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RADIOLOGICAL CONTROL ORGANIZATION
TECHNICAL BASIS DOCUMENT

TECHNICAL BASIS FOR

RADIOLOGICAL ACCEPTANCE CRITERIA FOR URANIUM

AT THE Y-12 NATIONAL SECURITY COMPLEX

July 22, 2009

Technical Basis for Radiological Acceptance Criteria for Uranium at the Y-12 National Security Complex

July 22, 2009

Original signatures and dates are on file

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1 Executive Summary

The purpose of this report is to establish radiological acceptance criteria for uranium. Other factors for acceptance not considered include criticality safety concerns, contaminants to the process stream, and impacts to the Safety Basis for the affected facilities.

Three types of criteria were developed in this report. They include limits on external penetrating and non-penetrating radiation and on the internal hazard associated with inhalation of the material. These criteria are intended to alleviate the need for any special controls beyond what are normally utilized for worker protection from uranium hazards. Any proposed exceptions would require case-by-case evaluations to determine cost impacts and feasibility.

Since Y-12 has set rigorous ALARA goals for worker doses, the external limits are based on assumptions of work time involved in the movement of accepted material plus the desire that external doses normally received are not exceeded, and set so that no special personnel monitoring would be required. Internal hazard controls were established so that dose contributions from non-uranium nuclides would not exceed 10% of that expected from the uranium component. This was performed using a Hazard Index (HI) previously established for work in areas contaminated with non-uranium nuclides [1].

The radiological acceptance criteria for uranium are summarized in Table 1. Note that these limits are based on the assumption that radioactive daughter products have reached equilibrium.

Table 1: Acceptance Criteria

Parameter	Acceptance Criteria
Photon exposure, enriched uranium (processing)	$\leq$ 1 mR/hr at one foot from part
Photon exposure, enriched uranium (storage only)	$\leq$ 2 mR/hr at one foot from container
Photon exposure, depleted uranium	$\leq$ 5 mR/hr at one foot from part
$H_p(10)$ (γ+ neutron) (processing)	$\leq$ 1 mrem/hr at one foot from part
$H_p(10)$ (γ+ neutron) (storage only)	$\leq$ 2 mrem/hr at one foot from container
$H_p(0.07)$ enriched uranium	$\leq$ 100 mrem/hr on contact
$H_p(0.07)$ depleted uranium	$\leq$ 250 mrem/hr on contact
Outer container surface contamination (storage only)	As listed in 10CFR835 Appendix D[a]
Internal dose (H^{50}_{Inh})	HI $\leq$ 10%
Volumetric TRU contamination associated with weapons grade Plutonium (mass-based), enriched uranium (Material not to be chemically processed/recovered)	15.6 ppm[b]
Volumetric TRU contamination associated with weapons grade Plutonium (mass-based), enriched uranium (Material to be chemically processed/recovered)	0.8 ppm[c]
Volumetric TRU contamination, depleted uranium	TRU: 25 pCi/g DU[d] (by isotope) TRU: 100 pCi/g DU[d] (total) ^{99}Tc: 750 pCi/g DU[d]

[a] Except for Tritium as listed in DOE Order 5400.5 [2]
[b] Refer to Table 8 of Section 4.1.1 for detailed limits
[c] Refer to Table 9 of Section 4.1.1 for detailed limits
[d] Refer to Section 4.4 for detailed limits

2 Introduction

The primary mission of the Y-12 National Security Complex is the inspection, manufacture, and storage of enriched and depleted uranium in various chemical forms. The Radiological Control Organization (RADCON) is responsible for the protection of workers against radiological hazards. To accomplish this, engineering controls, air monitoring, and internal and external dosimetry programs are used to limit and monitor the exposure of personnel to uranium. Most uranium isotopes are long-lived and decay by the emission of alpha particles and associated low-energy gamma and x-rays. As a result, the radiological hazard of uranium is low when compared to certain other nuclides. Since radiological protection systems at Y-12 have been designed primarily for uranium isotopes, it is essential to limit exposures to other nuclides in terms of its uranium equivalent. Although the majority of dose received by workers at the Y-12 Complex is a result of internal exposures, external dose rates produced by material to be accepted should also be considered.

Historically, the predominant nuclides at Y-12 are ^{234}U, ^{235}U, ^{238}U. ^{236}U activity is typically included in the total uranium activity for routine purposes. Throughout the text, these may be referred to simply as "uranium". Other isotopes of uranium (^{232}U and ^{233}U, for instance) are not routinely encountered and must be treated separately.

3 Internal and External Dosimetry

3.1 Dose Limits

Under 10CFR835, annual dose limits have been established for radiation workers [3]. Additionally, the Department of Energy (DOE) and Y-12 have placed more restrictive administrative control levels (ACLs) on annual radiation dose for workers in an attempt to maintain doses as low as reasonably achievable (ALARA).

The dose quantity used to limit uniform whole-body exposure is the effective dose (E) and is arrived at by coupling individual organ equivalent dose values with ICRP-defined organ weighting factors [4]. The Total Effective Dose (TED) is the sum of the internal Committed Effective Dose (H^{50}_{Inh}) and the external Effective Dose, E. For operational purposes, the personal dose equivalent, $H_p(10)$ (also referred to as "equivalent dose to the whole body" in 10CFR835) is used to approximate E. The 10CFR835-defined quantity "equivalent dose to the skin", which is analogous to the ICRU quantity $H_p(0.07)$, contributes to E since the skin is a defined organ under ICRP-60 guidelines, and is used to limit the dose received by the skin from non-penetrating radiation [3, 4].

Table 2 lists the 10CFR835, DOE, and Y-12 dose levels in place at this time. Note also that

Table 2: 10CFR835, DOE, and Y-12 Dose Levels

Quantity	10CFR835 Limit	DOE ACL	Y-12 ACL	Y-12 ALARA Level
E (TED)	5 rem/year	2 rem/year	1 rem/year	0.75 rem/year
H^{50}_{Inh}	5 rem/year	2 rem/year	1 rem/year[a]	0.75 rem/year
$H_p(10)$	5 rem/year	2 rem/year	1 rem/year[a]	0.75 rem/year[a]
$H_p(0.07)$	50 rem/year	—	—[b]	—[b]

[a] Monitoring is performed if the potential exists for the worker to exceed 100 mrem/year

[b] Note that $H_p(0.07)$ does not have a corresponding ALARA limit associated with it. However, monitoring is required if the dose to any extremity is expected to exceed 5 rem/year

personal dose equivalent is define for various locations and depths on the human body and is suitable for dosimeter calibrations. When field surveys are performed the appropriate quantity is the ambient dose equivalent, $H^*(10)$ [5]. For purposes of this work it is assumed these quantities are equal.

