Attachment G

Oil and Gas Exploration and Production (E&P) NORM Waste Acceptance

SOP 15.WAC.04 was developed to implement a method to evaluate shipments of oil and gas field waste to verify that the material meets radioactive material license concentration limits for disposal. NORM waste from Oil and Gas Exploration and Production is ubiquitous in the environment and stakeholders desire more efficient methods of waste characterization. CHDT believes that careful screening procedures can ensure that this waste is well within license limits.

This new SOP provides a method for accepting waste streams consisting of NORM-containing filter socks and tank bottoms generated by oil and gas field operations. The method in the SOP requires generators to document the process (oil and gas exploration and production) that produced the waste along with detailed field surveys of the waste in order to characterize it for disposal at CHDT. Limitation to use of the SOP to oil and gas production and exploration provides assurance that the potential hazards and radionuclides are well understood.

The method in this SOP implements exposure rate measurements of each container upon arrival at CHDT along with gamma isotope identification to verify that the waste meets the requirements of the Adams County Amended Certificate of Designation and the Colorado Department of Public Health and Environment Radioactive Materials License.

A Technical Basis document is provided that derives and documents the exposure rate limits that must be met to verify that material meets the CHDT Waste Acceptance Criteria. The technical basis uses the radium isotopic mixture found in these oil and gas field waste streams.

Dade Moeller & Associates, Inc.

Technical Report

TECHNICAL BASIS FOR THE ACCEPTANCE OF OIL PRODUCTION PIPING AND EQUIPMENT CONTAINING RADIOACTIVE PIPE SCALE AT THE DEER TRAIL LANDFILL

DMA-TR-42, REV. 0
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EXECUTIVE SUMMARY

Naturally Occurring Radioactive Materials (NORM), in the form of radium containing pipe scale, is commonly found in the oil production industry. Oil field waste, largely surplus pipe and equipment, is candidate for disposal at the Deer Trail landfill because of its radioactive properties. However, to ensure safe disposal within the Deer Trail radioactive materials license conditions, Standard Operating Procedures (SOPs) are needed for radiation surveys prior to shipment (by the waste generator) and upon receipt (by Deer Trail personnel). This report establishes the relationship between external exposure rates and the average radionuclide concentrations per waste shipment, in a conservative manner, so that external exposure rate measurements can be used as the basis of the SOPs. Modeling results, using the MicroShield computer program and an empirical equation from the literature, are compared and used to establish the exposure rate/radionuclide concentration relationship. The analysis considers the potential variability encountered from pipes of selected diameters and schedules, ^{226}Ra to ^{228}Ra pipe scale ratios, and single pipe versus multiple pipe configurations. The Wilcoxon Rank Sum (WRS) test is used to establish the number of exposure rate measurements that are required to statistically characterize the average exposure rate from a shipment of oil field pipe and equipment. The analysis considers the potential radon post-closure impacts from receiving radium containing oil field waste. From the results of this analysis, a conservative basis for the SOPs is recommended.

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1. INTRODUCTION

The purpose of this report is to develop a technical basis for Standard Operating Procedures (SOPs) for radiation surveys prior to shipment (by the waste generator) and upon receipt (by Deer Trail personnel) of oil field pipe and equipment containing radium-contaminated pipe scale. The goal of this analysis is to determine the relationship between external exposure rates and average radionuclide concentrations per waste shipment in a conservative manner, so that external exposure rate measurements can be used as the basis of SOPs. The overall approach is to compare modeling exposure rate results, using MicroShield (Grove 2009) and an empirical formula from the literature (API 1997) for various pipe diameters, schedules, and scale thicknesses. From this comparison, the exposure rates are correlated with internal pipe scale concentrations so that measurements can be used to determine if the Deer Trail waste acceptance criteria are met for specific shipments of oil field pipe and equipment. Using MicroShield, sensitivity studies are conducted to determine the potential variability encountered from various ^{226}Ra to ^{228}Ra ratios on measured radiation exposure rates, and the variability from stacks or jumbles of pipe and equipment with various scale concentrations on waste acceptance decisions. A statistical evaluation is conducted to determine the number of exposure rate measurements required per shipment to implement the SOPs and reduce uncertainty. Finally, an evaluation of potential long-term radon concerns following closure is conducted. From the results, a conservative basis for the SOPs is recommended.

1.1 Background Information – The Nature of Radium Pipe Scale

Naturally occurring radioactive material (NORM) has been detected in pipe scale associated with the oil and gas industry. Radium is preferentially soluble in brine solutions (often common to oil production) compared to relatively insoluble uranium and thorium. Under certain conditions, the dissolved minerals in the brine solution lead to the formation of pipe scale. NORM pipe scale is produced when radium is co-precipitated with silicates, sulfates, or carbonates. The concentration of radium in oil field pipe scale is a function of the underlying geology and age of the well; more brine solution is typically recovered from older wells as oil production decreases. The scale can buildup in the pipe to thicknesses of up to several centimeters, depending on the pipe diameter, the total time the pipe is in service, and the total brine production. As a result, the concentration and quantity of radium in pipe scale can vary over a wide range. It has been reported that radium scale can vary in concentration from background to 410,000 picoCuries per gram (pCi/g) radium (EPA 1993), with a reported average of 480 pCi/g. The scale will typically contain predominantly ^{226}Ra and ^{228}Ra and their associated decay chains in equilibrium. Although the concentration ratio of ^{226}Ra to ^{228}Ra is highly variable, a typical ratio of 3:1 has been reported and used in previous assessments (Smith et al. 1996).

1.2 Radium Decay Chains

As part of the ^{238}U decay series, separated ^{226}Ra has eight radioactive decay chain products for a total of nine radionuclides. These include:

^{226}Ra (half-life ~ 1,660 years) decays by alpha/gamma radiation to:
^{222}Rn (half-life ~ 3.8 days) decays by alpha/gamma radiation to:
^{218}Po (half-life ~ 3.1 minutes) decays by alpha radiation to:

^{214}Pb (half-life 27 minutes) decays by beta/gamma radiation to:
^{214}Bi (half-life ~ 20 minutes) decays by beta/gamma radiation to:
^{214}Po (half-life ~ 160 microseconds) decays by alpha/gamma radiation to:
^{210}Pb (half-life ~ 22 years) decays by beta/gamma radiation to:
^{210}Bi (half-life ~ 5 days) decays by beta radiation to:
^{210}Po (half-life ~140 days) decays by alpha/gamma radiation to Stable ^{206}Pb

