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Recommended Surface Contamination Levels for Waste Packages Prior to Placement in the Repository

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DISCLAIMER

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CHANGE HISTORY

Revision Number	Interim Change No.	Effective Date	Description of Change
000	N/A		Initial issue

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EXECUTIVE SUMMARY

Radioactive contamination levels were derived for acceptance of waste packages into the subsurface repository. The derivation of these contamination levels on the external surface of waste packages was based on limiting the annual average concentrations of airborne effluents at the subsurface exhaust to the airborne effluent concentration limits specified in 10 CFR Part 20 Appendix B. The derived radioactive contamination levels should be compared to radiological survey data to demonstrate compliance with effluent concentration limits. The results of the analysis indicates that the effluent concentration limits would not be exceeded at the subsurface ventilation exhausts during the pre-closure period of the repository if the contamination levels presented in this report are not exceeded. The derived contamination levels are presented in Table 3: "Derived Contamination Levels". A recommended conceptual survey process that could be utilized to demonstrate compliance is provided.

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ACRONYMS AND ABBREVIATIONS

ALARA	as low as is reasonably achievable
BWR	boiling water reactor
ECL	effluent concentration limit
NRC	U.S. Nuclear Regulatory Commission
PWR	pressurized water reactor
TEDE	total effective dose equivalent
WP	waste package

UNITS OF MEASURE

Ci	curie
cm	centimeter
dpm	disintegration's per minute
mrem	millirem
MTU	metric tons uranium
rem	roentgen equivalent man
s	second
yr	year

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1. PURPOSE

The purpose of this report is to provide a technical basis and justification for radioactive contamination levels on the external surfaces of waste packages before entering the repository for emplacement. This report also provides an approach for surveying waste packages for demonstrating compliance during the pre-closure period of the repository. The values represent the average contamination levels of radioactivity on the external surfaces of waste package before the waste package is to be transported to the subsurface facility for final emplacement.

2. QUALITY ASSURANCE

This technical report was prepared in accordance with AP-3.11Q, *Technical Reports*. Because the contamination levels derived from this report will be used to control potential radioactive contamination at the repository facilities, this document is directly related to the radiological safety of the repository and is subject to the requirements of *Quality Assurance Requirements and Description* (DOE 2003, Section 2.2.2).

3. METHOD

The method used to derive the contamination levels on the external surface of waste packages is expanded from that described in *Subsurface Contamination Control* (Yuan, Y. 2001, Section 3.1). The derived values represent the average concentration of contamination on the external surfaces of each waste package before the waste package is transported to the subsurface facility for emplacement. The derivation was based on limiting the annual average concentrations of airborne effluents at the subsurface exhaust to the airborne effluent concentration limit (ECL) specified in Table 2 of Appendix B to 10 CFR Part 20. As stated in 10 CFR 20.1302(b)(2)(i) the annual average concentrations of radioactive material released in gaseous effluents at the boundary of the unrestricted area cannot exceed the ECL values (see Section 4.4.2). The ECL values given in Table 2 of Appendix B to 10 CFR Part 20 are equivalent to the radionuclide concentrations that, if inhaled continuously over the course of a year, would produce a total effective dose equivalent of 50 mrem. In addition, the effluents released at the exhaust shaft will be below the levels stated in 10 CFR 20 Table 2 of Appendix B. This ensures that the dose to an off-site member of the general public is ALARA and 1) meets the annual dose limit of 10 mrem from airborne effluents in 10 CFR 1101 (d) and 2) meets the annual dose limit of 15 mrem in 10 CFR 63.204 from waste packages emplaced in the subsurface.

3.1 EQUATIONS

The contamination activity in the repository drifts due to contamination on waste package surfaces can be calculated by using the mass balance approach:

{rate of contamination accumulation} = {rate of contamination in} – {rate of contamination out}

The mass balance equation for the total waste package surface contamination in the repository with respect to time resulting from emplacement operation is:

$$\frac{dS}{dt} = MC_sA - kS \qquad \text{(Eq. 1)}$$

The various contributions to the mass balance are:

Time rate of change of contamination activity in repository drifts: $\frac{dS}{dt}$

Rate of waste package contamination entering into repository: MC_sA

Rate of contamination resuspended and released from repository drifts: kS

where:

S = time dependent total contamination on the waste packages in the repository (Ci)
A = surface area of the average waste package, (m^2)
M = rate of waste package emplacement or number of waste packages emplaced per year, (1/yr)
C_s = The average contamination for a specific radionuclide on the surface of waste package during emplacement, (Ci/m^2)
$k = \lambda + \omega$ (1/yr)
λ = radioactive decay constant (1/yr)
ω = re-suspension rate of the surface contamination (1/yr)

The solution for equation (1) at time t following the start of emplacement operation at t = 0 is:

$$S = \frac{MC_sA}{k}(1 - e^{-kt})$$

Since the second derivative $\frac{d^2S}{dt^2} = -k$ is less than zero, the maximum value of S can be obtained by setting $\frac{dS}{dt} = 0$ in Eq. 1 and therefore:

$$kS = M\,C_s\,A \qquad \text{(Eq. 2)}$$

Equation 2 implies that at some time in the future after startup of the emplacement operations, the total amount of contamination present in the subsurface will reach and remain at a constant maximum level (i.e. the amount of contamination entering is equal to the amount of contamination exiting).

Assuming radioactive decay is negligible (Assumption 4.2.4), the maximum contamination resuspension rate, R, may be calculated by:

$$R = \omega C_R = M C_s A \quad \text{(Eq. 3)}$$

At the ventilation exhaust shaft, the average effluent concentration is:

$$C_a = R/V = (M C_s A)/V \quad \text{(Eq. 4)}$$

where V (m^3/yr) is the ventilation airflow rate of the exhaust shaft.

Setting the annual average concentration, C_a, to the annual ECL, making C_s equal to the contamination level, or G, yields the following equation:

$$G = (ECL \times V) / (A \times M) \quad \text{(Eq. 5)}$$

where G represents the annual average contamination level for an emplacement rate of M number of waste packages per year.

Equation 5 is derived for calculating a single-radionuclide contamination level. The calculation of the contamination level or G for radionuclides of radiological importance in the repository is presented in Appendix A.