3.2 External Dose Calculations

3.2.1 Photon Dose

Performing external photon dose calculations involves several steps whereby energy dependent mass energy absorption coefficients are folded with the corresponding flux (or fluence) energy densities. In practice, specific gamma-ray constants (Γ) are commonly used to determine the expected dose (or dose equivalent rates) for point-source geometries and are derived for specific isotopes based on their photon energy flux. These assumptions can be important, since the Gamma factors may not be applicable in instances where point-source-geometry assumptions are not valid or when self-absorption is not negligible. Occasionally, these factors are expressed in terms of exposure in air per unit activity, but can be expressed in dose units by applying an additional conversion coefficient. Since the quality factor for photons is typically assumed to be unity over all energy ranges, the absorbed dose (in tissue) is equal to the dose equivalent.

The photon dose rate may be monitored in terms of the exposure rate in air (or air kerma depending on the instrument used) since these quantities have been demonstrated to be reasonably similar [6] for most photon fields encountered at Y-12. Thus, the monitoring result given in exposure units (Roentgen) can be assumed to be reasonably similar to the ambient dose equivalent, $H^*(d)$, for photon energies between about 50 keV and 10 MeV. If a count rate meter is used then an acceptable calibration must be performed as a function of photon energy since if an activity calculation is to be performed based on count rate data an absolute detector efficiency must be known [7].

3.2.2 Neutron Dose

Although neutron sources are not commonly encountered at Y-12, they can be produced in various (α, n) reactions involving alpha emitters (such as uranium) and low-Z target materials such as fluorine and beryllium. Neutron dosimetry is complicated by the energy-dependent quality factor that must be applied to account for the increased relative biological effectiveness of neutrons relative to the reference radiation (usually mid-energy photons). Also, the types of interactions that neutrons undergo are dependent on their incident energy. Fast neutrons transfer their energy primarily through elastic and inelastic scattering interactions, while thermal neutrons may undergo capture reactions which typically result in recoil nuclei. In each case, heavy charged particles are created. In many materials, the neutron absorption cross section decreases as $1/\sqrt{E}$.

To calculate the neutron absorbed dose given an energy spectrum $\phi(E)$:

$$D = \int_{E_{min}}^{E_{max}} \phi(E) \cdot d(E) dE \tag{1}$$

where $d(E)$ is the energy dependent dose conversion coefficient and E_{min} and E_{max} define the energy range considered. To determine the dose equivalent, energy-dependent quality factors are

folded with the neutron spectra for the source and multiplied by the integrated dose. Conversion coefficients and quality factors are available in the literature [8, 9].

A common survey instrument is one having a thermal neutron detector at the center of a 9 inch diameter spherical polyethylene moderator. This instrument, commonly referred to as a Rem-ball, has a response which roughly approximates that of $H^*(10)$ and therefore accounts for the energy dependence of both the absorbed dose and quality factor for neutrons [10].

3.2.3 Charged Particle Dose

Calculating absorbed doses for charged particles is difficult due to the types of interactions that can occur. In the case of uncharged particles, the interactions take place either with the atomic nuclei in the presence of electron fields or with the bound electrons themselves. For charged particles, the additional Coulombic interactions must be considered. As a result of the Coulombic repulsion effect, not all interactions lead to localized energy depositions since Coulombic repulsion effects can effectively take place outside the volume of interest. For the simplest case, to calculate the absorbed dose due to charged particles a number of assumptions must be made each related to the concept of a thin foil through which a parallel broad beam of charged particles pass. First, the collisional stopping power dT/dx is assumed to remain practically constant. The second condition is the charged particles pass straight through the foil and any scatter that occurs is negligible. Since charged particles can produce other charged particles as a result of collisions (delta rays), the third assumption is that either the energy carried by delta rays produced in the thin foil is negligible or that delta ray charged particle equilibrium exists within the foil [11].

The energy lost in collision interactions by a fluence of charged particles of energy T passing through a thin foil of thickness ρt is

$$E = \Phi \left(\frac{dT}{\rho dx} \right)_c \rho t \tag{2}$$

where ρdx is the mass collision stopping power of the foil and ρt is the pathlength through the foil.

Assuming that the third condition described above applies (energy lost by the particles remains in the foil), the absorbed dose is

$$\begin{aligned} D &= \frac{\Phi \left(\frac{dT}{\rho dx} \right)_c \rho t}{\rho t} \\ D &= \Phi \left(\frac{dT}{\rho dx} \right)_c \qquad (\text{MeV gm}^{-1}) \\ D &= 1.602 \cdot 10^{-10} \, \Phi \left(\frac{dT}{\rho dx} \right)_c \qquad (\text{Gy}) \end{aligned} \tag{3}$$

For most practical purposes, charged particle dosimetry is unnecessary since a minimal amount of shielding (including air) effectively reduces the primary dose rate to very low levels. However,

secondary radiation (for example bremsstrahlung) that will be produced when the charged particles are slowed down must be considered [11].

3.3 Internal Dose Calculations

3.3.1 ICRP-66 Respiratory Tract Model

The ICRP-66 [12] based respiratory tract model is currently used at Y-12 to assess internal doses from uranium and will be used for all radionuclides beginning in 2010 with the implementation of the latest revision of 10CFR835. The ICRP-66 report includes a considerably more detailed respiratory tract model than ICRP-30 [13] and includes a revision to the biokinetic model describing a radioisotopes behavior once it is absorbed into the bloodstream. Based on these new models, ICRP Report 54 has been replaced by ICRP Publication 78 [14] which is more appropriate for use with ICRP-66. The ICRP-66 deposition model provides estimates of the regional deposition for a variety of particle sizes. The default particle size used in the newer model is 5 μm compared to the ICRP-30 recommended 1 μm.

ICRP-30 assigned aerosols to one of three clearance classes: D, W, and Y that represented clearance half-times in the deep lung on the order of days, weeks, and years, respectively [13, 15]. ICRP-78 uses a slightly different convention, namely, Type F, M, and S which represent Fast, Moderate, and Slow [12, 14] to describe a material's solubility in the lung. For uranium compounds, ICRP-78 suggests the classification in Table 3 [14].

Table 3: Classification of Uranium Compounds

Type	Compound
F	UF_6, UO_2F_2, $UO_2(NO_3)_2$ and most hexavalent compounds
M	UO_3, UF_4, UCl_4 and other hexavalent compounds
S	UO_2, U_3O_8 and other highly insoluble compounds

The Y-12 National Security Complex obtained approval from DOE to use the ICRP-66 lung model to calculate internal doses from uranium and at the time of this exemption the use of the ICRP-66 model was not approved for assessing doses received from other isotopes and doses arising from intakes of radionuclides other than uranium are calculated using the ICRP-30 lung model. However, since the methods described in ICRP-66 are more technologically advanced, its methods are often used to determine the protection factors described in detail later in this report. This is consistent with the latest revision of 10CFR835 which requires the use of the ICRP-66 internal dosimetry models.

At Y-12, all uranium activity is assumed to be due to ^{234}U for internal dosimetry calculations. Attributing all of the uranium activity to ^{234}U in the dose assessment process, rather than assessing separate doses for ^{234}U, ^{235}U, ^{236}U, and ^{238}U reduces the computational time involved for routine

dose evaluations. In addition, it has a conservative (overestimates the dose by approximately 15% for depleted uranium) and in many cases negligible (overestimates the dose by about 0.5% for 93% enriched uranium)) impact on the end result. For instance, consider a 1000 dpm intake of 100% Type S 5 μm AMAD uranium. If the uranium were assumed to be 93% enriched, the resulting H^{50}_{Inh} would be 11.30 mrem. Assuming it was depleted uranium would change the H^{50}_{Inh} to 9.79 mrem. Using the total uranium assumption (all ^{234}U), the H^{50}_{Inh} is 11.33 mrem [16].