To meet the 2,000 pCi/g Deer Trail license limit, this means that, for a pure ^{226}Ra decay chain mixture, each member of the ^{226}Ra decay chain can have only one ninth (1/9th) of the total activity, or 220 pCi/g. As part of the ^{232}Th decay series, separated ^{228}Ra has nine radioactive decay chain products for a total of ten radionuclides. These include:

^{228}Ra (half-life ~ 5.8 years) decays by beta radiation to:
^{228}At (half-life ~ 6.1 hours) decays by beta/gamma radiation to:
^{228}Th (half-life ~ 1.9 years) decays by alpha/gamma radiation to:
^{224}Ra (half-life ~ 3.6 days) decays by alpha/gamma radiation to:
^{220}Rn (half-life ~ 55 seconds) decays by alpha/gamma radiation to:
^{216}Po (half-life ~ 0.15 seconds) decays by alpha radiation to:
^{212}Pb (half-life ~ 11 hours) decays by beta/gamma radiation to:
^{212}Bi (half-life ~ 5 days) decays 64% of the time by beta/gamma radiation to:
^{212}Po (half-life ~ 300 nanoseconds) decays to Stable ^{208}Pb
Or: ^{212}Bi (half-life ~ 5 days) decays 36% of the time by alpha/gamma radiation to:
^{208}Tl (half-life ~3.1 minutes) decays by beta/gamma radiation to Stable ^{206}Pb

Again, to meet the 2,000 pCi/g limit, this means that, for a pure ^{228}Ra decay chain mixture, each member of the ^{228}Ra decay chain can have only one tenth (1/10th) of the total activity, or 200 pCi/g.

As shown in **bold** in the decay chains above, each chain contains a noble gas member, radon. If radon is released from the scale, the subsequent radionuclides in each chain may not be in equilibrium with the parent radium radionuclides. The literature reports average radon emanation fraction of 0.001 to 0.002, which means that it can be assumed that equilibrium conditions exist between all decay chain members (Wilson 1994).

2. MODELS AND DATA

This section describes the modeling approaches, data, and assumptions used to determine the average radioactive contamination per shipment per unit mass based on external exposure rates in contact with pipe or equipment. Models are also described for statistical methods for determining the number of measurements required to characterize a pipe shipment and reduce uncertainty, and for evaluating potential post closure radon emanation rates. The results of the analyses are provided and discussed in Section 3.0, and overall recommendations are provided in Section 4.

2.1 Average Radium Concentrations

The radioactive concentration per shipment of contaminated oil field pipe is a function of the total quantity of radium in the pipe scale and the total weight of the pipe in a shipment. Relevant to this report is the concentration of total radioactivity averaged in each pipe or waste shipment (pipe scale plus pipe), not the concentration of radium in pipe scale alone. That is, the pCi/g of radium averaged between the scale and pipe, for each load of waste. The current Deer Trail waste acceptance criteria receipt of NORM waste that is less than 2,000 pCi/g.

Oil field pipe scale can vary in density over a wide range, and has been reported to vary from 1.7 to 3.5 g/cm^3 (API 1997). For this analysis, it is assumed that the average density of pipe scale is 2.6 g/cm^3. Steel pipe is assumed to have a density of 7.8 g/cm^3. The weight of pipe is a function of the pipe internal diameter and the pipe schedule, which determines the pipe wall thickness. Information from the American National Standards Institute (ANSI) for steel pipes of selected diameters and schedules is shown in Table 2-1 in units of pounds per foot for pipe weight, and inches for wall thickness (ANSI 2009).

Table 2-1. Steel pipe weights and wall thicknesses for selected pipe diameters.

Pipe Size (I.D. inches)	Schedule 40 (lb/ft)	Wall Thickness (inches)	Schedule 80 (lb/ft)	Wall Thickness (inches)	Schedule 160 (lb/ft)	Wall Thickness (inches)
2	3.653	0.154	5.022	0.218	7.462	0.344
3	7.576	0.216	10.25	0.3	14.32	0.437
6	18.97	0.28	28.57	0.432	45.3	0.718
8	28.55	0.322	43.39	0.5	74.69	0.906
10	40.48	0.365	64.33	0.593	115.7	1.125
12	49.56	0.375	88.51	0.687	160.3	1.312

The calculation of average radium activity simply accounts for the mass of scale (for selected scale thicknesses) and the mass of pipe (for selected pipe diameters and schedules). The resulting average concentrations are used as input for the modeling analyses to estimate radiation exposure rates in contact with single or multiple pipe configurations. The details of the analysis conducted using simple spreadsheets are provided in Appendix A.

The relationship between scale activity and total activity to equal 220 pCi/g for selected pipe diameters and schedules, for ^{226}Ra plus its decay chain members is shown in Figure 2-1. As is shown Figure 2-1, when the pipe diameter increases, the total activity needed to equal 220 pCi/g total (pipe plus scale) for ^{226}Ra increases because of the additional pipe mass encountered. In a

similar manner, as the pipe schedule increases, so does the total activity to equal 220 pCi/g total for ^{226}Ra, again because of the increased pipe mass with increased pipe wall thickness.