3.2 IMPLEMENTATION OF THE RADIOACTIVE CONTAMINATION LEVELS FOR RADIATION SURVEYS

When applying the derived single-radionuclide contamination level for surveying a loaded waste package prior to emplacement, the sum-of-fractions rule may be applied (10 CFR 20 Appendix B, footnote four). That is, the annual average summation of the fractional contamination level of radionuclides remaining on the emplaced waste packages should not be greater than unity, or:

$$\bar{F} = \frac{1}{N}\sum_{i=1}^{N} F_i \leq 1 \quad \text{(Eq. 6)}$$

or

$$\sum_{i=1}^{N} F_i \leq N \quad \text{(Eq. 6a)}$$

where

$$F_i = \sum_j \frac{C_{i,j}}{G_j} \quad \text{(Eq. 7)}$$

$\bar{F}$ is the annual average summation of the fractional contamination level of radionuclides remaining on the waste packages for the year; F_i is the summation of the fractional contamination level remaining on the i-th waste package; $C_{i,j}$ is the average concentration of the j-th nuclide on the surface of the i-th waste package; and G_j is the single-radionuclide contamination level derived for the j-th nuclide. N is the number of waste packages emplaced over a period of up to 90 days (quarterly) with a constraint that the number of waste packages emplaced over any continuous 365

days, is less than M (600 waste packages/yr, Assumption 4.2.2), the emplacement rate used to derive the contamination level in Equation 5.

Effluent monitoring will be performed to demonstrate that effluent concentration limits are not exceeded during the pre-closure period of the Monitored Geological Repository. The monitoring is also consistent with assuring the ALARA requirements for constraints on dose due to effluent releases is achieved.

Unless distribution of nuclides is determined, the contamination on the waste packages would consist of unknown mixtures of radionuclides. To simplify the waste package contamination survey process and for conservatism, initial radiation measurements are performed only for gross α and β/γ activities. Equation 6 can be written as:

$$\bar{F}(\alpha\beta) = \frac{1}{N}\sum_{i=1}^{N} F_i(\alpha\beta) \leq 1 \qquad \text{(Eq. 8)}$$

or

$$\sum_{i=1}^{N} F_i(\alpha\beta) \leq N \qquad \text{(Eq. 8a)}$$

where

$$F_i(\alpha\beta) = F_i(\alpha) + F_i(\beta) \qquad \text{(Eq. 9)}$$

where $\bar{F}(\alpha\beta)$ is the annual average summation of the fractional contamination level of gross α and β/γ activities remaining on the waste packages for the year under consideration and $F_i(\alpha\beta)$ is the summation of the fractional contamination level of gross α and β/γ activities remaining on the i-th waste package. $F_i(\alpha)$ and $F_i(\beta)$, respectively, are the individual fractional contamination level of gross α and gross β/γ activities remaining on the i-th waste package and are calculated by:

$$F_i(\alpha) = \frac{C_i(\alpha)}{G_\alpha} \qquad \text{(Eq. 10)}$$

$$F_i(\beta) = \frac{C_i(\beta)}{G_\beta} \qquad \text{(Eq. 11)}$$

where $C_i(\alpha)$ and $C_i(\beta)$, respectively, are the average concentrations of gross α and gross β/γ activities on the i-th waste package. G_α and G_β, respectively, are the contamination level derived for the gross α and gross β/γ activities. The determination of G_α and G_β levels is given in Section 6.1.

4. DESIGN INPUTS

The technical product input and sources of the input used in the development of this report are documented in this section.

4.1 DESIGN PARAMETERS

The design parameters used in this report are identified and provided in the following sections. The following parameters are used either in Section 6 or the Appendices.

4.1.1 Emplacement Drift Ventilation Parameters

The emplacement drift and raise ventilation airflow rates are taken from *Underground Layout Configuration* (BSC 2003b, Section 8.6) and are provided as follows:

- Emplacement Drift airflow intake rate: 15 m^3/s
- Emplacement Drift airflow output rate: 17 m^3/s

4.1.2 Radiological Parameters

The contamination levels are derived only for those radionuclides that were found to be radiologically important in the repository. The specific method used to obtain the radionuclide lists is described in Appendix A. Table 1 lists the selected radionuclides and their airborne ECL values (Column 1 of Table 2 of Appendix B to 10 CFR Part 20). Gaseous radionuclides, such as tritium and Kr-85 are excluded from the table because they would not be present as surface contamination on a waste package. The airborne ECL represents the concentration, which if inhaled continuously over the course of a year, would produce a total effective dose equivalent (TEDE) of 50 mrem (10 CFR Part 20, Appendix B). The ECL values are used in Section 6.1 to derive the contamination levels.

Table 1. Airborne Effluent Limits

Radionuclide	ECL(Ci/m^3)[a]
Co-60	5×10^{-11}
Ni-63	2×10^{-9}
Sr-90	6×10^{-12}
Cs-137	2×10^{-10}
I-129	4×10^{-11}
Pm-147	3×10^{-10}
Sm-151	2×10^{-10}
Eu-154	3×10^{-11}
Pu-238	2×10^{-14}
Pu-239	2×10^{-14}
Pu-240	2×10^{-14}
Pu-241	8×10^{-13}
Am-241	2×10^{-14}
Am-243	2×10^{-14}

Cm-243	2×10^{-14}
Cm-244	3×10^{-14}

NOTE: [a]Source: 10 CFR Part 20 Appendix B, Column 1 of Table 3. This column lists the "radionuclide concentrations which, if inhaled... continuously over the course of a year, would produce a TEDE of 50 mrem." For conservatism, the lowest or most restrictive listed values are used in this report.

4.2 ASSUMPTIONS

The following assumptions are used for the derivation of the contamination levels.

4.2.1 Waste Package Surface Area

The waste package surface area is assumed to be 32 m^2. This is the same assumption that was used in *Subsurface Contamination Control* (Yuan Y. 2001, Section 4.2.10), and is used in Section 6.1 of this report to derive the contamination levels. This area was taken from *Retrieval Equipment and Strategy for WP on Pallet* (CRWMS M&O 2000, p. 63).

4.2.2 Waste Package Emplacement Rate

The number of waste packages emplaced annually is assumed to be 600 (rounded from 605). The number (600) is taken from the assumption made in *Subsurface Contamination Control* (Yuan Y. 2001, Section 4.2.13). The waste packages emplacement rate in Yuan Y., was taken from the assumption made in *Retrieval Equipment and Strategy for WP on Pallet* (CRWMS M&O 2000, p. 21). The number is consistent with Cloud, J.D. 2003. It represents an upper bound estimate of the waste package emplacement rate. This assumption is used in Section 3.2, 6.1 and Appendix A of this report.