3.4 Dose Conversion Coefficients

To calculate internal and external doses in operational environments, dose conversion factors are commonly used. These factors provide a convenient means to estimate doses from field measurements (particle fluence for example) or from intake estimates. The conversion factors can vary depending on the quantity they are intended to estimate, geometry effects, and calculational assumptions used in their derivation. For this paper, the conversion factors found in ICRP reports 68 and 74 and Federal Guidance Reports (FGR) 11 and 12 will be used [8, 17, 18, 19]. A note on the derivation of values given in these reports is in order. Since internally deposited radionuclides can decay while in the body, radioactive daughter products which themselves are radioactive must also be considered. The conversion coefficients include radioactive daughter products produced after the parent has been introduced into the body. Therefore, daughter products that are present before the intake occurs must be treated separately. For instance, consider the decay

$$^{235}\text{U} \rightarrow {}^{231}\text{Th} \rightarrow {}^{231}\text{Pa}$$

The inhalation DCF for ^{235}U includes the dose from ^{235}U as well as all its daughter products. If pure ^{235}U were inhaled, the DCF could be applied. However, if the inhaled material included ^{235}U and ^{231}Th (regardless of the origin of the ^{231}Th), doses from intakes of both nuclides would have to be assessed separately, unless it can be determined that the dose contribution from the daughter products is negligible. FGR11 utilized the ICRP-30 lung model in its derivation of conversion coefficients. Conversion coefficients found in FGR12 were calculated under guidelines provided in ICRP-26, while those given in ICRP reports 68 and 74 are based on newer recommendations.

3.5 Derived Air Concentration

The DCF values can be used to determine air concentrations that will produce the 5 rem H^{50}_{Inh} limit and these quantities are termed Derived Air Concentrations (DAC). A number of assumptions are necessary and include a breathing rate of 1.2 m^3 hr^{-1}, an 8 hour workday, 5 day workweek, and 50 working weeks per year. Thus, to calculate the DAC for a nuclide from its DCF:

$$\text{DAC} = \frac{5}{2400 \cdot \text{DCF}} \tag{4}$$

where the DCF is given in rem Bq^{-1} and the DAC is in units Bq m^{-3}. This must be qualified by the assumption that the 5 rem H_{Inh}^{50} limit is more restrictive than the non-stochastic limit of 50 rem to any organ or tissue.

4 Hazard Control

4.1 Internal Dose Control

The current administrative control level at Y-12 is 1 rem TED per year. The federal regulations require that internal monitoring be performed for each radionuclide if the potential exists for a person to receive at least 100 mrem H_{Inh}^{50} from that nuclide. To limit personnel exposures, controls are established so that the 1 rem per year limit will not be exceeded under normal operating assumptions. Since internal doses typically account for the majority of the TED received by workers at Y-12, the focus of the radiation safety program is towards limiting internal intakes. Most of the work performed at Y-12, and consequently most of the internal monitoring, is directed towards uranium in various chemical and physical states. For this reason, it is essential to limit exposures to other nuclides in terms of the uranium dose.

To ensure that a worker does not receive greater than 1 rem internal dose, the internal dosimetry program has established monitoring requirements (bioassay sampling, sampling frequencies, etc.) to alert the dosimetrist if a worker approaches a dose threshold. For uranium compounds containing mixtures of nuclides other than uranium, monitoring for these other nuclides is required if a 100 mrem/year internal dose is expected, although the type of monitoring required is dependent on the quantities of non-uranium materials inhaled, bioassay detection limits, and other factors. The monitoring levels and requirements are discussed in detail in References [1] and [16]. If the expected dose from non-uranium nuclides is limited to 10% of the dose received from uranium for a given intake and the uranium dose is limited to 1000 mrem per year, the non-uranium dose will not exceed 100 mrem for that year. Mathematically, this can be represented by

$$100 * \frac{\sum_{i=1}^{n} H_{50,E_i}^{NonU}}{H_{50,E}^{U}} \leq 10\% \tag{5}$$

where $H_{50,E}^{NonU}$ is the H_{Inh}^{50} from non-uranium nuclide i, $H_{50,E}^{U}$ is the H_{Inh}^{50} from uranium, and n is the number of non-uranium nuclides considered [1]. If the activity (or quantity) of each non-uranium nuclide is known, Equation 5 can be used to determine the dose contribution from non-uranium nuclides relative to the dose from uranium by applying the appropriate dose conversion coefficients for each nuclide.

Derived Air Concentrations (DAC) are defined as the air concentration of a radionuclide that would deliver a limiting quantity of dose (either 5 rem H_{Inh}^{50} or 50 rem Committed Dose Equivalent, $H_T(50)$, to an organ or tissue) to a worker in 2000 hours. DAC values are readily available or may be calculated from the dose conversion factors (DCF) found in ICRP-68. For example, the DCF for insoluble (Type S) ^{234}U is given in ICRP-68 as 6.8E-6 Sv Bq^{-1} (0.68 mrem Bq^{-1}). The reference man breathing rate is 20 liters per minute (0.02 m^3 min^{-1}) and it is assumed that the average worker works 50 weeks each year (2000 hours). Thus, the worker breathes 2400 m^3 of air

per year. The stochastic DAC value can be calculated as

$$\begin{aligned} \text{DAC} &= \frac{5000 \text{ mrem}}{(0.68 \text{ mrem Bq}^{-1} * 2400 \text{ m}^3)} * \frac{60 \text{ dpm}}{\text{Bq}} \\ \text{DAC} &\simeq 184 \frac{\text{dpm}}{\text{m}^3} \end{aligned} \tag{6}$$

To relate the dose potential from a mixture containing uranium and non-uranium nuclides, the hazard index (HI) can be determined. The HI is defined as

$$HI = \frac{\sum_{i=1}^{n} \frac{Q_i}{DAC_i}}{\frac{Q_U}{DAC_U}} * 100 \tag{7}$$

where Q_i is the concentration of the i^{th} nuclide, DAC_i is the DAC of the i^{th} nuclide, Q_U is the concentration of uranium, DAC_U is the uranium DAC value, and n is the number of nuclides considered. For convenience, selected coefficients from ICRP-68 [17] are listed in Table 4. Note that these coefficients are in terms of the ICRP-60 defined quantity Committed Effective Dose [4], but that equivalent dose values are not being calculated. The material will be considered to meet radiological acceptance criteria for internal dose purposes if the HI is less than 10%.

