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Exam Preview:

1. It has been demonstrated that in some operations (such as welding over a short period of time) differences of as much as a factor of 10 between the right and left lapel PAS measurements can be expected.
    a. True
    b. False
2. Operations involving significant amounts of elements, such as ^{239}Pu, should be conducted in a ventilated glove-box environment and with monitoring systems capable of detection of small releases involving 1 derived air concentration (DAC) averaged over _ hours under laboratory conditions.
    a. 2
    b. 4
    c. 6
    d. 8
3. Rate of particle dissolution is divided into three categories by the ICRP Publication 68 model. Types F (fast), M (moderate), and S (slow) refer to the rate of absorption of the material in the pulmonary region of the lungs. Which of the following retention half-life ranges corresponds to Type S?
    a. Less than 10 days
    b. 10 to 100 days
    c. Greater than 100 days
    d. Greater than 200 days
4. Filters should have high collection efficiencies for particles over a wide range of sizes.
    a. True
    b. False

5. Effective dose rates of up to 150 mrem/h, attributed to radium accumulation, have been measured from neoprene liner material. Dose rates from furnace lids and crucibles have been measured as high as __ rad/h.
    a. 20
    b. 30
    c. 50
    d. 80
6. Internal doses are not directly measured, but are estimated or calculated based on knowledge of the material to which a worker may be exposed and it's known or assumed biokinetic behavior.
    a. True
    b. False
7. Using Table 4-1. 10 CFR § 835 Appendix D Surface Contamination Values (dpm/100cm^2) which of the following radionuclides corresponds to a removable surface contamination value for 200 dpm/100cm$^{2?}$?
    a. U-nat
    b. Transuranics
    c. Th-nat
    d. Tritium and STCs
8. Workers who are considered likely to have intakes resulting in excess of 0.1 rem CED are required to participate in a bioassay program. The workers at highest risk of incurring an intake are the ones in closest contact with the material.
    a. True
    b. False
9. Fecal analysis is often more likely to detect exposure to highly insoluble Type S material than urinalysis. The ratio between the fecal excretion level per day and the urine excretion level per day is greater than __, as calculated for a 90-day sampling interval.
    a. 8
    b. 7
    c. 6
    d. 5
10. For natural and enriched uranium, the energy most commonly used for in vivo monitoring is the ___-keV gamma that is emitted with 54% abundance from the decay of ^{235}U.
    a. 59
    b. 88
    c. 93.8
    d. 185

FOREWORD

This Technical Standard discusses, but does not establish any, requirements for DOE uranium facilities. Its purpose is to provide information that will assist DOE and DOE-contractor health and safety professionals in developing programs that will provide an appropriate level of protection to both affected workers and members of the public affected by DOE uranium-handling activities. This Technical Standard provides guides to good practice, updates existing reference material, and discusses practical lessons learned relevant to the safe handling, processing, and storage of uranium. The technical rationale for the guidance provided herein is explained to allow affected individuals to adapt the recommendations to similar situations throughout the DOE complex. This Technical Standard provides information to assist uranium facilities in complying with Title 10 of the Code of Federal Regulations, Part 835 (10 CFR Part 835), *Occupational Radiation Protection* and various DOE Orders. This technical standard supplements DOE G 441.1-1C, *Radiation Protection Programs Guide for Use with Title 10, Code of Federal Regulations, Part 835*, *Occupational Radiation Protection* (DOE, 2008a) and DOE-STD-1098-2008, *Radiological Control* (DOE, 2009c).

This Technical Standard has been updated to include provisions in the 2007 amendment to 10 CFR Part 835. This amendment updated the dosimetric terms and models for assessing radiation doses, both internal and external. Of particular interest for this Standard, the biological transportability of material is now classified in terms of absorption types: F (fast), M (medium) and S (slow). Previously this was classified in terms of material class: D (days), W (weeks) and Y (years). Throughout this Standard, discussions of previous studies describing the biological transportation of material in the body will continue to use D, W and Y, as appropriate. Discussions of other requirements which have not amended their dosimetric terms and models continue to use the older terminology.

This Technical Standard does not include every requirement applicable to DOE uranium facilities. Individuals responsible for developing and implementing radiation protection programs at uranium facilities should be knowledgeable of the requirements that apply to their facilities.

4 CONTAMINATION CONTROL

Contamination control is an important part of the overall radiological control program. There are four main aspects to this: 1) control of the release of contamination into the work-place environment; 2) control of personnel exposure to the contamination that does get into the work place; 3) protection of personnel from intake of contaminants and 4) prevention of release of contamination to the public and the environment. Effective control of personnel exposure to uranium and its decay products is accomplished mainly by controlling the potential for inhalation and ingestion of radioactive materials. Monitoring provides an indication of the effectiveness of physical design features and administrative controls in controlling exposure to radioactive material.

This chapter addresses the basic features of an effective contamination control program and the technical considerations of implementing the program. A release of radioactive material from containment typically results in surface contamination and airborne dispersion. Airborne contaminants are continuously cleared from the work place by ventilation. Strategic air sampling detects the release of an airborne contaminant and provides the means for control, minimization of personnel exposure, and evaluation of inhalation exposure. Considerations for design of an air monitoring program are followed in this chapter by a section on surface contamination control. Finally, protection of personnel from contaminant intake is accomplished with protective clothing and respiratory protection.

4.1 Air Monitoring

The most common route of uranium intake for workers is by inhalation. Airborne particles deposit throughout the respiratory tract. Some of the deposited particles are swallowed, contributing to ingestion, requiring that both inhalation and ingestion be considered with an exposure to airborne material. The particle size distribution that determines deposition in the respiratory tract is affected by the mechanism of dispersion and the nature of the source material. Characterization of inhalation exposure should make use of all available information about the chemical and physical form of airborne material. This information, along with spatial and temporal distribution, provides the basis to minimize personnel exposure for air contamination control.

4.1.1 Internal Versus External Dose Philosophy

The widespread application of methods to contain uranium in DOE facilities has resulted in a history of relatively minor internal exposures. The methods used to control internal dose have been developed for a variety of reasons:

a. The assessment of internal dose requiring bioassay is difficult, imprecise, time-consuming, and offensive to personnel as compared to external dosimetry. For example, an accidental internal uptake may require the subject to submit dozens of biological

samples over the span of many months, as well as requiring extensive analytical support for measurement of sample content, considerable time of trained professionals to analyze data and calculate the internal dose, and long lapses before dose estimates are available, thus handicapping the assessment of the occupational exposure status and treatment of the worker.

b. Prevention of internal exposure is often more feasible and successful than prevention of external exposure. Contained radioactive material may continue to produce external penetrating fields of radiation, but no internal exposure potential. Portable protective devices (respiratory protection equipment) can minimize internal exposure when containment is not practical.

c. Recommendations of the ICRP in formulating a dose limit system have resulted in combining internal and external dose. Again, the difficulty and time delay of internal dosimetry make elimination of significant internal exposure an economic incentive.

In facilities that process large quantities of uranium, however, there may be situations in which exposure to work-place airborne activity at low levels occurs daily. The fact that tons of material are handled, rather than gram quantities, and that the material is less toxic (on a mass basis because of low specific activity), make total containment impractical.

4.1.2 Purpose of Air Monitoring

The goal of the air monitoring program is to identify, evaluate, and control internal dose received by workers from routine occupational exposure to airborne radioactive materials, to confirm that source controls are functioning properly, and to assess the exposure resulting from an unusual event. There are two general aspects of air sampling that must receive equal consideration in a properly executed monitoring program. The first involves the methods and equipment by which a sample is collected and analyzed to yield an accurate measurement of the specific radionuclides. The second is the protocol of sampling location, duration, and frequency that focuses on determination of the radionuclide exposure in the work area.

Air monitoring should include both active and passive air samplers. A continuous air monitor (CAM) provides for immediate alarm, warning workers of an unusual release of high levels of airborne radioactive material. This active monitoring is needed for high hazard and high potential areas to provide immediate and timely protective response, while passive sampling provides high-sensitivity activity records, trends, continuous documentation, etc. Three types of air samplers are used to accomplish the air monitoring: general area sampling (GAS), breathing zone sampling (BZS), and personal air sampling (PAS).

The CAM continuously draws air through a sampler that has an active radiation detector. The sampled air is automatically monitored for an increase above normal or background levels of contamination. When airborne activity exceeds the alarm level, workers are warned of the potential problem and prompted to follow alarm procedures. This type of monitor is usually

practical only for stationary samplers (GAS or BZS). It is important that a CAM be placed to sample air that accurately represents the most likely area of material release. This will protect most workers from a worst-case exposure and minimize total work-force exposure.

4.1.2.1 General Air Samplers (GAS)

Air sampling is performed at a single point in the general area of a site where work with radioactive material is being performed. The sampler is placed in a position to give the best overall representation of the area, often in the main airflow exiting the area. Airflow patterns can be determined by tests with tracer smoke or balloons. This method is typically used to measure airborne radioactivity for the following purposes:

a. to determine if the work-place environments are free of significant contamination and are inherently safe for routine occupational activities

b. to detect measurable air activity which would signal the need for use of respiratory protection equipment

c. to detect unexpected loss of containment or malfunction of systems (which may not be detected by a CAM), and provide the basis to initiate corrective actions

d. to detect low-level trends in activity which can signal a gradual loss of containment in early stages

e. to estimate personnel exposure retrospectively and evaluate compliance with applicable requirements

4.1.2.2 Breathing Zone Samplers (BZS)

Breathing zone sampling is performed by placing air samplers in the immediate area in which workers will spend the majority of their time. The intent is to measure the air activity concentrations to which the workers are actually exposed. The purposes of breathing zone sampling are the same as those listed for general air sampling, but involve a greater number of samples, which gives more realistic information. Breathing zone samplers give earlier, more sensitive detection of release from containment.

Samples should be collected on a schedule corresponding to individual worker activities to best represent inhalation exposure. GAS is generally not a good measurement with which to estimate internal dose. A well-placed network of BZS gives a better representation of inhalation exposure.

4.1.2.3 Personal Air Samplers (PAS)

Personal air sampling should give the most realistic measurement of individual worker exposure. This involves greater expense, however, to equip personnel with samplers and to

process all of the individual samples. Personal air sampling is performed with a small, battery-powered, low-volume (approximately 2-L/min) sampler worn by the worker, with the filter located near the worker's face. This type of sampler is potentially subject to many inaccuracies caused by improper handling, which requires trained personnel to handle the equipment operation. Personal air sampling is often used to validate breathing zone sampling strategy and to conduct special investigations.

4.1.3 Regulations and Limits

The regulations, standards, and limits pertaining to exposure of radiation workers to airborne activity in the work place are based on the probability of injury to internal organs and the total body by radioactive materials taken into the body. To facilitate control of intake in the work place, standard-setting authorities have calculated derived air concentration (DAC) and annual limit on intake (ALI) as a control to limit resultant dose to internal organs. Operational hazards are directly controlled by the observance of DAC and ALI values.

The ICRP and the NCRP are independent, non-governmental organizations which set standards and guidance for control of radiation hazards. Governmental agencies implement these recommendations by establishing Federal policy for the protection of workers.

Formal rules for air monitoring for DOE facilities are provided in 10 CFR Part 835. Efforts have been made to keep these rules consistent with ICRP Publication 60 (1991a) and NCRP Report 91, *Recommendations on Limits for Exposure to Ionizing Radiation* (1987). DOE-STD-1098-2008 (2009c) detailed guidance on the best practices currently available in the area of radiological control. More specific guidance is given in DOE G 441.1C, Chapter 10 - *Air Monitoring,* and technical standards, such as this one.

Limits of chemical exposure also need to be monitored, especially for materials of low specific activity, such as depleted uranium or non-radioactive materials. The threshold limit value time-weighted average (TWA) for natural uranium is 0.2 mg m^{-3} (ACGIH, 2005). TWA is the chemical analog of DAC. In the case of reactor fuel uranium, enriched to about 3%, this corresponds to 4 x 10^{-10} µCi mL^{-1}, which is comparable to the DAC for soluble forms of uranium. However, the OSHA Permissible Exposure Limit for soluble uranium is 0.05 mg m^{-3}, which is more restrictive than the DAC. Soluble forms of such materials can be monitored directly by routine urinalysis, or indirectly by BZS and PAS. Internal deposits of insoluble forms may only be estimated by BZS and PAS, as with asbestos, for example.

4.1.4 Theoretical Considerations and Uncertainties

A discussion of the theoretical aspects of air contamination monitoring, and inherent uncertainties, should be useful in placing air monitoring programs in their proper perspective. In general, air sampling should not be the primary measurement of internal dose, except when bioassay information is unavailable or unobtainable. Evaluation of worker exposure

potential in terms of DAC-hours, however, may be a legitimate control measure and may demonstrate compliance with federal directives.

4.1.4.1 Airborne Concentration

An appropriate air-sampling method should provide samples which accurately represent the average airborne concentration of radioactive materials present in the work place, but should not be used as a measurement of individual exposures, except in unusual circumstances. If air activity data must be used for exposure records, these samples should be collected from the breathing zones of the workers, or by using an established conversion factor for the existing sampler configuration. In contaminated areas subject to significant temporal and spatial variations in the activity concentrations, only personal air samples or virtually continuous samples collected from within the breathing zone of workers can provide reliable breathing zone concentration measurements.

A restricted area, having good ventilation and point sources of contamination, will have substantial variations in the activity concentrations observed at different locations, particularly if the movements of the workers cause resuspension of the activity. The worker often spends time closer to the source of contaminant dispersion than is the location of the nearest BZS. Several researchers have investigated the relationship between fixed air samplers and spot samples collected at various locations in typical working areas. Discrepancies as great as two orders of magnitude are not unusual.

This deficiency of GAS monitoring for individual exposure records is caused by the high dilution factors that tend to reduce the airborne concentrations before and after contamination reaches the filter head. Much of the air sampled by a GAS originates in another part of the area and does not pass near enough to pick up contamination from the source, effectively reducing the measured concentration by dilution of the collected sample. A release of activity from a malfunctioning containment system can produce large activity concentrations in the breathing zone of the worker. These concentrations are effectively diluted in an unpredictable manner by one or two orders of magnitude before the contamination reaches a monitor located only a meter away. It has been demonstrated that in some operations (such as welding over a short period of time) differences of as much as a factor of 5 between the right and left lapel PAS measurements can be expected.

Most of the field studies that have compared urinalysis results with air sampling in natural uranium facilities have, in general, indicated very poor correlation between the estimated exposures and the bioassay data. This suggests that individual exposure records of uranium workers based on GAS methods have limited validity.

The potential for release of gaseous UF_6, and subsequent generation of its soluble hydrolysis product UO_2F_2, requires special air-sampling considerations in uranium conversion and gaseous diffusion plants, relative to those plants handling less reactive compounds. In these

plants, effective processing, as well as worker safety, requires a high degree of containment. Continuous GAS operation to detect loss of containment, coupled with spot air samples, constitute the typical sampling strategy. A study conducted at the Oak Ridge Gaseous Diffusion Plant, concluded that shift-long air samples collected in the general working areas were of little use in predicting worker's urinary uranium excretion. The slight correlation observed was not statistically significant at the 95% confidence level. Thus, gaseous contaminants behave much like particulate contaminants in that localized concentrations can be much greater than the average concentration measured by GAS. These researchers also found that smear samples of alpha activity on work surfaces in the area may provide a better indicator of uranium intake than the GAS records.

Although transuranic material is handled by DOE uranium facilities only as feed contamination, the unusual characteristics of the transuranic elements make them worthy of separate consideration. The low maximum permissible concentrations specified for these elements and their frequently low specific activities cause extreme difficulties in detection of significant airborne activity. Operations involving significant amounts of elements, such as ^{239}Pu, should be conducted in a ventilated glove-box environment and with monitoring systems capable of detection of small releases involving 1 DAC averaged over 8 hours (8 DAC-hours) under laboratory conditions. Special CAMs (GAS) and fixed BZSs are the standard air-sampling methods used in facilities of this category in the United States.

A clear example of the wide variations in observed air-activity concentrations that can occur with different sampling techniques is provided by data from the Three Mile Island Nuclear Generating Station, which is typical of operations in a large open building (EGG 1988). Between June and September 1983, over 40 multi-person entries were made into the containment building, providing 949 work-hours of PAS data. Five stationary air monitors were operated continuously at strategic locations throughout the building, and each entry was preceded by the collection and analysis of a high-volume grab sample. All samples were analyzed by a gamma spectrometer, primarily to detect cesium-137, and by gross beta counting.

The five continuous air samples exhibited good internal agreement when averaged over either 12- or 24-hour periods. However, the grab samples averaged a factor of 3 higher than the continuous air-sampler readings, and PAS samples were a factor of 34 higher. The major reason for this large difference was attributed to resuspension of the surface contamination by the work in progress. These data, coming from a thoroughly monitored and carefully analyzed air-sampling effort, are further evidence that GAS methods should be viewed with caution.

Even when the airborne-activity concentration in the breathing zone of a worker has been accurately measured, there are other physical and physiological parameters that can produce significant uncertainties in dose assessment. The established DACs are derived for each

radionuclide assuming a standard volume of air breathed in occupational situations, specified pathways to critical organs, the "standard man" metabolic and elimination patterns, and the physical and biological properties of the isotope. Large variations are encountered, however, in breathing rates and tidal volumes (which depend on working conditions), and there are individual variations in such physiological parameters as lung clearance and metabolic rates. The particle-size distribution of the aerosol and the actual solubility of the inhaled particles can significantly affect the deposition and retention of airborne activity in the respiratory tract. The potential uncertainty in the total dose assessment should include consideration of all of these factors, as discussed in the following paragraphs.

4.1.4.2 Particle-Size Distribution

In the absence of actual measurement of particle-size distributions, an activity median aerodynamic diameter (AMAD) of 5 μm and a geometric standard deviation (GSD) of 2 is often assumed as a conservative estimate, as laid out in the ICRP Publication 60 (1991a) methodology. Particles of this size are likely to result in the greatest deposition in the pulmonary region of the lungs. The actual size distribution can be measured with instruments such as cascade impactors, but these are not practical for continuous operation in the work-place environment. Electronic instruments can give continuous information about the optical particle size, but not the AMAD. Thus, particle size can only occasionally be measured to typify the size distribution in a particular situation.

Size-selective inlets for air samplers have been developed to mimic deposition in the respiratory tract, giving more accurate estimates of deposition in the pulmonary region. Non-respirable or non-inhalable particles are removed by the inlet, and the respirable or inhalable fraction is collected on a filter. These devices can be useful in minimizing the dose assessment errors resulting from uncertainties regarding the actual aerosol-size distribution; however, they require additional handling and care, and require separate samplers for total airborne activity. If the AMAD is often substantially greater than 1 μm in an area, the addition of samplers with size-selective inlets may be worthwhile. Regulations allowing the substitution of size-selective samplers are not established, however, so special arrangements may be needed with regulatory agencies.

4.1.4.3 Breathing Rates and Tidal Volumes

The actual air intake of a worker can vary from 5 L min^{-1} to 100 L min^{-1}, although typical variations from the assumed 20 L min^{-1} standard will probably be no larger than a factor of 3. Total air intake depends on the rate of breathing and on the volume of tidal air. The velocity of this air influences the regional deposition of aerosol particles. Newer, more sophisticated lung models include this breathing-rate effect in calculation of dose distribution. Information about individual breathing behavior may be useful in the application of the newer lung dosimetry models. Simpler models, such as ICRP Publication 30 (1979), assume that regional deposition

is independent of breathing rate, with total deposition determined only by the volume breathed.

4.1.4.4 Particle Solubility and Lung Clearance

When particles are deposited in the respiratory tract, they are cleared from airway surfaces by several mechanisms. Insoluble particles are cleared by the biomechanical means of macrophage and mucociliary transport, while some particles are retained in pulmonary tissues. Particles of soluble material dissolve, making the contaminant available for other means of transport such as absorption into the blood. Dosimetry of the contaminant depends on how fast the particles dissolve.