4.2.3 Shaft Exhaust Ventilation Rate

The shaft exhaust ventilation rate is assumed to be 340 m^3/s. This flow rate represents four shafts and two raises (BSC 2003b, Section 8.6), utilizing 106 emplacement drifts as stated in BSC 2003b, with four shafts and two rises (106 drifts / 6 shafts (which represents six release points) = 17.6 , where 17.6 is rounded to twenty), each with an airflow output of 17 m^3/s (Section 4.1.1, 20 drifts x 17 m^3/s per drift = 340 m^3/s). For all other exhaust flow rates (V), the derived contamination levels should be multiplied by a flow-rate-adjusting factor. This assumption is used in Section 6.1, and to derive the contamination levels in Appendix A. For other exhaust airflow rate (V) the values of the derived contamination levels in Table 3 should be multiplied by φ, where φ is the flow-rate-adjusting factor and is given as:

$$\varphi = V\ (m^3/s) / 340 \qquad \text{(Eq. 12)}$$

4.2.4 Radioactive Decay from Waste Package Surface Contamination

Radioactive decay is assumed to be negligible in deriving the contamination levels in Section 3.1. This assumption is appropriate because it yields the most restrictive or the smallest contamination level.

4.3 CRITERIA

Criteria applicable to this report are taken from *Project Design Criteria Document* (Minwalla 2003, Sections 4.9.3.2 and 4.9.3.3) which states that as low as is reasonably allowable (ALARA) is a regulatory requirement for the Yucca Mountain repository that must be addressed.

The ALARA design goals for occupational workers are to ensure that both individual and collective annual doses are maintained at ALARA levels during normal operations and as a result of Category 1 event sequences. Category 1 event sequences will be included in ALARA dose assessments. The following ALARA design goals are established for the design process.

- The ALARA goal for on-site members of the general public is to maintain individual doses less than 10 to 20 percent of the annual TEDE limit of 100 mrem in 10 CFR 20.1301.

- The general public dose shall comply with the pre-closure performance objectives as specified in 10 CFR 63.111, as well as the annual effluent dose limit of 10 mrem in 10 CFR 20.1101(d).

4.4 REGULATIONS AND GUIDES

The following regulations appear in this report.

4.4.1 10 CFR Part 63

10 CFR 63.111(a)(1) states that "the geological repository operations area must meet the requirements of part 20 of this Chapter [10 CFR Part 20]."

4.4.2 10 CFR Part 20

10 CFR 20.1302(b) requires that "A licensee shall show compliance with the annual dose limit in Sec. 20.1301 by (1) Demonstrating by measurement or calculation that the total effective dose equivalent to the individual likely to receive the highest dose from the licensed operation does not exceed the annual dose limit; or (2) Demonstrating that (i) The annual average concentrations of radioactive material released in gaseous and liquid effluents at the boundary of the unrestricted area do not exceed the values specified in Table 2 of Appendix B to Part 20."

20.1101(d) requires that "To implement the ALARA requirements of Sec. 20.1101(b), and notwithstanding the requirements in Sec. 20.1301 of this part, a constraint on air emissions of radioactive material to the environment, excluding Radon-222 and its daughters, shall be established by licensees ... such that the individual member of the public likely to receive the highest dose will not be expected to receive a total effective dose equivalent in excess of 10 mrem (0.1 mSv) per year from these emissions."

4.4.3 Regulatory Guide 4.20

Regulatory Guide 4.20 *Constraint on Releases of Airborne Radioactive Materials to the Environment for Licensees Other Than Power Reactors* (1996) Section 2.1, describes four screening techniques that can be used to implement the ALARA requirements. One of the screening techniques, which is acceptable by NRC for demonstrating compliance with 10 CFR 20.1101(d), conservatively assumes that the air concentration at the receptor is equal to the air concentration measured at the point of release (Regulatory Guide 4.20 1996, Subsection 2.1). This is analogous to demonstrating that the annual average concentrations of radioactive material released in gaseous effluents at the location of the exhaust do not exceed the values specified in Table 2 of Appendix B to 10 CFR Part 20 which are the airborne effluent concentration limits (ECL).

4.4.4 49 CFR 173.443

The following table which states the maximum permissible limits on the exterior of a shipping package is from Table 11 of 49 CFR 173.443. The contamination limits for the Department of Transportation are listed for comparison purposes to the individual contamination levels that will be derived.

Table 2. Non-Fixed External Radioactive Contamination-Wipe Limits

Contaminant	Maximum permissible limits		
	Bq/cm^2	$\mu Ci/cm^2$	dpm/cm^2
Beta and gamma emitters and low toxicity alpha emitters	0.4	10^{-5}	22
All other alpha emitting radionuclides	0.04	10^{-6}	2.2

5. USE OF SOFTWARE

Microsoft Excel 97, a spreadsheet computational software, was the only software used in this calculation. Excel is a commercial off-the-shelf spreadsheet program designed to assist in performing calculations and is an exempt software product in accordance with Section 2.1.6 of AP-SI.1Q, *Software Management*, and Section 5.2.3c) of AP-3.11Q, Technical Reports. The formulas used in the spreadsheets for all calculations were verified, in accordance with AP-SI.1Q, *Software Management* (Section 2.1.6), to be working correctly by hand calculations.

6. ANALYSIS

The contamination levels for accepting waste packages into the repository are derived in this section. A recommended survey procedure in the transfer cells is also described.

6.1 DERIVATION OF RADIOACTIVE CONTAMINATION LEVELS

The derived contamination levels represent the average concentration of radioactivity on the external surfaces of a waste package that should not be exceeded before the waste package is transported to the subsurface repository. The derivation of the contamination level is based on the

requirement that the annual average concentrations of radioactive material released at the repository shaft exhaust do not exceed the airborne ECL specified in Table 1 (see Section 4.1.2).

The contamination levels, G (Ci/m^2), are derived based on the methodology in Section 3.1, the input parameters in Section 4.1, and applicable assumptions in Section 4.2. Specifically, the G values presented in Appendix A are calculated by Equation 5 with:

ECL = values in Table 1 (Ci/m^3)
V = 340 m^3/s (Assumption 4.2.3)
= 1.07E10 m^3/yr
A = 32 m^2 (Assumption 4.2.1)
M = 600 /yr (Assumption 4.2.2)

The calculated contamination levels for radionuclides of radiological importance are listed in Columns 3 and 4 of Table A-3, respectively, in Ci/m^2 and dpm/100 cm^2.

During a confirmation survey of a waste package when applying Equations 8 through 11 (Section 3.2), the most restrictive contamination levels derived for alpha and beta/gamma emitting nuclides present in Table A-3 should be used. That is G_α = 250 dpm/100 cm^2 for α emitting nuclides and G_β = 9,900 dpm/100 cm^2 for β/γ emitting nuclides, respectively.

