is "one foot or two" of shielding and we can rule out effects on efficiency of operation, pipe routing, etc., as insignificant, then we could look at only X and Y. This is true because with more shielding the cost goes up but the dose goes down, and vice versa. In this case, V and P would not depend on the value of the thickness, and we could write

$$dB/dt = d/dt\ (V - P - X - Y) = -\ d/dt\ (X+Y)$$

We can set this equal to zero so as to solve for the value of t that makes B a maximum and the sum of X + Y a minimum. Note that neither X nor Y is necessarily a minimum, but rather the sum must be a minimum. This approach works where X and Y (and possibly V and P as well) are functions of a continuous variable. There are four main steps in performing a CBA, as given below.

1. Describe each feature or measure, including the status quo where applicable. This should include the radiological implications of its adoption.
2. Estimate or calculate the applicable costs, benefits, and doses for each feature or measure. If any dose exceeds a regulatory limit or a contract provision, or there is any other such absolute barrier, then the feature or measure can be ruled out at once and the analysis need not be completed.
3. Determine the net benefit or a shorter cost expression of each feature or measure and then compare them. Or, where applicable, form the cost-dose difference ratio and compare it to the value of a person-rem.
4. Where the results are close or where there are uncertainties, perform a subjective factors analysis or a sensitivity analysis or both.

f. Discuss methods to minimize total effective dose equivalent (TEDE) by evaluating the trade-offs in considering the internal and external dose components.

The following is taken from DOE-HDBK-1110-2008.

Airborne radioactive materials can deliver both external and internal doses. External doses result from the worker being surrounded by a radioactive cloud; whereas internal dose results from the radioactive material entering the worker's body and organs. Protection measures against airborne radioactive materials include the following:

1. For external doses, a protective plastic suit can be worn as shielding against weakly penetrating radiation from airborne radioactive materials. This shielding will stop alphas and most betas and radioactive material, such as tritium, that can be also absorbed through the skin.
2. For internal doses, one can:
   - wear a respirator, or
   - wear a nonporous suit in atmospheres containing absorbable radionuclides.

3. Engineered features, which are the primary defenses against airborne radioactive materials, include:
   - ventilation cleanup systems,
   - liquid filtration and processing systems,
   - containment devices, and
   - airborne radioactive monitoring systems.

DOE's limits on airborne radioactivity are expressed in terms of the Derived Air Concentrations (DACs) that are given in 10 CFR 835, Appendices A and C. Breathing 1 DAC for 2,000 working hours (1 year) would result in the annual limit on intake (ALI), corresponding to 5 rem committed effective dose (CED) or 50 rem (CED or organ dose), whichever is more limiting. An equivalent DAC for a mixture of radionuclides can also be calculated. Areas with atmospheres containing a radionuclide or a mixture of radionuclides > the DAC or where an individual present in the area without respiratory protection could receive an intake exceeding 12 DAC-hours in a week must be posted as airborne radioactivity areas.

The 10 CFR 835 design objective for airborne radioactive material states that "under normal conditions, to avoid releases to the workplace atmosphere, and in any situation, to control the inhalation of such materials to levels that are ALARA." If airborne radioactivity cannot be avoided, it is best to design an operation to try to stay below the 10 percent DAC level. At levels >10 percent DAC, more restrictive administrative and/or increased engineered controls may become necessary.

The primary method of controlling airborne contamination should be to use reasonable engineering design features. Good work practices and contamination control at the source should also be performed (e.g., flushing a pipeline to remove the radioactive source prior to maintenance). Only when these features and controls are not feasible or effective (or while they are being evaluated) should respirators be prescribed.

To be ALARA, routine respirator use must be kept to a minimum. Design that requires the constant use of respirators in frequently or regularly occupied areas or during routine work is not acceptable.

In specific situations, the use of respiratory protection may not be suitable due to physical limitations or the potential for increased external exposure. It is often generally perceived

that personnel should not be allowed to work in an airborne radioactivity area without a respirator. However, brief entries in airborne radioactivity areas may result in doses negligible compared to the risk of heat stress or the risk associated with not doing the job. Also, it has been shown that in airborne radioactivity areas with elevated external dose rates, total doses (i.e., internal plus external) to workers without respirators may be lower than total doses to those workers with respirators. This is due to the extra length of time it takes the worker with respirators to perform tasks as a result of restricted movement, blurred vision, impaired breathing, and limited communication. In such cases, the radiological control staff and the operational supervisor may agree to waive respirator use.

g. Using knowledge of ALARA principles, discuss how to perform an evaluation of a radiation job plan and the associated worker job performance.

Note: This is a site-specific activity. The Qualifying Official will evaluate its completion.

8. Radiation protection personnel shall demonstrate a working level knowledge of the application of engineered radiological controls and facility design, including containment/confinement systems.

a. Discuss the general principles relating to the design and installation of radiation protection containment/confinement systems, including the following radiological protection considerations:
- **Layout design for nuclear facilities**
- **Design and selection of components for nuclear facilities**
- **Selection of materials and the associated surfaces for components used in radiological control areas**
- **Design, construction, and operation of containment/confinement systems to minimize internal radiation exposure, including:**
  - **Engineered ventilation**
  - **Engineered containment**
  - **Hot cells**
  - **Radioactive liquid and solid waste processing facilities**
- **Design, construction, and operation of systems that minimize personnel external radiation exposure, including:**
  - **Shielding**
  - **Interlock systems**

Layout Design for Nuclear Facilities

The following is taken from 10 CFR 835.1001.

Measures shall be taken to maintain radiation exposure in controlled areas ALARA through engineered and administrative controls. The primary methods used shall be physical design features (e.g., confinement, ventilation, remote handling, and shielding).

Administrative controls shall be employed only as supplemental methods to control radiation exposure. For specific activities where use of engineered controls is demonstrated to be impractical, administrative controls shall be used to maintain radiation exposures ALARA.

Design and Selection of Components for Nuclear Facilities

The following is taken from DOE-STD-1189-2008.

This Standard provides the Department's expectations for incorporating safety into the design process for new or major modifications to DOE Hazard Category 1, 2, and 3 nuclear facilities, the intended purpose of which involves the handling of hazardous materials, both radiological and chemical, in a way that provides adequate protection for the public, workers, and the environment. The Standard describes the Safety-in-Design philosophies to be used with the project management requirements of DOE O 413.3A, Change (Chg) 1, *Program and Project Management for the Acquisition of Capital Assets*, and incorporates the facility safety criteria in DOE O 420.1B, *Facility Safety*, as a key foundation for Safety-in-Design determinations. The requirements provided in the above DOE Orders and the expectations in this Standard provide for identification of hazards early in the project and the use of an integrated team approach to design safety into the facility. The basic Safety-in-Design precepts are as follows:

- appropriate and reasonably conservative safety structures, systems, and components are selected early in project designs;
- project cost estimates include these structures, systems, and components; and
- project risks associated with safety structures, systems, and components selections are specified for informed risk decision-making by the Project Approval Authorities**.**

The provisions of this Standard, when implemented in conjunction with DOE O 413.3A, Chg 1, and its guidance documents, are consistent with the core functions and guiding principles of Integrated Safety Management (ISM), as described in DOE P 450.4, Integrated Safety Management Policy.

Selection of Materials and the Associated Surfaces for Components Used in Radiological Control Areas

The following is taken from DOE-STD-1128-98.

In general, the design features that aid in contamination control during operation also facilitate decommissioning. The inclusion of all the building materials suggested in this section may be cost-prohibitive, but they should be considered if the budget allows. The maintenance procedures that are used during operation are also important in controlling the spread of contamination to clean areas and, therefore, they facilitate decommissioning, too.

Less permeable building materials are more easily decontaminated. Any concrete with uncoated surfaces that comes in contact with plutonium solutions or plutonium contaminated air will require surface removal and disposal as radioactive waste at the end of its life. If there are cracks through which contaminated solutions have penetrated, the entire structure may need to be disposed of as radioactive waste.

Metal surfaces may also require decontamination. In general, the more highly polished the surface, the easier it will be to decontaminate. If feasible, all stainless steel that will come into contact with plutonium should be electropolished before being placed into service. If high-efficiency particulate air (HEPA) filtration has failed at any time during facility operation, roofs may require decontamination.

Metal roofs are easiest to decontaminate, but even these may contribute to the volume of radioactive waste unless unusual measures are taken to clean them. Built-up and composition roofs will be difficult to clean to unrestricted release levels.

Interior surfaces are most easily cleaned if they were completely primed and painted before the introduction of radioactive materials into the facility. If interior surfaces are repainted during operation, their disposal as clean waste is likely to require removal of the paint. However, if the paint has deteriorated, cleaning for unrestricted use may be as difficult as if the material had never been painted. Wood will almost certainly become contaminated, as will plasterboard and other such materials.

Floor surfaces are likely to be a problem. Concrete should be well sealed and covered with a protective surface. Single sheet, vinyl flooring with heat-sealed seams is preferable to asphalt or vinyl tile because it is more easily cleaned. If the floor needs resurfacing, it is preferable to overlay new flooring material rather than remove the old material and expose the underlying floor.

Design, Construction, and Operation of Containment/Confinement Systems to Minimize Internal Radiation Exposure, including:

Engineered Ventilation

The following is taken from DOE-HDBK-1145-2008.

Ventilation systems are engineered so that air flows from areas of low contamination to areas of greater contamination. Air balance is maintained by using damper controls, air locks, and backup safety systems. Air balance is also maintained by controlling the position of the inside doors. Ventilation control doors should not be blocked open, or ventilation balance could be lost. Also, do not operate any equipment unless you have been trained and authorized to do so.

Devices such as tents or glovebags are used to provide local containment for maintenance activities. These containments normally have local ventilation and exhaust filtration. A bagless transfer system for moving items out of gloveboxes has been developed for use throughout the complex. The best way to maintain contamination control after a loss of containment is to decontaminate to low or non-detectable levels. However, in some instances, contamination must be fixed in place. This is usually done by painting the surfaces of gloveboxes, walls, floors, etc. Because there is a potential for contamination control problems if these surfaces are disturbed, individuals should not scrape surfaces or remove tape unless precautions are taken.

Engineered Containment

The following is taken from DOE-HDBK-1145-2008.

There are many different types of containments that, when used in conjunction with ventilation, help prevent the loss of material and thus minimize dose to the workers. Gloveboxes, tanks, and piping are examples of “primary containments,” because there are no system openings. Gloveboxes have ports with long plastic sleeves attached that allow material to be “sealed in” or “sealed out” from the glovebox without breaching the containment.

Types of equipment such as fumehoods are “primary confinements,” since they are the barrier closest to the source. Primary barriers require good ventilation to maintain contamination control. The room that encloses the primary system and is intended to provide

containment if the primary fails is called secondary containment. The building that encloses the system is the final barrier.

Hot Cells

The following is taken from DOE-HDBK-1132-99.

Hot cell exhaust systems considerations are as follows:

- Exhaust prefilters and HEPA filters should be installed to facilitate filter replacement and repair. Use of a bag-in/out type filter house can lessen personnel exposures.
- Standby filters should be considered to provide backup protection and facilitate primary filter replacement without shutting down the exhaust fans. Standby filters should be installed outside the cell and sealed in an acceptable enclosure for direct maintenance. Note: Air leakage through isolation valves/dampers should be evaluated to avoid the bypassing of filtration devices; the reduction of exhaust flow from recirculation through the standby filters; the exposure of personnel changing the isolated filter elements; and the premature loading of the standby filters.
- Exhaust systems should have monitors that provide an alarm if the concentration of radioactive material in the exhaust exceeds specified limits.
- If radioiodine may be present, consideration should be given to the installation of radioiodine-absorber units.

In facilities where plutonium or enriched uranium is processed, the following are additional considerations:

- Wherever possible, the designer should provide enclosures for confining process work on plutonium and enriched uranium. When these confinement enclosures are specified and designed, consideration should be given to whether room ventilation air for either a secondary or tertiary confinement can be recirculated. If recirculation ventilation system is provided, the design should provide a suitable means for switching from recirculation to once-through ventilation.
- If advantageous to operations, maintenance, or emergency personnel, the ventilation system should provide for independent shutdown. Such a shutdown should be considered in light of its effect on the airflow in other interfacing ventilation systems. When a system is shut down, positive means of controlling backflow of air to uncontaminated spaces should be provided by positive shutoff dampers, blind flanges, or other devices.
- Equipment to continuously monitor oxygen levels should be provided for occupied working areas of facilities equipped with significant quantities of inert or oxygen-deficient process glovebox lines. Allowable leakage rates for ductwork systems should be taken into consideration.
- The supply of air to primary confinement, such as enclosures that confine the processing of plutonium and enriched uranium, should be filtered by HEPA filters at the ventilation inlets to the enclosures and area confinement barriers to prevent the transport of radioactive contamination in the event of a flow reversal.
- If room air is recirculated, the recirculation circuit should provide at least one stage of HEPA filtration. The design should include redundant filter banks and fans. If recirculation systems are used, contaminated process enclosure air should be prevented from exhausting into the working area rooms. Process enclosure air (from

hoods, gloveboxes, etc.) should be treated and exhausted without any potential for recirculation to occupied areas.

- The designer should specify and locate components in the exhaust systems to remove radioactive materials and noxious chemicals before the air is discharged to the environment. These components should be capable of handling combustion products safely. Exhaust system design should safely direct effluents through the appropriate ventilation ducts and prevent spread beyond the physical boundary of the ventilation system until treated.
- HEPA filters should be installed at the interface between the enclosures that confine the process and the exhaust ventilation system to minimize the contamination of exhaust ductwork. Prefilters should be installed ahead of HEPA filters to reduce HEPA filter loading. The filtration system should be designed to allow reliable in-place testing of the HEPA filter and to simplify filter replacement.
- Separate exhaust ventilation system ductwork and the initial two stages of filtration should be designed for exhaust air from enclosures that confine the process (e.g., gloveboxes). These systems should maintain a negative pressure inside the enclosure with respect to the operating area. These systems should be designed to remove moisture, heat, explosive and corrosive gases, and other contaminants. These systems should also be designed to automatically provide adequate inflow of air through a credible breach in the enclosure confinement.
- Enclosures that confine the process and are supplied with gases at positive pressure should have positive-acting pressure-relief valves that relieve the exhaust system to prevent over-pressurization of the process confinement system.
- The design of air cleaning systems for normal operations, anticipated operational occurrences, and accident conditions should consider use of the following equipment as appropriate:
  - prefilters;
  - scrubbers;
  - HEPA filters;
  - sand filters;
  - glass filters;
  - radioiodine absorbers;
  - condenser distribution baffles; and
  - pressure and flow measurement devices.

Airborne contaminant cleaning systems should be designed for convenient maintenance and the ability to decontaminate and replace components in the supply, exhaust, and cleanup systems without exposing maintenance or service personnel to hazardous materials. Filtration systems should be designed so that a bank of filters can be completely isolated from the ventilation systems during filter element replacement.

Where the confinement system's ventilation ducting penetrates fire barriers, fire dampers should be appropriately used to maintain barrier integrity. However, the closure of such dampers should not compromise confinement system functions where the loss of confinement might pose a greater threat than the spread of fire. In such cases, alternative fire protection means (e.g., duct wrapping) should be substituted for fire barrier closure. In no case should a sprinkler system be considered a fire barrier substitute. (All penetrations of a

fire barrier should be sealed, including conduit, cable trays, piping, and ductwork. In the selection of seals, requirements for pressure and water-tightness should be considered.)

Radioactive liquid and solid waste processing facilities

The following is taken from DOE-HDBK-1110-2008.

Title 10 CFR 835 has a design objective that the design or modification of a facility and the selection of materials shall include features that facilitate operations, maintenance, decontamination, and decommissioning. Provision for decontamination must be made in the design of any component, system, or area where the potential for leakage of radioactive materials exists. Facilities may be released for public use if contamination levels meet established limits (not zero).

The methods one can use to control and contain radioactive sources are:

- Containment
  - Leak-tight or controlled-opening enclosure to keep radioactive materials confined within (e.g., fuel cladding, piping, hot cells, and curbing around tanks that contain radioactive materials). Examples of temporary containments are tents and glove bags.
  - Examples of structural containments for the control of radioactive materials are walls, windows, doors, floors, transfer ports, and ceilings, with appropriate gaskets, caulking, etc.
- Ventilation: Provision of air and other gas flow direction and rate such that airborne material and radioactive gases are captured and directed to filters and an appropriate release point. Note that negative pressure is an important aspect of control for some contaminants.
- Filtration: The capture of airborne material on a medium, thus confining them to a small and disposable volume.

Protective designs include such items as:

- Ventilated fume hoods,
- Gloveboxes for handling radioactive material,
- Exhaust systems,
- Water filtration and processing systems,
- Conservatively sized ventilation cleanup systems, and
- Double-walled pipes and tanks, canned pumps, and leak-tight valves.

Design, Construction, and Operation of Systems That Minimize Personnel External Radiation Exposure Including:

Shielding

The following is taken from DOE-HDBK-1110-2008.

The fifth ALARA design principle is to provide shielding between the worker and the radiation source by providing permanent or temporary shielding between sources and the workers. In general, any material through which ionizing radiation passes absorbs some or all of the radiation. This attenuation depends on the type and energy of the radiation, as well as the thickness and composition of the shielding material.

Design Factors. Considerable thought should be given to incorporating adequate shielding structures during the design phase of a nuclear facility. This shielding can be quite elaborate in some cases and may even consist of several layers of different materials best suited for different types of radiation.

Considerations for shielding design include:

- Anticipation of crud buildup and hot spots;
- Use of labyrinths for shielding penetrations and cubicle entrances;
- Installation of special shields such as hot spot covers, leaded windows, and shielded carts and forklifts;
- Allow for adequate space and access for installing temporary shielding for anticipated hot spots or in frequent jobs; and
- Use of appropriate shielding materials based on the type and level of radiation.

Shielding materials. The choice of shielding material depends on the type(s) of radiation to be shielded.

- Alpha (can be stopped by a single sheet of paper): Due to the extremely low penetrating ability of alpha particles, shielding is not considered necessary.
- Beta (can be stopped by 1/2-inch Plexiglas; 1/4-inch aluminum, wood, rubber): Due to the potential creation of bremsstrahlung when the beta particles are slowed or stopped, consideration must be given to shielding these x-rays whenever beta radiation is present.

This phenomenon is strongest when beta particles are stopped by materials with a high atomic number (such as steel or lead), making these materials inappropriate for shielding beta particles unless they are sufficiently thick to stop the bremsstrahlung also.

- Gamma (lead, concrete, steel): The denser the material, the better it is suited for attenuation of gamma and x-rays.
- Neutron (water, polyethylene, concrete, boron): Neutron-absorbing radionuclides (such as ^{10}Bo) and materials that contain large amounts of hydrogen make efficient neutron shields. The production of "capture gammas" in some materials must also be considered.

Fortuitous Shielding. Fortuitous shielding should be used when possible. Fortuitous shielding is material placed in an area for reasons other than shielding but acting as a shield because of its location, composition and thickness. Steel cabinets, steel security doors, concrete columns, and similar objects can serve as fortuitous shielding. These objects should be permanently mounted if relied on as shielding, however.

Sequence of Shielding. Shielding should be correctly layered for structural integrity and attenuation of different types of radiation. For example, with gammas and strong betas, a layer of plastic might precede a layer of lead, so that betas would be captured in the plastic and not produce bremsstrahlung in the lead.

Interlock Systems

The following is taken from DOE G 441.1-1C.

If an area radiation monitor is incorporated into a safety interlock system, the circuitry should be such that a failure of the monitor shall either prevent normal access into the area or operation of the RGD. Administrative procedures should be implemented to ensure that the RGD installation and the RGD safety interlock control devices are such that:

- radiation cannot be produced until the interlock system logic has been completely satisfied;
- production of radiation cannot be resumed by merely reestablishing the interlock circuit at the location where an interlock was tripped; and
- the safety circuit cannot be re-energized or reestablished automatically (i.e., there should be a manual safety circuit reset on or near the main control console).

b. Discuss the design and application of temporary engineered radiological controls.

The following is taken from DOE G 441.1-1C.

Temporary engineered controls are used to protect individuals when permanent controls cannot adequately contain radioactive material. When engineered controls, such as ventilation, vacuum cleaners, or containment devices, are used to reduce or maintain airborne radioactivity concentrations, air monitoring should be performed to determine the adequacy and effectiveness of the engineered controls. Generally, for installed controls, such as fume hoods, fixed-location air sampling is preferred, whereas for temporary controls, such as portable ventilation or use of vacuum cleaners, grab sampling is preferred. Real-time air monitoring for determining the adequacy of installed controls may also be appropriate or required. Chapters 4 and 11 of this Guide and the RCS provide guidance on the use of engineered controls.

The following is taken from DOE-STD-1098-2008.

The installation, use, and removal of temporary shielding needed to prevent exposure in high radiation areas should be controlled by postings or procedure. The effects of the additional weight of temporary shielding on systems and components should be evaluated and established to be within the design basis prior to installation. Installed temporary shielding should be periodically inspected and surveyed to verify effectiveness and integrity. Installed temporary shielding should be periodically evaluated to assess the need for its removal or replacement with permanent shielding.

Radiation surveys should be performed during the alteration or removal of installed temporary shielding as appropriate. Removable shielding needed to prevent access to a high radiation area should be visibly marked or labeled with the following or equivalent wording: “Radiation Shielding - Do Not Remove without Permission from Radiological Control.” Site procedures may identify specific shielding applications, such as the shielding of low activity sources or samples that fall outside the recommendations of this Article [314].

9. Radiation protection personnel shall demonstrate a familiarity level knowledge of the radiological hazards associated with the following and a working level knowledge for site specific radiological hazards:

- **Plutonium operations**
- **Uranium operations**
- **Tritium operations**

- **Nuclear explosive operations**
- **Production/experimental reactors**
- **Accelerator operations**
- **Waste handling/processing operations**
- **Decontamination and decommissioning**
- **Use of radiation generating devices**
- **Environmental restoration activities**

Note: The following knowledge, skills, and abilities are discussed in the order of the items listed in the competency statement. Not all KSAs apply to all the items and therefore may not be discussed.

a. Discuss the basic function and work activities associated with the above list.

b. Discuss fundamental characteristics of the major radiological hazards at the above listed activities. This could include discussion of:
- **Mode of decay**
- **Source**
- **Energies of major radiations emitted**
- **Relative principle biological hazard**
- **Half-life**

c. Discuss unique radiological exposure control techniques associated with the above listed activities.

Plutonium Operations

The following is taken from DOE-STD-1128-98.

Basic Function and Work Activities

Plutonium is the first man-made element produced on an industrial scale. The special nuclear properties of ^{239}Pu and ^{238}Pu have led scientists to focus their efforts on these two isotopes. The fission cross-section of ^{239}Pu makes it a useful energy source for atomic weapons and nuclear power reactors. The 87.7-year half-life of ^{238}Pu makes it an excellent heat source for space applications. Unfortunately, the same nuclear properties of plutonium that make it attractive to science also make this element hazardous to human beings. All 15 plutonium isotopes are radioactive, with half-lives ranging from 26 minutes for ^{235}Pu to 7.6 x 10^7 years for ^{244}Pu.

In the past, most plutonium in DOE facilities was produced for nuclear weapons and was composed of greater than 90 wt% ^{239}Pu and about 6 to 8 wt% ^{240}Pu. This material has been referred to as “weapons grade” or “low exposure” plutonium. It is produced on a large scale by irradiating ^{238}U in moderated production reactors. Plutonium has also been produced as a byproduct in the operation of research reactors, and commercial nuclear power plants. It is recovered and purified by solvent extraction and ion exchange processes. The resulting highly concentrated $Pu(NO_3)^4$ product solution is converted to a nonhygroscopic PuF_4 intermediate by one of the several processes before being reduced to metal with calcium. Plutonium is also produced from the waste streams of the conversion processes and scrap recovery operations, which include material from research and development efforts. Other

processes for reduction to metal include direct reduction of the oxide and electrolytic reduction.

Fundamental Characteristics of Major Radiological Hazards

The decay modes of some important plutonium and other isotopes and decay products are shown in Table 4. For brevity, only the most abundant radiations have been included in the table. Most of the isotopes are strong alpha-emitters, making alpha heating a problem for the storage and handling of large amounts of plutonium. Kilogram quantities of ^{239}Pu or gram quantities of ^{238}Pu can generate enough heat to melt plastic bags. Sources of ^{238}Pu must be handled with insulated gloves, and special precautions must be taken to ensure a good thermal heat sink during shipping and storage.

Table 4. Radioactive decay properties of selected isotopes and decay products, excluding spontaneous fission

Isotope	Half-Life	Mode of Decay Particle	Energy MeV	Yield, d%	X-ray [b] Energy MeV	Yield %	Gamma Ray MeV	Yield %
^{236}Pu	2.851 y	α	5.77	69.3	L's 0.011-0.021	13[c]	0.0476	6.6×10^{-2}
		α	5.72	30.6			0.109	1.2×10^{-2}
^{238}Pu	87.7 y	α	5.50	71.0	L's 0.011-0.021	15[c]	0.0425	3.95×10^{-2}
		α	5.46	28.8			0.0999	7.35×10^{-3}
^{239}Pu	2.41×10^{4} y	α	5.157	73.1	L's 0.0116-	5.0[c]	0.099	1.22×10^{-3}
		α	5.144	15.0	0.0215		0.129	6.41×10^{-3}
		α	5.106	11.8			0.375	1.55×10^{-3}
							0.414	1.46×10^{-3}
^{240}Pu	6564 y	α	5.168	72.8	L's 0.0115-	10.8[c]	0.0452	4.50×10^{-2}
		α	5.124	27.1	0.0215		0.104	7.08×10^{-3}
^{241}Pu	14.35 y	β	0.0052[d]	100.00	--	--	0.077	2.20×10^{-5}
		α	4.896	2.04×10^{-3}			0.1037	1.01×10^{-4}
							0.114	6.0×10^{-6}
							0.149	1.9×10^{-4}
							0.160	6.71×10^{-5}
^{242}Pu	3.73×10^{5} y	α	4.901	77.5	L's 0.0116-	9.1[c]	0.0449	3.6×10^{-2}
		α	4.857	22.4	0.0215		0.104	7.8×10^{-3}
^{241}Am	432.2 y	α	5.486	85.2	L's 0.0119-	42[c]	0.0263	2.4
		α	5.443	12.8	0.0222		0.0332	1.2×10^{-1}
		α	5.388	1.4			0.0595	35.7
^{237}U	6.75 d	β	0.039[e]	0.8	L's 0.0119-	70[c]	0.0263	2.43
		β	0.050[e]	3.4	0.0206		0.0595	34.5
		β	0.065[e]	51		53	0.0648	1.28
		β	0.069[e]	42	K's 0.097-0.114		0.165	1.85
							0.208	21.1
							0.268	7.1×10^{-1}
							0.332	1.2
							0.335	9.5×10^{-2}
							0.369	4.0×10^{-2}
							0.371	1.1×10^{-1}

Source: DOE-STD-1128-98.

The radiobiological properties of plutonium and other transuranic (TRU) elements are known primarily from experiments performed on rats, dogs, baboons, and rabbits. Human data on plutonium are limited.

Radioactive material can enter the body by four different pathways: by inhalation, through a wound, by ingestion, or by absorption through intact skin. These pathways may occur singly or in any combination.

- Inhalation is probably the most prevalent mode for occupational intake of plutonium. It also provides a generally conservative assumption of intake for designing bioassay programs.
- Wounds are potentially the most serious mode of intake because of the high dose-per-unit uptake of plutonium. Wounds can result from direct penetration by an object (i.e., a puncture or cut), from abrasion, or from burning by an acid, caustic, or thermal source.
- Occupational ingestion of plutonium poses a relatively small risk because the uptake factor from the GI tract to the blood is quite small and because most of the alpha energy from transformations within the GI tract is absorbed by the contents of the GI tract, rather than by the target tissues of the tract itself.
- Absorption of plutonium through intact skin is, for practical purposes, almost nonexistent. However, when removing skin contamination, care must be taken to ensure that the skin integrity is not damaged by rough or extensive decontamination procedures. If the skin integrity is damaged, the result can be considered a wound, regardless of how it occurred.

Three commonly encountered biokinetic models have been promulgated by the ICRP for the internal distribution and retention of plutonium. These models are identified by the ICRP publications in which they were first reported: ICRP 30, Part 1 (1979), ICRP 48 (1986), and ICRP 30, Part 4 (1988b). The models are similar with regard to the organs of significance, but differ with regard to the fraction of uptake deposited in the organ and its respective retention (or clearance) half-time in the organ. The three models represent the ones most widely used in dosimetry and in commercially available computer codes.

In all three ICRP models, once plutonium has reached the bloodstream, it is translocated primarily to the liver and skeleton. In the skeleton, it is deposited primarily on the endosteal surfaces of mineral bone, from which it is gradually redistributed throughout the bone volume by resorption and burial. Because of the extremely slow nature of this redistribution, plutonium is considered to be uniformly distributed over bone surfaces at all times following skeleton deposition. A small fraction of the translocated plutonium reaches the gonads. Although the gonadal fraction is different for males and females, the calculated gonadal doses are the same regardless of gender because the plutonium concentration in the tissues is assumed to be the same. The ICRP assumes that the remainder goes directly to excretion. Although the ICRP did not specifically state the fraction of systemic excretion occurring by urine as opposed to feces, a 0.5 fraction for each is often assumed.

Exposure Control Techniques

Special devices can be used to provide exposure control and/or containment when it may not be practical without them. These include temporary shields, tents or greenhouses, portable fans, ductwork and filters, and special fixtures to hold highly radioactive materials requiring detailed inspections, repairs, modification, or fabrication. Such devices can permit doing difficult work at low radiation doses, which might not be possible otherwise.

Vacuuming is particularly effective in dry-atmosphere and inerted enclosures where the levels of radioactive dust can quickly increase personnel exposure. In many cases, vacuuming reduces the exposure level more than a wipe down with a damp cloth and it can be done more quickly and with less waste material generated. Two factors weigh against vacuuming: possible safety hazards from electrical sparks, and the occasional difficulty of operating in inert atmospheres (although the last item need not be of importance). However, in dry glove boxes with dusty operations using high-exposure plutonium, personnel exposure control is a problem and vacuuming is a quick and effective method of keeping the dust and exposure rates under control and should be considered.

Uranium Operations

The following is taken from DOE-STD-1136-2004.

Basic Function and Work Activities

Naturally occurring uranium consists of a mixture of ^{234}U, ^{235}U and ^{238}U isotopes, along with their decay products. Uranium is relatively abundant in nature. The primary isotopes of uranium are long-lived alpha-emitters with energies between 4.15 and 4.8 MeV. Their progeny include numerous other radionuclides, some of which are radiologically significant at uranium facilities, the degree of significance depending upon the history of the uranium materials and the processing.

Through proper processing, uranium can be used as a fuel in nuclear reactors to generate electricity on a commercially-viable scale. The ^{235}U isotope is readily fissioned by slow, "thermal" neutrons with the release of a large amount of energy. The percentage of ^{235}U present (referred to as "enrichment") determines the fuel reactivity and the criticality hazard of the material. By concentrating the amount of the ^{235}Uisotope in the uranium, the quantity of fuel and the size of the reactor needed for production decreases. This concentration of natural uranium to enriched uranium is carried out by special processes such as gaseous diffusion, centrifuging, or laser separation. The uranium by-product of the enrichment process is reduced in ^{235}U content and is called "depleted" uranium. Uranium is commonly classified by its ^{235}U enrichment as natural uranium, enriched uranium, or depleted uranium.