Rate of particle dissolution is divided into three categories by the ICRP Publication 68 model (1994b). Types F (fast), M (moderate), and S (slow) refer to the rate of absorption of the material in the pulmonary region of the lungs. The approximate half-times of clearance that these absorption rates correspond to are

- Type F (fast): 10 min (100%)
- Type M (moderate): 10 min (10%); 140 d (90%)
- Type S (slow): 10 min (0.1%); 7000 d (99.9%)

A retention half-time of less than 10 days is retention Type F, a half-time of 10 to 100 days is class M, and half-time greater than 100 days is Type S. Some materials have been described to have characteristic rates of dissolution and are associated with a particular retention class. Many factors can affect the dissolution rate, however, so general assignments to retention classes should be regarded with caution.

The health physicist may have some prior knowledge of the chemical compounds of the nuclides present in an area and may be able to assign them to retention classes. The ICRP Publication 60 (1991a) dosimetry model provides for a lung retention class designation of aerosols depending on the rate of dissolution; however, actual determination of the lung class for dose assessment can best be determined after an exposure utilizing appropriate chemical and bioassay data, but this can only be accomplished in retrospect (ICRP 1991). A prospective approach uses measured dissolution rate of potential contaminants for analysis and treatment of an accidental exposure. Determination of retention class should be a valuable precaution in uranium facilities.

A realistic determination of retention class can be made by collecting a sample of airborne material by using a size-selective sampler and drawing the sample from a process that has a potential for a significant release. The material collected on the filter represents that which would be deposited in the lungs by inhalation. Methods and instruments are now available with the sensitivity needed to precisely measure the rate of dissolution of this small mass of

uranium in simulated lung fluid. The same methods can be used on filter samples in operation at the time of an accidental exposure, but the time required to measure dissolution rate (at least 60 days) makes the information essentially retrospective. Prospective measurement of retention class provides for better risk assessment.

4.1.5 Samplers and Instrumentation

Air sampling equipment and monitors exist in a wide range of designs and capabilities, with characteristics specific to the application and need. Samplers range from small portable units that can be worn by an individual to high-volume units permanently mounted in the facility. Flow rates are from a few liters per minute to a few cubic meters per minute.

4.1.5.1 Key Factors in Selecting Air Samplers

Sensitivity of Detection. In general, the sensitivity required is at least DAC levels; however, in some applications, sensitivity to a small fraction of DAC is desired for early detection of loss of containment, low level trends, etc. Continuous air monitors may only need to alarm at multiple DAC levels in order to be effective in preventing or mitigating personnel exposures to an accidental airborne release.

Type of Sample. In most uranium facilities, particulates in the air are the primary concern, although gaseous forms may be most important in some areas. It may be of interest to collect samples that will allow characterization of the particle size distribution or define a "respirable fraction." In each application, the sample type will dictate the sampler design, filter media, flow rate, etc.

Convenience. Available space, noise level tolerance, portability, and weight also dictate specific designs and capabilities of air samplers and monitors.

Power Requirements. Requirements for battery-powered versus 110-120-VAC line power may dictate sampler selection.

Accuracy. Some sampling is performed to simply detect or make relative measurements of activity levels for which the accuracy requirements are not great. In other situations, accurate measurements of the air breathed by personnel may require an entirely different sampler design to achieve the needed quality assurance.

Reliability and Maintainability. Cost-effective operation and reliability need to be considered for selection of equipment design and for redundancy of components. Sensitivity to Electromagnetic fields should also be considered.

4.1.5.2 Filter Media

Filters should have high collection efficiencies (i.e., >99%) for particles over a wide range of sizes. Many cellulose ester (acetate, nitrate, or mixed ester) or glass-fiber filters meet these requirements and are commonly available. Other filters with reasonably high collection efficiency may be used if required for special applications or assay methods. Selection of a filter type generally involves compromises between filter efficiency, flow resistance, and requirements imposed by the desired assay method.

The specifications of a filter medium often include pore size and filter efficiency. Pore size is determined by filtration of a liquid; the particle size at which the collection efficiency is 95% in water is given as the effective pore size. Filtration efficiency for particles in air, however, is dramatically different. Aerodynamic effects make the collection efficiency dependent on the face velocity through the filter. Airborne particles of aerodynamic size equal to the pore-size rating of a filter are usually collected with high efficiency (>99%). Smaller particles may also be collected efficiently; however, some sizes may substantially penetrate the filter. Particles in the range 0.1- to 1.0 micron diameter are most likely to penetrate a filter. Many manufacturers use dioctylphthalate (DOP) to produce an aerosol of particles 0.3 micron in diameter for testing filter efficiency. Thus, if a filter is rated for efficiency by DOP retention, collection of other particle sizes will be more efficient. Collection efficiency is also increased by higher flow rate for particles >0.1 micron.

Cellulose ester membrane filters have interconnecting pores of uniform size. They typically produce a higher resistance to flow than glass-fiber filters and collect most particles near the surface of the filter.

Glass-fiber filters are made of a mat of randomly oriented glass fibers. They have lower flow resistance than most membrane filters, but trap an appreciable fraction of the particles within the filter mat. This interferes with detection of alpha radiation from the filter.

Cellulose filters are often used for air sampling. They have moderate flow resistance, but relatively poor collection efficiency. Their use may be justified in some situations, but only with the recognition that efficiency for certain particle sizes may be low. Generally, if analytical and sample-handling requirements allow, glass-fiber or cellulose-ester membrane filters are a better choice than cellulose filters.

Each type of filter has inherent advantages and disadvantages. The higher flow resistance of membrane filters may overtax the capabilities of older models of some PAS pumps although membrane filters can be used successfully with many of the new models of pumps. Glass-fiber filters should be substituted if a significant pressure drop occurs with the sampler being utilized.

The surface-collection properties of membrane filters can be an advantage when sampling for alpha and weak beta-emitting materials. Deposition of particles on the surface minimizes energy absorption by the filter medium. This is especially important for alpha spectrometry, where the energy spectrum is substantially degraded. Membrane filters are also advantageous if the assay procedure involves ashing or dissolution of the filters, but they are relatively fragile.

4.1.5.3 Filter Holders

Criteria for filter holders are simple, but critical. For the collection of large-volume air samples, filter holders should be open-face such that sample air is drawn directly onto the filter surface from the atmosphere without passing through a tube, orifice, or other obstruction. This precludes loss of the radionuclide to surfaces upstream from the filter. The holder should face downward to avoid collection of large, non-inhalable particles, unless a different position is required. Closed-face cassettes are recommended for small PAS, to protect the filter from direct contamination. Research studies of commonly-available types of closed-face cassettes with 4-mm inlets indicate that these designs have good particle-collection characteristics (at a flow rate of 2 L min^{-1}) and reduce sample contamination problems. Other closed-face filter inlet diameters, geometries, and flow rates may also be acceptable, but have not been characterized.

The filter should receive adequate support so that it is not stretched or torn by the pressure drop caused by the flow of sample air. The filter holder should be free of air leakage around the filter as well as into or through the holder's component parts. Metallic filter holders are generally more reliable and durable than plastics. Finally, filter-changing and holder replacement should be convenient and positive.

4.1.5.4 Size-Selective Devices

Size-selective devices fall into two categories: respirable-fraction samplers and instruments for measuring particle-size distributions. A respirable-fraction sampler collects a range of particle sizes, with collection efficiency decreasing for larger particle sizes. Particles that penetrate the size-selector represent those that would deposit in the pulmonary region of the lungs. A particle-size distribution instrument collects all particles with classification of particle size. Size-distribution data can be used to calculate the expected deposition of particles throughout the respiratory tract.

Particle-Sizing Devices. Particle-size distribution measuring devices are typically more complex and require more sample analysis than a size-selective sampler. The major advantage in using these devices is that the size distribution of airborne contaminants is useful for estimating regional deposition of inhaled particles in the respiratory tract. This information is more accurate than that provided by a simple size-selective sampler, especially if a large part of the airborne material has particle size less than about 2 μm. Particle-size measurement

should be performed only by properly trained individuals, as an investigative tool for evaluating the health hazard posed by a process or procedure suspected of generating airborne contamination.

The cascade impactor is the most commonly available particle-sizing device. Aerosol passing through a cascade impactor is forced through a series of increasingly rapid changes of velocity. The inertia of the particles causes them to deviate from the direction of the airstream at locations where the particle speed and direction are changing most rapidly. Particles of different aerodynamic size deflect to different extents so that larger particles contact the surface of the collection stage. The quantity of material deposited on each stage is measured and the size distribution calculated for the sampled aerosol.

There are some drawbacks to the use of impactors. Cascade impactors subdivide the sample so that more sensitive assay methods may be required for successful use. There is a limit to the mass of material that can be collected on each stage before overloading; inactive dust particles contribute to this mass, but not to the analyte. Each stage of the impactor is a separate fraction of the sample which must be analyzed. This multiplies sample number-capacity requirements of the activity measurement system. Careful calibration of a precisely controlled airflow rate is required for accurate particle-size measurement.

Optical particle-sizing instruments, such as a laser particle-size spectrometer, have the advantage of giving practically real time information. Most of these instruments give only an optical particle size, however, which must be converted to an aerodynamic size to be useful for dose estimation. They are generally expensive tools used mostly for research.

Respirable-Fraction Samplers. A number of respirable-fraction samplers have been developed, but the cyclone separator is the most widely used and best characterized type. The cyclone is specified by NIOSH and Mine Safety and Health Administration (MSHA) for personal respirable-mass sampling in coal mines. NIOSH and MSHA currently certify entire sampling systems (PAS pump, cyclone, filter head, and filters) for personal respirable-fraction sampling. This "system" approach may be modified as the result of recent research; however, it does provide an interim standard for performance. The performance of cyclones, pumps, and filters may be characterized to allow intermixing of sampling-train components in future work; at present, however, theoretical prediction of performance of mixed systems is not reliable.

Cyclones are aerodynamic particle sizers, as are impactors, but have some different operating features. They are not affected by loading, so dusty environments are not a problem, although filter loading may limit sampling time. Cyclones are rated for performance at a particular flow rate. Performance at other flow rates cannot easily be predicted and should be determined by testing. In contrast, impactors do follow a simple, well-defined relation between flow rate and size separation.

Alternatives to mechanical methods of particle-sizing exist and other respirable-fraction separators may be available in the future. Combined total and respirable-fraction samplers would be desirable; such designs retain both the respirable and non-respirable fractions so that total airborne activity can be estimated.

4.1.6 Sample Activity Measurement

Most sample analyses at uranium facilities are performed by quantifying the radioactivity by counting the samples collected. Some fluorometric analyses are performed with equivalent sensitivity. Kinetic phosphorescence analysis is available with substantially greater sensitivity.

Alpha Counting. Alpha particles can be counted with ionization, proportional, scintillation, or other solid state detectors. The major drawback is that relatively little particle penetration, in the filter or in the dust loading, can result in a low reading caused by self-absorption of the alpha particles.

Alpha Spectrometry. Measurement of the energy spectrum of alpha-emitters on a filter paper is possible and very beneficial in some applications in identifying or verifying the identity of the isotopes present. Typically, semiconductor detectors are the choice, and membrane filters or other surface-collecting filters are used with very low dust loading.

Beta Counting. Thin-window GM, ionization, proportional, and solid state detectors are used for beta counting. Because of the wide range of beta-particle energies of even a "single energy" emitter, careful energy calibration is necessary. Beta counting results are less dependent on self-absorption effects.

Beta Spectrometry. Beta spectrometry has recently become feasible through developments in tissue-equivalent plastic detectors. For routine isotopic identification, this method is not as useful, but it may provide valuable shielding information, etc.

Gamma Spectrometry. Sodium Iodide (NaI) and High Purity Germanium (HPGe) detectors can provide essential isotopic identification of gamma-emitters.

Precautions. The intricacies and procedures of sample analysis are beyond the scope of this manual. However, a few general precautions are important to mention. The naturally occurring radionuclides, radon and thoron and their decay products, are present in all atmospheres in widely varying concentrations. These radionuclides are typically present in higher concentrations than the isotopes of interest, and tend to interfere with radiometric analysis, unless the short-lived progeny are given time to decay after sample collection. Radon progeny, which are much more abundant than thoron progeny in most areas, decay with an effective half-life of about 30 minutes and a counting delay of 3 hours may be adequate. Thoron progeny decay with an effective half-life of 10.6 hours, and where they exist in

significant concentrations, a counting delay of several days is advisable. The presence of either radionuclide on a filter can be detected by recounting two or three times at intervals of a few hours.

The sensitivity of any counting method depends primarily on the background count rate of the counting instrument; estimates of low radionuclide concentrations can be seriously in error if the counting background is not accurately known. Even in stable instruments for which the background count may be quite constant, a daily check is advisable because of the possibility of contamination from sample material. Background counts should be made with a blank filter in place because some filter media contain trace amounts of radioactivity.

Counting instruments also require periodic standardization. Standard sources used for this purpose should match the samples both in size and energy.

The active (upstream) sides of filters collected in clean atmospheres can be difficult to identify. Some convention should be followed by sampling personnel to ensure that the proper sides of filters will be counted. This may consist of marking one side of the filter or placing the filter in the sample holder consistently with the exposed side toward the identifying number or label on the holder.

4.1.7 Continuous Air Monitors (CAM)

The combination of an air sampler and an activity counter into a single device for automatic operation and alarm control constitutes a CAM. Modern CAMs include the ability to automatically change the filter media, perform spectral analysis for isotopic identification, and to distinguish radon/radon progeny. They can evaluate the airborne contamination levels against several DACs simultaneously. CAMs have local visual and audible alarms and most have the ability to provide remote alarm signals to a control room. Some CAMs have the ability to use a remote head which allows for the sampling head to be close to the workers breathing zone. Other CAMs use stack monitoring attachments to sample stack airflows. CAMs need to be calibrated periodically and routinely source checked to verify their operability.

4.1.8 Monitoring Strategies and Protocols

Designing an air-sampling program for the work place is a complex task because each facility has unique design and operational characteristics. It is important that the radiological control personnel who coordinate the sampling program have a thorough understanding of basic facility operations, especially with respect to the potential each operation has for generating airborne material. In addition, these personnel should be familiar with the working habits of potentially-exposed workers. The success of most sampling programs depends on the ability of radiological control personnel to accurately assess worker exposure risk and properly select

workers for personal air sampling. This can only be accomplished by well-trained, observant safety personnel.

The following questions should be considered for an airborne activity hazard evaluation:

a. Where are the potential aerosol generation and release locations in the work-site, and what is the magnitude of potential exposures associated with each?

b. How effective or failure-prone are the physical and procedural barriers that protect the worker from airborne radioactive material generated at these locations?

4.1.8.1 Potential Sources of Airborne Contamination

Virtually every work site has at least one of the fundamental mechanisms for the generation and suspension of particulate material. The following descriptions of some of the basic mechanisms of aerosol generation are intended to help radiation safety personnel recognize processes which have inherently higher risk:

a. Mechanical fragmentation, i.e., grinding, abrasive saws, sandblasting

b. Combustion, burning materials producing smoke, fumes, etc.

c. Heating - many materials produce aerosols when heated, without actually igniting

d. Formation from bubbles, foams, or highly agitated liquids - fine solid aerosol particles can form from larger, evaporating liquid droplets

e. Condensation of liquid or solid particles from the gas phase

f. Formation of particles from the products of gas-phase reactions, e.g., $UF_6 + 2\ H_2O \rightarrow UO_2F_2 + 4\ HF$

g. Formation of solid, radioactive nuclides from gaseous parent nuclides - these radionuclides usually attach to existing, nonradioactive aerosol particles

h. Adsorption of gaseous, radioactive nuclides on non-radioactive aerosols

The program designer should be familiar with the routines and working habits of workers, especially those in situations where there is a greater potential for generating locally high concentrations of airborne contamination. This will assist in planning for exposure prevention and in selecting suitable sampling methods. Some factors to consider are the following:

a. Worker location and mobility - If the worker stays in a fixed location, fixed breathing-zone sampling may be useful for individual exposure estimation. This sampling may be performed using moderate flow-rate pumps (30 to 90 L min^{-1}) which can be located within a few feet of the worker. Mobile workers should be surveyed using PAS to obtain a breathing-zone sample.

b. Direct versus remote handling of radioactive material - Remote-handling facilities such as hot cells or caves usually restrict the workers to a fixed location. Well-located fixed sampling heads may be adequate for breathing-zone sampling at these work areas, provided that they have been properly located. As previously noted in this section, determining the proper sampling points for fixed breathing-zone sampling at fume hoods, glove boxes, etc., is not a straightforward exercise, and PAS may be the most expedient means for sampling a worker's true breathing zone.

   Direct-handling is commonly performed on material with relatively low intrinsic hazard, e.g., uranium metal or compounds. This sort of material may be moved around the work site and directly manipulated at a number of locations. Fixed breathing-zone samplers usually will be unsatisfactory in these situations, and PASs would be required for estimating an individual worker's exposure in DAC-hours.

c. Material with high intrinsic hazard is usually well contained, but if it is moved over wide areas in process flows, there is a potential for release at any point. The effectiveness of containment, in the process flow at locations where workers have access, is a major factor when considering use of PASs.

When evaluating risks associated with direct handling of radioactive materials, the variation in techniques employed by different workers to perform the same task must also be considered. No two workers perform the same operation in exactly the same manner. Aerosol production may depend on how each individual performs the operation (e.g., rate, accuracy, operating temperatures).

4.1.8.2 Characterization of Controls

For the purpose of evaluating work-place controls, work sites can be characterized as either "tightly controlled" or "loosely controlled." Tightly controlled work areas are preferred in all cases, but there are situations where good control is difficult or not reasonably achievable. PAS monitoring can help define those operations that pose the greatest radiological control problems and thus facilitate decisions to improve specific work situations.

Significant exposure incidents in highly controlled (i.e., tightly controlled) areas usually are the result of isolated and unforeseeable events, which are complete departures from the normal material-processing routine. These events usually include loss of containment. In tightly controlled areas, PASs can serve as a means of detecting a failure of containment because

work locations may be located near potential release points, and inadequate physical controls may be apparent only during an operation performed by a worker.

4.2 Surface Contamination Control

Uranium contamination on plant surfaces, such as floors and walls, does not present a significant risk to personnel unless the uranium becomes airborne by resuspension and is inhaled. The probability of significant airborne concentrations resulting from resuspension of uranium as a result of normal activities (such as walking) is low; however, any activity that vigorously disturbs the surface (such as floor sweeping) increases the probability of significant airborne concentrations of uranium. Resuspension is a function of both the chemical and physical forms of the uranium contamination. External exposure hazards from surface contamination can become an important concern when uranium decay products and/or fission products accumulate on surfaces. In some instances, efforts to decontaminate uranium compounds may leave behind insoluble uranium and decay product compounds which could present an external exposure hazard. Good industrial housekeeping practices and normal standards of personal hygiene will usually ensure that uranium surface contamination does not present a significant exposure hazard. However, even if the probability of resuspension is low, surface contamination on floors can result in contamination of shoes and thereby result in the potential for tracking of contamination into uncontrolled areas. Thus, contamination on surfaces must also be adequately controlled to prevent transfer of contamination above acceptable levels.

Several other contamination control objectives can be accomplished by a program of monitoring and control of surface contamination:

a. The program can be designed to provide information to detect containment failures or departures from good operating practices.

b. It can provide information that will assist in the design and evaluation of personnel monitoring, bioassay, and air monitoring programs.

c. The contamination monitoring and control program will provide information to establish operating zones, guidelines and constraints for radiation protection, and operational procedures.

d. The program will provide practical assurance that uranium contamination is confined to the operating areas of the plant and that the potential is minimized for contamination of personnel, the environment, and sensitive analytical areas.

Contamination control of work surfaces such as tools, equipment to be worked on (e.g., disassembly, machining), desks or tables in process areas is of greater concern than

contamination on floors. The likelihood of personnel contamination, ingestion of material through hand contamination, or inhalation of resuspended uranium compounds through work activities represents a significant potential for exposure of personnel. Work activities that involve the destruction of surfaces such as grinding, machining, drilling, or cutting can generate significant levels of airborne uranium compounds. Operations such as welding, burning, heating, etc. can alter the physical and/or chemical state of uranium compounds that are on the surfaces of equipment. Job-specific monitoring is required to establish protection requirements as a function of surface contamination levels.