As an example, consider the distribution of nuclides listed in Table 5 and assume all nuclides are in an insoluble chemical form and the 5 rem stochastic limit is applicable.[1] Using Equation 7 the Hazard Index can be calculated :

$$\begin{aligned} HI &= \frac{\sum_{i=1}^{n} \frac{Q_i}{DAC_i}}{\frac{Q_U}{DAC_U}} * 100 \\ HI &= \frac{\frac{850 \text{ Bq}}{1.89 \text{ Bq-m}^{-1}} + \frac{950 \text{ Bq}}{2.51 \text{ Bq-m}^{-1}} + \frac{1750 \text{ Bq}}{2.51 \text{ Bq-m}^{-1}}}{\frac{25000 \text{ Bq}}{3.06 \text{ Bq-m}^{-1}}} * 100 \\ HI &= 18.7\% \end{aligned}$$

The interpretation of this example is that an additional 187 mrem of internal dose would be expected if the worker received 1000 mrem of internal dose from uranium and the material would not be acceptable from an internal dosimetry standpoint.

4.1.1 Volumetric Plutonium Contamination

Special interest lies in the ability to ship high enriched uranium (HEU) that is volumetrically contaminated with plutonium to Y-12. This problem has been addressed separately [20], but will

[1] Note that in this case the 5 rem stochastic limit for Pu isotopes may not be applicable, but is used here to show how the HI calculation is performed using the values listed in Table 4

Table 4: ICRP-68 Committed Effective Dose Conversion Coefficients for Internally Deposited Radionuclides [17]. Values Listed are in Sv Bq^{-1}. Type F and M Compounds are Considered Soluble.

Nuclide	Soluble DCF	Insoluble DCF
^{60}Co	1.2E-12	1.2E-12
^{90}Sr	3.0E-08	7.7E-08
^{90}Y	1.6E-09	1.7E-09
^{129}I	5.1E-08	n/a
^{137}Cs	6.7E-09	n/a
^{237}Np	1.5E-05	n/a
^{238}Pu	3.0E-05	1.1E-05
^{239}Pu	3.2E-05	8.3E-06
^{240}Pu	3.2E-05	8.3E-06
^{241}Pu	5.8E-07	8.4E-08
^{242}Pu	3.1E-05	7.7E-06
^{241}Am	2.7E-05	n/a
^{226}Th	7.4E-08	7.8E-08
^{227}Th	6.2E-06	7.6E-06
^{228}Th	2.3E-05	3.2E-05
^{229}Th	6.9E-05	4.8E-05
^{230}Th	2.8E-05	7.2E-06
^{231}Th	3.7E-10	4.0E-10
^{232}Th	2.9E-05	1.2E-05
^{234}Th	5.3E-09	5.8E-09
^{244}Cm	1.7E-05	n/a
^{232}U	4.8E-06	2.6E-05
^{233}U	2.2E-06	6.9E-06
^{234}U	2.1E-06	6.8E-06

Table 5: Example Distribution of Nuclides Contained in an Insoluble Sample.

Nuclide	Activity (Bq)
^{238}Pu	850
^{239}Pu	950
^{240}Pu	1750
^{234}U	25000

be updated here to include the use of ICRP-68 DCF values. Specifically, to perform handling and chemical operations without significant program modifications, certain criteria should be met. The intent of the volumetric limits is similar to those of the hazard index: to ensure that a 1000 mrem H^{50}_{Inh} from uranium material will not result in greater than 100 mrem H^{50}_{Inh} from TRU material. The calculations presented here assume the mass fractions listed in Table 6.

Table 6: Radionuclide Data for 20 Year Old Weapons Grade Plutonium [20]

Nuclide	$t_{1/2}$ (years)	S.A. (Ci/g)	Mass Fraction	Activity Fraction
Np-237	2.14E+06	7.05E-04	0.00007	0.00000
Pu-238	8.64E+01	1.74E+01	0.00004	0.00181
Pu-239	2.44E+04	6.14E-02	0.93028	0.14848
Pu-240	6.58E+03	2.27E-01	0.06197	0.03657
Pu-241	1.32E+01	1.12E+02	0.00267	0.77735
Pu-242	3.79E+05	3.90E-03	0.00005	0.00000
Am-241	4.58E+02	3.24E+00	0.00425	0.03579
TRU mixture	N/A	3.85E-01	1.00[a]	1.00

[a]This mixture contains a small amount (< 0.07%) of uranium isotopes that are excluded from this table

The assumed specific activity of HEU (93% ^{235}U for this work) is 6.51E-5 Ci/g. The specific activity (SA) for the TRU mixture listed in Table 6 is derived using

$$SA = \sum_{i=1}^{n} SA_i * MF_i \tag{8}$$

where SA_i and MF_i are the specific activity and mass fraction of the i^{th} nuclide, respectively and n is the number of nuclides.

The default material type for uranium at the Y-12 complex is 100% Type S, however since the Type M DCF values listed in ICRP-68 are the most restrictive for TRU material, these will be used. Table 7 lists the 5 μm AMAD Type M DCF values given in ICRP-68 [17, 20]. The activity fractions in Tables 6 and 7 were calculated using

$$AF = MF\left(\frac{SA_i}{SA_M}\right) \tag{9}$$

where AF is the activity fraction, SA_i is the nuclide specific activity, and SA_M is the mixture specific activity [20].

The dose per gram of TRU mixture (column 6 of Table 7) is calculated by

$$\frac{D_i}{\text{gram of TRU mixture}} = MF_i * SA_i * DCF_i \tag{10}$$

where DCF_i is the DCF for the i^{th} nuclide and D_i is the dose from the i^{th} nuclide. The nuclide dose fraction per gram of TRU mixture is calculated in a similar manner. The column labeled 'TRU dose trigger' relates how the 100 mrem TRU dose for each nuclide is distributed amongst

Table 7: ICRP-68 Dose Conversion Factors [17] for Weapons Grade Plutonium Listed In Table 6

Nuclide	DCF (Sv/Bq)	S.A. (Ci/g)	Mass Fraction	Activity Fraction	Dose per gram TRU mix.(mrem)	Dose frac. per gram TRU mix.	TRU dose trigger (mrem)
Uranium (Type S)	6.80E-06	6.51E-05	N/A	N/A	N/A	N/A	N/A
Np-237	1.50E-05	7.05E-04	0.00007	1.28E-07	2.74E+03	2.60E-07	0.0
Pu-238	3.00E-05	1.74E+01	0.00004	1.81E-03	7.73E+07	7.34E-03	0.7
Pu-239	3.20E-05	6.14E-02	0.93028	1.48E-01	6.76E+09	6.43E-01	64.3
Pu-240	3.20E-05	2.27E-01	0.06197	3.66E-02	1.67E+09	1.58E-01	15.8
Pu-241	5.80E-07	1.12E+02	0.00267	7.77E-01	6.42E+08	6.10E-02	6.1
Pu-242	3.10E-05	3.90E-03	0.00005	5.07E-07	2.24E+04	2.13E-06	0.0
Am-241	2.70E-05	3.24E+00	0.00425	3.58E-02	1.38E+09	1.31E-01	13.1
TRU mix.	7.39E-06	0.385	1.00	1.00	1.05E+10	1.00	100.0

the mixture. The TRU mixture DCF is calculated by

$$\text{TRU Mix. DCF} = \left[\frac{100 \text{ mrem}}{\sum_{i=1}^{n} \left(\frac{TD_i}{DCF_i} \right)} \right] \tag{11}$$

where TD_i is the nuclide trigger dose and DCF_i is the dose conversion factor for the i^{th} nuclide [20].