The following is taken from DOE-HDBK-1113-2008.

Uranium is found in the earth's crust and is mined as ore. The average concentration is 2 parts per million (ppm) in the crust and less than 2 parts per billion (ppb) in the oceans. During the 1960's and 1970's, a program titled the Natural Uranium Resource Exploration was funded by the government to identify the locations of desirable uranium ore throughout the United States. It was determined that the most desirable locations of uranium are in the Colorado Plateau, the Wyoming Basin, and the flanks of the Black Hills in South Dakota. In those locations, the uranium concentration is much higher than 2 ppm. Uranium is also found on the African Continent. The ore is removed from either shallow open pits (less than 300-foot, or 100m, depths) or underground mines (greater than 300-foot depths). The typical uranium content of the ore is 0.15 - 0.3% and is in the form of U_3O_8, which is called "yellowcake." Uranium is also found in secondary minerals in the following forms: complex oxides, silicates, phosphates, and vanadates.

Uranium processing is dependent upon the desired product, but it generally involves the following cycle:

- mining and milling;
- conversion;
- enrichment;
- fabrication;
- use;
- waste disposal/storage; and
- decontamination and decommissioning.

The primary goal of the uranium fuel cycle process is to yield enriched uranium. This product can be used for:

- power reactors;
- research reactors;
- nuclear weapons; and
- naval propulsion reactors.

There are also a number of uses for uranium metal depleted in the ^{235}U isotope, such as:

- radiation shielding;
- armor-piercing bullets;
- catalysts for chemical reactions,;
- armor plating; and
- counter weights.

Depleted uranium typically is cast into ingots or billets, and then shipped to production facilities for appropriate reshaping.

Fundamental Characteristics of Radiological Hazards

As uranium goes through radioactive decay, it produces other radioactive elements known as radioactive decay products (also called progeny or daughter products). These radioactive decay products are also radioactive and have to be taken into account for radiological protection purposes.

Both alpha and beta particles are emitted as part of decay series. For example, ^{238}U decays by alpha emission to ^{234}Th; ^{234}Th decays by beta emission to ^{234}mPa; and so on, until stable ^{206}Pb is finally reached. Table 5 provides half-lives and natural percent abundance information for important uranium isotopes in the nuclear fuel cycle.

Table 5. Half-lives and natural percent abundance for important uranium isotopes in the nuclear fuel cycle

ISOTOPE	HALF-LIFE	NAT ABUND	IMPORTANCE
^{232}U	70 y	0%	An unwanted byproduct of ^{233}U production in a breeder reactor. Due to its much shorter half-life, ^{232}U contributes most of the radioactivity in samples of ^{233}U.
^{233}U	1.6×10^{5} y	0%	Manufactured by irradiating ^{232}Th with neutrons. It is a criticality hazard because it is fissile.
^{234}U	2.5×10^{5} y	0.0055%	A decay product of ^{238}U. It is concentrated with ^{235}U during enrichment. Highly enriched uranium contains about 1% ^{234}U. Most of the radioactivity of enriched uranium is from the ^{234}U.
^{235}U	7.1×10^{8} y	~0.7%	Fissile with slow neutrons; therefore, it is of primary interest for reactors and weapons. If not handled safely, an accumulation of ^{235}U could become critical.
^{236}U	2.3×10^{7} y	0%	Some ^{235}U is converted to ^{236}U in reactors. It is also present in reprocessed reactor fuel.
^{238}U	4.5×10^{9} y	~99.3%	The most abundant uranium isotope. It is fissionable with fast neutrons; however, it is not fissile (i.e., with thermal neutrons) so it is not a criticality hazard.

Source: DOE-HDBK-1113-2008.

Uranium has two naturally occurring decay series: the "actinium" series, which has ^{235}U as its parent; and the "uranium" series, which has ^{238}U as its parent. Many of our everyday encounters with radioactivity come from these decay series; examples are radon gas and radium. There are also man-made isotopes of uranium - ^{232}U and ^{233}U.

The decay products from these radionuclides must be considered in the implementation of a radiological control program at a facility where these uranium nuclides are present.

Criticality. Uranium is a fissionable material, which means it can undergo nuclear fission. Nuclear fission is a process in which a very heavy, unstable atom splits in two, or "fissions". When an atom fissions, one large atom primarily becomes two smaller atoms, between one and seven neutrons are given off (which may cause fission in nearby atoms), and a great deal of energy is given off as radiation and in other forms, such as kinetic energy of the fission fragments. The radiation created could result in the creation of radiological areas, such as high or very high radiation areas.

Toxilogical/biological effects. The principal entry of uranium into the human system is due to either inhalation or ingestion. Inhalation occurs either from release of volatile uranium compound or from suspension of volatile uranium-laden aerosols. Ingestion can occur when the uranium is introduced into water for consumption or the food chain by plant uptake. When uranium is either ingested or inhaled, it is removed from the body with a biological half-life varying between 6 and 5000 days, depending on which organ has become contaminated.

Uranium tends to concentrate in the kidneys and the bones. Additionally, if inhaled, the lungs are exposed. Internal exposure to uranium is controlled by limiting the ingestion and inhalation of this element. Most heavy metals, such as uranium, are toxic to humans depending on the amount introduced into the body. For short-term (acute) exposures, the toxicological effects are the primary concern, and acute exposures to significant amounts of uranium may result in kidney damage. However, as the enrichment of the uranium in the ^{235}U isotope increases, so too do the effects of radiation exposure in relation to toxicological effects.

Past industrial experience has proven that if there is a long-term exposure of small amounts of uranium (chronic exposure), the radiological effects are the primary biological concern. In fact, for chronic exposures, a development of tolerance against the toxicological effects may occur. The principal radiological hazard associated with uranium is due to the relatively high energy alpha particles its radionuclides and daughters emit. A chronic exposure to these radionuclides results in an increased risk of cancer, typically in the bones, kidney, and lungs, since these are the organs where uranium is deposited.

Exposure Control Techniques

[Note: Exposure control techniques for uranium are the same as for plutonium, described above, and not repeated here.]

Tritium Operations

The following is taken from DOE-HDBK-1105-2002.

Basic Function and Work Activities

There are three primary sources of tritium in our environment. Tritium is present in our environment from both man-made and natural sources as discussed below. Natural tritium is indistinguishable from man-made tritium. Tritium occurs naturally. It is formed by the reactions between cosmic rays and the nitrogen in the upper atmosphere. Nitrogen makes up 80% of the earths atmosphere. Cosmic rays generate approximately 4 million curies of tritium per year. With tritium being continually produced and at the same time decaying, the natural tritium in our environment is about 70 million curies.

The production of tritium from power reactors around the world is less than one-half that naturally formed (approximately 1 to 2 million curies a year).The amount of tritium in the world from weapons testing has been steadily declining since the 1970's when atmospheric testing was curtailed. Atmospheric testing from 1945-1975 produced about 8 billion curies. This has decayed to about 400 million curies. DOE has produced tritium at the Savannah

River Site with the use of a reactor. Tritium is commercially available from Canada and the European Community for non-weapons use.

Uses of tritium

- Consumer products
  - Gaseous tritium light sources
    - exit signs (1 Ci to tens of Ci's)
    - aviation landing aids (30-165 Ci per light) (1 curie of tritium has a mass of approximately 0.1 mg)
  - Luminizing industry:
    - self-luminous compounds for dials (several mCi's);and
    - controls as well as other general industry uses.
- Research - Tritium labeling
  - Tracers for medical and laboratory research.
- Department of Energy
  - Weapons development and applications
  - Fusion energy: As a fuel source

Radioactive decay of tritium

$^{3}_{2}$ He= Helium
υ = anti - neutrino
β^{-}= beta minus

Tritium "decays" by emitting a beta particle and becoming an atom of helium. A very low energy beta is emitted.

Max energy = 18.6 keV
Ave. energy = 5.7 keV

Tritium is not an "external" radiation hazard.

- Travels less than 1/4 inch in air
- Cannot penetrate through the dead layer of the skin
- Cannot penetrate clothing or gloves

Half-life

Radioactive half-life = 12.3 years.

Fundamental Characteristics of Radiological Hazards

Tritium can deliver a radiation dose if it gets inside our bodies. Modes of entry include:

- inhalation
- absorption
- injection (cuts/wounds)
- ingestion

Inhalation

Tritium gas: is only slightly incorporated into the body when inhaled. Most tritium gas inhaled is subsequently exhaled. There are other chemical and physical processes to convert tritium gas to tritiated water.

Tritiated water vapor: Nearly 100% of tritiated water vapor inhaled is incorporated into body fluids/tissues.

Hazard: Exposure to tritiated water is approximately 25,000 times more hazardous than exposure to elemental tritium gas.

Special Tritium Compounds: consist of organically bound tritium and tritium particulate aerosols. Exposure to organically bound tritium can be up to approximately 13 times more hazardous than exposure to tritium oxide (HTO). Exposure to tritium particulate aerosols can deliver up to approximately 20 times more dose to the whole body than exposure to HTO. The dose to the lungs from tritium particulate aerosols could be 2 orders of magnitude higher than from HTO.

Ingestion

Ingestion may occur by eating, drinking, chewing tobacco, and applying makeup where tritium contamination is present. Always wash hands thoroughly when leaving areas where there is a potential for contamination, and never eat, drink etc. where tritium contamination may be present.

Absorption

Absorption is also a hazard because an individual can receive, in certain situations, 1/3 of their uptake from absorption through the skin if not properly using personal protective equipment.

- Tritium gas: There is negligible skin absorption for tritium gas.
- Tritiated water: Tritiated water can be absorbed through the skin. It has been observed that moisture on hands enhances absorption.
- Solvents: Some solvents (organically bound tritium) can also go through the skin. Tritium will penetrate gloves; therefore gloves must be changed at a prescribed routine basis.

Exposure Control Techniques

The following is taken from DOE-HDBK-1079-94.

At most tritium facilities, the most commonly encountered forms of tritium are tritium gas and HTO. Other forms of tritium may be present, such as metal tritides, tritiated pump oil, and tritiated gases such as methane and ammonia. As noted earlier, deuterated and tritiated compounds generally have the same chemical properties as their protium counterparts, although some minor isotopic differences in reaction rates exist. These various tritiated compounds have a wide range of metabolic properties in humans under similar exposure conditions. For example, inhaled tritium gas is only slightly incorporated into the body during exposure, and the remainder is rapidly removed (by exhalation) following the

exposure. On the other hand, tritiated water vapor is readily taken up and retained in the body water.

Tritium released to room air moves readily with normal air current. The room or building ventilation system should be designed to prevent the air from being carried to uncontaminated areas, such as offices or other laboratories where tritium is not allowed. For that reason, differential pressure zoning is commonly used, and released tritium is directed outside through the building stack. In some newer facilities where the large quantities of tritium are being handled, room air cleanup systems are available for emergency use. Following a significant release, the room ventilation system is effectively shut down, the room is isolated, and cleanup of room air is begun.

The most important control for preventing a release of tritium to the room atmosphere is the use of containment around the source of tritium. This containment usually takes the form of a glove box, which is then a secondary containment if the tritium is already contained within the process plumbing, which is the primary containment. Even if the tritium is on the outside surface of a piece of equipment and located inside the glove box, through popular usage, the box is still referred to as the secondary containment.

At times, maintenance or repair work is done on equipment that cannot be moved into a glove box or fume hood and that has a high potential to release tritium. For these activities a temporary box ("tent"), may be constructed over the equipment, and an existing cleanup system installed to process the air. Alternatively, if the tritium at risk is not significant, the enclosing atmosphere may be purged to the stack. If the enclosure is small, gloves and glove ports may be fitted to the side of the enclosure. For larger enclosures entry may be required. In such cases, personnel must work in air-supplied suits inside the enclosure.

In spite of the greater protection afforded by glove boxes, fume hoods are commonly used at tritium facilities for handling or storing material with low quantities of tritium or with low-level contamination. Limits are generally imposed on the quantities used or stored in these hoods.

Fume hoods are also used to protect personnel at the outside door of glove-box pass boxes where materials are passed into and out of the boxes. Ideally, any tritium released in a hood from out-gassing or a leaky container, for instance, is routed to the hood's exhaust duct. However, turbulence may occur at the hood entrance, resulting in backwash and possible contamination of personnel if the face velocity is not adequate for the design of the hood, the activities in the hood, or the local conditions (such as traffic in front of the hood). No hood should be used that has not been thoroughly surveyed and judged acceptable for tritium use.

In general, only supplied-air respirators are effective in preventing inhalation of airborne tritium. Two types of air-supplied respirators are available: self-contained breathing apparatus and full-face supplied air masks. Because of the inherent disadvantages associated with respirators and other breathing apparatus, supplied-air plastic suits that completely enclose the body are often used by facilities that handle large quantities of tritium. Although they afford reasonably complete body protection, they are slow to don and cumbersome to wear. For these reasons, they are not favored for rescue work where time and mobility are

important considerations. For certain maintenance operations outside of glove boxes with a high degree of risk, supplied-air suits may be quite useful.

Nuclear Explosive Operations

The following definition is taken from DOE O 452.1C.

Basic Function and Work Activities

Nuclear explosive operation associated activities [are] activities directly associated with a specific nuclear explosive operation in a nuclear explosive area, such as work on a bomb nose or tail subassembly, even when physically separated from the bomb's nuclear explosive subassembly.

Fundamental Characteristics of Radiological Hazards

Note: Radiological hazards from nuclear explosive operations are site- and element-specific.

Exposure Control Techniques

Note: Exposure control techniques associated with nuclear explosive operations are site- and element specific.

Production/Experimental Reactors

Note: Information for KSAs "b" and "c" for the following discussion can be found in the previous discussion, and will not be repeated here.

Basic Function and Work Activities for Production Reactors

The following is taken from DOE-HDBK-1113-2008.

Uranium was used in plutonium production reactors. Uranium fuel and targets were coated with aluminum or zirconium metal and placed in the reactor. As they were irradiated with neutrons, a small fraction of the uranium was converted to plutonium. The irradiated fuel was then removed from the reactor, but the plutonium and uranium had to be separated from the fission products created during irradiation.

Basic Function and Work Activities for Experimental Reactors

The following is taken from DOE-STD-1128-98.

Plutonium and plutonium-uranium fuel mixtures were developed and tested in experimental reactors to prove the feasibility of operating power reactors. These fuels included both liquids and solids consisting of alloys and ceramic mixtures. Wick (1967) and Schneider and Roepenack (1986) provide comprehensive lists of fuels. Because of their pyrophoric nature, some of these alloys and compounds require special care and handling when exposed to reactive liquids or gases.

Accelerator Operations

The following is taken from DOE-HDBK-1108-2002

Basic Function and Work Activities

Accelerators are devices employing electrostatic or electromagnetic fields to input kinetic energy to molecules, atomic or subatomic particles. Accelerators also are capable of creating a radiological area and other radiological hazards.

In the early 1900s, radioactive particles could be obtained only from materials found in nature. The studies that physicists wanted to perform required both higher intensities and higher energies than were obtainable from the natural sources. The ability to vary energy and intensity to suit a particular experiment was also desirable. In the 1930s, scientists began to build machines that produced the needed degree of control. These machines were called accelerators.

The earliest accelerators were simple vacuum tubes in which electrons were given an increase in energy by the voltage difference between two oppositely charged electrodes. The acceleration of the electron by this electrical force also increases the energy of the electron. The amount of acceleration is determined by the potential difference measured in volts (V) in this electrical field. One electron volt (eV) is the energy gained by an electron accelerated through a potential difference of one volt. An electron accelerated across a gap by means of a 10,000 volt, or 10 kilovolt (kV), potential difference is said to have gained 10,000 electron volts (10 keV) of energy after crossing the gap.

One of the first machines to produce laboratory-accelerated particles was the Van de Graaff generator. The Van de Graaff consists of a polished metal sphere and a moveable belt. The function of the belt is to carry an electrical charge up to the sphere where it is stored. This process can be continued until a very high potential is developed in the sphere. In 1929, Van de Graaff built a pilot machine capable of generating 80,000 volts. In 1931, a 1.5 million-volt (MeV) machine was built at Princeton. A 25 MeV machine was built and operated at Daresbury Laboratory in the United Kingdom.

In 1932 in England, John D. Cockcroft and Ernest T.S. Walton constructed what is called a linear accelerator using a high-voltage source to accelerate protons through 700,000 volts (700 keV). From the first simple machines (Cockcroft-Walton and Van de Graaff machines) evolved the larger and more elaborate machines. The modern example of this type of accelerator is the linear accelerator, a sophisticated machine used in many scientific and medical applications.

The great breakthrough in accelerator technology came in 1930 with Ernest O. Lawrence's invention of the cyclotron. In the cyclotron, magnets guide the particle along a spiral path, allowing a single electric field to apply many cycles of acceleration. Soon unprecedented energies were achieved, and the steady improvement of Lawrence's simple machine has led to today's synchrotrons, whose endless circular flight paths allow particles to gain huge energies by passing millions of times through the electric fields that accelerate them.

A synchrotron accelerates particles using electric fields over and over in a circular path. Magnetic fields are used to bend the particles' trajectories and keep them moving in a circle. The accelerated particles lose energy rounding the curves, so energy must be continuously supplied. The beam is extracted heading toward targets and detectors. Until 30 years ago, all

accelerators were so-called fixed-target machines in which the speeding particle beam was made to hit a stationary target of some chosen substance.

In the early 1960s, physicists had gained enough experience in accelerator technology to be able to build colliders in which two carefully controlled beams are made to collide with each other at a chosen point. The beams for colliders may come from two synchrotrons or two linear accelerators.

Accelerators were originally designed to study (research) the structure of matter. Accelerators today are used not only for basic research purposes, but for many other applications such as:

- Production of radioisotopes, such as tritium.
- Generation of bremsstrahlung for radiography and radiation therapy.
- Induction of fusion.
- Pumping for lasers.
- Detoxification of hazardous waste.
- Actinide transmutation.
- Production of synchrotron radiation.
- Sterilization of food and surgical equipment.
- Medical radiation therapy.

Fundamental Characteristics of Radiological Hazards

Accelerators employ devices to impart energy to particles or redirect them during the acceleration process. These devices may emit ionizing radiation while they are operating. Klystrons provide power to accelerate charged particles. They emit x-rays during operation. Radiofrequency cavities accelerate charged particles using electromagnetic fields. Electrical discharges within the RF cavity cause photon (ionizing radiation) emission. Electrostatic separators/Septa split a particle beam into two beams using static electric fields. The high voltages associated with these devices cause electrons to accelerate in the vacuum within the beamline. They emit x-rays. Septa are also a high source of activation and residual radiation.

Exposure Control Techniques

Controls are used at accelerator facilities to protect personnel from exposure to ionizing radiation and other hazards including:

- Electrical.
- Mechanical.
- Cryogenic.
- Non-ionizing radiation.

<u>Engineered Controls.</u> Engineered controls include equipment and structures (passive or active) designed to protect personnel from hazards.

1. Passive engineered controls

   Once installed, passive engineered controls require no further action to perform their intended function. Passive engineered controls may include:

   - radiation shielding: such as concrete blocks, iron plates, lead bricks and earth berms.
   - barriers: such as fences, locked gates, and doors.
   - facility-specific: facility-specific passive engineered controls.

2. Active engineered controls

Active engineered controls include devices that sense changing conditions and can trigger a safety action.

- Safety interlock devices.
- Area radiation monitors.
- Access sensors, magnetic and mechanical.
- "Crash" or "scram" buttons.

Administrative Controls. Programs and activities which personnel must implement to provide protection from hazards.

- Search and secure (sweep) procedures.
- These are procedures used to verify that no personnel remain in a beamline enclosure when it is being prepared to receive beam.
- Controlled access procedures (including key controls).
- Procedures that allow personnel to access a beamline enclosure while it remains interlocked. There is no physical search of the area before the beam is restored.
- Radiological work permits (RWPs).
- RWPs provide written documentation of job descriptions, radiological conditions, and the required protective controls.
- Configuration control procedures.
- Procedures to ensure that important information about the configuration of a facility is accurate and that the configuration retains its functional purpose.

Waste Handling/Processing Operations

The following is taken from DOE O 435.1.

Basic Function and Work Activities

DOE radioactive waste management activities shall be systematically planned, documented, executed, and evaluated. Radioactive waste shall be managed to:

- Protect the public from exposure to radiation from radioactive materials. Requirements for public radiation protection are in DOE [Order] 5400.5, *Radiation Protection of the Public and the Environment*.
- Protect the environment. Requirements for environmental protection are in DOE [Order] 5400.1, *General Environmental Protection Program*, and DOE [Order] 5400.5, *Radiation Protection of the Public and the Environment*.
- Protect workers. Requirements for radiation protection of workers are in 10 CFR 835, "Occupational Radiation Protection;" requirements for industrial safety are in DOE O 440.1A, *Worker Protection Management for DOE Federal and Contractor Employees*.
- Comply with applicable Federal, state, and local laws and regulations.

These activities shall also comply with applicable Executive Orders and other DOE directives. All radioactive waste shall be managed in accordance with the requirements in DOE M 435.1-1, *Radioactive Waste Management Manual*. DOE, within its authority, may impose such requirements, in addition to those established in this Order, as it deems appropriate and necessary to protect the public, workers, and the environment, or to minimize threats to property.

Fundamental Characteristics of Radiological Hazards

Note: Radiological hazards related to waste handling and processing are element-specific, which are discussed above.

Exposure Control Techniques

Note: Exposure control techniques related to waste handling and processing are element-specific, and are discussed above.

Decontamination and Decommissioning

The following is taken from DOE-STD-1128-98.

Basic Function and Work Activities

No single DOE regulation covers all decontamination and decommissioning (D&D) requirements due to the wide variety of issues encompassed by D&D. These issues include project management, environmental surveillance, health and safety of workers and the public, engineering design, characterization survey techniques, D&D techniques, waste management, and waste transport. The primary DOE Orders pertaining to D&D activities are DOE Order 4700.1, Lifecycle Asset Management (DOE, 1998d); DOE Order 5400.5, Ch. 2, Radiation Protection of the Public and Environment (DOE, 1993c); DOE O 231.1, Environment Safety and Health Reporting (DOE, 1996e); DOE O 420.1A, Facility Safety (DOE, 2002a). The DOE operations offices may have implementation procedures corresponding to these Orders that which contractors will also need to comply.

DOE O 430.1B, Real Property Asset Management, provides the requirements to ensure a disciplined, systematic, and coordinated approach to project management. All projects, including D&D projects, should have clearly defined goals and objectives that support program requirements. Specific objectives include (1) promoting project execution that meets technical, schedule, and cost objectives, (2) meeting all applicable environmental, health and safety, and quality assurance requirements, and (3) avoiding a commitment of major resources before project definition. Good program management techniques should consider D&D costs as part of the lifecycle cost and select a tentative D&D method during the facility design phase.

The following is taken from DOE-HDBK-1110-2008.

Decontamination is any process or method of removing contamination. Frequently, there is a need for decontamination to reduce the radioactive source, and thus avoid more significant exposures to workers or, if the radioactive material escapes from the facility, to the public. Unfortunately, the process of decontamination itself may involve some dose to workers, e.g. radioactive material may enter the body through broken skin.

Title 10 CFR 835 has a design Objective that the design or modification of a facility and the selection of materials shall include features that facilitate operations, maintenance, decontamination, and decommissioning. Provision for decontamination must be made in the design of any component, system, or area where the potential for leakage of radioactive materials exists. Facilities may be released for public use if contamination levels meet established limits (not zero). Planning for the likelihood of decontamination should be done for any operation which may involve the spread or generation of significant amounts of

radioactive materials. Factors that affect the choice of decontamination method include type and quantity of radioactive leakage, item to be decontaminated, expense, practicality, etc.

Potential decontamination must be considered for fixed and removable equipment. Laundries and decontamination cells may be necessary for such items as respirators, clothing, or removable pumps. For equipment that must be decontaminated in place, provisions must be made for decontamination supplies (water, chemical, air) and electrical power.

Fundamental Characteristics of Radiological Hazards

Note: Radiological hazards related to decommissioning and decontamination are element-specific, and are discussed above.

Exposure Control Techniques

Note: Exposure control techniques related to decommissioning and decontamination are element-specific, and are discussed above.

Use of Radiation Generating Devices

The following is taken from DOE-STD-1098-2008.

Basic Function and Work Activities

Special considerations associated with the use of radiation generating devices include the presence of extremely high dose rates and the potential for uncontrolled exposures. Operation of these devices requires stringent physical and administrative controls to prevent overexposure to operating and support personnel a d those in adjacent work areas.

The following procedures should be considered when developing site-specific procedures for applicable types of radiation generating devices:

- ANSI N43.3, *American National Standard for General Radiation Safety-Installations Using Non-Medical X-Ray and Sealed Gamma-Ray Sources, Energies up to 10 MeV*, establishes acceptable guidelines for operations involving the irradiation of materials.
- ANSI N43.2, *Radiation Safety for X-Ray Diffraction and Fluorescence Analysis Equipment*, provides guidelines for operations involving the following devices :
  - Analytical diffraction and fluorescence
  - Flash x-ray
  - Sealed source irradiators used for diffraction studies.
- Line management, in conjunction with the radiological control organization, should establish the radiological control requirements for incidental x-ray devices such as electron microscopes and electron beam welders.
- Devices for medical use should be registered with the appropriate regulatory agency.
- Control requirements for radiographic devices include the following:
  - Title 10 CFR 34, “Licenses for Industrial Radiography and Radiation Safety Requirements for Industrial Radiographic Operations,” establishes acceptable guidelines for operations with devices containing sealed sources.
  - ANSI/HPS N43.3-2008: *For General Radiation Safety - Installations Using Non-Medical X-Ray and Sealed Gamma-Ray Sources, Energies up to 10 MeV*, establishes acceptable guidelines for on-site operations with devices other than sealed sources for radiographic use.

- On-site operations conducted by off-site contractors should be approved by line management in coordination with the site radiological control organization. This process should ensure the contractor has a valid Nuclear Regulatory Commission or Agreement State license and that the operational and emergency procedures are current and available.

Safety devices and interlocks at fixed installations that are required to ensure compliance with 10 CFR 835.501 shall be operational prior to and during generation of a radiation field. Operational status should be periodically verified by testing. Safety devices and interlocks should be fail-safe.

Fundamental Characteristics of Radiological Hazards

Note: Radiological hazards related to radiation generating devices are type-specific, and are discussed above.

Exposure Control Techniques

Note: Exposure control techniques related to radiation generating devices are type-specific, and are discussed above.

Environmental Restoration Activities

The following is taken from DOE-STD-1120-2008.

Basic Function and Work Activities.

Environmental restoration and decommissioning projects generally consist of multiple work tasks that must be evaluated throughout the life of the project as specific tasks are planned and scheduled. The work control process assures that each project task will be conducted in a safe manner in accordance with all pertinent requirements and controls. Work control activities such as task-level planning and analysis should be integrated with the unreviewed safety question (USQ) process to ensure that project tasks are conducted within the safety envelope analyzed by the DSA. The process for linking work control and the USQ process should be described in the DSA.

Task hazard analyses should be conducted throughout the life of the project as disposition tasks are planned and scheduled. The following guidelines should be used when conducting a task hazard analysis:

- The DSA should be used as the basis and an input for performing a task hazard analysis. The DSA analysis and control set provides an umbrella for all other work activities and provides controls at the project or facility level.
- The analysis should evaluate each step in the task's work instruction for hazards in the workplace and those introduced from chosen work methods. This process is accomplished most effectively by performing a walk-down of the work area, as needed, feasible, and permissible, based on existing facility hazards (e.g., high radiation areas), using the workers who will perform the task. The analysis should review task steps and evaluate hazardous substances and physical hazards. This typically provides the basis for selecting the appropriate immediate worker protection measures such as Personal Protective Equipment (PPE) or local monitoring. DOE O 440.1 and its implementation guide DOE G 440.1-1, Worker Protection Management

for DOE Federal and Contractor Employees Guide, provides further guidance on evaluation of worker hazards.

- The analysis should involve a multi-disciplinary team with the appropriate subject matter experts.
- Tasks should be screened against the DSA to ensure planned work is within the analyzed safety basis and to determine whether updates to documentation are necessary. This screening is accomplished consistent with the change control process

Fundamental Characteristics of Radiological Hazards

Note: Radiological hazards related to environmental restoration activities are site- and element-specific, and are discussed above.

Exposure Control Techniques

Note: Exposure control techniques related to environmental restoration activities are site- and type-specific, and are discussed above.

10. Radiation protection personnel shall demonstrate an expert level knowledge of the DOE radiation protection system for occupational workers as set forth in the following policy, requirements, and guidance documents:

- **10 CFR 835, Occupational Radiation Protection**
- **Implementation Guidance for use with 10 CFR 835, Occupational Radiation Protection**
- **DOE P 441.1, Department of Energy Radiological Health and Safety Policy**

a. Discuss the relationship of the above documents in defining the DOE system of radiation protection.

The following is taken from 10 CFR 835.1.

The rules in this part establish radiation protection standards, limits, and program requirements for protecting individuals from ionizing radiation resulting from the conduct of DOE activities.

The following is taken from DOE G 441.1-1C.

In March 2007 the U.S. Department of Energy (DOE) published an DOE G 441.1-1C, *Radiation Protection Programs Guide for Use with Title 10, Code of Federal Regulations, Part 835, Occupational Radiation Protection,* which discussed acceptable methods for ensuring that the functional elements of radiological activities will be managed and administered in accordance with 10 CFR 835, "Occupational Radiation Protection." The Guide was part of DOE's efforts to eliminate redundant requirements and guidance and compiled the guidance previously provided in a set of 13 Implementation Guides.

On June 8, 2007, the DOE published an amendment to 10 CFR 835. This Guide reflects the June 8, 2007, amendment to 10 CFR 835 and continues to provide cross-references to other guides, DOE-STD-1098-2008, *Radiological Control*, hereinafter referred to as the RCS, DOE directives, and industry consensus standards that provide detailed guidance for implementing specific requirements in 10 CFR 835.

DOE is in the process of updating all of its guidance documents for occupational radiation protection to reflect the 2007 amendment to 10 CFR 835. This Guide is one of the first documents to be updated, and as such, the cross-references to other DOE guidance documents will change as additional updated guidance documents are finalized. The references to other DOE guidance documents, which are scheduled to be updated, will include the notation "Use the revised version, reflecting the 2007 amendment to 10 CFR 835, when available." This Guide provides guidance with respect to implementing the provisions of all the functional areas contained in 10 CFR 835.

[Note: DOE P 441.1 is discussed in "b" below.]

b. Give examples of how DOE P 441.1, Department of Energy Radiological Health and Safety Policy, is reflected in requirements and guidance.

It is the policy of the Department of Energy to conduct its radiological operations in a manner that ensures the health and safety of all its employees, contractors, and the general public. In achieving this objective, the Department shall ensure that radiation exposures to its workers and the public and releases of radioactivity to the environment are maintained below regulatory limits and deliberate efforts are taken to further reduce exposures and releases as low as reasonably achievable. The Department is fully committed to implementing a radiological control program of the highest quality that consistently reflects this policy.