4.2.1 Reporting and Documenting Contamination Levels

Radiological control programs require the performance of contamination surveys to determine existing conditions in a given location. Maps with sufficient detail to permit identification of original survey locations should be maintained. Records shall contain sufficient detail to be meaningful even after the originator is no longer available. Contamination surveys should be recorded on appropriate standard forms and include the following common elements:

a. Date, time, and purpose of the survey

b. General and specific location of the survey

c. Name and signature of the surveyor and analyst

d. Pertinent information needed to interpret the survey results

e. Reference to a specific radiological work permit if the survey is performed to support the permit

In addition, records of contamination surveys should include, at a minimum, the following information:

a. Model and serial number or other unique identifier of counting equipment

b. Contamination levels (using appropriate units) and appropriate supporting parameters, including counting efficiency, counting time, correction factors, type of radiation, and whether the contamination was fixed or removable

c. Location of areas found to contain hot particles or high concentrations of localized contamination

d. Follow-up survey results for decontamination processes cross-referenced to the original survey

Records for the release of material and equipment from radiological areas to controlled areas should describe the property, the date on which the release survey was performed, the identity of the individual who performed the survey, the type and identification number of the survey instrument used, and the results of the survey. Additional details on radiation records can be obtained from DOE G 441.1-1C (2008a) and DOE-STD-1098-2008 (2009c).

All skin and personal property contaminations should be documented and evaluated to help improve the contamination control program. Documentation should include the following:

a. The person's name and work group

b. The location, amount, and type of skin or personal property contamination

c. The results of decontamination

d. A description of circumstances involved in the occurrence, such as radiation work permit number, protective clothing required, and protective clothing actually used

4.2.2 Monitoring

Radiological workers are often assigned tasks that could expose them to radioactive material. It is not sufficient to rely exclusively on equipment design to minimize contamination and exposure in the work place. A radiation protection program shall include both monitoring of the workers (discussed in Section 4.3) and monitoring of the conditions in the workplace (10 CFR § 835.401 - § 835.403, § 835.1101- § 835.1102). Both functions are essential to a good radiation monitoring program.

Continuous monitoring should be provided during the periods of high or unusual risk associated with the work in the area. Periods of high or unusual risk include the potential or actual breaching of the integrity of the glove-box or associated systems, including such maintenance as replacement of panels, glove changes, bag-out operations, replacement of filters, or repair of vacuum systems. Work that involves the use of temporary enclosures (greenhouses) should be provided with continuous coverage by an RCT. For decommissioning, most activities will be new, unique, and have no historical precedent. Consequently, high and unusual risks may become the norm and the use of temporary controls and continuous coverage the routine.

Monitoring of the work place is an essential element of every routine surveillance program. It can be effectively accomplished using any or all of the techniques that are discussed in this section. The rigor with which all of the various elements of a radiation monitoring program are applied should be tailored to meet the needs of the individual work areas and depend on the

kind and quantity of radioactive material present and its potential for dispersion. Each program should be designed to meet existing needs, but also should be flexible to allow for incorporation of the possible advantages to be provided by the various available monitoring practices. Monitoring practices include, but are not limited, to the following:

a. Contamination surveys of the workplace

b. Release surveys

c. External exposure surveys

d. Airborne contamination surveys

e. Routine surveillance by an RCT

4.2.2.1 Contamination Surveys of the Workplace

The radiation monitoring program should include documented survey procedures, a system for maintaining survey results, and contamination control limits for "fixed" and "removable" contamination. The results of contamination surveys should be reported in activity per area (e.g., dpm/100 cm^2) except for large-area swipes and swipes of very small items. This permits interpretation of the recorded data without requiring knowledge of instrument efficiency or geometry.

All workplaces should be monitored for contamination levels on a regularly scheduled basis. The frequency of such surveys will depend on the potential for dispersion of the radioactive material. As a minimum, all gloves, work surfaces, floors, equipment, etc., within the workplace should be surveyed according to the frequencies listed in DOE-STD-1098-2008 (2009c).

The change room and other support facilities within the controlled area should be surveyed for contamination daily. Continuous air monitors, survey instruments at step-off pads, and hand and shoe counters should be functionally tested daily or once per shift in support of the weekly and monthly surveys.

These frequent surveys are also part of the routine surveillance program and permit immediate follow-up if low-level contamination is detected to minimize the potential for major incidents. Some fixtures and support areas outside the controlled area, such as door knobs and telephones of adjacent offices and the lunchroom, should also be surveyed daily. Other support areas should be surveyed monthly. If routine survey results detect any contamination in a given area, more detailed surveys should be performed to determine the extent and source of the contamination.

Two principles should be adopted to preclude the possibility that contaminated waste would be disposed of as ordinary waste: 1) all process and controlled area waste should be considered contaminated, and 2) mechanisms should be established that prevent the mixing of contaminated and non-contaminated waste. In addition, mixing radioactive waste with RCRA-regulated hazardous waste should be avoided.

4.2.2.2 Release Surveys

As stated in Section 2.1.4.1., transuranics exist in small quantities of recycled or reclaimed feed materials. In many instances, these isotopes may be limiting for release of materials. For transuranic and uranium radionuclides, the contamination level (fixed and removable) at which surfaces are considered contaminated are listed in Appendix D of 10 CFR Part 835. That document also specifies the criteria for the release of materials and equipment from radiological areas to controlled areas.

Detailed requirements for unrestricted release of materials and equipment from controlled areas are found in DOE Order 458.1, *Radiation Protection of the Public and Environment* (2011d).

4.2.2.3 External Exposure Surveys

To delineate the levels involved, measurements of external exposure should be made at the time a program is established at all locations where personnel exposure occurs. Additional photon and neutron measurements should be made at the same frequency as the contamination surveys. The buildup of contamination in glove boxes and on gloves and equipment may contribute substantially to the external dose rates.

4.2.2.4 Measurement and Survey Techniques

This section discusses four types of contamination surveys that are typically used in DOE facilities. Surveys for removable contamination include a large-area wipe survey and a swipe or smear survey. Surveys for total/fixed contamination include a scan survey and a statistically-based survey. These surveys, or a combination of them, are used to survey material for release from radiological control. The appropriate use of each type of survey is discussed.

Surveys for Removable Contamination

Two types of surveys are used for removable contamination: a large-area wipe survey and a swipe or smear survey.

A large-area wipe survey is used to qualitatively detect gross removable contamination. A large-area wipe survey is typically performed using a large floor cloth and a dust mop type

handle to wipe large areas. This technique tends to concentrate any low levels of removable contamination that may be present. The surface to be wiped and the wiping material should be industrially clean (e.g., free of debris, grease) to reduce self-absorption of alpha contamination. The survey is performed by wiping the surface of the area being surveyed and conducting frequent checks of the cloth using a portable instrument. For detection of alpha-emitting isotopes, a nonabsorbent material should be used. Removable contamination will be accumulated and concentrated on the wipe, increasing the probability of its detection. Checking for contamination is conducted by placing an alpha-measurement instrument approximately 0.25 in. (0.6 cm) from the surface of the wipe for 5 seconds, and the count rate observed. If no radioactivity above background is measured, then the material is not contaminated with removable contamination. If radioactivity above background is measured, the material is contaminated. Technical smears (i.e. 100 cm^2) need to be taken to quantify removable contamination levels. Depending upon the specific circumstances, a series of smears may be required to locate and quantify the contamination within the area covered by the large-area wipe. In most instances, if contamination is detected on the large-area wipe, decontamination should be considered.

For transuranic radionuclides, the guideline values for removable contamination are lower than the minimum detectable activity (MDA) of portable instruments. During a wipe survey, the surface area of the material must be large enough that the quantity of radioactivity collected on the wipe will be greater than the MDA of the instrument. Wipe surveys of areas smaller than this minimum surface area require more sophisticated measuring instruments, such as a scaler measurement, and the entire surface of the material should be wiped. The minimum area for using a large-area wipe survey is given by where GV is the guideline value of the potential contaminant, given in Table 4-1.

$$A_{min} = \frac{MDA}{GV_{removable}}\ x\ 100\ cm^2;$$

Where:

A_{min} is the minimum area for using a large-area wiper survey,
$GV_{removable}$ is the guideline value of the potential contaminant
MDA is the minimum detectable activity.

The purpose of a smear survey is to locate and quantify removable contamination that is known or suspected to exist. For small items, a smear may be used at any time to verify the item's contamination status. A smear or swipe survey is performed by wiping a cloth, paper, plastic foam, or fiberglass disk over a 100-cm^2 area of the surface. The swipe should be taken with a dry medium using moderate pressure except for tritium. A common field practice is to use two fingers to press the swipe medium against the surface to be swiped. The swipe is then moved along an "S" shaped path that has a nominal length of 8 in. (20 cm) to 10 in. (25 cm). When the potential contaminant emits alpha radiation, paper or fiberglass filter papers should

be used so that alpha activity is not attenuated by becoming imbedded in the wipe. If the contaminant is an alpha-emitter and the surface is wet, the smear should be dried before counting. To improve the detection limit, smears may be taken over areas larger than 100 cm^2. However, the size of the area smeared should be limited to prevent buildup of material (radioactive or otherwise) that would attenuate alpha radiation. The current practice at DOE facilities is to use the 100 cm^2 area as the minimum size of objects being smeared. Appropriate corrections should be made for objects smaller than 100 cm^2.
If contamination is detected during a scan survey for fixed contamination, a swipe survey for removable contamination should be performed to determine if the contamination is fixed and to quantify any removable contamination. If no contamination above the guideline values for removable contamination in Table 4-1 is detected during the smear survey, the contamination is fixed, and the area should be posted appropriately.

A smear survey may be used routinely to detect removable contamination, especially for contamination surveys of radiological areas.

Table 4-1. 10 CFR § 835 Appendix D Surface Contamination Values[(1)] (dpm/100cm^2)

Radionuclide	Removable [2,4]	Total (Fixed + Removable)[2,3]
U-nat, U-235, U-238, and associated decay products	[7] 1,000	[7] 5,000
Transuranics, Ra-226, Ra-228, Th-230, Th-228, Pa-231, Ac-227, I-125, I-129	20	500
Th-nat, Th-232, Sr-90, Ra-223, Ra-224, U-232, I-126, I-131, I-133	200	1,000
Beta-gamma emitters (nuclides with decay modes other than alpha emission or spontaneous fission) except Sr-90 and others noted above[5]	1,000	5,000
Tritium and STCs[6]	10,000	See Footnote 6

[1] The values in this table, with the exception noted in footnote 6 below, apply to radioactive contamination deposited on, but not incorporated into the interior or matrix of, the contaminated item. Where surface contamination by both alpha- and beta-gamma-emitting nuclides exists, the limits established for alpha- and beta-gamma-emitting nuclides apply independently.

[2] As used in this table, dpm (disintegrations per minute) means the rate of emission by radioactive material as determined by correcting the counts per minute observed by an appropriate detector for background, efficiency, and geometric factors associated with the instrumentation.

[3] The levels may be averaged over one square meter provided the maximum surface activity in any area of 100 cm^2 is less than three times the value specified. For purposes of averaging, any square meter of surface shall be considered to be above the surface contamination value if: (1) from measurements of a representative number of sections it is determined that the average contamination level exceeds the applicable value; or (2) it is determined that the sum of the activity of all isolated spots or particles in any 100 cm^2 area exceeds three times the applicable value.

[4] The amount of removable radioactive material per 100 cm^2 of surface area should be determined by swiping the area with dry filter or soft absorbent paper, applying moderate pressure, and then assessing the amount of radioactive material on the swipe with an appropriate instrument of known efficiency. (Note - The use of dry material may not be appropriate for tritium.) When removable contamination on objects of surface area less than 100 cm^2 is determined, the activity per unit area shall be based on the actual area and the entire surface shall be wiped. It is not necessary to use swiping techniques to measure removable contamination levels if direct scan surveys indicate that the total residual surface contamination levels are within the limits for removable contamination.

[5] This category of radionuclides includes mixed fission products, including the Sr-90 which is present in them. It does not apply to Sr-90 which has been separated from the other fission products or mixtures where the Sr-90 has been enriched.

[6] Tritium contamination may diffuse into the volume or matrix of materials. Evaluation of surface contamination shall consider the extent to which such contamination may migrate to the surface in order to ensure the surface contamination value provided in this appendix is not exceeded. Once this contamination migrates to the surface, it may be removable, not fixed; therefore, a "Total" value does not apply. In certain cases, a “Total” value of 10,000 dpm/100 cm^2 may be applicable either to metals of the types from which insoluble special tritium compounds are formed, that have been exposed to tritium, or to bulk materials to which insoluble special tritium compound particles are fixed to a surface.

[7] These limits apply only to the alpha emitters within the respective decay series.

Scan Survey for Fixed Contamination

A scan survey for fixed contamination requires passing a detector attached to a portable instrument over the surface of the area being surveyed at a fixed, known scan speed and at a specified distance from the surface. Typically, the scan speed is 2 in./s (5 cm/s) and the maximum distance is 0.25 in. (0.6 cm) for alpha-contamination instruments. A scan survey should be used to survey material that resides in an area controlled for contamination purposes, an area where unsealed radioactive sources are used, or an area surrounding an area controlled for contamination purposes.

During the performance of scan surveys, the audible response of the instrument is faster than the needle deflection. Therefore, audible response should be used in conjunction with meter readings. For alpha surveys, the surveyor should pause for 3 to 5 seconds each time an individual pulse is detected in order to allow a longer count time at the location of the detected pulse, until it is determined whether the response indicates random background noise or detected contamination

The most critical factor affecting a scan survey measurement is the speed at which scan surveys are performed. Counting time is inversely proportional to scan speed. For instruments with larger detector faces, the scan speed is faster for a given rate of meter movement because a point on the surveyed surface remains beneath the window longer. To ensure that low levels of contamination can be detected, it is necessary that a maximum scan speed be mandated and that this speed be implemented during field measurements. Empirical information is available indicating that, for most instruments in current use, a maximum scan speed of 2 in./s (5 cm/s) can detect contamination at or above the total contamination values specified in Table 4.1 for nearly all radionuclides with 67% confidence.

For a rectangular probe, the detection probability can be increased by moving the probe lengthwise.

4.2.3 Release Criteria

Material in contamination, high contamination, or airborne radioactivity areas, shall be treated as radioactive material and shall not be released to controlled areas if either of the following conditions exist:

a. Measurements of accessible surfaces show that either the total or removable contamination levels exceed the values specified in Table 4-1.

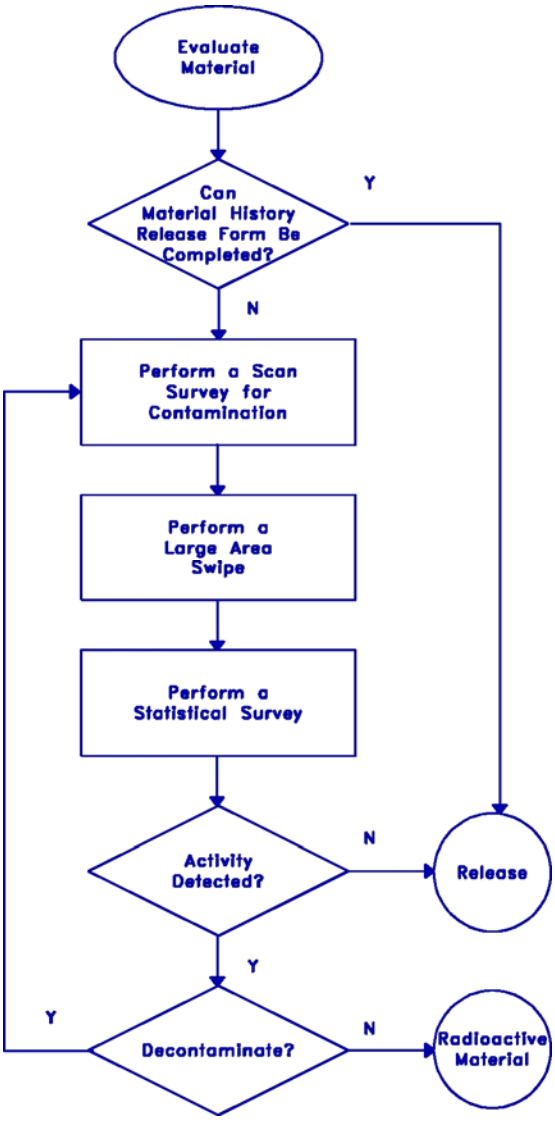
b. Prior use suggests that the contamination levels on the inaccessible surfaces are likely to exceed the values specified in Table 4-1 (10 CFR § 835.1101).

Usually, an ALARA evaluation is needed to ensure the contamination levels cannot be reduced even further below the DOE Order 458.1 release limits for a reasonable cost.

Material that has never been in a contaminated or airborne radioactivity area may be removed to controlled areas without survey. If the history of the item is unknown, it is appropriate to assume that it may have been in a contaminated or airborne radioactivity area.

To release material from radiological control, a methodology has been developed to reduce the time required to perform a survey while meeting DOE requirements. A logic diagram of the protocol is shown in Figure 4-1. The methodology ensures, with 67% confidence, that the guideline values of DOE Order 458.1 and 10 CFR Part 835 are met. The most current preapproved authorized limits for support of DOE Order 458.1 are listed in DOE G 441.1-XX (DRAFT), *Control and Release of Property with Residual Radioactive Material for use with DOE 5400.5, Radiation Protection of the Public and the Environment.*

Figure 4-1. Protocol for Release of Materials from Radiological Control



The material release methodology has four main components: material evaluation, scan survey for fixed contamination, large-area wipe survey for removable contamination (described above) followed by technical smears as necessary, and statistical survey for fixed contamination. The material evaluation process involves consideration of the previous known uses of the material, as well as typical uses and the environment in which the material was used. Material evaluation places the material into one of two categories: not potentially contaminated or potentially contaminated.

Non-radioactive material can be released without an instrument survey if its documented history ensures the following:

a. That it has never been used or stored in an area controlled for contamination

purposes (i.e., a contamination area, high contamination area, or airborne radioactivity area).

b. That it has never come into contact with unsealed radioactive sources.

c. That it has not been stored or used in a radiological buffer area (RBA) surrounding a contamination area, high contamination area, or airborne radioactivity area.

This material may be considered to be not contaminated and an instrument survey is not necessary. A material history release form should be used to document the release of material that is known to be free of contamination by its history of use. If the material history release form cannot be completed, or if the history of the material is unknown, an instrument survey must be made of the material. Material released from RBAs around contamination areas, high contamination areas, or airborne radioactivity areas should also be evaluated using an instrument survey.

The material evaluation process should also consider the nuclides to which the material was potentially exposed. If the material was exposed to significant quantities of nuclides that are difficult to detect, including tritium ^{14}C, ^{125}I, or ^{129}I, an appropriate survey methodology should be applied.

A scan survey for fixed contamination requires passing the detector of an alpha and a beta/gamma survey instrument, as applicable, over the accessible surface of the material. The detector should be moved at a constant rate that allows detection of contamination at a level equal to three times the guideline value. If a change in the audible output of the instrument is heard, the area under the window of the instrument should be re-surveyed using a stationary measurement for 3 to 5 seconds. If the increase does not persist, the scan should continue. If the elevated counts persist, it is good practice to consider the material is contaminated and should not be released. This procedure should be followed until the surface of the material has been surveyed.

The scan survey for fixed contamination ensures that none of the material's surface is contaminated above three times the guideline value. If no contamination above background is detected during the scan survey, a large-area wipe survey for removable contamination should be performed. If contamination above background is detected, then decontamination of the material should be considered and the methodology described in this document should not be applied.

Following the scan and large-area wipe surveys, a statistical survey for fixed contamination should be performed. The survey methodology should be used for both beta/gamma and alpha contamination, unless only one type of potential contaminant exists in the facility. If no measurements above background are observed, the material may be released from

radiological control.

The fixed survey measurements should be chosen using random detector placements over the entire surface of the material. It may be prudent to bias some of the measurements toward those areas that are more likely to be contaminated, including handles, horizontal surfaces, stains, cracks, and other surface anomalies in which foreign material typically accumulates. This type of selection bias will further increase the confidence associated with the statistical survey method.