Using this information, it can be determined that an intake of 8116 dpm will result in a 100 mrem H_{Inh}^{50} from the TRU mixture. The HEU:TRU mixture activity that will result in a 1000 mrem H_{Inh}^{50} from the HEU is

$$\begin{aligned} \text{HEU:TRU activity} &= \frac{\text{HEU Activity to Deliver 1000 mrem}}{\text{TRU Activity to Deliver 100 mrem}} \\ \text{HEU:TRU activity} &= \frac{88235 \text{ dpm}}{8116 \text{ dpm}} \\ \text{HEU:TRU activity} &= 10.9 \end{aligned} \tag{12}$$

The mass ratio can also be calculated as

$$\begin{aligned} \text{HEU:TRU Mix. Mass Ratio} &= \text{HEU:TRU Activity Ratio} \left(\frac{\text{TRU Mix. Spec. Act.}}{\text{HEU Spec. Act.}} \right) \\ \text{HEU:TRU Mix. Mass Ratio} &= 10.9 \left(\frac{0.385 \text{ Ci/g}}{6.51 * 10^{-5} \text{ Ci/g}} \right) \\ \text{HEU:TRU Mix. Mass Ratio} &= 6.42 * 10^4 \end{aligned} \tag{13}$$

which corresponds to a TRU mixture concentration of 15.6 parts per million [20]. These calculations are summarized in Table 8. Although these limits were derived using 20 year old weapons

Table 8: Summary of TRU Internal Dosimetry Analysis (Type S Uranium)

Mixture	DCF Sv/Bq	HEU:Mixture Activity Ratio for 1000 mrem H^{50}_{Inh} from HEU	HEU:Mixture Mass Ratio for 1000 mrem H^{50}_{Inh} from HEU	Maximum Mixture ppm for no TRU Bioassay Monitoring HEU dose $<$ 1000 mrem
Type S HEU 5 μm AMAD - 1% ^{234}U - 93% ^{235}U - 6% ^{238}U	$6.80 * 10^{-6}$	N/A	N/A	N/A
Type M TRU Mixture (5 μm AMAD)	$7.39 * 10^{-6}$	11	$6.42 * 10^{4}$	15.6

grade plutonium, limits for materials of different age and constituents can be determined in a similar fashion. These limits apply to operations that do not include the potential for concentrating the TRU constituents. For material that will be chemically processed or recovered, the TRU contaminants can be concentrated in the waste streams. Typical concentration factors based on past processing experience have been shown [20] to be about 30. Other considerations must be taken into account, including the lower dose potential for routine chemical operations at the Y-12 site. Past experience shows that typical chemical operators receive at the most about 200 mrem per year from routine chronic exposure to uranium. The scenario of a chemical operator receiving the 1000 mrem limit used previously is unlikely unless a significant incident occurred. Thus, the limits derived in Table 8 must be revised if the material is to be chemically processed or if recovery operations are to be performed. Using the ICRP-68 DCF for Type M uranium of 2.1E-6 Sv/Bq the revised limits, shown in Table 9, are calculated based on a uranium H^{50}_{Inh} of 200 mrem and a TRU H^{50}_{Inh} limit of 100 mrem.

4.2 External Dose Control

The external dose from a nuclide is dependent on the energies and types of radiation emitted. For low-penetrating radiations (α and β), the dose rate will decrease significantly as a function of distance due to attenuation in air. However, for more strongly penetrating (γ and neutron) radiation, this attenuation will be negligible in most cases over short distances. Two quantities must be considered for external dosimetry purposes: the personal dose equivalent, $H_p(10)$, also referred to as equivalent dose to the whole body in 10CFR835 and the equivalent dose to the skin, $H_p(0.07)$. Personal dose equivalent, $H_p(10)$ results primarily from penetrating radiations (since it is measured at a depth of 1 cm), while $H_p(0.07)$ is also influenced by weakly-penetrating emissions.

Table 9: Summary of TRU Internal Dosimetry Analysis (Type M Uranium). This Table applies to material that is to be chemically proceeded or if recovery operations will be performed.

Mixture	DCF Sv/Bq	HEU:Mixture Activity Ratio for 200 mrem H^{50}_{Inh} from HEU	HEU:Mixture Mass Ratio for 200 mrem H^{50}_{Inh} from HEU	Maximum Mixture ppm for no TRU Bioassay Monitoring HEU $H^{50}_{\text{Inh}} < 200$ mrem
Type M HEU 5 μm AMAD - 1% ^{234}U - 93% ^{235}U - 6% ^{238}U	2.10 E-6	N/A	N/A	N/A
Type M TRU Mixture (5 μm AMAD)	7.39E-6	7	4.16E+4	0.8[a]

[a]This includes a concentration factor of 30

4.2.1 Photon Dose Equivalent

Enriched uranium emits alpha, beta, and low-energy photons (γ and x-rays) produced in various abundance. The most abundant photon emitted from ^{235}U is the 0.1857 MeV gamma emitted in 54% of the transformations and the bulk of the $H_p(10)$ produced by ^{235}U is a result of this photon. The daughter products of ^{238}U emit photons of various energies, but the specific activity of ^{238}U is much lower than ^{235}U because of its longer half life. In general, the primary hazard associated with uranium is the internal dose since the dose rate produced by these emissions can be relatively low compared to other nuclides. Nuclides with short half lives (and corresponding high specific activities) that emit higher energy photons, will produce significantly higher absorbed doses. Most of the radiation shielding at Y-12 has been designed to protect workers from the low-energy photon emissions from uranium and if higher energy emissions are introduced, additional shielding or protective measures may be necessary. One such example is irradiated reactor fuel rods since fission products produced in the fuel emit energetic photons and have much shorter half-lives. To reduce the exposure of personnel to external photons, the acceptance criteria limits the exposure rate from material to less than 2 mR/hour at one foot when the material reaches equilibrium with its daughters if the material is for storage only. If the material is to be processed, the dose rate shall be less than 1 mR/hour at one foot.

4.2.2 Neutron Dose Equivalent

Neutron production can occur when alpha particle emitters are combined with low-Z materials including beryllium, fluorine, aluminum. Since these materials are commonly used with uranium,

doses arising from these neutrons must be considered. As discussed earlier, the calculation of dose equivalent, $H_p(10)$, is complicated by the energy dependence of the quality factor. For this reason the neutron spectrum must be known if the dose equivalent from a neutron source is to be accurately calculated. If the chemical composition and enrichment of uranium is known, the resulting emission spectrum can be easily calculated [21] although spectral degradation (thermalization) should be considered. Other measurement techniques can be used if an approximation of the neutron dose equivalent is desired. These include Bonner sphere measurements and so-called 'rem meters' [22]. The rem meter's (typically a thermal neutron detector scintillator surrounded by an 8, 9, or 10 inch polyethylene sphere) energy response roughly follows that of the dose equivalent delivered per neutron as a function of energy. Thus, readings will include the appropriate quality factor resulting in a meaningful measure of the overall dose equivalent produced by the neutron spectrum. The total dose equivalent rate (neutron + photon) from material accepted to Y-12 will not exceed 2 mrem/hr at one foot (storage only) or 1 mrem/hr at one foot (for material that is to be processed). It should be noted that beryllium (or other low-Z materials) content in the material can strongly influence the neutron generation. The acceptable Be (or other low-Z materials) content may be limited by other (industrial hygiene or criticality safety) criteria.