In meeting this policy, the Department shall:

- Establish and maintain a system of regulatory policy and guidance reflective of national and international radiation protection standards and recommendations;
- Ensure personnel responsible for performing radiological work activities are appropriately trained;
- Ensure the technical competence of personnel responsible for implementing and overseeing the radiological control program;
- Establish and maintain, at all levels, line management involvement and accountability for departmental radiological performance;
- Ensure radiological measurements, analyses, worker monitoring results and estimates of public exposures are accurate and appropriately made;
- Conduct radiological operations in a manner that controls the spread of radioactive materials and reduces exposure to the workforce and the general public and that utilizes a process that seeks exposure levels as low as reasonably achievable;
- Incorporate dose reduction, contamination reduction, and waste minimization features into the design of new facilities and significant modifications to existing facilities in the earliest planning stages; and
- Conduct oversight to ensure departmental requirements are being complied with and appropriate radiological work practices are being implemented.

c. Explain how the 10 CFR 835 Implementation Guides are used to develop and implement local programs to comply with the radiation protection requirements at the site/facility level.

Note: This activity is site- and facility-specific. The local Qualifying Official will evaluate the explanation.

d. Discuss methods of meeting the key requirements in Subpart A (General Provisions). Include:
- **Scope and exclusions**
- **Definitions**
- **Radiological Units**

The following is taken from 10 CFR 835.1 through .4.

Scope and Exclusions

General. The rules in this part establish radiation protection standards, limits, and program requirements for protecting individuals from ionizing radiation resulting from the conduct of DOE activities.

Exclusion. Except as provided in paragraph (c) of this section, the requirements in this part do not apply to:
- Activities that are regulated through a license by the Nuclear Regulatory
- Commission or a State under an Agreement with the Nuclear Regulatory Commission, including activities certified by the Nuclear Regulatory
- Commission under section 1701 of the Atomic Energy Act;
- Activities conducted under the authority of the Deputy Administrator for Naval Reactors, as described in Pub. L. 98–525 and 106–65;
- Activities conducted under the Nuclear Explosives and Weapons Surety Program relating to the prevention of accidental or unauthorized nuclear detonations;
- DOE activities conducted outside the United States on territory under the jurisdiction of a foreign government to the extent governed by occupational radiation protection requirements agreed to between the United States and the cognizant government;
- Background radiation, radiation doses received as a patient for the purposes of medical diagnosis or therapy, or radiation doses received from participation as a subject in medical research programs; or
- Radioactive material on or within material, equipment, and real property which is approved for release when the radiological conditions of the material, equipment, and real property have been documented to comply with the criteria for release set forth in a DOE authorized limit which has been approved by a Secretarial Officer in consultation with the Chief Health, Safety and Security Officer.
- Radioactive material transportation not performed by DOE or a DOE contractor.
- Occupational doses received as a result of excluded activities and radioactive material transportation listed in paragraphs (b)(1) through (b)(4) and (b)(7) of this section, shall be included to the extent practicable when determining compliance with the occupational dose limits at 10 CFR 835.202 and 835.207, and with the limits for the embryo/ fetus at 10 CFR 835.206. Occupational doses resulting from authorized emergency exposures and planned special exposures shall not be considered when determining compliance with the dose limits at 10 CFR 835.202 and 835.207.
- The requirements in subparts F and G of this part do not apply to radioactive material transportation by DOE or a DOE contractor conducted:
  - Under the continuous observation and control of an individual who is knowledgeable of and implements required exposure control measures, or
  - In accordance with Department of Transportation regulations or DOE orders that govern such movements.

Definitions

As used in this part:

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 this part.

Activity median aerodynamic diameter (AMAD) means a particle size in an aerosol where fifty percent of the activity in the aerosol is associated with particles of aerodynamic diameter greater than the AMAD.

Airborne radioactive material or airborne radioactivity means radioactive material dispersed in the air in the form of dusts, fumes, particulates, mists, vapors, or gases.

Airborne radioactivity area means any area, accessible to individuals, where:

- The concentration of airborne radioactivity, above natural background, exceeds or is likely to exceed the derived air concentration (DAC) values listed in appendix A or appendix C of this part; or
- An individual present in the area without respiratory protection could receive an intake exceeding 12 DAC hours in a week.

ALARA means "As Low As is Reasonably Achievable," which is the approach to radiation protection to manage and control exposures (both individual and collective) to the work force and to the general public to as low as is reasonable, taking into account social, technical, economic, practical, and public policy considerations. As used in this part, ALARA is not a dose limit but a process which has the objective of attaining doses as far below the applicable limits of this part as is reasonably achievable.

Annual limit on intake (ALI) means the derived 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 of 5 rems (0.05 sieverts (Sv)) (1 rem = 0.01 Sv) or a committed equivalent dose of 50 rems (0.5 Sv) to any individual organ or tissue. ALI values for intake by ingestion and inhalation of selected radionuclides are based on International Commission on Radiological Protection Publication 68, Dose Coefficients for Intakes of Radionuclides by Workers, published July, 1994 (ISBN 0 08 042651 4).

Authorized limit means a limit on the concentration of residual radioactive material on the surfaces or within the property that has been derived consistent with DOE directives including the as low as is reasonably achievable (ALARA) process requirements, given the anticipated use of the property and has been authorized by DOE to permit the release of the property from DOE radiological control.

Background means 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 o producing nominal amounts of radiation.

Bioassay means 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.

Calibration means to adjust and/or determine either:

- The response or reading of an instrument relative to a standard (e.g., primary, secondary, or tertiary) or to a series of conventionally true values; or
- The strength of a radiation source relative to a standard (e.g., primary, secondary, or tertiary) or conventionally true value.

Contamination area means any area, accessible to individuals, where removable surface contamination levels exceed or are likely to exceed the removable surface contamination values specified in appendix D of this part, but do not exceed 100 times those values.

Controlled area means any area to which access is managed by or for DOE to protect individuals from exposure to radiation and/or radioactive material.

Declared pregnant worker means 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 as provided in 10 CFR 835.206. This declaration may be revoked, in writing, at any time by the declared pregnant worker.

Derived air concentration (DAC) means, for the radionuclides listed in appendix A of this part, the airborne concentration that equals the ALI divided by the volume of air breathed by an average worker for a working year of 2000 hours (assuming a breathing volume of 2400 m^3). For the radionuclides listed in appendix C of this part, the air immersion DACs were calculated for a continuous, non-shielded exposure via immersion in a semi-infinite cloud of radioactive material. Except as noted in the footnotes to appendix A of this part, the values are based on dose coefficients from International Commission on Radiological Protection Publication 68, Dose Coefficients for Intakes of Radionuclides by Workers, published July, 1994 (ISBN 0 08 042651 4) and the associated ICRP computer program, The ICRP Database of Dose Coefficients: Workers and Members of the Public, ISBN 0 08 043 8768).

Derived air concentration-hour (DAC hour) means the product of the concentration of radioactive material in air (expressed as a fraction or multiple of the DAC for each radionuclide) and the time of exposure to that radionuclide, in hours.

Deterministic effects means effects due to radiation exposure for which the severity varies with the dose and for which a threshold normally exists (e.g., radiation-induced opacities within the lens of the eye).

DOE means the United States Department of Energy.

DOE activity means an activity taken for or by DOE in a DOE operation or facility that has the potential to result in the occupational exposure of an individual to radiation or radioactive material. The activity may be, but is not limited to, design, construction, operation, or decommissioning. To the extent appropriate, the activity may involve a single DOE facility or operation or a combination of facilities and operations, possibly including an entire site or multiple DOE sites.

Entrance or access point means any location through which an individual could gain access to areas controlled for the purpose of radiation protection. This includes entry or exit portals of sufficient size to permit human entry, irrespective of their intended use.

General employee means an individual who is either a DOE or DOE contractor employee; an employee of a subcontractor to a DOE contractor; or an individual who performs work for or in conjunction with DOE or utilizes DOE facilities.

High contamination area means any area, accessible to individuals, where removable surface contamination levels exceed or are likely to exceed 100 times the removable surface contamination values specified in appendix D of this part.

High radiation area means any area, accessible to individuals, in which radiation levels could result in an individual receiving an equivalent dose to the whole body in excess of 0.1 rems (0.001 Sv) in 1 hour at 30 centimeters from the radiation source or from any surface that the radiation penetrates.

Individual means any human being.

Member of the public means an individual who is not a general employee. An individual is not a “member of the public” during any period in which the individual receives an occupational dose.

Minor means an individual less than 18 years of age.

Monitoring means the measurement of radiation levels, airborne radioactivity concentrations, radioactive contamination levels, quantities of radioactive material, or individual doses and the use of the results of these measurements to evaluate radiological hazards or potential and actual doses resulting from exposures to ionizing radiation.

Occupational dose means an individual’s ionizing radiation dose (external and internal) as a result of that individual’s work assignment. Occupational dose does not include doses received as a medical patient or doses resulting from background radiation or participation as a subject in medical research programs.

Person means any individual, corporation, partnership, firm, association, trust, estate, public or private institution, group, Government agency, any State or political subdivision of, or any political entity within a State, any foreign government or nation or other entity, and any legal successor, representative, agent or agency of the foregoing; provided that person does not include DOE or the United States Nuclear Regulatory Commission.

Radiation means ionizing radiation: alpha particles, beta particles, gamma rays, x-rays, neutrons, high-speed electrons, high-speed protons, and other particles capable of producing ions. Radiation, as used in this part, does not include non-ionizing radiation, such as radio waves or microwaves, or visible, infrared, or ultraviolet light.

Radiation area means any area, accessible to individuals, in which radiation levels could result in an individual receiving an equivalent dose to the whole body in excess of 0.005 rem (0.05 mSv) in 1 hour at 30 centimeters from the source or from any surface that the radiation penetrates.

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 of this part.

Radioactive material transportation means the movement of radioactive material by aircraft, rail, vessel, or highway vehicle. Radioactive material transportation does not include preparation of material or packagings for transportation, storage of material awaiting transportation, or application of markings and labels required for transportation.

Radiological area means any area within a controlled area defined in this section as a "radiation area," "high radiation area," "very high radiation area," "contamination area," "high contamination area," or "airborne radioactivity area."

Radiological worker means 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 Sv) per year total effective dose.

Real property means land and anything permanently affixed to the land such as buildings, fences and those things attached to the buildings, such as light fixtures, plumbing and heating fixtures.

Real-time air monitoring means measurement of the concentrations or quantities of airborne radioactive materials on a continuous basis.

Respiratory protective device means an apparatus, such as a respirator, worn by an individual for the purpose of reducing the individual's intake of airborne radioactive materials.

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 non-radioactive material, or firmly fixed to a non-radioactive surface by electroplating or other means intended to prevent leakage or escape of the radioactive material. Sealed radioactive sources do not include reactor fuel elements, nuclear explosive devices, and radioisotope thermoelectric generators.

Source leak test means a test to determine if a sealed radioactive source is leaking radioactive material.

Special tritium compound (STC) means any compound, except for H_2O, that contains tritium, either intentionally (e.g., by synthesis) or inadvertently (e.g., by contamination mechanisms).

Stochastic effects mean malignant and hereditary diseases for which the probability of an effect occurring, rather than its severity, is regarded as a function of dose without a threshold, for radiation protection purposes.

Very high radiation area means any area, accessible to individuals, in which radiation levels could result in an individual receiving an absorbed dose in excess of 500 rads (5 grays) in one hour at 1 meter from a radiation source or from any surface that the radiation penetrates.

Week means a period of seven consecutive days.

Year means the period of time beginning on or near January 1 and ending on or near December 31 of that same year used to determine compliance with the provisions of this part. The starting and ending date of the year used to determine compliance may be changed, provided that the change is made at the beginning of the year and that no day is omitted or duplicated in consecutive years.

As used in this part to describe various aspects of radiation dose:

Absorbed dose (D) means the average energy imparted by ionizing radiation to the matter in a volume element. The absorbed dose is expressed in units of rad (or gray) (1 rad = 0.01 grays).

Committed effective dose (E_{50}) means the sum of the committed equivalent doses to various tissues or organs in the body ($H_{T,50}$), each multiplied by the appropriate tissue weighting factor (w_T)—that is, $E_{50} = \Sigma w_T H_{T,50} + w_{Remainder} H_{Remainder,50}$. Where $w_{Remainder}$ is the tissue weighting factor assigned to the remainder organs and tissues and $H_{Remainder,50}$ is the committed equivalent dose to the remainder organs and tissues. Committed effective dose is expressed in units of rem (or Sv).Committed equivalent dose ($H_{T,50}$) means the equivalent dose 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 equivalent dose is expressed in units of rem (or Sv).Cumulative total effective dose means the sum of all total effective dose values recorded for an individual plus, for occupational exposures received before the implementation date of this amendment, the cumulative total effective dose equivalent (as defined in the November 4, 1998 amendment to this rule) values recorded for an individual, where available, for each year occupational dose was received, beginning January 1, 1989.

Dose is a general term for absorbed dose, equivalent dose, effective dose, committed equivalent dose, committed effective dose, or total effective dose as defined in this part.

Effective dose (E) means the summation of the products of the equivalent dose received by specified tissues or organs of the body (H_T) and the appropriate tissue weighting factor (w_T)— that is, $E = \Sigma w_T H_T$. It includes the dose from radiation sources internal and/or external to the body. For purposes of compliance with this part, equivalent dose to the whole body may be used as effective dose for external exposures. The effective dose is expressed in

units of rem (or Sv).

Equivalent dose (H_T) means the product of average absorbed dose ($D_{T,R}$) in rad (or gray) in a tissue or organ (T) and a radiation (R) weighting factor (w_R). For external dose, the equivalent dose to the whole body is assessed at a depth of 1 cm in tissue; the equivalent dose to the lens of the eye is assessed at a depth of 0.3 cm in tissue, and the equivalent dose to the extremity and skin is assessed at a depth of 0.007 cm in tissue. Equivalent dose is expressed in units of rem (or Sv).

External dose or exposure means that portion of the equivalent dose received from radiation sources outside the body (i.e., "external sources").

Extremity means hands and arms below the elbow or feet and legs below the knee.

Internal dose or exposure means that portion of the equivalent dose received from radioactive material taken into the body (i.e., "internal sources").

Radiation weighting factor (w_R) means the modifying factor used to calculate the equivalent dose from the average tissue or organ absorbed dose; the absorbed dose (expressed in rad or gray) is multiplied by the appropriate radiation weighting factor. The radiation weighting factors to be used for determining equivalent dose in rem are as follows:

Table 6. Radiation weighting factors [1,] w_R-continued

Type and energy range	Radiation Weighting Factor
Neutrons, energy 10 keV to 100 keV [2,3]…….	10
Neutrons, energy > 100 keV toMeV[2,3]……...	20
Neutrons, energy > 2 MeV to 20 MeV [2,3]…….…….	10
Neutrons, energy > 20 MeV [2,3]……...	5
Protons, other than recoil protons, energy > 2MeV……	5
Alpha particles, fission fragments, heavy nuclei…….	20

[1] All values relate to the radiation on the body, or for internal sources, emitted from the source.
[2] When spectral data are insufficient to identify the energy of the neutrons, a radiation weighting factor of 20 shall be used.
[3] When spectral data are sufficient to identify the energy of the neutrons, the following equation may be used to determine a neutron radiation weighting factor value:

$$w_R = 5 + 17 \exp\left[\frac{-(\ln(2E_n))^2}{6}\right] \text{ Where } E_n \text{ is the neutron energy in MeV.}$$

Source: 10 CFR 835.2

Tissue weighting factor (w_T) means the fraction of the overall health risk, resulting from uniform, whole body irradiation, attributable to specific tissue (T). The equivalent dose to tissue, (H_T), is multiplied by the appropriate tissue weighting factor to obtain the effective dose (E) contribution from that tissue. The tissue weighting factors are as follows:

Table 7. Tissue weighting factor for various organs and tissues

Organs or tissues, T	Tissue weighting factor w_T
Gonads…….	0.20
Red bone marrow……	0.12
Colon..…….	0.12
Lungs.……..……	0.12
Stomach……...	0.12
Bladder……	0.05
Breast……..	0.05
Liver……	0.05
Esophagus……	0.05
Thyroid……	0.05
Skin……..	0.01
Bone surfaces……..	0.01
Remainder[1]…….…….	0.05
Whole body[2]……..…….……	1.00

[1] "Remainder" means the following additional tissues and organs and their masses, in grams, following parenthetically: adrenals (14), brain (1400), extrathoracic airways (15), small intestine (640), kidneys (310), muscle (28,000), pancreas (100), spleen (180), thymus (20), and uterus (80). The equivalent of the remainder tissues ($H_{Remainder}$) is normally calculated as the mass-weighted mean dose to the preceding ten organs and tissues. In those cases in which the most highly irradiated remainder tissue or organ receives the highest equivalent dose of all the organs, a weighting factor of 0.025 (half of remainder) is applied to that tissue or organ and 0.025 (half the remainder) to the mass-weighted equivalent dose in the rest of the remainder tissues and organs to give the remainder equivalent dose.

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

Source: 10 CFR 835.2.

Total effective dose (TED) means the sum of the effective dose (for external exposures) and the committed effective dose.

Whole body means, for the purposes of external exposure, head, trunk (including male gonads), arms above and including the elbow, or legs above and including the knee.

(c) Terms defined in the Atomic Energy Act of 1954 or in 10 CFR 820 and not defined in this part are used consistent with their meanings given in the Atomic Energy Act of 1954 or in 10 CFR 820.

Radiological Units

Unless otherwise specified, the quantities used in the records required by this part shall be clearly indicated in special units of curie, rad, roentgen, or rem, including multiples and subdivisions of these units, or other conventional units, such as, dpm, dpm/100 cm^2 or mass units. The SI units, becquerel, gray, and sievert may be provided parenthetically for reference with scientific standards.

e. Discuss methods of meeting the key requirements in Subpart B (Management and Administrative Requirements) based upon the guidance in the Radiation Protection Program Implementation Guide and the Occupational ALARA Program Implementation Guide, including:

- **Radiation Protection Program**
- **Internal audits**
- **Education, Training and Skills**
- **Written Procedures**

The following is taken from 10 CFR 835.101-104.

Radiation Protection Programs

The following are the management and administrative requirements for DOE's radiation protection programs.

- A DOE activity shall be conducted in compliance with a documented radiation protection program (RPP) as approved by the DOE.
- The DOE may direct or make modifications to a RPP.
- The content of each RPP shall be commensurate with the nature of the activities performed and shall include formal plans and measures for applying the as low as reasonably achievable (ALARA) process to occupational exposure.
- The RPP shall specify the existing and/or anticipated operational tasks that are intended to be within the scope of the RPP. Except as provided in 10 CFR 835.101(h), any task outside the scope of a RPP shall not be initiated until an update of the RPP is approved by DOE.
- The content of the RPP shall address, but shall not necessarily be limited to, each requirement in this part.
- The RPP shall include plans, schedules, and other measures for achieving compliance with regulations of this part. Unless otherwise specified in this part, compliance with the amendments to this part published on June 8, 2007 shall be achieved no later than July 9, 2010.
- An update of the RPP shall be submitted to DOE:
- Whenever a change or an addition to the RPP is made;
- Prior to the initiation of a task not within the scope of the RPP; or
- Within 180 days of the effective date of any modifications to this part.
- Changes, additions, or updates to the RPP may become effective without prior Department approval only if the changes do not decrease the effectiveness of the RPP and the RPP, as changed, continues to meet the requirements of this part. Proposed changes that decrease the effectiveness of the RPP shall not be implemented without submittal to and approval by the Department.
- An initial RPP or an update shall be considered approved 180 days after its submission unless rejected by DOE at an earlier date.

Internal Audits

Internal audits of the radiation protection program, including examination of program content and implementation, shall be conducted through a process that ensures that all functional elements are reviewed no less frequently than every 36 months.

Education, Training and Skills

Individuals responsible for developing and implementing measures necessary for ensuring compliance with the requirements of this part shall have the appropriate education, training, and skills to discharge these responsibilities.

Written Procedures

Written procedures shall be developed and implemented as necessary to ensure compliance with this part, commensurate with the radiological hazards created by the activity and consistent with the education, training, and skills of the individuals exposed to those hazards.

f. Discuss methods of meeting the key requirements in 10 CFR 835, Subpart C (Standards for Internal and External Exposure) based upon the guidance in the Internal Dosimetry Program Implementation Guide, the External Dosimetry Program Implementation Guide, the Radiation-Generating Devices Implementation Guide, and the Evaluation and Control of Fetal Exposure Implementation Guide, including:

- **Occupational Limits for general employees**
- **Combining internal and external dose equivalents resulting from DOE activities**
- **Determination of compliance for non-uniform exposure of the skin**
- **Limits for the embryo/fetus**
- **Limits for members of the public and minors entering a controlled area**
- **Concentrations of radioactive materials in air**

Occupational Limits for General Employees

The following is taken from 10 CFR 835.202-209.

Except for planned special exposures conducted consistent with 10 CFR 835.204 and emergency exposures authorized in accordance with 10 CFR 835.1302, the occupational dose received by general employees shall be controlled such that the following limits are not exceeded in a year:

- A total effective dose of 5 rems (0.05 Sv);
- The sum of the equivalent dose to the whole body for external exposures and the committed equivalent dose to any organ or tissue other than the skin or the lens of the eye of 50 rems (0.5 Sv);
- An equivalent dose to the lens of the eye of 15 rems (0.15 Sv); and
- The sum of the equivalent dose to the skin or to any extremity for external exposures and the committed equivalent dose to the skin or to any extremity of 50 rems (0.5 Sv).

All occupational doses received during the current year, except doses resulting from planned special exposures conducted in compliance with 10 CFR 835.204 and emergency exposures authorized in accordance with 10 CFR 835.1302, shall be included when demonstrating compliance with 10 CFR 835.202(a) and 835.207.

Doses from background, therapeutic and diagnostic medical radiation, and participation as a subject in medical research programs shall not be included in dose records or in the assessment of compliance with the occupational dose limits.

Combining Internal and External Dose Equivalents Resulting from DOE Activities

The total effective dose during a year shall be determined by summing the effective dose from external exposures and the committed effective dose from intakes during the year.

Determinations of the effective dose shall be made using the radiation and tissue weighting factor values provided in 10 CFR 835.2.

Determination of Compliance for Non-uniform Exposure of the Skin

Non-uniform exposures of the skin from x-rays, beta radiation, and/or radioactive material on the skin are to be assessed as specified in this section.

For purposes of demonstrating compliance with 10 CFR 835.202(a)(4), assessments shall be conducted as follows:

- Area of skin irradiated is 100 cm^2 or more. The non-uniform equivalent dose received during the year shall be averaged over the 100 cm^2 of the skin receiving the maximum dose, added to any uniform equivalent dose also received by the skin, and recorded as the equivalent dose to any extremity or skin for the year.
- Area of skin irradiated is 10 cm^2 or more, but is less than 100 cm^2. The non-uniform equivalent dose (H) to the irradiated area received during the year shall be added to any uniform equivalent dose also received by the skin and recorded as the equivalent dose to any extremity or skin for the year. H is the equivalent dose averaged over the 1 cm^2 of skin receiving the maximum absorbed dose, D, reduced by the fraction f, which is the irradiated area in cm^2 divided by 100 cm^2 (*i.e.*, H = fD). In no case shall a value of f less than 0.1 be used.
- Area of skin irradiated is less than 10 cm^2. The non-uniform equivalent dose shall be averaged over the 1 cm^2 of skin receiving the maximum dose. This equivalent dose shall:
  - Be recorded in the individual's occupational exposure history as a special entry; and
  - Not be added to any other equivalent dose to any extremity or skin for the year.

Limits for the Embryo/Fetus

- The equivalent dose limit for the embryo/fetus from the period of conception to birth, as a result of occupational exposure of a declared pregnant worker, is 0.5 rem (0.005 Sv).
- Substantial variation above a uniform exposure rate that would satisfy the limits provided in 10 CFR 835.206(a) shall be avoided.
- If the equivalent dose to the embryo/ fetus is determined to have already exceeded 0.5 rem (0.005 Sv) by the time a worker declares her pregnancy, the declared pregnant worker shall not be assigned to tasks where additional occupational exposure is likely during the remaining gestation period.

Limits for Members of the Public and Minors Entering a Controlled Area

The total effective dose limit for members of the public exposed to radiation and/or radioactive material during access to a controlled area is 0.1 rem (0.001 Sv) in a year. The dose limits for minors occupationally exposed to radiation and/or radioactive materials at a DOE activity are 0.1 rem (0.001 Sv) total effective dose in a year and 10 percent of the occupational dose limits specified at 10 CFR 835.202(a)(3) and (a)(4).

Concentrations of Radioactive Materials in Air

The derived air concentration (DAC) values given in appendices A and C of this part shall be used in the control of occupational exposures to airborne radioactive material.

The estimation of internal dose shall 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.

g. Discuss methods of meeting the key requirements in 10 CFR 835, Subpart E (Monitoring of Individuals and Areas) based upon guidance in the External Dosimetry Program Implementation Guide, the Internal Dosimetry Program Implementation Guide, the Evaluation and Control of Fetal Exposure Implementation Guide, the Instrument Calibration for Portable Survey Instruments Implementation Guide and the Workplace Air Monitoring Implementation Guide, including:

- **General monitoring requirements**
- **Individual monitoring**
- **Area monitoring**
- **Radioactive contamination control and monitoring**
- **Receipt of Packages containing radioactive material**

General Monitoring Requirements

The following is taken from 10 CFR 835.401.

Monitoring of individuals and areas shall be performed to:

- Demonstrate compliance with the regulations in this part;
- Document radiological conditions;
- Detect changes in radiological conditions;
- Detect the gradual buildup of radioactive material;
- Verify the effectiveness of engineered and administrative controls in containing radioactive material and reducing radiation exposure; and
- Identify and control potential sources of individual exposure to radiation and/or radioactive material.

Instruments and equipment used for monitoring shall be:

- Periodically maintained and calibrated on an established frequency;
- Appropriate for the type(s), levels, and energies of the radiation(s) encountered;
- Appropriate for existing environmental conditions; and
- Routinely tested for operability.

Individual Monitoring

The following is taken from 10 CFR 835.402.

For the purpose of monitoring individual exposures to external radiation, personnel dosimeters shall be provided to and used by:

- Radiological workers who, under typical conditions, are likely to receive one or more of the following:
  - An effective dose of 0.1 rem (0.001 Sv) or more in a year;
  - An equivalent dose to the skin or to any extremity of 5 rems (0.05 Sv) or more in a year;
  - An equivalent dose to the lens of the eye of 1.5 rems (0.015 Sv) or more in a year;
- Declared pregnant workers who are likely to receive from external sources an equivalent dose to the embryo/ fetus in excess of 10 percent of the applicable limit at 10 CFR 835.206(a);
- Occupationally exposed minors likely to receive a dose in excess of 50 percent of the applicable limits at 10 CFR 835.207 in a year from external sources;
- Members of the public entering a controlled area likely to receive a dose in excess of 50 percent of the limit at 10 CFR 835.208 in a year from external sources; and
- Individuals entering a high or very high radiation area.

External dose monitoring programs implemented to demonstrate compliance with 10 CFR 835.402(a) shall be adequate to demonstrate compliance with the dose limits established in subpart C of this part and shall be:

- Accredited, or excepted from accreditation, in accordance with the DOE Laboratory Accreditation Program for Personnel Dosimetry; or
- Determined by the Secretarial Officer responsible for environment, safety and health matters to have performance substantially equivalent to that of programs accredited under the DOE Laboratory Accreditation Program for Personnel Dosimetry.

For the purpose of monitoring individual exposures to internal radiation, internal dosimetry programs (including routine bioassay programs) shall be conducted for:

- Radiological workers who, under typical conditions, are likely to receive a committed effective dose of 0.1 rem (0.001 Sv) or more from all occupational radionuclide intakes in a year;
- Declared pregnant workers likely to receive an intake or intakes resulting in an equivalent dose to the embryo/ fetus in excess of 10 percent of the limit stated at 10 CFR 835.206(a);
- Occupationally exposed minors who are likely to receive a dose in excess of 50 percent of the applicable limit stated at 10 CFR 835.207 from all radionuclide intakes in a year; or
- Members of the public entering a controlled area likely to receive a dose in excess of 50 percent of the limit stated at 10 CFR 835.208 from all radionuclide intakes in a year.

Internal dose monitoring programs implemented to demonstrate compliance with 10 CFR 835.402(c) shall be adequate to demonstrate compliance with the dose limits established in subpart C of this part and shall be:

- Accredited, or excepted from accreditation, in accordance with the DOE Laboratory Accreditation Program for Radiobioassay; or,
- Determined by the Secretarial Officer responsible for environment, safety and health matters to have performance substantially equivalent to that of programs accredited under the DOE Laboratory Accreditation Program for Radiobioassay.

Area Monitoring

[Note: 10 CFR 835.403 is entitled "Air Monitoring." The following discussion is taken from that citation.]

Monitoring of airborne radioactivity shall be performed:

- Where an individual is likely to receive an exposure of 40 or more DAC hours in a year; or
- 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.

Radioactive Contamination Control and Monitoring

The following is taken from DOE G 441.1-1C.

Work with unsealed quantities of radioactive material creates the potential for generating radioactive contamination. 10 CFR 835 requires, in part, a contamination control program sufficient to provide warning of the presence of surface contamination and to prevent the inadvertent transfer of contamination at levels exceeding specified values outside of radiological areas under normal operating conditions.

An acceptable contamination control program incorporates two types of control: (1) Engineered control, and (2) administrative control. Contamination monitoring is part of and verifies the effectiveness of the contamination control program. In implementing a contamination control program, engineered controls that control contamination at the source are the most important element. Engineered controls incorporated into older facilities may not be sufficient to meet modern contamination control standards. The engineered controls used in a contamination control program may include temporary containment and ventilation, which may be the primary methods of controlling airborne radioactivity and internal exposures to workers in older facilities, during relatively short-term operations and maintenance, and in other situations in which permanent engineered controls are unavailable or inadequate. For example, a permanently installed high efficiency particulate air (HEPA)-filtered ventilation system may be included as a physical design feature in a facility to control airborne radioactive material concentrations during routine operations, but a temporary HEPA-filtered ventilation system may be used as an engineering control during certain maintenance activities. Similarly, a drain system may be included as a physical design feature to route contaminated fluids to a controlled collection point, but temporary drains may be installed as engineered controls during system breach. Finally, administrative controls, including access restrictions and the use of specific work practices designed to minimize contamination transfer, should be used as the tertiary method to control exposure to contamination hazards. These elements of a contamination control program are not

independent. The permanent engineered controls included in a facility will dictate the types and levels of administrative controls and engineered controls that are possible and necessary.

A contamination control program is an essential element of a comprehensive radiological control program. In this Guide, when another element of a radiological control program interfaces with the contamination control program, the appropriate chapter is referenced and the topic of interest is listed as it applies to contamination control. Because of these interfaces, individuals involved with the contamination control program should interact with personnel working in other elements of the radiological control program, particularly with individuals involved in instrument calibration, posting and labeling, air monitoring, internal and external dosimetry, As Low As Is Reasonably Achievable (ALARA), training, and record-keeping programs.

Receipt of Packages Containing Radioactive Material

The following is taken from 10 CFR 835.405.