Measurements performed to release material should be made in a low-background area unless the MDA of the instrument in a high-background area is known and appropriate considerations are made. If material is being surveyed for release from a radiological area, performing measurements in a low-background area may not be possible. If background count rates are high enough that the release guideline values cannot be measured in the radiological area by using portable survey instruments, a survey for removable contamination should be performed to avoid spreading removable contamination from the radiological area. If the survey for removable contamination does not indicate the presence of contamination in excess of background levels, the material may be moved to an area with a lower background for an immediate fixed contamination survey.

4.2.3.1 Uranium Contamination Detection

The detection and measurement of uranium contamination is necessary to ensure control of contamination and compliance with DOE requirements. Typically, detection of uranium contamination has been performed using the alpha activity. However, for some conditions and situations, detection of the beta/gamma radiations from uranium decay products may be a more sensitive and more appropriate monitoring technique. For natural uranium, depleted uranium, and the lower levels of enriched uranium that are in equilibrium with their decay products, the detection sensitivity for the beta/gamma radiations is about five times more sensitive than by the detection of the alpha alone. If the uranium is highly enriched or has been very recently processed, detection using the alpha radiation is necessary because of the high alpha:beta ratio.

Detection of uranium contamination may require use of beta/gamma-sensitive instruments when surveying upholstery material, rugs, cloth, and wet surfaces. Because of the range and ease of shielding alpha particles, burial or surface liquid may preclude the detection of the alpha radiation. The use of GM detectors, such as the thin-window detector probe, is particularly useful in these situations. In some instances, a thin Nal detector may be better than a GM detector for detecting low-energy photons from uranium contamination.

Many of the processes used in uranium facilities may separate and/or concentrate impurities or decay products of uranium. Examples of these processes are uranium recovery from ore, reduction of green salt to metal, UF_6 conversion, casting of metal, and uranium oxidation.

Radionuclides of particular importance are ^{234m}Pa and other decay products and trace impurities such as ^{99}Tc, ^{239}Pu, and ^{231}Np. In addition to the separation processes, some of the decay products of uranium may be selectively accumulated in tank and pipe liner material. Effective dose rates of up to 150 mrem/h, attributed to radium accumulation, have been measured from neoprene liner material. Dose rates from furnace lids and crucibles have been measured as high as 30 rad/h.

Detection and measurement of uranium contamination, both surface and airborne, require a knowledge of the process and of the separation and concentration mechanisms. Depending upon the process, the time since separation, and the isotopic ratios of the uranium, contamination resulting from uranium operations may be almost totally alpha or totally beta/gamma-emitters. Consequently, detection techniques may require the capability to detect all types of radiations. Appropriate monitoring in most facilities requires both types of surveys, but on differing frequencies.

4.2.4 ALARA Guidelines

Contamination levels should be maintained ALARA to minimize the potential for the spread of contamination and to reduce the protective measures and equipment required. Control of radioactive material at the source and prevention of the generation of contamination are generally more effective and less costly than remediation.

4.3 Personnel Contamination Control

Contamination control should be achieved primarily by physical design features, including engineering controls (see the discussion above), such as containment, confinement, and ventilation control. Only if the primary controls fail or if there is a potential for personnel contamination during an activity are controls such as protective clothing and respirators advisable.

4.3.1 Monitoring Philosophy

Although the primary hazard to personnel from uranium is from internal exposure, contamination is also of concern because of potential skin doses. Additionally, an objective of the contamination control program is to confine uranium contamination to production/work areas and to prevent any spread of contamination to areas outside the plant or to the public. Therefore, guidelines for allowable contamination on personnel and personal clothing/shoes both inside the plant and prior to exiting radiological areas are required. Also, a personnel monitoring program must be developed with adequate monitoring equipment and sensitivity to provide assurance that contamination is effectively controlled. The guidelines should be developed considering the following factors:

a. The need to prevent detectable activity from appearing outside the controlled area

b. The degree of risk to the health of the employees, their families, and the public from contamination removed from the plant

c. The technical feasibility of measurement of the guide levels

d. Commitment to the policy of keeping contamination to the minimum practical level

e. The presence of other radionuclides due to the presence of recycled uranium contaminants or uranium daughters

4.3.2 Monitoring Program

Instrumentation should be provided and persons in a uranium work station should be required to survey themselves at established frequencies. As a minimum, workers should survey their gloves and coverall sleeves each time they are withdrawn from a glove box (or similar containment system) and after each glove replacement or bag-out operation.

In addition to mandatory monitoring at the exit to areas controlled for contamination, personnel monitoring for contamination should be mandatory at the egress from controlled areas and be conducted in a verifiable manner. Portal monitors, hand-and-shoe counters, and/or portable survey instruments may be used for this purpose. If employees are instructed to perform self-monitoring, the equipment should be set up in a "go/no-go" mode and employees should be clearly instructed in the required actions to take if predetermined action levels are exceeded. Frequent audits should be performed to verify that controls are adequate. Limiting the number of egress points and controlling personnel movement can minimize the numbers of locations where positive control of personnel monitoring must be maintained.

Monitoring of shoes, clothing, and hands should be required prior to leaving a work station where uranium or uranium contaminated material was handled. Following routine work, self-monitoring upon exit is usually considered adequate if the person has received proper training in the use of the instrument provided. The instrument should clearly detect an unacceptable level of contamination.

For work that involves a high potential for intake of radioactive material in excess of the regulatory limits, engineering and administrative controls should be implemented to eliminate this potential. If these controls are inadequate, then appropriate PPE, including respirators, should be implemented.

After performing work that, in retrospect, involved a high potential for intake of radioactive material, each worker should provide a swipe of the nasal passages, to be counted

immediately. If respiratory protection was worn, there is no need for nasal swipes unless a breach of the respirator seal is suspected. If facial contamination is detected during the exit contamination monitoring, a nasal swipe should be taken and counted immediately. Chapter 5 provides guidance on the actions to be taken if a nasal swipe is positive.

4.3.3 Protective Clothing

Various types of protective clothing, including laboratory coats, shoe covers, gloves, coveralls, plastic or rubber suits, and air-purifying or atmosphere-supplying respiratory protective equipment, may be required for operations with transuranic radionuclides. The use of company-issue shoes and clothing for employees with work assignments in process areas can be a major aid in contamination control. Some facilities are using disposable anti-contamination clothing. This may be a cost savings from a handling standpoint. However, disposal costs should be considered.

4.3.4 Respiratory Protection

While every attempt should be made to control uranium hazards utilizing physical design features, including engineering controls, the use of respiratory protection is an essential part of the radiological control program.

As with personnel protective equipment, respiratory equipment utilized must also provide protection from the full range of airborne hazards that may be encountered in the work environment. For example, a uranium metal machining operation may have both an airborne uranium oxide hazard and an airborne hazard from solvent vapors. The respirator utilized must be effective for both types of hazard. Also, one airborne contaminant may interfere with the effectiveness of the canister in an air-purifying device that is designed for a different contaminant. For example, a corrosive gas, such as hydrofluoric acid (HF), may attack a HEPA filter and render the filter ineffective. It is important to coordinate the use of respiratory protection requirements with other health protection groups. The respiratory protection program should also be in compliance with ANSI Standard Z88.2-2015, *American National Standard for Respiratory Protection* (2015a) requirements. In specifying respirators for various applications, one should always know the applicable protection factors to determine that the range of hazard that may be encountered will be covered. While the specification of respiratory protection should normally be made a result of personal and/or area sampling results, the use of respirator guides based on surface contamination monitoring results is also acceptable.

4.3.5 ALARA Guidelines

The total dose to an individual and the collective dose to the work force should be ALARA. When applied to personnel contamination or internal intakes, this generally means less-than-detectable dose with the best available commercial technology.

4.3.6 Release Criteria

The decision to release personnel with detectable uranium contamination is made on a case-by-case basis. If the individual is injured and needs prompt medical attention, medical treatment will always take precedence, with compensatory measures made for protecting medical personnel and facilities. If injuries are absent or do not require immediate attention, decontamination is preferable to ensure that the dose to the contaminated individual and the potential for inhalation by the victim and medical staff are minimized and the spread of contamination is prevented.

In a case where decontamination is incomplete due to injury to the skin or other reasons, the individual may be provisionally released with measures to prevent the spread of contamination.

4.4 Decontamination and Decommissioning Techniques

This section concentrates on decontamination techniques to be used in the final decommissioning of a uranium-contaminated facility for unrestricted release. Some of these techniques are similar to those used during routine operations (e.g., personnel decontamination and some equipment and building surface decontamination). Contamination detection methods are similar for routine and decommissioning operations.

4.4.1 Personnel Decontamination

Skin decontamination should be performed by health physics technicians or other members of the health physics staff. The treatment and decontamination of wounds should be performed by medical staff.

Non-abrasive methods should be used for skin decontamination to protect the tissues from deeper contamination. Masking tape should be used to remove dry contamination. Wet decontamination should be used to remove residual contamination. The skin should be gently scrubbed with soap and water. Diluted bleach with water may be applied as needed to decontaminate more effectively. The following procedure is recommended:

a. Survey the worker to determine the contaminated areas of the skin. Have the medical staff treat and decontaminate breaks in the skin.

b. Wipe loose contamination with a gauze sponge or cotton applicators dipped in mild antiseptic detergent. Do not spread contamination to uncontaminated areas.

c. Rub the skin with the applicators to produce good lather.

d. Use soft bristle scrub brushes for fingernails and other difficult-to-clean areas as long as the skin barrier is maintained intact. It may be difficult to decontaminate the cuticles and under the nails.

e. Dry the skin area with cleansing tissue.

f. After the skin is thoroughly dry, survey it for any remaining contamination.

g. If no contamination is detected, apply a good-quality hand cream to prevent chapping.

Another effective non-abrasive decontamination method involves placing the contaminated hand in a cotton glove and then a Latex glove (causing the hand to perspire).

The decontamination factor is the ratio of the initial contamination level to the contamination level after decontamination methods are applied, as determined by survey instrument readings. Non-abrasive methods should be repeated until the decontamination factor between washes drops below 2 or 3 with significant contamination still remaining.

If contamination persists on the skin, a more abrasive decontamination method may be necessary. The decision to proceed with a more abrasive method should be based on the effectiveness of the decontamination. An abrasive soap should be applied with a moist gauze sponge or soft brush while rubbing the skin to develop a soapy lather. Care should be exercised to prevent damage to the skin surface. If contamination persists after using the abrasive soap, potassium permanganate ($KMnO_4$) and sodium bisulfite ($NaHSO_3$) should be considered. Paint the contaminated skin with $KMnO_4$ using cotton-tipped applicators, allow the solution to dry, and paint it again two or three more times, allowing the solution to dry thoroughly between each application. The skin will then appear almost black. Applicators should be discarded after each use to avoid spreading contamination to the solutions. Then, rub the treated area with sodium bisulfate using cotton applicators, until the brown discoloration is removed. Rinse the skin with water to remove the remaining $KMnO_4$, and dry the area thoroughly and survey it for contamination.

If contamination persists after all the above decontamination efforts, wrap the contaminated area to control the contamination and consult with medical personnel.

Liberal irrigation with room temperature water or saline solution (preferable) is recommended for eye, nose, and mouth contamination. These procedures are performed by the medical staff to remove contamination.

4.4.2 Equipment and Surface Decontamination

Decontamination of surface areas may be as simple as hosing off the floors with water,

washing surfaces with detergent and water, or wiping with household dust cloths. Waste material generated from decontamination activities (e.g., water and wipe material) must be contained and disposed of as radioactive waste. For some locations, vacuuming the surfaces may be appropriate. If vacuuming is used, HEPA-filtered vacuum systems are required to keep airborne radioactive material out of the vacuum exhaust.

For some operations, periodic surface flushing with water may be adequate to maintain acceptable contamination levels. Precautions should ensure control and collection of run-off water so material may be recovered and waste water analyzed before discharge. Depending upon which isotope of uranium is involved, geometrically safe containers may be required for collecting and holding the liquid.

Depending upon the physical and chemical form of the uranium and the type of surface, uranium may become imbedded in the surface. Removal of embedded material may require physical abrasion, such as scabbling, grinding, sand blasting, or chipping, or it may be accomplished using chemical etching techniques. If the surface is porous, complete replacement could be necessary. The use of high-pressure water (hydroblasting) has been quite successful for metal and concrete surfaces.

Ultrasonic cleaning techniques, electro-polishing, or chemical baths may be useful for decontamination of high-cost items if the chemicals used are compatible with the material to be cleaned.

A description of different decontamination techniques is found in DOE/EM-0383, *DOE Decommissioning Handbook* (2000), and publications by the Electric Power Research Institute. The *DOE Decommissioning Handbook* also includes guidance on decontamination techniques, assessment of environmental impacts, disposition of wastes, and preparation of decommissioning cost estimates.

5 INTERNAL DOSIMETRY

Internal dosimetry is an essential part of a comprehensive radiological control program at every facility where uranium is handled or processed. The purpose of an internal dosimetry program is to monitor workplace activities, assess accidental or inadvertent intakes of radioactive material, and conduct internal dose assessments from bioassay measurement data.

DOE requires that new or modified facilities be designed, operated, and remediated to prevent intakes of radioactive materials in accordance with 10 CFR § 835.1002. Radiological controls for the workplace shall ensure that under normal conditions, radionuclides are contained and handled properly, and that intakes are as low as reasonably achievable.

Experience has shown that the most common route for inadvertent uranium intake is inhalation. The uranium may be in natural, enriched, or depleted form, or a combination thereof. Intakes can also occur by accidental ingestion or by wound contamination. Surveillance programs should be designed to rapidly detect a release in the event of a loss of radioactive material containment. Internal dosimetry programs should be tailored to the needs of each uranium-handling facility so that workers' internal doses are determined by appropriate methods.

When workers are inadvertently exposed to radioactive material, appropriate corrective action should be taken to ensure that control and containment are re-established. Prompt detection by routine workplace monitoring practices is essential to regaining control after any contamination spread or loss of containment. Prompt workplace indications of potential intake are also crucial to ensure timely initiation of special bioassay monitoring for intake and dose assessment. An early assessment of the probable severity of an intake and its corresponding dose, preferably within the first two hours of the intake, is needed for decisions on dose reduction therapy and event reporting. Uranium is both a radiological and chemical hazard. Because the total risk must be considered, both hazards must be considered. For uranium intakes, it may take many months to obtain the bioassay data necessary for final dose assessment. Until such data become available, ongoing preliminary assessments of intake and dose may be necessary to provide guidance for the administrative and medical management of the workers.

5.1 Internal Dose Evaluation Program

Internal doses are not directly measured, but are estimated or calculated based on knowledge of the material to which a worker may be exposed and it's known or assumed biokinetic behavior. The common approach to internal dosimetry is to calculate an occupational intake based on worker bioassay measurements or workplace air-sample data and assumed breathing rates. Once an intake is calculated, appropriate equivalent doses to organs and

tissues of concern can be estimated by using fundamental dosimetry principles, by various intake-to-dose conversion factors, that incorporate assumed biokinetic models, or by use of an appropriate computer code. Intake-to-dose conversion factors can be found in ICRP Publication 68 (1994b). Further discussion on intake and dose assessment is provided in Section 5.8.

Participation in internal dose evaluation programs (which include routine bioassay programs) is required for conditions identified in 10 CFR § 835.402(c). The internal dose evaluation program must address both general workplace conditions and individual intakes. Workplace conditions are monitored through air and surface contamination monitoring programs. Individual monitoring for intakes is commonly performed using bioassay procedures. Bioassay monitoring includes both direct (in vivo) measurements of radioactivity in the body and indirect (in vitro) measurements of material excreted or removed from the body.

10 CFR § 835.402 requires participation in a bioassay program if a general employee is likely to exceed 0.1 rem CED from all intakes for all radionuclides in a year. Participation in a bioassay program is generally based on the possibility that a single intake causing a dose in excess of 0.1 rem CED might occur.

Indications of intake include (but are not limited to) detection of facial or nasal contamination, positive air monitoring or sampling results that may indicate internal exposure, or any wound in which contamination is detected or suspected. The most common internal exposure monitoring program for workers is the bioassay program, which must be designed for the specific nuclides and forms of material at a particular facility. Likely candidates for internal exposure monitoring include personnel who may be routinely exposed to surface or airborne contamination, or those identified by workplace indicators.

Workplace monitoring for potential internal exposures is performed to verify the adequacy of containment and work practices. This monitoring includes air sampling, continuous air monitoring, personal contamination surveys, and workplace contamination surveys. Facilities are to be designed and operated to minimize internal exposure. Details regarding workplace monitoring and control practices are discussed in Chapter 4, Contamination Control.

5.1.1 Performance Capabilities for Internal Exposure Monitoring

Bioassay programs must be capable of showing compliance with the 5-rem/year stochastic and 50-rem/year deterministic dose limits of 10 CFR § 835.202. 10 CFR § 835.402(c)(1) identifies 0.1 rem CED for all likely intakes in a year as a level above which workers must participate in a bioassay program. Therefore, such bioassay monitoring programs must be capable of detecting individual doses at that level. To meet this requirement, reliance must be placed on workplace monitoring to identify potential intakes at the time they occur so that special bioassay monitoring can be initiated.

Performance capabilities for bioassay and internal dosimetry programs can be expressed as the minimum detectable dose, based on some combination of minimum detectable activity and frequency of measurement or time post-intake at which the measurement is made. The term "minimum detectable dose" is preferred over any variants of the occasionally encountered terms "dose-missed" or "potentially undetected dose," which were usually defined as the same thing. The connotation of the latter terms is that of an actual intake which was not detected, whereas the intent was to define a measure of program sensitivity to doses that might have gone undetected had an intake occurred. The preferred term "minimum detectable dose" (MDD) ties the concept to the recognized terminology of MDA.

The MDD for a bioassay program must meet the aforementioned dose limit requirements of 10 CFR § 835.202. A design goal of 0.1 rem CED from all intakes of similar nuclides in a year is desirable but unrealistic for a routine program. To meet these requirements, bioassay programs should have measurement sensitivities (i.e., MDAs for bioassay measurements) established based on the material to which workers might be exposed. Examples of such sensitivities are given in Tables 5-1 through 5-3 for pure ^{238}U monitored by urinalysis, fecal analysis, and lung counting, respectively. The bioassay goals are calculated by multiplying the intake (nCi) by the intake retention fraction (IRF) and by a correction factor of 2,220 dpm/nCi, where intake is the dose limit divided by a calculated dose conversion factor (rem/nCi). For uranium-238, for simplicity, the dose limit goal is based on the 50-rem committed equivalent dose (for type S uranium-238, the stochastic limit is slightly (5rem) is slightly more restrictive); the other goals are based on the 0.1 rem CED monitoring threshold. Table 5-1 through Table 5-3 give values for bioassay goals and the calculations used to derive them.

There may be circumstances in which the measurement technology is not available to provide the sensitivities required for the 0.1 rem goal using routine, periodic measurements at reasonable frequencies. Therefore, because the goal of 0.1 rem CED cannot be met through routine bioassay, the radiation protection organization should take the following administrative actions:

a. Ensure that adequate control measures are applied to prevent intakes.

b. Document the adequate control measures for auditing purposes.

c. Upgrade bioassay measurement systems and workplace monitoring practices to provide state-of-the-art measurements.

d. Ensure that internal dose assessments use state-of-the-art technology.