4.2.3 Shallow Dose Equivalent

The shallow dose equivalent is measured at a depth of 0.07 mm. As a result, non-penetrating radiations that do not contribute significantly to $H_p(10)$ can profoundly affect $H_p(0.07)$ and must be considered. These non-penetrating radiations include low-energy photon and β^- radiations. To properly estimate the $H_p(0.07)$ produced by a nuclide, daughter products should be considered. In the case of uranium these daughter products include ^{231}Th resulting from the alpha decay of ^{235}U and both ^{234}Th and ^{234}Pa produced in the decay of ^{238}U. ^{234}U daughter products were not included since they will not become significant until at least 5 half lives of ^{230}Th have passed (370,000 years) and near secular equilibrium.

As a rule, 93% enriched uranium does not present a serious shallow dose equivalent hazard, but the presence of ^{238}U in any appreciable quantity can introduce a significant beta-dose contribution. To prevent special monitoring requirements (e.g. extremity dosimetry), the shallow dose equivalent produced by enriched forms of uranium shall be less than 100 mrad/hr at contact. Since the expected shallow dose rate of depleted uranium materials will be higher, the shallow dose equivalent from depleted uranium will be less than 250 mrad/hr at contact.

4.2.4 ^{233}U Considerations

Recall that for purposes of this report, ^{233}U is not included in the generic "uranium", but must be treated separately. In practice, ^{233}U is usually accompanied by a significant amount of ^{232}U. ^{208}Tl is a member of the ^{232}U decay chain and emits a 2.61 MeV photon which is much more penetrating than the weaker x-rays and low-energy photons emitted by other uranium isotopes. The shielding considerations for these strongly penetrating photons are much more strenuous than

for the low-energy photons commonly encountered at Y-12. Therefore, particular attention should be paid to the U-233 and/or U-232 content.

4.3 Acceptance Criteria for Storage Only Material

For material that is to be accepted for storage only, the requirements should be based on the hazard associated with the storage and not the contents. External dose rates should be limited to the criteria given earlier with all daughter products in secular equilibrium.

If contamination is present on the outside of the container, the appropriate limits as listed in Appendix D of 10CFR835 should be met. These limits are listed in Table 10.

Table 10: 10CFR835 Contamination Limits [3]. Limits are listed in units of dpm/100 cm^2

Radionuclide	Removable Limit	Total (Fixed + Removable) Limit
Natural U, ^{235}U, ^{238}U, and associated decay products	1000 (Alpha)	5000 (Alpha)
Transuranics, ^{226}Ra, ^{228}Ra, ^{230}Th, ^{228}Th, ^{231}Pa, ^{227}Ac, ^{125}I, ^{129}I	20	500
Natural Th, ^{232}Th, ^{90}Sr, ^{223}Ra, ^{224}Ra, ^{232}U, ^{126}I, ^{131}I, ^{133}I	200	1000
Beta-Gamma emitters except ^{90}Sr and others noted above	1000	5000[a]
Tritium and tritiated compounds	10000	N/A

[a]May be difficult to discriminate fixed contamination from emissions from material in container

4.4 Depleted Uranium Acceptance Criteria

The current acceptance criteria established specifically for depleted uranium are based on the guidelines contained in the procurement specification [23]. This specification states the following:

> Prior to packaging for shipment, the subcontractor shall verify that all derby material in a shipping lot to Y-12 meets the required limits established for any of the transuranic constituents as specified.
>
> 1. The maximum concentration of Np-237, Pu-238, Pu-239/240, and Am-241 is less than or equal to 25 pCi per gram of depleted uranium (each) or the sum for all transuranic constituents is less than or equal to 100 pCi per gram of depleted uranium.

2. The maximum value for Tc-99 is less than or equal to 750 pCi per gram of depleted uranium.
3. If the isotopic concentration of Np-237, Pu-238, Pu-239/240, Am-241, and Tc-99 exceed the limits in 1 and 2, the contractor will perform a dose analysis based on these concentrations. The derby material shipment may be acceptable if the Company's calculation for dose analysis provides evidence that the dose contribution of the transuranic and fission products is less than 100 mrem.

A study of the actual concentration levels of representative accepted material was conducted [24, 25] in 2000 by staff at Y-12. This study determined that an acceptance criteria of 0.17 ppm volumetric TRU contamination (approximately 1366 pCi TRU/g depleted uranium) would limit internal dose potential from TRU contaminants to 100 mrem for a 2000 mrem intake of depleted uranium. These calculations were performed using the ICRP-68 DCF for uranium and ICRP-30 (Class W) DCFs for the TRU mixture. Table 11 lists the average composition of the DU samples that were used for these calculations. Calculations similar to those performed in Section 4.1.1 were

Table 11: Average Depleted Uranium TRU Contaminants [24]

Isotope	TRU mix mass fraction	Isotope S.A. (Ci/g)	Isotope S.A. (Bq/g)	TRU Mix S.A. (Bq/g)
Np-237	0.9500	7.05E-04	2.61E+07	2.48E+07
Pu-238	0.0001	1.74E+01	6.44E+11	3.80E+07
Pu-239	0.0488	6.14E-02	2.27E+09	1.11E+08
Pu-240	0.0000	2.27E-01	8.40E+09	0.00E+00
Pu-241	0.0000	1.12E+02	4.14E+12	0.00E+00
Pu-242	0.0000	3.90E-03	1.44E+08	0.00E+00
Am-241	0.0010	3.24E+00	1.20E+11	1.15E+08
TRU Mix	1.0000			2.89E+08

applied to the distribution listed in Table 11. Since no chemical recovery or processing operations are performed with depleted uranium at Y-12, all constituents were considered to be insoluble (Type S), except Np-237 and Am-241 which were assumed to be Type M. Again, the intent of the depleted uranium limit was to limit the TRU H_{Inh}^{50} to 100 mrem assuming a 1000 mrem H_{Inh}^{50} intake of uranium. Using the information provided in Table 11, a DCF for the TRU constituents of 4.82E+3 Sv/g was derived (e.g. intake of 2.07E-7 grams would result in 100 mrem H_{Inh}^{50}). Using a DCF for Type S uranium of 6.8E-6 Sv/Bq and a specific activity of 1.76E+4 Bq/g, the quantity of uranium that would result in 1000 mrem H_{Inh}^{50} is 8.35E-2 g. Therefore, the TRU concentration that would result in 100 mrem TRU H_{Inh}^{50} assuming a 1000 mrem H_{Inh}^{50} from uranium is 2.5 ppm (assuming the concentrations listed in Table 11). The average TRU activity in the samples was 19.34 pCi/g DU or

