ENGINEERING-PDH.com
ONLINE CONTINUING EDUCATION

GUIDE TO RADIOLOGICAL PROTECTION IN URANIUM FACILITIES - VOL 3 OF 3

Main Category:	Nuclear Engineer
Sub Category:	-
Course #:	NUC-142
Course Content:	95 pgs
PDH/CE Hours:	8

COURSE/EXAM PREVIEW

(HIT BACK TO RETURN TO ENGINEERING-PDH.COM)

NUC-142 EXAM PREVIEW

Instructions:

- Review the course & exam preview below.
- Click “Add to Cart” from the course page on the website. You can “Continue Shopping” to add additional courses, or checkout. Don’t forget to apply your coupon code if you have one before checkout.
- After checkout you will be provided with links to download the official courses/exams.
- At your convenience and own pace, you can review the course material. When ready, select “Take Exam” to complete the live graded exam. Don’t worry, you can take an exam as many times as needed to pass.
- Upon a satisfactory completion of the course exam, which is a score of 70% or better, you will be provided with your course completion certificate. Be sure to download and print your certificates to keep for your records.

Exam Preview:

1. According to the reference material, beta radiation fields are usually the dominant external radiation hazard in facilities requiring contact work with unshielded forms of uranium.
   a. True
   b. False
2. Using Table 6-2. Radiation Weighting Factors, w_R, what is the radiation weighting factor for Neutrons, energy > 20 MeV?
   a. 20
   b. 10
   c. 5
   d. 1
3. Typical portable survey instruments demonstrate a fairly flat energy response above 250 keV, while the response below 250 keV can be variable to a greater or lesser degree depending upon the instrument design
   a. True
   b. False
4. According to the reference material, potentially significant skin exposure from uranium occurs primarily from the ^{234m}Pa betas at tissue depths of _ mg/cm^2 and greater.
   a. 3
   b. 4
   c. 5
   d. 7

5. Using Table 6-9. Gamma Flux and Ratios at Various Locations and Sources at Fernald Plant which if the following integrated gamma flux values corresponds to the 675 to 1050 KeV range for an Open UO_3 barrel?
   a. 18
   b. 58
   c. 154
   d. 232
6. Beta dose rates from uranium and its decay products decrease slightly with distance from the source due to geometry and air shielding while gamma and neutron radiation decreases faster with distance due to scattering buildup.
   a. True
   b. False
7. Using Table 6-10. Performance Test Categories, Radiation Sources, and Test Ranges for the DOELAP and NVLAP Programs, which of the following test ranges corresponds to high-energy photons (low dose)?
   a. 0.1 - 50 Gy
   b. 1.5 - 100 mSv
   c. 0.3 - 100 mSv
   d. 2.0 - 50 mSv
8. The PNADs should be capable of determining gamma dose from 10 rad to 1000 rad with an accuracy of ±30% and neutron dose from 1 rad to 1000 rad with an accuracy of ±20% without dependence upon fixed-unit data.
   a. True
   b. False
9. Using Table 6-11. Uranium Beta Shielding, which of the following materials requires the least thickness, in cm, required to stop ^{234m}Pa Betas?
   a. Air
   b. Aluminum
   c. Pyrex
   d. Lead
10. Uranium facilities shall have a waste-minimization and pollution prevention program (DOE, 2011). The objective of such a program is the cost-effective reduction in the generation and disposal of hazardous, radioactive, and mixed waste. The preferred method is to reduce the total volume and/or toxicity of hazardous waste generated at the source, which minimizes the volume and complexity for waste disposal.
    a. True
    b. False

6 EXTERNAL DOSIMETRY 140
6.1 DOSE LIMITS 140
6.1.1 Limiting Quantities 140
6.1.2 Operational Quantities 144
6.2 RADIATIONS IN URANIUM FACILITIES 144
6.2.1 Alpha and Beta Doses 148
6.2.2 Gamma Doses 149
6.2.3 Neutron Dose Equivalents 149
6.3 RADIATION DETECTION AND EVALUATION 152
6.3.1 Portable Survey Instruments--Beta Radiation Response 152
6.3.2 Portable Survey Instruments--Gamma Radiation Response 156
6.3.3 Portable Survey Instruments--Neutron Response 158
6.4 PERSONNEL DOSIMETRY 159
6.4.1 Energy Dependence 160
6.4.2 Angular Dependence 160
6.4.3 Dosimetry Practices 160
6.4.4 Extremity Dosimetry 161
6.4.5 Dose to Lens of Eye 163
6.5 External Dose Control 163
6.5.1 Time 163
6.5.2 Distance 164
6.5.3 Shielding 164
6.5.4 Geometry 166
6.6 Recordkeeping 166
7 NUCLEAR CRITICALITY SAFETY 167
7.1 REGULATIONS AND STANDARDS 167
7.2 Criticality Control Factors 168
7.2.1 Controllable Factors 168
7.2.2 Administrative Practices 170

7.3 CRITICALITY ACCIDENT EXPERIENCE ... 173
7.3.1 Types of Criticality Accidents ... 173
7.3.2 Summary of Past Criticality Accidents ... 174
7.4 CRITICALITY ALARMS AND NUCLEAR ACCIDENT DOSIMETRY ... 174
7.4.1 Criticality Accident Alarm System (CAAS) ... 174
7.4.2 Nuclear Accident Dosimetry ... 175
7.5 RESPONSIBILITIES OF RADIOLOGICAL CONTROL STAFF ... 179
7.5.1 Routine Operations ... 179
7.5.2 Emergency Response Actions ... 179
7.5.3 Special Considerations during Decommissioning Activities ... 180
8 WASTE MANAGEMENT ... 181
8.1 POTENTIALLY CONTAMINATED WASTES ... 181
8.1.1 Solid Waste ... 182
8.1.2 Liquid Waste ... 182
8.2 DESIGN OF WASTE PROCESSING SYSTEMS ... 182
8.2.1 Objectives ... 182
8.2.2 Effluents ... 183
8.3 Treatment ... 183
8.3.1 Airborne Wastes ... 183
8.3.2 Liquid Waste ... 184
8.3.3 Solid Waste ... 184
8.4 Monitoring ... 185
8.4.1 Air and Gaseous Effluents ... 185
8.4.2 Liquid Effluents ... 185
8.4.3 Water Collection System ... 186
8.5 Waste Minimization ... 186
9 EMERGENCY MANAGEMENT ... 189
9.1 Emergency Management in DOE ... 189

9.1.1 Basis for DOE Emergency Management Policy ... 189

9.1.2 Requirements Pertaining to All DOE Operations ... 190

9.2 Specific Guidance on Emergency Management for Uranium Facilities ... 191

9.2.1 Technical Planning Basis ... 191

9.2.2 Program Elements ... 195

10 DECOMISSIONING ... 200

10.1 REGULATORY FRAMEWORK ... 200

10.1.1 Resource Conservation and Recovery Act of 1976 (RCRA) ... 200

10.1.2 Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) ... 200

10.1.3 National Environmental Policy Act ... 201

10.1.4 DOE Directives and Standards ... 201

10.2 Residual Radioactivity Levels ... 202

10.2.1 Public Dose Limits ... 202

10.2.2 ALARA process requirements ... 202

10.2.3 Dose Constraints ... 203

10.2.4 Authorized Limits ... 203

10.2.5 Survey and Monitoring ... 204

10.3 Design Features for New Facilities ... 204

10.3.1 Building Materials ... 205

10.4 Ventilation Systems ... 206

10.4.1 Piping Systems ... 206

10.4.2 Soil-Contamination Considerations ... 206

10.4.3 Other Features ... 206

10.5 DECOMMISSIONING PROGRAM REQUIREMENTS ... 207

10.5.1 Pre-Operational and Operational Activities ... 208

10.5.2 Post-Operational Activities ... 208

10.5.3 Decommissioning Activities ... 208

10.5.4 Post-Decommissioning Activities ... 214

10.5.5 Quality Assurance ... 214

10.6 Decontamination and Decommissioning Experience ... 214

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.

6 EXTERNAL DOSIMETRY

The external dosimetry program is an integral part of the external dose control program. DOE G 441.1-1C provides detailed guidance for implementing an external dosimetry program that meets the requirements of 10 CFR Part 835. The reference section of that Guide lists specific documents applicable to external dosimetry. Because the requirements and recommendations are explicitly given in these documents, they will not be discussed in any great detail in this chapter. Rather, the emphasis will be on items that are unique to uranium facilities and the radiological aspects for safe handling of uranium.

Measuring the external radiation exposure and the resultant dose is complicated by the many radiations involved in uranium handling. Chapter 2 of this Technical Standard discusses the radioactive decay schemes for and radiations emitted by the uranium isotopes and their radioactive daughter products. Uranium has a wide distribution of beta and gamma energies, with a 2.29-MeV beta as the most significant of these. The dose rate from photons is relatively low. Uranium also emits alpha particles that may generate ~2 MeV neutrons as a result of interactions with the nuclei of fluorine or other low-Z atoms. The magnitude of the neutron fluence depends on the enrichment of the uranium and on the interacting chemical.

The elements of the external dose control program are: detection and characterization of the beta, gamma, and neutron radiation fields; measurement and quantification of these fields; measurement of personnel dose; and determination and establishment of dose control practices.

6.1 DOSE LIMITS

10 CFR Part 835 specifies the applicable limits used for control of external radiations. Table 6-1 lists the appropriate depths in tissue for measurement of doses to the whole body, lens of the eye, "unlimited areas of skin," and extremities.

Table 6-1. Effective Depth of Tissue for Various Organs

Tissue Type	Depth of tissue, (mg/cm^2)
Deep (penetrating)	1000
Lens of eye	300
Shallow (skin, extremities)	7

6.1.1 Limiting Quantities

In 1994, the ICRP introduced a major revision in recommended radiation protection practice with the introduction of ICRP Publication 60 (1991a). The new methodology expands on a "risk-based" system of dose limitation. The ICRP introduced the terms *stochastic* and *deterministic* for radiation effects and set limits for both types of effect. Stochastic effect is defined as one for which the probability of the effect occurring (as opposed to the degree or severity of effect) is a function of radiation dose. Deterministic effect is defined as one for which the severity of the effect is a function of the dose; a threshold may exist. Limits were established such that the risk of stochastic effects occurring was equivalent to about the same risks faced by workers in "safe" industries who were not occupationally exposed to radiation in the workplace. Limits were also established for deterministic effects that prevented these effects from occurring even if the exposure occurred at the annual limit over the lifetime of the worker.

Equivalent dose (H_T), which is the absorbed dose averaged over tissue or organ due to radiation, is given by the equation:

$$H_T = \sum_R w_R \; D_{T,R}$$

Where:
W_R is the radiation weighting factor (See Table 6-2)
$D_{T,R,}$ is the average absorbed dose (rad) in a tissue or organ

The equivalent dose is not always the appropriate quantity for use in relations to deterministic effects because the values of radiation weighting factors (w_T) have been chosen to reflect the relative biological effectiveness of the different types of energies of radiation in producing stochastic effects.

Table 6-2. Radiation Weighting Factors, w_R

Type and Energy Range	Radiation Weighting Factor (w_R)
Photons, electrons and muons, all energies	1
Neutrons, energy<10keV	5
Neutrons, energy 10 keV to 100 keV	10
Neutrons, energy > 100 keV to 2 MeV	20
Neutrons, energy > 2 MeV to 20 MeV	10
Neutrons, energy > 20 MeV	5
Protons, other than recoil protons, energy > 2 MeV	5
Alpha particles, fission fragments, heavy nuclei	20

The Effective dose (E) is the summation of the products of the equivalent dose received by specified tissues or organs of the body and the appropriate tissue weighting factor (w_T). Effective dose is expressed in terms of rem (or Sv).

$$E = \sum w_T \, H_T$$

Where:
W_T is the tissue weighting factor (See Table 6-3)
H_T is the equivalent dose in rem (or Sv)

Table 6-3. Tissue Weighting Factors

Organs or tissues, T	Tissue weighting factor, w_T
Gonads	0.20
Red bone marrow	0.12
Colon	0.12
Lungs	0.12
Stomach	0.12
Bladder	0.05
Breast	0.05
Liver	0.05
Esophagus	0.05
Thyroid	0.05
Skin	0.01
Bone surfaces	0.01
Remainder[a]	0.05
Whole body[b]	1.00

a. "Remainder" means the following additional tissues and organs and their masses, in grams, following parenthetically: adrenals (14), brain (1400), extrathoracic airways (15), small intestine (640), kidneys (310), muscle (28,000), pancreas (100), spleen (180), thymus (20), and uterus (80). The equivalent dose to the remainder tissues ($H_{remainder}$), is normally calculated as the mass-weighted mean dose to the preceding ten organs and tissues. In those cases in which the most highly irradiated remainder tissue or organ receives the highest equivalent dose of all the organs, a weighting factor of 0.025 (half of remainder) is applied to that tissue or organ and 0.025 (half of remainder) to the mass-weighted equivalent dose in the rest of the remainder tissues and organs to give the remainder equivalent dose.

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

Table 6-4 lists the annual radiation dose limits for DOE activities. However, DOE contractors usually establish lower annual administrative control levels, typically 500 mrem/year or less.

In practice, it is difficult to measure the effective dose equivalents specified in Table 6-4 because it is necessary to know not only the type of radiation but also its energy and direction. If the flux, energy, and direction of incidence are known, it is possible to calculate effective dose equivalent using fluence to effective dose equivalent conversion coefficients presented in ICRP Publication 51 (1987), which presents the effective dose equivalent as a function of energy for various irradiation geometries. Conversion coefficients for mono-directional beams of neutrons can be found in an article by Stewart et al (1994). Conversion coefficients for photons in various irradiation geometries, including planar sources, can be found in a report by Zankl et al. (1994). This approach will provide more accurate values of effective dose equivalent, as opposed to numerically setting the value of effective dose equivalent equal to dose equivalent geometries. Conversion coefficients for mono-directional beams of neutrons can be found in an article by Stewart (Stewart et al. 1994). Conversion coefficients for photons in various irradiation geometries, including planar sources, can be found in a report by Zankl et al. (1994). This approach will provide more accurate values of effective dose equivalent, as opposed to numerically setting the value of effective dose equivalent equal to dose equivalent.

Table 6-4. Radiation Dose Limits for DOE and DOE Contractors

Type of Radiation Exposure	Limit
Occupational Exposures of Adults	
Stochastic Effects	5-rem total per year (sum of effective dose from external exposures and CED received during year)
Deterministic Effects	
Lens of eye	15-rem equivalent dose per year
Extremity	50-rem equivalent dose per year
Skin	50-rem equivalent dose per year
Individual organ or tissue	50-rem dose equivalent per year (sum of equivalent dose from external exposures and CED received during the year)
Occupational Exposures of Minors	
Stochastic Effects	0.1-rem per year (sum of effective dose from external exposures and CED received during year)
Deterministic Effects (Lens of eye, extremity, skin, individual organ or tissue)	10% of occupational dose limits for adults
Embryo/fetus of a Declared Pregnant Worker	0.5-rem equivalent dose per gestation period
Planned Special Exposure	Same as routine occupational dose limits in a year (but accounted for separately) 5 times the routine occupational dose limits over an individual's lifetime

6.1.2 Operational Quantities

Because of the difficulties in determining effective dose equivalent from direct measurements, the concept of *operational quantities* has been introduced to be more closely related to measurable quantities. Operational quantities include *ambient dose equivalent* used for area monitoring and *personal dose equivalent* used for personnel dosimetry. Operational quantities are designed to be a conservative estimator of effective dose equivalent, i.e., the values of the operational quantities will be equal to or higher than the effective dose equivalent specified for the limiting quantities.

The ambient dose equivalent, $H^*(d)$, is the dose equivalent at a depth, d, in a 30-cm-diameter sphere of tissue, where: a) the radiation field has the same fluence and energy distribution as the point of reference for the measurement; and b) the fluence is unidirectional (i.e., the sphere can be viewed as being in an aligned radiation field). Most survey instruments are designed to measure ambient dose equivalent, and international standards are based on the ambient dose equivalent concept. The depth of interest is typically 1 cm of soft tissue, as specified in 10 CFR Part 835.

The personal dose equivalent, $H_p(d)$, is the dose equivalent in soft tissue at the appropriate depth, d, below a specified point on the body. Obviously, personnel dosimeters should be calibrated in terms of personal dose equivalent.

In reality, most instruments and personnel dosimeters used at DOE facilities are calibrated in terms of equivalent dose. For example, consider the case in which personnel neutron dosimeters are calibrated on acrylic plastic phantoms at a specified distance from a calibrated neutron source. For DOELAP testing, the dose equivalent at this point has been calculated in accordance with ICRP Publication 74 (1996).

In most instances, the present methods based on dose equivalent overestimate effective dose equivalent. In cases where personnel are approaching dose limits, it may be prudent to more accurately evaluate effective dose equivalent using special calibrations. Depending on the irradiation geometry and energy, effective dose equivalent may be as much as a factor of two less than dose equivalent.

6.2 RADIATIONS IN URANIUM FACILITIES

As outlined in Section 2.0 of this Technical Standard, the uranium isotopes are primarily alpha-emitters and their progeny emit a wide variety of radiations, including alpha and beta particles, as well as more penetrating x rays and gamma rays. Alpha-neutron interactions (and the small cross-section for spontaneous fission) add the potential for neutron exposure to the radiation mix. This section outlines methods to calculate the dose equivalents from radiations emitted by uranium and its progeny. Examples of measured dose rates are also included.

The design of an external dose control program, including instrument and dosimeter selection, is dependent upon the type and intensity of the radiation fields to which the workers will be exposed. Many factors can affect the radiation field:

a. Enrichment (mix of uranium isotopes)

b. Emissions from parent radionuclide(s)

c. Emissions from daughter radionuclide(s)

d. Emissions from impurity radionuclide(s)

e. Type of radiation emitted (e.g., beta, gamma)

f. Energies of emitted radiation

g. Specific activity of the source material

h. Self-shielding of source material

i. Shielding provided by process equipment

j. Shielding provided by personal protective equipment

k. Distance and geometry factors

The ratio of uranium isotopes in a specific process (a function of enrichment) will determine the source term by which the radiation fields can be predicted. This mix of uranium isotopes and daughter radionuclides may be estimated by using an equation developed to predict specific activity as a function of enrichment.

Radiation fields from uranium are frequently dominated by contributions from daughter product or impurity radionuclides. For example, nearly all of the beta radiation field from depleted uranium comes from the daughter radionuclide ^{234}mPa, and to a lesser extent from ^{234}Th. During melting and casting operations, these daughter elements may concentrate on the surface of the castings and equipment, producing beta radiation fields up to 20 rad per hour.

Figure 6-1. Beta Radiation Readings at Surface of Uranium Metal vs. % Enrichment by Weight

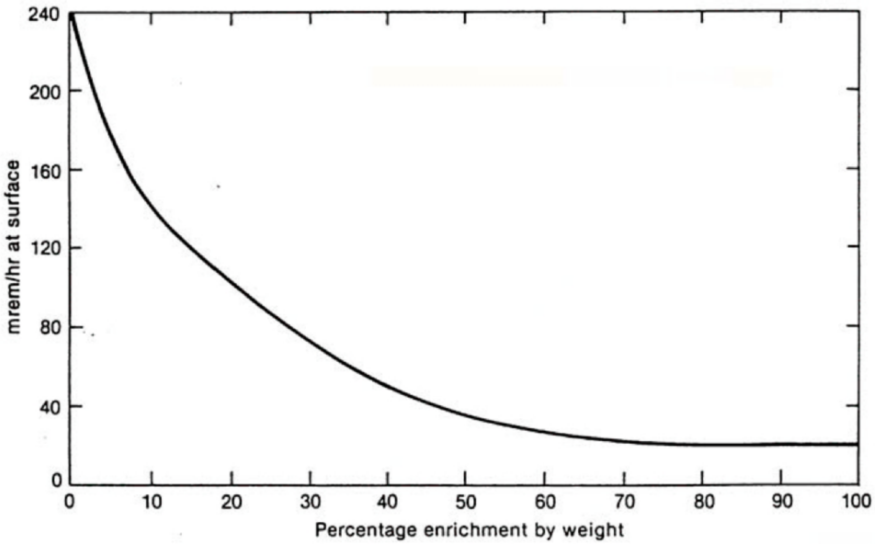


Figure 6-2. Absorbed Dose Rate as a Function of Depth in Mylar

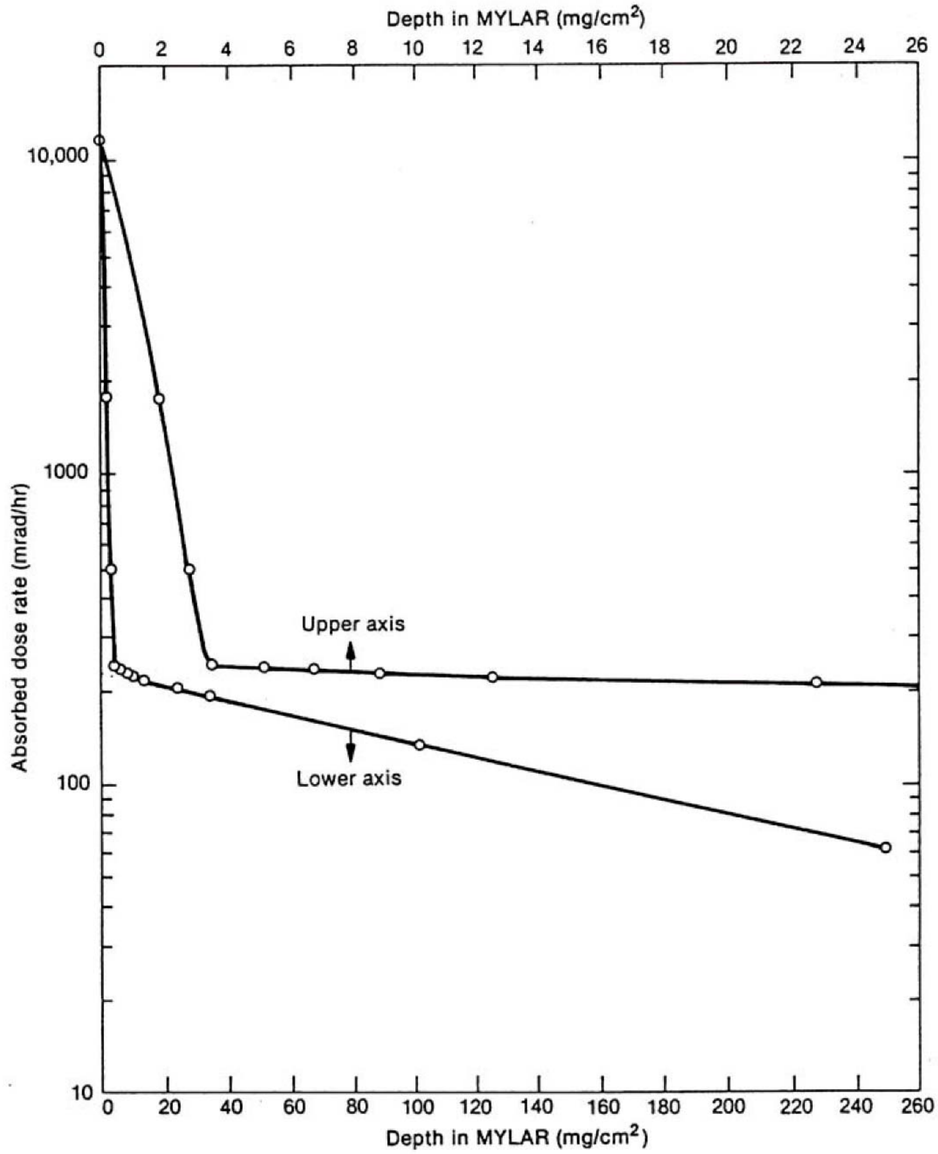
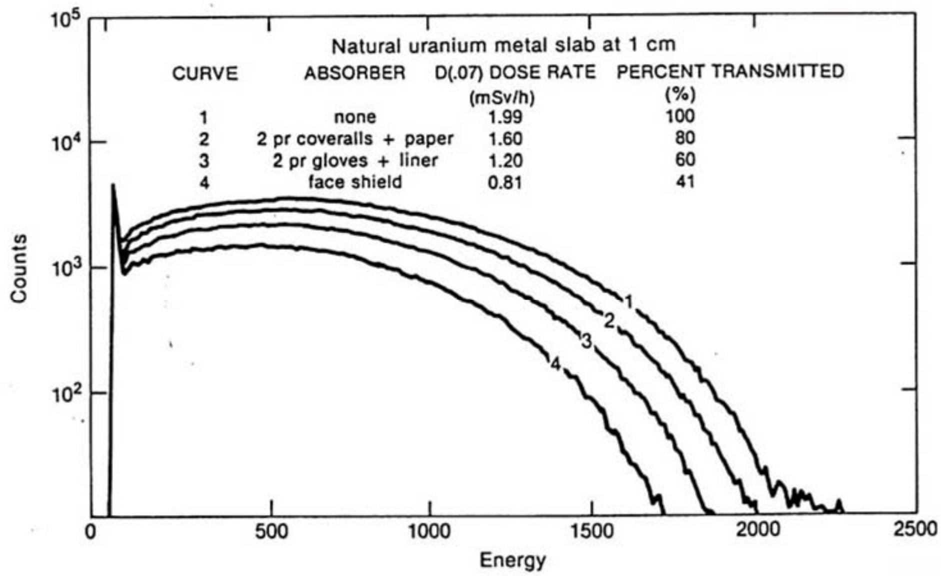


Figure 6-3. Changes in Beta Energy Spectra and Shallow Dose Rate from a Natural Uranium Metal Slab Source Caused by Protective Apparel (Note the bremsstrahlung peak in the low energy ranges).