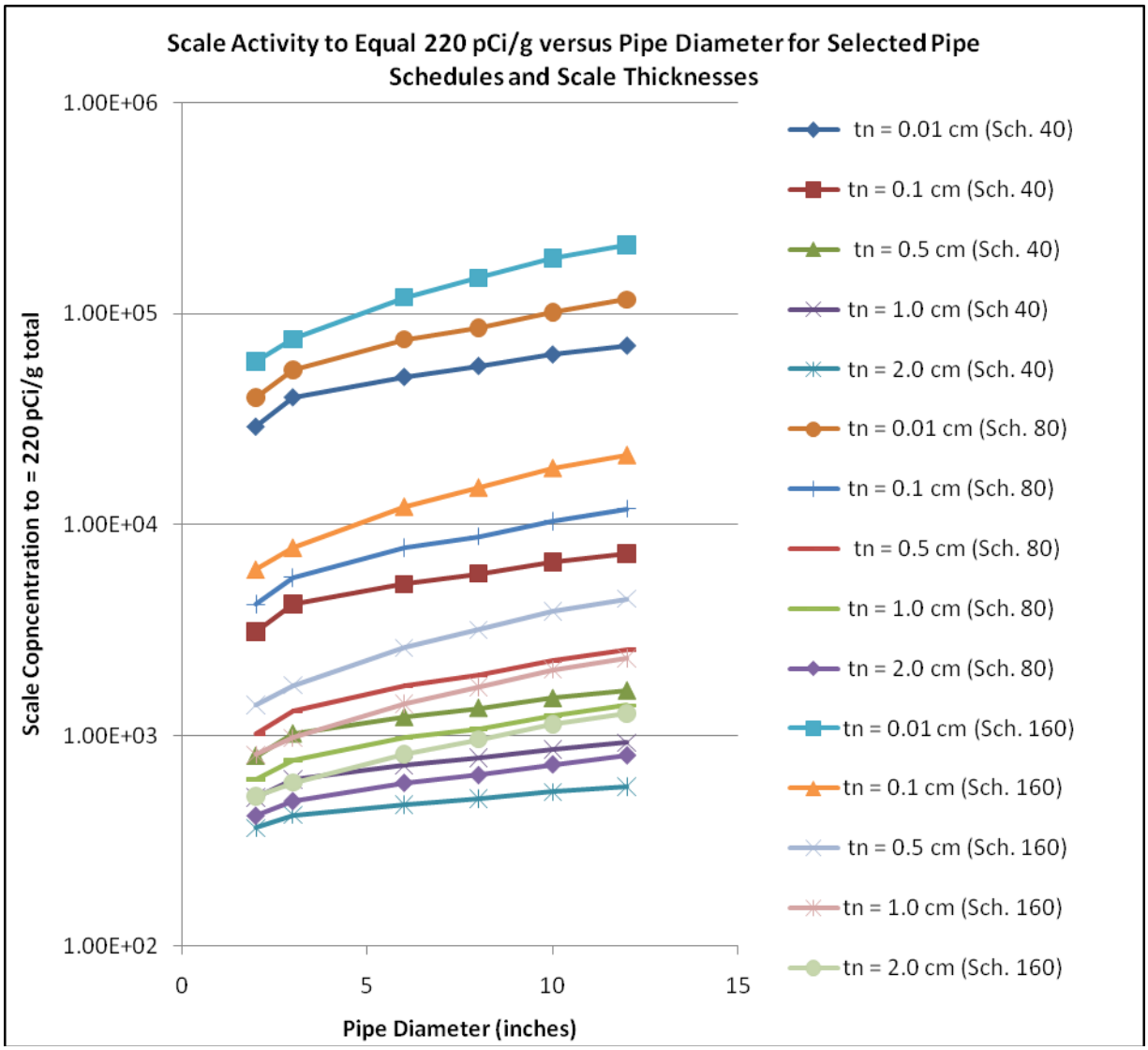

Figure 2-1. Scale activity to equal 220 pCi/g total pipe plus scale activity for selected pipe diameters and schedules: ^{226}Ra.

2.2 Estimated Radiation Exposure Rates

Radiation surveys for the shipment and receipt of oil field pipe and equipment containing radium-contaminated pipe scale for the Deer Trail landfill must rely on external radiation measurements made in the field to confirm that the concentrations encountered meet the waste acceptance criteria. The exposure rate results from two modeling methods are compared to determine the best approach for establishing the external exposure rate to pipe scale radium concentration relationship. The first applies the MicroShield (Grove 2009) computer program, and the second applies an empirical formula derived by the American Petroleum Institute (API 1997) based on measured data.

2.2.1 Modeling Estimates of External Exposure Rates Using MicroShield

Radiation exposure rates are estimated using the MicroShield (Grove 2009) computer program. MicroShield is a widely used and comprehensive photon/gamma-ray shielding and exposure assessment program for designing shields, estimating source strength from radiation measurements, minimizing exposure to people, and as a tool for teaching shielding principles. The program uses a point-kernel calculation with integration parameters to define the size of each source, and takes that amount of activity at that location along with the line of sight distance to the receptors to calculate the receptor dose. The program uses buildup and reduction factors based on the intervening (shield) materials.

MicroShield features include:

Sixteen defined geometries are provided to accommodate offset dose points and as many as 10 standard shields as well as source self-shielding and cylinder cladding.

- Dimensional data are accepted in meters, centimeters, feet, or inches with a rotatable three-dimensional display for viewing and printing;
- Sources can be created, saved, and moved among cases as radionuclides or photon energies, as either individual concentrations or mixture totals. Several photon-grouping methods are provided including user-defined custom groupings.
- Source decay can be calculated with progeny in-growth.
- As many as 25 energy groups ranging from 15 keV to 10 MeV can be used; input can be individual concentrations or mixture totals.
- Sensitivity of exposure rate to time, source dimension, shield thickness, or distance can be investigated.
- Up to six receptors or dose points can be defined for a single case for most geometries.

The standard geometry consisted of a 30 cm length of pipe (cylindrical geometry) with the exposure point 0.1 cm from the midpoint of the pipe (to approximate surface contact exposure rates). The analysis considered the various combinations of pipe diameter, pipe schedule (pipe wall thickness), and scale thickness.

2.2.2 Modeling Estimates of External Exposure Rates Using an Empirical Equation

In November 1997, the American Petroleum Institute (API) published a report entitled *Methods for Measuring Naturally Occurring Radioactive Materials (NORM) in Petroleum Production Equipment* (API 1997). In response to industry concerns regarding the potential concentrations of NORM radionuclides in oil and gas production equipment, the API developed a correlation between radium concentrations in scales and sludge and the external radiation measured with scintillation detectors and Geiger- Mueller (GM) tubes. Their correlation (an empirical equation) was validated with field measurements, and was used to estimate the lowest limits of detection of

radium in piping and equipment. The report provided commentary and experimental data for the characteristics of the field-survey instruments and the NORM distributions that may be encountered. The report recognized that correlations were needed to quantitatively relate survey readings. The correlations were dependent on the NORM scale density, volume, thickness, radionuclide composition, detector efficiency, and geometric efficiency (measurement position). They also depend on the thickness of the equipment wall (i.e., pipe schedule). Actual NORM scale sources from oil fields, and surrogates sources using uranium tailings, were used to obtain measured data to serve as the empirical basis of the correlations. Although separate correlations were developed for thin scales in gas plant equipment and for NORM contamination in exposed surface soils, only the evaluations relevant to oil field piping are considered in this analysis.

The API study benchmarked 159 field-measured data points and recommended the following correlation between measured external radiation and internal NORM concentration for oil field equipment and pipe:

$$C = (0.031/\ t_n\ \rho_n)\ (1 + st_n/4 + 2t_w)\ D$$

Where: t_n = NORM scale thickness
ρ_n = NORM density
s = 2.6 for dense scales
t_w = pipe wall thickness
D = gamma detector count rate (counts/minute).