If packages containing quantities of radioactive material in excess of a Type A quantity (as defined at 10 CFR 71.4) are expected to be received from radioactive material transportation, arrangements shall be made to either:

- Take possession of the package when the carrier offers it for delivery; or
- Receive notification as soon as practicable after arrival of the package at the carrier's terminal and to take possession of the package expeditiously after receiving such notification.

Upon receipt from radioactive material transportation, external surfaces of packages known to contain radioactive material shall be monitored if the package:

- Is labeled with a Radioactive White I, Yellow II, or Yellow III label (as specified at 49 CFR 172.403 and 172.436–440); or
- Has been transported as low specific activity material (as defined at 10 CFR 71.4) on an exclusive use vehicle (as defined at 10 CFR 71.4); or
- Has evidence of degradation, such as packages that are crushed, wet, or damaged.

The monitoring required by paragraph (b) of this section shall include:

- Measurements of removable contamination levels, unless the package contains only special form (as defined at 10 CFR 71.4) or gaseous radioactive material; and
- Measurements of the radiation levels, if the package contains a Type B quantity (as defined at 10 CFR 71.4) of radioactive material.

The monitoring required by paragraph (b) of this section shall be completed as soon as practicable following receipt of the package, but not later than 8 hours after the beginning of the working day following receipt of the package.

Monitoring pursuant to 10 CFR 835.405(b) is not required for packages transported on a DOE site which have remained under the continuous observation and control of a DOE employee or DOE contractor employee who is knowledgeable of and implements required exposure control measures.

h. Discuss methods of meeting the key requirements in 10 CFR 835, Subpart F (Entry Control Program), including:
- **Radiological Areas**
- **Radiation Areas**
- **High Radiation Areas**
- **Very High Radiation Areas**

The following is taken from 10 CFR 835.501-502.

Radiological Areas

- Personnel entry control shall be maintained for each radiological area.
- The degree of control shall be commensurate with existing and potential radiological hazards within the area.
- One or more of the following methods shall be used to ensure control:
- Signs and barricades;
- Control devices on entrances;
- Conspicuous visual and/or audible alarms;
- Locked entrance ways; or
- Administrative controls.
- Written authorizations shall be required to control entry into and perform work within radiological areas. These authorizations shall specify radiation protection measures commensurate with the existing and potential hazards.
- No control(s) shall be installed at any radiological area exit that would prevent rapid evacuation of personnel under emergency conditions.

Radiation Areas

[Note: 10 CFR 835, Subpart F does not contain discussion on radiation areas per se. Rather, it contains discussion on the next two items, which follows.]

High Radiation Areas

The following measures shall be implemented for each entry into a high radiation area:
- The area shall be monitored as necessary during access to determine the exposure rates to which the individuals are exposed; and
- Each individual shall be monitored by a supplemental dosimetry device or other means capable of providing an immediate estimate of the individual's integrated equivalent dose to the whole body during the entry.

Physical controls

One or more of the following features shall be used for each entrance or access point to a high radiation area where radiation levels exist such that an individual could exceed an equivalent dose to the whole body of 1 rem (0.01 sievert) in any one hour at 30 centimeters from the source or from any surface that the radiation penetrates:
- A control device that prevents entry to the area when high radiation levels exist or upon entry causes the radiation level to be reduced below that level defining a high radiation area;
- A device that functions automatically to prevent use or operation of the radiation source or field while individuals are in the area;

- A control device that energizes a conspicuous visible or audible alarm signal so that the individual entering the high radiation area and the supervisor of the activity are made aware of the entry;
- Entryways that are locked. During periods when access to the area is required, positive control over each entry is maintained;
- Continuous direct or electronic surveillance that is capable of preventing unauthorized entry;
- A control device that will automatically generate audible and visual alarm signals to alert personnel in the area before use or operation of the radiation source and in sufficient time to permit evacuation of the area or activation of a secondary control device that will prevent use or operation of the source.

Very High Radiation Areas

- In addition to the above requirements, additional measures shall be implemented to ensure individuals are not able to gain unauthorized or inadvertent access to very high radiation areas.
- No control(s) shall be established in a high or very high radiation area that would prevent rapid evacuation of personnel.

i. Discuss methods of meeting the key requirements in 10 CFR 835, Subpart G (Posting and Labeling) based upon the guidance in the Posting and Labeling for Radiological Control Implementation Guide, including:

- **General posting and labeling requirements**
- **Controlled areas**
- **Radiological areas**

The following is taken from 10 CFR 835.601-606.

General Posting and Labeling Requirements

Except as otherwise provided in this subpart, postings and labels required by this subpart shall include the standard radiation warning trefoil in black or magenta imposed upon a yellow background.

Controlled Areas

Each access point to a controlled area (as defined at 10 CFR 835.2) shall be posted whenever radiological areas or radioactive material areas exist in the area. Individuals who enter only controlled areas without entering radiological areas or radioactive material areas are not expected to receive a total effective dose of more than 0.1 rem (0.001 sievert) in a year.

Signs used for this purpose may be selected by the contractor to avoid conflict with local security requirements.

Radiological Areas

Each access point to radiological areas and radioactive material areas (as defined at 10 CFR 835.2) shall be posted with conspicuous signs bearing the wording provided in this section.

Radiation Area

The words “Caution, Radiation Area” shall be posted at each radiation area.

High Radiation Area

The words “Caution, High Radiation Area” or “Danger, High Radiation Area” shall be posted at each high radiation area.

Very High Radiation Area

The words “Grave Danger, Very High Radiation Area” shall be posted at each very high radiation area.

Airborne Radioactivity Area

The words “Caution, Airborne Radioactivity Area” or “Danger, Airborne Radioactivity Area” shall be posted at each airborne radioactivity area.

Contamination Area

The words “Caution, Contamination Area” shall be posted at each contamination area.

High Contamination Area

The words “Caution, High Contamination Area” or “Danger, High Contamination Area” shall be posted at each high contamination area.

Radioactive Material Area

The words “Caution, Radioactive Material(s)”shall be posted at each radioactive material area.

j. Discuss methods of meeting the key requirements in 10 CFR 835, Subpart H (Records) based upon the guidance in the Occupational Radiation Protection Record-Keeping and Reporting Implementation Guide, including:

- **Individual monitoring records**
- **Monitoring and workplace records**
- **Administrative records**

The following is taken from 10 CFR 835.702-704.

Individual Monitoring Records

Except as authorized by 10 CFR 835.702(b), records shall be maintained to document doses received by all individuals for whom monitoring was conducted and to document doses received during planned special exposures, unplanned doses exceeding the monitoring thresholds of 10 CFR 835.402, and authorized emergency exposures.

Recording of the non-uniform equivalent dose to the skin is not required if the dose is less than 2 percent of the limit specified for the skin at 10 CFR 835.202(a)(4). Recording of internal dose (committed effective dose or committed equivalent dose) is not required for any monitoring result estimated to correspond to an individual receiving less than 0.01 rem (0.1 mSv) committed effective dose. The bioassay or air monitoring result used to make the estimate shall be maintained in accordance with 10 CFR 835.703(b) and the unrecorded internal dose estimated for any individual in a year shall not exceed the applicable monitoring threshold at 10 CFR 835.402(c).

The records required by this section shall:

- Be sufficient to evaluate compliance with subpart C of this part;
- Be sufficient to provide dose information necessary to complete reports required by subpart I of this part;
- Include the results of monitoring used to assess the following quantities for external dose received during the year:
- The effective dose from external sources of radiation (equivalent dose to the whole body may be used as effective dose for external exposure);
- The equivalent dose to the lens of the eye;
- The equivalent dose to the skin; and
- The equivalent dose to the extremities.
- Include the following information for internal dose resulting from intakes received during the year:
- Committed effective dose;
- Committed equivalent dose to any organ or tissue of concern; and
- Identity of radionuclides.
- Include the following quantities for the summation of the external and internal dose:
- Total effective dose in a year;
- For any organ or tissue assigned an internal dose during the year, the sum of the equivalent dose to the whole body from external exposures and the committed equivalent dose to that organ or tissue; and
- Cumulative total effective dose.
- Include the equivalent dose to the embryo/fetus of a declared pregnant worker.

Documentation of all occupational doses received during the current year, except for doses resulting from planned special exposures conducted in compliance with 10 CFR 835.204 and emergency exposures authorized in accordance with 10 CFR 835.1302(d), shall be obtained to demonstrate compliance with 10 CFR 835.202(a). If complete records documenting previous occupational dose during the year cannot be obtained, a written estimate signed by the individual may be accepted to demonstrate compliance.

For radiological workers whose occupational dose is monitored in accordance with 10 CFR 835.402, reasonable efforts shall be made to obtain complete records of prior years occupational internal and external doses. The records specified in this section that are identified with a specific individual shall be readily available to that individual. Data necessary to allow future verification or reassessment of the recorded doses shall be recorded. All records required by this section shall be transferred to the DOE upon cessation of activities at the site that could cause exposure to individuals.

Monitoring and Workplace Records

The following information shall be documented and maintained:

- Results of monitoring for radiation and radioactive material as required by subparts E and L of this part, except for monitoring required by 10 CFR 835.1102(d);
- Results of monitoring used to determine individual occupational dose from external and internal sources;
- Results of monitoring for the release and control of material and equipment as required by 10 CFR 835.1101; and

- Results of maintenance and calibration performed on instruments and equipment as required by 10 CFR 835.401(b).

Administrative Records

Training records shall be maintained, as necessary, to demonstrate compliance with 10 CFR 835.901. Actions taken to maintain occupational exposures as low as reasonably achievable, including the actions required for this purpose by 10 CFR 835.101, as well as facility design and control actions required by 10 CFR 835.1001, 835.1002, and 835.1003, shall be documented.

Records shall be maintained to document the results of internal audits and other reviews of program content and implementation. Written declarations of pregnancy, including the estimated date of conception, and revocations of declarations of pregnancy shall be maintained.

Changes in equipment, techniques, and procedures used for monitoring shall be documented. Records shall be maintained as necessary to demonstrate compliance with the requirements of 10 CFR 835.1201 and 835.1202 for sealed radioactive source control, inventory, and source leak tests.

k. Discuss methods of meeting the key requirements in 10 CFR 835, Subpart I (Reports to Individuals) based upon the guidance in the Occupational Radiation Protection Record-Keeping and Reporting Implementation Guide, including:
- **Annual Dose Report to Monitored Individuals**
- **Termination Report**

The following is taken from 10 CFR 835.801.

Annual Dose Report to Monitored Individuals

Radiation exposure data for individuals monitored in accordance with 10 CFR 835.402 shall be reported as specified in this section. The information shall include the data required under 10 CFR 835.702(c). Each notification and report shall be in writing and include: the DOE site or facility name, the name of the individual, and the individual's social security number, employee number, or other unique identification number.

Termination Report

Upon the request from an individual terminating employment, records of exposure shall be provided to that individual as soon as the data are available, but not later than 90 days after termination. A written estimate of the radiation dose received by that employee based on available information shall be provided at the time of termination, if requested. Each DOE- or DOE-contractor-operated site or facility shall, on an annual basis, provide a radiation dose report to each individual monitored during the year at that site or facility in accordance with 10 CFR 835.402.

Detailed information concerning any individual's exposure shall be made available to the individual upon request of that individual, consistent with the provisions of the Privacy Act (5 U.S.C. 552a). When a DOE contractor is required to report to the Department, pursuant to Departmental requirements for occurrence reporting and processing, any exposure of an

individual to radiation and/or radioactive material, or planned special exposure in accordance with 10 CFR 835.204(e), the contractor shall also provide that individual with a report on his or her exposure data included therein. Such report shall be transmitted at a time not later than the transmittal to the Department.

I. Discuss methods of meeting the key requirements in 10 CFR 835, Subpart J (Radiation Safety Training) based upon the guidance in the Radiation Safety Training Implementation Guide, including:

- **General employee training**
- **Radiological worker training**
- **Radiological control technician training**
- **Use of escorts**

The following is taken from 10 CFR 835.901.

General Employee Training

Each individual shall complete radiation safety training on the topics established at 10 CFR835.901(c) commensurate with the hazards in the area and the required controls:

- Before being permitted unescorted access to controlled areas; and
- Before receiving occupational dose during access to controlled areas at a DOE site or facility.

Radiological Worker Training

Each individual shall demonstrate knowledge of the radiation safety training topics established at 10 CFR 835.901(c), commensurate with the hazards in the area and required controls, by successful completion of an examination and performance demonstrations:

- Before being permitted unescorted access to radiological areas; and
- Before performing unescorted assignments as a radiological worker.

Radiological Control Technician Training

Radiation safety training shall include the following topics, to the extent appropriate to each individual's prior training, work assignments, and degree of exposure to potential radiological hazards:

- Risks of exposure to radiation and radioactive materials, including prenatal radiation exposure; Basic radiological fundamentals and radiation protection concepts;
- Physical design features, administrative controls, limits, policies, procedures, alarms, and other measures implemented at the facility to manage doses and maintain doses ALARA, including both routine and emergency actions;
- Individual rights and responsibilities as related to implementation of the facility radiation protection program;
- Individual responsibilities for implementing ALARA measures required by 10 CFR 835.101; and Individual exposure reports that may be requested in accordance with 10 CFR 835.801.

Use of Escorts

When an escort is used in lieu of training in accordance with paragraph (a) or (b) of this section, the escort shall:

- Have completed radiation safety training, examinations, and performance demonstrations required for entry to the area and performance of the work; and
- Ensure that all escorted individuals comply with the documented radiation protection program.

Radiation safety training shall be provided to individuals when there is a significant change to radiation protection policies and procedures that may affect the individual and at intervals not to exceed 24 months. Such training provided for individuals subject to the requirements of 10 CFR 835.901(b) (1) and (b) (2) shall include successful completion of an examination.

m. Discuss methods of meeting the key requirements in 10 CFR 835, Subpart K (Design and Control) based upon the guidance in the Occupational ALARA Program Implementation Guide, including:
- **Design features, administrative controls and procedural requirements**
- **Facility design and modification**
- **Control features**

The following is taken from 10 CFR 835.1001-1003.

Design Features, Administrative Controls and Procedural Requirements

Measures shall be taken to maintain radiation exposure in controlled areas ALARA through engineered and administrative controls. The primary methods used shall be physical design features (e.g., confinement, ventilation, remote handling, and shielding). Administrative controls shall be employed only as supplemental methods to control radiation exposure. For specific activities where use of engineered controls is demonstrated to be impractical, administrative controls shall be used to maintain radiation exposures ALARA.

Facility Design and Modification

During the design of new facilities or modification of existing facilities, the following objectives shall be adopted:
- Optimization methods shall be used to assure that occupational exposure is maintained ALARA in developing and justifying facility design and physical controls.
- The design objective for controlling personnel exposure from external sources of radiation in areas of continuous occupational occupancy (2000 hours per year) shall be to maintain exposure levels below an average of 0.5 millirem (5 ⌈Sv) per hour and as far below this average as is reasonably achievable. The design objectives for exposure rates for potential exposure to a radiological worker where occupancy differs from the above shall be ALARA and shall not exceed 20 percent of the applicable standards in 10 CFR 835.202.
- Regarding the control of airborne radioactive material, the design objective shall be, under normal conditions, to avoid releases to the workplace atmosphere and in any situation, to control the inhalation of such material by workers to levels that are ALARA; confinement and ventilation shall normally be used.
- The design or modification of a facility and the selection of materials shall include features that facilitate operations, maintenance, decontamination, and decommissioning.

Control Features

During routine operations, the combination of engineered and administrative controls shall provide that:

- The anticipated occupational dose to general employees shall not exceed the limits established at 10 CFR 835.202; and
- The ALARA process is utilized for personnel exposures to ionizing radiation.

n. Discuss methods of meeting the key requirements in 10 CFR 835, Subpart L (Radioactive Contamination Control).

The following is taken from 10 CFR 835.1101-1102.

Except as provided in paragraphs (b) and (c) of this section, material and equipment in contamination areas, high contamination areas, and airborne radioactivity areas shall not be released to a controlled area if:

- Removable surface contamination levels on accessible surfaces exceed the removable surface contamination values specified in appendix D of this part; or
- Prior use suggests that the removable surface contamination levels on inaccessible surfaces are likely to exceed the removable surface contamination values specified in appendix D of this part.

Material and equipment exceeding the removable surface contamination values specified in appendix D of this part may be conditionally released for movement on-site from one radiological area for immediate placement in another radiological area only if appropriate monitoring is performed and appropriate controls for the movement are established and exercised.

Material and equipment with fixed contamination levels that exceed the total contamination values specified in appendix D of this part may be released for use in controlled areas outside of radiological areas only under the following conditions:

- Removable surface contamination levels are below the removable surface contamination values specified in appendix D of this part; and
- The material or equipment is routinely monitored and clearly marked or labeled to alert personnel of the contaminated status.

Appropriate controls shall be maintained and verified which prevent the inadvertent transfer of removable contamination to locations outside of radiological areas under normal operating conditions. Any area in which contamination levels exceed the values specified in appendix D of this part shall be controlled in a manner commensurate with the physical and chemical characteristics of the contaminant, the radionuclides present, and the fixed and removable surface contamination levels.

Areas accessible to individuals where the measured total surface contamination levels exceed, but the removable surface contamination levels are less than, corresponding surface contamination values specified in appendix D of this part, shall be controlled as follows when located outside of radiological areas:

- The area shall be routinely monitored to ensure the removable surface contamination level remains below the removable surface contamination values specified in appendix D of this part; and

- The area shall be conspicuously marked to warn individuals of the contaminated status.

Individuals exiting contamination, high contamination, or airborne radioactivity areas shall be monitored, as appropriate, for the presence of surface contamination. Protective clothing shall be required for entry to areas in which removable contamination exists at levels exceeding the removable surface contamination values specified in appendix D of this part.

o. Discuss methods of meeting the key requirements in 10 CFR 835, Subpart M (Sealed Radioactive Source Control). Use the guidance in the Sealed Radioactive Source Accountability Implementation Guide to support the discussion on sealed source accountability.

The following is taken from 10 CFR 835.1201-1202.

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

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

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

Except for sealed radioactive 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.

Notwithstanding the requirements of paragraph (b) of this section, 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 as required by paragraph (a) of this section, and subject to source leak testing prior to being returned to service.

Notwithstanding the requirements of paragraphs (a) and (b) of this section, 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.

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

p. Discuss methods of meeting the key requirements in 10 CFR 835, Subpart N (Emergency Exposure Situations), including:

- **General provisions**
- **Emergency exposure situations**
- **Nuclear accident dosimetry**

The following is taken from 10 CFR 835.1301-1304.

General Provisions

A general employee whose occupational dose has exceeded the numerical value of any of the limits specified in 10 CFR 835.202 as a result of an authorized emergency exposure may be permitted to return to work in radiological areas during the current year providing that all of the following conditions are met:

- Approval is first obtained from the contractor management and the Head of the responsible DOE field organization;
- The individual receives counseling from radiological protection and medical personnel regarding the consequences of receiving additional occupational exposure during the year; and
- The affected employee agrees to return to radiological work.

All doses exceeding the limits specified in 10 CFR 835.202 shall be recorded in the affected individual's occupational dose record. When the conditions under which a dose was received in excess of the limits specified in 10 CFR 835.202, except those received in accordance with 10 CFR 835.204, have been eliminated, operating management shall notify the Head of the responsible DOE field organization.

Operations which have been suspended as a result of a dose in excess of the limits specified in 10 CFR 835.202, except those received in accordance with 10 CFR 835.204, may be resumed only with the approval of DOE.

Emergency Exposure Situations

The risk of injury to those individuals involved in rescue and recovery operations shall be minimized. Operating management shall weigh actual and potential risks against the benefits to be gained. No individual shall be required to perform a rescue action that might involve substantial personal risk. Each individual authorized to perform emergency actions likely to result in occupational doses exceeding the values of the limits provided at 10 CFR 835.202(a) shall be trained in accordance with 10 CFR 835.901(b) and briefed beforehand on the known or anticipated hazards to which the individual will be subjected.

Nuclear Accident Dosimetry

Installations possessing sufficient quantities of fissile material to potentially constitute a critical mass, such that the excessive exposure of individuals to radiation from a nuclear accident is possible, shall provide nuclear accident dosimetry for those individuals. Nuclear accident dosimetry shall include the following:

- A method to conduct initial screening of individuals involved in a nuclear accident to determine whether significant exposures to radiation occurred;
- Methods and equipment for analysis of biological materials;
- A system of fixed nuclear accident dosimeter units; and
- Personal nuclear accident dosimeters.

q. Discuss methods of meeting the key requirements on Administrative Control Levels, Work Authorizations, Radiation Safety Training, and Posting.

Administrative Control Levels

The following is taken from DOE-HDBK-1098-2008.

To accomplish DOE's objective of maintaining individual doses well below regulatory limits, challenging numerical administrative control levels should be established below the regulatory limits to administratively control and help minimize individual and collective radiation dose. These control levels should be multi-tiered with increasing levels of authority required to approve higher administrative control levels. Unless otherwise indicated, administrative, lifetime, and special control levels and dose limits are stated in terms of the total effective dose, which is the sum of the doses received from internal and external sources.

Facility management should establish an annual facility administrative control level based upon an evaluation of historical and projected radiation exposures, work load, and mission. This control level should be reevaluated annually. The choice of a low level for one year does not preclude choosing either a higher or lower level in a subsequent year. The facility administrative control level should be approved by the contractor senior site management.

When there is wide variation in the expected doses to the various work groups at a single facility, facility management should develop work group-specific administrative control levels to control worker doses below the regulatory limits.

No individual should be allowed to exceed the facility administrative control level without the prior written approval of the radiological control organization and cognizant facility management. Authorization by the contractor senior management is recommended.

Work Authorizations

The following is taken from DOE-STD-1136-2004.

Written authorizations shall be required to control entry into and work within radiological areas and shall specify radiation protection measures commensurate with the existing and potential hazards (10 CFR 835.501(d)). ALARA considerations need to be included in the work authorization. One approach that works well is the inclusion of an ALARA worksheet with the radiological work permit (RWP). Although the written work authorizations may take any appropriate form (e.g., written procedures, policy statements, technical work documents, etc.), RWPs are most often used. RWPs should be used for entry into high and very high radiation areas, high contamination areas, and airborne radioactivity areas. RWPs should also be used to control entry into radiation and contamination areas and for handling materials with removable contamination. The RWPs should be initiated by the work group responsible for the activity. All RWPs should be reviewed and approved by the radiological control staff and cognizant line management. The RCS provides detailed guidance for RWPs.

Radiological workers should read and understand the applicable RWP before entering the affected area. Copies of the RWP should be located at the access point to the applicable area. Workers should acknowledge by signature or through electronic means that they have read, understood, and will comply with the RWP before they initially enter the area and after changes. Out-of-date RWPs should be removed.

Radiation Safety Training

[Note: This topic is discussed in detail in item "l" above.]

Posting

[Note: This topic is discussed in detail in item "i" above.]

r. Explain how the Radiation Control Technical Positions, 10 CFR 835 exemption decisions, and official interpretations of 10 CFR 835 are used to adapt the radiation protection requirements to unique conditions at DOE sites and facilities.

This is a site-specific requirement. The Qualifying Official will evaluate its completion.

11. Radiation protection personnel shall demonstrate a working level knowledge of the following DOE Policy, Order, and Manual Directives, and Technical Standards related to radiation protection:

- **DOE P 450.2A, Identifying, Implementing, and Complying with Environmental, Safety, and Health Requirements**
- **DOE Order 5400.5, Ch. 2, Radiation Protection of the Public and the Environment, or when promulgated, 10 CFR 834, Radiation Protection of the Public and the Environment and associated Implementation Guides**
- **DOE Order 5480.4, Ch. 4, Environmental Protection, Safety, and Health Protection Standards**
- **DOE M 231.1-2, Occurrence Reporting and Processing of Operations Information**
- **DOE Order 5480.19, Ch. 2, Conduct of Operations Requirements for DOE Facilities**
- **DOE-STD-1098-99, Department of Energy Standard — Radiological Control**
- **DOE-STD-1121-98, Department of Energy Standard — Internal Dosimetry**

a. Describe the relevant requirements, interrelationships, and importance of the listed Orders, notices, codes and regulations, guides, and technical manual(s).

DOE P 450.2A, Identifying, Implementing, and Complying with Environmental, Safety, and Health Requirements

DOE Policy 450.2A sets forth the framework for environmental, safety, and health (ES&H) requirements so that work is performed in the DOE complex in a manner that ensures adequate protection of workers, the public and the environment. This includes

- integrated review of safety requirements
- transition to rules and revised orders
- ensuring adequate protection at diverse facilities
- continuity of ongoing efforts
- implementation plans
- guidance documents (including technical standards)
- compliance with requirements by contractual mechanisms and
- compliance with nuclear safety requirements, including the enforcement through civil and criminal penalties

DOE Order 5400.5, Ch. 2, Radiation Protection of the Public and the Environment

Environmental requirements for facility operations, such as permissible public doses, limits on effluent discharge, and the release of waste from DOE sites, are stated in DOE Order 5400.5 and will be released as 10 CFR 834.

DOE Order 5400.5 establishes standards and requirements for operations of DOE and its contractors with respect to protection of the public and the environment against undue risk from radiation. The Order is divided into four chapters that discuss the general topics covered in the Order, requirements for radiation protection of the public and the environment, derived concentration guides (DCGs) for air and water, and residual radioactive material.

DOE Order 5480.4, Ch. 4, Environmental Protection, Safety, and Health Protection Standards

[Note: This Order has been canceled.]

DOE M 231.1-2, Occurrence Reporting and Processing of Operations Information

This manual provides detailed requirements to supplement DOE O 231.1A, *Environment, Safety, and Health Reporting*, dated 08-19-03. This manual is approved for use by all DOE elements and their contractors.

This manual provides detailed information for reporting occurrences and managing associated activities at DOE facilities, including NNSA facilities. It complements DOE O 231.1A, Environment, Safety, and Health Reporting and its use is required by that Order.

Information gathered in response to the requirements in this order and manual is used for analysis of the Department's performance in environmental protection and the safety and health of its workers and the public. This information is also used to develop lessons learned and document events that impact DOE operations.

DOE Order 5480.19, Ch. 2, Conduct of Operations Requirements for DOE Facilities

The purpose of this Order is to provide requirements and guidelines for Departmental elements to use in developing directives, plans, and/or procedures relating to the conduct of operations at DOE facilities. The implementation of these requirements should result in improved quality and uniformity of operations.

The Order is divided into 18 chapters. A summary of the requirements for each chapter follows:

- Chapter I, Operations Organization and Administration — DOE facility policies should describe the philosophy of standards of excellence under which the facility is operated, and clear lines of responsibility for normal and emergency conditions should be established.
- Chapter II, Shift Routines and Operation Practices — Standards for the professional conduct of operations personnel should be established and followed so that operator performance meets the expectations of DOE and facility management. This chapter describes important aspects of routine shift activities and watchstanding practices.
- Chapter III, Control Area Activities — Control area activities should be conducted in a manner that achieves safe and reliable facilities operations. Guidelines are presented for control area access, professional behavior, and monitoring control panels and equipment.
- Chapter IV, Communications — This chapter provides important guidelines for a plant program for audible communications, including emergency communications systems, public address (PA) systems, contacting operators, and radios.

- Chapter V, Control of On-Shift Training — On-shift training should be conducted so that the trainee satisfactorily completes all of the required training objectives and receives maximum learning benefits. This chapter includes documentation of instructor qualifications and supervision and control of trainees.
- Chapter VI, Investigation of Abnormal Events — A program for the investigation of abnormal events should ensure that facility events are thoroughly investigated to assess the impact of the event, to determine the root cause of the event, and to ascertain whether the event is reportable.
- Chapter VII, Notifications — Guidelines to ensure timely notification of appropriate DOE personnel and other agencies, when required, should be employed to ensure that the facility is responsive to public health and safety concerns.
- Chapter VIII, Control of Equipment and System Status — Overall guidelines on good operating discipline should ensure that facility configuration is maintained in accordance with design requirements. The operating shift should know the status of equipment and systems.
- Chapter IX, Lockouts and Tagouts — The important elements of a lockout/tagout program are covered in this chapter. DOE's intention is to meet the requirements of 29 CFR 1910.
- Chapter X, Independent Verification — Guidelines to provide a high degree of reliability in ensuring the correct facility operations and the correct position of components such as valves, switches, and circuit breakers are provided in this chapter.
- Chapter XI, Log Keeping — Records should contain a narrative log of the facility's status and all events, as required, to provide an accurate history of facility operations.
- Chapter XII, Operations Turnover — Guidelines for the important aspects of a good shift turnover, including checklists, and control panel walkdowns are provided in this chapter.
- Chapter XIII, Operations Aspects of Facility Chemistry and Unique Processes — Operational monitoring of facility chemistry or unique process data and parameters should ensure that parameters are properly maintained.
- Chapter XIV, Required Reading — Proper use of a required reading file by operations personnel should ensure that appropriate individuals are made aware of important information that is related to job assignments.
- Chapter XV, Timely Orders to Operators — Key features for a means of operations management to communicate short-term information and administrative instructions to operations personnel are covered in this chapter.
- Chapter XVI, Operations Procedures — Operations procedures should provide appropriate direction to ensure that the facility is operated within its design bases and should be effectively used to support safe operations of the facility.
- Chapter XVII, Operator Aid Postings — Operator aid programs should be established to ensure that operator aids, which are posted, are current, correct, and useful.
- Chapter XVIII, Equipment and Piping Labeling — A well established and maintained labeling program should ensure that facility personnel are able to positively identify equipment they operate, in accordance with OSHA regulations.

DOE-STD-1098-99, Radiological Control

[Note: DOE-STD-1098-99 has been superseded by DOE-STD-1098-2008, from which the following is taken.]

The Department of Energy (DOE) has developed this standard to assist line managers in meeting their responsibilities for implementing occupational radiological control programs. DOE has established regulatory requirements for occupational radiation protection in 10 CFR 835, "Occupational Radiation Protection," amended 2007. Failure to comply with these requirements may lead to appropriate enforcement actions as authorized under the Price Anderson Act Amendments (PAAA). While this Standard does not establish requirements, it does restate, paraphrase, or cite many (but not all) of the requirements of 10 CFR 835 and related documents (e.g., occupational safety and health, hazardous materials transportation, and environmental protection standards).

Because of the wide range of activities undertaken by DOE and the varying requirements affecting these activities, DOE does not believe that it would be practical or useful to identify and reproduce the entire range of health and safety requirements in this Standard and therefore has not done so. In all cases, DOE cautions the user to review any underlying regulatory and contractual requirements and the primary guidance documents in their original context to ensure that the site program is adequate to ensure continuing compliance with the applicable requirements.

DOE-STD-1121-98, Internal Dosimetry

This technical standard is created to provide a resource for those engaged in the science and practice of internal dosimetry within the DOE complex. This standard defines minimum levels of acceptable performance and provides basic procedural guidelines for evaluating the internal radiation dose equivalent that may be received by radiation workers from intakes of radionuclides. This set of defined internal dosimetry performance criteria meets the requirements set forth in 10 CFR 835 for monitoring the workplace, for assessing internal radiation doses to workers at DOE facilities, and for recording and reporting requirements as they apply to internal dosimetry programs.

b. Discuss the role of radiation protection personnel with respect to these Orders and regulations.