6.2.1 Alpha and Beta Doses

Uranium is an alpha-emitter and is of concern if inhaled or ingested into the body. The skin is an effective barrier to alpha particles, and contamination is only a problem if there is a wound or break in the skin.

Beta radiation fields are usually the dominant external radiation hazard in facilities requiring contact work with unshielded forms of uranium. Figure 6-1 gives the estimated beta dose rates from a semi-infinite slab of uranium metal of various enrichments. For uranium enrichments up to 30%, the beta radiation field is dominated by contributions from ^{238}U decay products. Thus, for uranium of these enrichments, one is dealing essentially with 2.29-MeV (E_{max}) beta particles from ^{234m}Pa, the most energetic contributor to the beta exposure.

Beta doses to the skin, extremities, and (sometimes) the lens of the eye can be limiting in facilities that process unshielded depleted, natural, or low-enrichment uranium. Absorbed dose rates as a function of depth were measured by Plato (1979) with an extrapolation chamber in a tissue equivalent medium (Mylar) (See Figure 6-2). Skin doses at less than 4 mg/cm^2 resulting from alpha particles are of no concern from an external radiation exposure standpoint. Potentially significant skin exposure from uranium occurs primarily from the ^{234m}Pa betas at tissue depths of 4 mg/cm^2 and greater.

Processes that separate and sometimes concentrate beta-emitting uranium daughters are not uncommon in DOE uranium facilities. Surface beta dose rates on the order of 1 to 20 rad per hour have been observed in such circumstances. Exposure control is complicated by the fact that considerable contact work takes place in facilities that process uranium metal. Beta particles are shielded by rubber gloves or other protective devices or are usually absorbed within the dead layer of skin. The actual beta dose to live tissue would depend on the energy of the beta particles and the thickness and types of intervening shielding.

The data in Figure 6-3 were obtained with a tissue equivalent plastic scintillation detector and demonstrate the spectral changes and the resultant exposure rates under typical protective clothing. It can be seen from Figures 6-2 and 6-3 that significant fractions of the uranium beta radiation will penetrate typical protective clothing worn in facilities which process uranium.

6.2.2 Gamma Doses

Gamma radiation from uranium is normally not the controlling challenge to radiation protection. For example, the contact beta radiation field from depleted uranium is approximately 240 mrem/h, while the contact gamma radiation field is less than 10 mrem/h. Although gamma radiation fields from uranium are not usually the dominant concern, significant gamma fields can exist in areas where large quantities of uranium are stored. Bremsstrahlung from the 2.29 MeV ^{234m}Pa beta can contribute up to 40% of the photon dose from uranium metal. Neutron fields from enriched uranium fluoride compounds can also add to this area of concern. Care should be taken that dose-equivalents from such fields are kept to levels that are ALARA.

Although beta radiation fields from unshielded uranium tend to present the most intense radiation problem, storage of large quantities of uranium can create widespread, low-level (<5 mrem/h) gamma radiation fields. Such fields can create ALARA problems--particularly when significant numbers of people must work in adjacent areas.

6.2.3 Neutron Dose Equivalents

In uranium processes that create fluoride compounds (e.g., UF_4, UF_6), the a-n reaction with this light nuclide can result in neutron radiation fields, the intensity of which are a function of the compound, mixing, storage configuration, and enrichment. As indicated in Section 2.0, low enriched UF_6 (< 5%) in large storage containers can result in neutron radiation in the 0.2 mrem/h range, while highly enriched (> 97%) UF_6 can create fields in the 4 mrem/h range. At high enrichments, the neutron fields can be up to a factor of 2 higher than the gamma fields and be the limiting source of whole body exposure. Neutron radiation from uranium metals and low enriched compounds is considerably lower than the gamma component and, consequently, is not limiting.

Neutron dose equivalent rates can be calculated accurately with computer codes, such as the Monte Carlo Neutron Program (MCNP) (Breismeier, 1986). The MCNP code has the advantage that it can calculate both neutron and photon doses through shielding and in complex arrays. The Monte Carlo codes can also calculate the effects of neutron multiplication in systems containing large amounts of uranium. However, neutron dose equivalent rates can also be calculated from simple empirical formulas. Unlike gamma doses, there is very little self-shielding for neutrons in sub-kilogram masses of uranium.

Table 6-5 lists spontaneous fission yields for uranium isotopes that may be found in facilities within the DOE complex. These data are taken from NUREG/CR-5550 (NRC 1991) and are believed to be more current then the previously published PNL values (Backenbush et al., 1988). As a rule of thumb, nuclides with even numbers of protons and neutrons have the highest spontaneous fission neutron emission rates. The spontaneous fission rate for odd-even nuclides is about 1000 times less, and the rate for odd-odd nuclides is about 100,000 less. Spontaneous fission neutrons are emitted with a Maxwellian energy distribution given by the equation:

$$N(E) \doteq \sqrt{E} * \exp(E/1.43\,MeV)\,;$$

Where N(E) is the number of neutrons as a function of the energy E in MeV.

Table 6-5. Spontaneous Fission Neutron Yields

Isotope	Total Half Life (years)	Spontaneous Fission Half Life (years)	Spontaneous Fission Yield (n/sec gram)
^{232}U	71.7	8×10^{13}	1.3
^{233}U	1.59×10^{5}	1.2×10^{17}	8.6×10^{-4}
^{234}U	2.45×10^{5}	2.1×10^{16}	5.02×10^{-3}
^{235}U	7.04×10^{8}	3.5×10^{17}	2.99×10^{-4}
^{236}U	2.34×10^{7}	1.95×10^{16}	5.49×10^{-3}
^{238}U	4.47×10^{9}	8.20×10^{15}	1.36×10^{-2}

Energetic alpha particles can overcome coulomb barriers in low-atomic-number elements and create an unstable nucleus that emits neutrons. Because of the high alpha activity of uranium, this can be a significant source of neutrons. There are two nuclear reactions that are of importance:

$$\propto\ + {}^{18}O \rightarrow {}^{21}Ne + n$$

$$\propto\ + {}^{19}F \rightarrow {}^{22}Na + n$$

Table 6-6 lists the alpha-neutron yields for oxides and fluorides for the uranium isotopes. Note that the neutron yields are normalized per gram of nuclide, not per gram of compound. These data are taken from NUREG/CR-5550 (NRC 1991).

The total neutron yield per gram of uranium can be found by summing the contributions from

a. Spontaneous fission (from Table 6-5)

b. Alpha-neutron reactions in oxides or fluorides (from Table 6-6)

c. Neutrons from low-atomic-number impurities (from Table 6-7)

Multiplying the specific neutron yield (neutrons per second per gram (nps/g) of uranium by the mass of uranium (grams) gives S, the neutron emission rate (neutrons per second).

In order to sum the three contributions, it is necessary to convert the values in Table 6-7.

The specific activity of U-234 is converted from Ci/g to disintegrations per second per gram (dps/g):

$6.2\text{x}10^{-3}$ Ci/g x $3.7\text{x}10^{10}$ dps/Ci = $2.3\text{x}10^{8}$ dps/g.

Now we set (for Be): 10^{6} disintegrations = 44 neutrons

$2.3\text{x}10^{8}$ dps/g x 44 neutrons/10^{6}dis ≈ 10^{4} nps/g

Table 6-6. Neutron Yields from Alpha-Neutron Reactions for Oxides and Fluorides

Isotope	Alpha Decay Half Life	Alpha Yield (alpha/s g)	Average Alpha Energy (MeV)	alpha, n Yield in Oxides (n/s g)	alpha, n Yield in Fluorides (n/s g)
^{232}Th	1.41×10^{10} y	4.1×10^{3}	4.00	2.2×10^{-5}	
^{232}U	71.7 y	8.0×10^{11}	5.30	1.49×10^{4}	2.6×10^{6}
^{233}U	1.59×10^{5} y	3.5×10^{8}	4.82	4.8	7.0×10^{2}
^{234}U	2.45×10^{5} y	2.3×10^{8}	4.76	3.0	5.8×10^{2}
^{235}U	7.04×10^{8} y	7.9×10^{4}	4.40	7.1×10^{-4}	0.08
^{236}U	2.34×10^{7} y	2.3×10^{6}	4.48	2.4×10^{-2}	2.9
^{238}U	4.47×10^{9} y	1.2×10^{4}	4.19	8.3×10^{-5}	0.028

Table 6-7. Neutron Yields for Trace Impurities of Uranium

Element	Neutron Yield per 10^6 Alphas at 4.7 MeV (^{234}U)
Li	0.16 ± 0.04
Be	44. ± 4
B	12.4 ± 0.6
C	0.051 ± 0.002
O	0.040 ± 0.001
F	3.1 ± 0.3
Na	0.5 ± 0.5
Mg	0.42 ± 0.03
Al	0.13 ± 0.01
Si	0.028 ± 0.002
Cl	0.01 ± 0.01

6.3 RADIATION DETECTION AND EVALUATION

This section describes the response of portable instruments, personnel dosimeters, and nuclear accident dosimeters to the radiations emitted by uranium, which are primarily alpha and beta particles and photons. Neutron emissions may range from negligible to significant. Data are also included on special spectrometry instruments used to calibrate dosimeters in the field.

6.3.1 Portable Survey Instruments--Beta Radiation Response

The primary exposures of concern when handling bare uranium materials come from the beta radiation. The accuracy and precision of survey instruments used for measurement of beta radiation fields depend upon some or all of the following factors:

a. beta energy response

b. angular response of instrument

c. source-detector geometry factors

d. detector construction (e.g., window thickness)

6.3.1.1 Energy Dependence

Most commercially-available radiation survey instruments under-respond to beta radiation fields from uranium. Figures 6-4 and 6-5 show the beta and gamma spectra measured with a tissue equivalent plastic scintillation. Table 6-8 presents typical survey instrument response to uranium fields specifically. At best, typical "beta correction factors" (true dose rate/indicated dose rate) are on the order of 1.5 to 2. This under-response is due primarily to a) the angular response of the detector and b) attenuation of the dose-rate by the detector window and the sensitive volume of the detector.

Figure 6-4. Meter Readings for a Depleted Uranium Ingot

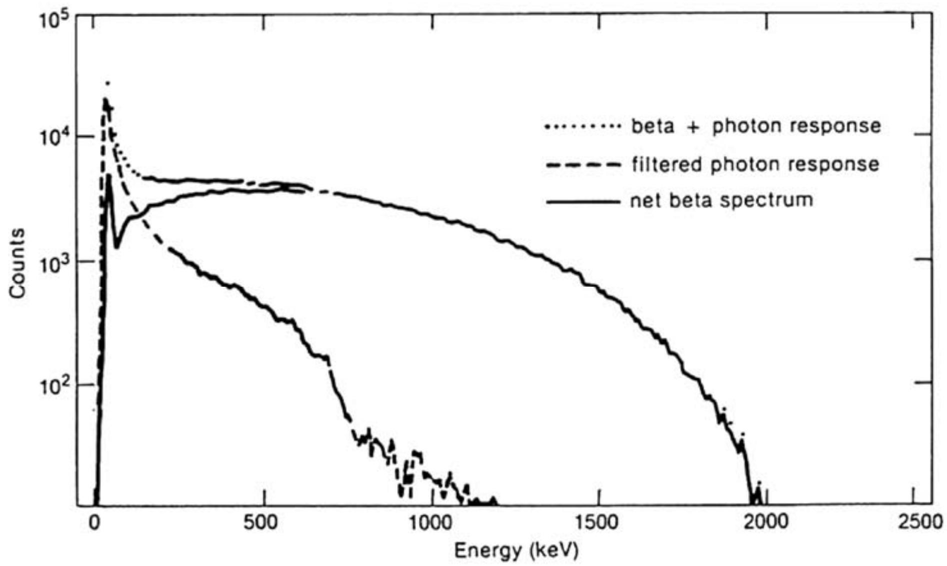


Figure 6-5. Meter Readings for an Open Drum of UF_4 (green salt)

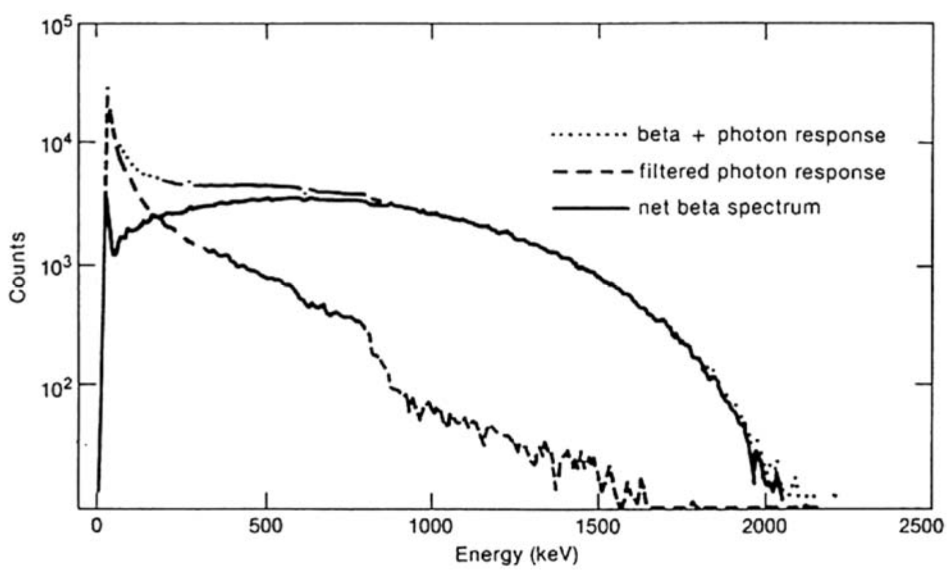


Table 6-8. Instrument Response to Uranium Beta Fields

Instrument	Window (mg/cm^2)	Beta Correction Factor[a]	Exposure Geometry
Victoreen 471	1.1	1.4	30 cm from U foils
Eberline RO-2	7	2.0	30 cm from U foils
Eberline RO-2A	7	4.0	Contact with DU slab
Aluminum-walled GM Detector	30	1.7	30 cm from U foils
Victoreen Radector III	34	14	Contact with DU slab
HPI-1075	7	1.8	Contact with DU slab
Teletector	30 (low range)	50	Contact with DU slab
Eberline PIC-6A	30	40	Contact with DU slab
British BNL-3	7	1.3	1.5 cm from 100 cm^2 DU
(a) True reading/measured value.			

Currently, skin dose measurements are related to the dose at a depth of 7 mg/cm^2 in tissue. Window thicknesses of commonly available survey instruments typically range from on the order of 7 mg/cm^2 to several hundred mg/cm^2.

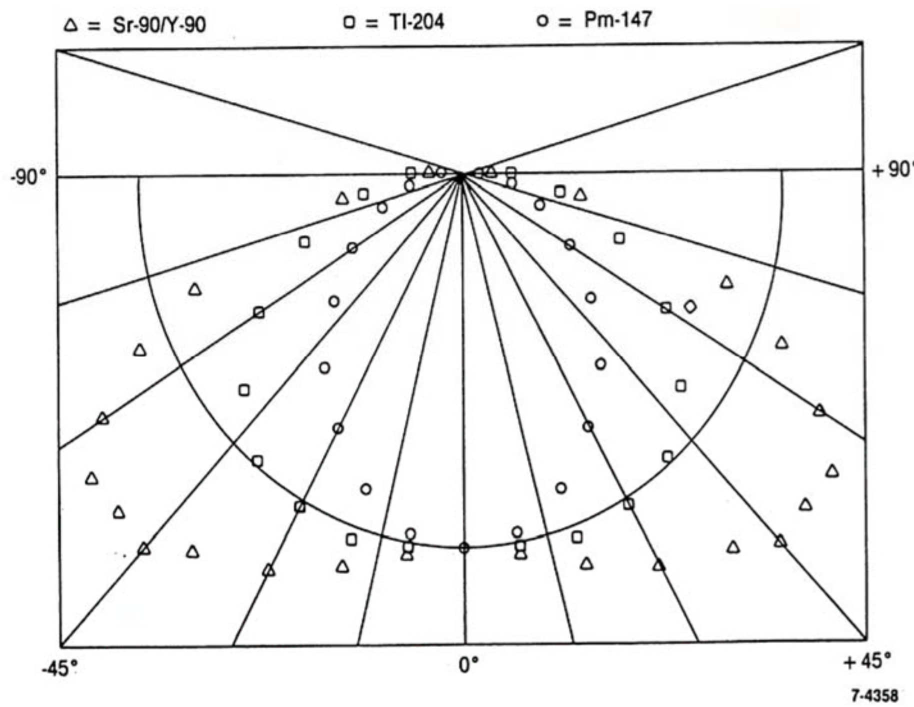
Even if the window provides only minimal attenuation, the attenuation of the beta field through the sensitive volume of large detectors remains a problem. The detector indicates the average dose-rate throughout the sensitive volume. The "true" dose-rate is that which occurs at the plane of the detector incident to the radiation source. The instrument will under-respond by the ratio of this average dose-rate to the incident dose-rate. This sensitive volume under-response is a function of the beta energy distribution and of the size and shape of the sensitive volume.

6.3.1.2 Angular Response

The construction of most survey instruments (e.g., "cutie pie") leads to a severe angular dependence when measuring beta radiation fields. This angular dependence results from the attenuation of the beta field by the walls of the detector as the window is moved away from the source.

Figure 6-6 demonstrates the response of a tissue equivalent response (a 5 mg/cm^2 detector under a 5 mg/cm^2 window and mounted in a TE phantom) to off-axis (non-incident) ^{90}Sr/^{90}Y betas (energies similar to those from uranium), ^{204}Tl, and ^{147}Pm. Skin tissue dose response is greater to off-axis betas; survey instruments, which effectively shield these high angle particles, will under-respond compared to skin tissue.

Figure 6-6. Measured Angular Response to the INEL TE Survey Meter to Parallel Beams of Beta Particles from Three Standard Beta Sources.



6.3.1.3 Source-Detector Geometry

Measurements taken close to small beta sources may be inaccurate due to non-uniform irradiation of the sensitive volume of the detector. Uranium in most DOE facilities tends to present wide-area sources of beta radiation. However, adjustments would need to be made if significant non-uniform irradiation was encountered.

6.3.1.4 Detector Construction and Use

Characteristics of instrument construction may significantly affect their response and use. For example, many survey instruments have "beta windows" that are intended to discriminate between beta and gamma radiation. Obviously, measurements of beta dose-rate must be made with the beta window open. It should be noted, however, that a number of instruments have beta windows that are only a few hundred mg/cm^2 thick. Such windows can transmit a significant fraction of the dose-rate from high-energy beta-emitters (e.g., ^{234m}Pa). Thus, up to 10% or 20% of the "gamma only" reading may be due to the higher-energy betas penetrating the so-called beta window.

Occasionally, survey instruments are placed in plastic bags or covered to protect them from becoming contaminated. Bagging the instrument places additional absorber between the radiation field and sensitive volume of the detector. Calibration of the instrument (or application of a correction factor) should take this additional shielding into account.

6.3.2 Portable Survey Instruments--Gamma Radiation Response

Although the external dose resulting from gamma and x-ray radiation from bare uranium is a small fraction of the total, it represents the "penetrating" or whole body dose source and is the only source of radiation from contained facilities (i.e., those having glove boxes). Survey instruments are typically calibrated with ^{137}Cs (0.662-MeV) photons. Typical portable survey instruments demonstrate a fairly flat energy response above 250 keV, while the response below 250 keV can be variable to a greater or lesser degree depending upon the instrument design. Figures 6-7 and 6-8 show average response of a group of commercial survey instruments. Figure 6-9 shows a typical gamma spectrum from a uranium oxide source while Table 6-9 illustrates the wide variation that can occur in the photon spectra at various locations in a single plant. This demonstrates the desirability of using ion chambers or compensated beta instruments for dose-rate measurements. It also indicates the need to have knowledge of the energy response of the instrument used and the value, or at least qualitative knowledge, of the photon spectra at the various work stations.