All confirmed occupational intakes of uranium, regardless of magnitude, should be assessed. The results of all bioassay and other measurements needed to demonstrate the quality of measurements and dose assessment should be recorded and maintained. The recording and

reporting requirements for internal dosimetry data are set forth in Section 3.7 of this technical standard

Table 5-1. Urine Bioassay Goals[(a)] for ^{238}U

Days Post-Intake	Urine Intake Retention Fraction[(b)]	Dose Limit Goal[(c)] (dpm)	100-mrem CED Goal[(d)] (dpm)
	Type F	**Inhalation**	
1	1.85E-01	5.55E+05	1.91E+04
7	3.47E-03	1.04E+04	3.58E+02
30	6.78E-04	2.03E+03	7.00E+01
60	2.26E-04	6.78E+02	2.33E+01
90	1.20E-04	3.62E+02	1.24E+01
200	2.44E-05	7.32E+01	2.52E+00
400	4.58E-06	1.37E+01	4.73E-01
1000	2.31E-06	6.93E+00	2.39E-01
10000	2.32E-07	6.96E-01	2.40E-02
20000	9.63E-08	2.89E-01	9.94E-03
	Type M	**Inhalation**	
1	2.34E-02	5.40E+04	8.78E+02
7	6.43E-04	1.48E+03	2.41E+01
30	2.65E-04	6.12E+02	9.94E+00
60	1.67E-04	3.85E+02	6.26E+00
90	1.25E-04	2.88E+02	4.69E+00
200	5.76E-05	1.33E+02	2.16E+00
400	1.79E-05	4.13E+01	6.71E-01
1000	1.12E-06	2.58E+00	4.20E-02
10000	5.74E-08	1.32E-01	2.15E-03
20000	2.37E-08	5.47E-02	8.89E-04
	Type S	**Inhalation**	
1	7.13E-04	3.28E+02	7.50E+00
7	1.91E-05	8.80E+00	2.01E-01
30	7.71E-06	3.55E+00	8.11E-02
60	5.18E-06	2.39E+00	5.45E-02
90	4.28E-06	1.97E+00	4.50E-02
200	3.18E-06	1.46E+00	3.35E-02
400	2.55E-06	1.17E+00	2.68E-02
1000	1.52E-06	7.00E-01	1.60E-02
10000	1.01E-07	4.65E-02	1.06E-03
20000	2.56E-08	1.18E-02	2.69E-04

(a) The goals reflect the activity in a 24 hour urine void corresponding to either a 50 rem committed equivalent dose or a 0.1 rem CED.

(b) IRF values obtained from "Intake Retention Functions Developed from Models Used in the Determination of Dose Coefficients Developed for ICRP Publication 68 – Particulate Inhalation" (Potter, 2002).

(c) Calculated as Goal (dpm) = Intake ∗ IRF ∗ 2220 dpm/nCi, where Intake (nCi) is the 50 rem committed equivalent dose limit/ dose conversion factor and IRF is the intake retention fraction.

The dose conversion factor (committed dose per unit intake) derived from the ICRP Publication 68 Database is shown below:

Type F, rem/nCi	Type M, rem/nCi	Type S, rem/nCi
3.70E-02 (Bone Surface)	4.81E-02 (Lungs)	2.41E-01 (Extrathoracic Airways)

(d) Calculated as Goal (dpm) = Intake * IRF * 2220 dpm/nCi, where Intake (nCi) is the 0.1 rem CED limit/ dose conversion factor and IRF is the intake retention fraction.

The dose conversion factor (committed dose per unit intake) derived from the ICRP Publication 68 Database is shown below:

Type F, rem/nCi	Type M, rem/nCi	Type S, rem/nCi
2.15E-03	5.92E-03	2.11E-02

Table 5-2. Fecal bioassay Goals[(a)] for ^{238}U

Days Post-Intake	Urine Intake Retention Fraction[(b)]	Dose Limit Goal[(c)] (dpm)	100-mrem CED Goal[(d)] (dpm)
	Type F	**Inhalation**	
1	5.65E-02	1.68E+05	5.82E+03
7	8.54E-04	2.58E+03	9.00E+01
30	3.17E-06	9.60E+00	3.30E-01
60	1.41E-06	4.20E+04	1.44E-01
90	7.83E-07	2.34E+04	7.80E-02
200	1.62E-07	4.86E-01	1.68E-02
400	3.04E-08	9.00E-02	3.12E-03
1000	1.53E-08	4.56E-02	1.56E-03
10000	1.54E-09	4.62E-03	1.62E-04
20000	6.39E-10	1.92E-03	6.60E-05
	Type M	**Inhalation**	
1	1.08E-01	2.53E+04	4.02E+03
7	2.21E-03	5.10E+03	8.40E+01
30	2.72E-04	6.00E+02	1.02E+01
60	1.25E-04	2.88E+02	4.68E+04
90	6.22E-05	1.44E+02	2.34E+04
200	1.15E-05	2.64E+01	4.32E-01
400	2.78E-06	6.60E+00	1.02E-01
1000	8.10E-08	1.86E-01	3.06E-03
10000	3.81E-10	0.9E-01	1.44E-05
20000	1.57E-10	3.60E-04	5.88E-06
	Type S	**Inhalation**	
1	1.16E-01	5.34E+03	1.20E+03
7	2.42E-03	1.14E+02	2.52E+01
30	3.50E-04	1.62E+01	3.66E+04
60	1.86E-04	8.40E+00	1.98E+04
90	1.07E-04	4.92E+00	1.14E+04
200	3.30E-05	1.50E+00	3.48E-01
400	2.12E-05	9.60E-01	2.22E-01
1000	1.11E-05	5.10E-01	1.14E-01
10000	6.00E-08	2.76E-03	6.00E-04
20000	6.69E-09	3.06E-04	7.20E-05

(a) The goals reflect the activity in a 24 hour fecal sample corresponding to either a 50 rem committed equivalent dose or a 0.1 rem CED.

(b) IRF values obtained from “Intake Retention Functions Developed from Models Used in the Determination of Dose Coefficients Developed for ICRP Publication 68 – Particulate Inhalation”

(Potter, 2002).

(c) Calculated as Goal (dpm) = Intake ∗ IRF ∗ 2220 dpm/nCi, where Intake (nCi) is either the 50 rem committed equivalent dose or threshold/committed equivalent dose conversion factor and IRF is the intake retention fraction.

The dose conversion factor (committed dose per unit intake) derived from the ICRP Publication 68 Database is shown below:

Type F, rem/nCi	Type M, rem/nCi	Type S, rem/nCi
3.70E-02 (Bone Surface)	4.81E-02 (Lungs)	2.41E-01 (Extrathoracic Airways)

(d) Calculated as Goal (dpm) = Intake * IRF * 2220 dpm/nCi, where Intake (nCi) is the 0.1 rem CED threshold/CED conversion factor and IRF is the intake retention fraction.

The dose conversion factor (committed dose per unit intake) derived from the ICRP Publication 68 Database is shown below:

Type F, rem/nCi	Type M, rem/nCi	Type S, rem/nCi
2.15E-03	5.92E-03	2.11E-02

Table 5-3. In Vivo Measurement Bioassay Goals for ^{238}U[(a)]

Days Post-Intake	Urine Intake Retention Fraction[(b)]	Dose Limit Goal[(c)] dpm	100-mrem CED Goal[(d)] dpm
	Type F	**Inhalation**	
1	3.65E-01	1.10E+06	3.77E+04
7	7.27E-02	2.18E+05	7.51E+03
30	3.63E-02	1.09E+05	3.75E+03
60	2.49E-02	7.47E+04	2.57E+03
90	2.00E-02	6.00E+04	2.07E+03
200	1.38E-02	4.14E+04	1.42E+03
400	1.17E-02	3.51E+04	1.21E+03
1000	9.94E-03	2.98E+04	1.03E+03
10000	3.29E-03	9.87E+03	3.40E+02
20000	1.79E-03	5.37E+03	1.85E+02
	Type M	**Inhalation**	
1	4.74E-01	1.09E+06	1.78E+04
7	6.39E-02	1.47E+05	2.40E+03
30	4.51E-02	1.04E+05	1.69E+03
60	3.34E-02	7.71E+04	1.25E+03
90	2.64E-02	6.09E+04	9.90E+02
200	1.42E-02	3.28E+04	5.33E+02
400	6.28E-03	1.45E+04	2.36E+02
1000	2.59E-03	5.98E+03	9.71E+01
10000	8.08E-04	1.86E+03	3.03E+01
20000	4.39E-04	1.01E+03	1.65E+01
	Type S	**Inhalation**	
1	4.89E-01	2.25E+05	5.14E+03
7	6.24E-02	2.87E+04	6.57E+02
30	5.04E-02	2.32E+04	5.30E+02
60	4.25E-02	1.96E+04	4.47E+02
90	3.82E-02	1.76E+04	4.02E+02
200	3.16E-02	1.46E+04	3.32E+02
400	2.60E-02	1.20E+04	2.74E+02
1000	1.55E-02	7.14E+03	1.63E+02
10000	1.06E-03	4.88E+02	1.12E+01
20000	2.86E-04	1.32E+02	3.01E+00

(a) Suitable for elemental or any long half-life uranium isotope.

(b) IRF values obtained from "Intake Retention Functions Developed from Models Used in the Determination of Dose Coefficients Developed for ICRP Publication 68 – Particulate Inhalation" (Potter, 2002).

(c) Calculated as Goal (dpm) = Intake ∗ IRF ∗ 2220 dpm/nCi, where Intake (nCi) is the 50 rem committed equivalent dose limit/dose conversion factor and IRF is the intake retention fraction.

The dose conversion factor (committed dose per unit intake) derived from the ICRP Publication 68 Database is shown below:

Type F, rem/nCi	Type M, rem/nCi	Type S, rem/nCi
3.70E-02 (Bones Surface)	4.81E-02 (Lungs)	2.41E-01 (Extrathoracic Airways)

(d) Calculated as Goal (dpm) = Intake * IRF * 2220 dpm/nCi, where Intake (nCi) is the 0.1 rem CED limit/dose conversion factor and IRF is the intake retention fraction.

The dose conversion factor (committed dose per unit intake) derived from the ICRP Publication 68 Database is shown below:

	Type F, rem/nCi	Type M, rem/nCi	Type S, rem/nCi
Effective Dose	2.15E-03	5.92E-03	2.11E-02

Radiation exposure records programs must also provide for the summation of internal and external doses, as required by 10 CFR § 835.702. While the summation process is not necessarily performed under a site internal dosimetry program, it is recommended that the program coordinator recognize what is required. The following summations are identified by 10 CFR § 835.702(c)(5):

a. Total effective dose (TED) in a year is defined as the summation of effective dose from external exposure and the CED to the whole body in a year

b. For any organ or tissue assigned an internal dose during the year, the sum of the equivalent dose from external exposure and the committed equivalent dose to organs or tissues of concern

c. Cumulative TED defined as 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

d. For the embryo/fetus of a declared pregnant worker, the summation of the equivalent

dose to the mother from external exposure during the entire gestation period and the CED to the embryo/fetus from intakes by the mother during the entire gestation period

Doses must be calculated and recorded (if greater than 10 mrem) for any confirmed uranium intake. What constitutes a confirmed intake is discussed in Section 5.7. Along with the doses, supporting records must be maintained, including the bioassay data, assumptions, biokinetic models, and calculational methods used to estimate the doses. These may be included in letter-report dose assessments, databases, technical basis documents, and similar records, either singly or in combination.

5.1.2 Protection of the Embryo/Fetus, Minors, and Members of the Public

The effective dose limit for the embryo/fetus of a declared pregnant worker is 0.5 rem for the entire gestation period, defined as the summation of external dose received and internal dose received during the gestation period (not the 50-year committed internal dose). Internal exposure monitoring is required if an intake is likely to result in more than 10% of that limit (i.e., 50 mrem for the gestation period). Providing adequate protection to keep the mother's intakes below the occupational limits will also provide adequate protection for the embryo/fetus. Thus, special bioassay for uranium during pregnancy is not required. As a matter of caution, some sites try to obtain baseline bioassays as soon as a pregnancy is declared, with another baseline bioassay following the end of pregnancy. Some sites also offer to restrict pregnant workers from jobs with relatively high potential for occupational intakes.

Minors and members of the public are limited by 10 CFR § 835.207 and 10 CFR § 835.208 to a total effective dose of 0.1 rem/year. Minors are also limited to 10% of the occupational dose limits of 10 CFR § 835.202(a)(3) and (a)(4). Internal exposure monitoring is required if an intake is likely to result in 50% of that limit (0.05 rem) from all radionuclide intakes in a year. With routine fecal sampling for insoluble uranium (Type S) a routine bioassay program is sufficiently sensitive to identify such intakes. Enhanced workplace surveillance or restriction of access may be required.

5.2 Characterization of Internal Hazards

Monitoring for uranium poses special problems for the following reasons.

a. Uranium presents both chemical and radiological toxicity risks, the relative importance of which depends on its transportability from the lung.

b. Uranium usually exists in mixed transportability classes.

c. Small, recent intakes easily mask larger, older intakes because nearly 50% of the uranium going to blood is cleared immediately through the urine.

d. An intake of Type S material potentially resulting in a CED of 0.1 rem generally cannot be detected by routine urinalysis alone. Monitoring of the workplace to document the working environment and to provide immediate indication of an intake is essential unless a combination of urine and fecal sampling is used in the routine bioassay program.

e. Low-level chronic intakes are common, so the bioassay program must monitor for long-term buildup as well as for potentially significant acute intakes.

f. Individual and temporal variability in the environmental background of uranium complicates interpretation of urinalysis and fecal results.

Consequently, the proper bioassay monitoring program for uranium workers is best determined on a case-by-case basis in consultation with an internal dosimetry specialist. As part of the program technical basis, the uranium mixtures need to be determined. In addition, determinations should be made at the time of identified incidents of potential intake. Methods for such determination may include radiochemical analysis or chemistry followed by mass spectrometry.

Solubility is of major importance in uranium inhalation toxicology. Soluble uranium compounds are absorbed and rapidly transported to kidney and bone, or excreted in urine. Because uranium damages kidney tissues by the same mechanisms as other heavy metals, dissolved uranium is considered to be a chemical toxicant. Dissolved uranium also deposits in bone and is retained for long periods of time, such that sufficiently enriched uranium can deliver an accumulated radiation dose sufficient to be considered a radiological hazard to bone (Morrow, 1986).

Oxides of uranium tend to exhibit inhalation Type S behavior, slightly more soluble compounds are assigned to Type M, and soluble compounds are assigned to Type F. Note that some compounds that have been classified as type S have shown a more rapid clearance from the lung than for other Type S compounds, i.e., having a 100-day effective half-life in the lung compared to the class S compounds that have a 500-day effective half-life. This may be due to the existence of mixtures having more than one physicochemical form (ICRP 1988b; NUREG/CR-5566, 1990). As uranium ages in a residual, loose contamination form, such as might be found in old duct work, glove boxes, or other such components, it can be expected to undergo slow oxidation to a more insoluble form. Thus, Type S forms of uranium may be reasonable assumptions of what to expect during many decommissioning operations.

For depleted uranium to present a chemical toxic hazard from inhalation, the depleted uranium would have to be subdivided into soluble particles that can be inhaled, transported into the lungs, and transferred to the blood for transport to the kidneys. Depleted uranium

metal is not readily subdivided into small, respirable particles. However, depleted uranium metal can slowly oxidize under ambient environmental conditions (corrosion), resulting in formation of small particles. The solubility of uranium oxides may be affected by the rate of oxidation, which will vary with the temperature and water vapor present in the air.

Following an accidental release from a nuclear reactor, fission and activation products may be present in fragments of irradiated fuel, of which the matrix is predominately uranium oxide (Devell, 1988; Begichev et al., 1989; Toivonen et al., 1992). Studies of the in vitro dissolution of particles released from the Chernobyl accident, seven out of ten of which consisted mainly of uranium (Cuddihy et al., 1989), were consistent in assigning all the gamma-emitting radionuclides to Type M (ICRP 71).

Particle size is an important consideration for inhalation exposures. The normal practice for an aerosol is to identify the activity median aerodynamic diameter (AMAD) and its associated particle-size distribution. Particle sizes of 10 µm or less are considered respirable. For compliance with 10 CFR Part 835, the common practice is to assume a 5-µm particle size for dosimetry purposes unless actual particle size information is available. Particle size data are most readily obtainable for chronic exposure situations. Unless representative air sampling is performed in the immediate proximity of a worker during abnormal working conditions, the practical likelihood of obtaining good particle-size information is slim.

5.3 Scope of Bioassay Program

For Types F and M uranium compounds, the monitoring programs need to be designed to maintain exposures, including those from single acute intakes, below levels that will cause transient kidney damage due to the heavy metal toxicity of uranium. Typically, urine sampling is the preferred method of monitoring for Types F and M uranium. For Type S natural uranium and all types of highly enriched uranium, radiological considerations are more limiting. In addition, local factors concerning the diversity of chemical forms of uranium must be taken into account when designing a bioassay monitoring program. For these materials, a combination of direct and indirect monitoring may be required.

5.3.1 Classification of Bioassay Measurements

Bioassay measurements can be classified according to the primary reason for their performance. This is a useful practice for historically documenting why a worker participated in a bioassay program. Numerous reasons for bioassay measurements may be defined for specific facilities; some suggested common classifications are as follows:

a. **Baseline measurements** are used to establish a pre-exposure condition, either for a new employee or as a result of a new work assignment. DOE-STD-1098-2008 recommends baseline measurements if workers are considered likely to receive intakes resulting in greater than 100-mrem CED. It is a good practice to perform such

measurements for newly hired employees, intra-company transferees, or workers transferred from facilities where bioassay measurements may not have been required. In addition, baseline measurements can verify workers' status for special work assignments. For uranium bioassay, baseline measurements made before any occupational exposure can be expected to yield only dietary levels of uranium in urine or feces.

b. A special consideration is the evaluation of intakes that include natural materials such as uranium. The sensitivity of urine sampling as a uranium bioassay tool is limited by the presence of environmental levels of uranium, which is subject to some uncertainty in interpretation. Knowledge of background level of uranium excretion is an important factor in analysis and interpretation of urine or feces for uranium bioassay purposes. In ICRP Publication 23 (1975), model values for excretion of uranium by Reference Man are given as 0.05 to 0.5 µg/day in urine and 1.4 to 1.8 µg/day in feces (ICRP, 1975). There are two distinct decisions to be made: whether a result differs from an analytical blank, and if so, whether the amount detected is greater than what would be expected in a population that is not occupationally exposed (Long et al., 1994). For example, the internal dosimetry program at Hanford distinguishes between the environmental decision level L_C and the analytical decision level DL (Carbaugh et al., 1995) using inductively coupled plasma mass spectrometry (ICP/MS) to look for the presence of ^{236}U. Since the ^{236}U isotope does not occur in nature, it is used as a flag to indicate occupational exposure.

c. Exempting workers from baseline bioassay implies accepting any detectable results as likely attributable to current occupational exposure. However, requiring baseline measurements can potentially impact the schedule of short-term jobs; the time required to obtain a chest count and a large-volume urine sample may add a day or two delay to entry procedures. Moreover, missing a baseline for a long-term employee who will be placed on a routine bioassay program is not likely to be as troublesome as not obtaining a baseline for a short-term worker who provides a termination sample that shows detection of uranium after the worker has left the site and is difficult to reach for follow-up.

d. **Routine, or periodic, measurements** are performed on a predetermined schedule (e.g., an annual or quarterly frequency).

e. **Special bioassay measurements** are performed as follow-up to unusual routine results or suspected intakes.

f. **End of assignment or termination measurements** are performed following completion of specific work or at the time of termination of employment. DOE-STD-1098-2008 recommends that workers who participate in bioassay programs have

appropriate termination measurements.

Bioassay classification is important because the purpose of a sample may affect the collection and analysis or monitoring method chosen. For example, single-void urine samples are not adequate for routine monitoring of potential uranium exposure, but can provide important information for dose-reduction therapy following a suspected intake; samples representative of excretion over a 24-hour period should be collected for quantitative intake and dose assessment. The date of sample collection (and possibly the time of collection) can be very important to special monitoring performed to assess intake. However, these are much less important with regard to periodic monitoring, for which measurements are not expected to show detectable activity and when any detection whatsoever is likely to initiate investigation and special bioassay.

5.3.2 Monitoring Requirements and Selection of Employees

Workers who are considered likely to have intakes resulting in excess of 0.1 rem CED are required to participate in a bioassay program. The workers at highest risk of incurring an intake are the ones in closest contact with the material. Typically, these are the operators, maintenance, and radiological control personnel handling uranium or uranium-contaminated objects in the course of routine glove-box, maintenance, or decommissioning operations. In the event of containment system failure, it is these workers who will most likely incur exposure and subsequent intake. These workers should be on a routine bioassay program designed to meet the requirements of 10 CFR Part 835 as a kind of safety net to identify intakes which might have gone undetected by workplace monitoring.