The role of radiation protection personnel is covered in item "a" of this competency statement.

c. Discuss how conduct of operations is applied to radiation protection activities.

Radiological controls fall under the operation of a facility and, therefore, DOE Order 5480.19 applies to the radiological control organization. The following is taken from DOE Order 5480.19 regarding the conduct of operations.

A high level of performance in DOE operations is accomplished by establishment of high-operating standards by management, communicating operating standards to the working level, providing sufficient resources to the operations department, ensuring personnel are well trained, closely monitoring performance in operations, and holding workers and their supervisors accountable for their performance in conducting activities.

Senior management establishes operating standards, considering input from the working level when appropriate. The working level will more eagerly support the standards when they have had input into the development of those standards. The standards should define operating objectives, establish expected performance levels, and clearly define responsibility in plant operations. Standards for operating activities should also be integrated into procedures and programs. Operating standards should also be communicated to the working level by training workers in operating practices and by supervisory monitoring and guidance of work. Sufficient staff, equipment, and funding should be allocated to permit the operations department to effectively perform its functions.

Performance in operations should be closely monitored by facility management, and operating reports and goals should be used so that the performance of the operating department can be effectively measured. Operations personnel should be held accountable for their performance through supervisory counseling, performance appraisals, and when necessary, disciplinary measures. Remedial training should be provided when appropriate.

Procedures or other definitive documentation should specify policies that are to be applied to operations. These policies should specify goals and the means to achieve those goals. These documents should also provide for the types of controls necessary to implement policies.

Operating procedures should be based on facility and DOE guidance for operations. Responsibilities for implementing these policies, including the responsibility of shift personnel, if applicable, should be clearly defined. Operations personnel should clearly understand their authority, responsibility, accountability, and interfaces with other groups.

The operations supervisor for DOE facilities should be provided with sufficient resources in materials and personnel to accomplish assigned tasks without requiring excessive overtime by the operations staff. These resources should include technical personnel needed to support the operations. A long-range staffing plan that anticipates personnel losses should be developed and implemented.

As described in chapter VI of DOE Order 5480.19, operating problems should be documented and evaluated. Based on assessments of these problems, corrective actions should be taken to improve the performance of the operations department's performance. Additionally, frequent direct observation of operations activities by supervisors and managers is essential to monitoring operations performance.

Safety, environment, and operating goals should be used as a management tool for involving cognizant groups or individuals in improving operating performance and for measuring operating effectiveness. Operations goals should be established.

d. Discuss how the standard, Radiological Control, is now applied (i.e., as a requirement or as a technical standard) in your program, or at the site(s) or facility(s) for which you have responsibility.

This is a site-specific requirement. The Qualifying Official will evaluate its completion.

e. Discuss the following as they relate to occurrence reporting:

- **How soon after an event or condition is identified must it be characterized?**
- **Who must be notified at the facility where it occurred?**
- **What are the two broad groups or conditions in which a health physicist would likely be involved in identifying the reportable event?**

The following is taken from DOE M 231.1-2

The Facility Manager must categorize all occurrences, except Operational Emergencies, within 2 hours of discovery by the cognizant facility staff following the site/facility-specific procedures developed in accordance with section 9 of this Manual.

The Facility Manager must notify the DOE Facility Representative (in a manner determined locally) and the DOE Headquarters Operations Center DOE HQ OC), as required, of the following reportable occurrences as soon as practical (i.e., promptly), but no later than 2 hours after categorization.

The following is taken from DOE-STD-1121-2008.

Radiation protection and health physics expertise is rare in occupational medicine physicians and medical staff. Thus the health physicist will likely need to work closely with medical staff in dose reduction therapy. The decision to commence therapy for dose reduction is a medical decision which cannot be delegated to the health physicist. However, the health physicist can identify the circumstances under which therapy would seem appropriate, and advise the medical staff on the likely efficacies of treatment alternatives. Once therapy has commenced, bioassay measurements are required to determine the efficacy of therapy. The interpretation of those bioassay measurements will likely fall to the health physicist.

DOE facility health physics staff should establish contact with the cognizant medical staff prior to an emergency. Once a significant potential intake event occurs, the administrative and technical pressures associated with response and case management can become intense. Prior efforts to establish good communications will pay dividends.

12. Radiation protection personnel shall demonstrate a familiarity level knowledge of the identification, reporting, investigation, and enforcement related to potential noncompliance with nuclear safety requirements.

a. Describe the purpose and scope of the Price-Anderson Amendments Act.

The following is taken from Public Law 100-408.

The Price-Anderson Act was enacted into law in 1957 and has been revised several times. It constitutes section 170 of the Atomic Energy Act. The latest revision was enacted through the “Energy Policy Act of 2005,” and extended it through December 31, 2025.

The main purpose of the Price-Anderson Act is to ensure the availability of a large pool of funds (currently about $10 billion) to provide prompt and orderly compensation of members of the public who incur damages from a nuclear or radiological incident no matter who might be liable. The Act provides “omnibus” coverage, that is, the same protection available for a covered licensee or contractor extends through indemnification to any persons who may be

legally liable, regardless of their identity or relationship to the licensed activity. Because the Act channels the obligation to pay compensation for damages, a claimant need not sue several parties but can bring its claim to the licensee or contractor.

The Price-Anderson Act requires Nuclear Regulatory Commission licensees and Department of Energy contractors to enter into agreements of indemnification to cover personal injury and property damage to those harmed by a nuclear or radiological incident, including the costs of incident response or precautionary evacuation and the costs of investigating and defending claims and settling suits for such damages. The scope of the Act includes nuclear incidents in the course of the operation of power reactors; test and research reactors; Department of Energy nuclear and radiological facilities; and transportation of nuclear fuel to and from a covered facility.

b. Discuss the Price-Anderson Amendments Act's applicability to the Department's nuclear safety activities.

The following is taken from DOE-STD-1098-2008.

DOE has established regulatory requirements for occupational radiation protection in Title 10 of the Code of Federal Regulations, Part 835 (10 CFR 835), "Occupational Radiation Protection," amended 2007. Failure to comply with these requirements may lead to appropriate enforcement actions as authorized under the Price Anderson Act Amendments (PAAA). While this Standard does not establish requirements, it does restate, paraphrase, or cite many (but not all) of the requirements of 10 CFR 835 and related documents (e.g., occupational safety and health, hazardous materials transportation, and environmental protection standards). Because of the wide range of activities undertaken by DOE and the varying requirements affecting these activities, DOE does not believe that it would be practical or useful to identify and reproduce the entire range of health and safety requirements in this Standard and therefore has not done so. In all cases, DOE cautions the user to review any underlying regulatory and contractual requirements and the primary guidance documents in their original context to ensure that the site program is adequate to ensure continuing compliance with the applicable requirements.

c. Discuss the purpose and scope of the current nuclear safety rules, including:
- **10 CFR 708, DOE Contractor Employee Protection Program**
- **10 CFR 820, Procedural Rules for DOE Nuclear Activities**
- **10 CFR 830, Nuclear Safety Management**

10 CFR 708, DOE Contractor Employee Protection Program

This part provides procedures for processing complaints by employees of DOE contractors alleging retaliation by their employers for disclosure of information concerning danger to public or worker health or safety, substantial violations of law, or gross mismanagement, as well as for participation in Congressional proceedings, or for refusal to participate in dangerous activities.

10 CFR 820, Procedural Rules for DOE Nuclear Activities

This part sets forth the procedures to govern the conduct of persons involved in DOE nuclear activities and, in particular, to achieve compliance with the DOE Nuclear Safety

Requirements by all persons subject to those requirements. Questions that are not addressed in this part shall be resolved at the discretion of the DOE Official.

Activities and facilities covered under Executive Order (E.O.) 12344, 42 U.S.C. 7158 note, pertaining to Naval nuclear propulsion, are excluded from the requirements of subparts D and E of this part regarding interpretations and exemptions related to this part. The Deputy Assistant Secretary for Naval Reactors or his designee will be responsible for formulating, issuing, and maintaining appropriate records of interpretations and exemptions for these facilities and activities.

10 CFR 830, Nuclear Safety Management

This part governs the conduct of DOE contractors, DOE personnel, and other persons conducting activities, including providing items and services that affect, or may affect, the safety of DOE nuclear facilities.

d. Discuss the Department's enforcement program, including:

- **Identification and reporting of potential noncompliance with nuclear safety requirements**
- **Roles and responsibilities of DOE employees**

The following is taken from 10 CFR 820.24.

If the Director has reason to believe a person has violated or is continuing to violate a provision of the Act or a DOE Nuclear Safety Requirement, he may file a Preliminary Notice of Violation. The Notice and any transmittal documents shall contain sufficient information to fairly apprise the respondent of the facts and circumstances of the alleged violations and the basis of any proposed remedy, and to properly indicate what further actions are necessary by or available to respondent.

Within 30 days after the filing of a preliminary notice of violation, the respondent shall file a reply.

The reply shall be in writing and signed by the person filing it. The reply shall contain a statement of all relevant facts pertaining to the situation that is the subject of the Notice. The reply shall state any facts, explanations and arguments which support a denial that a violation has occurred as alleged; demonstrate any extenuating circumstances or other reason why the proposed remedy should not be imposed or should be mitigated; and furnish full and complete answers to the questions set forth in the notice. Copies of all relevant documents shall be submitted with the reply. The reply shall include a discussion of the relevant authorities which support the position asserted, including rulings, regulations, interpretations, and previous decisions issued by DOE.

The respondent may terminate an enforcement action if the reply agrees to comply with the proposed remedy and waives any right to contest the Notice or the remedy. If a respondent elects this option, the preliminary notice of violation shall be deemed a final order upon the filing of the reply.

13. Radiation protection personnel shall demonstrate a familiarity level knowledge of radioactive waste management:

a. Discuss the Department's policy regarding the handling and management of waste as described in DOE O 435.1, Ch. 1, Radioactive Waste Management.

The following is taken from DOE O 435.1.

DOE radioactive waste management activities shall be systematically planned, documented, executed, and evaluated. Radioactive waste shall be managed to

- protect the public from exposure to radiation from radioactive materials;
- protect the environment;
- protect workers;
- comply with applicable Federal, state, and local laws and regulations.

All radioactive waste shall be managed in accordance with the requirements in DOE M 435.1-1, *Radioactive Waste Management Manual*.

DOE, within its authority, may impose such requirements, in addition to those established in this order, as it deems appropriate and necessary to protect the public, workers, and the environment, or to minimize threats to property.

b. Define the following terms:

- **Low-level waste**
- **High-level waste**
- **Transuranic waste**
- **Mixed waste**

The following definitions are taken from DOE M 435.1-1.

Low-Level Waste

Low-level radioactive waste is radioactive waste that is not high-level radioactive waste, spent nuclear fuel, transuranic waste, byproduct material (as defined in section 11e.(2) of the Atomic Energy Act of 1954, as amended), or naturally occurring radioactive material.

High-Level Waste

High-level waste is the highly radioactive waste material resulting from the reprocessing of spent nuclear fuel, including liquid waste produced directly in reprocessing and any solid material derived from such liquid waste that contains fission products in sufficient concentrations; and other highly radioactive material that is determined, consistent with existing law, to require permanent isolation. [Adapted from: *Nuclear Waste Policy Act of 1982*, as amended]

Transuranic Waste

Transuranic waste is radioactive waste containing more than 100 nanocuries (3700 becquerels) of alpha-emitting transuranic isotopes per gram of waste, with half-lives greater than 20 years, except for:

- High-level radioactive waste;

- Waste that the Secretary of Energy has determined, with the concurrence of the Administrator of the Environmental Protection Agency does not need the degree of isolation required by the 40 CFR 191 disposal regulations; or
- Waste that the Nuclear Regulatory Commission has approved for disposal on a case-by-case basis in accordance with 10 CFR 61.

Mixed Waste

Waste that contains both source, special nuclear, or by-product material subject to the *Atomic Energy Act of 1954*, as amended, and a hazardous component subject to the *Resource Conservation and Recovery Act*. [Adapted from: *Federal Facility Compliance Act of 1992*]

c. Discuss the Department's policies on waste management, including:

- **Generation reduction**
- **Segregation**
- **Minimization**
- **Pollution prevention**
- **Disposal**

Generation Reduction

The following is taken from DOE O 450.1A.

The purpose of DOE O 450.1A is to implement sound stewardship practices that are protective of the air, water, land, and other natural and cultural resources impacted by DOE operations, and by which DOE cost effectively meets or exceeds compliance with applicable environmental, public health, and resource protection requirements. The objectives are—

- To implement sustainable practices for enhancing environmental, energy, and transportation management performance, as stipulated in section 3(a) of E.O. 13423, "Strengthening Federal Environmental, Energy, and Transportation Management," through environmental management systems that are part of Integrated Safety Management (ISM) systems established pursuant to DOE P 450.4, *Safety Management System Policy*, dated 10-15-96.
- To achieve the DOE Sustainable Environmental Stewardship goals found in the attachment to this order.

Segregation

The purpose of sustainable environmental stewardship is to establish Department of Energy (DOE) goals that advance the sustainable practices for enhancing environmental, energy, and transportation management performance, as stipulated in E.O. 13423, "Strengthening Federal Environment, Energy, and Transportation Management."

The objective of the goals is to reduce environmental hazards, protect environmental resources, minimize life-cycle cost and liability of DOE programs, and maximize operational sustainability by eliminating or minimizing the generation of wastes and other pollutants, through source reduction including segregation, substitution, and reuse that would otherwise require storage, treatment, disposal, and long-term monitoring and surveillance (i.e., future environmental legacies).

Minimization

The following is taken from DOE P4 and DOE G 450.1-5.

Minimization is defined as "actions that economically avoid or decrease waste production by reducing waste generation at the source, reducing the toxicity of hazardous waste, improving efficiency of energy usage, or recycling wastes."

48 CFR 970.5223-1, "Integration of Environment, Safety, and Health into Work Planning and Execution," states that contractors "shall ensure that management of ES&H functions and activities becomes an integral but visible part of the contractor's work planning and execution processes." The regulation clarifies that safety includes P2 and waste minimization.

Pollution Prevention

The following is taken from DOE G 450.1-1A.

A source reduction as defined in the Pollution Prevention Act and other practices that reduce or eliminate the creation of pollutants through (1) increased efficiency in the use of raw materials, energy, water, or other natural resources or (2) protection of natural resources by conservation. The Department of Energy has expanded this definition to include recycling.

Disposal

The following is taken from DOE M 435.1-1.

Disposal is defined as "emplacement of waste in a manner that ensures protection of the public, workers, and the environment with no intent of retrieval and that requires deliberate action to regain access to the waste." [Adapted from: DOE 5820.2A]

Site office managers are responsible for ensuring radioactive waste is disposed of in a manner that protects the public, workers, and the environment, and in accordance with a radioactive waste management basis. They are also responsible for reviewing specific transuranic or low-level waste documentation, including the performance assessment and composite analysis, or appropriate CERCLA documentation, prior to forwarding them to Headquarters (HQ) for approval, and for obtaining and ensuring that the facility is operated in accordance with the disposal authorization statement. Site office managers must also conduct performance assessments and composite analysis maintenance.

Disposal of high-level waste must be in accordance with the provisions of the Atomic Energy Act of 1954, as amended, the Nuclear Waste Policy Act of 1982, as amended or any other applicable statutes.

Transuranic waste shall be disposed of in accordance with the requirements of 40 CFR 191, "Environmental Radiation Protection Standards for Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes."

A disposal authorization statement shall be obtained prior to construction of a new low-level waste disposal facility. Field elements with existing low-level waste disposal facilities shall obtain a disposal authorization statement in accordance with the schedule in the Complex-Wide Low-Level Waste Management Program Plan. The disposal authorization statement

shall be issued based on a review of the facility's performance assessment, composite analysis, performance assessment and composite analysis maintenance, preliminary closure plan, and preliminary monitoring plan. The disposal authorization statement shall specify the limits and conditions on construction, design, operations, and closure of the low-level waste facility based on these reviews. A disposal authorization statement is a part of the radioactive waste management basis for a disposal facility.

A preliminary monitoring plan for a low-level waste disposal facility shall be prepared and submitted to Headquarters for review with the performance assessment and composite analysis. The monitoring plan shall be updated within one year following issuance of the disposal authorization statement to incorporate and implement conditions specified in the disposal authorization statement.

The site-specific performance assessment and composite analysis shall be used to determine the media, locations, radionuclides, and other substances to be monitored.

The environmental monitoring program shall be designed to include measuring and evaluating releases, migration of radionuclides, disposal unit subsidence, and changes in disposal facility and disposal site parameters which may affect long-term performance.

The environmental monitoring programs shall be capable of detecting changing trends in performance to allow application of any necessary corrective action prior to exceeding the facility's performance objectives.

d. Discuss the process for determining whether or not waste is classified as mixed waste.

The following is taken from U.S. EPA, *Wastes*, Hazardous Waste

Mixed waste is waste containing both radioactive and hazardous components as defined by the Atomic Energy Act (AEA) and the Resource Conservation and Recovery Act (RCRA), respectively. Mixed waste contains listed or characteristic RCRA hazardous waste and regulated radioactive material. The radioactive material may be waste which contains SNM, regulated under the AEA of 1954, or may be waste or residue which contains NRC nuclear material, such as medical isotopes or commercial spent nuclear fuel (SNF). Categorically, however, SNM and SNF that are contaminated with hazardous waste are not considered mixed waste under RCRA.

Mixed waste also refers to waste that contains RCRA hazardous waste constituents, but is also managed under the Toxic Substances Control Act (TSCA) regulations. This type of waste is called TSCA mixed waste.

14. Radiation protection personnel shall demonstrate a working level knowledge of DOE requirements and guidance related to safety management.

a. Describe the relevant requirements, purpose, interrelationships, and importance of the following requirements and guides to radiation protection activities:
- **10 CFR 830, "Nuclear Safety Management"**
- **DOE M 411.1-1C, Safety Management Functions, Responsibilities, and Authorities**
- **DOE O 414.1-1A, Ch. 1, Quality Assurance**
- **DOE G 414.1-2, Quality Assurance Management System Guide for use with 10 CFR 830.120 and DOE O 414.1-1A**
- **DOE O 420.1A, Facility Safety**
- **DOE G 421.1-1, DOE Good Practices Guide: Criticality Safety Good Practices Program Guide for DOE Nonreactor Nuclear Facilities**
- **DOE G 421.1-2, Implementation Guide for Use in Developing Documented Safety Analyses to Meet Subpart B of 10 CFR 830**
- **DOE G 423.1-1, Implementation Guide for Use in Developing Technical Safety Requirements**
- **DOE G 424.1-1, Implementation Guide for Use in Addressing Unreviewed Safety Question Requirements**
- **DOE O 430.1B, Real Property Asset Management**
- **DOE-STD-1073-2003, Configuration Management Program**
- **DOE-STD-3009-94, Ch. 2, Preparation Guide for U.S. DOE Nonreactor Nuclear Facility Safety Analysis Reports**
- **DOE-HDBK-3010-94, Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities**
- **DOE-STD-3011-2002, Guidance for Preparation of Basis for Interim Operation (BIO) Documents**
- **DOE-STD-1027-92, Ch. 1, Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports**

10 CFR 830, Nuclear Safety Management

This document establishes safety basis requirements for hazard category 1, 2, and 3 DOE nuclear facilities.

In establishing the safety basis for a hazard category 1, 2, or 3 DOE nuclear facility, the contractor responsible for the facility must
- define the scope of the work to be performed;
- identify and analyze the hazards associated with the work;
- categorize the facility consistent with DOE-STD-1027-92, Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports, Change Notice 1, September 1997;
- prepare a documented safety analysis for the facility; and
- establish the hazard controls upon which the contractor will rely to ensure adequate protection of workers, the public, and the environment.

DOE M 411.1-1C, Safety Management Functions, Responsibilities, and Authorities

This DOE manual defines safety management functions, responsibilities, and authorities for DOE senior management with responsibilities for line, support, oversight, and enforcement

actions. It provides detailed requirements to supplement DOE P 411.1, Safety Management Functions, Responsibilities, and Authorities Policy, dated 1-28-97.

The Secretary of Energy has the primary responsibility for ensuring that work at DOE or Department facilities and sites is performed in a manner that adequately protects the worker, the public, and the environment. This responsibility flows from the Secretary through line management to the individuals performing the work. The goal of performing work safely is reflected in the guiding principles and core management functions established in DOE P 450.4, Safety Management System Policy, dated 10-15-96, and is codified in the Department of Energy Acquisition Regulations, found at Title 48 of the Code of Federal Regulations (CFR), 970.5204-2, and 970.5223-1 (48 CFR 970.5204-2 and 970.5223-1).

These guiding principles include the following:

- Line management is responsible for protection of employees, the public, and the environment.
- DOE and its contractors must clearly define and maintain the lines of responsibility for ensuring protection of ES&H at all organizational levels.

This manual addresses both of these guiding principles for DOE by documenting DOE senior management functions, responsibilities, and authorities relating to safety management. The term "safety management" for purposes of this manual refers to those DOE functions and responsibilities that pertain to and govern the safety of operations and activities at DOE sites and facilities.

This document is required by DOE P 411.1, Safety Management Functions, Responsibilities, and Authorities Policy, dated 1-28-97, which mandates the development of a corporate-level document to establish the clear lines of responsibilities and authorities that are necessary to

- develop and implement requirements and standards that are necessary to provide reasonable assurance that workers, the public, and the environment are adequately protected;
- define essential safety management functions and establish unambiguous DOE roles, responsibilities, and authorities for executing them to accomplish the authorized work;
- clarify the roles, responsibilities, lines of authority, and delegations between HQ and field organizations;
- ensure compliance with legal requirements and manage against contractual requirements;
- define functional relationships and responsibilities among DOE line, support, oversight, and enforcement organizations;
- address the coordination of line direction from multiple program offices at a single site.

This manual is the corporate-level document that defines safety management functions, responsibilities, and authorities for DOE senior management with responsibilities for line, support, oversight, and enforcement functions.

DOE O 414.1A, Ch. 1, Quality Assurance

[Note: DOE O 414.1A has been superseded by DOE O 414.1C, from which the following is taken.]

The objectives of DOE O 414.1C, Quality Assurance are the following:

- To ensure that Department of Energy (DOE), including National Nuclear Security Administration (NNSA), products and services meet or exceed customers' expectations;
- To achieve quality assurance (QA) for all work based upon the following principles.
  - That quality is assured and maintained through a single, integrated, effective QA program (i.e., management system);
  - That management support for planning, organization, resources, direction, and control is essential to QA;
  - That performance and quality improvement require thorough, rigorous assessment and corrective action;
  - That workers are responsible for achieving and maintaining quality;
  - That environmental, safety, and health risks and impacts associated with work processes can be minimized while maximizing reliability and performance of work products;
- To establish quality process requirements to be implemented under a QA program (QAP) for the control of suspect/counterfeit items (S/CIs), safety issue corrective actions, and safety software.

DOE G 414.1-2A, Quality Assurance Management System Guide for Use with 10 CFR 830.120 and DOE O 414.1-1A, Quality Assurance

This guide provides information on principles, requirements, and practices used to establish and implement an effective quality assurance program (QAP) or quality management system consistent with the requirements of 10 CFR 830, subpart A, and DOE O 414.1C.

This guidance includes methods for the interrelated functions and responsibilities of managing, performing, and assessing work. Implementation of a quality management system will contribute to improved safety, management, and reliability of DOE products and services. The methods and references described in this guide are not mandatory and do not add, modify, or delete any requirements identified in the QA rule and Order. Use of this guide in conjunction with appropriate standards will facilitate development and approval of a QAP compliant with the QA rule and Order. An organization may select alternative methods to document and implement its quality management system as long as the requirements of the QA rule and Order are satisfied. The content of the quality management system must be based on an organization's unique set of responsibilities, its product/service realization process, hazards, and customer expectations.

DOE O 420.1A, Facility Safety

[Note: DOE O 420.1A has been superseded by DOE O 420.1B, from which the following is taken.]

This order includes requirements for nuclear and explosives safety design criteria, fire protection, nuclear criticality safety, natural phenomena hazards mitigation, and the system engineer program. The order also includes a list of positional responsibilities associated with

these requirements. Finally, the order establishes contractor requirements in a contractor requirements document.

DOE G 421.1-1, DOE Good Practices Guide: Criticality Safety Good Practices Program Guide for DOE Nonreactor Nuclear Facilities

This guide illustrates and suggests practices and procedures for conducting a nuclear criticality safety (NCS) program at DOE nonreactor nuclear facilities having significant quantities of fissionable materials. These DOE practices and procedures are relevant to NCS program administration and oversight, NCS personnel selection and training, performance of NCS evaluations and analyses, emergency response, and programmatic control of processes, storage, procedures, hardware, and software. Throughout the text of this document, the term "fissionable material," used for concision, refers to the term that is specifically relevant to criticality safety concern, "significant quantity of fissionable material."

DOE G 421.1-2, Implementation Guide for Use in Developing Documented Safety Analyses to Meet Subpart B of 10 CFR 830

This guide elaborates on the documented safety analysis (DSA) development process and the safe harbor provisions of the appendix to 10 CFR 830, subpart B. Title 10 CFR 830, subpart B, Safety Basis Requirements, requires the contractor responsible for a DOE nuclear facility to analyze the facility, the work to be performed, and the associated hazards, and to identify the conditions, safe boundaries, and hazard controls necessary to protect workers, the public, and the environment from adverse consequences. Reviews provide an independent confirmation of readiness to start or restart operations.

Title 10 CFR 830, subpart B, "Safety Basis Requirements," requires the contractor responsible for a DOE nuclear facility to analyze the facility. These analyses and hazard controls constitute the safety basis upon which the contractor and DOE rely to conclude that the facility can be operated safely. Performing work consistent with the safety basis provides reasonable assurance of adequate protection of workers, the public, and the environment.

DOE G 423.1-1, Implementation Guide for Use in Developing Technical Safety Requirements

This guide provides elaboration on the content of TSRs. Section 10 CFR 830.205 of the nuclear safety management rule requires DOE contractors responsible for category 1, 2, and 3 DOE nuclear facilities to develop TSRs. These TSRs identify the limitations to each DOE-owned, contractor-operated nuclear facility based on the DSA and any additional safety requirements established for the facility.

The TSR rule requires contractors to prepare and submit TSRs for DOE approval. This guide provides guidance in identifying important safety parameters and developing the content for the TSRs that are required by 10 CFR 830.205.

The appendix to subpart B of the nuclear safety management rule specifies the types of safety limits, operating limits, surveillance requirements, and administrative controls that define the safety envelope necessary to protect the health and safety of the public and workers. The TSR derivation chapter in the DSA is the key component that provides the basis for TSRs.

DOE G 424.1-1, Implementation Guide for Use in Addressing Unreviewed Safety Question Requirements

This guide provides information to assist in the implementation and interpretation of Title 10 CFR 830.203, "Unreviewed Safety Question Process," of the nuclear safety management rules for applicable nuclear facilities owned or operated by DOE, including the NNSA.

Section 830.203, "Unreviewed Safety Question Process," allows contractors to make physical and procedural changes and to conduct tests and experiments without prior DOE approval if the proposed change can be accommodated within the existing safety basis. The contractor must carefully evaluate any proposed change to ensure that it will not explicitly or implicitly affect the safety basis of the facility. The USQ process is primarily applicable to the DSA. Although the rule references only the DSA, the DSA must include conditions of approval in safety evaluation reports and facility-specific commitments made in compliance with DOE rules, Orders, or policies. Because application of the USQ process depends on facility-specific information, results of a USQ determination in one facility generally cannot be extrapolated to other facilities. DOE approval of the procedure to implement the USQ process is required by 10 CFR 830.203.

DOE O 430.1B, Real Property Asset Management

The objective of this directive is to establish a corporate, holistic, and performance-based approach to real property life-cycle asset management that links real property asset planning, programming, budgeting, and evaluation to program mission projections and performance outcomes. To accomplish the objective, this Order identifies requirements and establishes reporting mechanisms and responsibilities for real property asset management. This Order implements Department of Energy (DOE) P 580.1, *Management Policy for Planning, Programming, Budgeting, Operation, Maintenance and Disposal of Real Property,* dated 05-20-02.

DOE-STD-1073-2003, Configuration Management

The purpose of this standard is to define the objectives of a configuration management process for DOE nuclear facilities, including activities and operations, and to provide detailed examples and supplementary guidance on methods of achieving those objectives. Configuration management is a disciplined process that involves both management and technical direction to establish and document the design requirements and the physical configuration of the nuclear facility and to ensure that they remain consistent with each other and the documentation.

The size, complexity, and missions of DOE nuclear facilities vary widely, and configuration management processes may need to be structured to individual facilities, activities, and operations. It would generally be inappropriate to apply the same configuration management standards to widely different activities, for example, a reactor facility and a small, simple laboratory. The detailed examples and methodologies in this standard are provided to aid those developing their configuration management processes; however, they are provided for guidance only and may not be appropriate for application to all DOE nuclear activities. The individuals defining the configuration management process for a particular nuclear activity will need to apply judgment to determine if the examples and methods presented in this standard are appropriate for the activity.

Nevertheless, the basic objectives and general principles of configuration management are the same for all activities. The objectives of configuration management are

- to establish consistency among design requirements, physical configuration, and documentation (including analysis, drawings, and procedures) for the activity;
- to maintain this consistency throughout the life of the facility or activity, particularly as changes are being made.

DOE-STD-3009-94, Ch. 2, Preparation Guide for U.S. DOE Nonreactor Nuclear Facility Safety Analysis Reports

This standard describes a DSA preparation method that is acceptable to the DOE. It was developed to assist hazard category 2 and 3 facilities in preparing safety analysis reports (SARs) that will satisfy the requirements of 10 CFR 830. Hazard category 1 facilities are typically expected to be category A reactors for which extensive precedents for SARs already exist.

Guidance provided by this standard is generally applicable to any facility that is required to document its safety basis in accordance with 10 CFR 830, "Nuclear Safety Management." For new facilities in which conceptual design or construction activities are in progress, elements of this guidance may be more appropriately handled as an integral part of the overall design requirements. The methodology provided by this standard focuses more on characterizing facility safety with or without well-documented information than on the determination of facility design. Accordingly, contractors for facilities that are documenting conceptual designs for a preliminary DSA should apply the process and format of this standard to the extent it is judged to be of benefit.

Beyond conceptual design and construction, the methodology in this standard is applicable to the spectrum of missions expected to occur over the lifetime of a facility. As the phases of facility life change, suitable methodology is provided for use in updating an existing DSA and in developing a new DSA if the new mission is no longer adequately encompassed by the existing DSA. This integration of the DSA with changes in facility mission and associated updates should be controlled as part of an overall safety management plan.

This standard addresses the following tasks related to implementing the requirements of 10 CFR 830:

- It ensures consistent and appropriate treatment of all DSA requirements for the variety of DOE nonreactor nuclear facilities.
- It provides final facility hazard categorization and considers and incorporates the categorization into programmatic requirement measures to protect workers, the public, and the environment from hazardous and accident conditions. TSRs and safety-significant SSCs that are major contributors to worker safety and defense in depth, are identified in the hazard analysis.
- It designates safety-class SSCs and safety controls as a function of the evaluation guideline.
- It provides a consistent and measured treatment of this concept, including guidance on the minimum acceptable DSA content.