Figure 6-7. Average Ion Chamber Survey Meter Response by Group to X or Gamma Photon Radiation

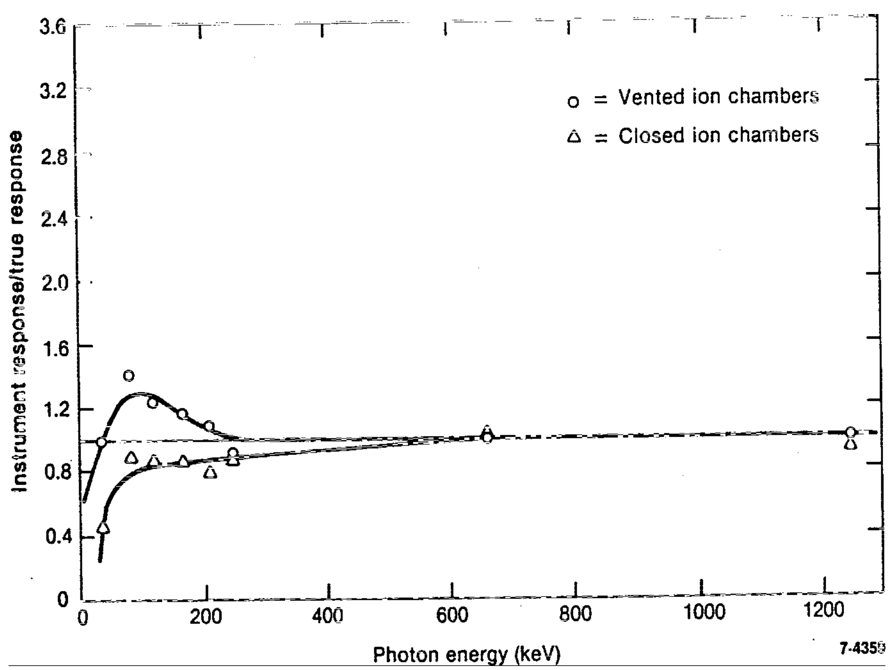


Figure 6-8. Average GM Survey Meter Photon Energy Response by Group

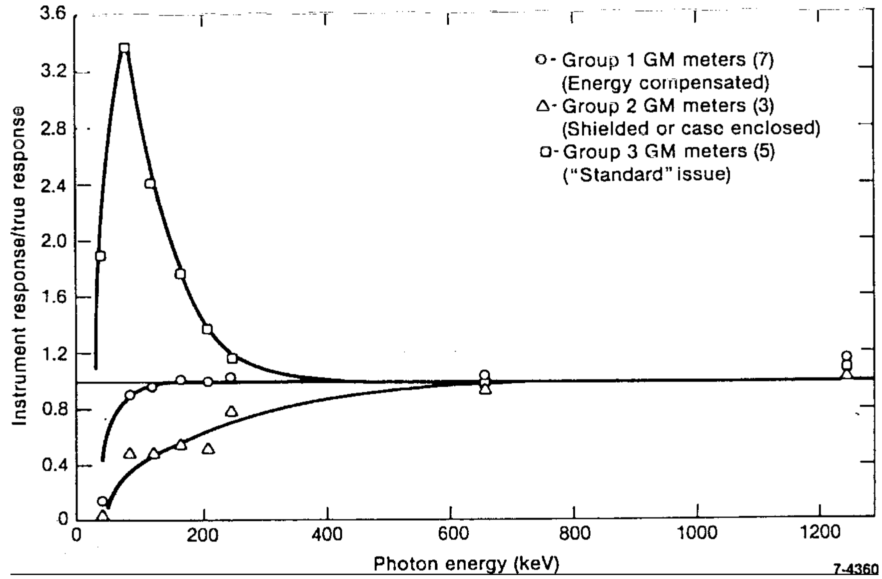


Figure 6-9. High Resolution Gamma Spectrum of Slightly Enriched Uranium Oxide (1% ^{235}U) record with Ge(Li) Detector

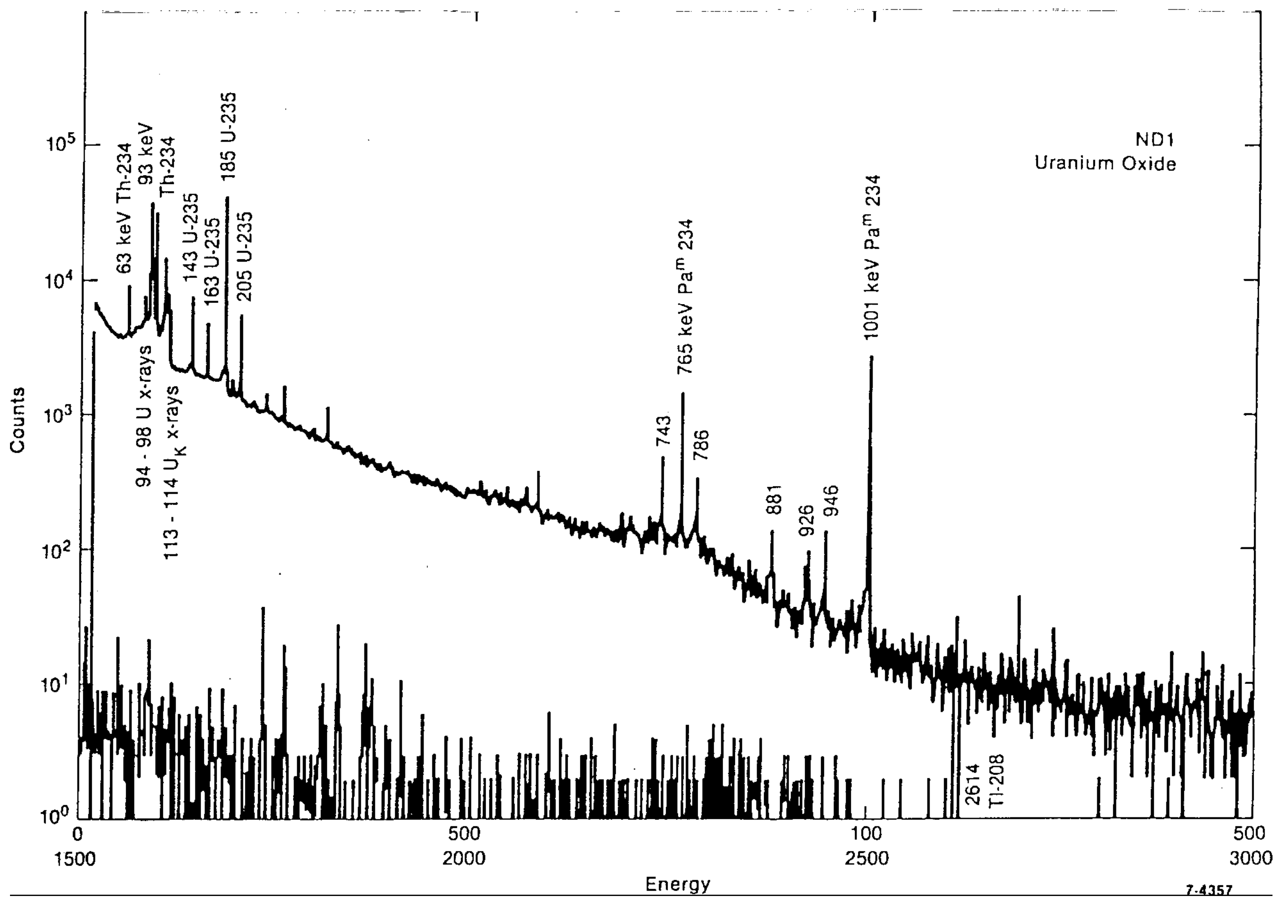


Table 6-9. Gamma Flux and Ratios at Various Locations and Sources at Fernald Plant

	Integrated Gamma Flux (photons/cm^2/sec)		
Source Description or Location	**30 to 225 keV**	**675 to 1050 keV**	**Ratio**
Crucible load station - 55-gal. drum	990	348	2.8
Beside UO_3 barrel	538	159	3.4
Open UO_3 barrel	919	232	4.0
Tube-cutting work station, metal	253	58	4.4
Outside Plant 9 south entrance, near exhaust fan	776	165	4.7
Box of black top crop at 25 cm	848	154	5.5
Lathe work station	424	76	5.6
Background outside Building 3045	35	5	7.0
Near "thorium" hopper	424	58	7.3
Plant 9 west wing, SW hot area	708	72	9.8
Crucible burnout station	776	69	11.2
Plant 9 HP change room	5	<0.4	12.5
Background 75 ft from Bldg. 3045	25	2	12.5
Graphite crucible (G-8010) 30 cm	183	11	16.6
Graphite crucible (3898) 30 cm	310	18	17.2

6.3.3 Portable Survey Instruments--Neutron Response

The need for neutron surveys at uranium facilities depends on the quantity of uranium present, its form, and the potential for (α,n) reactions, such as occurs with uranium fluoride. In facilities where such monitoring is required, selection of instruments with appropriate energy characteristics is important because of the energy and angular dependence with most instruments. Fortunately, uranium compounds emit neutrons in the MeV range, where problems with energy and angular dependence are minimal. Calibration with sources that emit neutron energies similar to those in the facility will assist in accurately measuring the radiation fields and selecting appropriate factors in calculating personnel doses.

6.4 PERSONNEL DOSIMETRY

It is important to verify and document that personnel dosimetry systems provide accurate measurements and records of the occupational radiation doses received by workers in uranium facilities. To provide a level of confidence in dosimetry services in DOE facilities, the DOELAP accreditation program has been established. 10 CFR Part 835 requires participation in the DOELAP program (or specific exceptions or other approvals) by all DOE facilities that are subject to the individual external dose monitoring requirements. Previously, the National Institute of Standards and Technology (NIST) established the National Voluntary Laboratory Accreditation Program (NVLAP) for testing and accreditation of dosimeter processors serving the commercial nuclear power industry and medical facilities. The DOELAP standard includes some tests that differ from those in ANSI N13.11-2009 (2009), on which the NVLAP program is based. Both DOELAP and NVLAP accreditation programs use performance tests that evaluate the accuracy and precision of personnel dosimetry measurements. The accuracy is determined by comparing the measured dose equivalent to the "conventionally true dose equivalent" derived from calibration standards directly traceable to NIST in carefully controlled conditions.

DOE G 441.1-1C (2008a) provides detailed guidance for developing and implementing an external dosimetry program that will comply with the requirements of 10 CFR Part 835. This section will focus on dosimetry problem areas specific to uranium facilities and possible solutions.

Personnel dosimeters produce the data that become the formal or "legal" record of personnel exposure. However, these detectors experience many of the same energy dependence and angular response problems encountered by survey instruments. The most difficult problem is relating badge results to the shallow or skin dose.

Thermoluminescent dosimeters (TLDs), currently the dosimeter of choice in most DOE uranium facilities, provide the most accurate and precise means of measuring doses received by workers. Film badges and nuclear track detectors are other types of dosimeters. Although the following discussion focuses on the more widely used TLD detector systems, the basic principles apply to film badges, with the added uncertainties associated with the increased susceptibility of film to environmental influences, such as temperature, humidity, pressure, etc. Great care is necessary to ensure that the shallow and deep doses are accurately discriminated and measured.

An ideal dosimeter would directly measure doses at 7, 1000, and perhaps 300 mg/cm^2 (shallow, deep, and lens of eye doses). In practice, the dose at such depths in tissue must be inferred from a combination of measurements with different filters. TLD and film elements are mounted in a badge arrangement, which is covered by at least 10 to 30 mg/cm^2 of Mylar, paper, or other covering for mechanical and/or protective reasons.

6.4.1 Energy Dependence

Personnel dosimeters are beta energy-dependent for the same reason that survey instruments are beta energy-dependent. That is, the reading obtained from the dosimeter is proportional to the average rate of energy deposition through the "sensitive volume" or body of the element. If this average energy deposition is less than the deposition at 7 mg/cm^2, then the dosimeter will under-respond.

TLD chips of lithium fluoride (0.32 cm x 0.32 cm) are about 240 mg/cm^2 thick. Significant attenuation of the beta field takes place through the body of the chip. As a result, these types of TLD chips under-respond to uranium decay betas by a factor of about 2.

Other TLD systems minimize this problem by adhering a thin layer of TL powder onto a plastic backing. Current TLD personnel dosimeters typically use multiple detectors (typically, four) under different filter thicknesses. The different responses of each element are used as input to an algorithm which provides an estimate of the effective radiation energy and the doses at depths of interest.

Detectors that are very thin minimize energy-dependence. Film detectors demonstrate a high energy-response dependence for low-energy photons, as well as beta energy-response dependence (though the beta response is less variable than that of TLD chips).

Current systems could potentially provide accurate and precise information; however, their complexity can lead to problems. Calibration of these systems should be performed by a person with specific expertise in the detector’s system and knowledge of badge response to high beta or mixed beta and gamma radiation fields.

6.4.2 Angular Dependence

The dosimeter elements must be mounted in a badge or element holder. The assembled badge usually displays severe angular dependence. Fortunately, in most cases, a worker’s normal movements will tend to average out some of this dependence. Some badge holder arrangements can flip the badge completely over so that the "beta window" of the badge is facing the worker, not the source. The design of the badge holder or strict administrative controls should be utilized to minimize this problem.

6.4.3 Dosimetry Practices

Beta and gamma fields in working areas should be well-characterized. See previous figures and tables as examples. An attempt should be made to correlate survey instrument and dosimeter badge results. Badge reading frequency should be long enough to accumulate a significant dose (100-mrem range) and short enough to allow adequate control. Dosimeter change frequencies can vary with the specific work-site conditions.

Although multiple badging is not usually necessary, it should be considered for use in very high beta fields produced by separated uranium decay products. The dosimetry system used shall meet or be specifically excepted from DOELAP standards (10 CFR § 835.402(b)) and be specifically designated for measuring both shallow and deep doses from uranium.

Dosimetry systems should be capable of providing routine results within a reasonable time period. The system of badge collection and re-distribution should be well defined and minimize the possibility of lost badges.

Badge reading systems should have established "action levels" to alert technicians or operators of unusual results. Such results should include readings or TLD element ratios in excess of certain levels. If possible, the system should automatically save glow curves of any unusual results.

The potential for badge contamination should be minimized. Where the potential for badge contamination exists, badges should be frequently checked for contamination.

6.4.4 Extremity Dosimetry

Doses to the extremities from uranium processing and handling can involve significant exposures to the skin of the hands and forearms. Doses over small areas of the skin, including those from hot particles, are discussed in detail in DOE G 441.1-4 and will not be discussed here.

Measurement of the dose to the hands and/or forearms typically are made with TLD chips or TL powder in finger rings or wrist dosimeters. Such devices do not allow for all of the sophisticated energy discrimination just discussed. The non-homogeneity of beta radiation fields coupled with the angular dependence of commonly-available extremity dosimeters can result in a probability of underestimating the dose. However, by carefully considering the typical exposure conditions at the work site (e.g., handling metal pieces, glove box work) and calibrating the dosimeters with appropriate sources (e.g., uranium plaque sources), extremity doses can be measured with acceptable accuracy for protective purposes.

Care should be exercised in preventing "obvious" underestimations of extremity dose. For example, finger rings worn on the "top" of the finger (opposite the palm side of the hand) will not measure the dose received by the palm side when handling metal rods, etc. Dosimeters worn on the wrist have been shown to underestimate the beta dose to the fingers and palm. Reference to the Bibliography information sources will provide further information in current techniques and considerations.

The general methods used to calibrate dosimeters are given in the National Bureau of

Standards Special Publication 633, *Procedures for Calibrating Neutron Personnel Dosimeters* (Schwartz and Eisenhauer, 1982). The Radiological and Environmental Laboratory (RESL) of Idaho Falls, Idaho conducts the performance test irradiations for the DOELAP program. Processors submit dosimeters for testing to the performance testing laboratories in the categories listed in Table 6-10. If the dosimeter processor passes certain accuracy and tolerance testing criteria, a team of dosimetry experts visit the processor and assess the operation of the dosimetry program, including dosimetry records and data retrieval systems, before the dosimeter processor is certified. These requirements are given in DOE STD-1111-2013, *Department of Energy Laboratory Accreditation Program Administration* (2013a) and its associated guidance documents.

Table 6-10. Performance Test Categories, Radiation Sources, adn Test Ranges for the DOELAP and NVLAP Programs

Category	Radiation Source	Test Range
Low-energy photons (high dose)	IST x-ray Beam code M150	0.1 - 50 Gy
High-energy photons (high dose)	^{137}Cs	0.1 - 50 Gy
Low-energy photons (low dose)	NIST x-ray Beam codes: M30 M50(a) S60 S75(a) M100(a) M150 H150(b)	0.3 - 100 mSv
High-energy photons (low dose)	^{137}Cs	0.3 - 100 mSv
Low-energy photons (monoenergetic)	15 - 20 keV(b) 55 - 65 keV(b)	0.3 - 50 mSv
Beta particles	^{204}Tl $^{90}Sr/^{90}Y$ Natural or depleted uranium (slab)(b)	1.5 - 100 mSv 1.5 - 50 mSv
Neutrons	^{252}Cf moderated ^{252}Cf unmoderated(b)	2.0 - 50 mSv
Photon mixtures Photon/beta mixtures Photon/neutron mixtures		2.0 - 50 mSv

(a) Category unique to the NVLAP program.
(b) Category unique to the DOELAP program. Note also that ^{241}Am (59-keV photons) may be used in place of the mono-energetic photon (55 - 65 keV) fluorescent x-ray source.

At present, only personnel dosimeters for whole body irradiations are required to be tested, but a DOE working group developed an extremity dosimetry performance testing standard. Extremity dosimeters may be voluntarily tested. DOE also conducts an inter-comparison of calibration sources used for radiation protection purposes, but in the near future DOE secondary calibration laboratories will be established to increase the consistency of radiation protection instrument calibrations to national standards.

There is some question about the correct quality factor to apply to extremity neutron dosimeters. Most quality factors are defined in terms of linear energy transfer (LET), so a numerical value for quality factor can be readily derived by calculation or measurement of the neutron energy spectra. However, the relationship between quality factor and LET was derived from biological experiments on cancer induction, especially leukemia in blood-forming organs. There are no blood-forming organs in the extremities, so there is no biological basis for large values of quality factors for extremity exposures. However, regulatory agencies typically apply quality factors derived for whole-body exposures to the extremities; thus, for compliance purposes, qualify factors should be applied for extremity exposures.

6.4.5 Dose to Lens of Eye

It is sometimes assumed that if the skin dose limit is not exceeded, the dose limit to the lens of the eye will not be exceeded. Such assumptions should be well supported by calculations or (preferably) actual measurements. See Figure 6-3 for data indicating significant uranium beta penetration of even face shields. It is suggested and is a common practice in most fabrication areas to require the use of safety glasses, a practice which tends to mitigate this concern.

6.5 External Dose Control

Reduction of personnel doses to levels that are ALARA is largely a matter of common sense applied to the principles of time, distance, and shielding. The first step in any dose control program is to adequately identify, characterize, and measure the radiation fields. Only after this step has been performed can optimum dose control be achieved for a given amount of time, money, and energy. However, other considerations may be just as important. Good housekeeping practices are vital to keep dose rates low. Even invisible dust layers on the interior surfaces of glove boxes can increase radiation fields. Storing gloves inside the glove box when not in use and placing lightweight "pie plate" shields over the glove-port openings are examples of practices that can significantly reduce dose rates.

6.5.1 Time

As a general rule, a reduction in exposure time will yield a reduction in doses. Any operation that involves high dose rates (more than a few mrem/hour) or extended exposures should be reviewed for possible reductions in a worker's exposure time. Traffic and material flow in proposed facilities should be closely examined for opportunities to reduce exposure time.

6.5.2 Distance

Beta dose rates from uranium and its decay products decrease rapidly with distance from the source due to geometry and air shielding while gamma and neutron radiation decrease less with distance due to scattering buildup. Because uranium facilities usually involve a high percentage of contact work, considerable dose reduction can result from simple techniques to make operations semi-remote and allow workers to function. Even short distances can effect significant dose reductions.

6.5.3 Shielding

Shielding is probably the most widely used (and most effective) method of reducing beta doses from uranium. Relatively lightweight, cheap, and flexible shielding (e.g., plastic or rubber mats) has been used effectively. Figure 6-3 demonstrates the spectral basis for shielding and lists a few protective clothing reduction factors. Table 6-11 lists the thicknesses of common shielding materials necessary to stop essentially all of the beta particles from uranium (i.e., ^{234m}Pa). Generally, the lower atomic number and less dense shielding materials are used whenever possible to eliminate bremsstrahlung as well as beta radiation fields.

Protective clothing commonly worn in the nuclear industry can also afford beta dose reduction. Figure 6-3 and Table 6-12 list approximate dose reduction factors provided by such clothing. Particular attention should be paid to the use of gloves for "hands-on" work. Although lightweight rubber gloves provide some reduction, consideration should be given to using heavy leather or even leaded gloves for operations that do not require manual dexterity. Such gloves can be particularly effective in handling materials emitting high beta fields from unsupported uranium decay products.

Table 6-11. Uranium Beta Shielding

Material	Approximate Material Thickness Required to Stop ^{234m}Pa Betas, cm
Air	850
Aluminum	0.41
Lead	0.10
Lucite	0.92
Pyrex Glass	0.49
Polyethylene	1.2
Stainless Steel (347)	0.14
Water	1.1
Wood	1.7 (approx.)
Uranium	0.06

Table 6-12. Uranium Beta Dose Reduction Factors

Item	Fraction of Beta Dose Remaining
Vinyl surgeon's gloves	0.95
Latex surgeon's gloves	0.87
Lead loaded, 10-mil lead equivalent	0.77
Lead-loaded, 30-mil lead equivalent	0.13
Pylox gloves	0.62
Leather, medium weight	0.62
White cotton gloves	0.89
"Tyvek" coveralls	0.98
"Durafab" paper lab coat	0.96
65% Dacron/35% cotton lab coat	0.91

Contamination build-up inside of work gloves has led to unacceptable hand doses in some facilities. Re-use of leather or cloth gloves should be reviewed carefully for such build-up. Workers should wear thin, anti-contamination gloves inside the heavy gloves.

Dose to the lens of the eye can be effectively reduced through the use of ordinary glasses, safety glasses, or face shields. Such eye protection should be required when workers are dealing with the high beta fields from concentrated uranium decay products.

6.5.4 Geometry

The beta radiation field from uranium is strictly a surface phenomenon. Dose reduction programs can take advantage of this fact in some circumstances. For example, large plates or sheets of uranium metal, if stored in racks ’ledge on," will present less of a beta (and gamma) radiation field.

6.6 Recordkeeping

10 CFR Part 835 establishes specific requirements for maintenance of records associated with area and individual monitoring. DOE G 441.1-1C (2008a) and DOE-STD-1098-2008 (2009c) provide guidance for achieving compliance with these requirements. There are no occupational radiation protection recordkeeping requirements that are unique to uranium facilities.

7 NUCLEAR CRITICALITY SAFETY

This chapter emphasizes present-day criticality concerns from the standpoint of what nuclear criticality safety and radiological control personnel in a uranium facility need to know for the DOE mission to be accomplished in a safe and cost-effective manner. It provides an overview of the administrative and technical elements of current nuclear criticality safety programs. It does not provide a definitive discourse on nuclear criticality safety principles or repeat existing guidance. For radiological control personnel who require a greater understanding of nuclear criticality safety, the listed references provide a source of detailed requirements and information.

Health physicists and other radiation protection personnel have the technical responsibility to understand nuclear principles and the impact of these principles, in the form of the radiological conditions that exist in DOE facilities as the result of the processing, handling, and storage of radioactive and/or fissile materials. Radiation protection personnel provide an additional knowledgeable resource to help recognize workplace situations that might lead to the violation of a nuclear criticality control parameter that could contribute to an inadvertent nuclear criticality event. There have been occasions in which radiation protection personnel have observed and stopped unsafe actions by facility personnel that, if allowed to continue, might have resulted in a nuclear criticality accident. Radiation protection personnel must also be aware of the potential impacts of their actions that would be viewed as routine for normal radiation protection practice, but which could result in the violation of a nuclear criticality safety control parameter. Finally, radiation protection personnel are the focus of emergency response actions should an inadvertent nuclear criticality occur. These actions include use of emergency instrumentation, accident dosimetry, radiological dose assessment, and recovery.

This section reviews 1) nuclear criticality safety regulations and standards, including technical safety requirements (TSRs) applicable to DOE facilities, 2) criticality control factors, 3) past criticality accidents and associated lessons learned, 4) roles, responsibilities, and authorities of radiological control staff with regard to nuclear criticality safety, and 5) the content of an acceptable nuclear criticality safety program.

7.1 REGULATIONS AND STANDARDS

Nuclear criticality safety program requirements for DOE facilities are presented in DOE O 420.1C Chg 1, *Facility Safety* (2015), which requires the development of a well-documented Criticality Safety Program and incorporates compliance with ANSI/ANS-8 nuclear criticality safety standards. ANSI/ANS-8 standards provide guidance and criteria on good practices for the prevention and mitigation of criticality accidents during handling, processing, storing, and transporting special nuclear materials at fuels and material facilities.

The DOE detailed requirements for criticality safety are contained in Attachment 2, Chapter 3 of the DOE Order 420.1C (2015a). Criticality safety requirements support the documented safety analysis required by 10 CFR Part 830, Subpart B.

There are two objectives for nuclear criticality safety in the Order: 1) nuclear criticality safety is comprehensively addressed and receives an objective review, with all identifiable risks reduced to acceptably low levels and management authorization of the operation documented, and 2) the public, workers, property, both government and private, the environment, and essential operations are protected from the effects of a criticality accident.

7.2 Criticality Control Factors

As noted in ANSI/ANS-8.1, *Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors* (1998), nuclear criticality safety is achieved by controlling one or more factors of the system within subcritical limits.

Control may be exercised through the following:

- Administrative procedures (e.g., requiring a mass not exceed a posted limit)
- Physical restraints (e.g., confiding a solution to a cylindrical vessel with a diameter no greater than a specified value below the subcritical limit)
- Instrumentation (e.g., utilize devices that measures the fissile concentration and prevent its buildup in a chemical system)
- Chemical means (e.g., prevention of conditions that allow precipitation, thereby maintaining concentration characteristic of an aqueous solution)
- Reliance on the natural course or credible course of events (e.g., by relying on the nature of a process to keep the density of uranium oxide less than a specified fraction of theoretical)

7.2.1 Controllable Factors

Some of the criticality safety controls used maintain a system in a subcritical state are described below.

7.2.1.1 ^{235}U Enrichment

Enriched uranium is normally required to provide sufficient fissionable material to sustain a critical nuclear reaction in a small enough mass to meet the needs of the system. Handling of natural (0.7% ^{235}U) or depleted (<0.2% ^{235}U) uranium is generally safe at DOE uranium-processing facilities because deliberate engineering efforts, such as moderation with heavy water, reactor-grade graphite, etc., are required to create a critical mass with natural uranium. However, safe-handling measures should always be observed when handling uranium of any enrichment.