Other workers (e.g., supervisors, inspectors, observers, guards, and tour groups) who work in or visit a uranium facility but are not directly working with the material or contaminated objects are normally at a substantially lower risk for incurring an intake. Although these people may not need to be on a routine bioassay program, they should be subject to participation in a special bioassay program if workplace indications suggest loss of control or containment.

Routine bioassay monitoring should be implemented whenever quantities of uranium handled exceed the respective quantities in Table 5-4. Although these values were derived using the older ICRP 30 models (1979), on this kilogram scale, there would not be much difference with the newer models.

Table 5-4. Minimum Uranium Bioassay Monitoring[a,b]

TYPES OF OPERATION	MASS	ACTIVITY AMOUNT[c]
Processes in open room on bench top, with possible escape from process vessels	0.5 kg[d]	320 µCi
Process with possible escape of uranium that are carried out within a fume hood of adequate design, face velocity, and performance reliability with approved method of usage (e.g. sash height)	5 kg	3,200 µCi
Process carried out within gloveboxes that are ordinarily closed, but with possible release from process vessels and occasional exposure to contaminated box and leakage	50 kg	32,000 µCi

(a) From ANSI/HPS N13.22-2013 (2013a).

(b) Values chosen as conservative for any transportability class or mixture of isotopes of uranium. For a particular type of operation, the value of mass or activity that is more restrictive for the mixture should be used.

(c) Obtained from DAC values for pure ^{235}U (see Appendix A.2 of ANSI/HPS N13.22-2013 (2013a)).

(d) From ANSI/HPS N13.22-2013 (2013a), Appendix A.1.

5.3.3 Selection of Bioassay Monitoring Techniques

Bioassay monitoring techniques fall into two broad categories: direct measurement of radioactive materials in the body (in vivo counting) and analysis of material removed from the body for laboratory (in vitro analysis). In vivo counting includes measurements of the chest, lung, skeleton, liver, and wounds. In vitro measurements include urinalysis, fecal analysis, and occasionally analysis of tissue, sputum, or blood samples. Methods for in vitro analysis include liquid scintillation counting, fluorescence measurements, gamma spectrometry, chemical separation followed by electrodeposition, and counting with radiation detectors. A brief overview of bioassay techniques and capabilities has been developed (Selby et al., 1994). Further discussion of the techniques is provided below.

In addition, to ensure that adverse chemical toxicity effects are unlikely, bioassays for uranium should be performed when intakes of 1 mg or more of soluble uranium are likely to occur in any one work day (ANSI, 2013a).

5.3.3.1 In Vivo Counting

Direct bioassay (in vivo counting) is the measurement of radiations emitted from radioactive material taken into and deposited in the body. Direct bioassay is appropriate for detection and

measurement of photons emitted by uranium and its decay products. Lung, wound, and skeleton counting are examples of in vivo monitoring most commonly used for uranium and its progeny.

When direct bioassay is used, the detection system should be calibrated for the radionuclides to be measured in the appropriate organs. All calibration procedures, calibration records, and quality control data should be maintained.

A uranium facility should have the capability to detect and assess depositions of uranium in the lungs of affected workers. The major objective of lung counting is to provide measurements of suspected intakes triggered by workplace monitoring results. Lung measurements should be made to provide an early estimate of the magnitude of the intake and resulting lung deposition.

The most widely used systems for lung counting are high-purity germanium detectors, thin sodium-iodide detectors, phoswich detectors, and proportional counters. Multiple high-purity germanium detectors have advantages over the other detector systems because of their good resolution, allowing better identification of the radionuclide, better detectability, and better background prediction capability. The main disadvantages of germanium detector arrays are their higher cost and lower reliability relative to other types of in vivo detectors. Germanium detectors also must be continuously cooled with liquid nitrogen.

For natural and enriched uranium, the energy most commonly used for in vivo monitoring is the 185-keV gamma that is emitted with 54% abundance from the decay of ^{235}U (ANSI, 2013a; Gerber and Thomas, 1992). For natural uranium, approximately 50% of the activity is due to decay of ^{234}U. For enriched uranium, ^{234}U is the major contributor to total activity. To calculate dose, one needs to know the total uranium activity and the isotopic distribution of the material.

For natural or depleted uranium, detection of the 92.4-keV and 92.8-keV K x-rays emitted by the ^{234}Th daughter product are most commonly used (ANSI, 2013a; Gerber and Thomas, 1992). This monitoring method would not be appropriate for freshly separated uranium as the ^{234}Th will not be in equilibrium with the ^{238}U and would potentially result in an underestimate or overestimate of the actual internal burden.

Measurement equipment to detect and measure uranium contamination in wounds should be available at all uranium facilities. Instrumentation used for this purpose may include thin-crystal NaI(Tl), intrinsic germanium, or Si(Li) detectors. Correction for depth due to absorption of photons in the overlying tissues should be considered. Collimated detectors are useful for determining the location of the uranium in wounds.

Estimates of the depth of uranium contamination in a wound may be made using solid-state

germanium or Si(Li) detectors to measure the relative absorption of the low-energy x-rays emitted by uranium. Information about depth is important for determining whether tissue excision is necessary to remove the contamination.

5.3.3.2 In Vitro Analysis

The two most common forms of in vitro analysis are urinalysis and fecal analysis.

Urinalysis. Urine sampling provides useful information about the amount of uranium excreted following an intake. After chemical isolation, the uranium in urine samples may be determined by alpha spectrometry (gas-flow proportional or surface-barrier detection), alpha counting (zinc sulfide or liquid scintillation counting), or track counting. Analytical procedures for in vitro measurement of uranium and other radionuclides have been published (Volchok and dePlanque, 1983; Gautier, 1983).

Urine samples should be collected away from the uranium facility to minimize cross-contamination. Samples should be collected in contamination-free containers; measures should be considered for minimizing plateout on walls of container surfaces (such as by addition of trace amounts of gold, oxalate, or nitric acid).

Fecal Analysis. Fecal analysis is a useful procedure for evaluating the excretion of uranium and many other radioactive materials because more than half of the material deposited in the upper respiratory tract is cleared rapidly to the stomach and gastrointestinal (GI) tract.

The total fecal plus urinary elimination for the first few days after exposure, combined with in vivo counts that might be obtained, may provide the earliest and most accurate assessment of intake. Fecal samples taken during the second and third day after an inhalation incident are likely to provide the most useful data because the gastrointestinal hold-up time may vary from a few hours to a few days.

Fecal sampling is primarily a monitoring procedure for confirming and evaluating suspected intakes, but is used at some uranium facilities for routine periodic monitoring as well. Workers may find fecal sampling unpleasant or objectionable, and laboratory technicians may also have aversion to fecal sample analysis. Some of these problems may be minimized if commercial fecal sample collection kits are used for convenient collection and handling of samples. Collection kits also provide a means for collecting uncontaminated samples. Fecal samples may require additional sample preparation before analysis.

5.4 Establishing Bioassay Frequency

The bioassay measurement frequency should be based on: 1) the potential risks of an intake occurring; and 2) the sensitivity of a bioassay program to detecting potential intakes. The bioassay program sensitivity can be selected using specified intervals between measurements

based on the MDD associated with an interval.

The rationale for the selected bioassay measurement frequency should also be documented. It is appropriate to evaluate the probability of intake and to modify the sampling frequency based on that probability.

The frequency of bioassay measurements should normally not be decreased because analytical results are below the detection level. The bioassay program should be maintained to confirm the proper functioning of the overall internal exposure control program and to document the absence of significant intakes of radionuclides.

5.4.1 Frequency Based on Program Sensitivity

The minimum detectable dose concept refers to the potential dose associated with an MDA bioassay measurement at a given time interval post-intake. The pattern of retention of activity in the body, the MDA for a bioassay measurement technique, and the frequency with which that technique is applied define a quantity of intake that could go undetected by the bioassay program. An intake of such a magnitude would not be detected if it occurred immediately after a bioassay measurement and if it were eliminated from the body at such a rate that nothing was detected during the next scheduled measurement. The dose resulting from such an intake would be the MDD for that particular measurement technique and frequency.

Estimates of MDD in terms of CED should be documented for each measurement technique, MDA, and frequency. The MDA is defined in ANSI/HPS N13.30 (2011b) as a measure of the detection limit. Analytical radiobioassay laboratories should meet the Acceptable MDAs (AMDAs) recommended in ANSI N13.30-2011 (2011b) as a minimum. The AMDAs for U bioassay are shown in Table 5-5.

Table 5-5. Categories and Performance Criteria for Uranium Bioassay

Direct Bioassay		
CATEGORY	**ORGAN**	**AMDA[(a)]**
Measurement of ^{234}Th	Lung	3 nCi*
Measurement of ^{235}U	Lung	0.2 nCi
* Based on 10 mg ^{238}U.		
Indirect Bioassay		
CATEGORY	NUCLIDE	AMDA[(a)]
Alpha (Urine)	^{234}U, ^{235}U, ^{238}U	0.1 pCi/L
Isotope specific measurements		
Mass determination	Uranium (natural)	5 µg/L

(a) Note: The "Acceptable MDAs (AMDAs)" were removed from later drafts of the ANSI standard due to possible misinterpretation of the word "acceptable". The AMDAs have been replaced with test ranges for externally conducted quality control tests that take into consideration the need to be several times MDA or more before reasonably low coefficients of variation can be obtained for individual sample measurements. In this way, bias as well as precision can be estimated from reasonably small samples within each test category.

Retention functions specific to the various chemical forms and particle size distributions found in the facility should be used. Examples of MDD tabulations can be found in La Bone et al. (1993) and Carbaugh et al. (1994). In establishing MDD tables, it is important to consider dose contributions from all appropriate radionuclides in any mixture, rather than just the dose contribution from the bioassay indicator nuclide.

The minimum frequency for routine bioassay programs is interrelated to action levels, as specified in Table 5-6 (ANSI, 2013a). Special bioassays are taken as needed.

Table 5-6. Minimum Suggested Frequencies for Routine Bioassay for Uranium[(a)]

ABSORPTION TYPE	SITUATION	FREQUENCY		
		URINE	FECAL	IN VIVO
	Radiological			
F		Monthly	(b)	(b)
M		Quarterly		Annually
S		(b)		Annually
	Chemical Toxicity			
F and M		Monthly		Annually[(c)]

(a) From ICRP Publication 72 (1995c).
(b) The method of analysis not usually used.
(c) For Type F.

5.4.2 Frequency Based on Potential Risk of Intake

Although uranium workers are not generally considered to be at high risk of incurring intakes that might result in CEDs of 0.1 rem or more, any uranium worker can be considered to have the potential for such an intake (see Section 5.3.2). However, having the potential for intake does not mean that they are likely to incur an intake.

Workers who have the highest potential risk for an intake are those most closely working with uranium or uranium-contaminated material. Typically, these workers are glove-box workers, maintenance workers, and operational radiological control surveillance staff. These workers should be on a routine uranium bioassay program, including urinalysis and in vivo measurements. Such programs are relatively insensitive compared to the 0.1 rem CED monitoring threshold and are a safety net intended to catch intakes of significance relative to regulatory limits, rather than substantially lower administrative levels. Selection of bioassay frequency depends on the facility experience with potential intakes, the perceived likelihood of intake, and the MDD of a program. Annual urinalyses and in vivo chest counts are fairly typical. More frequent (e.g., semi-annual or quarterly) measurements may permit more timely review of workplace indicators in the event that an abnormal bioassay result is obtained, but do not necessarily mean a more sensitive program.

5.4.3 Special Bioassay as Supplements to Routine Bioassay Programs

Special bioassay programs for workers with known or suspected acute inhalation intakes of uranium or other alpha-emitting radionuclides should include both urine and fecal sampling. Special bioassay measurements should be initiated for each employee in a contaminated work area when surface contamination is detected by routine surveillance if it is possible that the

contamination resulted in a CED of 0.1 rem or greater. Excreta samples should not be collected where they may be contaminated by external sources of uranium. Ideally, total urine and feces should be collected for about a week following intake. This permits a sensitive assessment of potential intake and internal dose. Longer-term special samples collected at various times from a month to a year following intake can help to discriminate between ingestion, Type M inhalation, and Type S inhalation.

5.4.4 Long-term Follow-up Bioassay Programs

Following an intake, a long-term follow-up bioassay program may be required for a worker to compare the actual excreta or in vivo results with those projected by the evaluation. This is important to verify the accuracy of intake and dose assessments. The frequency and duration of a special program is dependent upon the projected values; it is suggested that as long as a worker continues to have detectable bioassay results, he or she should continue to be monitored. It is particularly important to have good baseline data and projections for individuals who return to uranium work.

The ability of a bioassay program to distinguish between an established, elevated baseline and a new potential intake is important in the continued monitoring of workers once an intake has occurred. Because of statistical fluctuations in low-level uranium measurements, it can be very difficult to identify a new intake by routine bioassay if a worker has an elevated baseline.

5.4.5 Other Frequency Situations

For chronic exposures to soluble uranyl compounds approaching the occupational exposure limits, more frequent bioassays should be taken. Some suggested frequencies are to sample after each work break and to sample at the beginning or end of the work week.

If exposure to pure class Y material occurs, monitoring may be done either by fecal analysis, or urinalysis methods with lower MDAs. As a minimum, the monitoring must be adequate to show compliance with the dose limits (10 CFR § 835.402(d)). Increased frequency is one way to lower MDAs for urinalysis for the average of a number of measurements.

5.5 Administration of a Bioassay Program

Administering a bioassay program requires that the policies, procedures, materials, support facilities, and staff be in place to enable a bioassay program to commence. Among the administrative items to address are the following:

a. Management policy requiring participation in bioassay program by appropriate workers (may be part of an overall radiation protection policy)

b. Implementing procedures (e.g., criteria for who should participate, scheduling, sample

kit instructions, sample kit issue/receipt, follow-up to unsuccessful sample or measurement attempts, data-handling)

c. Arrangements with appropriate analytical laboratories, including specifications of analysis sensitivity, processing times, reporting requirements, and quality assurance provisions

d. Onsite support facilities (e.g., sample kit storage locations, sample kit issue/collection stations, measurement laboratory facilities, equipment maintenance)

e. Staff selection, qualification, and training

f. Total CED from all intakes during a year

g. Committed equivalent dose to organs or tissues of concern from all intakes during a year

h. Magnitude of intake for each radionuclide during a year

i. Data necessary to allow subsequent verification, correction, or recalculation of doses

j. Gestation period equivalent dose to the embryo/fetus from intake by the mother during the entire gestation period

Recommendations for testing criteria for radiobioassay laboratories are in ANSI N13.30. These recommendations include calculational methods and performance criteria for bias, precision, and testing levels. The establishment of minimum detection capability must be driven by programmatic needs, ideally related to some concept of a minimum detectable dose, rather than as a single magnitude number.

Some sites have established brief flyers or brochures describing their bioassay measurements. These may be distributed to workers during classroom training, upon notification of scheduled measurements, or at the time of the measurement or sample.

The choice of the measurement technique, or of a combination of techniques, depends on the radioisotopes, physicochemical forms, and exposure pathway.

Because of the wide range of chemical and physical forms of uranium, an appropriate bioassay program is one that does not rely on assumed transportability and will provide data from which radiation dose can be calculated that will not be dependent on the chemical form. This will normally require both in vivo and in vitro bioassay. If the uranium being handled has been shown to be of medium to high transportability, then the bioassay program must be designed

to demonstrate that 3 µg U/g kidney has not been exceeded.

Uranium Type S materials cannot be effectively detected at the levels listed in ICRP Publication 54 by ordinary methods available for either lung in vivo counts or urinalysis (ICRP, 1989). This is shown by the fact that the DIL (based on 0.3 ALI as per ANSI/HPS N13.22-2013 (2013a)) was 0.06 pCi L^{-1}, which is below the MDA suggested as reasonable for routine uranium alpha urinalysis (0.1 pCi L^{-1}) in the standard. A combination of urine and fecal sampling for Type S materials will allow for adequate detection of uranium.

5.5.1 In Vivo Monitoring

The scheduling and measurement process for obtaining in vivo measurements is usually straightforward. Workers are scheduled for the measurements and results are available shortly after the measurement is completed. The long counting times can impose limitations on the throughput of workers through a measurement facility, making scheduling an important issue. Procedures should be in place to ensure that workers arrive for scheduled measurements and that follow-up occurs when a measurement is not completed or a worker fails to show.

Occasionally, workers are found who are claustrophobic when placed inside in vivo counter cells. Leaving the cell door partially open may help reduce some of the anxiety, but will also likely compromise the low background for which the system is designed.

Many workers want to know the results of their measurements. While a simple statement by the in vivo measurement technician may be adequate, a form letter stating that results were normal (or showed no detection of any of the nuclides of concern) can provide permanent verification. If results are not normal, a form letter can also be used to explain what happens next.

In vivo analysis is most useful for characterizing inhalation exposure of Type M or S compounds of uranium by lung counting. MDAs are generally not sufficiently low to provide reliable information about systemic distribution of soluble uranium at occupational levels. The ^{235}U decays with emission of an energetic (186-keV) photon in high abundance that is used for in vivo monitoring of enriched uranium workers. The other long-lived uranium isotopes emit only low yields of low-energy photons (<60 keV), which are easily attenuated by body tissue and have limited usefulness for in vivo analysis. Internal exposures to aged depleted uranium can be measured in vivo by taking advantage of several photons of moderate energy (63-93 keV) emitted by the ^{234m}Pa daughter of ^{234}Th, which are both short-lived daughters of ^{238}U.

An important aspect of any in vivo measurement program is the calibration and verification testing of the measurement equipment. In vivo measurement results are highly dependent on the determination of a background result. Likewise, calibration using known activities in

appropriate phantoms is also important.

5.5.2 Urine Sampling

Urine sampling programs can be effectively administered using either workplace or home collection protocols. Workplace sampling protocols must determine whether adequate precautions are taken to prevent external contamination of the sample by levels of activity well below the detection capabilities of friskers and workplace monitors. Home collection protocols have the advantage of being sufficiently removed from the workplace to render essentially nonexistent the potential of very low-level contamination of the sample from external sources of uranium. Avoidance of very minor external contamination of the samples is extremely important due to the dosimetric implications of uranium in urine.

Large-volume urine samples are necessary for bioassay monitoring due to the very small urinary excretion rates. Ideally, 24-hour total samples would be preferred; however, such samples often impose substantial inconvenience on workers, resulting in noncompliance with the instructions. As an alternative, total samples can be simulated by either time-collection protocols or volume normalization techniques.

One method of time-collection simulation (NCRP 87, 1987a; Sula et al., 1991) is to collect all urine voided from 1 hour before going to bed at night until 1 hour after rising in the morning for two consecutive nights. This technique has been reviewed with regard to uranium (Medley et al. 1994) and found to underestimate daily urine excretion by about 14%. Such a finding is not unexpected, since the time span defined by the protocol is likely to be about 18 to 22 hours for most people.

The volume normalization technique typically normalizes whatever volume is collected to the ICRP Reference Man daily urine excretion volume of 1400 ml. Reference Woman excretion (1000 mL/d) may be used for gender-specific programs. As a matter of practicality, routine monitoring programs do not usually use gender as a basis of routine data interpretation, particularly since results are anticipated to be nondetectable under normal conditions.

A third method calls for collection of a standard volume (e.g., 1 liter) irrespective of the time over which the sample is obtained. This method uses the standard volume as a screening tool only for routine monitoring. It does not attempt to relate measured routine excretion to intake, relying on well-defined and timely supplemental special bioassay to give true or simulated daily excretion rates.

The most common sample collection containers are 1-liter polyethylene bottles. Although glass bottles are also used, they pose additional risks of breakage. Wide-mouthed bottles are preferred for convenience and sanitation. The number of bottles included in the kit should be appropriate to the protocol; for a total 24-hour protocol, as much as 3 liters can be expected.