The detector used in the API study was a 1 by 1 inch sodium iodine (NaI) detector. Solving for D the equation becomes:

$$D = (t_n\ \rho_n/0.031)\ (4 + 2t_w/1 + st_n)\ C$$

The solution for this equation, using the previously calculated values of C (equal to 220 pCi/g total pipe plus scale, derived in Appendix A), are shown in Appendix B. The empirical results are in terms of the detector response in units of counts per minute using a NaI detector. The API study concluded that, although there will be variability in scale distributions and radium concentrations, external measurements represent averages over significant areas (API 1997). They further concluded that external radiation measurements give as accurate an estimate of the average radium concentration compared with a single grab sample (API 1997).

2.3 Evaluation of Multiple Pipe Configurations

The modeling conducted thus far consider exposure rates from a single pipe with radium-contaminated inner scale. The potential importance of multiple pipes on the measured exposure rate for a multiple pipe shipment received at Deer Trail is evaluated using a MicroShield analysis. First, the relative contribution from adjacent contaminated pipes is evaluated for small diameter (two inch) and larger diameter (six inch) pipes, schedule 80, for ^{226}Ra contaminated pipe scale. This analysis was conducted by modeling each adjacent pipe, then summing the contributions of each pipe in the array, with consideration of the shield afforded by the pipe walls between each pipe and the instrument location. The analysis was conducted using a configuration of 28, two inch internal diameter, schedule 80 pipes, and a configuration of 21 six

inch internal diameter, schedule 80 pipes, intended to determine the difference between small and large diameter pipes.

2.4 Statistical Determination of the Required Number of Exposure Rate Measurements

Several sources of parameter and measurement variability that lead to uncertainty in the measured exposure rates have been identified, including:

- Variability of scale layers and distributions within a pipe shipment,
- Variability in the gamma attenuation factors for steel pipe across the incident energy groups,
- Geometric measurement efficiency,
- Variation cause by the instrumentation and the detector efficiency, and
- Measurement variability and errors.

However, the uncertainty produced can be managed my making multiple measurements, and using the average of those measurements in making decisions regarding waste acceptance. Thus, in addition to determining a defensible exposure rate, a sufficient number of measurements must be made for each multiple-pipe shipment to assure that statistically-defendable decisions are made about the average radionuclide concentrations encountered, consistent with the Deer Trail radioactive materials license. A determination of how many measurements are required can be derived through a statistical analysis. The problem of determining average concentrations per shipment, and the number of measurements required to make that determination, is parallel to making release decisions for contaminated property using the Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM) (EPA 2000). Specifically, the exposure rate is comparable to the MARSSIM Derived Concentration Guideline Level (DCGL). The exposure rate is the measurement of the activity/exposure rate relationship corresponding to the Deer Trail waste acceptance criteria. When the specific radionuclide activity is present in background (such as NORM), the Wilcoxon Rank Sum (WRS) test is used. The WRS evaluation is provided in Appendix C. The WRS test relies on managing two types of decision errors: Type I, or α (in our case, accepting waste that exceeds the waste acceptance criteria) and Type II, or β (in our case, rejecting waste that meets the waste acceptance criteria).

2.5 Post Closure Radon Concerns

Finally, an evaluation of potential post-closure radon emanation rates is conducted. Radon flux rate calculations can be made for up to eight different soil and sub-soil layers using the Uranium Mill Tailings Cover Calculator (Wise Uranium Project 2004). For this analysis, the following assumptions are made:

- The waste layer is 4 m thick, with a concentration of 220 pCi/g of ^{226}Ra. This is a conservative concentration since it does not account for the presence of other non-NORM waste or daily soil cover materials that would reduce the average concentration of ^{226}Ra in the waste layer.

- The waste is assumed to have a dry weight porosity of 0.44%, moisture content of 11%, and diffusion coefficient of 2.6×10^{-6} m^2/second (based on ranges of literature values).

- The radon emanation fraction from scale is assumed to be 0.1 (a factor of 100 higher than literature estimates) (Wilson 1994). Note, for pipe scale, diffusion would only occur at the ends of the pipe, which would greatly reduce the average emanation fraction.

- The cover layer thicknesses and properties are consistent with typical RCRA closure designs: 0.15 m of clay, 0.91 m of compressed fill material, and 0.2 m of cover soil.

- The assumed radium concentration in the cover materials (2 pCi/g) is consistent with average soil concentrations in the U.S. (NCRP 2009, Table 3.4).

3. RESULTS AND DISCUSSION

The results of applying the models described in Section 2 for the analysis of exposure rates from oil field pipe and equipment containing radium-bearing pipe scale, using the average scale plus pipe concentrations reported in Appendix A, are presented in this section.

3.1 Single Pipe MicroShield Results

A summary of the MicroShield exposure rate results for the single pipe analysis are shown graphically in Figure 3-1 for selected pipe internal diameters, schedules, and scale thicknesses for ^{226}Ra and ^{228}Ra, using the average concentrations reported in Appendix A. Note that there is a linear relationship between the exposure rate and the radium concentration in scale; that is, an increase in scale concentration will result in a similar, linear increase in the measured exposure rate.