7.2.1.2 Mass

The minimum mass of uranium that will sustain a chain reaction under specified conditions is called the critical mass. The minimum critical mass depends on ^{235}U enrichment and other factors, such as the amount of moderator.

7.2.1.3 Density or Concentration

Density or concentration is defined as mass per unit volume (e.g., grams/liter). A uniform solution or slurry less than 10.8 gm ^{235}U /l will be subcritical at any volume, while a concentration four or five times greater could result in the critical mass.

7.2.1.4 Moderation and Reflection

A moderator is a material that slows down fast neutrons. The most effective moderators are those materials having a low atomic weight, such as hydrogen, deuterium, beryllium and carbon. The moderator concentration is usually expressed as the ratio of the number of hydrogen atoms to the number of fissionable atoms of the isotope; thus, the extent of moderation in an aqueous solution of ^{235}U may be expressed as the $H/^{235}U$ ratio. The ratio $H/^{235}U$ may range from zero for metal, or a dry unhydrated salt, to several thousand for a dilute aqueous solution. Over this concentration range and with the assumed spherical geometry, the critical mass may vary from a few tens of kilograms (with little hydrogen) through a minimum of a few hundred grams (at optimum moderation) to infinity in a very dilute solution where the neutron absorption by hydrogen makes a chain reaction impossible. A moderated and/or reflected system allows a smaller mass of ^{235}U to become critical.

A reflected system is an assembly where the fissionable material is partly or wholly surrounded by another material having a greater neutron scattering cross-section than air. (Technically, air is a reflector, but its effect is usually negligible). In a reflected system, a fraction of the neutrons leaving the fissionable material (core) is reflected back into the fissionable material where they may induce additional fissions. The effect of a reflection is to reduce the minimum critical mass. A good reflector is a material that has a low neutron absorption cross-section. Water, concrete, graphite, and stainless steel are typically "good" reflectors, although any material will serve as a reflector. A "fully reflected" system is one where the fissionable material is totally surrounded by a reflector such that increasing the reflector thickness results in little or no decrease in the critical mass. For example, experiments at various laboratories have shown that increasing the thickness of water surrounding the fissionable material beyond 8 inches does not significantly decrease the critical mass (Paxton et. al., 1986).

7.2.1.5 Geometry or Shape

Leakage of neutrons from a system depends on the shape of the system and on the neutron-reflecting properties of surrounding materials. The shape and size of containers are determined by considering the ratio of surface area (S) to volume (V). The ratio S/V is maintained at a value that prevents a chain reaction regardless of the quantity of material contained.

7.2.1.6 Interaction or Arrays

Interaction is the exchange of neutrons between separate containers containing uranium material. An increase in the exchanged neutrons increases the fission reaction rate. Units that are subcritical individually can be made into a critical array if brought near each other.

7.2.1.7 Neutron Poisons (Absorbers)

Neutron absorbers (poisons) are nonfissionable materials that capture neutrons, thus reducing the number of neutrons available for a fission reaction. Cadmium, boron, and chlorine are examples of neutron absorbers. Boron in borosilicate glass Raschig rings and chlorine in polyvinyl chloride (CPVC) rings are poisons used in some applications.

7.2.1.8 Monitoring for Deposits for Nuclear Safety Control

One concern in many older facilities is the potential for accumulation of uranium compounds in ventilation ductwork and process piping. A program must be in effect to routinely monitor such equipment to identify uranium compound deposits in quantities that may present nuclear criticality safety concerns. The need for such a program should be determined by nuclear criticality safety specialists, based on the enrichment of material processed (both past and present) and the geometry of the ductwork or piping. Such a review and survey should also be conducted prior to shut down and decommissioning of uranium facilities. In general, the use of NaI detectors, in conjunction with single or multichannel analyzers, can often provide adequate sensitivity to determine holdup deposits. If intervening shielding reduces sensitivity and/or background gamma radiation levels are too great, neutron detectors may be effective in identifying uranium deposits, particularly for highly enriched uranium. Since the hold-up measurements are generally taken in "cpm" for maximum sensitivity, it is useful to have a correlation from "cpm" to exposure or dose units to facilitate an understanding of the relative radiological hazard.

7.2.2 Administrative Practices

Administrative practices consist of personnel, programs, plans, procedures, training, audits and reviews, and quality assurance practices used to administer a nuclear criticality safety program. Administrative controls are used in addition to physical design features, including engineered controls, to ensure nuclear criticality safety. ANSI/ANS-8.19, *Administrative Practices for Nuclear Criticality Safety* (2005) outlines administrative practices. An effective nuclear criticality safety program requires a joint effort by managers, supervisors, workers, and nuclear criticality safety staff and relies on conformance with operating procedures by all involved personnel. The following sections describe the key elements of a nuclear criticality safety program.

7.2.2.1 Nuclear Criticality Safety Program

Management should develop a nuclear criticality safety policy and ensure it is distributed to fissionable material workers. They should also delegate authority to implement the policy, monitor the nuclear criticality safety program, and periodically participate in audits of the program. Supervisory staff should ensure that nuclear criticality safety procedures are written and staff are trained in those procedures. The nuclear criticality safety staff should provide technical guidance for equipment and process design and for operating procedure development. The nuclear criticality safety staff should perform a nuclear criticality safety evaluation before starting a new operation with fissionable materials or before changing an existing operation. An independent expert should evaluate the technical adequacy of the nuclear criticality safety program periodically.

7.2.2.2 Nuclear Criticality Safety Organization

Like the radiation protection program, the nuclear criticality safety organization should report to the highest level of facility management independent of operations. Management should clearly communicate nuclear criticality safety organization responsibilities and authorities to other facility personnel. Organizational and procedural documents should clearly define lines of interaction and interfaces with other facility organizational components. Management should assign the responsibility for nuclear criticality safety in a manner that is consistent with other safety disciplines. The organization should also have an independent nuclear criticality safety review committee and have access to consultants to assist in the conduct of the criticality safety program.

7.2.2.3 Plans and Procedures

Facility nuclear criticality safety plans and procedures are critical components of the overall facility operation. The purpose of procedures is to facilitate the safe and efficient conduct of operations. These documents provide the means by which the program is conducted and prescribe how nuclear criticality safety is to be achieved. The plans and procedures describe administrative activities and the technical aspects of nuclear criticality safety analysis. The processes of procedure development, review, training, and approval should have sufficient controls to ensure that nuclear criticality concerns are properly addressed. These controls include periodically reviewing and reaffirming procedures, and properly investigating procedure deviations and reporting them to facility management and, if appropriate, to DOE.

The controls should also ensure such deviations do not recur.

Procedures should exist that address the determination and posting of nuclear criticality safety parameters. These procedures should include a description of how the limits are to be determined and how workstations are to be posted as to form, geometry controls, mass limits, moderator limits, etc.

Management should provide fire-fighting guidelines to ensure fire-fighting techniques do not violate a criticality control limit that might lead to an inadvertent nuclear criticality event. These guidelines should include the posting of specific rooms with acceptable fire-suppression techniques that can be used for a specific location or the use of notations on facility fire pre-plans (operating procedures) located at fire stations.

Recovery procedures should be in place to provide for the recovery from a nuclear criticality control limit violation. A limit violation involves exceeding the fissionable material mass limit or the moderator liquid limit, or violating any other criticality control in an operations procedure. This process should separately address both static and dynamic cases, as responses to these violations may be quite different.

Management should develop and implement nuclear criticality safety training plans and procedures for all personnel working with or near fissionable materials, as required by ANSI/ANS-8.20, *Nuclear Criticality Safety Training* (1991). This program and its associated procedures describe the program, training requirements, recordkeeping, content, responsibilities, and objectives of a facility nuclear criticality safety program.

Inspections and audits are performed to assess the success of the nuclear criticality safety program. Qualified individuals who are independent of the operation should perform the inspections and audits. The audits and inspections should verify that operating procedures and other safety standards are being followed and identify any weaknesses in the nuclear safety program. Deficiencies should be formally addressed, tracked, reported, and resolved.

7.2.2.4 Nuclear Facility Safety Analysis

Documented Safety Analyses (DSA) document the analysis and potential consequences of accidents and abnormal occurrences at nuclear facilities. Per 10 CFR § 830.204 (2011f), with respect to a nonreactor nuclear facility with fissionable material in a form and amount to pose a potential for criticality, the DSA defines a criticality safety program that: ensures that operations with fissionable material remain subcritical under all normal and credible abnormal conditions, identifies applicable nuclear criticality safety standards, and describes how the program meets applicable nuclear criticality safety standards.

7.3 CRITICALITY ACCIDENT EXPERIENCE

Criticality accidents, sometimes called criticality excursions, can either be single pulse, multiple pulse, or "steady state" (continuous) excursions.

7.3.1 Types of Criticality Accidents

In a pulse-type criticality accident, there is an initial pulse of 10^{16}-10^{18} fissions over a short time-period (less than 1 second), sometimes followed by additional lower-intensity pulses. In a fissionable material solution, the pulse or spike is terminated by the heating and consequent thermal expansion of the solution and by bubble formation that serves to reconfigure the fissionable mass into a noncritical configuration (Paxton, 1966). If the initial pulse results in a loss of solution from the container (e.g., by splashing) or redistribution of material, the criticality event may conclude without further pulses. However, if there is no loss of material as the solution cools, it may form a critical mass once again and pulse with slightly lower fission yield.

Criticality accidents can result in lethal doses of neutron and gamma radiation at considerable distances from the accident site (on the order of tens of meters). There can also be high level beta-gamma residual radiation levels from fission products after the excursion is concluded. The heat generated during the excursion can melt parts of the system that contain the fissionable material (Moe 1988).

Moe reviewed estimated prompt radiation doses from excursions in a moderated system and a metallic system, as well as dose rates from residual contamination left by a criticality excursion. Assuming a burst of 10^{18} fissions in an unshielded, water-moderated system, the total absorbed dose is estimated to be >600 rad up to 6 m and >100 rad up to about 15 m. The gamma/neutron ratio of the total absorbed dose was 2.8. An excursion of 3 x 10^{15} fissions in a metallic, partially reflected ^{239}Pu assembly, assuming no shielding, yielded total absorbed doses of >600 rad up to approximately 10 m and >100 rad up to approximately 25 m. The gamma/neutron absorbed dose ratio was 0.1. In general, for a moderated system, the gamma dose would be expected to be higher than the neutron dose and, for a metal system, the neutron dose would be expected to be higher than the gamma dose.

Moe (Moe 1988) noted that for an excursion of >10^{18} fissions, dispersion of the fissionable material and fission products would occur, resulting in heavy local contamination and subsequent high residual dose rates. This dose rate was estimated at >1000 rad/h at 100 ft shortly after the burst and >10 rad/h at 30 ft an hour after the burst. This is the basis for instructing workers to immediately run from the work area when the criticality alarm is sounded. Seconds can save significant dose, if not from the excursion itself, then from any residual radiation that is in the area.

7.3.2 Summary of Past Criticality Accidents

Current criticality safety practice has been influenced both by the overall experience of the nuclear industry and by the analysis of the accidental criticality excursions that have occurred. Los Alamos National Laboratory has published LA-13638, A *Review of Criticality Accidents* (McLaughlin et al., 2000) which provides a description of 60 criticality accidents. According to LA-13638, there have been 22 criticality accidents in chemical process facilities. Twenty-one of the 22 occurred with fissile material in solutions or slurries, one occurred with metal ingots. No accidents occurred with powders.

Overall, the consequences from the 22 accidents have been 9 deaths, 3 survivors with limbs amputated, minimal equipment damage, and negligible loss of fissionable material. One of these incidents resulted in measurable exposure to the general public (well below allowable worker annual exposures). All accidents have been dominated by design, managerial, and operational failures. The focus for accident prevention should be on these issues.

7.4 CRITICALITY ALARMS AND NUCLEAR ACCIDENT DOSIMETRY

Guidelines for criticality alarm systems and nuclear accident dosimetry are presented in this section. Criticality alarm systems provide rapid warning to individuals in the immediate accident location and nearby locations to evacuate to a predesignated assembly location. Specific requirements for the criticality alarm system are found in ANSI/ANS-8.3, *Criticality Alarm System (1997)*. Key requirements that may be of interest for the radiological control staff are summarized in Section 7.4.1. Paxton noted lives have been saved in past criticality accidents by radiation alarms coupled with effective evacuation procedures. Nuclear accident dosimetry, discussed in Section 7.4.2, provides the means for determining the dose to workers in the vicinity of the excursion.

7.4.1 Criticality Accident Alarm System (CAAS)

As specified in ANSI/ANS-8.3 (1997), the need for a CAAS shall be evaluated for all activities in which the inventory of fissionable material in individual unrelated work areas exceeds 700 g of ^{235}U, 520 g of ^{233}U, 450 g of ^{239}Pu or 450 g of any combination of these three isotopes.

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

b. If the fissionable material mass exceeds the ANSI/ANS-8.3 limits and the probability of criticality is greater than 10^{-6} per year, but there are no occupied areas in which the expected dose exceeds 12 rad in free air, then only a criticality detector system (i.e., nuclear accident dosimetry) is needed.

c. If the fissionable material mass exceeds the ANSI/ANS-8.3 limits, but a criticality accident is determined to be impossible or less than 10^{-6} per year (per a Documented Safety Analysis), then neither a criticality alarm nor nuclear accident dosimetry is needed.

The alarm signal shall be for immediate evacuation purposes only and of sufficient volume and coverage to be heard in all areas that are to be evacuated. Information on sound levels of the alarm can be found in ANSI/ANS-8.3 (1997). The alarm trip point shall be set low enough to detect the minimum accident of concern. The minimum accident of concern may be assumed to deliver the equivalent of an absorbed dose in free air of 20 rad at a distance of 2 meters from the reacting material within 60 seconds. The alarm signal shall activate promptly (i.e., within 0.5 second) when the dose rate at the detectors equals or exceeds a value equivalent to 20 rad/min at 2 meters from the reacting material. A visible or audible warning signal shall be provided at a normally occupied location to indicate system malfunction or loss of primary power. Each alarm system should be tested at least once every three months. An evacuation drill shall be conducted at least annually.

Criticality accident alarm systems may consist of one to several detectors per unit. In multi-detector units (e.g., three detectors), at least two detectors shall be at the alarm level before initiating the alarm; in redundant systems, failure of any single channel shall not prevent the CAAS from functioning.

7.4.2 Nuclear Accident Dosimetry

Nuclear accident dosimetry shall be provided for installations that have sufficient quantity of fissionable material such that the excessive exposure of individuals to radiation from a nuclear criticality accident is possible (10 CFR § 835.1304(a)).

Requirements for nuclear accident dosimetry programs at DOE facilities are found in 10 CFR Part 835. A nuclear accident dosimetry program shall include the following:

a. a method to conduct initial screening of individuals involved in a nuclear accident to determine whether significant exposures have occurred

b. a system of fixed nuclear accident dosimeter units. Sometimes referred to as area dosimeters, the dosimeters should be capable of yielding estimated radiation dose and the approximate neutron spectrum at their locations

c. personal nuclear accident dosimeters (PNADs)

d. methods and equipment for analysis of biological materials (such as ^{24}Na activity in blood and ^{32}P activity in hair)

7.4.2.1 Initial Screening Evaluation

A nuclear accident dosimetry program should provide absorbed dose information within 24 hours after the incident. A method should be established for immediately identifying individuals exposed to total absorbed doses exceeding 50 rad (ANSI N13.3-2013, 2013b). Discussions on initial screening evaluations to segregate exposed from unexposed individuals (sometimes referred to as "quick sort techniques") are found in several references (Moe, 1988; Delafield, 1988; Petersen and Langham, 1966; Hankins, 1979; Swaja and Oyan, 1987).

A common initial screening method is to provide all workers in areas requiring nuclear accident dosimetry with an indium foil in their personnel dosimeter or security badge. During a criticality excursion, the foil will become activated by neutrons per the ^{115}In (n, gamma) ^{116m}In reaction and can be measured with a portable beta-gamma survey instrument or ion chamber. The ^{116m}In has a 54-minute half-life and releases a 1-MeV beta (maximum energy) and a 1.3-MeV gamma (80% of the time).

An alternate screening is to measure body activity due to neutron activation of the sodium in the blood via the ^{23}Na(n, gamma)^{24}Na reaction. Sodium-24 has a 15-hour half-life and releases a 1.4-MeV beta (maximum energy) and two gammas (1.37 MeV and 2.75 MeV). A beta-gamma survey meter is used to measure the ^{24}Na activity in the blood by placing the detector probe against the individual's abdomen and having the individual bend forward to enclose the detector (Moe 1988). Alternatively, the probe can be positioned under the armpit with the open window facing the chest area. Moe (1988) noted this method is less sensitive than the use of indium foils and even a small reading can indicate a significant exposure. An approximate equation to calculate worker dose (D) based on body weight and instrument reading is shown in the following Equation:

$$D(Gy) = \frac{80 \text{ (instrument reading in mR/h)}}{\text{Body weight (lb)}} \qquad ($$

Differences in incident neutron energy spectrum, orientation, and measurement techniques relative to conditions used to develop activity-dose correlations can cause significant errors in estimated radiation dose based on quick-sort surveys. Swaja and Oyan (1987) showed radiation doses estimated from induced body activity can vary by a factor of approximately 2 because of neutron energy spectrum or orientation effects and by as much as 30% due to probe position. Doses based on indium foil activity can vary by a factor of approximately 9 due to neutron energy spectrum effects, a factor of 3 depending on foil orientation relative to the incident field, and a factor of approximately 2 due to probe window setting. Swaja and Oyan (1987) recommended those count rates above background during quick-sort techniques should be initially interpreted only as an indication that the person has been exposed.

7.4.2.2 Fixed and Personnel Nuclear Accident Dosimeters

A comprehensive nuclear criticality dosimetry system should consist of stationary (fixed-location, area) dosimeters, neutron and gamma dosimeters worn by personnel (i.e., PNADs), and specialized laboratory equipment to evaluate the dosimeters.

Fixed nuclear accident dosimeter units should be capable of determining neutron doses in the range of 10 rad to 10,000 rad with an accuracy of ±25%. They should also be capable of providing the approximate neutron spectrum to permit the conversion of rad to rem. The gamma-measuring component of the dosimeter should be capable of measuring doses in the range of 10 rem to 10,000 rem in the presence of neutrons with an accuracy of about ±20%. The number of fixed dosimeter units needed and their placement will depend on the nature of the operation, structural design of the facility, and accessibility of areas to personnel. Generally, dosimeters should be placed so there is as little intervening shielding and as few obstructions as possible. The number and placement of dosimeters should be periodically re-verified to reflect changes in building design and operations. Ease of dosimeter recovery after a criticality event should be considered in their placement, including the possible need for remote retrieval.

PNADs should be worn by all individuals who enter a controlled area, with locations requiring an installed criticality alarm system. The PNADs should be capable of determining gamma dose from 10 rad to 1000 rad with an accuracy of ±20% and neutron dose from 1 rad to 1000 rad with an accuracy of ±30% without dependence upon fixed-unit data.

The general criteria of ANSI N13.3-2013 (2013b) for nuclear accident dosimeters are reviewed below. Dosimeters, both fixed and personnel, should be protected against radioactive contamination to avoid false measurements. Periodic inventory methods should be established and audits made to ensure the dosimeters are not removed or relocated without appropriate approvals. Techniques for estimating the effect of body orientation at the time of the exposure should also be developed.

- *Neutron-Measuring Component of Dosimeter.* Criticality accidents create a wide range of neutron energies. Since the neutron dose per unit fluence is strongly dependent on neutron energy, knowledge of the neutron energy spectrum is important in accident dosimetry. In criticality accidents, neutrons with energies greater than 100 keV contribute most of the dose; therefore, measurement of the fast neutron dose is of the most importance. See Delafield (Delafield 1988) for a review of the different types of neutron dosimeters available for accidents.

- *Gamma-Measuring Component of Dosimeter.* Delafield noted the ratio of the gamma rays to neutron dose will vary according to the type of critical assembly and whether or not additional shielding is present. For unshielded assemblies, the gamma-to-neutron ratio can range from 0.1 for a small heavy-metal system up to approximately 3 for a small hydrogen-moderated solution system. A concrete or hydrogenous shielding material will increase the gamma-to-neutron ratio. Gamma dose can be determined by TLD, film, or radiophotoluminescent glass.

- *Dosimeter Comparison Studies.* Sims and Dickson (Sims and Dickson, 1979; Sims, 1989) present a summary of nuclear accident dosimetry intercomparison studies performed at the Oak Ridge National Laboratory Health Physics Research Reactor. The more recent summary showed that of the 22 studies conducted over 21 years, 68% of the neutron dosimeter results were within the ±25% accuracy standard and 52% of the gamma dosimeter results were within the ±20% accuracy standard. Most measurements that failed to meet the accuracy standards overestimated the actual dose. Some of their other findings include the following:

  - Doses from hard neutron energy spectra are more accurately measured than those from soft energy spectra.

  - The threshold detector unit (TDU) is the most accurate type of nuclear accident neutron dosimeter; however, its use is declining due to increasingly strict control of small quantities of fissionable materials.

  - Activation foils (ACT) are the most popular nuclear accident neutron dosimeter.

  - For gamma dosimeters, TLDs are the most popular and the least accurate, and film is the least popular and the most accurate.

7.4.2.3 Biological Indicators

Earlier in this section, a quick-sort method was described that uses neutron activation of sodium in the blood as an indicator of worker exposure. More sophisticated laboratory analysis of blood samples can be performed to obtain a more accurate estimate of worker dose (Delafield, 1988; Hankins, 1979). The use of neutron activation of sulfur in hair ($^{32}S(n,p)^{32}P$) is another method to estimate absorbed dose for workers involved in a criticality accident. The orientation of the subject can also be determined by taking samples of hair from the front and back of the person. Hankins described a technique for determining neutron dose to within ±20-30% using a combination of blood and hair activations. The evaluation was independent of the worker's orientation, of shielding provided by wall and equipment, and of neutron leakage spectra.

7.5 RESPONSIBILITIES OF RADIOLOGICAL CONTROL STAFF

The radiological control staff should have a basic understanding of program structure, engineering criteria, and administrative controls as related to nuclear criticality safety and reviewed in earlier sections of this chapter. However, the health physicist's primary responsibilities with regard to nuclear criticality safety include emergency instrumentation and emergency response actions.

7.5.1 Routine Operations

During routine operations, the radiological control staff's primary responsibility related to nuclear criticality safety will include calibrating, repairing, and maintaining the neutron criticality alarm detectors and nuclear accident dosimeters, and maintaining appropriate records. The radiological control staff should be knowledgeable of criticality alarm systems, including alarm design parameters, types of detectors, detector area coverage, alarm set-points, and basic control design. The staff should also be familiar with locations and scenarios for designing the fixed nuclear accident dosimetry program and formulating plans for emergency response.

The radiological control staff should maintain an adequate monitoring capability for a nuclear criticality accident. In addition to the criticality alarm systems and the fixed nuclear accident dosimeters discussed above, remotely operated high-range gamma instruments, personal alarming dosimeters for engineering response/rescue teams, neutron-monitoring instrumentation (in case of a sustained low-power critical reaction), and an air-sampling capability for fission gases shall be maintained.

Other support activities may include assisting the nuclear criticality safety engineer or operations staff in performing radiation surveys to identify residual fissionable materials remaining in process system or ventilation ducts.

7.5.2 Emergency Response Actions

The priorities of the radiological control staff during a criticality event should be to rescue personnel, prevent further incidents or exposures, and quickly identify those who have been seriously exposed. To support these emergency response actions, the radiological control staff should be trained in facility emergency procedures. These emergency procedures include evacuation routes, personnel assembly areas, personnel accountability, care and treatment of injured and exposed persons, means for immediate identification of exposed individuals, instrumentation for monitoring the assembly area, and re-entry and formation of response teams.

Emergency response procedures for conducting the initial quick-sort of workers should specify measurement techniques and require that surveyors record methods and instrument settings used for quick-sort operations to ensure proper interpretation of the results. Surveyors/analysts should compare field results to pre-established activity-dose relationships developed as part of emergency response procedures to determine if a worker was exposed. Other indicators, such as a discharged self-reading dosimeter, could also indicate a possible exposure.