$$\text{TRU concentration} = \left(\frac{19.34 \text{ pCi TRU}}{\text{g DU}}\right)\left(\frac{0.037 \text{ Bq}}{\text{pCi}}\right)\left(\frac{1 \text{ g}}{2.89 * 10^8 \text{ Bq}}\right)$$

$$\text{TRU concentration} = \frac{2.48 * 10^{-9} \text{ g TRU}}{\text{g DU}}$$

$$\text{ppm TRU} = 2.48 * 10^{-3}$$

which is approximately 1000 times less than the 2.5 ppm concentration that would deliver a 100 mrem TRU H^{50}_{Inh} given an intake that resulted in 1000 mrem H^{50}_{Inh} from the uranium, or about

$$\begin{aligned} \text{SA} &= \left(\frac{2.5 \text{ ppm}}{2.5 * 10^{-3} \text{ ppm}}\right)(19.34 \text{ pCi/g}) \\ \text{SA} &= 20000 \text{ pCi/g} \end{aligned}$$

which is much higher than that allowed under the procurement specification.

Dose rates from depleted uranium material received from various vendors are typically less than 4 mR/hr at one foot. Therefore, the acceptance criteria for depleted uranium is 5 mR/hr.

5 Acceptance Criteria

5.1 Proposed Limits

The radiological acceptance criteria for uranium as derived in this work are summarized in Table 12.

Table 12: Acceptance Criteria

Parameter	Acceptance Criteria
Photon exposure, enriched uranium (processing)	$\leq$ 1 mR/hr at one foot from part
Photon exposure, enriched uranium (storage only)	$\leq$ 2 mR/hr at one foot from container
Photon exposure, depleted uranium	$\leq$ 5 mR/hr at one foot from part
$H_p(10)$ (γ+ neutron) (processing)	$\leq$ 1 mrem/hr at one foot from part
$H_p(10)$ (γ+ neutron) (storage only)	$\leq$ 2 mrem/hr at one foot from container
$H_p(0.07)$ enriched uranium	$\leq$ 100 mrem/hr on contact
$H_p(0.07)$ depleted uranium	$\leq$ 250 mrem/hr on contact
Outer container surface contamination (storage only)	As listed in 10CFR835 Appendix D[a]
Internal dose (H^{50}_{Inh})	HI $\leq$ 10%
Volumetric TRU contamination associated with weapons grade Plutonium (mass-based), enriched uranium (Material not to be chemically processed/recovered)	15.6 ppm[b]
Volumetric TRU contamination associated with weapons grade Plutonium (mass-based), enriched uranium (Material to be chemically processed/recovered)	0.8 ppm[c]
Volumetric TRU contamination, depleted uranium	TRU: 25 pCi/g DU[d] (by isotope) TRU: 100 pCi/g DU[d] (total) ^{99}Tc: 750 pCi/g DU[d]

[a] Except for Tritium as listed in DOE Order 5400.5 [2]
[b] Refer to Table 8 of Section 4.1.1 for detailed limits
[c] Refer to Table 9 of Section 4.1.1 for detailed limits
[d] Refer to Section 4.4 for detailed limits

5.2 Past Limits

A number of acceptance criteria have been used at Y-12 since 1958. In order to provide a means of comparison between the limits defined in this report with those previously issued, Table 13 lists the past limits and provides a brief description of each.

Table 13: Review of Past Acceptance Criteria by Year

Reference	Year	Parameter	Acceptance Criteria
[26]	1958	Pu contamination	1 dpm Pu per 700 dpm total (Pu+U)
[27]	1960	Pu contamination	1 dpm Pu per 700 dpm total (Pu+U)
[28]	1960	Pu contamination	1 dpm Pu per 700 dpm total (Pu+U)
[28]	1960	Pu contamination	1.5 dpm/m^3 Pu
[29]	1965	α Activity from TRU	1 dpm TRU per 700 dpm (TRU+U)
[29]	1965	Gross γ activity	3.5 μg Radium equivalent/g U
[29]	1965	α Activity from all U isotopes	170 dpm/μg
[29]	1965	^{232}U Activity	1 dpm ^{232}U per 300 dpm total U
[30]	1975	α Activity Non-U actinides	0.10 μCi/g U
[30]	1975	α Activity from actinides	1 dpm TRU per 700 dpm (non-U+U)
[30]	1975	β^- Activity from actinides	2/(fraction of ^{238}U * activity from unirradiated uranium of similar enrichment)
[30]	1975	γ Activity from fission products	0.2 μCi/g U
[30]	1975	Total γ activity	2.0 μg Radium equivalent/g U
[30]	1975	^{232}U Content	0.03 ppm ^{232}U/g U
[31]	1985	TRU α activity	0.1% of U α activity
[31]	1985	β^- Activity	$\frac{\beta^-\text{ activ. recycle material}}{\beta^-\text{ activ. 93\% enriched U}} \leq 1.25$
[31]	1985	Total γ activity (U compound)	1.2 μCi/g U
[31]	1985	Ce γ activity (U compound)	0.3 μCi/g U
[31]	1985	Ru γ activity (U compound)	0.3 μCi/g U
[31]	1985	Cs γ activity (U compound)	0.1 μCi/g U
[31]	1985	Zr-Ni γ activity (U compound)	0.5 μCi/g U
[31]	1985	Any other nuclide γ activity (U compound)	0.1 μCi/g U
[31]	1985	Total γ activity (U metal)	0.3 μCi/g U
[31]	1985	Ce γ activity (U metal)	0.05 μCi/g U
[31]	1985	Ru γ activity (U metal)	0.05 μCi/g U
[31]	1985	Cs γ activity (U metal)	0.05 μCi/g U
[31]	1985	Zr-Ni γ activity (U metal)	0.1 μCi/g U
[31]	1985	Any other nuclide γ activity (U metal)	0.05 μCi/g U
[32]	1997	α Activity (TRU, ^{232}U,^{233}U)	7.0% of nominal U
[32]	1997	β^- Activity	1.25 times β^- activity of 93% U
[32]	1997	γ Activity (fission prod.)	1.2 μCi/g total U in secular equil.
[32]	1997	γ Activity (isotope)	By isotope as listed in Ref. [31]
[32]	1997	(α,n) Hazard	1% by weight Be
[20]	2000	Pu contamination (2 rem CEDE)	2.1 ppm
[20]	2000	Pu contamination (200 mrem CEDE)	0.2 ppm

5.3 Comparison of Proposed Limits with Past Limits

To provide some means of comparing the new limits with those used previously, the limits listed above were recalculated in terms of the new limits. The results of the comparison are given in Table 14 for internal limits and Table 15 for external limits. In each case, a number of assumptions were necessary and are listed below.