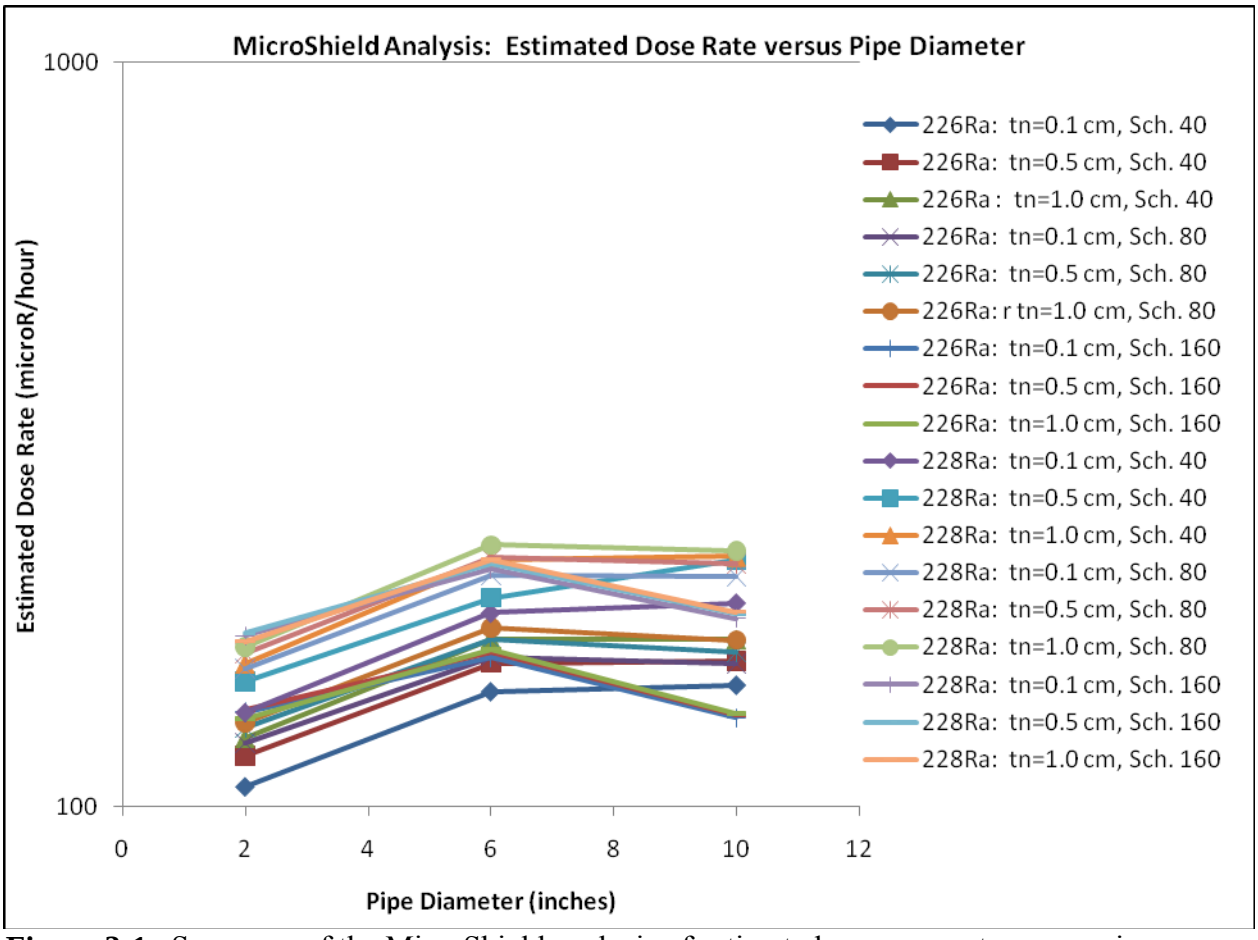

Figure 3-1. Summary of the MicroShield analysis of estimated exposure rate versus pipe diameter for selected pipe diameters, pipe schedules, and scale thicknesses.

This figure shows several features:

- All of the exposure rate results, across pipe diameters, pipe schedules, scale thicknesses, and decay chains are within about a factor of 2; this is close agreement for modeling results,
- Since the activity concentrations for each of these estimates (scale plus pipe) is the same, the theoretical external exposure rates from all pipe scale/pipe combinations should be roughly the same, with some variation induced by geometry effects associated with the pipe diameter and wall thickness,
- For a given pipe diameter the results across all schedules are in close agreement (within about 50%),
- The results for the ^{228}Ra decay chain produce radiation exposures that are about a factor of two greater than comparable exposures for the ^{226}Ra decay chain, and
- For the larger pipe diameters, the estimated exposure rate at a point on the surface at the pipe centerline for the higher pipe schedules decreases, reflecting the geometry effect of additional shielding by the thicker steel walls compared to the increased radius on the modeling results.

3.2 Single Pipe Empirical Equation Results

The empirical equation results obtained from the API study equation, using the detector response (D) values calculated in Appendix B, are next converted to the detector response in terms of exposure rate (μR/hour). The API study indicates that the background count rate for the one-inch NaI detector is 1,504 counts per minute for background, with a background exposure rate of 15 μR/hour. Figure 3-2 shows the empirical exposure rates produced by converting the count rates in Appendix B to exposure rates.

This figure includes the following features:

- For a given pipe diameter the results across all schedules are in close agreement (within about 30%); however, there seems to be some sensitivity to pipe diameter, especially for the lower pipe schedules, and
- The empirical equation includes a mixture of roughly 66% ^{226}Ra decay chain members and 33% % ^{228}Ra decay chain members.

3.3 Multiple Pipe Results

The MicroShield exposure rate modeling results for the 28 pipe configuration of two inch internal diameter, schedule 80 pipes, is shown in Figure 3-3. The relative contribution of each pipe to the exposure rate at the instrument location (at the centerline of the middle pipe on the top row) is shown within each pipe.