As an immediate follow-up action on workers identified as being exposed during a quick-sort procedure, a more accurate dose estimate should be made using PNADs, fixed-location accident dosimeters, or biological activity analyses (^{24}Na in the blood or ^{32}P in the hair). The more accurate analyses should include: 1) better definition of source characteristics, 2) location of moderating materials, and 3) location and orientation of the person(s) at the time of exposure and action of the person following the irradiation. If the radiological control staff are involved in the rescue and initial monitoring procedures, they can provide valuable information to support this analysis, particularly regarding the location and orientation of workers to the excursion.

Radiological control staff should be responsible for retrieving fixed nuclear accident dosimeters and ensuring that PNADs from any exposed workers are submitted for analysis.

7.5.3 Special Considerations during Decommissioning Activities

Before decommissioning or disposal of any facilities or equipment, an evaluation should be performed to assess the potential holdup of fissionable material in any equipment. These types of measurements may require the assistance of radiological control staff.

Some strippable coatings and surface-fixing films are effective neutron moderators. Nuclear criticality safety specialists should be consulted when using these coatings to decontaminate surfaces because criticality could be a concern, depending on the geometry of the removed coating when in the disposal unit.

8 WASTE MANAGEMENT

A material is a waste once there is no identified use or recycle value for it. Normally, wastes are considered by their physical form as either solids, liquids, or gasses, except that containerized liquids are considered solid waste under some of the current regulations. Although these forms are each processed differently, there are interrelationships. For example, it may be possible to reduce solid waste by replacing disposable protective clothing with reusable clothing that must be laundered. The laundry will produce liquid waste. In treating liquid waste, solids may be generated, e.g., filters or ion exchange resins. By careful engineering, waste generation, and treatment alternatives, a site can minimize the total waste volume and elect to generate types of waste that can be disposed of. The following sections address potentially contaminated waste and waste terminology and handling of airborne waste, solid waste, and liquid waste. The treatment of excess materials to reclaim uranium is not a waste treatment process and is not discussed here.

The Atomic Energy Act, 1954, as amended, categorizes uranium as source material (natural) or as special nuclear material (enriched). For this reason uranium is exempted from regulation by the Resource Conservation and Recovery Act (RCRA) in 40 CFR § 261.4(a)(4). However, if these materials are mixed with other wastes, they may be subject to RCRA. For this reason, it is important to prevent comingling, mixing, dissolving, etc. of uranium with hazardous waste.

8.1 POTENTIALLY CONTAMINATED WASTES

Wastes are generated within a plant or facility as a consequence of creating the uranium product(s) for which the plant was designed. Uranium may be entrained in the air, may contaminate equipment, materials, or other scrap, or may be contained in low concentrations in liquid wastes and effluents. Wastes resulting from operation of a uranium facility may include radioactive, nonradioactive, and mixed materials in the form of liquids and gaseous effluent or solids requiring disposal.

Uranium recovery operations and processes are an operational feature of most major facilities handling large quantities of material for at least two major purposes, i.e., to salvage valuable material and to reduce effluent concentrations and volumes to acceptable levels.

The facility and all waste systems must be designed to minimize wastes that result in the release of radioactive materials, during normal plant operation, the occurrence of a Design Basic Accident (DBA) meeting the regulatory limits, and conditions in which dose is kept as low as reasonably achievable. Waste systems include retention containers, cleanup systems for liquids and solids, and analytical equipment.

Accounting for waste management for solid and liquid wastes is discussed below.

8.1.1 Solid Waste

Facilities should provide for the safe collection, packaging, inventory, storage, and transportation of solid waste that is potentially contaminated with radioactive materials. Such provisions include adequate space for sorting and temporary storage of solid waste, equipment for assay of the waste, and facilities for volume reduction appropriate to the types and quantities of solid waste expected. All packages containing potentially contaminated solid waste should be appropriately monitored, both before being moved to temporary storage locations and before being loaded for transport to a disposal site.

8.1.2 Liquid Waste

Industrial wastes such as discharge from mop sinks, overflow from positive pressure circulating waste systems, and process steam condensate (if existing) should be analyzed, collected and transferred to a liquid waste treatment plant or similar treatment area if mandated by the chemical analysis. Provisions should be made for continuous monitoring and recording of radioactivity, flow volume, and pH. The radioactivity monitor should have an alarm located in the liquid waste treatment plant or area. Consideration should be given to retention systems.

Liquid process wastes should be collected and monitored near the source of generation before batch transfer through appropriate pipelines or tank transfer to a liquid waste treatment plant or area. These wastes should be individually collected at the facility in storage tanks that are equipped with stirrers, sampling and volume-measuring devices, and transfer systems. Waste storage tanks and transfer lines should be designed and constructed so that they are fully inspectable and that any leakage can be detected and contained before it reaches the environment. Consideration of nuclear criticality (See previous section) is necessary to prevent accidents.

Sanitary wastes include the nonradioactive wastes usually found at a facility, e.g., discharges from non-contaminated chemical laboratories, showers, and lavatories. The sanitary waste system and the uranium-handling area should not be connected. Sanitary sewers should discharge into an onsite, approved sanitary-sewage treatment system. Current Federal, state, and local codes regarding the discharge of sanitary wastes must be met.

8.2 DESIGN OF WASTE PROCESSING SYSTEMS

Process system designs may be characterized by their design objectives and the effluents of concern.

8.2.1 Objectives

A principal design objective for process systems is to minimize production of wastes at the source. One of the primary design objectives of any Waste Management Program is to provide facilities and equipment to handle the wastes generated and further reduce the amounts and volume of the waste. Volume-reduction facilities and equipment for liquid and solid wastes are required, as is air filtration to reduce the concentration of contaminants in the air effluent.

8.2.2 Effluents

Airborne and liquid effluents released uncontrolled to the environment are of particular concern when societal emphasis on environmental pollution control is high. Process and monitoring equipment are critical to maintaining acceptable operations.

Effluents (both radioactive and nonradioactive) from the uranium-handling facility include air and other gaseous exhausts and liquid wastes. The contamination in the effluents should be kept ALARA, commensurate with best available technology at the time of design. Emphasis should be placed on reducing total quantities of effluents (both radioactive and nonradioactive) released to the environment. Filter systems should be designed so that the effluent concentrations of uranium should not exceed the inhaled air Derived Concentration Standard (DCS) for releases, as described in DOE-STD-1196-2011, *Derived Concentration Technical Standard* (2011a) for uncontrolled areas measured at the point of discharge (e.g., exhaust ducts and stacks) during normal operations. Consideration should be given to recirculation systems for process ventilation where feasible. Provisions should be made for retention systems for liquid effluents. All effluent streams should be sampled or monitored as appropriate to ensure accurate measurements of all releases under normal and DBA conditions.

8.3 Treatment

The following sections provide information about treating airborne, liquid, and solid wastes.

8.3.1 Airborne Wastes

Ventilation control systems within a plant are designed to move air from outside "clean" areas to process areas and then to air-cleanup systems. Occupied area off-gas systems are also vented to the atmosphere and may have cleanup systems of their own. Process off-gas treatment systems consist of any or all of the following:

Wet scrubbers are generally used in dusty process off-gas situations, in which large amounts of uranium are present. The scrubbers are capable of removing and processing large quantities and serve as a prefilter to the remaining cleanup units.

Prefilter systems other than the wet scrubber are bag filters or other rough/coarse filters. The prefilters are used to remove significant quantities of particulate material from the air off-gas and are generally placed before high-efficiency particulate air (HEPA) filters in order to extend the life of the more expensive filters.

HEPA filters generally are the final filter in the process off-gas and serve to reduce the particulate effluent to insignificant or permissible levels. They may be placed in series to provide the required filtering efficiency. See section 8.3.3 for disposition of HEPA filters.

8.3.2 Liquid Waste

Because liquid effluents are generally released to the environment, liquid wastes are of equal concern with airborne wastes. Liquid effluents become available for dispersion and reconcentration in food chains, and may otherwise result in population exposure potential. In the case of liquid wastes, the concern for chemical pollutants is generally of equal concern to that of radiological contaminants. Liquid process wastes are generally collected in hold tanks, monitored, processed or treated, and released.

Hold tanks are used to collect liquid effluent prior to release in order that analyses can be performed to establish that the concentrations or total quantities are below permissible levels prior to release. The liquid can be processed or treated to remove radioactive material or neutralize chemicals.

Settling basins are frequently used to provide a means of reducing effluents further before releasing them to offsite areas.

Filtration is a simple method of removing insoluble particulate materials entrained in the liquid streams. For some processes, it is an effective and inexpensive method. The particulate material collected and filter must be periodically removed and treated as solid waste.

Ion exchange is a cleanup system for removing soluble ions from the liquid streams by collecting the material on resin columns. The contaminants must be periodically removed by a regeneration process and the materials processed, concentrated, etc., or by replacing the resin completely and treating it as solid waste.

Conversion to solid forms is a function of nearly all the processes mentioned which converts the materials removed from the liquid and airborne waste streams to more manageable forms for handling and permanent disposal.

8.3.3 Solid Waste

Solid waste come from a variety of sources in the plant from machining chips to contaminated clothing. The solid wastes should be concentrated (if possible and/or practical), packaged, and stored on the plant site for an interim time period prior to permanent disposal. Careful documentation is necessary to establish: a) quantities and nature of the waste being disposed, and b) compliance with the Resource Conservation and Recovery Act (RCRA) and other disposal and shipping/handling requirements.

Onsite volume-reduction facilities, such as incinerators, compactors, or chemical leach from metallic waste sources, can result in volume reduction in the range of 1 to 400 or more.

8.4 Monitoring

Monitoring the airborne effluents is an important aspect of control and documentation. Monitoring should be done in the stack at the discharge point and at the boundary of the uncontrolled area. In addition, total activity discharged and total mass of uranium discharged should be determined and documented to ensure that concentration requirements are not exceeded. If stack monitoring cannot be performed (e.g., in instances where the facility design is such that there are no stacks), then the reason for the monitoring method selected should also be documented.

Monitors are of two general types: continuous and passive. Continuous monitors are constructed with a radiation detector which is placed in a shielded container such that it "views" the activity as it is being collected on a filter from a sample of the stack effluent. The continuous level of radioactivity on the filter is recorded and set up in such a way that preset levels trigger an alarm. This type of monitor is less sensitive but provides an alarm in the event of mishap or equipment failure in time to take effective mitigating action.

Passive monitors consist of a continuous (isokinetic, if practical) sample collected of the effluent in the stack. The filter is periodically removed and submitted to radiological and/or chemical analyses. The sensitivity or level of detection is lower for passive sampling systems than for continuous stack samplers, and provide after-the-fact information only.

8.4.1 Air and Gaseous Effluents

All air and other gaseous effluents from confinement areas should be exhausted through a ventilation system designed to remove particulates. All exhaust ducts (or stacks) that may contain fissile contaminants should be provided with two monitoring systems. One should be of the continuous type (CAM) and the other a passive sampler. These systems may be a combination unit. The probes for sampling purposes should be designed for isokinetic sampling and located according to good industrial hygiene practices. The design of effluent monitoring systems should appropriately meet the requirements of ANSI N42.18-2014, Specification and Performance of Onsite Instrumentation for Continuously Monitoring Radioactive Effluents (2014b). Nuclear criticality safety should be considered in the design of equipment used to treat and clean up radioactive gaseous effluents.

8.4.2 Liquid Effluents

Emphasis should be placed on reducing total quantities of liquid effluents released to the environment. The contamination in the effluents should be ALARA, commensurate with the latest accepted technology at the time of design. All effluent streams should be sampled or monitored, as appropriate, to ensure accurate measurement of all releases under normal and DBA conditions. The design of effluent monitoring systems should appropriately meet the provisions of ANSI N42.18-2014 (2014b).

8.4.3 Water Collection System

Collection systems should be considered and provided where practical for water runoff from nuclear facilities containing radioactive material, such as from firefighting activities. Nuclear criticality, confinement, sampling, volume determination, and retrievability of liquids and solids should be required in the design of collection systems. The size of the collection system for firefighting water should be based on the maximum amount of water which would be collected in fighting the Design Basis Fire (DBF). The configuration of the system components should be based on conservative assumptions as to the concentration of fissile material which might collect in the system. Recirculating systems should also be considered when there is no possibility of contamination.

For special facilities that process, handle, or store uranium, the water runoff collection system should be designed with the following nuclear criticality safety considerations: 1) the maximum uranium mass loading that could be in the runoff system; 2) the most disadvantageous uranium concentrations, particle size, and uranium dispersion in the water slurry; and 3) the change in concentration of uranium and geometric configuration of the slurry as the uranium settles out of the water.

8.5 Waste Minimization

Uranium facilities shall have a waste-minimization and pollution prevention program (DOE, 2011). The objective of such a program is the cost-effective reduction in the generation and disposal of hazardous, radioactive, and mixed waste. The preferred method is to reduce the total volume and/or toxicity of hazardous waste generated at the source, which minimizes the volume and complexity for waste disposal. It is necessary to minimize the generation of RCRA mixed waste because of the difficulty and cost of later separation.

The waste minimization program applies to all present and future activities of the facilities that generate hazardous, radioactive, and/or mixed wastes. Furthermore, waste minimization is to be considered for all future programs and projects in the design stages, and should be included in all maintenance and/or construction contracts.

All managers of facilities or activities that generate hazardous, radioactive, and mixed waste are responsible for:

- minimizing the volume and toxicity of all radioactive, hazardous, and radioactive mixed waste generated, to the extent economically practicable,
- preparing and updating waste minimization plans for their waste-generating facilities or activities (small waste generators in a larger facility may be grouped with others in a facility or activity plan),
- implementing the facility-specific or activity-specific waste minimization plan,
- providing input to the organization responsible for waste characterization and minimization, to support the waste minimization program,
- communicating waste minimization plans to their employees, and ensuring that employees receive appropriate training,
- ensuring that existing system/equipment replacement or modification is designed and installed to minimize generation of waste,
- developing new waste-minimization strategies and identifying cognizant staff for waste minimization communications between facility personnel, and
- identifying new waste-generating facilities or activities and significant process changes to existing facilities or activities to the waste characterization and waste-minimization organization.

Waste volume control, or waste minimization, involves limiting the amount of material that becomes contaminated, segregating clean and contaminated material, and prolonging the useful life of equipment and material to minimize replacement. Sometimes, materials can be completely cleaned so that disposal as non-radioactive solid waste (or refurbishment in clean areas) is an option.

Program design decisions can affect uranium waste generation. For example, the quantity of protective clothing may be a significant factor. If an incinerator is available, combustible protective clothing may be selected to have a low ash content and generate a minimum of harmful effluents, such as oxides of nitrogen or halogenated compounds. In other facilities, water-washable, reusable protective clothing may minimize waste disposal.

In many nuclear facilities, contamination of packaging materials is a problem. For example, if a tool or material (e.g., a pump or some ion exchange resin) is to be used in a contaminated area, as much of the packaging material must be removed as possible before the material enters the radiological area.

Another opportunity for waste minimization occurs when materials are used as a contingency protection against contamination. For example, strippable coatings may be applied to an area that is not expected to become contaminated or may receive only minor contamination so that it can be easily cleaned. Another example involves disposable surgeons' gloves, which are routinely worn inside glove-box gloves. Unless there are serious contamination control problems in the facility, these can be surveyed and disposed of as non-radioactive solid waste rather than LLW or TRU waste.

If a piece of equipment is to have more than a single use in a contaminated environment, every possible measure should be taken to ensure its continued reliability rather than relying on frequent replacements. Tools should be of the highest quality and maximum flexibility consistent with the situation. For example, if a wrench is needed to maintain a piece of equipment in a glove-box, consideration should be given to future needs and storage provisions. A socket set with interchangeable sockets may ultimately create less waste than a box-end wrench of each size that is needed.

Likewise, all tools and equipment to be placed in a contaminated environment should be tested for reliability and preferably used on a clean mock-up to ensure their serviceability before they become contaminated in order to avoid unnecessary waste volume.

9 EMERGENCY MANAGEMENT

It is DOE policy that all DOE facilities and activities be prepared to deal with operational emergencies in a way that minimizes consequences to workers, the public, and the environment. Formal emergency management programs are the final element of DOE's defense-in-depth against adverse consequences resulting from its operations.

9.1 Emergency Management in DOE

DOE Order 151.1D (2016a) requires each DOE location, including secure transportation activities, administrative offices in the field, and headquarters offices to plan and prepare for the management of emergencies. The following discussion of emergency management principles, requirements and guidance is generally applicable to DOE uranium facilities. Specific facility requirements are in accordance with the individual facility DOE contract. The Emergency Management Guides (EMGs) provide guidance for implementing DOE Order 151.1D (2016a).

9.1.1 Basis for DOE Emergency Management Policy

DOE emergency management policy and direction is based on the following: planning and preparedness commensurate with hazards; integrated planning for health, safety, and environmental emergencies; classification of and graded response to emergencies; and multiple levels (tiers) of emergency management responsibility.

Note on Terminology: Within the Emergency Management System (EMS), "planning" includes the development of emergency plans and procedures and the identification of personnel and resources necessary to provide an effective response. "Preparedness" is the procurement and maintenance of resources, training of personnel, and exercising of plans and procedures. "Response" is the implementation of the plans during an emergency to mitigate consequences and to initiate recovery.

9.1.1.1 Planning and Preparedness Commensurate with Hazards

Because of the wide range of activities and operations under DOE's authority, standards and criteria suited to one type of facility or hazard may be inappropriate for another. To deal with this diversity in circumstances while ensuring an adequate overall state of preparedness, DOE Order 151.1D (2016a) requires that the details of each feature be tailored to the unique hazards of the facility. This approach provides a more complete and quantitative understanding of the hazards while providing for focused and cost-effective emergency planning and preparedness.

9.1.1.2 Integrated Planning for Health, Safety, and Environmental Emergencies

A wide variety of different types of Operational Emergencies can occur at DOE operations. Some may involve loss of control over radioactive or other hazardous materials unique to DOE operations, while others may involve security, transportation activities, natural phenomena impacts, environmental damage, or worker safety and health concerns. Planning, preparedness and response requirements applicable to DOE facilities and activities for some types of emergency conditions are specified by other agencies. For example, Federal regulations on occupational safety, environmental protection and hazardous waste operations have consequent “emergency planning” requirements. Rather than meet these requirements piecemeal through separate programs, each DOE/NNSA site/facility must have an Emergency Management Core Program that implements the requirements of applicable Federal, State, and local laws/regulations/ordinances for fundamental worker safety programs (e.g., fire, safety, and security). These requirements are not unique to DOE/NNSA operations.

9.1.1.3 Classification of Emergencies and Graded Response

Operational emergencies involving the airborne release of hazardous materials are grouped into one of three classes according to magnitude or severity. Classification of events is intended to promote more timely and effective response by triggering planned response actions appropriate to all events of a given classification. This principle, termed "graded response," is embodied in DOE Order 151.1D requirements and is important to the management of response resources (See DOE Order 151.1D Attach. 3, Sec. 8, *Emergency Categorization)* .

9.1.1.4 Tiers of Emergency Management Responsibility

Within the EMS, responsibility for emergency management extends from the individual facility level to the cognizant DOE field element and culminates at the cognizant Headquarters Program Secretarial Office (PSO). The responsibilities vested at each level of the hierarchy are specified in DOE Order 151.1D (2016a). The responsibility and authority for recognizing, classifying, and mitigating emergencies always rests with the facility staff. The cognizant Field Element Managers oversees the response of contractors and supports the response with communications, notifications, logistics, and coordination with other DOE elements. The DOE Headquarters (HQ) Emergency Operations Center (EOC) receives, coordinates, and disseminates emergency information to HQ elements, the cognizant PSO, Congressional offices, the White House, and other Federal Agencies (See DOE Order 151.1D Attach. A, *Responsibilities*).

9.1.2 Requirements Pertaining to All DOE Operations

DOE Order 151.1D (2016a) identifies 14 core program elements that compose each DOE facility emergency management program. The 14 core elements form a standard framework, with the details of each program element varying according to the nature and magnitude of the facility hazards and other factors. The Order requires that a Hazard Survey be used to identify the generic emergency events or conditions that define the scope of the emergency management program. Where hazardous materials, such as uranium, are present in quantities exceeding the quantity that can be "easily and safely manipulated by one person" and whose potential release would cause the impacts and require response activities characteristic of an Operational Emergency, the Order requires a facility-specific Emergency Planning Hazards Assessment (EPHA) be conducted and the results used as the technical basis for the program element content. Using the results of an objective, quantitative, and rigorous hazards assessment as its basis, each program is configured to the specific hazards and response needs of the facility.

Detailed guidance on the implementation of the Order requirements has been published by the DOE Office of Emergency Management. These EMGs provide acceptable methods for meeting the EMS Order requirements. Individual guides have been published for the technical planning basis (i.e., Hazards Survey/EPHA) processes and for programmatic and response program elements.

9.2 Specific Guidance on Emergency Management for Uranium Facilities

This section provides technical guidance that is specifically applicable to the development and implementation of emergency management programs for uranium facilities. It is intended to supplement, not replace, the more general recommendations provided in the EMGs.

9.2.1 Technical Planning Basis

9.2.1.1 Hazards Survey

The Emergency Management Core Program must be based on a Hazards Survey. A Hazards Survey establishes the planning basis for the emergency management program. It is an examination of the features and characteristics of the facility or activity to identify the generic emergency events and conditions (including natural phenomena such as earthquakes and tornadoes; wild land fires; and other serious events involving or affecting health and safety, the environment, safeguards, and security at the facility) and the potential impacts of such emergencies.
In addition to the Emergency Management Core Program requirements (DOE O 151.1D Attachment 3), DOE sites, facilities, and activities must establish and maintain the Emergency Management Hazardous Materials Program (DOE O 151.1D Attachment 4) if any site, facility or activity contains hazardous materials that were not screened out by the hazardous material screening process detailed in DOE O 151.1D (2016a) Attachment 3.

Each Hazards Survey must

a. Describe the applicable potential health, safety, or environmental impacts.

b. Identify the need for developing of further planning and preparedness beyond the

Emergency Management Core Program requirements that will apply to each type of hazard.

c. Be submitted for approval to the Field Element Manager or appropriate Federal Manager; and be updated every 3 years from date of issuance, and when there are significant changes to site/facility/activity operations or to hazardous material inventories;

d. Address natural hazards, technological hazards, and human-caused incidents.

e. Include conducting a Site Threat and Hazard Identification and Risk Assessment (Site THIRA).

A Hazardous Material Screening Process must identify specific hazardous materials and quantities that, if released, could produce impacts consistent with the definition of an Operational Emergency. The potential release of these materials to the environment requires further analysis in an EPHA. The release of hazardous materials less than the quantities listed below does not require quantitative analysis in an EPHA.

a. In general, to meet the definition of an Operational Emergency, the release of a hazardous material must: immediately threaten or endanger personnel and emergency responders who are in close proximity of the event; have the potential for dispersal beyond the safety of onsite personnel or the public in collocated facilities, activities, and/or offsite; and have a potential rate of dispersal sufficient to require a time-urgent response to implement protective actions for workers and the public.

b. The hazardous material screening process must identify all hazardous materials in a facility/activity that require further analysis in an EPHA. Specifically, for radioactive materials:

    1. All radioactive materials in a facility/activity must be subjected to a hazardous material screening process.

    2. Radioactive materials that may be excluded from further analysis in an EPHA include: sealed radioactive sources that are engineered to pass the special form testing specified by the Department of Transportation (DOT) or the American National Standards Institute (ANSI); materials stored in DOT Type B shipping containers with overpack, if the Certificates of Compliance are current and the materials stored are authorized by the Certificate; and materials used in exempt, commercially available products.

    3. Radioactive hazardous materials that require further analysis in an EPHA are those

associated with a defined Hazard Category 1, 2, or 3 nuclear facility per 10 CFR Part 830. Specifically those material contributing to the categorization of such a facility when in quantities greater than the largest Category 3 value (or if the sum of the ratios) exceeds any of the following:

- DOE-STD-1027-92, *Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports* (1997)*;*

- NA-1 SD G 1027, CN1, *Guidance on Using Release Fraction and Modern Dosimetric Information Consistently with DOE STD 1027-92* (2011).