However, since Pu-241, a very low energy beta emitter (20 keV maximum), is undetectable by conventional survey instruments, its potential presence on the waste packages has to be treated separately from the other β/γ emitters. Equation 9 is rewritten in the following form by separating Pu-241 from the rest of the β/γ emitters:

$$F_i(\alpha\beta) = F_i(\alpha) + \frac{C_{i,Pu-241}}{G_{Pu-241}} + F_i(\beta) \qquad \text{(Eq. 13)}$$

where $C_{i, Pu-241}$ is the average activity concentration of Pu-241 on the i-th waste package and G_{Pu-241} is the contamination level derived for Pu-241 (9,900 dpm/100 cm^2). By separating Pu-241 from the rest of β/γ emitters, G_β in Equation 13 now has a value of 74,000 dpm/100 cm^2 and is represented by the second most limiting contamination level for β/γ emitting nuclides listed on Table A-3 (Sr-90).

In order to apply Equation 13, Pu-241 concentration, $C_{i, Pu-241}$, must be determined. Because Pu-241 is found only with other plutonium isotopes (i.e., Pu-238, Pu-239, and Pu-240), which are alpha-emitters, its contamination level therefore can be inferred from the concentrations detected for these plutonium isotopes. The following equation is used to estimate $C_{i, Pu-241}$:

$$C_{i, Pu-241} = \eta C_{i, \alpha-Pu} \qquad \text{(Eq. 14)}$$

or

$$\eta = \frac{C_{i, Pu-241}}{C_{i, \alpha-Pu}} \qquad \text{(Eq. 15)}$$

where $C_{i,\ \alpha\text{-Pu}}$ represents the gross activity concentration of α emitting plutonium isotopes. To apply the η value to Equation 13, it is conservatively assumed that α activities detected on waste packages would be derived from α emitting plutonium isotopes or:

$$C_{i,\ \text{Pu-241}} = \eta C_i(\alpha) \qquad \text{(Eq. 16)}$$

Applying Equation 16 to Equation 13, Equation 17 is obtained:

$$F_i(\alpha\beta) = \frac{C_i(\alpha)}{G_\alpha} + \frac{\eta C_i(\alpha)}{G_{\text{Pu-241}}} + F_i(\beta) \qquad \text{(Eq. 17)}$$

Equation 17 can be re-arranged in the following form:

$$F_i(\alpha\beta) = \xi F_i(\alpha) + F_i(\beta) \qquad \text{(Eq. 18)}$$

where

$$\xi = 1 + \frac{\eta G_\alpha}{G_{\text{Pu-241}}} \qquad \text{(Eq. 19)}$$

The calculation of a conservative value of ξ that can be applied to spent nuclear fuel is presented in Appendix B (Table B-1). The calculation is based on the relative radionuclide inventory of the plutonium isotopes estimated for the average and maximum pressurized water reactor (PWR) and boiling water reactor (BWR) spent nuclear fuel. An average ξ value of 1.25 is the result (see Appendix B).

Since the differences among contamination levels derived for alpha emitting nuclides (Table A-3) are not significant, individual α emitting nuclides are not measured during confirmation survey. For β/γ emitting nuclides, it may be desirable in some cases to analyze specific β/γ emitting nuclides. In those cases, $F_i(\beta)$ may be calculated by:

$$F_i(\beta) = \frac{C_i(\beta) - \sum_k C_{i,\beta_k}}{G_\beta} + \sum_k \frac{C_{i,\beta_k}}{G_{\beta_k}} \qquad \text{(Eq. 20)}$$

where G_{β_k} is the derived contamination level for the k-th nuclide. $C_{i,\ \beta_k}$ is the average concentration of the k-th β/γ emitter identified on the surface of the i-th waste package. To simplify the survey process, only two β/γ emitting nuclides of radiological significance may be measured, if required. They are Co-60 and Cs-137 (Ba-137m). Cs-137 (Ba-137m) is selected because of its relatively large inventory in the spent nuclear fuel (BSC 2003a, Attachment IX and BSC 2003c, Attachment XIII). Co-60 is selected (NRC 2000, Section 9.5.2.2, Table 9.2 footnote) because it is a major activated corrosion product (crud) that could be removed from the fuel rod rather easily and become potential contamination on the waste packages.

Table 3 summarizes the derived contamination levels (see Table A-3) that may be used to demonstrate compliance for radiation survey of the waste packages. The derived contamination levels for gross alpha and gross beta/gamma activities, respectively, are 250 dpm/100 cm^2 and 74,000 dpm/100 cm^2. The derived levels for Co-60 and Cs-137, respectively, are 620,000 dpm/100 cm^2 and 2,500,000 dpm/100 cm^2. The contamination levels for these two nuclides are included in Table 3 for waste package confirmation surveys that require identification of specific nuclides.

Table 3. Derived Contamination Levels

	Derived Contamination Levels[a]	
	(Ci/m^2)	($dpm/100\ cm^2$)
Alpha-emitting nuclides (G_α)	1.1E-8	250 (α)
Beta/Gamma emitting nuclides (G_β)	3.4E-06	74,000 (β/γ)
Co-60 ($G_{Co\text{-}60}$)	2.8E-05	620,000 (β/γ)
Cs-137 ($G_{Cs\text{-}137}$)	1.1E-04	2,500,000 (β/γ)

NOTE: [a]Derived contamination levels was based on an exhaust ventilation flow rate of 340 m^3/s. This flow rate represents the ventilation airflow of a total of 20 emplacement drifts (Assumption 4.2.3). For all other exhaust flow rates (V), the derived contamination levels should be multiplied by a flow-rate-adjusting factor $\varphi = V\ (m^3/s)/340$.

Verification Calculation utilizing Co-60
(5E-11 uCi/cc*340 m^3/s*3600 s/hr *24 hr/day *365 day/yr*1E-6 Ci/uCi)/(32 m^2*600 per yr) = 2.8E-05 Ci/m^2
or 2.8E-05 Ci/m^2 *1 m2/1002 cm^2 * 3.7E10 dis/s *60 s/min *100 $cm^2/(100cm^2)$ = 620,000 $dpm/100\ cm^2$