Special provisions, such as a funnel or transfer cup, may improve the esthetics of sample collection and provide for added worker cooperation.

Some concerns can exist with length of sample storage before analysis. Storage may come from delays before batching samples in-house or due to transportation times to an offsite laboratory. The longer a sample stands, the more chemical and biological change it can undergo, typically manifesting itself as sedimentation and plate-out on container walls. While samples can be preserved by acidification or freezing, good radiochemistry techniques should ensure essentially complete recovery of any plate-out or sediment. Samples sent offsite for analysis can be preserved with acid, but this method imposes hazardous material shipping requirements. Freezing samples can preserve them, but plate-out and sedimentation upon thawing should still be expected.

Precautions are necessary if a lab uses an aliquot for analysis and extrapolates the aliquot result to the total sample. The aliquoting procedure should be tested using spiked samples to determine if it is representative.

A quality control (QC) verification program should exist for laboratory analyses, including use of known blank samples and samples spiked with known quantities of radioactivity. Ideally, the samples should not be distinguishable by the analytical laboratory from actual worker samples. The number of QC verification samples may range from 5% to 15% of the total samples processed by a large-volume program; a small program focused on submittal of special samples following suspected intakes may have a much higher percentage of controls. An additional QC provision may be to request the analytical lab to provide results of their in-house QC results for independent review.

There are no standard or regulatory requirements for bioassay sample chain-of-custody provisions, nor has there been consensus on their need. Tampering with samples has not been a widely reported or suspected problem. Site-specific chain-of-custody requirements should be based on balancing the need with the resources required to implement them. Some sites have no chain-of-custody requirements associated with bioassay sample collection. At other sites, a simple seal placed on a sample container following collection by the subject worker is an effective means of providing a small degree of chain-of custody. At the more complex level would be strict accountability requiring signature of issue, certification of collection, and signature of submittal.

Procedures describing details of the bioassay program should be documented. These procedures should include a description of sample collection, analysis, calibration techniques, QC, biokinetic modeling, and dose calculational methods used.

5.5.3 Fecal Sampling

Fecal analysis is most useful in the first few days after a known acute exposure, since a large fraction of either an ingestion or inhalation deposition is excreted in feces. Chronic inhalation exposures to Type M or S uranium can also be characterized by fecal analysis, since a large fraction of the material clears to the GI tract and is eliminated in feces. Urinalysis is the only reliable method for determining inhalation exposures to Type F uranium and for monitoring the excretion of systemic uranium. It also provides complementary information, which, when used with in vivo or fecal monitoring results, contributes to greater accuracy in internal dose assessments. Because urinalysis is generally less disruptive to work schedules than in vivo monitoring and more acceptable to workers than fecal monitoring, it occupies a prominent place in most uranium bioassay programs.

Fecal analysis is often more likely to detect exposure to highly insoluble Type S material than urinalysis. The ratio between the fecal excretion level per day and the urine excretion level per day is greater than 7, as calculated for a 90-day sampling interval. All action levels are above the typically attainable MDA for fecal analysis of 0.1 pCi per L (ANSI, 2011b). Thus, it is recommended that facilities that have a significant Type S uranium exposure potential should have fecal analysis capabilities available to them, unless they have urinalysis methods that have MDAs well below the 0.1 pCi per sample (ANSI, 2013a).

A fecal sampling program must be designed to optimize worker cooperation, whether collecting samples at home or in the workplace. Since the frequency of fecal voiding varies greatly from person to person, the sample collection program must be adaptable. Flexibility in sample dates is important. It is suggested that when a fecal sample is required, the worker be provided with a kit and instructed to collect the sample, noting the date and time of voiding on the sample label. This practice can reduce the likelihood of unsuccessful samples. If multiple samples are required (for example, to collect the total early fecal clearance following an acute inhalation exposure), the worker may be given several kits and told to collect the next several voidings, noting the date and time of each.

Since the total fecal voiding should be collected, thought must be given to the kit provided. Fecal sampling kits can be obtained from medical supply companies or designed by the site. A typical kit might include a large plastic zipper-closure bag to hold the sample, placed inside a 1- to 2-liter collection bucket with a tight-fitting lid. The bucket and bag can be held in place under a toilet seat by a trapezoid-shaped bracket with a hole through it sized to hold the bucket. After sample collection, the zipper bag is sealed, the lid is snapped tight on the bucket, and the bucket placed in a cardboard box.

Following collection, the provisions for sample handling, control, analytical, and QC are similar to those described above for urine samples. One particular concern for fecal analysis is the potential difficulty of dissolving class S uranium in the fecal matrix. While nitric acid

dissolution may be adequate, enhanced digestion using hydrofluoric acid may be preferred.

5.5.4 Conditions for Adjustments of Action Levels

When workers are potentially exposed to other radiation sources or toxic agents, the action levels should be reevaluated. Since uranium has both chemical and radiological toxicity characteristics, urinalysis results should be interpreted both in terms of mass and radioactivity to ensure that the most appropriate set of action levels is used (ANSI, 2013a).

5.6 Interpretation of Bioassay Results

Bioassay measurements detecting uranium in workers can be initially interpreted as indicating that occupational intakes may have occurred. Standard bioassay procedures are not sufficiently sensitive to detect or differentiate occupational intakes from the range of environmental background levels in vivo or in excreta. For example, there may be significantly elevated uranium bioassay results in certain populations who obtain their drinking water from wells. Since most uranium bioassay measurement procedures include counting for radioactivity as the final step in the measurement process, they are also subject to the statistics associated with the counting process.

Two key questions associated with bioassay data are:

1) When does a sample result indicate the presence of something (i.e., when is the analyte detected?

2) What is the overall capability of the bioassay method for continual assurance of detection of the analyte?

The decision level L_c (also called the critical level for detection) is the level for a given measurement that indicates the likely presence of the analyte. The L_c is dependent on the probability of obtaining false positive results (type I, or alpha, error) that is acceptable to the program. A 5% probability of false-positive results is a common design parameter of measurement programs, implying that for a large number of measurements, 5% of the time results will be indicated as positive when in fact there is no activity present. The L_c is calculated from results of analyses of blank samples. Once a measurement is performed, it is appropriate to compare it with the L_c to determine whether or not the result is "positive" (i.e., the analyte is detected).

The MDA is the level at which continued assurance of detection can be provided. The MDA is a function of the probabilities of both false positive and false negative (type II, or beta) errors and is typically based on a 5% probability for each kind of error. The MDA is also determined from analysis of blank samples, but is substantially higher than the L_c. The MDA is appropriate for use in designing bioassay programs and as the basis for estimating minimum detectable

intakes and doses as indicators of program sensitivity. The MDA should not be used as a comparison with actual measurements to determine whether or not activity is present (i.e., <MDA is not an appropriate use of the concept).

Methods for calculating both L_c and MDA are given in ANSI N13.30 (2011b).

As an alternative to the L_c and MDA of classical statistics, there have been proposals (Miller et al., 1993) to use Bayesian statistical methods for evaluating bioassay data.

General follow-up actions to abnormal bioassay measurements should include data checks, timely verification measurements, work history reviews, and performance of special in vivo measurements or excreta sample analyses for intake and dose assessments.

5.6.1 In Vivo Count Results

In vivo uranium measurements are generally relatively insensitive with regard to levels of occupational exposure concern. This applies particularly to routine chest or lung counting, skeleton counting, and liver counting. For that reason, any detection of uranium should be investigated. The investigation should address the validity of the measurement by reviewing the spectrum and its associated background subtraction. These reviews are particularly important if the result is near the L_c. Follow-up to a positive result should include a confirming measurement. Ideally, this should be an immediate (same day) recount of equal or higher sensitivity. The farther removed in time a verification measurement is from the original measurement, the more important it becomes to factor in potential lung clearance in comparing the two measurements. A follow-up measurement taken 30 days after an initial high-routine may not be capable of providing verification if the material of concern exhibits type M behavior.

Chest-wall thickness has a significant impact on chest counting. Corrections are commonly made using a height-to-weight ratio or ultrasonic methods (Kruchten and Anderson, 1990).

Corrections may be required to address apparent detection in one tissue resulting from photon crossfire from another tissue. For example, chest counting is performed primarily to estimate activity in the lung. Yet, there is substantial bone over the lungs (rib cage, sternum) and behind the lungs (vertebrae). Plutonium and uranium are both bone-seeking radionuclides which will deposit on those bone surfaces and can interfere with chest counting. It is possible for a person having a systemic burden of uranium from a wound in the finger to manifest a positive chest count from material translocated to the skeleton, axillary lymph nodes, or liver (Carbaugh et al. ,1989; Graham and Kirkham, 1983; Jefferies and Gunston, 1986). Interpreting such a chest count as a lung burden can render dose estimates somewhat inaccurate.

When comparing in vivo measurements made over many years, it is important to make sure that the measurements are, in fact, comparable. One consideration is to make sure that corrections have been consistently applied to all similar measurements. It is not unusual for measurement systems to be replaced or to change the algorithms used for calculating results over time. Step changes in data can occur and should be addressed in monitoring long-term detectable trends (Carbaugh et al., 1988).

In vivo wound counting for uranium is usually one facet of special bioassay. While a portable alpha survey meter may show if surface contamination is present at the wound site or contamination of the wounding object, alpha detectors are not capable of measuring imbedded activity or activity masked by blood or serum. Thus, uranium facilities should have available a wound counter utilizing a thin sodium iodide or semi-conductor (e.g., planar germanium) detector. Such detectors are capable of measuring the low-energy photons emitted from uranium. The ability to accurately quantify wound activity is highly variable, depending on the calibration of the equipment and how deeply imbedded material is in the wound. If the object causing a wound and blood smears taken at the time of a wound show no detectable activity, then a wound count also showing no detectable activity is probably sufficient to rule out an intake. If the wounding object or the blood smears show detectable activity, special urine samples should be obtained regardless of the wound count result. In this latter circumstance, lack of detectable activity on a wound count could be attributable to deeply imbedded material at the wound site or to rapid transportation of material from the wound to the systemic compartment.

5.6.2 Urine Sample Results

Detection of uranium activity in a routine or special urine sample using commonly available radiochemical measurement techniques should be investigated as a potential intake. A data review should be made to determine if the sample result was correctly determined, and batch QC sample data should be verified.

If the result is near the L_C, it is possible that statistical fluctuation of the measurement process could account for the apparent detection. Recounting the final sample preparation once or twice can be a helpful technique to verify a result or classify it as a false-positive. If the first recount also detects the analyte, it can be concluded that the sample does contain the analyte (the likelihood of two consecutive false positives at a 5% type I error per measurement is 0.0025, or 0.25%.) If the first recount does not detect the analyte, a second recount can be performed as a tie-breaker.

An investigation should be initiated for any abnormal uranium urinalysis result. "Abnormal" for a person with no prior history of intake should be interpreted as any detectable activity.

Once an intake is confirmed, obtaining sufficient samples is necessary to establish a

reasonably anticipated baseline against which future measurements can be compared. This is important both to provide future verification of the accuracy of the assessment and to identify potential additional intakes.

The statistical fluctuation of low-level measurements can be particularly troublesome for long-term excretion patterns. Factors of two can be easily expected due to day-to-day variability and imprecise adherence by the worker to urine collection protocols.

5.6.3 Fecal Sample Results

Fecal samples are much more sensitive to detection of intakes than are urine samples and, consequently, are an important part of follow-up bioassay monitoring for potential intakes initially identified by workplace indications. Pitfalls to the data interpretation include highly variable individual fecal voiding patterns, ranging from more than one per day to one every few days. This makes it extremely important to know what time interval is represented by a collected fecal sample. While a single set of fecal data can be normalized to a daily excretion rate for Reference Man, it is not likely to improve the quality of assessment.

The preferred fecal sampling protocol following an intake is to collect all the early fecal clearance (meaning total feces for the first 5 to 7 days). This method will allow a good estimation of inhalation or ingestion intake, but does not readily permit discrimination of inhalation from ingestion, or identify whether inhaled material exhibits Type F, M, S clearance patterns. For optimum interpretation, total fecal collection should be interpreted in light of early urine and in vivo data for preliminary estimates. The urine data is likely to be particularly valuable in conjunction with fecal data to classify an intake as Type M or S. Longer-term follow-up fecal samples at nominally 30, 60, and 90 days post-intake should substantially improve the classification of material as Type M or S.

5.6.4 Use of Air Sample Data in Internal Dosimetry

Results of air sampling and continuous air monitoring implying more than 40 DAC-hours exposure should be used to initiate special bioassay to assess intakes of uranium. Although bioassay data are the preferred method for assessing intakes and internal doses, air sample data can be used if bioassay data are unavailable or determined to be inadequate or nonrepresentative. Air sample data can be used to calculate an exposure to airborne material either in terms of DAC-hours or potential radioactivity intake as follows:

$$DAC - hours = \frac{Air\ Concentration}{DAC}\ x\ Duration\ (hours)$$

$$Intake = Air\ Concentration\ x\ Breathing\ Rate\ x\ Time$$

If air sample results are $H_{E,50}$ representative of air breathed by individuals, then doses can be

calculated using the 5-rem stochastic limit for CED (E_{50}) or the 50-rem nonstochastic limit for committed tissue dose equivalent ($H_{T,50}$) and the respective stochastic or nonstochastic DAC or ALI conversion factor, as shown below:

$$E_{50} = (DAC - hours)\ x \frac{Dose\ Limit}{2000\ DAC - hours}$$

$$E_{50} = Intake\ x \frac{Dose\ Limit}{ALI}$$

If respiratory protection is worn by workers, the appropriate respirator protection factor may be applied to the above calculations (i.e., dividing the calculated result by the protection factor.)

General air sampling programs should be augmented by breathing zone sampling when air concentrations to which individuals are exposed might be highly variable. Breathing zone sampling may include both fixed-location and personal (lapel) air samplers. Personal air samples are more likely to be representative of actual exposure conditions than are samples collected at fixed locations, and they can be particularly useful for assessing potential intakes involving short-term exposure to well-monitored air concentrations.

5.7 Dose Assessment

Understanding the behavior of uranium in the body is essential for being able to model the uranium body burden on the basis of a bioassay. Knowledge of pathways the radionuclides follow, the organs and systems that make up the pathways, the rates at which the radionuclides travel along these pathways, and the rates at which they are eliminated from the body are essential for determining radiation dose. The US Department of Health and Human Services Agency for Toxic Substances and Disease Registry, *Toxicological Profile for Uranium* (2013) details the toxicokinetics of uranium in the body.

Dose assessment involves collecting and analyzing information concerning a potential intake and developing a conclusion regarding the magnitude of intake and its associated committed dose equivalents. Dose assessments are conducted by investigating the nature of a potential intake and by analyzing bioassay measurement results or other pertinent data. Biokinetic models are used in conjunction with bioassay data to evaluate the intake, uptake, and retention of uranium in the organs and tissues of the body. Intake estimates can then be used to calculate committed effective and organ dose equivalents. It is essential that good professional judgement be used in evaluating potential intakes and assessing internal doses. A number of considerations for dose assessments have been identified (Carbaugh 1994).

Computer codes are commonly used for assessment of intakes, dose calculation, and bioassay or body content projections. An overview of what should be considered in selecting a

computer code, as well as descriptions of a number of internal dosimetry codes available in 1994, has been developed (La Bone 1994). Internal dosimetry code users should understand how the code works and be aware of its limitations. Computer codes merely provide the logical result of the input they are given. Use of a particular computer code does not necessarily mean a dose estimate is correct.

As used in this section, the definition of "intake" is the total quantity of radioactive material taken into the body. Not all material taken into the body is retained. For example, in an inhalation intake, the ICRP Publication 68 (1994b) respiratory tract model predicts that, for 5-μm particles, 82% of the intake will be deposited in the respiratory tract; the other 18% is immediately exhaled. For a wound intake, material may be initially deposited at the wound site. Once the material has been deposited, it can be taken up into systemic circulation either as an instantaneous process (e.g., direct intravenous injection of a dissolved compound) or gradually (e.g., slow absorption from a wound site or the pulmonary region of the lung). Both the instantaneous and slow absorption processes are often referred to as uptake to the systemic transfer compartment (i.e., blood). Once material has been absorbed by the blood, it can be translocated to the various systemic organs and tissues.

An understanding of this terminology is important to review of historical cases. Before DOE Order 5480.11, many sites reported internal doses not as dose equivalent estimates but as an uptake (or projected uptake) expressed as a percentage of a maximum permissible body burden (MPBB). The standard tabulated values for MPBBs were those in ICRP Publication 2 (1959). Many archived historical records may have used this approach. DOE Order 5480.11 (now superseded) required calculation of dose equivalent. Now, 10 CFR Part 835 has codified the calculation of intakes and committed doses.

5.7.1 Methods of Estimating Intake

There are several published methods for estimating intake from bioassay data (Skrable et al., 1994; Strenge et al., 1992; ICRP, 1989; King, 1987). These methods each employ an idealized mathematical model of the human body showing how materials are retained in and excreted from the body over time following the intake. An intake retention function (IRF) is a simplified mathematical description of the complex biokinetics of a radioactive material in the human body. These functions are used to predict the fraction of an intake that will be present in any compartment of the body, including excreta, at any time post-intake. Intake retention functions incorporate an uptake retention model that relates uptake to bioassay data and a feed model that relates intake to uptake and bioassay data. ICRP Publication 54 (1989) and others (Lessard et al., 1987) contain compilations of IRFs.

In its simplest form, a compartment content at any time post-intake (Q_t) can be expressed as the product of intake multiplied by the intake retention function value for compartment Q at time t post-intake, or:

$$Q_t = Intake\ x\ IRF(Q_t)$$

Results predicted by the model can then be compared with the observed bioassay data. Such results are often referred to as expectation values.

Simple algebraic manipulation of the model allows calculation of intake from the compartment content at time t, as shown below:

$$Intake = \frac{Q_t}{IRF(Q_t)}$$

When multiple data points are available for a compartment, the intake can be estimated using an unweighted or weighted least-squares fitting procedure, as described by Skrable (1994) and Strenge (1992) or as can be found in most statistics textbooks. As an alternative, data can be fit by eye to a graphical plot; however, the apparent fit can be misleading if data have been logarithmically transformed.

Intake can also be estimated from air sample data, as described in Section 5.7.4. This method is appropriate if bioassay data are not available or insufficiently sensitive. Intake estimates based on air samples and bioassay data are also appropriate as a check on each other. Valid bioassay data showing detectable results should be given preference over intake estimates based on air sample results.

5.7.2 Alternate Methods of Intake Assessment

Historically, intake as described in the foregoing section was not always calculated when assessing uranium exposures. Estimates of uptake using recognized methods (Langham 1956, Healy 1957, Lawrence 1987) focused on assessing the magnitude of radioactivity retained in the body, rather than intake (which includes material not retained and of no dosimetric significance). These methods were (and are) dosimetrically sound in so far as estimates of deposition and uptake are concerned.

5.7.3 Estimating Effective Dose from Intakes of Uranium

The committed equivalent dose resulting from an intake of uranium may be calculated by multiplying the estimated intake (I) by an appropriate dose conversion factor (DCF):

$$H_{50} = I\ x\ DCF$$

Dose conversion factors can be obtained from tabulated data in ICRP Publication 68 (1994b), or calculated directly using computer programs.

Values for simplified dose conversion factors can be obtained by dividing a dose limit by the

corresponding value for the ALI. A caution must be observed with this approach: not all tabulated values of ALIs are the same. The ALIs are commonly rounded in most tabulations to one significant figure. Substantial variation can occur as a result of unit conversion. Such rounding errors can introduce significant discrepancies in dosimetry calculations. This method also raises a question about which ALI should be used if compliance monitoring is being based on comparison with secondary limits, such as the ALI rather than the primary dose limits.

Where individual-specific data are available, the models should be adjusted. However, the general lack of capability to monitor organ-specific retention for uranium (i.e., content and clearance half-times) makes the use of default models most practical.

Ideally, one should obtain as much bioassay information as possible to determine the intake and track the retention of uranium in the body to reduce the uncertainty associated with the daily variation in the measurements. A regression analysis should be used to fit the measurement values for estimating the initial intake and clearance half-times.