- LA-12981-MS, *Table of DOE-STD-1017-92 Hazard Category 3 Threshold Quantities for the ICRP-30 List of 7574 Radionuclides, LANL Fact Sheet* (2002)

- LA-12846-MS, *Specific Activities and DOE-STD-1027-92 Hazard Category 2 Thresholds*, LANL Fact Sheet (1994)

4. The threshold screening quantities should not be used to eliminate from consideration very low-specific activity substances, such as depleted, natural, or low-enriched uranium in soluble forms. For those materials, chemical- not radiological-toxicity may be the dominant concern.

5. In a few circumstances, the chemical toxicity of a radioactive substance may actually be of greater health concern than the potential radiation dose. Because the DOE Category 3 radionuclide thresholds are based on radiation dose alone, chemical toxicity may need to be considered when applying the screening values to very low specific activity radionuclides (or mixtures) that are known to also be chemically toxic. For practical purposes, the concern is limited to uranium of low enrichment in the form of compounds that are relatively soluble in body fluids (such as nitrates, fluorides, and sulfates). Depending on the exact proportions of the different uranium isotopes, the chemical toxicity concern becomes dominant as the nominal enrichment (^{235}U weight percent) decreases through the range from about 16 percent to 5 percent. (Stannard, 1988).

9.2.1.2 Emergency Planning Hazards Assessment (EPHA)

Unique properties and characteristics of uranium and its compounds should be considered at certain steps in the EPHA process.

a. **Description of Facility and Operations.** The properties of the hazardous material do not significantly affect the manner in which this step of the EPHA is performed, except to the extent that uranium safety considerations may mandate more detailed descriptions of certain facility physical or operational features.

b. **Characterizing the Hazards.** The objective of this step is to describe the hazardous materials in sufficient detail to allow accurate modeling of releases and calculation of consequences. The following properties of uranium and its compounds strongly influence the release potential and consequences.

- **Chemical form.** The chemical toxicity and reactive properties of any uranium compound must be weighed against the inherent toxicity of the compound or the uranium alone. For example, gaseous uranium hexafluoride (UF_6) reacts with atmospheric moisture and undergoes hydrolysis, producing uranyl fluoride (UO_2F_2) and hydrogen fluoride (HF), a highly corrosive and toxic gas. Depending on the temperature, humidity, and uranium enrichment, the HF may be a more serious health and safety concern than either the UF_6 or the contained uranium. Some uranium compounds ignite violently on contact with air, water, or hydrocarbons.

- **Physical form.** Physical form influences the release potential and toxicity of uranium and its compounds in numerous ways. Large, monolithic pieces of uranium metal may be relatively benign; however, they can develop a pyrophoric surface due to effects of air and moisture. Finely divided metallic uranium can react violently with numerous other materials or self-ignite in air, yielding respirable particles of uranium compounds. UF_6 is a solid at ambient temperature but goes directly to a gaseous state above ~270 °F at atmospheric pressure.

- **Solubility.** For air exposure, permissible exposure levels for soluble uranium compounds are based on the chemical toxicity (particularly to the kidney), while for insoluble compounds, radiotoxicity (radiation dose to the lung) is limiting.

- **Particle size.** Particle size and the range of sizes have a large effect on the radiotoxicity of inhaled materials. Larger particles will be cleared rapidly from the upper respiratory regions, delivering little radiation dose to the lung tissues. Small particles are deposited deeper in the lung and are cleared very slowly, producing a much larger dose per unit activity inhaled.

- **Enrichment.** Enrichment, or specific activity of the isotope mixture, often determines the relative importance of radiological and chemical toxicity for more soluble materials.

c. **Developing Event Scenarios.** Properties of the hazardous material do not significantly affect the manner in which this step of the hazards assessment is performed.

d. **Estimating Potential Event Consequences.** For the scenarios developed in the previous step, this step determines the area potentially affected, the need for protective actions, and the time available to take those actions. The way these consequences are determined will depend on properties of the hazardous material. For uranium and its compounds, the following possibilities should be considered.

- **Model types.** Depending on the relative significance of radiological and chemical toxicity, the analyst may need to calculate either radiation dose, air concentration, or both for the postulated releases. For a specific scenario, different models may be needed to analyze different consequences to determine which effect is limiting (for example, radiation dose, soluble uranium intake, or HF concentration).

- **Model features.** For reactive species, the ability to model the transformation and depletion of material during transport is important to a sound analysis. Because the hydrated uranyl fluoride formed by hydrolysis of UF_6 is a solid, some will be lost due to gravitational settling as a plume moves away from the release point. When analyzing consequences of a postulated accidental criticality, correcting for the decay during transport of the short-lived fission product gases will produce a more accurate assessment of consequences.

9.2.2 Program Elements

Properties and characteristics of uranium and its compounds will also need to be considered in formulating the emergency management program elements. Following are specific program element considerations related to the hazardous properties of uranium.

9.2.2.1 Programmatic Elements

The specific properties of the hazardous material do not significantly affect the content of the programmatic elements: Program Administration, Training and Drills, Exercises, and Readiness Assurance.

9.2.2.2 Response Elements

a. **Emergency Response Organization.** The primary influence of uranium's hazardous properties on the Emergency Response Organization (ERO) is in the staffing of the consequence assessment component. As will be discussed below in Consequence Assessment, staff assigned to the ERO should be knowledgeable of, and able to quantitatively evaluate, both the health physics (radiological) and industrial hygiene (non-radiological) aspects of the hazard.

a. **Offsite Response Interfaces.** The specific properties of the hazardous material do not significantly affect the content of this program element.

b. **Operational Emergency Event Classes.** As with all hazardous materials, classification of emergencies for uranium facilities should be based on the predicted consequences at specific receptor locations, as compared with numerical criteria for taking protective action (dose, exposure, air concentration). If a material has two or more recognized modes of effect and associated protective action criteria, classification decisions should be based on the more limiting one.

c. Example: The postulated release of a quantity of a uranium compound will produce a radiological consequence corresponding to the classification criterion for Alert. The chemical toxicity of the uranium compound is such that the non-radiological consequence exceeds the criterion for Site Area Emergency. The postulated release should be classified as a Site Area Emergency.

   The appropriate classification for the postulated event or condition should be determined during the EPHA process and the observable features and indications identified as Emergency Action Levels (EALs) for that event/condition.

d. **Notification.** The specific properties of the hazardous material do not significantly affect the content of this program element.

e. **Consequence Assessment.** As discussed in section 9.2.1.2(d), models and calculation methods used for consequence assessment should be appropriate physical, chemical, and radiological properties of the hazards. Models used to calculate and project the radiological and non-radiological consequences of a release of uranium and its compounds should be the same ones used in the EPHA process. If the same models are not used, the differences between outputs should be characterized and documented to avoid the potential for confusion and indecision during response to an actual emergency.

   Environmental monitoring capability for assessing consequences of a uranium release should conform to several general principles.

   - Procedures for measurement of airborne uranium should provide for timely analysis and reporting of results in units that correspond to decision criteria. Decision points based on initial screening measurements with field instruments should account for the expected levels of radon progeny collected on the air sample media. Portable survey instruments capable of performing alpha spectroscopy measurements can be used to provide rapid isotopic analysis of uranium collected on sample media.

   - Measurement of uranium deposition should be planned and proceduralized to yield results that correspond to those produced by the predictive models used for emergency response. The correlation between direct or indirect radioactivity measurements (in units of activity) and measurement methods that give mass or concentration of uranium in a sample should be established for the expected enrichment values of material that might be released.

- If the potential exists for release of uranium compounds with high chemical toxicity (such as UF_6), it is not practical to plan to use survey teams to quantify concentrations in a plume. The high risk to survey personnel, the protective equipment necessary to minimize that risk, the time needed to prepare and position a team for such a survey, and the limited value of the information that could be gained all weigh against this approach to assessing the consequence of a highly toxic release.

- Environmental air concentrations are commonly measured continuously around the perimeter of some uranium facilities. Consequence assessment procedures should provide for the rapid retrieval and analysis of sample media from fixed samplers that may be operating in an area affected by a uranium release.

f. **Protective Actions.** Because the health consequences of a given intake of uranium, or its compounds, are highly dependent on properties such as enrichment, particle size, and solubility, facility and site-specific protective action criteria stated in terms of observable quantities and features of the release should be developed. In order for protective action criteria stated in terms of calculated dose or concentration to be valid, the calculational models should account for the properties of the material.

The Protective Action Guides (PAGs) published by the U.S. Environmental Protection Agency (EPA, 2016) have been adopted by DOE as its basic protective action criteria for planning and response.

Table 9-1. Summary Table of PAGs, Guidelines, and Planning Guidance for Radiological Incidents (EPA, 2016)

Phase	Protective Action Recommendation	PAG, Guideline, or Planning Guidance
Early Phase	Sheltering-in-place or evacuation of the public[b]	**PAG:** 1 to 5 rem (10 to 50 mSv) projected dose over four days[c]
	Supplementary administration of prophylactic drugs – KI[d]	**PAG:** 5 rem (50 mSv) projected child thyroid dose[e] from exposure to radioactive iodine
	Limit emergency worker exposure (total dose incurred over entire response)	**Guideline:** 5 rem (50 mSv)year (or greater under exceptional circumstances)[f]
Intermediate Phase	Relocation of the public	**PAG:** ≥ 2 rem (20 mSv) projected dose in the first year, 0.5 rem (5 mSv)/year projected dose in second and subsequent years
	Apply simple dose reduction techniques	**Guideline:** < 2 rem (20 mSv) projected dose in the first year
	Food interdiction[g]	**PAG:** 0.5 rem (5 mSv)/year projected whold body dose, or 5 rem (50 mSv)/year to any individual organ or tissue, whichever is limiting
	Alternative drinking water	**PAG:** pending finalization of proposal
	Limit emergency worker exposure (total dose incurred over entire response)	**Guideline:** 5 rem (50 mSv)/year
	Reentry	**Guideline:** Operational Guidelines[h] (stay times and concentrations) for specific reentry activities (see EPA, 2016 Section 4.6)
Late Phase	Cleanup[i]	**Planning Guidance:** Brief description of planning process (See EPA, 2016 Section 5.1)
	Waste Disposal	**Planning Guidance:** Brief description of planning process (See EPA, 2016 Section 5.1)

a. This guidance does not address or impact site cleanups occurring under other statutory authorities such as the United States Environmental Protection Agency's (EPA) Superfund program, the Nuclear Regulatory Commission's (NRC) decommissioning program, or other federal or state cleanup programs.
b. Should begin at 1 rem (10 mSv); take whichever action (or combination of actions) that results in the lowest exposure for the majority of the population. Sheltering may begin at lower levels if advantageous.
c. Projected dose is the sum of the effective dose from external radiation exposure (e.g., groundshine and plume submersion) and the committed effective dose from inhaled radioactive material.
d. Provides thyroid protection from radioactive iodines only. See the complete 2001 FDA guidance, "Potassium Iodide as a Thyroid Blocking Agent in Radiation Emergencies." Further information is also available in "KI in Radiation Emergencies, 2001 – Questions and Answers" 2002, and "Frequently Asked Questions on Potassium Iodide (KI)."
e. Thyroid dose. See Section 1.4.2. For information on radiological prophylactics and treatment other than KI, refer to http://www.fda.gov/Drugs/EmergencyPreparedness/BioterrorismandDrugPreparedness/ucm063807.htm, https://www.emergency.cdc.gov/radiation, and www.orau.gov/reacts.
f. When radiation control options are not available, or, due to the magnitude of the incident, are not sufficient, doses to emergency workers above 5 rem (50 mSv) may be unavoidable and are generally approved by competent authority. For further discussion see Chapter 3, Section 3.1.2. Each emergency worker should be fully informed of the risks of exposure they may experience and trained, to the extent feasible, on actions to be taken. Each emergency worker should make an informed decision as to how much radiation risk they are willing to accept to save lives.
g. For more information on food and animal feeds guidance, the complete FDA guidance may be found at http://www.fda.gov/downloads/MedicalDevices/DeviceRegulationandGuidance/GuidanceDocuments/UCM094513.pdf
h. For extensive technical and practical implementation information please see "Preliminary Report on Operational Guidelines Developed for Use in Emergency Preparedness and Response to a Radiological Dispersal Device Incident" (DOE 2009).
i. This cleanup process does not rely on and does not affect any authority, including the Comprehensive Environmental Response, Compensation and Liability Act (CERCLA), 42 U.S.C. 9601 et seq. and the National Contingency Plan (NCP), 40 CFR Part 300. This document expresses no view as to the availability of legal authority to implement this process in any particular situation.

Facilities having substantive and persuasive arguments for using other protective action threshold values may propose values that are specific to their radioactive material holdings and operations. Any alternative proposals should be supported by an analysis that addresses the four principles that form the basis for the selection of the EPA PAG values and the other considerations utilized in the selection process

g. **Medical Support.** If the potential exists for significant uranium intakes, the emergency management program should include specific planning for the quantification of exposure, diagnosis of health effects, and treatment. Medical facilities providing emergency medical support should be provided with references relating to uranium toxicity and treatment protocols. Criteria for implementing treatments such as surgical excision of contaminated tissue or use of chelating agents should be discussed with the medical staff and sources of real-time advice and assistance should be identified.

h. **Recovery and Reentry.** The specific properties of the hazardous material do not significantly affect the content of this program element.

i. **Public Information.** The specific properties of the hazardous material do not significantly affect the content of this program element.

j. **Emergency Facilities and Equipment.** Except for instruments and analysis methods used in consequence assessment, specialized facilities and equipment will not be required to meet the emergency management program needs of uranium facilities. Equipment and analytical techniques for detection and measurement of uranium in environmental sample media should have sufficient sensitivity to measure levels at or below those corresponding to decision criteria. Whereas larger sample sizes or longer counting times may be used to reduce the limit of detection for routine environmental surveillance, time constraints may dictate that more sensitive techniques be used for emergency response. Kinetic phosphorimetry, a fast, sensitive, and accurate method for direct determination of uranium, permits analysis of many sample media directly or with limited sample preparation.

10 DECOMISSIONING

At the end of the useful life, nuclear facilities are deactivated by placing the facility in a safe and stable condition by removing the fuel, draining and/or de-energizing nonessential systems, and removal of stored radioactive and hazardous materials and related actions. To minimize or eliminate long-term institutional control of a facility, with adequate regard for the health and safety of workers and the public and to protect the environment, a decision may be made to undergo a decommissioning program. Decommissioning generally takes place after deactivation, and includes surveillance and maintenance, decontamination, and/or dismantlement. The ultimate goal of decommissioning is unrestricted release or restricted use of the site.

This section provides guidance on establishing and implementing an effective decommissioning program. Major topic areas include regulations and standards, design features, decommissioning program, decommissioning techniques, and decommissioning experience. The following subsections concentrate on the radiation protection aspects of decommissioning at uranium-contaminated DOE facilities.

10.1 REGULATORY FRAMEWORK

Decontamination and decommissioning of a DOE uranium facility require adherence to a complex array of regulations and policies. The Resource Conservation and Recovery Act of 1976 (RCRA) and the Comprehensive Environmental Response, Compensation, and Liability Act of 1980 (CERCLA) are the two main regulatory programs that directly impact the decommissioning of DOE facilities.

10.1.1 Resource Conservation and Recovery Act of 1976 (RCRA)

The purpose of RCRA is to reduce the total quantity of hazardous waste generated and to prevent releases of hazardous waste to the environment by controlling waste from the *cradle-to-grave.* Section 6001 of RCRA details that the provisions of RCRA apply to Federal facilities. Section 6001 also subjects Federal facilities to civil penalties and confirms Federal employees are personally liable for criminal violations of the provisions of RCRA. If the facility is being decommissioned under a RCRA permit or order, the EPA or the State may have RCRA authority over actions taken at the facility during decommissioning. This may include authority to review documents, set timetables, or require specific closure or corrective actions. Even if a facility does not have a RCRA permit, RCRA requirements may be applicable and relevant or appropriate requirements under CERCLA.

10.1.2 Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA)

CERCLA, as amended by the Superfund Amendments and Reauthorization Act of 1986, was enacted to identify and remediate sites where hazardous substances were, or could be, released into the environment. Executive Order 12580, *Superfund Implementation*, delegated authority to the Secretary of Energy for evaluating whether conditions at sites under the jurisdiction, custody, or control of DOE meet the CERCLA definition of significant threat of release of hazardous substances. If so, DOE is authorized to conduct response activities consistent with 40 CFR Part 300, *National Oil and Hazardous Substances Pollution Contingency Plan*. For sites listed on the National Priorities List, the EPA has responsibility for ensuring that actions taken by DOE comply with CERCLA requirements and may take legal action against DOE in the event that EPA does not agree with DOE's determination.

10.1.3 National Environmental Policy Act

The National Environmental Policy Act (NEPA) establishes a national policy to ensure that environmental factors are considered in planning and decision-making by federal agencies. The decommissioning of a DOE uranium facility will require a determination whether the proposed action is a major federal action that may have significant environmental impacts and requires the preparation of an environmental impact statement (EIS). An environmental assessment (EA) may be prepared to assist in making this determination.. The EA or EIS must discuss the amount of material that will remain onsite and its effect, in addition to addressing the alternatives. The alternatives will include retaining radioactive material onsite under DOE control, cleaning the site to a level that would be acceptable for unrestricted release, and the null or no-action alternative of "walking away" from the site. If the action does not require an EA or EIS, either because the possible adverse effects are insignificant or because decommissioning was adequately addressed in a pre-operational or other EA or EIS, then the decommissioning can proceed in accordance with the information contained in other applicable regulations.

10.1.4 DOE Directives and Standards

No single DOE Directive covers all decommissioning requirements due to the wide variety of issues encompassed by decommissioning. These issues include project management, environmental surveillance, health and safety of workers and the public, engineering design, characterization survey techniques, decommissioning techniques, waste management, and waste transport.

10.1.4.1 DOE Order 435.1, *Radioactive Waste Management*

The management and disposal of radioactive waste resulting from environmental restoration activities, including decommissioning, must meet the substantive requirements for DOE O 435.1, *Radioactive Waste Management* (2007f), as well as DOE M 435.1-1, *Radioactive Waste Management Manual* (2011d). DOE radioactive waste must be appropriately managed so as to protect the public from exposure to radiation from radioactive materials; protect the environment; and protect workers, including following the requirements of 10 CFR Part 835.

10.1.4.2 DOE Order 430.1C, *Real Property Asset Management*

DOE Order 430.C, *Real Property Asset Management* (2016b), and associated guides establish an integrated corporate, holistic, and performance based approach to the life-cycle management of real property assets. It directs that acquisition, sustainment, recapitalization, and disposal is balanced to ensure real property assets are available, utilized, and in suitable condition to accomplish DOE missions. DOE O 430.1C further directs that industry standards, a balanced approach, and performance objectives are to be used in managing an asset throughout its life-cycle, including disposition. Disposition are activities that include preparation for reuse, surveillance, maintenance, deactivation, decommissioning, and long-term stewardship. DOE G 430.1-4, *Decommissioning Implementation Guide* (1999) was developed to aid in the planning and implementation of decommissioning activities in support of DOE O430.1C.

10.1.4.3 DOE Order 458.1, *Radiation Protection of the Public and the Environment* (2011d)

DOE Order 458.1, Ch. 3 (2013c), provides radiological protection requirements and guidelines for release and clearance of property and management, storage, and disposal of the wastes at DOE sites. This DOE Order establishes a basic public dose limit for exposure to residual radioactive material (in addition to naturally occurring background radiation exposures) not to exceed a total effective dose (TED) of 100-mrem (1-mSv) in a year. A more detailed discussion on authorized limits is presented in section 10.2.3.

10.2 Residual Radioactivity Levels

A primary concern in the decommissioning of any nuclear facility is the level of residual radioactivity that may be permitted for unrestricted use. Before a property that has the potential to contain residual radioactive material can be released from control, it must meet dose limits under any plausible use of the property and ALARA process requirements.

10.2.1 Public Dose Limits

DOE Order 458.1 requires radiological activities to be conducted so that exposure to members of the public to ionizing radiation will not cause a TED of greater than 100 mrem in a year, an equivalent dose to the lens of the eye not greater than 1500 mrem in a year, or an equivalent dose to the skin or extremities exceeding 5000 mrem in a year.

10.2.2 ALARA process requirements

In addition to public dose limits, a documented ALARA process must be implemented for decommissioning activities in a manner that ensures that dose to members of the public and releases to the environment are kept as low as reasonably achievable. This process must consider DOE sources, modes of exposure, and all pathways which could result in the release of radioactive materials into the environment or exposure to the public. Though the ALARA process is not applicable to non-routine events (i.e., accidental or inadvertent releases or exposures), it is applicable during recovery and remediation associated with non-routine events. DOE-HDBK-1215-2014 (2014c), *Optimizing Radiation Protection of the Public and the Environment For Use With DOE O 458.1, ALARA Requirements*, provides detailed information to assist in understanding what is necessary and acceptable for implementing the ALARA provision of the Order.

10.2.3 Dose Constraints

To ensure that public dose limits from exposure to residual radioactive material from all DOE sources are met, DOE O 458.1 applies dose constraints to each specific clearance of property for any actual or likely future use of the property. For real property, the dose constraint is set at a total effective dose of 25 mrem above background in any calendar year. For personal property, the dose constraint is set at 1 mrem above background in any calendar year.

10.2.4 Authorized Limits

Authorized limits must be established and approved for clearance of any property containing or potentially containing residual radioactive material. Authorized limits must

- Be developed in accordance with DOE O 458.1 (2013c), ALARA requirements;
- Be based on the applicable dose constraint, supported by a complete exposure pathway analysis using appropriate methodologies, techniques, parameters, and models that meet DOE quality assurance requirements under DOE O 414.1D, *Quality Assurance* (2013e);
- Be expressed in terms of concentration of radioactivity per unit surface area (e.g., dpm per 100 cm^2), radioactivity per unit mass (e.g., pCi per gram) or volume (e.g., pCi per ml), total radioactivity, or DOE controls and restrictions, if applicable;
- Explicitly state any restrictions or conditions on future use of the property necessary to ensure the basic dose limit and applicable dose constraint are not exceeded;
- Include written notification of applicable Federal, State, or local regulatory agencies or Tribal governments; and
- Be approved in accordance with DOE O 458.1 (2013c), DOE Approval of Authorized Limits.

10.2.4.1 Pre-approved authorized Limits

In accordance with DOE O 458.1 (2013c), pre-approved authorized limits may be used instead of developing specific authorized limits once the specific application of the Authorized Limits is approved by the responsible Field Element Manager. For uranium facilities, the preapproved Surface Contamination Guidelines are listed in Table 10-1.

Table 10-1. Surface Contamination Guidelines

Radionuclides	Averaged (dpm/100 cm^2)	Maximum (dpm/100 cm^2)	Removable (dpm/100 cm^2)
U^{232}	1,000	3,000	200
U-Nat, U^{235}, U^{238} and associated decay products	5,000	15,000	1,000

10.2.5 Survey and Monitoring

Survey and monitoring to support decommissioning activities must

- Use methodologies such as MARSSIM, MARSAME, or other DOE-approved methodologies.
- Meet measurement or data quality objectives.
- Use DOE-approved sampling and analysis techniques.
- Include an evaluation of non-uniformly distributed residual radioactive material, if applicable.

Survey and monitoring instrumentation must be capable of detecting and quantifying residual radioactive material, consistent with applicable authorized limits. In addition to MARSSIM and MARSAME, DOE-HDBK-1216-2015 (2015c), *Environmental Radiological Effluent Monitoring and Environmental Surveillance,* identifies procedures, systems, methods, instruments, and practices that may be used to plan and implement radiological monitoring and environmental surveillance.

10.3 Design Features for New Facilities

Design of the facility should allow easy decommissioning of equipment and materials. Details on designing facilities for ease of decommissioning are discussed in the following sections.

10.3.1 Building Materials

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

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

Metal surfaces may also require decontamination. In general, the more highly polished the surface, the easier it will be to decontaminate. If feasible, all stainless steel that will come into contact with uranium should be electro-polished before being placed into service. If high-efficiency particulate air (HEPA) filtration has failed at any time during facility operation, roofs may require decontamination. Metal roofs are easiest to decontaminate, but even these may contribute to the volume of radioactive waste unless unusual measures are taken to clean them. Built-up and composition roofs will be difficult to clean to unrestricted release levels. Interior surfaces are most easily cleaned if they were completely primed and painted before the introduction of radioactive materials into the facility. If interior surfaces are repainted during operation, their disposal as clean waste is likely to require removal of the paint. However, if the paint has deteriorated, cleaning for unrestricted use may be as difficult as if the material had never been painted. Wood will almost certainly become contaminated, as will plasterboard and other such materials.