6.2 PROPOSED WASTE PACKAGE SURVEY PROCESS

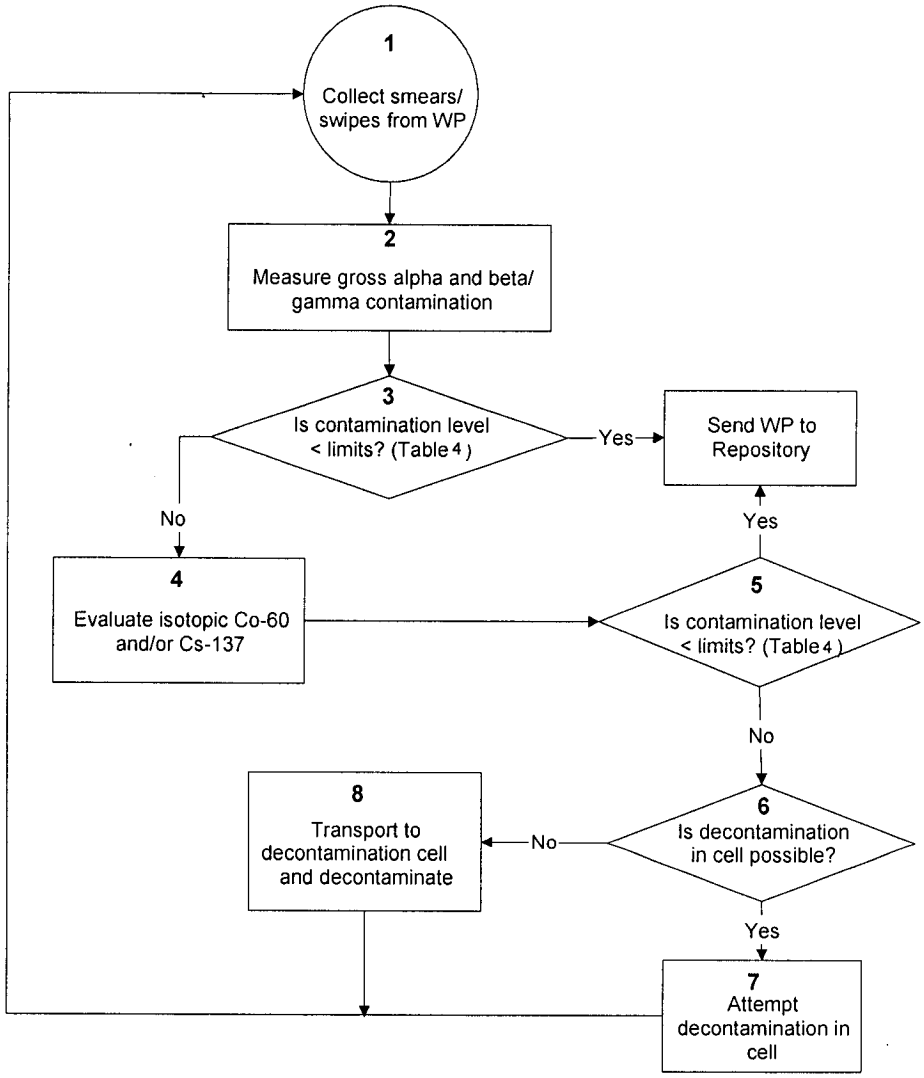
During repository operation, an operational procedure for waste package contamination surveys will be required to demonstrate that removable surface contamination in excess of the contamination levels is not present on the waste packages. This section describes an approach that may be used to demonstrate compliance. Table 4 summarizes the calculation steps and equations that may be used to demonstrate compliance following survey measurements. For accepting a waste package into the subsurface repository for emplacement, the average amount of radioactivity detected on the waste package should not exceed the criteria set forth by calculation steps A through D in Table 4. Figure 1 depicts the process for radiation surveys that may be used for demonstrating compliance.

Since very high radiation levels are expected in the transfer cells, it is not feasible to detect the contaminants with portable field instruments. Radiation surveys of waste packages during the pre-closure period of the repository, will therefore, be performed remotely. The recommended approach of a remote survey operation is provided in Appendix C.

Table 4. Calculation Steps for Demonstrating Compliance

Calculation Step		Equation	Source	Comments
A		$F_i(\alpha) = \frac{C_i(\alpha)}{G_\alpha}$	Eq. 10	Calculate $F_i(\alpha)$. (See Section 3.2) • $C_i(\alpha)$ is the measured average concentration of gross α activity of the i-th waste package. • G_α is the contamination level for gross α activity (see Table 3).
B	i	$F_i(\beta) = \frac{C_i(\beta)}{G_\beta}$ or	Eq. 11	Calculate $F_i(\beta)$. (See Section 3.2) • Eq. 11 is used if only gross β/γ activity is measured. • $C_i(\beta)$ is the measured average gross β/γ activity of the i-th waste package. • G_β is the contamination level for gross β/γ activity (see Table 3). • Eq. 20 is used if in addition to gross β/γ activity, Co-60 and/or Cs-137 activities are measured. (see Section 6.1). • C_{i,β_k} is the measured average concentration of these nuclides. • G_{β_k} is the single-nuclide contamination level derived for these nuclides (see Table 3).
	ii	$F_i(\beta) = \frac{C_i(\beta) - \sum_k C_{i,\beta_k}}{G_\beta} + \sum_k \frac{C_{i,\beta_k}}{G_{\beta_k}}$	Eq. 20	
C		$F_i(\alpha\beta) = 1.25 F_i(\alpha) + F_i(\beta)$	Eq. 18	Calculate $F_i(\alpha\beta)$. (See Section 6.1)
D		$\bar{F}(\alpha\beta) = \frac{1}{N}\sum_{i=1}^{N} F_i(\alpha\beta) \le 1$	Eq. 8	Calculate $\bar{F}(\alpha\beta)$ (See Section 3.2). • N is the number of waste packages emplaced over a period of up to 90 days including the current waste package. • If the calculated $\bar{F}(\alpha\beta)$ is less than or equal to 1.0, the WP is ready for emplacement. Otherwise, decontamination may be required (see Figure 1).

NOTE: WP = waste package.



1) Take smears from the waste package (see Appendix C for a more detailed description of the approach).
2) Measure smears using portable instrument that is in a shielded pass-through port that connects working area to the transfer cell via pass-through ports.
3) Calculate $\bar{F}(\alpha\beta)$ with calculation steps outlined in Table 4 using Step A, for gross alpha $F_i(\alpha)$, and Step B-i for gross beta/gamma $F_i(\beta)$, Step C for $F_i(\alpha\beta)$, and Step D for averaging. If $\bar{F}(\alpha\beta) \leq 1$ per Step D, the waste package is allowed to be transported to the subsurface facility for emplacement.
4) Measure Co-60 and/or Cs-137 using multi-detector instruments.
5) Calculate $\bar{F}(\alpha\beta)$ using Steps A, B-ii, C, and D outlined in Table 4. If $\bar{F}(\alpha\beta) \leq 1$, per Step D, the waste package is allowed to be transported to the subsurface facility for emplacement.
6) If surveys indicate that the waste package is grossly contaminated, operational constraints may dictate that decontamination in Decontamination Cell is more productive.
7) If the waste package is not grossly contaminated, in cell decontamination may be possible using vacuum and/or large area wipes.
8) Transport the waste package to Decontamination Cell. The waste package should leave the decontamination facility meeting the contamination levels.