5.8 Reference and Action Levels

Reference and action levels are essential to operation of a routine internal dosimetry program. Because a wide range of levels can be defined by various facilities and organizations, this document does not attempt to prescribe particular level titles. As used in this document, reference and action levels are simply workplace or bioassay measurements, or associated calculated doses, at which specific actions occur.

Notification levels based on workplace indicators for reacting to a potential intake are suggested in Table 5-9. The intent of these notification levels is to provide guidance for field response to any potential intake of radioactive material with a potential for a dose commitment that is >100-mrem CED. Table 5-10 suggests notification levels to the occupational medicine physician for possible early medical intervention in an internal contamination event. These tables, derived from Carbaugh et al. (1994), are based on general considerations and significant experience with past intakes of radioactive material and, because they are based on field measurements, do not correspond with any exact dose commitment to the worker.

Table 5-7. Uranium Levels for Internal Dosimetry Notification

Indicator	Notification Level
Nasal or mouth smears	Detectable activity
Facial contamination (direct measurement)	200 dpm
Skin breaks or blood smears	Any skin break while handling material other than sealed sources
Head, neck contamination	2,000 dpm
Contamination in respirator	Detectable activity inside respirator after use
Hands, forearms, clothing[a]	10,000 dpm
Airborne radioactivity	Acute intake equivalent to 40 DAC-hours after accounting for respiratory protection factor[b]

(a) Clothing contamination levels apply to exposure without respiratory protection, such as on inner coveralls or personal clothing.

(b) $DAC - hours = \frac{airborne\ concentration}{DAC} x\ hours\ of\ intake$

Table 5-8. Uranium Contamination Levels for Notification of Occupational Medicine Physician

Indicator	Medical Notification Level (dpm)
Nasal or mouth smears	1,000
Facial contamination	25,000
Skin breaks or wounds	100

The decision to administer treatment and the treatment protocol are solely the responsibilities of the physician in charge. The basic principle is that the proposed intervention should do more good than harm (Gerber and Thomas, 1992).

Guidelines for the medical intervention of a radionuclide intake can be found in several publications. *The Guidebook for the Treatment of Accidental Internal Radionuclide Contamination of Workers* (Gerber and Thomas, 1992) contains detailed guidance in intervention and medical procedures useful in mitigating radiation overexposures. The CEC/DOE Guidebook has been based on the ALI for action levels, rather than on CED, to overcome the problem of uncertainties in dose per unit intake. The ICRP recommends in Publication 60 (1991b) a limit of 2-rem/y (20-mSv/y) on effective dose (ICRP, 1991a). Thus, the ALIs found in ICRP Publication 61 and used in the Gerber and Thomas (1992) noted above are those which would provide a CED of 2-rem/y instead of current U.S. regulations of 5-rem/y (Gerber and Thomas, 1992).

Guidance in the CEC/DOE Guidebook can be summarized as follows:

a. When the estimated intake is below 1 ALI, treatment should not be considered.

b. When the estimated intake is between 1 and 10 times the ALI, treatment should be considered. Under these situations, short-term administration will usually be appropriate, except for intake of materials poorly transported from the lung (Type S).

c. When the estimated intake exceeds 10 times the ALI, then extended or protracted treatment should be implemented, except for materials poorly transported from the lung.

d. For poorly transported material in the lung, lung lavage is the only recommended treatment, and it is only a consideration for intakes exceeding 100 times the ALI.

Because the dose associated with the ALI in the Gerber and Thomas (1992) is 2-rem CED and because the upper administrative control level suggested by DOE-STD-1098-2008 (2009c) is 2 rem, intervention levels of 2 rem and 20 rem might be used for guidance in the manner presented in the Gerber and Thomas (1992):

a. When the CED estimated intake is below 2 rem, treatment is not generally recommended.

b. When the CED for an estimated intake is between 2 rem and 20 rem, treatment should be considered. Under these situations, short-term administration will usually be appropriate.

c. When the CED for an estimated intake exceeds 20 rem, then extended or protracted treatment is strongly recommended, except for poorly transported material in the lung.

A useful method to enhance excretion of uranium via the kidneys is the formation of radionuclide complexes using sodium bicarbonate. This type of complexation appears to be the only current method that has a reasonable chance of reducing or preventing kidney damage during the early period after incorporation of this chemotoxic heavy metal.

An initial prophylactic chelation therapy may be appropriate because bioassay measurements (particularly urinalysis) cannot usually be completed within the response time required for effective chelation therapy. Urinalysis becomes very helpful following administration of chelation therapy because there is a direct correlation between urinary excretion and dose averted because of uranium excreted. This provides a method of measuring the effectiveness of chelation therapy and determining if it is worthwhile to continue therapy. It is probable that the efficacy of treatment will decrease with continued administration as uranium is removed

and the rate of transfer into the systemic compartment decreases.

5.9 Response to Suspected Intakes

Experience has shown that most intakes of uranium are accidental. Uranium facilities and operating procedures are designed to prevent intakes. Nonetheless, it is important for management to prepare for the possibility that workers might receive an intake of uranium--even though the probability of an incident may be very small. Prompt and appropriate action following an accidental intake of uranium will allow for therapeutic measures to be taken to minimize the internal contamination and lessen the potential for harmful effects. The health physicist and medical staff should work closely to ensure that the proper course of action is followed.

All employees suspected of having received an intake of uranium should be referred for special bioassay measurements. Because a fraction of an intake by inhalation may be retained in the nasal passages for a few hours after exposure to airborne radioactive materials, any level of contamination on a nasal swab indicates an intake that should be followed up by a special bioassay measurement program. However, lack of detection on nasal smears cannot be taken as evidence that an intake did not occur either because the nasal passages can be expected to clear very rapidly or, alternatively, because the worker could be a mouth-breather. Special bioassay should also be initiated if uranium contamination is found on the worker in the vicinity of nose or mouth.

Developing specific field criteria to identify the need for medical response can be challenging. Inhalation intake estimations based on DAC-hours exposure are straightforward and discussed earlier in this document. Early bioassay measurement levels corresponding to the action levels have been calculated at Hanford and are summarized in Table 5-11. Another method is to develop field observation criteria (e.g., nasal smear or skin contamination criteria) which might imply that an action level has been exceeded. This latter approach is highly subjective with any number chosen likely to be arguable. Knowledge of facility operations, material forms, and past experience will likely play a key role in development of such criteria.

For acute intakes, direct bioassay measurements should be taken before, during, and after the period of rapid clearance of activity. Urine and fecal samples collected after known or suspected inhalation incidents should also be used to estimate the magnitude of the intake. Initial assessments of intakes from contaminated wounds are based primarily on wound count and urinalysis data.

If a significant intake is indicated, the worker should not return to further potential exposure to uranium until the intake has been thoroughly assessed and a predictable bioassay pattern established. This is particularly important because a new intake of a very low level may confound the interpretation of bioassay measurements for previous intakes of uranium.

Table 5-9. Early Bioassay Measurement Results Corresponding to the Therapeutic Intervention Action Levels Used at the Hanford Site (Carbaugh et al., 1995)

Isotope and Dose (*HE,50*)	Measurement	Result	Action	Possible Treatment
		Uranium, Soluble		
Potential kidney toxicity	Chest count	>MDA (14-21 mg)	Consider therapy	Na or Ca bicarbonate; intestinal adsorbents
	Second-void urine sample	>0.1 mg		
	12-hour urine sample	>0.5 mg		
		Uranium Insoluble[(a)]		
2 rem	Chest count	>MDA for ^{235}U or ^{234}Th	Consider therapy	None recommended
200 rem	Same	100 x ALI	Treatment strongly recommended	Lung lavage

a. If soluble component is present, then urine sampling is appropriate. Use same action levels as above for soluble uranium.

The health physicist must make important decisions for prompt action at the site of an accidental or suspected intake of uranium or other radioactive materials. Often, these decisions must be based on limited data. Information that may be available for initially estimating the amount and type of intake may include the following:

a. levels of measured contamination in the work area

b. skin contamination levels, affected areas, and whether the skin is damaged or punctured

c. wound contamination levels

d. chemical form of the material involved

e. results of air monitoring

f. nasal smear activity levels

g. sputum and/or mouth contamination

The special bioassay monitoring program is initiated following a known or suspected intake. This information is needed for dose assessment and future exposure management. The intake is confirmed if follow-up bioassay measurements indicate positive measurement results. Additional bioassay measurements may be needed to quantify the intake and provide data for determining the effective dose equivalent. The frequency of bioassay monitoring will depend on the specific case to be evaluated. Selection of the appropriate sampling frequency is based on the previously discussed performance capabilities for workplace monitoring programs, consultations with internal dosimetry specialists, and the cooperation of the affected employee.

5.9.1 Emergency Action Planning

The management at the uranium facility should be prepared to follow an emergency action plan for response to a uranium intake. If a worker accidentally inhales or ingests uranium or is injured by a uranium-contaminated object, the action plan should be initiated immediately. A rapid response is important because any delay in implementing appropriate action could lessen the effectiveness of decorporation therapy and increase the probability for internalized uranium to deposit in the kidneys or on bone surfaces.

5.9.2 Medical Emergency Response Plan

The health physicist and medical staff must establish an emergency action plan for the appropriate medical management of an accidental intake of uranium. The elements of the plan should include the following:

a. Decision levels for determining when monitoring data or accident events require emergency medical response

b. Responsibilities of the affected worker, health physicist, medical staff, and management or supervisory personnel

c. Instructions for immediate medical care, decontamination, monitoring, and longer-term follow-up response

d. Provisions for periodically reviewing, updating, and rehearsing the emergency action plan

The sequence and priority of the emergency action plan may vary with the magnitude and type of accidental conditions and their severity. An initial early assessment of the incident should focus, first, on treatment of life-threatening physical injuries and, second, on the radioactive contamination involved. Minor injuries should be treated after decontamination.

A rapid estimate of the amount of internal contamination by uranium or other alpha-emitters

may not be possible. If a significant intake (meaning one that exceeds 10 times the ALI) is suspected, medical staff should proceed with decorporation therapy after first treating major injuries.

5.9.3 Responsibilities for Management of Internal Contamination

Responsibilities should be assigned for action in response to an accidental internal uranium contamination. The affected worker has the responsibility to inform the health physicist, radiological control technician (RCT), or his immediate supervisor as soon as an intake is suspected. The health physicist or RCT should make an initial survey of the extent of the contamination and immediately contact his supervisor and, when action levels are exceeded, contact a member of the medical staff. Monitoring and radiation safety support to the medical staff and supervisors should continue during the management of the contamination incident. Care should be taken to limit the spread of radioactive contamination.

The health physicist should immediately begin to gather data on the time and extent of the incident. Contamination survey results should be recorded. Radionuclide identity, chemical form, and solubility classification should be determined. Nasal smears should be obtained immediately if an intake by inhalation is suspected. When action levels are exceeded, all urine and feces should be collected and labeled for analysis. Decontamination should proceed with the assistance of the medical staff. Contaminated clothing and other objects should be saved for later analysis.

5.9.4 Immediate Medical Care

The medical staff should provide immediate emergency medical care for serious injuries to preserve the life and well-being of the affected worker. Minor injuries may await medical treatment until after an initial radiation survey is completed and the spread of contamination is controlled. However, the individual should be removed from the contaminated radiation area as soon as possible. Chemical contamination and acids should be washed immediately from the skin to prevent serious burns and reactions.

A chelating agent should be administered by a qualified medical professional immediately following an accidental intake of uranium if the dose thresholds discussed in Section 5.9 (2 rem-20 rem) are exceeded... Sodium bicarbonate should be available for treating internal uranium contamination. The worker to be treated must first be informed of the proposed use of a chelating agent, instructed on the purpose of administering the chelating agent, and warned about the possible side-effects. The worker must then give signed consent before chelation therapy may be initiated. Even though sodium bicarbonate therapy is the only method available for reducing the quantity of uranium retained in the body, the affected worker has the right to refuse its use.

The recommended therapy for decorporation is a systemic administration of 250 mL of isotonic (1.4%) solution of sodium bicarbonate by slow intravenous injection (Gerber and Thomas, 1992). The sodium bicarbonate reacts with uranyl ions, UO_2^{++}, in body fluids to form an anionic complex, probably $UO_2(CO_3)_3$, which is rapidly excreted in urine. Treatment may be continued if bioassay indicates that decorporation therapy continues to enhance the urinary excretion of uranium. However, if treatment is extended over the days following the incident, the dosage should be adapted to prevent contraindications of alkalosis (bicarbonate solution is alkaline) and respiratory acidosis (Gerber and Thomas, 1992).

5.9.5 Contaminated Wounds

Medical treatment for contaminated wounds may include flushing with saline and decorporating solutions, debridement, and surgical excision of the wound. These measures are all the responsibility of trained medical staff operating under the direction of a physician. Radiological control personnel can provide valuable assistance by prompt assessment of materials removed from the wound and identification of magnitude of residual activity as decontamination proceeds. Decontamination should continue until all radioactivity has been removed or until risk of permanent physical impairment is reached.

11 APPENDIX A - REFERENCES

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12 APPENDIX B - GLOSSARY

Terms used consistent with their regulatory definitions.

abnormal situation: Unplanned event or condition that adversely affects, potentially affects, or indicates degradation in the safety, security, environmental or health protection performance or operation of a facility.

air sampling: A form of air monitoring in which an air sample is collected and analyzed at a later time, sometimes referred to as retrospective air monitoring.

air monitoring: Actions to detect and quantify airborne radiological conditions by the collection of an air sample and the subsequent analysis either in real-time or in off-line laboratory analysis of the amount and type of radioactive material present in the workplace atmosphere.

airborne radioactive material: Radioactive material in any chemical or physical form that is dissolved, mixed, suspended, or otherwise entrained in air.

alarm set point: The count rate at which a continuous air monitor will alarm, usually set to correspond to a specific airborne radioactive material concentration by calculating the sample medium buildup rate.

ambient air: The general air in the area of interest (e.g., the general room atmosphere) as distinct from a specific stream or volume of air that may have different properties.

breathing zone air monitoring: A form of air monitoring that is used to detect and quantify the radiological conditions of air from the general volume of air breathed by the individual, usually at a height of 1 to 2 meters. See *personal air monitoring.* (Air Monitoring Chapter of DOE G 441.1-1C)

continuous air monitor (CAM): An instrument that continuously samples and measures the levels of airborne radioactive materials on a "real-time" basis and has alarm capabilities at preset alarm set points. (Air Monitoring Chapter of DOE G 441.1-1C)

decontamination: The process of removing radioactive contamination and materials from personnel, equipment, or areas.

Department of Energy operations: Those activities for which DOE has authority over environmental, safety, and health protection requirements.

Department of Energy site: Either a tract owned by DOE or a tract leased or otherwise made available to the Federal Government under terms that afford to the Department of Energy rights of access and control substantially equal to those that the Department of Energy would possess if it were the holder of the fee (or pertinent interest therein) as agent of and on behalf of the Government. One or more DOE operations/program activities are carried out within the boundaries of the described tract.

detector: A device or component designed to produce a quantifiable response to ionizing radiation, normally measured electronically. (Portable Monitoring Instrument Calibration Chapter of DOE G 441.1-1C)

DOELAP: The Department of Energy Laboratory Accreditation Program defines a set of reference performance tests and provides a description of the minimum levels of acceptable performance for personnel dosimetry systems and radiobioassay programs under DOE-STD-1111-2013 (2013a). (External Dosimetry Program Chapter of DOE G 441.1-1C)

exposure: The general condition of being subjected to ionizing radiation, such as by exposure to ionizing radiation from external sources or to ionizing radiation sources inside the body. In this document, exposure does not refer to the radiological physics concept of charge liberated per unit mass of air. (Internal Dosimetry Chapter of DOE G 441.1-1C)

fixed contamination: Radioactive material that has been deposited onto a surface and cannot be readily removed by nondestructive means, such as casual contact, wiping, brushing, or laundering. Fixed contamination does not include radioactive material that is present in a matrix, such as soil or cement, or radioactive material that has been induced in a material through activation processes. (DOE-STD-1098)

fixed-location sampler: An air sampler located at a fixed location in the workplace.

grab sampling: A single sample removed from the workplace air over a short time interval, typically less than 1 hour.

high-efficiency particulate air (HEPA) filter: Throwaway extended pleated medium dry-type filter with 1) a rigid casing enclosing the full depth of the pleats, 2) a minimum particle removal efficiency of 99.97% for thermally generated monodisperse di-octyl phlalate smoke particles with a diameter of 0.3 µm, and 3) a maximum pressure drop of 1.0-in. w.g. when clean and operated at its rated airflow capacity. (DOE-STD-1098).

intake: The amount of radionuclide taken into the body by inhalation, absorption through intact skin, injection, ingestion, or through wounds. Depending on the radionuclide involved, intakes may be reported in units of mass (e.g., µg, mg), activity (e.g., µCi, Bq), or potential alpha energy (e.g., MeV, J) units. (Internal Dosimetry Program Chapter of DOE G 441.1-1C)

minimum detectable amount/activity (MDA): The smallest amount (activity or mass) of an analyte in a sample that will be detected with a probability, *B*, 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 (Type I error). The MDA is computed using the same value of α as used for the decision level (DL). The MDA depends on both α and *B*. Measurement results are compared to the DL,

not the MDA; the MDA is used to determine whether a program has adequate detection capability. The MDA will be greater than or equal to the DL. (Internal Dosimetry Program Chapter of DOE G 441.1-1C)

personal air monitoring: A form of breathing zone air monitoring that involves the sampling of air in the immediate vicinity (typically within one foot) of an individual's nose and mouth, usually by a portable sampling pump and collection tube (e.g., a lapel sampler) worn on the body. (Air Monitoring Chapter of DOE G 441.1-1C)

portable air sampler: An air sampler designed to be moved from area to area.

radiation-generating device (RDG): The collective term for devices which produce ionizing radiation, including certain sealed radioactive sources, small particle accelerators used for single purpose applications which produce ionizing radiation (e.g., radiography), and electron-generating devices that produce x-rays incidentally. (Radiation-Generating Devices Chapter of DOE G 441.1-1C)

radioactive material: Any material that spontaneously emits ionizing radiation (e.g., X- or gamma rays, alpha or beta particles, neutrons). The term "radioactive material" also includes materials onto which radioactive material is deposited or into which it is incorporated. For purposes of practicality, both 10 CFR Part 835 and this Standard establish certain threshold levels below which specified actions, such as posting, labeling, or individual monitoring, are not required. These threshold levels are usually expressed in terms of total activity or concentration, contamination levels, individual doses, or exposure rates. (DOE-STD-1098)

radiological work permit (RWP): The permit that identifies radiological conditions, establishes worker protection and monitoring requirements, and contains specific approvals for radiological work activities. The Radiological Work Permit serves as an administrative process for planning and controlling radiological work and informing the worker of the radiological conditions. (DOE-STD-1098)

radiological control organization: An organization responsible for radiation protection. (Sealed Radioactive Source Accountability and Control Chapter of DOE G 441.1-1C)

real-time air monitoring: Collection and real-time analysis of the workplace atmosphere using continuous air monitors (CAMs).

refresher training: The training scheduled on the alternate year when full retraining is not completed for Radiological Worker I and Radiological Worker II personnel. (DOE-STD-1098)

removable contamination: Radioactive material that can be removed from surfaces by nondestructive means, such as casual contact, wiping, brushing, or washing. (DOE-STD-1098)

representative air sampling: The sampling of airborne radioactive material in a manner such that the sample collected closely approximates both the amount of activity and the physical and chemical properties (e.g., particle size and solubility) of the aerosol to which the workers may be exposed.

source-specific air sampling: Collection of an air sample near an actual or likely release point in a work area using fixed-location samplers or portable air samplers.

survey: An evaluation of the radiological conditions and potential hazards incident to the production, use, transfer, release, disposal, or presence of radioactive material or other sources of radiation. When appropriate, such an evaluation includes a physical survey of the location of radioactive material and measurements or calculations of levels of radiation, or concentrations or quantities of radioactive material present. (DOE-STD-1098)

workplace monitoring: The measurement of radioactive material and/or direct radiation levels in areas that could be routinely occupied by workers.