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

Carpets are not recommended because they are difficult to clean and survey and bulky to dispose of and they do not adequately protect the underlying surface. In some areas, such as control rooms, their use may be justified by noise control requirements; however, their contamination control limitations should be considered. If used, carpets should be surveyed frequently and disposed of as radioactive waste when they become contaminated.

10.4 Ventilation Systems

In addition to decommissioning considerations, the design of the ventilation system will depend on the operations conducted in the facility. Adequate air flow for all operations and good design practices will help keep the facility clean during operations and will facilitate decommissioning. Fiberglass duct work may present a fire hazard and may be more difficult to decontaminate than stainless steel, especially stainless steel that has been electro-polished. Welded joints are less likely to collect contamination than bolted ones; however, bolted joints are easier to remove and the most contaminated areas are readily accessible for cleaning.

Filters should be positioned in ventilation systems to minimize contamination of ductwork (e.g., filtration of glove-box exhaust air before it enters a duct leading to a plenum).

10.4.1 Piping Systems

Potentially contaminated piping systems imbedded in concrete are a common and relatively expensive decommissioning problem. Most often, they must be sealed and removed last, after all other radioactive material has been removed and the building is being demolished by conventional methods. Often, they provide the major impetus for demolishing a building rather than converting it to some non-nuclear use. For this reason, it is best to run pipes in chases or tunnels that have been lined (usually with stainless steel) to prevent contamination from penetrating building surfaces. To minimize hand jackhammer work required during decommissioning, floor drains should not be enclosed in concrete.

10.4.2 Soil-Contamination Considerations

Depending on the activity levels found, locations where contaminated effluents have penetrated the ground may require excavation during decommissioning. The facility design should minimize such areas. Particular attention should be paid to storm runoff from roofs, storage areas, contaminated equipment storage, and liquid waste treatment impoundments (including sanitary sewage systems if they may receive some small amount of contamination during the life of the facility).

10.4.3 Other Features

Installed decontamination and materials-handling equipment that facilitates operation and maintenance also generally facilitates decommissioning in two ways. First, it can be used for its intended purposes of cleaning and moving equipment during the decommissioning phase. Even more important, it usually contributes to a cleaner, better-maintained facility, where nonfunctional equipment is moved out when it is no longer needed and work surfaces are kept free of spreadable contamination.

Other features include the following:

- minimizing service piping, conduits, and ductwork,
- caulking or sealing all cracks, crevices, and joints,
- using modular, separable confinements for radioactive or other hazardous materials to preclude contamination of fixed portions of the structure,
- using localized liquid transfer systems that avoid long runs of buried contaminated piping,
- using equipment that precludes the accumulation of radioactive or other hazardous materials in relatively inaccessible areas, including curves and turns in piping and ductwork,
- using designs that ease cut-up, dismantling, removal, and packaging of contaminated equipment from the facility,
- using modular radiation shielding, in lieu of or in addition to monolithic shielding walls,
- using lifting lugs on large tanks and equipment, and
- using fully drainable piping systems that carry contaminated or potentially contaminated liquids.

10.5 DECOMMISSIONING PROGRAM REQUIREMENTS

Planning for facility decommissioning should be initiated during the design phase for new facilities and before termination of operations for existing operational facilities. To assist in decommissioning activity planning the Office of Environmental Management distributed the “Decommissioning Resource Manual.” Refer to that document for guidance.

DOE Order 430.1C, *Real Property Asset Management* (DOE, 2016b), contains the requirements by which all applicable DOE projects must be managed. This technical standard sets for the requirements for real property asset life-cycle management including: planning and budgeting, acquisition, sustainment, disposition, performance measurement, and reporting systems.

10.5.1 Pre-Operational and Operational Activities

Determination of the natural background levels of radiation and of the background and fallout radionuclides is a critical step in decommissioning. These levels are best determined before the facility becomes operational. These levels need to be determined so the incremental dose occurring from material left onsite at the termination of operations can be assessed.

The contamination control practices and records maintained during facility operation will also be important. If paint is used in contamination fixation (seldom an optimum, but sometimes a necessary, practice), it should be of a distinctive color and the location should be permanently recorded. Other records are also helpful in planning and executing final decontamination for dismantling. Spills, pipe and tank leaks, ventilation failures, burial of low-level radioactive or potentially radioactive materials onsite, or other actions that might affect decommissioning shall become part of the permanent record of the facility and be considered in decommissioning planning. Insights from workers who worked at the facility during the operational phase can also provide useful information on past incidents.

10.5.2 Post-Operational Activities

DOE program organizations shall identify contaminated facilities under their jurisdiction, document the potential for reuse and recovery of materials and equipment, and develop decommissioning schedules. Before decommissioning activities begin, adequate surveillance and maintenance should be performed for inactive facilities that allow them to: 1) meet applicable radiation protection, hazardous chemical, and safety standards, 2) maintain physical safety and security standards, and 3) reduce potential public and environmental hazards. Deactivation operations, such as removing all high-level waste and stored hazardous materials, should be performed by the facility operator as part of the last operational activities before entering into the decommissioning phase.

10.5.3 Decommissioning Activities

The following discussion of decommissioning activities is divided into four phases: site characterization, environmental review process, general decommissioning planning, decommissioning project plan, and decommissioning operations.

10.5.3.1 Site Characterization

Characterization data shall be collected to support a thorough physical, chemical, and radiological characterization to fulfill the requirements of NEPA reviews, and the RCRA/CERCLA preliminary assessment/site investigations and detailed engineering studies. The facility characterization shall include the following:

- drawings, photographs, or other records reflecting the as-built and as-modified condition of the facility and grounds,

- the condition of all structures, existing protective barriers, and systems installed to ensure public, occupational, and environmental safety,

- the type, form, quantity, and location of hazardous chemical and radioactive material from past operations at the site, and

- information on factors that could influence the selection of decommissioning alternatives (safe storage, entombment, dismantling), such as potential future use, long-range site plans, facility condition, and potential health, safety, and environmental hazards.

One portion of the site-characterization process is a composite of several different types of surveys: background, scoping (or preliminary site characterization), and detailed characterization, as defined by Berger (1992). Guidance for conducting site characterizations as part of the remedial investigations/feasibility studies (RI/FS) under CERCLA can be found in EPA/540/G-89/004, *Guidance for Conducting Remedial Investigations and Feasibility Studies Under CERCLA* (1988).

The background survey information (i.e., direct radiation levels and concentrations of potential radionuclide contaminants in construction materials and soils) may be performed as part of the environmental baseline studies during pre-operational activities. Otherwise, background levels should be determined at onsite or immediately offsite locations that are unaffected by operations.

The scoping or preliminary site characterization study should be performed to identify the potential radionuclide contaminants at the site, the relative ratios of these nuclides, and the general extent of contamination. The survey provides the basis for initial estimates of the required decommissioning effort and a framework for planning the more detailed characterization study. A limited number of measurements will be made at locations that are most likely to have contamination. Scoping or preliminary site characterization surveys may be conducted during the post-operational phase.

The detailed characterization survey will more precisely define the extent and magnitude of contamination. The resulting data will be used to assist in planning for the decontamination effort, including decontamination techniques and health and safety considerations during decommissioning.

10.5.3.2 Environmental Review Process

Candidate decommissioning alternatives should be identified, assessed, and evaluated, and preferred decommissioning alternatives should be selected based on the results of the environmental review. The review should be performed according to the requirements of NEPA, RCRA, and CERCLA. Depending on the operation, the environmental review may consist of an EA or an EIS (see Section 10.2.1).

10.5.3.3 Decommissioning Planning

The first step in decommissioning planning is the development of a series of absolute criteria. These will necessarily include such items as compliance with DOE regulations and Orders, EPA regulations, interagency agreements, and other statutes. They may also include commitments to states, landowners, or others, or provisions of a Record of Decision.

As these criteria are developed, other high-value criteria may also be established. These are likely to include such considerations as maximizing the aesthetic and recreational value of the site, performing decommissioning within allocated funds, lowest worker dose, lowest population dose, lowest cost, lowest future surveillance commitment, and least effect in case of probable accidents. Depending on the viability of the decommissioning action, the decision-making process that has been established, and the level of public concern, notice of a scoping meeting may be published in the Federal Register and scoping meetings may be held. Similar actions may be taken to determine the applicable decommissioning criteria and the alternatives to be considered.

Whether or not a formal scoping meeting and EIS are used, it will be necessary to define the decommissioning options to be considered. Most of the analysis effort should be expended on those options that fulfill the absolute criteria so they can be ranked relative to the other high-value criteria. General options would typically include the following, which are taken from NUREG-0586, *Final Generic Environmental Impact Statement (GEIS) on Decommissioning of Nuclear Facilities* (NRC, 2002):

a. **Decontamination (DECON) -** Decontamination is the alternative in which contaminated equipment, structures, and portions of a facility are physically removed from the site or their radioactive contaminants are removed by chemical or abrasive means. This alternative is the preferred approach to decommissioning uranium-contaminated facilities.

b. **Safe Storage (SAFSTOR) -** SAFSTOR is sometimes referred to as "deferred decommissioning," the alternative in which nuclear facilities are placed and maintained in such a condition that the structure and contents can be safely stored and eventually decommissioned. In preparing a facility for SAFSTOR, the structure is left intact, but all nuclear fuels, radioactive fluids, and wastes are removed from the site. This alternative is generally considered when the following conditions occur:

1. Low-level waste disposal capacity is inadequate to implement DECON.

2. An adjacent operating nuclear facility would be adversely affected if the DECON alternative were implemented.

3. A positive benefit would be derived through a limited period of radioactive decay. A cost-benefit analysis should be performed, comparing total cost and radiation exposure resulting from DECON versus SAFSTOR. Then, a decision should be made whether any additional costs incurred for the SAFSTOR alternative are justified by the dose savings. Due to the long half-lives of uranium isotopes, radioactive decay is not a viable reason for using the SAFSTOR Deactivation and Decontamination option.

c. **Entombment (ENTOMB) -** The entombment alternative involves removing all nuclear fuels, radioactive fluids, and wastes from the site and encasing all structural and mechanical materials and components not decontaminated to acceptable levels in a structurally long-lived material, such as concrete. The entombed structure is maintained under appropriate continued surveillance until the radioactivity decays to a level permitting unrestricted release of the facility. The maximum allowable time in entombment should be less than 100 years. Due to the long half-lives of uranium isotopes, entombment is not a viable option for decommissioning of uranium-contaminated facilities.

d. **The no-action alternative, as required by NEPA -** In decommissioning, this is normally considered the "walk away" option.

Conversion of a facility for alternate nuclear or other controlled use has sometimes been considered a decommissioning mode; however, it is not truly decommissioned unless conversion involves removal of all radioactive material. Final disposition, when it occurs at the end of the new use, should consider the residual radioactivity onsite.

10.5.3.4 Decommissioning Project Plan

A decommissioning project plan should be prepared and should include the following:

a. physical, chemical, and radiological characterization data or references to such data,

b. a summary evaluation of decommissioning alternatives for the facility, including the preferred alternatives,

c. plans for meeting requirements from the environmental review process (NEPA, RCRA, and CERCLA),

d. radiological criteria to be used,

e. development of a health and safety plan for decommissioning,

f. projections of occupational exposure,

g. estimated quantities of radioactive waste to be generated, and

h. detailed administrative, cost-schedule, and management information.

If a contractor will be used to perform the decommissioning operations, the plan should include detailed technical specifications for selecting a contractor.

The site characterization survey should provide the necessary information on the type of facility or land area to be decommissioned and the type and amount of residual radioactive material that must be cleaned up. Other information to be considered in deciding the appropriate decommissioning alternative includes the following:

a. the availability of a final disposal facility for the radioactive waste, hazardous waste, or mixed waste,

b. the intended use of the site and components (e.g., Will the site be released for unrestricted or restricted use?),

c. the site characteristics (e.g., demography, accessibility, meteorology),

d. the cost-benefit analysis (CBA) results, and

e. the resource considerations.

10.5.3.5 Decommissioning Operations

Decommissioning operations shall be conducted according to the approved decommissioning project plan. Significant deviations from the decommissioning project plan should be approved by the responsible field organization in consultation with the appropriate program office.

During decommissioning operations, remediation control surveys (Berger 1992) should be conducted to guide the cleanup in the real-time mode. This will ensure that the decommissioning workers, the public, and the environment are all adequately protected against exposures to radiation and radioactive materials arising from the decommissioning activities.

The volume of waste and the associated cost of decommissioning the waste will be greatly reduced if equipment can be cleaned up and disposed of as either non-radioactive waste or as non-TRU waste. Numerous techniques have been developed for decontamination of equipment and materials. Established techniques and the latest technology should be considered in minimizing the quantity of contaminated equipment that requires disposal and the waste generated from the decontamination processes. These techniques are described in Section 10.4.2.

In establishing a radiological control program for decommissioning operations, the scope of the decommissioning effort should be identified. Factors to be considered in program development include:

a. the type of facility or land area to be cleaned up,

b. the type and amount of radioactive contaminated material, hazardous waste, and mixed waste,

c. the radiological and hazardous material cleanup levels, and

d. the decommissioning methods being used.

The extent of the radiological control program will depend on the selected decommissioning alternative. For the SAFSTOR alternative, the radiological control program would be minimal following deactivation (i.e., surveillance activities) until the decontamination phase is initiated, at which time a full radiological control program would be necessary. For the DECON alternative, a fully staffed radiological control program would be needed from the start of decontamination. Typically, this program would be similar to the program conducted during normal operations. Entombment is not a viable alternative for decommissioning of uranium-contaminated facilities.

Also, the hazardous and radioactive contaminants present and the specific decontamination techniques (e.g., mechanical methods, high-pressure water, abrasive cleaning, vibratory finishing, ultrasonics, electro-polishing, decontamination foams, strippable decontamination coatings, and dry ice blasting) used by each alternative will affect the extent of the radiological control program. For example, if an abrasive mechanical technique for decontaminating equipment (where airborne concentrations may be a concern) is chosen over just scrapping the equipment as waste, obviously the radiological control program will need to be more sophisticated.

10.5.4 Post-Decommissioning Activities

A final radiological and chemical survey report (or an independent verification survey report) and a project final report should be prepared. The final report should include a description of the project, the final status of the property, and the lessons learned from the project.

As defined in Berger (1992), confirmatory surveys may be performed by the regulatory agency to confirm the adequacy of the contractor's final radiological and chemical survey report. A confirmatory survey typically addresses from 1% to 10% of the site.

The responsible program organization should ensure any necessary long-term maintenance and surveillance or other safety controls are provided for the decommissioned property. The decommissioned property may be released from DOE ownership if the responsible program organization, in consultation with the Office of Associate Under Secretary for Environment, Health, Safety and Security (AU-1) certifies that the property meets applicable release criteria for residual radioactivity and hazardous chemicals. If appropriate release criteria are not met, the property may be reused for other program activities that may or may not involve radioactive or hazardous materials, provided a conversion method leading to reuse is described (DOE M 435.1-1).

10.5.5 Quality Assurance

Decommissioning activities shall be conducted according to the applicable requirements of the ANSI/ASME NQA-1, *Quality Assurance Program Requirements for Nuclear Facilities* (2015b) and other appropriate national consensus standards (e.g., EPA guidance documents in the EPA QA/R and EPA QA/G series should be used in the design of environmental monitoring programs). The quality assurance program for decommissioning activities should follow the guidelines in DOE Order 414.1D, *Quality Assurance* (2013e).

10.6 Decontamination and Decommissioning Experience

Discussions of decommissioning activities at several uranium facilities can be found in Adkisson (1987) and Wynveen et al. (1982). Decommissioning activities took place in several types of uranium facilities including an enriched uranium fuel fabrication plant, a mixed oxide (Pu/U) fuel fabrication and development plant, a research and development laboratory, and a depleted uranium manufacturing plant. Equipment decontaminated, dismantled, or removed included glove boxes, fume hoods, laboratory equipment, piping, ventilation ducts, uranium and thorium sediments from a settling lagoon, and soil from a small shallow burial area. Decontamination techniques included wiping with a damp cloth, strippable paint, acid wash, and removal of soil and sediments. Some lessons learned from these decommissioning operations included the following:

a. Waste management planning should begin early in the decommissioning planning stages and account for the possibility there may be more stringent regulations for shipping hazardous or radioactive wastes than disposing of it. Any waste package designs need to be reviewed to ensure compliance with all applicable waste management requirements.

b. Temporary contamination enclosures are effective in controlling contamination during size reduction of large equipment such as glove boxes. Any loose contamination on the equipment should be fixed prior to placing it in the enclosure.

   1. Criticality safety issues should be considered regarding the geometry of any waste material containing fissile material.

   2. Decommissioning operations must be prepared for changes in regulatory criteria and implementation of these new criteria.

   3. During decommissioning operations, personnel need to recognize the possibility of encountering elevated levels of contamination in unexpected locations such as the excavations for concrete structures or under existing roofing or flooring.

   4. It is necessary to establish and document criteria for implementing regulations in accordance with 10 CFR § 835.101. For example, in meeting surface contamination guidelines, it is important to establish the acceptable detection efficiency of the detector and areas for averaging measurements.

11 APPENDIX A - REFERENCES

Adkisson, R. J. 1987. *Fuel Fabrication Decommissioning Experience*. Proceedings of 1987 International Decommissioning Symposium, pp. III-50-III-54. CONF-87018, Vol. 1. Pittsburgh, Pennsylvania.

American Conference of Governmental Industrial Hygienists (ACGIH). 2015. *TLVs, Threshold Limit Values and Biological Exposure Indices.*

American National Standards Institute. 1979. *American National Standards Performance Specifications for Reactor Emergency Radiological Monitoring Instrumentation.* ANSI N320-1979 (R1993). New York, New York.

American National Standards Institute (ANSI). 1980. *Performance Criteria for Instrumentation Used for In-Plant Plutonium Monitoring.* ANSI N317-1980. Rev. 1991. New York, New York.

American National Standards Institute. 1990a. *Performance Specifications for Health Physics Instrumentation - Occupational Airborne Radioactivity Monitoring Instrumentation*. ANSI N42.17B-1989 (R1990). New York, New York.

American National Standards Institute (ANS). 1990b. *Performance Specifications for Health Physics Instrumentation - Portable Instrumentation for Use in Extreme Environmental Conditions.* ANSI N42.17C-1989. New York, New York.

American National Standards Institute. 1991. *Nuclear Criticality Safety Training*. ANSI/ANS-8.20 (R2015). American Nuclear Society, LaGrange Park, Illinois.

American National Standards Institute. 1997. *Criticality Accident Alarm System.* ANSI/ANS-8.3-1997 (R2012). American Nuclear Society, LaGrange Park, Illinois.

American National Standards Institute. 1998. *Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors.* ANSI/ANS-8.1-1998 (R2007). New York, New York.

American National Standards Institute. 2004a. *American National Standard Performance Specifications for Health Physics Instrumentation-Portable Instrumentation for Use in Normal Environmental Conditions*. IEEE/ANSI N42.17A-2003. New York, New York.

American National Standards Institute. 2004b. *American National Standard Specification and Performance of On-Site Instrumentation for Continuously Monitoring Radioactivity Effluents.* ANSI/IEEE N42.18-2004. (Reaffirmation of ANSI N42.18-1980; Redesignation of ANSI N13.10-1974) New York, New York.

American National Standards Institute. 2005. *Administrative Practices for Nuclear Safety*. ANSI/ANS-8.19. American Nuclear Society, LaGrange Park, Illinois.

American National Standards Institute. 2009. *Personnel Dosimetry Performance – Criteria for Testing.* ANSI N13.11-2009 (R2015). New York, New York.

American National Standards Institute. 2010. *Practice for Occupational Radiation Exposure Records Systems.* ANSI/HPS N13.6-2010. Health Physics Society, McLean, VA.

American National Standards Institute. 2011a. *Sampling and Monitoring Releases of Airborne Radioactive Substances from the Stacks and Ducts of Nuclear Facilities.* ANSI/HPS N13.1-2011. Health Physics Society, McLean, VA.

American National Standards Institute. 2011b. *Performance Criteria for Radiobioassay.* ANSI/HPS N13.30-2011. Health Physics Society, McLean, VA

American National Standards Institute. 2013a. *Bioassay Programs for Uranium.* ANSI/HPS N13.22-2013. Health Physics Society, McLean, VA.

American National Standards Institute. 2013b. *Dosimetry for Criticality Accidents.* ANSI N13.3-2013. New York, New York.

American National Standards Institute. 2014a. *American National Standard for Radiation Protection Instrumentation Test and Calibration, Survey Portable Instruments*. IEEE/ANSI N323AB-2013. New York, New York.

American National Standards Institute. 2015a. *American National Standard Practices for Respirator Protection.* ANSI/ASSE-Z88.2-2015. New York, New York.

American National Standards Institute. 2015b. *Quality Assurance Program Requirements for Nuclear Facilities.* ANSI/ASME NQA-1. Washington, DC.

Andersen, B. V., L. A. Carter, J. G. Droppo, J. Mishima, L. C. Schwendimen, J. M. Selby, R. I. Smith, C. M. Unruh, D. A. Waite, E. C. Watson, and L. D. Williams. 1974. *Technological Consideration in Emergency Instrumentation Preparedness. Phase II-B - Emergency Radiological and Meteorological Instrumentation for Mixed Oxide Fuel Fabrication Facilities.* BNWL-1742. Pacific Northwest Laboratory, Richland, WA.

Begichev, S.N., A.A. Borovoj, E.V. Burlakov, A. Ju Gagarinsky, V.F. Demin, I.L. Khodakovsky, A.A. Khrulev. 1989. *Radioactive Releases Due to the Chernobyl Accident*. Int. Sem. "Fission Product Transport Processes in Reactor Accidents. May 22-26, Dubrovnik, Yugoslavia.

Berger, J. D. 1992. *Manual for Conducting Radiological Surveys in Support of License Termination. Draft Report for Comment.* NUREG/CR-5849. U.S. Nuclear Regulatory Commission, Washington, DC.

Briesmeier, J. F., ed. 1986. *MCNP- A General Monte Carlo Code for Neutron and Photon Transport, Version 3A.* LA7396-M, Rev. 2. Los Alamos National Laboratory, Los Alamos, New Mexico.

Broadhead, B. L., C. M. Hopper, R. L. Childs and J. S. Tang. 1994. *An Updated Nuclear Criticality Slide Rule* NUREG/CR-5504. U.S. Nuclear Regulatory Commission. Washington, DC.

Brackenbush, L. W., K. R. Heid, W. N. Herrington, J. L. Kenoyer, L. F. Munson, L. H. Munson, J. M. Selby, K. L. Soldat, G. A. Stoetzel, and R. J. Traub. 1988. *Health Physics Manual of Good Practices for Plutonium Facilities.* PNL-6534. Pacific Northwest Laboratory, Richland, WA.

Carbaugh, E. H., M. J. Sula, T. L. Aldridge, and H. B. Spitz. 1988. *The Impact of Upgraded In Vivo Lung Measurement Capability on an Internal Dosimetry Program.* Annals of Occupational Hygiene 32(Supplement 1):885-892.

Carbaugh, E. H., W. A. Decker, and M. J. Swint. 1989. *Medical and Health Physics Management of a Plutonium Wound.* Radiation Protection Dosimetry. 26(1/4):345-349.

Carbaugh, E. H. 1994. *Practical Applications of Internal Dose Calculations*. Internal Radiation Dosimetry, Ch. 24 pp. 529-542. Medical Physics Publishing, Madison, Wisconsin.

Carbaugh, E. H., D. E. Bihl, J. A. MacLellan, and M. P. Long. 1994. *Hanford Internal Dosimetry Project Manual.* PNL-7001, Rev. 1. Pacific Northwest Laboratory, Richland, WA.

Carbaugh, E.H.; Bihl, D.E.; Houston, J.R. 1995. *Hanford Internal Dosimetry Project Manual*. PNL-MA-552 Rev. 2. Richland, WA: Pacific Northwest National Laboratory.