DOE-HDBK-3010-94, Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities

The purpose of this document is to provide a compendium and analysis of experimental data from which airborne release fractions and respirable fractions may be derived. Such values are needed to determine quantities of radioactive material driven airborne to estimate the scope of the potential release spectrum and potential downwind consequences from a given facility or activity. The information provided in this handbook aids in making such estimates.

The handbook discusses the following major topics:

- Source term formula: provides a computational formula for using the information gained from analysis.
- Applicability of data: distinguishes proper use of information
- Accident stresses: identifies the types of accident conditions for which this information is applicable.
- Handbook organization: explains the presentation of information and the use of examples.

The data in the handbook can be used in a variety of applications, such as safety and environmental analyses, and to provide information relevant to system and experiment design. However, the data and the analyses of the data contained therein need to be critically evaluated for applicability in each situation in which they are used, and represent only one source of information in a complete safety analysis or design process.

DOE-STD-3011-2002, Guidance for Preparation of Basis for Interim Operation Documents

This standard provides guidance for the development of Basis for Interim Operation (BIO) documents, which are an acceptable form of DSA under the provision of the 10 CFR 830 rule. In this regard, it supplements the guidance in DOE G 421.1-2, *Implementation Guide for Developing Documented Safety Analyses to Meet Subpart B of 10 CFR 830*.

DOE-STD-1027-92, Ch. 1, Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports

The purpose of this DOE standard is to establish guidance for the preparation and review of hazard categorization and accident analysis techniques, as required in DOE Order 5480.23, Nuclear Safety Analysis Reports.

This standard provides specific guidance on several of the requirements contained in DOE Order 5480.23, Nuclear Safety Analysis Reports. Section 1 establishes the threshold quantities of hazardous materials that, if exceeded, would mandate the development of a SAR under the Order. Section 2 discusses the SAR upgrade plan and schedule that must be submitted to each Secretarial Officer. Section 3 provides a uniform methodology for hazard categorization. Section 4 gives additional specific guidance on the use of the graded approach and accident/hazard analysis techniques for compliance with the Order.

b. Discuss the role of radiation protection personnel with respect to the listed requirements and guidance.

The following is taken from DOE P 441.1.

It is the policy of DOE to conduct its radiological operations in a manner that ensures the health and safety of all its employees, contractors, and the general public. In achieving this objective, the Department shall ensure that radiation exposures to its workers and the public and releases of radioactivity to the environment are maintained below regulatory limits and deliberate efforts are taken to further reduce exposures and releases as low as is reasonably achievable. The Department is fully committed to implementing a radiological control program of the highest quality that consistently reflects this policy. The means by which the Department shall comply with this policy are discussed below.

The Department shall establish and maintain a system of regulatory policy and guidance reflective of national and international radiation protection standards and recommendations. The Assistant Secretary for Environment, Safety, and Health has responsibility for promulgating and maintaining policies, standards, and guidance related to radiological protection. Departmental radiological protection requirements are, at a minimum, consistent with the presidentially approved Radiation Protection Guidance to the Federal Agencies for Occupational Exposure, which was developed by the Environmental Protection Agency in accordance with its mandated Federal guidance responsibilities. Departmental requirements often are more stringent and reflect, as appropriate, recommendations and guidance from various national and international standards-setting and scientific organizations, including the International Commission on Radiological Protection, the National Council on Radiation Protection and Measurements, the American National Standards Institute, and others. Departmental requirements related to radiological protection will be set forth, as appropriate, in rules and DOE Orders, and guidance documents will be issued on acceptable means to implement these requirements.

The Department shall ensure personnel responsible for performing radiological work activities are appropriately trained. Standards shall be established to ensure the technical competency of the Department's work force, as appropriate, through implementation of radiological training and professional development programs.

The Department shall ensure the technical competence of personnel responsible for implementing and overseeing the radiological control program. An appropriate level of technical competence gained through education, experience, and job-related technical and professional training is a critical component for achieving the goals of the Department's radiological control policy. Qualification requirements commensurate with this objective shall be established for technical and professional radiological control program positions and shall, at a minimum, be consistent with applicable industry standards and promote professional development and excellence in radiological performance as a goal.

The Department shall establish and maintain, at all levels, line management involvement and accountability for departmental radiological performance. The responsibility for compliance with Departmental radiological protection requirements, and for minimizing personnel radiation exposure, starts at the worker level and broadens as it progresses upward through the line organization. The Department's line managers are fully responsible for radiological performance within their programs and the field activities and sites assigned to them, and shall take necessary actions to ensure requirements are implemented and performance is monitored and corrected as necessary.

The Department shall ensure radiological measurements, analyses, worker monitoring results, and estimates of public exposures are accurate and appropriately made. The capability to accurately measure and analyze radioactive materials and workplace conditions, and to determine personnel radiation exposure, is fundamental to the safe conduct of radiological operations. Policy, guidance, and quality control programs shall be directed towards ensuring such measurements are appropriate, accurate, and based upon sound technical practices.

The Department shall conduct radiological operations in a manner that controls the spread of radioactive materials and reduces exposure to the workforce and the general public, and that uses a process that seeks exposure levels as low as is reasonably achievable. Radiological operations and activities shall be preplanned to allow for the effective implementation of dose and contamination reduction and control measures. Operations and activities shall be performed in accordance with departmental conduct of operations requirements and shall include reasonable controls directed toward reducing exposure, preventing the spread of radiological contamination, and minimizing the generation of contaminated wastes and the release of effluents.

The Department shall incorporate dose reduction, contamination reduction, and waste minimization features into the design of new facilities and significant modifications to existing facilities in the earliest planning stages. Wherever possible, facility design features shall be directed toward controlling contamination at the source, eliminating airborne radioactivity, maintaining personnel exposure and effluent releases below regulatory limits, and using a process that seeks exposure levels and releases as low as is reasonably achievable. Radiological design criteria shall reflect appropriate consensus recommendations of national and international standards-setting groups.

The Department shall conduct oversight to ensure departmental requirements are being complied with and appropriate radiological work practices are being implemented.

All Departmental elements shall conduct their radiological operations in a manner consistent with the above policies and objectives.

c. Define the following safety management terms:

- **Authorization basis**
- **Design basis**
- **Safety limit**
- **Administrative controls**

Authorization Basis

The following is taken from DOE M 450.4-1.

Those aspects of the facility design basis and operational requirements relied upon by DOE to authorize operation. These aspects are considered important to the safety of facility operations. The authorization basis is described in documents such as the facility Safety Analysis Report and other safety analyses; Hazard Classification Documents, the Technical Safety Requirements, DOE-issued safety evaluation reports, and facility-specific commitments made in order to comply with DOE Orders or policies.

Design Basis

The following is taken from DOE-STD-3009-94.

The design basis is the set of requirements that bound the design of systems, structures, and components within the facility. These design requirements include consideration of safety, plant availability, efficiency, reliability, and maintainability. Some aspects of the design basis are important to safety, although others are not.

Safety Limit

The following is taken from 10 CFR 830.3.

Safety limits means the limits on process variables associated with those safety class physical barriers, generally passive, that are necessary for the intended facility function and that are required to guard against the uncontrolled release of radioactive materials.

Administrative Controls

Administrative controls are the provisions relating to organization and management, procedures, recordkeeping, assessment, and reporting necessary to ensure the safe operation of a facility.

d. Define the following terms associated with nuclear criticality safety:

- **Criticality incident**
- **Double contingency principle**
- **Geometry control**
- **Nuclear criticality safety**
- **Significant quantity of fissionable material**

The following is taken from DOE G 421.1-1.

Criticality Incident

A criticality incident is defined as a change in process condition or a loss of control beyond the evaluated process variances of the nuclear criticality safety analysis.

Double Contingency Principle

The double contingency principle requires that process designs shall incorporate sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible. In all cases, no single failure shall result in the potential for a criticality accident.

Geometry Control

Geometry control refers to the passive engineering control by which container equipment design limits fissionable material dimensions and the spacing between adjacent equipment. Reliance on geometry control shall be the first priority.

Nuclear Criticality Safety

Nuclear criticality safety is the prevention or termination of inadvertent nuclear criticality and protection against injury or damage due to an accidental nuclear criticality. Generally, prevention is preferred.

Significant Quantity of Fissionable Material

Significant quantity of fissionable material refers to the minimum mass of fissionable material for which control of at least one parameter is required to ensure subcriticality under all normal and credible abnormal conditions.

e. Describe the responsibilities of Operating and Management (O&M) and Management and Integration (M&I) contractors for the development and maintenance of a Documented Safety Analysis (DSA).

The following is taken from DOE G 421.1-2.

Development of a DSA or preliminary documented safety analysis (PDSA) is the process whereby facility hazards are identified, controls to prevent and mitigate potential accidents involving those hazards are proposed, and commitments are made for design, construction, operation, and disposition so as to assure adequate safety at DOE nuclear facilities. DOE, in its review and approval role, may require modification or addition to these commitments by the responsible contractor. Throughout the life of the facility, from design and construction to mission-oriented operations, through deactivation, long-term surveillance and maintenance, to decontamination and decommissioning, there must be a safety basis in place that is appropriate to the activities (operations) occurring during each of those phases.

During design and construction, the governing safety basis document is the PDSA. It is updated as the design matures and is approved prior to procurement and construction activities. Until approval, the PDSA and its updates serve to keep DOE informed as to how DOE nuclear safety design criteria are being addressed in the design. Project design reviews provide the vehicle by which safety-related changes are reviewed, and DOE can provide guidance to the contractor.

Prior to operations, the PDSA evolves to a final DSA that reflects the facility as actually constructed. During mission-oriented operations, and for each phase thereafter through deactivation or decontamination and decommissioning, or until the facility falls below the Category 3 threshold for nuclear facilities, the DSA must be kept current, considering any changes to the facility or its operations. The USQ process is key to this requirement. The USQ process must be integrated with the configuration management process that must be a part of the safety management program commitments of a DSA. The USQ process is the tool by which it is determined when DOE must approve any changes to the facility or its operations.

A DSA must demonstrate the extent to which a nuclear facility can be operated safely with respect to workers, the public, and the environment. DOE expects a contractor to use a graded approach to develop a DSA and describe how the graded approach was applied. The level of detail, analysis, and documentation will reflect the complexity and hazard associated with a particular facility or activity. Thus, the DSA for a simple, low-hazard facility may be relatively short and qualitative in nature, while the DSA for a complex, high-hazard facility may be quite elaborate and more quantitative. DOE will work with its contractors to ensure a DSA is appropriate for the facility or activity for which it is being developed. DSAs are prepared in order to be the primary reference on facility safety. Contractor management uses the DSAs for new nuclear facilities to have an authoritative documented record of DSA-

derived and programmatic safety commitments made to DOE governing safety and health aspects of project management, engineering, design, procurement and construction of the facility or the development of the nuclear operation. DOE O 420.1, Facility Safety, contains requirements for the design of new nuclear facilities and mandates the use of safety analyses to guide safety aspects of design. In addition, 10 CFR 835, subpart K, Occupational Radiation Protection, gives regulatory requirements for design and control. These analyses should be summarized in the DSA to support the rationale for safety aspects of design.

f. Discuss the development and maintenance of site/facility safety management documents and procedures for modifications.

The following is taken from DOE G 421.1-2.

DOE does not expect a PDSA to be required for activities that do not involve significant construction, such as environmental restoration activities, decontamination and decommissioning activities, specific nuclear explosive operations, transition surveillance and maintenance activities, or for activities that are not major modifications. For activities that are not major modifications, the USQ process should be used to determine if DOE approval is needed. If so, a safety analysis that supports the request for approval should be developed. If the request is approved, then the safety analysis should be included in the DSA when the modification is completed.

The PDSA required by 10 CFR 830.206 may need updating to sustain the reliability of the information therein, until such time as it is superseded by a Final DSA. The contractor should update the PDSA as necessary to keep it applicable to the evolving design so that the Department can continue to rely on the information in the PDSA.

15. Radiation protection personnel shall demonstrate a familiarity level knowledge of Federal regulations and DOE Orders related to emergency planning and preparedness as they pertain to radiological incidents.

a. Describe the relevant requirements, purpose, interrelationships, and importance of the following Orders and regulation:
- **29 CFR 1910.120, "Hazardous Waste Operations and Emergency Response"**
- **DOE O 151.1A, Comprehensive Emergency Management and Series of Guides (G 151.1-1)**
- **DOE Order 5530.3, Ch. 1, Radiological Assistance Program**
- **DOE Order 5530.5, Ch. 1, Federal Radiological Monitoring and Assessment Center**

29 CFR 1910.120, Hazardous Waste Operations and Emergency Response

All suspected conditions that may pose inhalation or skin absorption hazards that are immediately dangerous to life or health (IDLH), or other conditions that may cause death or serious harm, shall be identified during a preliminary survey and evaluated during a detailed survey. Examples of such hazards include, but are not limited to, confined space entry, potentially explosive or flammable situations, visible vapor clouds, or areas where biological indicators such as dead animals or vegetation are located.

The following monitoring shall be conducted during initial site entry when the site evaluation produces information that shows the potential for ionizing radiation or IDLH conditions, or when the site information is not reasonably sufficient to eliminate these possible conditions:

- Monitoring with direct-reading instruments for hazardous levels of ionizing radiation;
- Monitoring the air with appropriate direct-reading test equipment (i.e., combustible gas meters, detector tubes) for IDLH and other conditions that may cause death or serious harm (combustible or explosive atmospheres, oxygen deficiency, toxic substances); and
- Visually observing for signs of actual or potential IDLH or other dangerous conditions.
- An ongoing air-monitoring program shall be implemented after site characterization has determined the site is safe for the start-up of operations.

Once the presence and concentrations of specific hazardous substances and health hazards have been established, the risks associated with these substances shall be identified. Employees who will be working on the site shall be informed of any risks that have been identified.

DOE O 151.1A, Comprehensive Emergency Management and Series of Guides (G 151.1-1)

[Note: DOE O 151.1A has been superseded by DOE O 151.1C, from which the following is taken.]

The objectives of DOE O 151.1-1C are as follows:

- To establish policy and to assign and describe roles and responsibilities for the Department of Energy (DOE) Emergency Management System. The Emergency Management System provides the framework for development, coordination, control, and direction of all emergency planning, preparedness, readiness assurance, response, and recovery actions. The Emergency Management System applies to DOE and to the National Nuclear Security Administration (NNSA).
- To establish requirements for comprehensive planning, preparedness, response, and recovery activities of emergency management programs or for organizations requiring DOE/NNSA assistance.
- To describe an approach to effectively integrate planning, preparedness, response, and recovery activities for a comprehensive, all-emergency management concept.
- To integrate public information and emergency planning to provide accurate, candid, and timely information to site workers and the public during all emergencies.
- To promote more efficient use of resources through greater flexibility (i.e., the graded approach) in addressing emergency management needs consistent with the changing missions of the Department and its facilities.
- To ensure that the DOE Emergency Management System is ready to respond promptly, efficiently, and effectively to any emergency involving DOE/NNSA facilities, activities, or operations, or requiring DOE/NNSA assistance.
- To integrate applicable policies and requirements, including those promulgated by other Federal agencies (e.g., stockpiling stable iodine for possible distribution as a radiological protective prophylaxis) and interagency emergency plans into the Department's Emergency Management System. In compliance with the statutory requirements in 42 USC 7274k, DOE hereby finds that this Order is necessary for the

fulfillment of current legal requirements and conduct of critical administrative functions.

- To eliminate duplication of emergency management effort within the Department.

DOE G 151.1-1 Series of Guides

[Note: DOE G 151.1-1 has been superseded by five guides, discussed below.]

DOE G 151.1-1A, Emergency Management Fundamentals and the Operational Emergency Base Program

This guide provides information about the emergency management fundamentals imbedded in the requirements of DOE O 151.1C, as well as acceptable methods of meeting the requirements for the Operational Emergency Base Program, which ensures that all DOE facilities have effective capabilities for all emergency response.

DOE G 151.1-2, Technical Planning Basis

This guide provides information about the emergency management fundamentals imbedded in the requirements of DOE O 151.1C, as well as acceptable methods of meeting the requirements for the Operational Emergency Base Program, which ensures that all DOE facilities have effective capabilities for all emergency response.

DOE G 151.1-3, Programmatic Elements

This guide provides acceptable methods of meeting the requirements of DOE O 151.1C for programmatic elements that sustain the emergency management program and maintain the readiness of the program to respond to an emergency.

DOE G 151.1-4, Response Elements

This guide provides acceptable methods for meeting the requirement of DOE O 151.1C for response elements that respond or contribute to response as needed in an emergency.

DOE G 151.1-5, Biosafety Facilities

This guide assists DOE/NNSA field elements and operating contractors in incorporating hazardous biological agents/toxins into emergency management programs, as required by DOE O 151.1C.

DOE Order 5530.3, Ch. 1, Radiological Assistance Program

DOE Order 5530.5, Ch. 1, Federal Radiological Monitoring and Assessment Center

[Note: DOE Orders 5530.3 and 5530.5 have been canceled.]

b. Describe what is meant by an operational emergency.

The following is taken from DOE O 151.1C.

In general, to be considered an operational emergency, an event or condition involving the uncontrolled release of a hazardous material must: immediately threaten or endanger personnel who are in close proximity of the event; have the potential for dispersal beyond the immediate vicinity of the release in quantities that threaten the health and safety of onsite personnel or the public in collocated facilities, activities, and/or offsite; and have a potential

rate of dispersal sufficient to require a time-urgent response to implement protective actions for workers and the public.

c. Describe how the following guides are used:
- **Protective action guide (PAG)**
- **Emergency response planning guide (ERPG)**

The following is taken from U.S. EPA, Protective Action Guides
http://www.epa.gov/radiation/rert/pags.html

Protective Action Guide

Protective action guides (PAGs) help state and local authorities make radiation protection decisions during emergencies. EPA developed the PAG Manual to provide guidance on actions to protect the public.

Emergency Response Planning Guide

The following is taken in part from the American Industrial Hygiene Association,

2006 Emergency Response Planning Guidelines (ERPG) and Workplace Environmental Exposure Levels (WEEL) Handbook.

ERPG values are intended to provide estimates of concentration ranges where one reasonably might anticipate observing adverse effects as described in the definitions for ERPG-1, ERPG-2, and ERPG-3 as a consequence of exposure to the specific substance.

ERPG-1 is the maximum airborne concentration below which it is believed that nearly all individuals could be exposed for up to one hour without experiencing other than mild transient adverse health effects or perceiving a clearly defined, objectionable odor.

ERPG-2 is the maximum airborne concentration below which it is believed that nearly all individuals could be exposed for up to one hour without experiencing or developing irreversible or other serious health effects or symptoms which could impair an individual's ability to take protective action.

ERPG-3 is the maximum airborne concentration below which it is believed that nearly all individuals could be exposed for up to one hour without experiencing or developing life-threatening health effects.

d. Discuss the conditions that would require an operational emergency to be classified as an:
- **Alert**
- **Site area emergency**
- **General emergency**

The following is taken from DOE O 151.1C.

Alert

An alert must be declared when events are predicted, are in progress, or have occurred that result in one or more of the following situations:

- An actual or potential substantial degradation in the level of control over hazardous materials.
- The radiation dose from any release to the environment of radioactive material or a concentration in air of other hazardous material is expected to exceed one of the following:
  - the applicable PAG or ERPG at or beyond 30 meters from the point of release to the environment.
  - a site-specific criterion corresponding to a small fraction of the applicable PAG or ERPG at or beyond the facility boundary or exclusion zone boundary.

- It is not expected that the applicable PAG or ERPG will be exceeded at or beyond the facility boundary or exclusion zone boundary.
- An actual or potential substantial degradation in the level of safety or security of a nuclear weapon, component, or test device that would not pose an immediate threat to workers or the public.
- An actual or potential substantial degradation in the level of safety or security of a facility or process that could, with further degradation, produce a site area emergency or general emergency.

Site Area Emergency

A site area emergency must be declared when events are predicted, in progress, or have occurred that result in one or more of the following situations:

- An actual or potential major failure of functions necessary for the protection of workers or the public. The radiation dose from any release of radioactive material or concentration in air from any release of other hazardous material is expected to exceed the applicable PAG or ERPG beyond the facility boundary or exclusion zone boundary. The PAG or ERPG is not expected to be exceeded at or beyond the site boundary.
- An actual or potential threat to the integrity of a nuclear weapon, component, or test device that may adversely impact the health and safety of workers in the immediate area, but not the public.
- Actual or potential major degradation in the level of safety or security of a facility or process that could, with further degradation, produce a general emergency.

General Emergency

A general emergency must be declared when events are predicted, in progress, or have occurred that result in one or more of the following situations:

- Actual or imminent catastrophic reduction of facility safety or security systems with potential for the release of large quantities of hazardous materials (radiological or non-radiological) to the environment. The radiation dose from any release of radioactive material or a concentration in air from any release of other hazardous material is expected to exceed the applicable PAG or ERPG at or beyond the site boundary.
- Actual or likely catastrophic failures in safety or security systems threatening the integrity of a nuclear weapon, component, or test device that may adversely impact the health and safety of workers and the public.

e. Discuss the role of radiation protection personnel with respect to the Orders and regulations in supporting knowledge and/or skill.

This varies from site to site. The Qualifying Official will evaluate the completion of this activity.

f. Discuss the emergency response assistance that is available from the following:
- **Nuclear emergency response team**
- **Accident response group**
- **Aerial measuring system**
- **Atmospheric release advisory capability**
- **Federal radiological monitoring and assessment center**
- **Radiation emergency assistance center/training site**
- **Radiological assistance program**

Nuclear Emergency Response Team

The following is taken from the U.S. EPA, Emergency Preparedness and Response

EPA prepares for and responds to emergencies involving radioactive materials. In the event of a radiological emergency, EPA's Radiological Emergency Response Team (RERT) works with other Federal agencies, state and local governments, and international organizations to monitor, contain, and clean up the release while protecting people and the environment from harmful exposure to radiation.

Accident Response Group

The following is taken from DOE O 153.1.

The accident response group is a deployable asset that
- manages technical resolution of accidents or significant incidents involving United States nuclear weapons that are in DOE custody at the time of an accident or incident;
- provides timely worldwide support to Department of Defense (DoD) in the technical resolution of accidents and significant incidents involving United States nuclear weapons in DoD custody;
- performs risk assessments to determine safe-to-ship status of damaged nuclear weapons before recommending transport to the designated disposition site; and
- accepts custody and control of nuclear weapons or weapons components on behalf of DOE.

Aerial Measuring System

The aerial measuring system is a deployable asset that measures and evaluates the radiological information necessary to address the relevant radiological impacts of accidents and radiological national security emergencies. It detects and measures radioactivity over large areas using both fixed- and rotary-wing aircraft.

Atmospheric Release Advisory Capability

The atmospheric release advisory capability (ARAC) is a laboratory-based program that provides near real-time computer-based predictive modeling to assess events involving release of hazardous radiological materials into the atmosphere. Predictions are produced for

local and Federal leaders to determine the protective actions necessary to ensure the health and safety of people in affected areas. The ARAC program includes:

- the National Atmospheric Release Advisory Center (NARAC) at Lawrence Livermore National Laboratory (LLNL), which has a staff of physical scientists, computer scientists, engineers, and technicians that provide services 24/7 for planning, emergency response, and detailed hazard studies;
- the High Consequence Assessment and Technology group from Sandia National Laboratories, which provides additional NARAC tools and expertise for nuclear and conventional explosive source characteristics/effects and dose/risk assessment;
- a suite of NARAC software tools including simple standalone, local-scale plume modeling tools for end-users' computers, and Web- and Internet-based software to access advanced three-dimensional modeling tools and expert analyses from the national center at LLNL;
- the Interagency Modeling Atmospheric and Assessment Center, which supports Federal responses to incidents of national significance involving hazardous material releases to the atmosphere; and
- the computer-based system at NARAC, which provides realistic plots or maps of potential radiation dose and exposure, and estimates of the paths of nuclear contaminants released into the atmosphere.

Federal Radiological Monitoring and Assessment Center (FRMAC)

FRMAC is a Federal interagency center responsible for coordinating monitoring and assessment activities with the affected state and local agencies. The Nuclear/Radiological Annex to the NRP assigns DOE responsibility for

- developing and maintaining FRMAC policies and procedures;
- determining FRMAC composition, and
- maintaining FRMAC operational readiness
- managing the FRMAC during initial phases of a radiological incident with transfer of the leadership role to the EPA at a mutually agreeable time. Consequence management teams provide command and control, planning, and technical elements needed to accomplish assigned FRMAC responsibilities.

All other departmental radiological emergency response assets will be prepared to contribute to the FRMAC through the consequence management teams.

Radiation Emergency Assistance Center/Training Site

REAC/TS is a medical consulting and/or deployable, tailored, program that provides a 24-hour response center for medical advice, specialized training, and unique on-site assistance in triage, diagnosis, and treatment of all types of radiation-induced injuries.

Radiological Assistance Program

The Radiological Assistance Program is a deployable, tailored capability that provides first-responder radiological assistance to protect the health and safety of the public and the environment. Upon request, RAP provides radiological assistance to other Federal agencies, state, tribal, and local governments, and private businesses or individuals in the detection, identification and analysis, and response to events/incidents involving radiological/nuclear materials.

16. Radiation protection personnel shall demonstrate a familiarity level knowledge of DOE Orders related to Federal and contractor personnel training and qualification.

a. Describe in general the training and qualification requirements for contractors specified in DOE Order 5480.20A, Ch. 1, Personnel Selection, Qualification, and Training Requirements for DOE Nuclear Facilities.

The purpose of DOE Order 5480.20A is to establish selection, qualification, and training requirements for management and operating (M&O) contractor personnel involved in the operation, maintenance, and technical support of DOE and NNSA category A and B reactors and non-reactor nuclear facilities.

The operating contractor shall establish one or more organizations to be responsible for the training of operating organization personnel. This organization shall be held accountable for providing facility line management with the support necessary to ensure that personnel in the operating organization are qualified to safely and effectively meet job requirements. In some cases, this function may be integrated into the operating organization and may not necessarily be officially designated as a training organization. The responsibilities, qualifications, and authority of training organization personnel shall be documented, and managerial responsibilities and authority clearly defined. This organization may include subcontracted personnel who conduct training activities. At sites where a central training organization is used, this organization may be separate from the facility operating organization for support in areas of regulatory training. For example, central training organizations that provide support to line operating organizations may conduct training for the operating organization in regulatory compliance issues (e.g., OSHA training, Radiation Worker training, supervisory/ management training, etc.) that have site-wide application, and which have content that is defined from other sources.

b. Describe the Technical Qualification Program for Federal employees delineated in DOE Order 360.1B, Training.

[Note: DOE O 360.1B does not contain a description of the Technical Qualification Program. DOE M 360.1-1B, *Federal Employee Training Manual* contains the TQP Program Components. Additional information on this program may be found in DOE M 426.1-1A, *Federal Technical Capability Manual*.]

The following is taken from DOE M 360.1-1B.

Technical Qualification Program Components

Plans and procedures.

Each DOE element required to establish a Technical Qualification Program must develop and implement a Technical Qualification Program Plan as a separate component of its training plan. Technical Qualification Program Plans must be approved by the head of the element and include the process and requirements for the following:

- identifying personnel and positions required to participate in the Technical Qualification Program;
- identifying and maintaining technical qualification standards or individual qualification requirements, as appropriate;

- evaluating employees against qualification standards and documenting the approval of equivalencies for DOE-wide competencies;
- establishing and updating individual development plans, training plans, qualification cards, or related records to document learning activities;
- implementing continuing training and requalification programs; and
- applying evaluation requirements for completing the technical qualification standard (e.g., written or oral examinations).

Qualification Requirements for the Technical Qualification Program.

Qualification requirements must be documented by each DOE element in qualification standards or other appropriate means. These requirements must be established using the systematic approach to training methodology and include the necessary basic technical knowledge; technical discipline competency requirements; and position-specific knowledge, skills, and abilities.

Documentation of qualification requirements must contain the following:

- general duties and responsibilities associated with the position;
- background and experience in terms of preferred education and experience;
- a list of the technical competencies that define the level of expected performance, with supporting knowledge and/or skill statements provided as guidance to describe the intent of the competency statements; and
- continuing training and proficiency requirements.

The following is taken from DOE M 426.1-1A.

Training, education, and experience combine to provide a workforce that ensures safe operation of defense nuclear facilities. The TQP establishes a process to objectively determine that individuals performing activities related to the technical support, management, oversight, or operation of defense nuclear facilities possess the necessary knowledge, skills, and abilities to perform their assigned duties and responsibilities.

The TQP specifically applies to DOE technical employees whose duties and responsibilities require them to provide assistance, guidance, direction, oversight, or evaluation of contractor activities that could impact the safe operation of a defense nuclear facility. This includes employees designated as safety system oversight personnel, facility representatives, senior technical safety managers, and employees on detail or temporary assignment.

The objectives of the TQP are as follows:

- Identify and document the functional competencies that individual employees must possess to ensure that DOE defense nuclear facilities and programs are operated in accordance with applicable safety, health, and environmental requirements.
- Establish a program that clearly identifies and documents the process used by senior line management to demonstrate employee technical competence, consistent with applicable industry standards for similar occupations.
- Ensure that employees maintain their technical competencies.
- Maintain a cycle of continuous performance improvement through structured, individualized training and development programs and through review and assessment of HQ and field element programs.

c. Discuss the purpose, scope, and application of DOE-STD-1107-97, Knowledge, Skills, and Abilities for Key Radiation Protection Positions at DOE Facilities.

DOE-STD-1107-97 provides detailed qualification criteria for contractor key radiation protection personnel. Appendix B provides detailed listings for knowledge, skills, and abilities for contractor and DOE Federal key radiation protection positions. This information may be used in developing position descriptions and individual development plans. Information provided in appendix C may be useful in developing performance measures and assessing an individual's performance in his or her specific position. Additionally, Federal personnel may use this information to augment their office/facility qualification standards under the Technical Qualification Program.

17. Radiation protection personnel shall demonstrate a working level knowledge of national and international radiation protection standards and recommendations.

a. Discuss the content and application of the following national and international documents on radiation protection:

- **Radiation Protection Guidance to the Federal Agencies for Occupational Exposure (52 FR 2822)**
- **Recommendations of the ICRP, Publication 26**
- **Recommendations of the ICRP, Publication 30**
- **Recommendations of the ICRP, Publication 60**
- **Recommendations of the ICRP, Publication 66**
- **Recommendations of the ICRP, Publication 68**
- **Recommendations of the ICRP, Publication 71**
- **Recommendations of the ICRP, Publication 72**
- **BEIR V Executive Summary**
- **Recommendations on Limits for Exposure to Ionizing Radiation, NCRP, Report No. 91**
- **Limitation of Exposure to Ionizing Radiation, NCRP, Report No. 116**
- **Practices for Respiratory Protection, ANSI (ANSI Z88.2-1992)**

Radiation Protection Guidance to the Federal Agencies for Occupational Exposure (52 FR 2822)

This guidance provides general principles, and specifies the numerical primary guides for limiting worker exposure. It applies to all workers who are exposed to radiation in the course of their work, either as employees of institutions and companies subject to Federal regulation or as Federal employees.

Recommendations of the ICRP, Publication 26

ICRP Publication 26 was first published in 1977 and, since that time, the recommendations which it includes have been increasingly incorporated into national legislation. This new [1987] edition reproduces the complete text of the original publication, but supplements it with the statements issued by the Commission in 1978, 1980, 1983, 1984, and 1985. These statements clarify, and in some cases, modify the recommendations given in Publication 26.