1. 1 dpm Pu per 700 dpm total (Pu+U) (Pu contamination): The HI was calculated assuming 100% Type S uranium and 100% Type S ^{239}Pu
2. 1.5 dpm/m^3 (Pu contamination): The air concentration was used in conjunction with the 100% Type S DCF listed in ICRP-68 and the reference man breathing rate and assumed 2000 working hours per year
3. 1 dpm TRU per 700 dpm (TRU+U) (α activity): Used the mixture listed in Table 6 and assumed 100% Type S for all nuclides and isotopes except ^{237}Np (Type M)
4. 3.5 μg Radium equivalent/g U (gross γ activity): Used a total gamma emission rate of 334750 γ/g enriched uranium
5. 170 dpm/μg (α Activity from all U isotopes): Used Y-12 specific activity of enriched uranium (143.21 dpm/μg) to determine increase in dose conversion coefficient
6. 1 dpm ^{232}U per 300 dpm total U (^{232}U activity): Used Γ factors given in reference [33] (including Tl-208)
7. 0.2 μCi/g U (γ Activity from fission products): Assumed 0.2 μCi ^{137}Cs per gram of ^{234}U
8. 2.0 μg Radium equivalent/g U (Total γ activity): Used a total gamma emission rate of 334750 γ/g enriched uranium
9. 0.03 ppm ^{232}U/g U (^{232}U Content): Used Γ factors from reference [33] (including Tl-208)
10. TRU α activity = 0.1% of U α activity (TRU α activity): Assumed 1 kBq 234U α activity and 1 Bq ^{239}Pu α activity
11. 1.2 μCi/g U (total γ activity): Assumed isotopic distribution listed in reference [31] and Γ factors given in reference [33]
12. TRU α activity 7.0% of nominal U α Activity (TRU, ^{232}U,^{233}U α activity): Assumed 100 Bq ^{234}U α activity and 7 Bq ^{239}Pu α activity
13. 1.2 μCi/g total U in secular equil. (fission product γ Activity): Attributed 1.2 μCi to ^{137}Cs and used Γ factors from reference [33]

Although the plutonium contamination limit of 1 dpm Pu per 700 dpm total (Pu+U) appears much more restrictive than the hazard index of 10% currently proposed, a number of qualifications

Table 14: Comparison of Proposed Internal Limits with Past Acceptance Criteria by Year

Year	Parameter	Conversion of Old Criteria to Equivalent New
1958	Pu contamination	HI = 0.17% (Type S ^{239}Pu)
1960	Pu contamination	1250 mrem (^{239}Pu, ref. man assumptions)
1965	α Activity from TRU	HI = 0.04% (mix listed in Table 6 for Type S)
1965	α Activity from all U isotopes	18% increase in H_{50}^{inh} (S.A.=143.21 dpm/μg)
1985	TRU α activity	HI = 0.1% (Type S ^{239}Pu)
1997	α Activity (TRU, ^{232}U,^{233}U)	HI = 8.5% (Type S ^{239}Pu)
2000	Pu contamination (2 rem CEDE)	156 ppm (1 rem H_{50}^{inh})
2000	Pu contamination (200 mrem CEDE)	15.6 ppm (100 mrem H_{50}^{inh})

Table 15: Comparison of Proposed External Limits with Past Acceptance Criteria by Year

Year	Parameter	Conversion of Old Criteria to Equivalent New
1965	Gross γ activity	$\sim$ 39% increase in γ activity
1965	^{232}U Activity	$\sim$ 7.7% increase in γ dose rate
1975	γ Activity from fission products	$\sim$ 1.5% increase in γ dose rate (for ^{137}Cs)
1975	Total γ activity	$\sim$ 22% increase in γ activity
1975	^{232}U Content	$\sim$ 122% increase in γ dose rate
1985	Total γ activity	$\sim$ 6.5% increase in γ dose rate
1997	γ Activity (fission prod.)	$\sim$ 9.2% increase in γ dose rate (for ^{137}Cs)
1997	(α,n) Hazard	Unchanged

must be addressed. The HI has been defined so that a 1000 mrem H_{50}^{inh} from uranium will not result in 100 mrem additional H_{50}^{inh} from non-uranium nuclides using ICRP-68 dose conversion coefficients. The HI value for this case (0.17%) was calculated assuming both the uranium and ^{239}Pu were Type S material using the DCF values given in ICRP-68 [17]. Under ICRP-30 guidelines [13], the DCF for Class W uranium (^{234}U) is 2.13E-6 Sv/Bq compared to 2.10E-6 Sv/Bq under ICRP-68 (Type M). The DCF for Class W ^{239}Pu is 1.16E-4 Sv/Bq in ICRP-30 and 3.20E-5 Sv/Bq in ICRP-68 (Type M). For Type M material (ICRP-68 DCF values), the HI becomes 2.18%. Using the ICRP-30 DCF values, the HI is 7.79. Each of these HI values assume the uranium limit is 1000 mrem (a 1 rem uranium CED will result in 77.9 mrem from ^{239}Pu). This 1000 mrem value is based on the Administrative Control Level (ACL) currently in place at Y-12. Previous values have been as high as 5 rem. If the 5 rem uranium H_{50}^{inh} limit is applied, the relative HI is 7.8%, however the H_{50}^{inh} from ^{239}Pu would become 390 mrem. Of course, these limits and DCF values from ICRP-30 were established in the late 1960's, with implementation at operational facilities coming later in most cases. The concept of ALARA is also fairly recent, so historically the 5 rem limit was applied more frequently. Under the Handbook 52 limits of 15 rem, the CEDE received from the non-uranium radionuclides would be even higher. This example does, however, show the difficulty in making a straight comparison between historical limits and those proposed here. To further elaborate, the internal dose consequences of the proposed limits are compared with those defined previously in Table 16. The column labeled 'ICRP-68' assumes the material is 100% Type

S, 5 μm AMAD, and the uranium H_{50}^{inh} limit is 1000 mrem while the column labeled 'ICRP-30' uses 100% Class W DCF values, 1 μm AMAD, and assumed the CEDE limit for intakes of uranium is 5000 mrem. The values contained in column two are the ICRP-60 H_{50}^{inh} that would be received from the non-uranium nuclides if the uranium limit were reached while column three lists the ICRP-30 CEDE values. The proposed limits for TRU contamination in depleted uranium samples have corresponding HI values listed in Table 17.

Table 16: Comparison of Internal Limits and ICRP-68 and ICRP-30 Models

Criteria	ICRP-68 1000 mrem U 100% Type S	ICRP-30 5000 mrem U 100% Class W
1 dpm ^{239}Pu per 700 dpm total (Pu+U)	1.75 mrem	390 mrem
0.10 μCi ^{239}Pu/g U	1.89 mrem	423 mrem
^{239}Pu α activity is 0.1% of U α activity	1.22 mrem	272 mrem
^{239}Pu α activity is 7% of U α activity	85 mrem	19060 mrem
Pu contamination	100 mrem	250 mrem
TRU contamination (depleted)	0.63 mrem	80 mrem
^{99}Tc contamination	0.0 mrem	0.0 mrem

Table 17: Hazard Indices for Depleted Uranium Contaminated with TRU

Criteria	ICRP-68 100% Type S	ICRP-30 100% Class W
100 pCi TRU/g DU	0.06%	1.6%
25 pCi ^{239}Pu/g DU	0.01%	0.4%
750 pCi ^{99}Tc/g DU	0.0%	0.0%

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