Figure 1. Waste Package Survey Operational Procedure Flowchart

7. CONCLUSIONS

The contamination levels were derived for acceptance of waste packages into the subsurface repository. The derivation of the contamination levels on the external surface of waste packages was based on limiting the annual average concentrations of airborne effluents at the subsurface exhaust to the airborne ECL values specified in Table 2 of Appendix B to 10 CFR 20. The derived contamination levels may be applied to survey data to demonstrate compliance. The results of the analysis indicate that the ECL constraints would not be exceeded at the subsurface ventilation exhausts during the pre-closure period of the repository if the contamination levels presented in Table 3 of this report are not exceeded. A proposed operational procedure for surveying waste packages to demonstrate compliance is depicted in Figure 1.

The derived contamination levels for gross alpha-emitting nuclides and gross beta/gamma emitting nuclides, respectively, are 250 dpm/100 cm^2 and 74,000 dpm/100 cm^2. The 250 dpm/100 cm^2 derived for alpha emitting nuclides is about the same as the maximum permissible limit of 220 dpm/100 cm^2 listed in Table 2 (Non-Fixed External Radioactive Contamination-Wipe Limits) from 49 CFR 173.443 for transportation packages. The 74,000 dpm/100 cm^2 derived for gross beta/gamma emitters is about 33 times higher than the maximum permissible limit of 2,200 dpm/100 cm^2 also listed in Table 11 of 49 CFR 173.443. The derived contamination level for Co-60 and Cs-137 are 620,000 dpm/100 cm^2 and 2,500,000 dpm/100 cm^2, respectively. These single-nuclide values are included for individual surveys that may require extra effort to identify specific nuclides for demonstrating compliance. It should be noted that the derived levels were based on an exhaust ventilation flow rate of 340 m^3/s from 20 emplacement drifts. For other exhaust flow rates, the contamination levels to be used should be adjusted accordingly.

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APPENDIX A— CALCULATION OF SINGLE-RADIONUCLIDE CONTAMINATION LEVELS

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APPENDIX A—CALCULATION OF SINGLE-RADIONUCLIDE CONTAMINATION LEVELS

In this report, gaseous radionuclides, such as tritium and Kr-85 are excluded because they would not be present as surface contamination on a waste package. Tables A-1 (PWR) and A-2 (BWR) present the individual radionuclide activity and its contribution in percent (%) for source term nuclides, and the selected radionuclides are in bold face to be included in Table A-3. The contamination levels in Table A-3 are derived only for the radionuclides that were identified to be radiologically important in Tables A-1 and A-2. The selection criteria for radionuclides identified as radiologically important are based on U.S. Nuclear Regulatory Commission (NRC) guidance documents (NRC 2000, Section 9.5.2.2; and NRC 2003, Section 3). The analysis includes the activity from the Co-60 in the form of an activated corrosion product (crud material), the activity for iodine, other fission products that contribute greater than 0.1 percent of design basis fuel activity, and actinide activity that contributes greater than 0.01 percent of the design basis activity.

Table A-1. Source Term Selection for Pressurized Water Reactor

Nuclide[a]	Avg. PWR Curies/FA[a,b]	% activity	Nuclide[b]	Avg. PWR Curies/FA[b]	% activity
H-3	1.14E+02	0.112%	**Eu-154[c]**	6.71E+02	0.657%
C-14	3.32E-01	0.000%	Eu-155	5.16E+01	0.051%
Cl-36	6.80E-03	0.000%	Ac-227	1.61E-05	0.000%
Fe-55	3.47E+00	0.003%	Th-230	1.48E-04	0.000%
Ni-59	2.09E+00	0.002%	Pa-231	2.97E-05	0.000%
Co-60[c]	3.13E+02	0.307%	U-232	2.05E-02	0.000%
Ni-63[c]	2.52E+02	0.247%	U-233	4.07E-05	0.000%
Se-79	4.57E-02	0.000%	U-234	6.77E-01	0.001%
Kr-85[c]	1.13E+03	1.107%	U-235	7.36E-03	0.000%
Sr-90[c]	2.72E+04	26.639%	U-236	1.72E-01	0.000%
Nb-93m	1.30E+01	0.013%	Np-237	2.47E-01	0.000%
Zr-93	8.94E-01	0.001%	**Pu-238[c]**	2.29E+03	2.243%
Nb-94	8.39E-01	0.001%	U-238	1.48E-01	0.000%
Tc-99	8.99E+00	0.009%	**Pu-239[c]**	1.77E+02	0.173%
Ru-106	1.23E-02	0.000%	**Pu-240[c]**	3.18E+02	0.311%
Pd-107	8.41E-02	0.000%	**Am-241[c]**	1.98E+03	1.939%
Cd-113m	7.66E+00	0.008%	**Pu-241[c]**	2.47E+04	24.190%
Sb-125	9.71E+00	0.010%	Am-242m	6.39E+00	0.006%
Sn-126	3.85E-01	0.000%	Cm-242	5.27E+00	0.005%
I-129	**2.20E-02**	**0.000%**	Pu-242	1.64E+00	0.002%
Cs-134	2.52E+01	0.025%	**Am-243[c]**	2.20E+01	0.022%
Cs-135	3.50E-01	0.000%	**Cm-243[c]**	1.03E+01	0.010%
Cs-137[c]	4.11E+04	40.252%	**Cm-244[c]**	1.36E+03	1.332%
Pm-147[c]	1.19E+02	0.117%	Cm-245	3.07E-01	0.000%
Sm-151[c]	2.11E+02	0.207%	Cm-246	1.04E-01	0.000%
Total				1.02E+05	100%

NOTES: [a]FA = Fuel Assembly per BSC 2003a, Attachment IX
[b]Source: BSC 2003a, Attachment IX.
[c]Radionuclides selected (in bold face) per NRC 2000 Section 9.5.2.2 and NRC 2003 Section 3 are: Fission products> 0.1% total activity; and Actinides> 0.01% total activity.