In addition to its basic recommendations, the ICRP publishes a wide range of other reports relating to the basis of the recommendations and to their application in particular

circumstances. The first edition of Publication 26 contained a bibliography of other publications of the Commission up to 1977. This has now been updated to include all reports either published or in preparation for publication up to the end of 1986.

Recommendations of the ICRP, Publication 30

This recommendation is titled Limits for Intakes of Radionuclides by Workers. This part gives metabolic data for 30 elements, including ALIs for their isotopes. The data given in this report are intended to be used together with the text and dosimetric models described in ICRP Publication 30, Part 1.

This part contains statements and recommendations of the ICRP from its 1980 meeting. The elements addressed are: fluorine, sodium, sulphur, chlorine, argon, potassium, calcium, chromium, iron, copper, zinc, bromine, rubidium, yttrium, technetium, ruthenium, rhodium, silver, cadmium, indium, xenon, barium, rhenium, osmium, iridium, gold, mercury, lead, bismuth, and neptunium.

Recommendations of the ICRP, Publication 60

Since 1977, when the Commission issued its basic recommendations as *ICRP Publication 26,* it has reviewed these recommendations annually and, from time to time, has issued supplementary Statements in the *Annals of the ICRP*. A complete list of the Commission's publications is given in Annex D. Developments in the last few years have now made it necessary to issue a completely new set of recommendations. In doing so, the Commission has had three aims in mind:

- to take account of new biological information and of trends in the setting of safety standards;
- to improve the presentation of the recommendations, and
- to maintain as much stability in the recommendations as is consistent with the new information.

The draft of the recommendations was prepared by a Task Group set up by the 1985-1989 Commission. The draft was discussed and adopted by the 1989-93 Commission in November 1990.

The Commission intends this report to be of help to regulatory and advisory agencies at national, regional, and international levels, mainly by providing guidance on the fundamental principles on which appropriate radiological protection can be based. Because of the differing conditions that apply in various countries, the Commission does not intend to provide a regulatory text. Authorities will need to develop their own structures of legislation, regulation, authorizations, licenses, codes of practice, and guidance material in line with their usual practices and policies. The Commission believes that these regulatory structures should be designed to be broadly consistent with the guidance in this report. In addition, the Commission hopes that the report will be of help to management bodies with responsibilities for radiological protection in their own operations, to the professional staff whom they use as their advisers, and to individuals, such as radiologists, who have to make decisions about the use of ionizing radiation.

Recommendations of the ICRP, Publication 66

This recommendation is titled Human Respiratory Tract Model for Radiological Protection. The report describes a revision of the model used in ICRP Publication 30 to calculate radiation doses to the respiratory tract of workers resulting from the intake of airborne radionuclides. This revision was motivated by the availability of increased knowledge of the anatomy and physiology of the respiratory tract and of the deposition, clearance, and biological effects of inhaled radioactive particles, and by greatly expanded dosimetry requirements. To meet fully the needs of radiation protection, a dosimetric model for the respiratory tract should:

- provide calculations of doses for individual members of the populations of all ethnic groups, in addition to workers;
- be useful for predictive and assessment purposes as well as for deriving limits on intakes;
- account for the influence of smoking, air pollutants, and respiratory tract diseases;
- provide for estimates of respiratory tract tissue doses from bioassay data; and
- be equally applicable to radioactive gases as well as to particles.

Additionally, large differences in radiation sensitivity among the respiratory tract tissues and the doses they receive from inhaled radionuclides argue for calculating radiation doses to specific tissues of the respiratory tract. To use these tissue doses effectively for radiation protection purposes, they must be compatible with the ICRP dosimetric system.

Addressing all of these requirements has resulted in a dosimetry model that is more complex than previous models. This complexity is reflected in the structure of this report, which includes chapters on respiratory tract morphometry, physiology, and radiation biology or health detriment. Other chapters deal with deposition and clearance of inhaled radioactive particles, inhalation of radioactive gases, and application of the model to estimate respiratory tract doses. The model provides most of the flexibility needed to calculate doses to the respiratory tract for a wide range of exposure conditions and for specific individuals. This flexibility also allows for revision of reference parameter values as new information becomes available without changing the model. The wide availability of personal computers allows for easy use of the model.

Recommendations of the ICRP, Publication 68

This recommendation is titled Dose Coefficients for Intakes of Radionuclides by Workers. The Commission's 1990 recommendations on radiation protection standards in ICRP Publication 60 were developed to take into account new biological information related to the detriment associated with radiation exposures, and to supersede the earlier recommendations in ICRP Publication 26. To permit immediate application of these new recommendations, revised values of the ALIs, based on the methodology and biokinetic information and incorporating the new dose limits and tissue weighting factors (w_T), were issued as ICRP Publication 61. Since issuing ICRP Publication 61, ICRP has published a revised kinetic and dosimetric model of the respiratory tract. The main aim of the present report is to give values of dose coefficients for workers using this new model.

Recommendations of the ICRP, Publication 71

Since issuing *ICRP Publication 56,* ICRP has issued in *ICRP Publication 66* a revised kinetic and dosimetric model of the respiratory tract, which is also age-specific. The main aim of the present report [Publication 71] is to give values of inhalation dose coefficients for members of the public using this new model, for selected radioisotopes of the 29 elements covered in *ICRP Publications 56, 67* and 69, and of calcium and curium, for which biokinetic models are given here.

Recommendations of the ICRP, Publication 72

This report is a compilation of age-dependent committed effective dose coefficients for members of the public from intakes by ingestion and inhalation of radioisotopes of the 31 elements covered by *ICRP Publications 56, 67, 79, and 71.* This report also gives committed effective dose coefficients for members of the public from radioisotopes of the additional 60 elements for which dose coefficients are given for workers in *ICRP Publication 68.* This report does not give committed equivalent doses to tissues and organs. In a few instances there are small changes in ingestion dose coefficients from those given in previous reports. In these cases, the values given here supersede the earlier values. These ICRP dose coefficients have been adopted in the International Basic Safety Standards and in the Euratom Directive. Values of dose coefficients for workers are also given in the International Basic Safety Standards and in the Euratom Directive. With the exception of inhalation dose coefficients for ^{226}Ra, these values are the same as those given in *ICRP Publication 68.*

BEIR V Executive Summary

This report, prepared by the National Research Council's Committee on the Biological Effects of Ionizing Radiations (BEIR), is the fifth in a series that addresses the health effects of exposure of human populations to low-dose radiation. Ionizing radiations arise from both natural and man-made sources and can affect the various organs and tissues of the body. Late health effects depend on the physical characteristics of the radiation as well as biological factors.

Well demonstrated late effects include the induction of cancer, genetically determined ill-health, developmental abnormalities, and some degenerative diseases (e.g., cataracts). Recent concern has centered on the risks of these effects following low-dose exposure, in part because of the presence of elevated levels of radon progeny at certain geographical sites and fallout from the nuclear reactor accidents at Three Mile Island in Pennsylvania in 1979 and Chernobyl in the USSR in 1986. In addition, there is concern about radioactivity in the environment around nuclear facilities and a need to set standards for cleanup and disposal of nuclear waste materials.

Recommendations on Limits for Exposure to Ionizing Radiation, NCRP, Report No. 91.

The National Council on Radiation Protection and Measurements (NCRP) has the responsibility under its charter from the Congress of the United States to:

- Collect, analyze, develop, and disseminate in the public interest information and recommendations about (a) protection against radiation and (b) radiation measurements, quantities, and units, particularly those concerned with radiation protection;

- Provide a means by which organizations concerned with the scientific and related aspects of radiation protection and of radiation quantities, units, and measurements may cooperate for effective utilization of their combined resources, and to stimulate the work of such organizations;
- Develop basic concepts about radiation quantities, units, and measurements, about the application of these concepts, and about radiation protection;
- Cooperate with the International Commission on Radiological Protection the Federal Radiation Council, the International Commission on Radiation Units and Measurements, and other national and international organizations, governmental and private, concerned with radiation quantities, units, and measurements with radiation protection.

Limits for exposure to ionizing radiation constitute a central element of the recommendations specified in the charter. The last general recommendations of the NCRP on such limits were issued in 1971. At that time, it was not possible to utilize estimates of the risk attributable to radiation exposure as a basis for the limits specified. Subsequent development of our knowledge of radiation risks has made it possible to base radiation exposure limits on risk estimates, and this Report uses that approach.

New recommendations are provided both for occupational and public exposure. The recommendations given are believed to be in concert with current concepts and understand of radiation risks and their magnitude. However, the NCRP keeps abreast of developments in the field of radiation protection and is prepared to make modifications as knowledge of radiation effects improves.

The International Systéme of Units (SI) is used in this Report followed by the conventional units in parentheses, in accordance with the procedure set forth in NCRP Report No. 82, *SI Units in Radiation Protection and Measurements.*

Limitation of Exposure to Ionizing Radiation, National Council on Radiation Protection, Report No. 116

This report updates and replaces National Council on Radiation Protection and Measurements (NCRP) Report No. 91, *Recommendations on Limits for Exposure to Ionizing Radiation.* Although the recommendations contained in this Report are similar to those in NCRP Report No. 91, the Council desires to reiterate and update its position on radiation protection issues following the publication of additional data on the biological effects of ionizing radiation by the National Academy of Sciences/National Research Council Committee on the Biological Effects of Ionizing Radiation (BEIR V), the United National Scientific Committee on the Effects of Atomic Radiation, and the review of these documents by Scientific Committee 1-2 of the NCRP that is being published as NCRP Report No. 115, *Risk Estimates for Radiation Protection Purposes* and the publication of the *1990 Recommendations of the International Commission on Radiation Protection.* Deviation in the recommendations of this Report from those of the ICRP reflect the Council's desire to incorporate greater flexibility or increased protection in its recommendations for those situations where it is reasonable to do so.

Practices for Respiratory Protection, American National Standards Institute (ANSI Z88.2-1992)

Scope

This standard sets forth accepted practices for respirator users; provides information and guidance on the proper selection, use, and care of respirators, and contains requirements for establishing and regulating respirator programs. The standard covers the use of respirators to protect persons against the inhalation of harmful air contaminants and against oxygen-deficient atmospheres in the workplace. The following subjects are not covered by this standard:

- Underwater breathing devices;
- Aircraft oxygen systems;
- Use of respirators under military combat conditions; and
- Medical inhalators and resuscitators.

Purpose

The purpose of this standard is to provide information and guidance on the proper selection and use of respirators that will help safeguard the life and health of users. This standard is written for all persons concerned with respiratory protection, but especially for those primarily responsible for establishing and administrating an acceptable respirator program. The standard contains requirements recommended for use by enforcement authorities in establishing regulations or codes on respiratory protection.

b. Discuss how the listed documents relate to DOE radiation protection requirements.

This varies from site to site. The Qualifying Official will evaluate the completion of this activity.

18. Radiation protection personnel shall demonstrate a familiarity level knowledge of the Federal regulations, guidelines, and DOE Orders pertaining to the decontamination and decommissioning of nuclear facilities.

a. Familiarity with the DOE policy and requirements regarding the management, control, and release of property containing residual radioactivity, as contained in:

- **DOE/EH-413-0002, Facility Disposition: Principles for Accelerated Project Management**
- **DOE Order 5400.5, Ch. 2, Radiation Protection of the Public and the Environment**
- **DOE-STD-1120-98, Integration of Environment, Safety, and Health into Facility Disposition Activities**

DOE/EH-413-0002, Facility Disposition: Principles for Accelerated Project Management

This technical support bulletin is intended primarily for project managers responsible for planning, organizing, and implementing facility disposition (deactivation and decommissioning) activities at sites throughout the DOE complex. Its goal is to assist in developing more efficient project plans, while concurrently abiding by all requirements and guidance outlined in: 1) DOE O 430.1B, *Real Property Asset Management*, and its associated

guides; 2) appropriate elements of CERCLA; and 3) DOE-STD-1120-2005, *Integration of Environment, Safety, and Health into Facility Disposition Activities.*

These facility disposition framework documents explicitly include DOE's Integrated Safety Management (ISM) Policy (DOE P 450.4), which provides the overall structure for conducting projects safely, including those related to facility disposition.

With the publication of these three documents, DOE has outlined a general approach to ensure project managers conduct adequate planning and have appropriate mechanisms in place to protect workers, meet environmental requirements, and achieve project objectives. To implement the elements of this approach as practically and cost-effectively as possible, this bulletin presents a detailed planning approach that is based on meeting the same challenges for very similar environmental restoration projects. Specifically, this bulletin introduces four underlying principles for project managers to apply to facility disposition planning. These principles are the distilled lessons learned from practical field experience implementing facility disposition and environmental restoration projects. Together with the core activities of ISM, the Orders, and guidance, these principles form a complete framework that will foster better communication and enhanced project planning and implementation.

Facility disposition encompasses three major, often interrelated phases of work: surveillance and maintenance, deactivation, and decommissioning.

Surveillance and Maintenance (S&M).

Surveillance includes any activity that involves the scheduled periodic inspection of a facility, equipment, or structure as required by Federal and state environmental, safety, and health laws and regulations, and DOE Orders. The purpose of surveillance is to demonstrate compliance, identify problems requiring corrective action, and determine the facility's present environmental, radiological, and physical condition. More specifically, surveillance includes activities performed to determine the operability of critical equipment, monitor radiological conditions, check safety-related items, provide for facility-security controls, and assess facility structural integrity.

Maintenance includes any activity that is required to sustain property in a condition suitable for the property to be used for its designated purpose, including preventative, predictive, and corrective maintenance.

Deactivation

Deactivation is the process of placing a facility in a stable condition for the purposes of minimizing existing risks and the life-cycle cost of an S&M program that is still protective of workers, the public, and the environment.

Decommissioning

Decommissioning generally takes place after deactivation, and includes surveillance and maintenance, decontamination, and/or dismantlement. These actions are taken at the end of the life of a facility to retire it from service with adequate regard for the health and safety of workers and the public and protection of the environment. The ultimate goal of decommissioning is generally unrestricted release or restricted use of the facility.

Additional information regarding DOE/EH-413-0002 is available at http://homer.ornl.gov/oepa/guidance/cercla/facilitydisposition.pdf.

DOE Order 5400.5, Ch. 2, Radiation Protection of the Public and the Environment

The purpose of this Order is to establish standards and requirements for operations of the DOE and DOE contractors with respect to protection of members of the public and the environment against undue risk from radiation.

It is DOE's objective to operate its facilities and conduct its activities so that radiation exposures to members of the public are maintained within the limits established in this Order and to control radioactive contamination through the management of real and personal property. It is also a DOE objective that potential exposures to members of the public be as far below the limits as is reasonably achievable (ALARA) and that DOE facilities have the capabilities, consistent with the types of operations conducted, to monitor routine and non-routine releases and to assess doses to members of the public.

Chapter IV of this Order presents radiological protection requirements and guidelines for cleanup of residual radioactive material, management of the resulting wastes and residues, and release of the property. These requirements and guidelines are applicable at the time the property is released. Properties subject to these criteria include, but are not limited to, sites identified by the Formerly Utilized Sites Remedial Action Program (FUSRAP) and the Surplus Facilities Management Program (SFMP). The topics covered are basic dose limits, guidelines and authorized limits for allowable levels of residual radioactive material, and control of the radioactive wastes and residues.

This chapter provides guidance on radiation protection of the public and the environment from

- residual concentrations of radionuclides in soil (for these purposes, soil is defined as unconsolidated earth material, including rubble and debris that might be present in earth material);
- concentrations of airborne radon decay products;
- external gamma radiation;
- surface contamination; and
- radionuclide concentrations in air or water resulting from, or associated with, any of the above.

A guideline for residual radioactive material is a level of radioactive material that is acceptable for use of property without restrictions due to residual radioactive material. Guidelines for residual radioactive material presented herein are of two kinds: generic and specific. The basis for the guidelines is generally a presumed worst-case plausible-use scenario for the property.

Generic guidelines, independent of the property, are taken from existing radiation protection standards. Generic guideline values are presented in this chapter.

Specific property guidelines are derived from basic dose limits using specific property models and data. Procedures and data for deriving specific property guideline values are given by DOE/CH-8901.

DOE-STD-1120-98, Integration of Environment, Safety, and Health into Facility Disposition Activities

[Note: This standard has been superseded by DOE-STD-1120-2005, from which the following is taken.]

Subsequent to the initial release of DOE-STD-1120-98, nuclear safety basis requirements were promulgated in 10 CFR 830, subpart B. The standard was identified as a "safe harbor" approach for preparing a DSA for decommissioning and environmental restoration activities. The May 1998 version of the standard was not explicit regarding compliance with safety basis requirements of 10 CFR 830. Therefore, volume 1 of DOE-STD-1120-98 has been revised to focus on DSA requirements.

Volume 2 still retains a broad focus on integrated safety management, and many of the appendices apply to all facility disposition activities including deactivation and long-term surveillance and maintenance. A number of topics previously covered in both volumes of STD-1120-98 were either reconfigured or not retained in the current revision.

Volume 1: *Documented Safety Analysis for Decommissioning and Environmental Restoration Projects*, has four sections, including an introductory section. Section 2 discusses general safety basis concepts that have a direct or indirect impact on the DSA. Section 3 provides guidance on preparing DSAs and TSRs that are compliant with 10 CFR 830, subpart B requirements and associated methodology for decommissioning of a nuclear facility. Section 4 provides guidance on preparing DSAs and TSRs that are compliant with 10 CFR 830, subpart B requirements and associated methodology for environmental restoration activities involving work not performed within a permanent structure.

Volume 2: *Appendices*, complements other sections of the standard with additional ES&H information. Appendix A provides a set of candidate DOE ES&H directives and external regulations, organized by hazard types that may be used to identify potentially applicable directives to a specific facility disposition activity. Appendix B offers examples and lessons learned that illustrate implementation of ES&H approaches discussed in section 3 of volume 1. Appendix C contains integrated safety management guidance that applies to all facility disposition projects. Appendix D provides supplemental safety basis guidance related to inactive waste sites. Appendix E provides example risk binning guidelines that can be used to support control selection. Appendix F provides guidance for readiness evaluations.

b. General familiarity with requirements and guidance from other Federal agencies, or from DOE in collaboration with other Federal agencies (e.g., NRC and EPA) regarding the decontamination, decommissioning, and release of property that may be applicable to the disposition of DOE sites and facilities.

This varies from site to site. The Qualifying Official will evaluate the completion of this competency. However, the following information may be helpful.

The following is taken from the DOE Policy on Decommissioning of Department of Energy Facilities under the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA). No formal policy number has been assigned.

This policy establishes the approach agreed upon by DOE and the EPA for the conduct of decommissioning projects consistent with CERCLA requirements. This policy creates a framework for the conduct of decommissioning of DOE facilities and provides guidance to EPA regions and DOE operations offices on the use of CERCLA response authority to decommission such facilities. The principal objectives of this policy are to ensure that decommissioning activities are protective of worker and public health and the environment, consistent with CERCLA and, where applicable, RCRA, ensure stakeholder involvement, and achieve risk reduction without unnecessary delay.

This policy builds on the foundation established in the recent EPA/DOE/DOD "Guidance on Accelerating CERCLA Environmental Restoration at Federal Facilities" (August 22, 1994). Specifically, this Policy represents the next step in realizing the goal of that guidance to "develop decisions that appropriately address the reduction of risk to human health and the environment as expeditiously as the law allows." To achieve that end, this policy endorses the use of removal action authority to conduct decommissioning, although DOE and EPA recognize that removal action will not necessarily be the final response action needed at a facility subject to decommissioning.

The DOE Office of Environmental Management and EPA Federal Facilities Restoration and Reuse Office and Federal Facilities Enforcement Office have developed this approach for applying CERCLA authority to decommissioning activities to encourage streamlined decision-making in decommissioning activities.

The following is taken from NUREG-1757.

As part of its redesign of the materials license program, the U.S. Nuclear Regulatory Commission (NRC), Office of Nuclear Material Safety and Safeguards (NMSS) has consolidated and updated numerous decommissioning guidance documents into a three-volume NUREG. Specifically, the three volumes address the following topics:

- "Decommissioning Process for Materials Licensees;"
- "Characterization, Survey, and Determination of Radiological Criteria;" and
- "Financial Assurance, Recordkeeping, and Timeliness."

This three-volume NUREG series replaces NUREG-1727 (NMSS Decommissioning Standard Review Plan) and NUREG/BR-0241 (NMSS Handbook for Decommissioning Fuel Cycle and Materials Licensees). This NUREG series is intended for use by NRC staff, licensees, and others.

Volume 2 of the NUREG series, entitled, "Consolidated Decommissioning Guidance: Characterization, Survey, and Determination of Radiological Criteria," provides guidance on compliance with the radiological criteria for license termination (License Termination Rule) in 10 CFR Part 20, subpart E. This guidance takes a risk-informed, performance-based approach to the demonstration of compliance. The approaches to license termination described in this guidance will help to identify the information (subject matter and level of detail) needed to terminate a license by considering the specific circumstances of the wide range of NRC licensees. Licensees should use this guidance in preparing decommissioning plans, license termination plans, final status surveys, and other technical decommissioning

reports for NRC submittal. NRC staff will use the guidance in reviewing these documents and related license amendment requests.

c. Familiarity with the Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM; NUREG-1575; EPA 402-R-97-016) for planning, conducting, evaluating, and documenting building surface and surface soil final status radiological surveys for demonstrating compliance with dose or risk-based regulations or standards.

The following is taken from the MARSSIM, which is a multi-agency consensus document, incorporating the other documents listed.

MARSSIM provides an approach that is technically defensible and flexible enough to be applied to a variety of site-specific conditions. Applying this guidance to a dose- or risk-based regulation provides a consistent approach to protecting human health and the environment. The manual's performance-based approach to decision making provides the flexibility needed to address compliance demonstration at individual sites.

The process described in MARSSIM begins with the premise that a release criterion has already been provided in terms of a measurement quantity. The methods presented in MARSSIM are generally applicable and are not dependent on the value of the release criterion.

A release criterion is a regulatory limit expressed in terms of dose (mSv/y or mrem/y) or risk (cancer incidence or cancer mortality). The terms release limit or cleanup standard are also used to describe this term. A release criterion is typically based on the TEDE, the CEDE, risk of cancer incidence (morbidity), or risk of cancer death (mortality), and generally cannot be measured directly. Exposure pathway modeling is used to calculate a radionuclide-specific predicted concentration or surface area concentration of specific nuclides that could result in a dose (TEDE or CEDE) or specific risk equal to the release criterion. In this manual, such a concentration is termed the derived concentration guideline level (DCGL). Exposure pathway modeling is an analysis of various exposure pathways and scenarios used to convert dose or risk into concentration. In many cases, DCGLs can be obtained from responsible regulatory agency guidance based on default modeling input parameters, while other users may elect to take into account site-specific parameters to determine DCGLs. In general, the units for the DCGL are the same as the units for measurements performed to demonstrate compliance (e.g., Bq/kg or pCi/g, Bq/m^2 or dpm/100 cm^2). This allows direct comparisons between the survey results and the DCGL. A discussion of the uncertainty associated with using DCGLs to demonstrate compliance is included in appendix D, section D.6.

An investigation level is a radionuclide-specific level based on the release criterion that, if exceeded, triggers some response such as further investigation or remediation. An investigation level may be used early in decommissioning to identify areas requiring further investigation, and may also be used as a screening tool during compliance demonstration to identify potential problem areas. A DCGL is an example of a specific investigation level.

MARSSIM addresses the concern for small areas of elevated activity by using a simple comparison to an investigation level as an alternative to statistical methods. Using the elevated measurement comparison (EMC) represents a conservative approach, in that every

measurement needs to be below the action level. The investigation level for this comparison is called the $DCGL_{EMC}$, which is the $DCGL_W$ modified to account for the smaller area. (The "W" in $DCGL_W$ stands for Wilcoxon Rank Sum test, which is the statistical test recommended in MARSSIM for demonstrating compliance when the contaminant is present in background. The Sign test recommended for demonstrating compliance when the contaminant is not present in background also uses the $DCGL_W$.)

While the derivation of DCGLs is outside the scope of MARSSIM, it is important to understand the assumptions that underlie this derivation. The derivation assumptions must be consistent with those used for planning a compliance demonstration survey. One of the most important assumptions used for converting a dose or risk limit into a media-specific concentration is the modeled area of contamination. Other considerations include sample depth, composition, modeling parameters, and exposure scenarios. MARSSIM defines two potential DCGLs based on the area of contamination.

- If the residual radioactivity is evenly distributed over a large area, MARSSIM looks at the average activity over the entire area. The DCGLw is derived based on an average concentration over a large area.
- If the residual radioactivity appears as small areas of elevated activity within a larger area, typically smaller than the area between measurement locations, MARSSIM considers the results of individual measurements. The $DCGL_{EMC}$ is derived separately for these small areas and generally from different exposure assumptions than those used for larger areas.

A site is any installation, facility, or discrete, physically separate parcel of land, or any building or structure or portion thereof that is being considered for survey and investigation.

Area is a very general term that refers to any portion of a site, up to and including the entire site.

Decommissioning is the process of safely removing a site from service, reducing residual radioactivity through remediation to a level that permits release of the property, and termination of the license or other authorization for site operation. Although only part of the process, the term decommissioning is used in this sense for the Radiation Survey and Site Investigation (RSSI) Process, and is used this way throughout MARSSIM.

A survey unit is a physical area consisting of structure or land areas of specified size and shape for which a separate decision will be made as to whether or not that area exceeds the release criterion. This decision is made as a result of the final status survey, the survey in the RSSI Process used to demonstrate compliance with the regulation or standard. The size and shape of the survey unit are based on factors such as the potential for contamination, the expected distribution of contamination, and any physical boundaries (e.g., buildings, fences, soil type, surface water body) at the site.

For MARSSIM, measurement is used interchangeably to mean: 1) the act of using a detector to determine the level or quantity of radioactivity on a surface or in a sample of material removed from a media being evaluated, or 2) the quantity obtained by the act of measuring. Direct measurements are obtained by placing a detector near the media being surveyed and inferring the radioactivity level directly from the detector response. Scanning is a

measurement technique performed by moving a portable radiation detector at a constant speed above a surface to semi-quantitatively detect areas of elevated activity. Sampling is the process of collecting a portion of an environmental medium as being representative of the locally remaining medium. The collected portion, or aliquot, of the medium is then analyzed to identify the contaminant and determine the concentration. The word sample may also refer to a set of individual measurements drawn from a population whose properties are studied to gain information about the entire population. This second definition of sample is primarily used for statistical discussions.

To make the best use of resources for decommissioning, MARSSIM places greater survey efforts on areas that have, or had, the highest potential for contamination. This is referred to as a graded approach. The final status survey uses statistical tests to support decision making. These statistical tests are performed using survey data from areas with common characteristics, such as contamination potential, which are distinguishable from other areas with different characteristics. Classification is the process by which an area or survey unit is described according to radiological characteristics. The significance of survey unit classification is that this process determines the final status survey design and the procedures used to develop this design. Preliminary area classifications, made earlier in the MARSSIM process, are useful for planning subsequent surveys.

Areas that have no reasonable potential for residual contamination are classified as non-impacted areas. These areas have no radiological impact from site operations and are typically identified early in decommissioning. Areas with reasonable potential for residual contamination are classified as impacted areas.

Impacted areas are further divided into one of three classifications:

- Class 1 areas are areas that have, or had prior to remediation, a potential for radioactive contamination (based on site operating history) or known contamination (based on previous radiation surveys) above the $DCGL_W$. Examples of class 1 areas include: 1) site areas previously subjected to remedial actions, 2) locations where leaks or spills are known to have occurred, 3) former burial or disposal sites, 4) waste storage sites, and 5) areas with contaminants in discrete solid pieces of material and high specific activity.
- Class 2 areas are areas that have, or had prior to remediation, a potential for radioactive contamination or known contamination, but are not expected to exceed the $DCGL_W$. To justify changing the classification from class 1 to class 2, there should be measurement data that provides a high degree of confidence that no individual measurement would exceed the $DCGL_W$. Other justifications for reclassifying an area as class 2 may be appropriate, based on site-specific considerations. Examples of areas that might be classified as class 2 for the final status survey include: 1) locations where radioactive materials were present in an unsealed form, 2) potentially contaminated transport routes, 3) areas downwind from stack release points, 4) upper walls and ceilings of buildings or rooms subjected to airborne radioactivity, 5) areas handling low concentrations of radioactive materials, and 6) areas on the perimeter of former contamination control areas.
- Class 3 areas are any impacted areas that are not expected to contain any residual radioactivity, or are expected to contain levels of residual radioactivity at a small fraction of the $DCGL_W$, based on site operating history and previous radiation

surveys. Examples of areas that might be classified as class 3 include buffer zones around class 1 or class 2 areas, and areas with very low potential for residual contamination but insufficient information to justify a non-impacted classification.

Class 1 areas have the greatest potential for contamination, and therefore receive the highest degree of survey effort for the final status survey using a graded approach, followed by class 2, and then by class 3. Non-impacted areas do not receive any level of survey coverage because they have no potential for residual contamination. Non-impacted areas are determined on a site-specific basis. Examples of areas that would be non-impacted rather than impacted usually include residential or other buildings that have, or had, nothing more than smoke detectors or exit signs with sealed radioactive sources.

If the radionuclide of potential concern is present in background, or if the measurement system used to determine concentration in the survey unit is not radionuclide-specific, background measurements are compared to the survey unit measurements to determine the level of residual radioactivity. The background reference area is a geographical area from which representative reference measurements are taken for comparison with measurements taken in specific survey units. The background reference area is defined as an area that has similar physical, chemical, radiological, and biological characteristics as the survey unit(s) being investigated, but has not been contaminated by site activities (i.e., it is non-impacted).

The process of planning the survey, implementing the survey plan, and assessing the survey results prior to making a decision is called the data life cycle. Survey planning uses the data quality objectives process to ensure that the survey results are of sufficient quality and quantity to support the final decision. Quality assurance and quality control (QA/QC) procedures are performed during implementation of the survey plan to collect information necessary to evaluate the survey results. Data quality assessment is the process of assessing the survey results, determining that the quality of the data satisfies the objectives of the survey, and interpreting the survey results as they apply to the decision being made.

A systematic process and structure for quality should be established to provide confidence in the quality and quantity of data collected to support decision making. The data used in decision making should be supported by a planning document that records how quality assurance and quality control are applied to obtain the type and quality of results that are needed and expected. There are several terms used to describe a variety of planning documents, some of which document only a small part of the survey design process. MARRSIM uses the term Quality Assurance Project Plan (QAPP) to describe a single document that incorporates all of the elements of the survey design. This term is consistent with consensus guidance ANSI/ASQC E4-1994 (ASQC 1995), *Specifications and Guidelines for Quality Systems for Environmental Data Collection and Environmental Technology Programs* and EPA guidance (EPA 1994c; EPA 1997a), and is recommended to promote consistency. The use of the term QAPP in MARSSIM does not exclude the use of other terms (e.g., decommissioning plan, sampling and analysis plan, field sampling plan) to describe survey documentation provided the information included in the documentation supports the objectives of the survey.

d. Knowledge of guidance regarding the development and analysis of property release options, and determination of authorized limits for property and material

to be managed or released from DOE, as contained in DOE Implementation Guide G-441.1-xx, Control and Release of Property with Residual Radioactive Material for use with DOE 5400.5, Radiation Protection of the Public and the Environment.