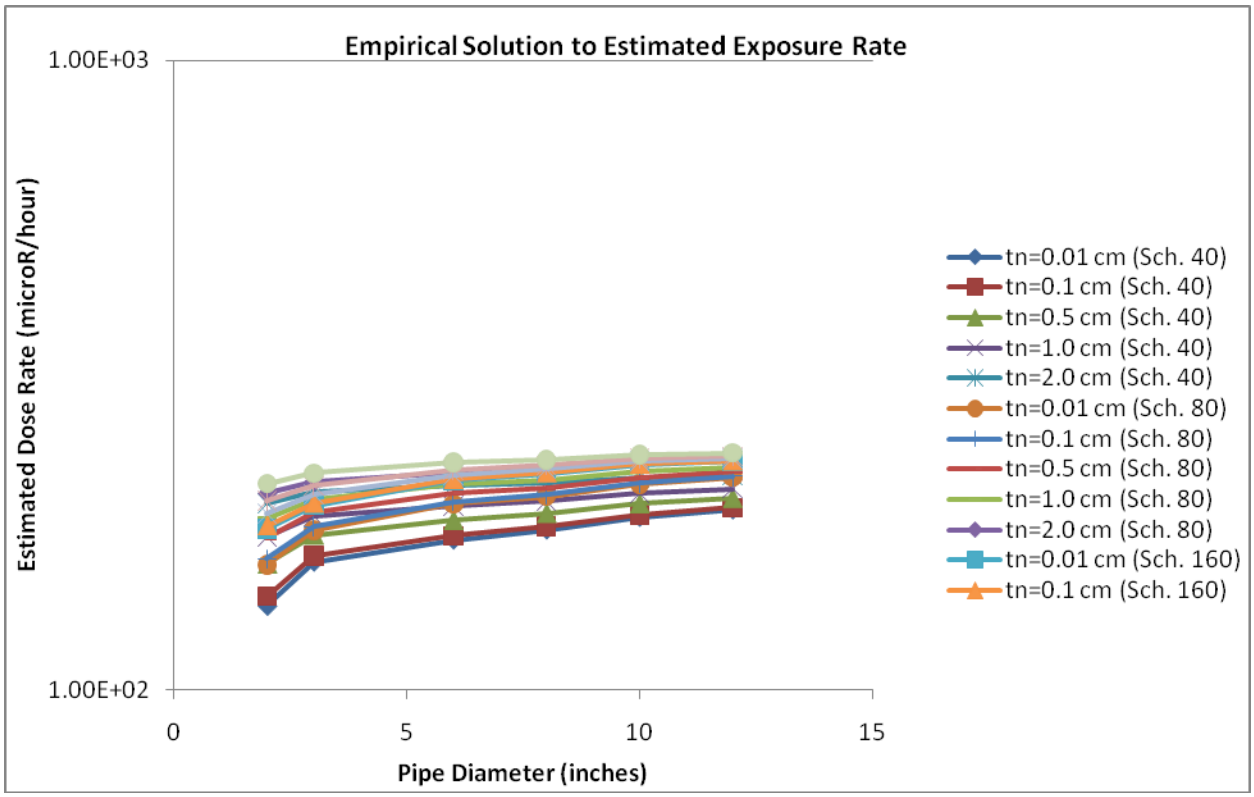

Figure 3-2. Empirical exposure rates for selected pipe diameters, pipe schedules, and scale thicknesses.

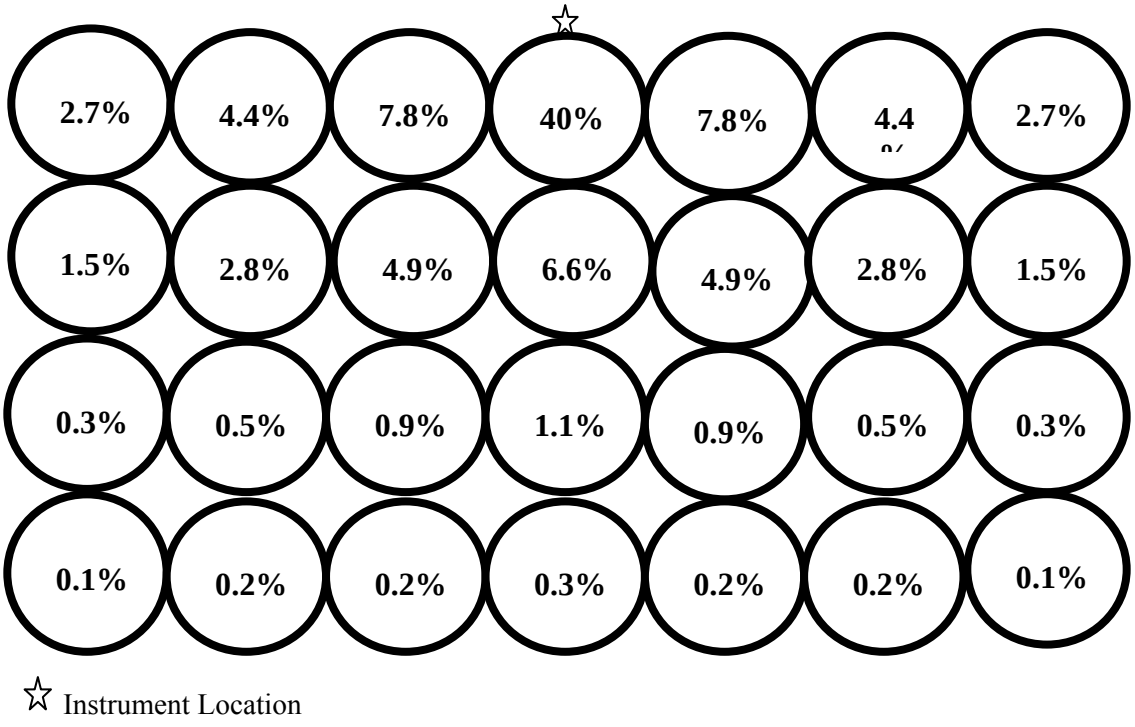

Figure 3-3. Relative contribution to the exposure rate from two inch, Schedule 80 adjacent pipes to the instrument location: ^{226}Ra contaminated pipe scale.

As this figure shows, pipes further away from the instrument location contribute less to the measured exposure rate than closer pipes. It also shows that the instrument reading would need to be increased by 40% to meet the 220 pCi/g criterion if all pipes in the array were uniformly contaminated. Assuming 30 foot lengths of pipe, using the information in Table 1-1, a shipment of 28 pipes would weigh about 2 tons. This analysis was repeated for six inch diameter pipe, using a configuration of 21 pipes, as shown in Figure 3-4.

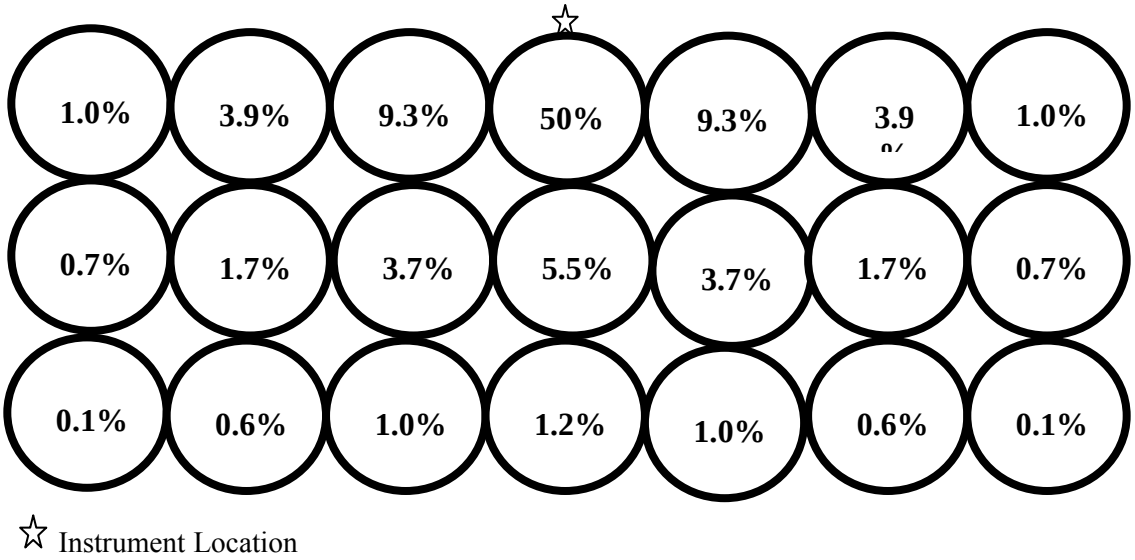

Figure 3-4. Relative contribution to the exposure rate from six inch, Schedule 80 adjacent pipes to the instrument location: ^{226}Ra contaminated pipe scale.