Table A-2. Source Term Selection for Boiling Water Reactor

Nuclide[a]	Avg. BWR Curies/FA[a]	% activity	Nuclide[a]	Avg. BWR Curies/FA[a]	% activity
H-3	3.95E+01	0.121%	**Eu-154[b]**	1.75E+02	0.552%
C-14	1.75E-01	0.001%	Eu-155	1.60E+01	0.050%
Cl-36	2.93E-03	0.000%	Ac-227	0.00E+00	0.000%
Fe-55	1.09E+00	0.003%	Th-230	6.09E-05	0.000%
Ni-59	5.02E-01	0.002%	Pa-231	1.39E-05	0.000%
Co-60[b]	4.39E+01	0.135%	U-232	4.63E-03	0.000%
Ni-63[b]	5.86E+01	0.180%	U-233	1.06E-05	0.000%
Se-79	1.59E-02	0.000%	U-234	2.50E-01	0.001%
Kr-85	3.81E+02	1.168%	U-235	2.62E-03	0.000%
Sr-90[b]	9.54E+03	29.237%	U-236	6.26E-02	0.000%
Nb-93m	4.73E-01	0.001%	Np-237	6.88E-02	0.000%
Zr-93	3.39E-01	0.001%	**Pu-238[b]**	5.85E+02	1.793%
Nb-94	1.87E-02	0.000%	U-238	6.32E-02	0.000%
Tc-99	3.20E+00	0.010%	**Pu-239[b]**	5.35E+01	0.164%
Ru-106	3.00E-03	0.000%	**Pu-240[b]**	1.14E+02	0.349%
Pd-107	2.65E-02	0.000%	**Am-241[b]**	5.58E+02	1.710%
Cd-113m	2.26E+00	0.007%	**Pu-241[b]**	6.78E+03	20.778%
Sb-125	2.89E+00	0.009%	Am-242m	2.17E+00	0.007%
Sn-126	1.27E-01	0.000%	Cm-242	1.78E+00	0.005%
I-129	**7.42E-03**	**0.000%**	Pu-242	5.08E-01	0.002%
Cs-134	6.32E+00	0.019%	**Am-243[b]**	5.34E+00	0.016%
Cs-135	1.39E-01	0.000%	**Cm-243[b]**	2.47E+00	0.008%
Cs-137[b]	1.39E+04	42.599%	**Cm-244[b]**	2.55E+02	0.785%
Pm-147[b]	3.98E+01	0.122%	Cm-245	4.03E-02	0.000%
Sm-151[b]	5.39E+01	0.165%	Cm-246	1.45E-02	0.000%
Total				3.26E+04	100.00%

NOTES: [a]Source: (BSC 2003c, Attachment XIII).
[b]Radionuclides selected (bold face) per NRC 2000, Section 9.5.2.2, and NRC 2003, Section 3 are: Fission products> 0.1% total activity; and actinides> 0.01% total activity.

Calculation of the Contamination Levels

The calculation of the contamination levels for selected radionuclides from Tables A-1 and A-2 is based on Equation 5 in Section 3.1. The results are listed in Table A-3.

The most limiting contamination levels in Table A-3 are 250 dpm/100 cm^2 for alpha emitters and 9,900 dpm/100 cm^2 for beta/gamma emitters. The second most limiting value for β/γ emitting nuclides is 74,000 dpm/100 cm^2 (Sr-90).

The derived values in Table A-3 should be considered as the "annual average levels" for the radionuclides of concern. During confirmation survey of a waste package, the most restrictive contamination level for alpha or beta/gamma emitters in Table A-3 should be used to demonstrate compliance unless the actual radionucldies are known.

Table A-3. Derivation of Radioactive Contamination Levels

Nuclide	(1) Airborne ECL (Ci/m^3)	Derived Contamination Level	
		(2) (Ci/m^2)	(3) (dpm/ 100 cm^2)
Beta/Gamma emitting nuclides:			
Co-60	5.0E-11	2.8E-05	6.2E+05
Ni-63	2.0E-09	1.1E-03	2.5E+07
Sr-90	6.0E-12	3.4E-06	7.4E+04
I-129	4.0E-11	2.2E-05	5.0E+05
Cs-137	2.0E-10	1.1E-04	2.5E+06
Pm-147	3.0E-10	1.7E-04	3.7E+06
Sm-151	2.0E-10	1.1E-04	2.5E+06
Eu-154	3.0E-11	1.7E-05	3.7E+05
Pu-241	8.0E-13	4.5E-07	9.9E+03
Alpha emitting nuclides:			
Pu-238	2.0E-14	1.1E-08	2.5E+02
Pu-239	2.0E-14	1.1E-08	2.5E+02
Pu-240	2.0E-14	1.1E-08	2.5E+02
Am-241	2.0E-14	1.1E-08	2.5E+02
Am-243	2.0E-14	1.1E-08	2.5E+02
Cm-243	2.0E-14	1.1E-08	2.5E+02
Cm-244	3.0E-14	1.7E-08	3.7E+02

NOTES: (1) Table 1.

(2) Equation 5: (1) x 340 (m^3/s) / [32 (m^2/WP) x 600 (WP/yr.)] x 86400 (s/d) x 365 (d/yr.)

(3) (2) x 2.22E12 (dpm/Ci) x 100 cm^2.

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APPENDIX B—DETERMINATION OF PU-241 ξ FACTOR

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APPENDIX B—DERIVATION OF PU-241 ξ FACTOR

The Pu-241 ξ factor in Equation 19 is derived in this Appendix with Equations 15 and 18 in Section 6.1..

$$\xi = 1 + \frac{\eta G_{\alpha}}{G_{Pu\text{-}241}} \qquad \text{(Eq. B-1)}$$

where

$$\eta = \frac{C_{i,\,Pu\text{-}241}}{C_{i,\,\alpha\text{-}Pu}} \qquad \text{(Eq. B-2)}$$

The calculated η value is presented in Table B-1. The calculation is based on the relative radionuclide inventory of the plutonium isotopes calculated for the average and maximum PWR and BWR spent nuclear fuel. An average η value of 10 (rounded from 9.8) is obtained.

Table B-1. Calculation of η Factor

Nuclide	Curie/Fuel Assembly[a]			
	Avg. PWR	Max PWR	Avg. BWR	Max BWR
Pu-238	2.29E+03	6.16E+03	5.85E+02	2.11E+03
Pu-239	1.77E+02	1.85E+02	5.35E+01	5.36E+01
Pu-240	3.18E+02	3.90E+02	1.14E+02	1.48E+02
Pu-242	1.64E+00	3.01E+00	5.09E-01	1.26E+00
Am-241	1.98E+03	8.71E+02	5.58E+02	2.66E+02
Am-242m	6.39E+00	1.02E+01	2.17E+00	3.40E+00
Cm-242	5.27E+00	3.43E+01	1.79E+00	1.13E+01
Am-243	2.20E+01	5.22E+01	5.35E+00	1.93E+01
Cm-243	1.03E+01	3.83E+01	2.48E+00	1.12E+01
Cm-244	1.36E+03	1.12E+04	2.56E+02	3.95E+03
Total Alpha (α)	6.17E+03	1.89E+04	1.58E+03	6.57E+03
Total α-emitting Pu	2.79E+03	6.74E+03	7.53E+02	2.31E+03
Pu-241	2.47E+04	7.91E+04	6.78E+03	2.25E+04
Pu-241/Total α	4.00	4.18	4.29	3.42
Pu-241/Total α-Pu	8.9	11.7	9.0	9.7
Average η Factor	**9.8[b]**			