[Note: DOE Implementation Guide G-441.1-xx has been superseded by DOE G 441.1-1C, from which the following is taken.]

The DOE operates a variety of nuclear facilities including reactors, accelerators, and weapons test facilities. Many of these facilities are undergoing remediation and decommissioning. This draft guide references DOE's requirements and presents DOE's guidance for the control and release of property that may contain residual radioactive material. Implementation guidance is provided for Department and contractor personnel who perform cleanup of property contaminated with residual radioactive material and who must determine the disposition of property under the requirements in DOE [Order] 5400.5, *Radiation Protection of the Public and the Environment*, and its proposed successor, 10 CFR 834, *Radiation Protection of the Public and the Environment*.

This guide is applicable to decommissioning, deactivation, decontamination, and remedial action of property with residual radioactive contamination. Guidance for implementing DOE Order 5400.5 requirements for the release of property has been provided by the Office of Environmental Policy and Guidance over the past 10 years through individual memorandums, guidance documents and handbooks, and modeling and analysis tools.

A principal objective of this guide is to integrate the key elements of these individual guidance documents and tools into one document as a principal resource for DOE and contractor personnel.

e. Familiarity with DOE dose and risk modeling tools applicable to the decontamination and decommissioning of sites and facilities, such as: (1) the RESRAD (RESidual RADioactivity) code (User's Manual for RESRAD Version 6, Argonne National Laboratory, ANL/EAD-4, 2001); and (2) the RESRAD-BUILD code (A Computer Model for Analyzing the Radiological Doses Resulting from the Remediation and Occupancy of Buildings Contaminated with Radioactive Material, ANL/EAD-LD-3, 1994).

RESRAD

In 1993, a manual on using version 5 of the RESRAD code to implement the DOE's residual radioactive material guidelines was released. Since then, as part of the RESRAD QA program, the RESRAD code has undergone extensive review, benchmarking, verification, and validation. The manual and code have been used widely by DOE and its contractors, the U.S. Nuclear Regulatory Commission, the U.S. EPA, the U.S. Army Corps of Engineers, industrial firms, universities, and foreign government agencies and institutions. New features, some in response to comments received from users, have been incorporated into the code to form RESRAD 6. These improvements have increased RESRAD's capabilities and flexibility and have enabled users to interact with the code more easily. With the improvements, the code has become more realistic in terms of the models and default parameters it uses. RESRAD 6 represents the sixth major version of the RESRAD code since it was first issued in 1989.

The RESRAD code can now perform uncertainty/probabilistic analyses with an improved probabilistic interface. It uses a preprocessor and a postprocessor to perform probabilistic dose and risk analyses. It incorporates default parameter distributions (based on national average data) for selected parameters. The code can provide analysis results as text reports, interactive output, and graphic output. The results of an uncertainty analysis can be used as a basis for determining the cost effectiveness of obtaining additional information or data on input parameters (variables).

The RESRAD code now allows users to calculate the time-integrated dose and risk at user-specified times. The instantaneous dose/risk can be calculated by setting the time integration point to one. The other major changes in the code are a new external exposure model, a new area factor model for the inhalation pathway, and a new dose conversion factor (DCF) editor. The DCF Editor gives users the ability to change dose conversion factors, transfer factors, and slope factors. Moreover, the RESRAD database was updated. It now includes inhalation and ingestion dose conversion factors from the EPA's Federal Guidance Report No. 11 (FGR-11), direct external exposure dose conversion factors from FGR-12, risk slope factors primarily from the latest health effects and summary tables, and radionuclide half-lives from ICRP Publication 38. The risk coefficients from FGR-13 are also available in the code.

Version 6 of RESRAD incorporates many improvements made since Version 5 was released. These include the addition of seven new radionuclides: ^{79}Se, ^{93}Zr, ^{93m}Nd, ^{133}Ba, ^{245}Cm, ^{246}Cm, and ^{247}Cm. The code was also modified to account for radioactive decay and in-growth during food storage times. The code now has an improved groundwater model to ensure convergence for distribution coefficient calculations when water concentrations are known; has an improved radon pathway to reduce execution time; allows sensitivity analyses of many more parameters, such as transfer factors, leach rate, and solubility; has a modified user interface to better check the sensitivity ranges; and contains many updates in its graphics. The code now allows users to input radionuclide activity in SI units, and the resultant doses can also be reported in SI units. QA files are now distributed along with the code so that users can ensure that the code is properly installed and verify RESRAD calculations on their computers.

The latest RESRAD code works only on computers with Windows operating systems. The DOS version of RESRAD is no longer supported. To help users familiar with the DOS interface to use the Windows interface, the new interface incorporates many features similar to those used in DOS, including a DOS main menu emulator, mapping of parameters to screens (windows), navigation (tab/F10/ESC), sensitivity settings and graphics, the format for results reports, and context-specific help. The new features of RESRAD for Windows (95, 98, 2000, and NT) include a graphical display of active pathways, a sensitivity summary bar, integrated plot options and results, color-coded default settings, an uncertainty summary window, button prompts to interpret icons, and soil strata graphic feedback. Furthermore, in order to take advantage of the latest computer technology, the RESRAD FORTRAN programs are now compiled by using the FORTRAN 95 (LAHEY/FUJITSU LF95) compiler. Also, the Windows interface, which was initially developed in 16-bit Visual Basic 4.0, is now upgraded to 32-bit Visual Basic 6.0.

The RESRAD code now has its own Web site (http://web.ead.anl.gov/resrad). The site has information on the updates to the code. The site enables users to download the most recent

releases of RESRAD, information on upcoming training workshops, and many documents relevant to the application of the code. Users can get technical assistance via e-mail (resrad@anl.gov). The entire user's manual is also found on the RESRAD Web site in Adobe portable document format.

The manual is a user's guide on the operation of RESRAD 6 and a reference manual on the algorithms and formulas used in RESRAD. To reflect the changes in the RESRAD code, this version is very different from previous versions. Some parts were removed (such as DOE guidance on procedures and requirements for obtaining approval for releases of material and property and an appendix on performing ALARA analyses). Other parts were added (such as a section on verification and validation, an appendix on uncertainty analysis, and an appendix on calculating time-integrated cancer risk). Many parts were extensively modified (such as the section describing the RESRAD user's guide and the appendixes on external ground radiation pathway factors, inhalation pathway factors, and the estimation of individual off-site doses). The discussion on exposure scenarios was expanded to include suburban resident, industrial worker, and recreationist scenarios.

RESRAD-BUILD Code

The RESRAD-BUILD computer code is a pathway analysis model developed to evaluate the potential radiological dose incurred by an individual who works or lives in a building contaminated with radioactive material. The radioactive material in the building structure can be released into the indoor air by mechanisms such as diffusion (radon gas and tritiated water), mechanical removal (decontamination activities), or erosion (removable surface contamination).

The transport of radioactive material within the building from one compartment to another is calculated with an indoor air quality model (see appendix A). The air quality model evaluates the transport of radioactive dust particulates and radon progeny due to (1) air exchange between compartments and with outdoor air, (2) the deposition and resuspension of particulates, and (3) radioactive decay and ingrowth. RESRAD-BUILD can model up to three compartments in a building, thereby making it possible to evaluate situations ranging from a one-room warehouse to a three-story house.

The design of RESRAD-BUILD is similar to that of the RESRAD code: the user can construct the exposure scenario by adjusting the input parameters. Typical building exposure scenarios include long-term occupancy (resident and office worker) and short-term occupancy (renovation worker and visitor). The long-term occupancy scenarios are usually low release scenarios, whereas short-term occupancy scenarios may involve a high release of contaminants in a short time period. Up to 10 receptor locations and 10 distinct source locations can be input into the RESRAD-BUILD code to calculate dose in a single run of the code. The calculated dose can be the total (individual) dose to a single receptor spending time at various locations, or the total (collective) dose to a workforce decontaminating the building. If a building is demolished, the RESRAD computer code may be used to evaluate the potential dose from the buried material.

The RESRAD-BUILD code considers seven exposure pathways: (1) external exposure directly from the source, (2) external exposure to materials deposited on the floor, (3) external exposure due to air submersion, (4) inhalation of airborne radioactive particulates,

(5) inhalation of aerosol indoor radon progeny (in the case of the presence of radon predecessors) and tritiated water vapor, (6) inadvertent ingestion of radioactive material directly from the source, and (7) ingestion of materials deposited on the surfaces of the building compartments. A detailed discussion of exposure scenarios and pathways is presented in section 3 of the user's manual.

The RESRAD-BUILD code can compute the attenuation due to a shielding material between each source-receptor combination when calculating the external dose. The user can select the shielding material from eight material types and input the thickness and density of the shielding material. The user may also define the source as a point, line, area, or volume source. Volume sources can be composed of up to five layers of different materials, with each layer being homogeneous and isotropic.

The radionuclides included in the RESRAD-BUILD code are similar to those included in the RESRAD code. Currently, the RESRAD-BUILD database contains 67 radionuclides, and additional radionuclides are being added. All these radionuclides have a half-life of six months or longer, and they are referred to as principal radionuclides. It is assumed that the short-lived decay products with half-lives of six months or less, referred to as the associated radionuclides, are in secular equilibrium with their parent (principal) radionuclides. For the 67 principal radionuclides in the current RESRAD-BUILD database, there are 53 associated radionuclides. Therefore, a total of 120 radionuclides are available in RESRAD-BUILD Version 3. A detailed list of these radionuclides can be found in table 3.1 of the RESRAD User's Manual.

f. Cursory knowledge of currently available technologies, and innovative technologies as available, that are applicable to the cleanup, decontamination, and decommissioning of DOE facilities.

The candidate should be able to discuss technologies that are available at the time of their evaluation. The Qualifying Official will evaluate their competency in this activity.

g. Knowledge, as appropriate to decontamination and decommissioning activities, regarding radiological control practices to minimize occupational exposures to ionizing radiation, as contained in the DOE Standard, Radiological Control (DOE-STD-1098-99).

[Note: DOE-STD-1098-99 has been superseded by DOE-STD-1098-2008, from which the following is taken.]

The following are radiological control practices appropriate to decontamination and decommissioning activities:

- Radiological work permits or technical work documents should include provisions to control contamination at the source to minimize the amount of decontamination needed.
- Work preplanning should include consideration of the handling, temporary storage, and decontamination of materials, tools, and equipment.
- Decontamination activities should be controlled to prevent the spread of contamination.

- Water and steam are the preferred decontamination agents. Other cleaning agents should be selected based on their effectiveness, hazardous properties, amount of waste generated, and ease of disposal.
- Facility line management should be responsible for directing decontamination efforts.

Technical requirements for the conduct of work, including construction, modifications, operations, maintenance, and decommissioning, should incorporate radiological criteria to ensure safety and maintain radiation exposures ALARA. In general, efforts to reduce individual dose should not be allowed to cause a concurrent increase in collective dose.

19. Radiation protection personnel shall demonstrate a familiarity level knowledge of the standards and DOE Orders pertaining to the packaging and transportation of radioactive materials.

a. Discuss the purpose and scope of the DOE O 460.1B, Packaging and Transportation Safety.

The objectives of this Order are to establish safety requirements for the proper packaging and transportation of DOE/NNSA offsite shipments and onsite transfers of hazardous materials and for modal transport. (Offsite is any area within or outside a DOE site to which the public has free and uncontrolled access; onsite is any area within the boundaries of a DOE site or facility to which access is controlled.) This Order is applicable to the following:

- Office of Civilian Radioactive Waste Management
- Office of Energy Efficiency and Renewable Energy
- Office of Environmental Management
- Office of Fossil Energy
- National Nuclear Security Administration
- Office of Nuclear Energy, Science, and Technology
- Office of Science
- Office of Security
- Bonneville Power Administration
- Southeastern Power Administration
- Southwestern Power Administration
- Western Area Power Administration

b. Describe the authorities and responsibilities of radiation protection personnel with respect to DOE O 460.1B, Packaging and Transportation Safety.

The following are special requirements for radioactive material packaging:

- Each person who offers for transportation or transports low specific activity materials or surface contaminated objects pursuant to 49 CFR 173.427 may use a strong, tight container or industrial packaging in accordance with the Hazardous Materials Regulations.
- Except for materials specified in subparagraph 4.a.(2)(a) of the Order and excepted quantities as defined in the Hazardous Materials Regulations, each person who offers for transportation or transports quantities of radioactive materials not exceeding A_1 or A_2, as appropriate, shall use a Type A package that is authorized by 49 CFR 173.415 or a Type B package authorized by subparagraph 4.a.(2)(c) of the Order.

- Each person who offers for transportation or transports quantities of radioactive materials exceeding A_1 or A_2, as appropriate, or fissile material, shall use a packaging that is certified by the Assistant Secretary for Environmental Management, or a Secretarial Officer/Deputy Administrator, NNSA, pursuant to 49 CFR 173.7(d), or that is otherwise authorized by 49 CFR 173.416 or 49 CFR 173.417.
- Each person who offers for transportation quantities of radioactive materials exceeding A_1 or A_2 or fissile quantity of radioactive material in a package certified by the Assistant Secretary for Environmental Management or another Secretarial Officer/Deputy Administrator, NNSA, shall
  - meet the conditions specified in the certificate for the package issued by the Assistant Secretary for Environmental Management or a Secretarial Officer/Deputy Administrator, NNSA or NRC; and
  - register in writing with the Assistant Secretary for Environmental Management or the responsible Secretarial Officer/Deputy Administrator, NNSA, prior to using the certificate.
- An application for a package certification must include a Safety Analysis Report for Packaging, which demonstrates that the packaging conforms with the standards of 10 CFR 71, subparts E, F, G, and H, and any other applicable standards that the Assistant Secretary for Environmental Management or a Secretarial Officer/Deputy Administrator, NNSA, may determine applicable for granting a certificate.

The following are additional requirements for plutonium packaging:

- Each person who offers for transportation or transports quantities of plutonium in excess of 20 curies per package may only use a packaging approved by the Assistant Secretary for Environmental Management or another Secretarial Officer/Deputy Administrator, NNSA, or the NRC as meeting the requirements of 10 CFR 71.63.
- Unless exempted for the purposes of national security pursuant to the provisions of 10 CFR 871, each person who offers Type B quantities of plutonium for air transportation shall use a packaging approved by the Assistant Secretary for Environmental Management or a Secretarial Officer/Deputy Administrator, NNSA, or the NRC as conforming to the requirements of 10 CFR 71.64.

20. Radiation protection personnel shall demonstrate a familiarity level knowledge of the Department's philosophy and approach to implementing Integrated Safety Management.

a. Explain the objective of Integrated Safety Management.

The following is taken from DOE M 450.4-1.

The objective of integrated safety management (ISM) is to perform work in a safe and environmentally sound manner. More completely, as described in DOE P 450.4, *Safety Management System Policy*: "The Department and Contractors must systematically integrate safety into management and work practices at all levels so that missions are accomplished while protecting the public, the worker, and the environment. This is to be accomplished through effective integration of safety management into all facets of work planning and execution. In other words, the overall management of safety functions and activities becomes an integral part of mission accomplishment." The desired result is that work is accomplished

in a safe manner. ISM is applicable to all facility life-cycle phases including design, construction, operation, and decontamination and decommissioning. In ISM, the term "safety" is used synonymously with ES&H to encompass protection of the public, the workers, and the environment.

b. Discuss the following existing Department programs and initiatives that lead to successful implementation of Integrated Safety Management:
- **Standards/Requirements Identification Documents (S/RIDs) and Work Smart Standards**
- **Contract reform and performance-based contracting**
- **Activities at research and development laboratories related to safety management (i.e., pilot oversight program for line ES&H management.)**
- **Operational Readiness Reviews (ORRs)**

S/RIDs and Work Smart Standards

The following is taken from DOE-HDBK-1148-2002.

DOE P 450.4, *Safety Management Policy*, commits the Department to conducting work efficiently and in a way that provides reasonable assurance of adequate protection of workers, the public and the environment. Demonstrating ISMS effectiveness is more than just an internal affair between the Department and its contractors. While protecting the environment and the safety and health of the public and workers, DOE is also committed to demonstrating good stewardship of resources, and to building public trust and confidence in its programs and plans.

The Department has deliberately adopted a standards-based approach to safety management that is intended to allow for good judgment in work design and resource allocation. This approach creates consistency and stability of expectations and accountability, permits judgment to be exercised at the level appropriate to effective management, and helps people do their jobs through teamwork. These features are the outward sign of an effective ISMS at work. Central to understanding ISM are the five core functions and seven guiding principles that undergird the contract requirements that mandate an ISMS. Key references for the establishment of ISMS include DOE P 450.4, *Safety Management Policy*; DOE G 450.4-1, *Integrated Safety Management System Guide*; DEAR 970.5204-2, "Integration of Environment, Safety, and Health into Work Planning and Execution;" and DEAR 970.5204-78; "Laws, Regulations, and DOE Directives." DEAR 970.5204-78(a) requires that the contractor assist DOE to comply with any applicable Federal, state, and local laws (List A), and conform to the agreed upon requirements of those Department of Energy directives, or parts thereof, identified in the List of Applicable Directives (List B). As noted in the ISMS Guide, ES&H requirements appropriate for work conducted by a contractor may be determined using a DOE-approved process to:
- evaluate the work and the associated hazards; and
- identify an appropriately tailored set of standards, practices, and controls. The resulting approved set shall be incorporated into the contract as required by DEAR 970.5204-78.

Approved processes for establishing ES&H requirements include the following:
- incorporation of a Standards/Requirements Identification Document (S/RID) into the contract,

- use of the Necessary and Sufficient Closure Process; and
- compliance with the DOE directives and other applicable laws and regulations.

Contract Reform and Performance-Based Contracting

The following is taken from 48 CFR 970.5223-1.

For the purposes of this clause,

- Safety encompasses environment, safety and health, including pollution prevention and waste minimization; and
- Employees include subcontractor employees.

In performing work under this contract, the contractor shall perform work safely, in a manner that ensures adequate protection for employees, the public, and the environment, and shall be accountable for the safe performance of work. The contractor shall exercise a degree of care commensurate with the work and the associated hazards. The contractor shall ensure that management of ES&H functions and activities becomes an integral but visible part of the contractor's work planning and execution processes. The contractor shall, in the performance of work, ensure that:

- Line management is responsible for the protection of employees, the public, and the environment. Line management includes those contractor and subcontractor employees managing or supervising employees performing work.
- Clear and unambiguous lines of authority and responsibility for ensuring ES&H are established and maintained at all organizational levels.
- Personnel possess the experience, knowledge, skills, and abilities that are necessary to discharge their responsibilities.
- Resources are effectively allocated to address ES&H, programmatic, and operational considerations. Protecting employees, the public, and the environment is a priority whenever activities are planned and performed.
- Before work is performed, the associated hazards are evaluated and an agreed-upon set of ES&H standards and requirements are established which, if properly implemented, provide adequate assurance that employees, the public, and the environment are protected from adverse consequences.
- Administrative and engineering controls to prevent and mitigate hazards are tailored to the work being performed and associated hazards. Emphasis should be on designing the work and/or controls to reduce or eliminate the hazards and to prevent accidents and unplanned releases and exposures.
- The conditions and requirements to be satisfied for operations to be initiated and conducted are established and agreed-upon by DOE and the contractor. These agreed-upon conditions and requirements are requirements of the contract and binding upon the contractor. The extent of documentation and level of authority for agreement shall be tailored to the complexity and hazards associated with the work and shall be established in a safety management system.

The contractor shall manage and perform work in accordance with a documented safety management system (SMS) that fulfills all conditions in paragraph (b) of this clause at a minimum. Documentation of the SMS shall describe how the contractor will:

- Define the scope of work;
- Identify and analyze hazards associated with the work;

- Develop and implement hazard controls;
- Perform work within controls; and
- Provide feedback on adequacy of controls and continue to improve safety management.

The SMS shall describe how the contractor will establish, document, and implement safety performance objectives, performance measures, and commitments in response to DOE program and budget execution guidance while maintaining the integrity of the System. The System shall also describe how the contractor will measure system effectiveness.

The contractor shall submit to the contracting officer documentation of its System for review and approval. Dates for submittal, discussions, and revisions to the SMS will be established by the contracting officer. Guidance on the preparation, content, review, and approval of the System will be provided by the contracting officer. On an annual basis, the contractor shall review and update, for DOE approval, its safety performance objectives, performance measures, and commitments consistent with and in response to DOE's program and budget execution guidance and direction. Resources shall be identified and allocated to meet the safety objectives and performance commitments as well as maintain the integrity of the entire SMS. Accordingly, the SMS shall be integrated with the contractor's business processes for work planning, budgeting, authorization, execution, and change control.

The contractor shall comply with, and assist the Department of Energy in complying with, ES&H requirements of all applicable laws and regulations, and applicable directives identified in the clause of this contract entitled "Laws, Regulations, and DOE Directives." The contractor shall cooperate with Federal and non-Federal agencies having jurisdiction over ES&H matters under this contract.

The contractor shall promptly evaluate and resolve any noncompliance with applicable ES&H requirements and the SMS. If the contractor fails to provide resolution or if, at any time, the contractor's acts or failure to act causes substantial harm or an imminent danger to the environment or health and safety of employees or the public, the contracting officer may issue an order stopping work in whole or in part. Any stop work order issued by a contracting officer under this clause (or issued by the contractor to a subcontractor in accordance with paragraph (i) of this clause) shall be without prejudice to any other legal or contractual rights of the Government. In the event that the contracting officer issues a stop work order, an order authorizing the resumption of the work may be issued at the discretion of the contracting officer. The contractor shall not be entitled to an extension of time or additional fee or damages by reason of, or in connection with, any work stoppage ordered in accordance with this clause.

Regardless of the performer of the work, the contractor is responsible for compliance with the ES&H requirements applicable to this contract. The contractor is responsible for flowing down the ES&H requirements applicable to this contract to subcontracts at any tier to the extent necessary to ensure the contractor's compliance with the requirements.

The contractor shall include a clause substantially the same as this clause in subcontracts involving complex or hazardous work on site at a DOE-owned or -leased facility. Such subcontracts shall provide for the right to stop work under the conditions described in

paragraph (g) of this clause. Depending on the complexity and hazards associated with the work, the contractor may choose not to require the subcontractor to submit a Safety Management System.

Activities at R&D Laboratories

The following is taken from DOE G 441.1-1C.

A radiological control organization should be established to support line managers and workers. To function effectively and be consistent, as necessary, with the requirements in DOE O 226.1, Implementation of Department of Energy Oversight Policy (DOE 2005b) the radiological control organization should be independent of the line organizational element responsible for production, operation, or research activities, and should have an equivalent reporting level. Radiological control organization function is discussed in detail in the RCS. Other organizational schemes that allow effective compliance with the standards set forth in 10 CFR 835 should be considered to address site- or facility-specific needs.

Operational Readiness Reviews

The following is taken from DOE O 425.1C.

The objective of DOE O 425.1C is to establish the requirements for the Department of Energy, including the National Nuclear Security Administration (NNSA), for startup of new nuclear facilities and for the restart of existing nuclear facilities that have been shut down. Nuclear facilities are activities or operations that involve radioactive and/or fissionable materials in such form or quantity that a nuclear hazard potentially exists to the employees or the general public.

The requirements specify a readiness review process that must, in all cases, demonstrate that it is safe to start (or restart) the applicable facility. The facility must be started (or restarted) only after documented independent reviews of readiness have been conducted and the approvals specified in this Order have been received. The readiness reviews are not intended to be tools of line management to achieve readiness. Rather, the readiness reviews provide an independent confirmation of readiness to start or restart operations.

c. Discuss the purpose, content, and application of DOE P 450.4, Safety Management System Policy, and DOE G 450.4-1B, Integrated Safety Management System Guide.

DOE P 450.4, Safety Management System Policy

Safety management systems provide a formal, organized process whereby people plan, perform, assess, and improve the safe conduct of work. The safety management system is institutionalized through DOE directives and contracts to establish the Department-wide safety management objective, guiding principles, and functions.

The Department is committed to conducting work efficiently and in a manner that ensures protection of workers, the public, and the environment. It is Department policy that safety management systems described in the policy shall be used to systematically integrate safety into management and work practices at all levels so that missions are accomplished while protecting the public, the worker, and the environment. Direct involvement of workers during

the development and implementation of safety management systems is essential for their success.

The DOE safety management system establishes a hierarchy of components to facilitate the orderly development and implementation of safety management throughout the DOE complex. The safety management system consists of six components:

- the objective;
- guiding principles;
- core functions;
- mechanisms;
- responsibilities; and
- implementation.

The objective, guiding principles, and core functions of safety management identified below shall be used consistently in implementing safety management throughout the DOE complex. The mechanisms, responsibilities, and implementation components are established for all work and will vary based on the nature and hazard of the work being performed. Throughout this policy statement, the term safety is used synonymously with ES&H to encompass protection of the public, the workers, and the environment.

DOE G 450.4-1B, Integrated Safety Management System Guide

This guide has two purposes. One purpose is to assist DOE contractors in developing, describing, and implementing an ISMS in compliance with DOE P 450.4, Safety Management System Policy (the SMS Policy); DOE P 450.5, Line Environment, Safety and Health Oversight; DOE P 450.6, Secretarial Policy Statement Environment, Safety and Health; DOE P 411.1, Safety Management Functions, Responsibilities, and Authorities (FRAM); and the following provisions of the Department of Energy Acquisition Regulation (DEAR):

- 48 CFR 970.5223-1, which requires integration of environment, safety, and health into work planning and execution
- 48 CFR 970.5204-2, which deals with laws, regulations, and DOE directives
- 48 CFR 970.1100-1, which requires performance-based contracting

Attachments 1 through 5 to volume 1 of this guide contain the full text of the policies and the relevant ISMS sections of the DEAR.

A second purpose of this guide is to assist DOE line managers and contracting officers (COs) who

- provide ISMS guidance and requirements;
- review and approve ISMS products;
- verify implementation of the ISMS;
- perform various integrating activities (e.g., planning, budgeting, review, approval, and oversight) that complement, or are required, for the ISMS.

21. Radiation protection personnel shall demonstrate a familiarity level knowledge of the Department's guidance for the structure, function, and operation of a radiation generating device (RGD) control program as discussed in Implementation Guide G 441.1-5, Radiation Generating Devices.

[Note: DOE G 441.1-5 has been canceled and superseded by DOE G 441.1-1C.]

a. Describe the different types of radiation-generating devices that may be used at DOE facilities, including:
- **X-ray machines**
- **Accelerators**
- **Irradiators**
- **Radiography sources**

X-Ray Machines

The following is taken from DOE G 441.1-1C.

X-ray machines are usually shielded to attenuate the emission of x-rays. Requirements for the exempt-shielded or shielded installations prevail. Examples include electron beam welders, electronic microscopes, pulse generators, and microwave cavities, if used as beam guides.

Accelerators

Note: Accelerators and their use are discussed in detail in Competency 9, KSA "b" above. Therefore, no additional discussion will be provided here.

Irradiators

The following definition is taken from DOE G 441.1-1C.

Irradiators are any gamma-emitting or neutron-emitting sealed radioactive materials that have the potential to create a radiation level exceeding 500 rads (5 grays) in 1 hour at 1 meter and are operated within the requirements of an RGD installation.

Radiography Sources

The following is taken from DOE-HDBK-1141-2008.

Sealed gamma ray sources are used for a variety of applications in industry. Gamma ray sources are the most common sealed source encountered, although others are used and are discussed later.

Radiography is probably the most common use, and may be performed with the gamma rays from sealed sources of ^{60}Co, ^{137}Cs, or ^{192}Ir. Other uses of sealed gamma ray sources are thickness gauges (e.g., to determine the thickness of sheet metal), level gauges (e.g., to determine a fluid level in a container), and density gauges (e.g., to measure the geologic formation porosity during oil and mineral logging).

The hazard from these sources is primarily an external dose hazard. The most common cause of overexposure incidents with gamma radiography sources results from radiographers failing to perform radiation surveys to verify that the gamma source is back in the shielded

position. Also, if mechanical damage to the source encapsulation occurs, radioactive material contamination will be a hazard as well.

b. Using Implementation Guide G 441.1-5, Radiation Generating Devices, discuss the difference between an open beam and cabinet x-ray system and the different types of controls (design, equipment, and administrative) that can be used to prevent radiation exposure above DOE limits.

[Note: DOE G 441.1-5 has been canceled and is superseded by DOE G 441.1-1C, from which the following is taken.]

A cabinet x-ray system is an x-ray system with the x-ray tube installed in an enclosure which, independent of existing architectural structures except the floor on which it may be placed, is intended to contain at least that portion of a material being irradiated, provide radiation attenuation, and exclude individuals from its interior during generation of x-radiation. Included are all the x-ray systems designed primarily for inspection of carry-on baggage at airline, railroad, and bus terminals, and in similar facilities. An x-ray tube used within a shielded part of a building, or x-ray equipment which may temporarily or occasionally incorporate portable shielding, is not considered a cabinet x-ray system.

The following is taken from DOE-HDBK-1130-2008.

Analytical x-ray devices use x-rays for diffraction or fluorescence experiments. These research tools are normally used in materials science. ANSI N43.2, *Radiation Safety for X-Ray Diffraction and Fluorescence Analysis Equipment,* defines two types of analytical x-ray systems: enclosed beam and open beam.

According to ANSI N43.2, a device that does not meet the enclosed-beam standards is classified as an open-beam system. In an open-beam system, one or more x-ray beams are not enclosed, making exposure of human body parts possible during normal operation. The open-beam system is acceptable for use only if an enclosed beam is impractical because of any of the following reasons:

- A need for frequent changes of attachments and configurations.
- A need for making adjustments with the x-ray beam energized.
- Motion of specimen and detector over wide angular limits.
- The examination of large or bulky samples.

The following safety features are essential in an open-beam x-ray system:

- Each port of the x-ray tube housing must have a beam shutter.
- All shutters must have a conspicuous SHUTTER OPEN indicator of failsafe design.
- Shutters at unused ports should be mechanically or electrically secured to prevent casual opening.
- Special rules apply when the accessory setup is subject to change as is the case with powder diffraction cameras. In these cases, the beam shutter must be interlocked so the port will be open only when the accessory is in place.
- Exposure rates adjacent to the system must not exceed 2.5 mrem/hour at 5 cm from the surface of the housing.
- A guard or interlock must prevent entry of any part of the body into the primary beam.

The following safety features are common to both systems:

- A fail-safe light or indicator is installed in a conspicuous location near the x-ray tube housing. These indicators are energized automatically and only when the tube current flows or high voltage is applied to the x-ray tube.
- Accessories to the equipment have a beam stop or other barrier.
- Shielding is provided.

c. Discuss possible exposure incidents (or ones that have actually happened in medicine, private industry, or at DOE facilities) as a result of improper practices with accelerators, irradiators, and radiography sources, or loss of control of sources.

This is a site-specific activity. The Qualifying Official will evaluate its completion.

d. Discuss possible actions to control sources at DOE facilities, especially radiography sources brought on DOE sites by subcontractors who may be unaware of the site Radiation Protection Program (RPP) and 10 CFR 835, Occupational Radiation Protection, requirements.

This is a site-specific activity. The Qualifying Official will evaluate its completion.

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