This figure again shows pipes further away from the instrument location contribute less to the measured exposure rate than closer pipes. It also shows that the instrument reading would need to be increased by 50% to meet the 220 pCi/g criterion if all pipes in the array were uniformly contaminated. Assuming 30 foot lengths of pipe, using the information in Table 1-1, a shipment of 21 pipes would weigh about 9 tons.

3.4 Statistical Results for Determining the Number of Measurements Required for a Multiple-Pipe Configuration

The API study concluded that the gamma measurements of pipe and equipment (valves) in the laboratory predicted radium concentrations that averaged within 20-50 percent of the reference values. Although this is an excellent single measurement result, the uncertainty can be reduced by making a number of measurements, and using the average of the results to determine compliance with the waste acceptance criteria (i.e., acceptable exposure rate).

The Wilcoxon Rank Sum (WRS) test was applied, as described in Appendix C, to determine the number of measurements required for a multiple-pipe configuration so that the mean concentration in the survey unit (a shipment of contaminated oil field pipe or equipment) is less than the identified exposure rate. This result would then indicate that the shipment is in compliance with the waste acceptance criteria. The WRS test relies on managing two types of decision errors: Type I, or α (in this case, accepting waste that exceeds the waste acceptance

criteria) and Type II, or β (in this case, rejecting waste that meets the waste acceptance criteria). For this analysis, α and β were selected to be 0.05 (or selecting a 95% confidence limit). The estimate of the mean concentration is the arithmetic average of a set of randomly selected locations for the multiple-pipe shipment. The results of the analysis indicated that 15 total number of measurements (N = measurements for pipe shipment plus background measurements) would be required. However, to assure sufficient data points to attain the desired power level (95% confidence limit) with statistical tests and to allow for possible unusable data, MARSSIM recommends increasing the number of points by 20%, making the calculated value of N equal to 18. Since N equals the number of measurements of the pipe shipment and background, the number for the shipment is one half of the total, or nine (rounding up to the nearest measurement). Thus, the average of nine exposure rate measurements per shipment, compared with the waste acceptance criteria exposure rate limit, provide a statistical basis for characterizing the multiple-pipe configuration for oilfield pipe and equipment.

3.5 Post Closure Radon Concerns

The input/output generated by the Uranium Mill Tailings cover Calculator is shown as:

Uranium Mill Tailings Cover Calculator Input/Output

---------- Input Parameters ----------
Number of Layers: 4
Radon Flux into Layer 1: 0 pCi/m2s
Surface Radon Concentration: 0 pCi/L
Bare Source Flux (Jo) from Layer 1: 77.61 pCi/m^2s
Specific Bare Source Flux from Layer 1: 0.353 pCi/m^2s per pCi_Ra-226/g

Layer No.	Thickness [m]	Ra-226 [pCi/g]	Emanation Fraction	Porosity [dry wt %]	Moisture	Diffusion Coefficient [m^2/s]
1	4	220	0.1	0.44	11	2.600E-6
2	0.15	2	0.3	0.37	8	220.0E-9
3	0.91	2	0.3	0.37	6.3	5.000E-6
4	0.2	2	0.3	0.37	8	1.700E-6

-------- Results of Radon Diffusion Calculation --------

Layer No.	Thickness [m]	Exit Flux [pCi/m^2s]	Exit Concentration [pCi/L]	MIC
1	4	23.83	52.39E3	0.720
2	0.15	20.37	12.93E3	0.728
3	0.91	15.63	5.351E3	0.786
4	0.2	15.68	0E0	0.728

Total cover radon retention: 79.80%

The results indicate a radon exit flux rate of about 16 pCi/m^2s, compared with the EPA uranium mill tailings limit of 20 pCi/m^2s. Again, this result is considered to be conservative since it did not account for mixing of the pipe scale waste with non-NORM waste or with the daily soil

cover, and the limited radon diffusion that would occur only at the ends of pipe. This result indicates that there should be no difficulty from receiving oil field piping and equipment containing radium scale. However, measurements of the radon flux above the cell prior to cell closure will confirm these results, or define the need for a thicker cover to provide an extra diffusion barrier to limit radon emissions.

3.6 Comparison of Results and Discussion

Figure 3-5 shows a comparison of the MicroShield (for ^{226}Ra and ^{228}Ra decay chain members) and empirical equation modeling exposure rate results for each scale thickness and each pipe schedule. For this figure, average exposure rate values were obtained across all internal diameters to show the potential effects of scale thickness and pipe schedule on the results.