Cuddihy, R.G. 1989. *Overview of NCRP Task Group on Respiratory Tract Modeling*. Health Physics 57:291.

Delafield, H.J. 1988. *Nuclear Accident Dosimetry - An Overview*. Radiation Protection Dosimetry 23(1/4):143-149.

Devell, L. 1988. *Nuclide Composition of Chernobyl Hot Particles*. In: Hot Particles from the Chernobyl Fallout. Proc. Int. Workshop, Theuern, October 1987. (Ed. By H. Von Philipsborn and F. Steinhausler), pp. 23-34. Bergbau and Industriemuseum Ostbayern, Band 16.

Durbin, P. 1986. *Metabolic Models of Uranium*. In: Biokinetics of Uranium in Man. In: Proceedings of a Colloquium, Richland, WA, August 8-9, 1984. (NTIS, Springfield, VA USUR-05, HEHF-47).

Electric Power Research Institute (EPRI). 1989. *A Review of Plant Decontamination Methods --1988 Update.* NP-6169 RP2296-15. Applied Radiological Control, Inc., Palo Alto, California.

Environmental Protection Agency (EPA). 2016. *PAG Manual: Protection Action Guides for Planning Guidance for Radiological Incidents.* Washinton, DC.

Eve, I.S. 1966. *A Review of the Physiology of the Gastrointestinal Tract in Relation to Radiation Doses from Radioactive Materials*." Health Physics 12:131-161.

Gautier, M. A., ed. 1983. *Manual of Analytical Methods for Radiobioassay.* LA09736-M. Los Alamos National Laboratory, Los Alamos, New Mexico.

Gerber, G. B., and R. G. Thomas, eds. 1992. *Guidebook for the Treatment of Accidental Internal Radionuclide Contamination of Workers*. Radiation Protection Dosimetry 41(1).

Graham, S. G., and S. J. Kirkham. 1983. *Identification of* ^{241}*Am in the Axillary Lymph Nodes with an Intrinsic Germanium Detector*. Health Physics. 44 (Supplement):343-352.

Hankins, D. E. 1979. *Dosimetry of Criticality Accidents Using Activations of the Blood and Hair.* Health Physics 38:529-541.

Health and Welfare Canada. 1987. *Bioassay Guideline 4, Guidelines for Uranium Bioassay.* 88-EHD-139. Communications Directorate, Department of National Health and Welfare, Ottawa, Canada.

Healy, J. W. 1957. *Estimation of Plutonium Lung Burden by Urine Analysis*. American Industrial Hygiene Association Quarterly 18(3):261-266.

International Atomic Energy Agency (IAEA). 1994. *International Basic Safety Standards.* Safety Series, 115-I IAEA, Vienna.

International Commission on Radiological Protection (ICRP). 1959. *Report of Committee II on Permissible Dose for Internal Radiation.* ICRP Publication 2. Pergamon Press, Oxford, U.K.

International Commission on Radiological Protection. 1964. *Recommendations of the International Commission on Radiological Protection.* ICRP Publication 6.

International Commission on Radiation Protection. 1975. *Report of the Task Group on Reference Man.* ICRP Publication 23.

International Commission on Radiological Protection. 1977. *Recommendations of the International Commission on Radiological Protection.* ICRP Publication 26. Ann. ICRP 1(3).

International Commission on Radiological Protection. 1979. *Limits for Intakes of Radionuclides by Workers:* ICRP Publication 30. Ann. ICRP 2 (3-4).

International Commission on Radiological Protection. 1987*. Data for Use in Protection Against External Radiation.* ICRP Publication 51. Ann. ICRP 17 (2-3).

International Commission on Radiological Protection. 1989. *Individual Monitoring for Intakes of Radionuclides by Workers: Design and Interpretation.* ICRP Publication 54. Ann. ICRP 19 (1-3).

International Commission on Radiological Protection. 1991a. *1990 Recommendations of the International Commission on Radiological Protection,* ICRP Publication 60. Ann. ICRP 21 (1-3).

International Commission on Radiological Protection. 1991b. *Annual Limits on Intake of Radionuclides by Workers Based on the 1990 Recommendations*. ICRP Publication 61. Ann. ICRP 21 (4).

International Commission on Radiological Protection. 1994a. *Human Respiratory Tract Model for Radiological Protection*. ICRP Publication 66. Ann. ICRP 24 (1-3).

International Commission on Radiological Protection. 1994b. *Dose Coefficients for Intakes of Radionuclides by Workers.* ICRP Publication 68. Ann. ICRP 24(4).

International Commission on Radiological Protection. 1995a. *Age Dependent Doses to Members of the Public from Intakes of Radionuclides: Part 3 Ingestion Dose Coefficients*. ICRP Publication 69, Ann. ICRP 25 (1).

International Commission on Radiation Protection. 1995b. *Age-Dependent Doses to Members of the Public from Intake of Radionuclides: Part 4 Inhalation Dose Coefficients*. ICRP Publication 71. Ann.

ICRP 25 (3-4)

International Commission on Radiation Protection. 1995c. *Age-dependent Doses to the Members of the Public from Intake of Radionuclides – Part 5 Compilation of Ingestion and Inhalation Coefficients.* ICRP Publication 72. Ann. ICRP 26 (1).

International Commission on Radiation Protection. 1996. *Conversion Coefficients for use in Radiological Protection against External Radiation.* ICRP Publication 77. Ann. ICRP 26 (3-4).

International Commission on Radiation Protection. 1997. *Individual Monitoring for Internal Exposure of Workers.* ICRP Publication 78. Ann. ICRP 27 (3-4).

International Commission on Radiation Protection. 2001. *Doses to the Embryo and Fetus from Intakes of Radionuclides by the Mother*. ICRP Publication 88. Ann. ICRP 31 (1-3).

International Electrotechnical Commission. 2002a. *Equipment for Continuously Monitoring Radioactivity in Gaseous Effluents.* 6 vols. IEC 60761. Geneva, Switzerland.

International Electrotechnical Commission. 2002b. *Radiation Protection Instrumentation – Alpha, beta and alpha/beta (beta energy >60 keV) contamination meters and monitors.* IEC Publication 60325:2002. Geneva, Switzerland.

International Organization for Standardization. 2006. *Nuclear Energy – Reference Beta Particle Radiation.* ISO 6980:2006 (R2015). Geneva, Switzerland.

Jefferies, S. J., and K. J. Gunston. 1986. *Further Work Concerning a Case of Am Contamination in the Lateral Axillary Lymph Nodes*. Health Physics. 50(6):839-842.

Just, R. A. and V. S. Emler. 1984. *Generic Report on Health Effects for the U. S. Gaseous Diffusion Plants.* DOE Report K/D-5573, Revision 1. Washington, DC.

King, W. C. 1987. *Description and Application of the AERIN Code at LLNL.* In Proceedings of the Department of Energy Workshop on Radiobioassay and Internal Dosimetry. NL-SA-14043. Pacific Northwest Laboratory, Richland, WA.

Kruchten, D. A., and A. L. Anderson. 1990. *Improved Ultrasonic Measurement Techniques Applied to Assay of Pu and Other Transuranics in Lung.* Health Physics. 59(1):117-123.

La Bone, T. R. 1994. *The Perfect Internal Dosimetry Code*. Chapter 27 of Internal Radiation Dosimetry, O. G. Raabe, ed., pp. 585-594. Medical Physics Publishing, Madison, Wisconsin.

La Bone, T. R., F. W. Boone, D. J. Fauth, E. M. Kim, T. J. Kirkham, F. T. Ogden, and S. A. Thomas. 1993. *Savannah River Site Internal Dosimetry Technical Basis Manual.* WSRC-IM-90-139, Rev. 2. Savannah River Site, Aiken, South Carolina.

Langham, W. H. 1956. *Determination of Internally Deposited Radioactive Isotopes from Excretion Analysis.* American Industrial Hygiene Association Quarterly 17(3):305-318.

Lawrence, J. N. P. 1984. *Uranium Internal Exposure Evaluation Based on Urine Assay Data*, LA-10246-MS.

Lawrence, J. N. P. 1987. *Some Further PUQFUA Studies*. In Proceedings of the Department of Energy Workshop on Radiobioassay and Internal Dosimetry, pp. 107-122. PNL-SA-14043. Pacific Northwest Laboratory, Richland, WA.

Leggett, R. W. 1989. *The Behavior and Chemical Toxicity of U in the Kidney: A Reassessment*. Health Physics 57(3):376-383.

Lessard, E. T., X. Yihua, K. W. Skrable, G. E. Chabot, C. S. French, T. R. La Bone, J. R. Johnson, D. R. Fisher, R. Belanger, and J. L. Lipsztein. 1987. *Interpretation of Bioassay Measurements.* NUREG/CR-4884. U.S. Nuclear Regulatory Commission, Washington, DC.

Long, M.P.; Carbaugh, E.H.; Fairrow, N.L. 1994. *Practical Issues in Discriminating between Environmental and Occupational Sources in a Uranium Urinalysis Bioassay Program*. PNL-SA-24340. Richland, WA: Pacific Northwest Laboratory.

Los Alamos National Laboratory Fact Sheet. 1994. *Specific Activities and DOE-STD-1027-95 Hazard Category 2 Thresholds. LA-12846-MS.* Los Alamos, NM.

Los Alamos National Laboratory Fact Sheet. 2002. *Hazard Category 3 Threshold Quantities for the ICRP-30 List of 757 Radionuclides*. LA-12981-MS. Los Alamos, NM.

McGuire, S.A. 1991. *Chemical Toxicity of Uranium Hexafluoride Compared to Acute Effects of Radiation.* NUREG-1391. U.S. Nuclear Regulatory Commission. Washington, DC.

McLaughlin, T. P., S. P. Monahan, N. L. Pruvost, V. V. Frolov, B. G. Ryazanov, V. I. Sviridov. 2000. *Review of Criticality Accidents* LA-13638. Los Alamos National Laboratory, Los Alamos, New Mexico.

Medley, D. W., R. L. Kathren, and A. G. Miller. 1994. *Diurnal Urinary Volume and Uranium Output in Uranium Workers and Unexposed Controls.* Health Physics 67(2):122-130.

Miller, G., W. C. Inkret, and H. F. Martz. 1993. *Bayesian Detection Analysis for Radiation Exposure.* Radiation Protection Dosimetry 48(3):251-256.

Mishima, J., J. Hunt, W. D. Kittinger, G. Langer, D. Ratchford, P. D. Ritter, D. Rowan, and R. G. Stafford. 1988. *Health Physics Manual of Good Practices for the Prompt Detection of Airborne Plutonium in the Workplace.* PNL-6612. Pacific Northwest Laboratory, Richland, WA.

Moe, H. J., and E. J. Vallario. 1988. *Operational Health Physics Training.* ANL-88-26. Argonne National Laboratory, Chicago, IL.

Morrow, P.E. 1986. *Biokinetics and Analysis of Uranium in Man*. Proceedings of a Colloquium, Richland, WA, August 8-9, 1984. (NTIS, Springfield, VA USUR-05, HEHF-47).

National Council on Radiation Protection and Measurements. 1987a. *Use of Bioassay Procedures for Assessment of Internal Radionuclide Deposition.* NCRP Report No. 87. Bethesda, Maryland.

National Council on Radiation Protection and Measurements. 1987b. *Recommendations on Limits for Exposure to Ionizing Radiation.* NCRP Report No. 91. Bethesda, MD.

Occupational Safety and Health Administration (OSHA). 2006. 29 Code of Federal Regulations Part 1910.1000, *Air Contaminants*.

Pacific Northwest Laboratory. 1990. *Evaluation of Health Effects in Sequoyah Fuels Corporation Workers from Accidental Exposure to Uranium Hexaflouride.* NUREG/CR-5566. Richland, WA.

Paxton, H. C. 1966. *Criticality Control in Operations with Fissile Material.* LA-3366. Los Alamos Scientific Laboratory, Los Alamos, NM.

Paxton, H. C., N. L. Provost. 1986. *Critical Dimensions of Systems Containing U-235, Pu-239, and U-233.* LA-10860-MS. Los Alamos Scientific Laboratory, Los Alamos, NM.

Petersen, D. F., and W. H. Langham. 1966. *Neutron Activation of Sulfur in Hair: Application in a Nuclear Accident Dosimetry Study.* Health Physics 12:381-384.

Plato, P., Absorbed Bose Rate Produced by Natural Uranium as a Function of Depth in Tissue, *The International Journal of Applied Radiation and Isotopes, 3*0, 1979, pp. 109-113.

Potter, C. A. 2002. *Intake Retention Fractions Developed from Models Used in the Determination of Dose Coefficients Developed for ICRP Publication 68-Particulate Inhalation.* Health Physics 83(5):594.

Priest, N.D., G.R. Howells, D. Green, J.W. Haines. 1982. *Uranium in Bone: Metabolic and Autoradiographic Studies in the Rat.* Human Toxicol. 1:97-114.

Reilly, D., N. Ensslin, and H. Smith, Jr. 1991. *Passive Nondestructive Assay of Nuclear Materials.* NUREG/CR-5550. U.S. Nuclear Regulatory Commission, Washington, DC.

Rich, B. L., et al. 1988. *Health Physics Manual of Good Practices for Uranium Facilities.* EGG-2530. EG&G Idaho, Idaho Falls, ID.

Rowland, R.E., and J.E. Farnham. 1969. *The Deposition of Uranium in Bone*. Health Physics 17:139-144.

Schwartz, R. B., and C. M. Eisenhauer. 1982. *Procedures for Calibrating Neutron Personnel Dosimeters.* NBS Special Publication 633, U.S. Department of Commerce, National Bureau of Standards, Washington, DC.

Selby, J. M., M. Lardy, E. H. Carbaugh, T. P. Lynch, and D. J. Strom. 1994. *Internal Dosimetry Monitoring Equipment: Present and Future*. Radiation Protection Dosimetry. 53(1-4):49-58.

Sikov, M. R., R. J. Traub, T. E. Hui, H. K. Meznarich, and K. D. Thrall. 1992. *Contribution of Materials Radionuclide Burdens to Prenatal Radiation Doses.* NUREG/CR-5631, PNL-7445, Rev. 1. U.S. Nuclear Regulatory Commission, Washington, DC.

Sims, C. S. 1989. *Nuclear Accident Dosimetry Intercomparison Studies.* Health Physics 57(3):439-448.

Sims, C. S., and H. W. Dickson. 1979*. Nuclear Accident Dosimetry Intercomparison Studies at the Health Physics Research Reactor: A Summary (1965-1978).* Health Physics 3:687-699.

Skrable, K. W., G. E. Chabot, C. S. French, and T. R. La Bone. 1994. *Use of Multi-Compartment Models of Retention for Internally Deposited Radionuclides*. Chapter 14 of Internal Radiation Dosimetry, O. G. Raabe, ed., pp. 271-354. Medical Physics Publishing, Madison, WI.

Stannard, J. N. 1988. *Radioactivity and Health – A History.* DOE/RL/01830-T59. Battelle Memorial Institute. .

Stehn, J.R., Goldberg, M.D., Wiener-Chasman, R., Mughabgharb, S.F., Magurno, B.A., and May, V.M., *Neutron Cross Sections*, Brookhaven National Laboratory Report BNL-325, Volume III, Z=88 to 98, 1965.

Stewart, R. D., J. E. Tanner, and J. A. Leonowich. 1994. *An Extended Tabulation of Effective Dose Equivalent from Neutrons Incident on a Male Anthropomorphic Phantom.* Health Physics 65(4):405-413.

Strenge, D. L., R. A. Kennedy, M. J. Sula, and J. R. Johnson. 1992. *Code for Internal Dosimetry (Cindy Version 1.2).* PNL-7493, Pt. 1, Rev. 1. Pacific Northwest Laboratory, Richland, WA.

Sula, M. J., E. H. Carbaugh, and D. E. Bihl. 1991. *Technical Basis for Internal Dosimetry at Hanford.* PNL-6866, Rev. 1. Pacific Northwest Laboratory, Richland, WA.

Swaja, R. E., and R. Oyan. 1987. *Uncertainties Associated With Using Quick-Sort Techniques to Estimate Neutron Doses Following Criticality Accidents*. Health Physics 52(1):65-68.

Toivonen, H., R. Pollanen, A. Leppanen, S. Klemola, J. Lahtinen. 1992. *Release from the Nuclear Power Plant in Sosnovyy Bor in March 1992*. Radiochimica Acta 57:169-172.

U.S. Department of Energy. 1997. *Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports.* DOE-STD-1027-92, CN1. Washington, DC.

U.S. Department of Energy. 1999. *Decommissioning Implementation Guide.* DOE G 430.1-4. Washington, DC.

U.S. Department of Energy. 2000. *Decommissioning Handbook: Procedures and Practices for Decommissioning.* DOE/EM-0383. Washington, DC.

U.S. Department of Energy. 2001a. Radioactive *Waste Management.* DOE O 435.1, Chg 1 (R2007). Washington, DC.

U.S. Department of Energy. 2007a. Emergency *Management Fundamentals and the Operational Emergency Base Program.* DOE G 151.1-1A. Washington, DC.

U.S. Department of Energy. 2007b. *Knowledge Skills and Abilities for Key Radiation Protection Positions at DOE Facilities.* DOE-STD-1107-2007, CN 1. Washington, DC.

U.S. Department of Energy. 2007c. Technical *Planning Basis.* DOE G 151.1-2. Washington, DC.

U.S. Department of Energy. 2007d. Programmatic *Elements.* DOE G 151.1-3. Washington, DC.

U.S. Department of Energy. 2007e. *Response Element.* DOE G 151.1-4. Washington, DC.

U.S. Department of Energy. 2007f. *Radioactive Waste Management.* DOE O 435.1. Washington, DC.

U.S. Department of Energy. 2008a. *Radiation Protection Programs Guide for use with Title 10, Code of Federal Regulations, Part 835, Occupational Radiation Protection,* DOE G 441.1-1C, Admin Chg 1. Washington, D. C.,

U.S. Department of Energy. 2009a. General *Employees Radiological Training.* DOE-HDBK-1131-2007, CN 1 (Reaffirmed 2013). Washington, DC.

U.S. Department of Energy. 2009b. *Guide to Good Practices for Occupational Radiological Protection in Uranium Facilities*, DOE-STD-1136-2009, Washington DC.

U.S. Department of Energy. 2009c. *Radiological Control*, DOE-STD-1098-2008, Washington DC.

U.S. Department of Energy. 2010. *Packaging and Transportation Safety.* DOE O 460.1C. Washington, DC.

U.S. Department of Energy. 2011a. *Derived Concentration Technical Standard*. DOE-STD-1196-2011. Washington, DC.

U.S. Department of Energy. 2011b*. Integrated Safety Management System Policy.* DOE P 450.4A. Washington, DC.

U.S. Department of Energy. 2011c. *Radiological Control Technician Training Program* DOE-HDBK-1122-2009, CN 2. Washington, DC.

U.S. Department of Energy. 2011d. *Radioactive Waste Management Manual*. DOE M 435.1-1 Admin Chg 2. Washington, DC.

U.S. Department of Energy. 2011e. Title 10 Code of Federal Regulation, Part 1008. *Records Maintained on Individuals (Privacy Act).* 10 CFR 1008. Washington, DC.

U.S. Department of Energy. 2011f. Title 10, Code of Federal Regulations, Part 830, *Nuclear Safety Management.* Washington, DC.

U.S. Department of Energy. 2011g. Title 10, Code of Federal Regulations, Part 835, *Occupational Radiation Protection*. Washington, DC.

U.S. Department of Energy. 2013a. *Department of Energy Laboratory Accreditation Program Administration.* DOE-STD-1111-2013. Washington, DC.

U.S. Department of Energy. 2013b. *Personnel Selection, Training, Qualification, and Certification*

Requirements for DOE Nuclear Facilities. DOE Order 426.2, Chg 1. Washington, DC.

U.S. Department of Energy. 2013c. *Radiation Protection of the Public and Environment.* DOE Order 458.1, Admin Ch.3. Washington, DC.

U.S. Department of Energy. 2013d. *Radiological Worker Training.* DOE-HDBK-1130-2008, CN 2 (Reaffirmed 2013). Washington, DC.

U.S. Department of Energy. 2013d. *Records Management Program.* DOE Order 243.1B, Chg 1. Washington, DC.

U.S. Department of Energy. 2013e. *Quality Assurance.* DOE Order 414.1D, Admin Chg 1. Washington, DC.

U.S. Department of Energy. 2014a. *Conduct of Operations.* DOE Order 422.1, Admin Chg 2. Washington, D. C.

U.S. Department of Energy. 2014b. *Primer on Spontaneous Heating and Pyrophoricity*. DOE-HDBK-1081-2014. Washington, DC.

U.S. Department of Energy. 2014c. Optimizing Radiation Protection of the Public and the Environment for use with DOE O 458.1, ALARA Requirements. DOE-HDBK-1215-2014. Washington, DC.

U.S. Department of Energy. 2015. *Facility Safety.* DOE O 420.1C, Chg 1. Washington, DC.

U.S. Department of Energy. 2015b. *Environmental Radiological Effluent Monitoring and Environmental Surveillance.* DOE-HDBK-1216-2015. Washington, DC.

U.S. Department of Energy. 2016a. *Comprehensive Emergency Management System.* DOE O 151.1D. Washington, DC.

U.S. Department of Energy. 2016b. *Real Property and Asset Management*. DOE O 430.1C. Washington, D. C.

U.S. Department of Energy. XXXX. *Control and Release of Property with Residual Radioactive Material for use with DOE 5400.5, Radiation Protection of the Public and the Environment.* DOE G 441.1-XX (DRAFT). Washington, D.C.

U.S. Department of Energy National Nuclear Security Administration. 2011. *Guidance on Using Release Fraction and Modern Dosimetric Information Consistently with DOE STD 1027-92, Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports, CN1. NA-1 SD G 1027.* Washington DC.

U.S. Department of Health and Human Services. Agency for Toxic Substance and Disease Registry. 2013. *Toxicological Profile for Uranium*. Atlanta, GA.

U.S. Environmental Protection Agency. 1988. *Guidance for Conducting Remedial Investigations and Feasibility Studies Under CERCLA.* EPA/540/G-89/004, OSWER Directive-9355.3-01. Washington, DC.

U.S. Environmental Protection Agency. 1992. *Manual of Protective Action Guides and Protective Actions for Nuclear Incidents.* EPA 400-4-92-001. Washington, DC.

U.S. Nuclear Regulatory Commission. 1992a. *Radiation Dose to the Embryo/Fetus.* Regulatory Guide 8.36. Washington, DC.

U.S. Nuclear Regulatory Commission, Washington, DC. 1992b.
Title 10 Code of Federal Regulations Part 20, *Standards for Protection Agai nst Radiation*.

U.S. Nuclear Regulatory Commission (NRC). 2002. *Final Generic Environmental Impact Statement (GEIS) on Decommissioning of Nuclear Facilities.* NUREG-0586. Washington, DC.

United States Code. 1970. *National Environmental Policy Act.* 42 USC 4321 et seq.

United States Code. 1980. *The Comprehensive Environmental Response, Compensation and Liability Act (CERCLA) of 1980.* 42 USC 9615 et seq.

Volchok, H. L., and G. dePlanque, eds. 1983. *EML Procedures Manual*. HASL-300. Environmental Measurements Laboratory, New York, NY.

Wrenn, M.E., et al. 1985. Metabolism of Ingested Uranium and Radium. *Health Physics.* 48:601-633.

Wrenn, M.E., Bertelli, L., Durbin, P.W., Singh, N.P., Lipsztein, J.L., and Eckerman, K.F. 1994. *A Comprehensive Metabolic Model for Uranium Metabolism and Dosimetry Based on Human and Animal Data.* Radiation Protection Dosimetry 53(1-4):255-258. (Wrenn et al., 1994)

Wynveen, R. A., H. J. Moe, and M. J. Robinet. 1982. *Waste Identification, Characterization and Disposal During the D&D of a Non-Reactor Nuclear Facility.*

Zankl, M., N. Petrossi, and G. Drexler. 1994. *The Calculation of Dose from External Photon Exposures Using Reference Human Phantoms and Monte Carlo Methods. Part VII: Organ Doses Due to Parallel and Environmental Exposure Geometries.* GSF-Bericht, Gesellschaft fur Strahlen and Umweltforschung GmbH, Munich, Germany.

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.