[a]Average PWR assembly: 4.0%, 48 GWd/MTU, 25 years, (BSC 2003a, Attachment IX)
Maximum PWR assembly: 5.0%, 75 GWd/MTU, 5 years, (BSC 2003a, Attachment IX)
Average BWR assembly: 3.5%, 40 GWd/MTU, 25 years (BSC 2003c, Attachment XIII)
Maximum BWR assembly: 5.0%, 75 GWd/MTU, 5 years (BSC 2003c, Attachment XIII)

[b]9.8 = (8.9+11.7+9.0+9.7)/4

Using the value of η = 10, $G_{Pu\text{-}241}$ from Table A-3 and G_{α} from Table 3, ξ is calculated to be 1.25 (= 1 + 10 x 250/9900).

Verification Calculation:
2.29E3+1.77E2+3.18E2+1.64 = 2.79E3
2.47E4/2.79E3 = 8.9
ξ = 1 + (8.9)*250/9900 = 1.22

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APPENDIX C—PROPOSED WASTE PACKAGE SURVEY

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APPENDIX C—PROPOSED WASTE PACKAGE SURVEY

A typical survey practice is to take a sample of removable surface contamination by wiping a sizable area with an absorbent material with moderate pressure, and measuring the activity of the sample. Sufficient measurements must be taken to yield representative assessment of non-fixed contamination levels. The sufficient number of measurements required will be provided to personnel that perform the survey per a procedure when needed.

Monitoring the removable surface contamination on a waste package will consist of utilizing conventional smear techniques for quantitative analyses and large area wipes for qualitative analyses. Measuring the removable surface contamination will be performed with direct frisking by relocating and/or shielding the smears and large area wipes from high background radiation areas to quantify and qualify the levels of removable contamination. When performing an evaluation of the amount of removable contamination present on a waste package both large area wipes and smears will be taken for each location.

The surveys will be performed in a transfer cell or a similarly equipment area. The transfer cell will contain the waste package with the equipment necessary to survey the exterior of the package for contamination. The cell would contain equipment such as a manipulator arm for collecting smears remotely, and a shielded pass-through port which will allow access between the cell and the counting station.

A general description of each process follows:

Large Area Wipes—The large area wipes consist of a cloth type material (e.g., masslin cloth) that will be used to wipe surfaces that potentially contain removable radioactive material. By using large area wipes, the direct removable surface contamination that is transferred to the large area wipe indicates if removable contamination is present on the surface being tested. Five large area wipes are proposed to achieve the previously described general survey process as discussed below. A typical survey with a large area wipe is approximately one square meter of surface area. By utilizing five large area wipes per waste package, approximately eighteen percent of the waste package surface area will be checked for contamination (CRWMS M&O 2000, Table 3).

Conventional Smears—A major distinction between large area wipes and smears is that a large area wipe is used for measuring the gross removable contamination present on the surface of objects. A smear is a circular pad, approximately the size of a silver dollar. Depending on the manufacture, the back of the smear may or may not contain an adhesive based material. Quantifying removable contamination with a smear is used by convention for comparison purposes because the surface area tested is the same size as the regulatory or administrative level being reviewed. For waste packages: the surface area of a waste package wiped with a smear is approximately 100 cm^2 such that an easy comparison with the levels can be made. The smears will be used to quantify removable surface contamination with portable instrumentation and, if needed, for more specific analysis of surface contamination.

Direct Frisking—Utilizing hand-held portable instrumentation, direct readings of smears and large area wipes will be made. The instruments used for frisking should be capable of detecting removable contamination at or below the total surface contamination values that will be the action level for further evaluation by the project. A major consideration is that the hand-held instrumentation is very sensitive to radiation and can not distinguish removable contamination from elevated background reading without shielding.

The recommended waste package survey, illustrated in Figure 1, is as follows:

1. After the spent fuel or high-level radioactive waste is placed in the waste package and the package sealed, a large area wipe and a smear will be taken on the top of the waste package utilizing remote methods. The large area wipe and smear will then be placed in a pass-through port utilizing care to assure that any potential removable contamination from within the transfer cell does not cross contaminate the large area wipe or smear.

2. The large area wipe and smear will then be checked for removable contamination utilizing direct frisking techniques. The hand-held instrumentation should be placed in a pass-through type glove port area between the transfer cell and the technician measuring the removable contamination to prevent the spread of contamination. Table 4, Steps A, B-i, C, and D will be utilized to determine if the gross contamination levels are below the derived values. If they are, no further analysis is required. If the contamination levels exceed the screening level, then a secondary analysis utilizing Steps A, B-ii, C, and D could be utilized (see Figure 1) to determine if the levels are below the derived values, based on the isotopic distribution, if needed.

3. The previous process will be repeated for the following locations: the weld area of the waste package, down one side of the waste package, down the other side of the waste package, and near the bottom of the waste package. The large area wipes and smears will follow the process described previously.

It should be noted that the bottom of the waste package will not be checked for contamination because the waste package will be loaded into the transfer cell after being placed on a trolley, and the bottom of the waste package is in direct contact with the bottom of the trolley. As such, the contamination that is capable of migration to the bottom surface is minimized. However, if a waste package is decontaminated, the bottom should be surveyed after decontamination.

The time required to complete the survey will depend on the time it takes to collect the samples (large area wipes and smears), to place them in the shielded pass-through port, and to check them for contamination. Utilizing portable instrumentation allows timely measurement of the contamination levels after the large area wipes and smears are placed in the pass-through area. The time required to take large area wipes and smears will depend on the speed of the equipment (e.g., manipulator arm) and the time required passing the large area wipes and smears through pass-through ports. Anticipated time for taking the large area wipes and smears is less

than one hour. Throughout the survey process, large area wipes will be used for a qualitative analysis of contamination. If the large area wipes indicate that the contamination levels could exceed the quantified levels for the smears (where the large area wipes are normalized to 100 cm^2 in surface area), further decontamination and/or surveys will be required depending on the contamination levels measured. If decontamination is required a survey will need to be performed prior to allowing placement into the subsurface to demonstrate that the decontamination process was successful. The smears will be used to quantify the amount of contamination in a specified location. When performing an evaluation of the amount of removable contamination present on a waste package, both large area wipes and smears will be taken for each location.