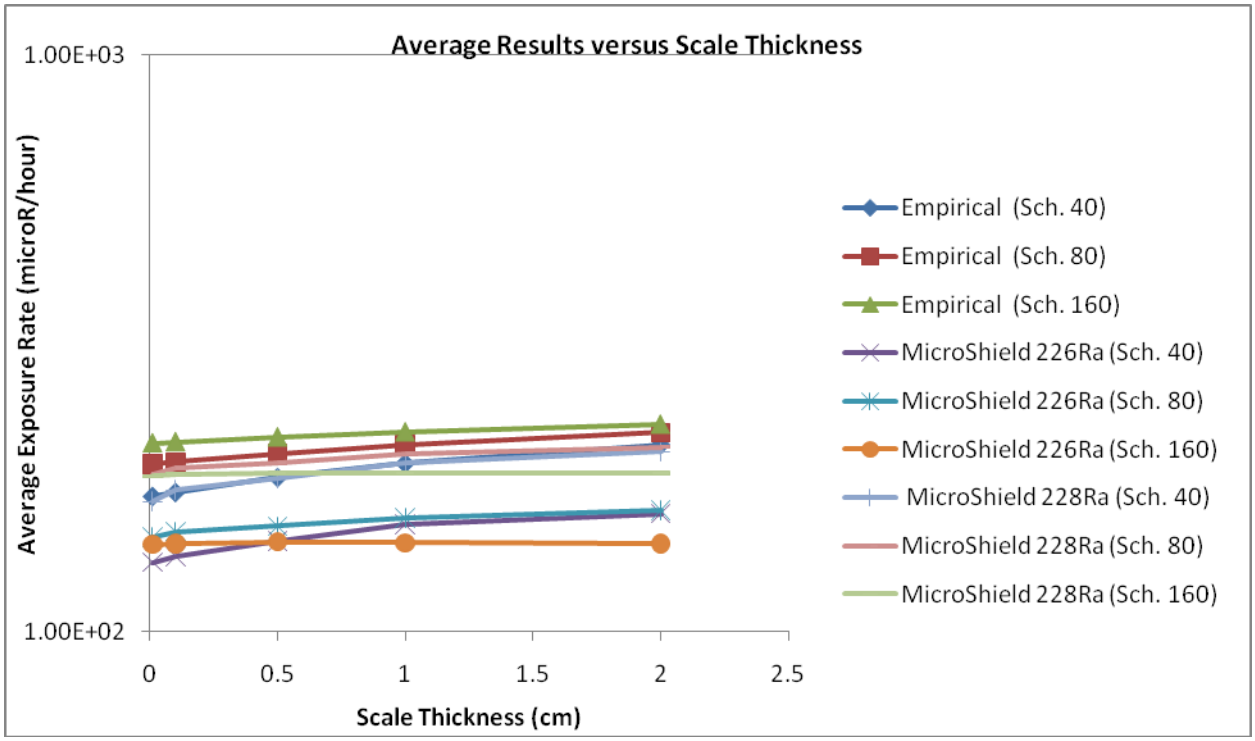

Figure 3-5. Average modeling exposure rate results comparison.

This figure shows the following features:

- All of the modeling exposure rate results are in close agreement, and are generally within about a factor of 1.6, independent of pipe schedule, indicating that scale thickness and pipe schedule are less significant than pipe diameter in estimating external exposures,
- All of the modeling exposure rate results are consistent across all scale thicknesses considered, and
- The empirical modeling exposure rate results are slightly larger than the MicroShield ^{226}Ra results, but almost equal to the MicroShield ^{228}Ra results – this partially reflects the inclusion of ^{228}Ra in the empirical results.

This comparison indicates that there is less variation in estimated exposure rate across scale thickness and pipe schedule compared with the variation produced across the selected pipe diameters (as shown in figures 3-1 and 3-2). The API study also concluded that, based on gamma measurements, the dependence of the scale thickness was low.

4. COMMENTARY AND RECOMMENDATIONS

The purpose of this report is to develop a technical basis for SOPs for radiation surveys prior to shipment and upon receipt of oil field pipe and equipment containing radium-contaminated pipe scale. The goal of this analysis is to determine the relationship between external exposure rates and average radionuclide concentrations per waste shipment in a conservative manner, so that external exposure rate measurements can be used as the basis of the SOPs. The overall approach is to compare modeling exposure rate results, using MicroShield (Grove 2009) and an empirical formula from the literature (API 1997) for selected pipe diameters, schedules, and scale thicknesses. From this comparison, the exposure rates are correlated with internal pipe scale concentrations so that measurements can be used to determine if the Deer Trail waste acceptance criteria are met for specific shipments of oil field pipe and equipment. As previously described, there was close agreement between the MicroShield and empirical modeling contact exposure rate results, which were well within a factor of two for all pipe sizes, schedules, and scale thicknesses. The results also quantified the difference between measurements made on single pipes versus multiple pipes for small and large diameter pipes. The following paragraphs provide final commentary and recommendations for the development of the Deer Trail oil field pipe and equipment SOPs.

4.1 Commentary

The following topics identified during this analysis require further commentary:

- Overall Ranges. The overall ranges of the estimated contact exposure rates for small and large pipes are summarized in Table 4-1 for both the MicroShield and empirical models, on a per pipe basis.

  Table 4-1. Summary of estimated exposure rates for small and large oil field pipes containing radium scale.

Modeling Method	Small Pipe Exposure Rate Range[a] (µR/hour)	Large Pipe Exposure Rate Range[b] (µR/hour)
Empirical – $^{226}Ra/^{228}Ra$ mixture	140 – 220	173 – 240
MicroShield – ^{226}Ra	106 – 131	140 – 160
MicroShield – ^{228}Ra	130 – 170	170 – 250

  a. Pipes with internal diameters < 4 inches.
  b. Pipes with internal diameters > 4 inches.

  These single pipe results can be thought of as the exposure rate per the same average radium concentration in pipe (scale plus pipe). Although all of these results show relatively close agreement, the empirical model exposure rate results are somewhat higher than the results produced by MicroShield. This means that it would be conservative to select the MicroShield model exposure rates as the basis for the SOPs since they would not underestimate the radium concentrations in pipe compared to the empirical results. Also, since actual pipe scale has both ^{226}Ra and ^{228}Ra, it would not be as appropriate to use the most conservative ^{226}Ra results as the sole basis for the SOPs since this value may underestimate the radium concentration when the contamination contains a good proportion of ^{228}Ra (total scale activity is typically about one fourth ^{228}Ra). Based on these considerations, it is concluded that the following external

exposure rates would conservatively indicate pipe scale concentrations of less than the Deer Trail waste acceptance limits: 120 μR/hour for small pipes less than four inches internal diameter (roughly the midpoint of the small pipe MicroShield exposure rate range for ^{226}Ra) and 150 μR/hour for large pipes greater than four inches internal diameter (roughly the midpoint of the MicroShield large pipe exposure rater range for ^{226}Ra). Again, note that there is a linear relationship between the exposure rate and the radium concentration in scale; that is, an increase in scale concentration will result in a similar, linear increase in the measured exposure rate.

- Multiple Pipes. As has been shown, when multiple pipes or equipment and pipes are encountered, the measured exposure rates increase, for the same scale concentration, because of the contributions of the scale in adjacent pipes. This means that for multiple pipe configurations, the small diameter pipe value can be increased by 1.4 times (to 170 μR/hour), and the large diameter pipe value can be increased by 1.5 times (to 230 μR/hour). All measured exposure rates would be exclusive of background. Based on practical considerations for measurements in the field, single pipe measurements would be appropriate for the waste generator to assure that the waste acceptance criteria are met, while multiple pipe measurements would be appropriate for waste received at Deer Trail. A summary of the final technical basis for the Deer Trail SOPs is shown in Table 4-2.

  Table 4-2. External exposure rates to meet the Deer Trail Waste acceptance criteria for oil field pipe and equipment.

Measurement Location and Type	Pipe I.D. < 4 inches (μR/hr at Contact)	Pipe I.D. > 4 inches (μR/hr at Contact)
Generator – Single Pipe	120	150
Receipt – Multiple Pipe	170	230

- "Junk" Shipments. It is most probable that the shipments will resemble shipments of "junk," that is random jumbles of various sizes of pipe, random valves, tanks, and or other equipment. For this situation, it would be appropriate to apply the small diameter pipe exposure rate criteria instead of the large diameter pipe criteria to assure that the waste acceptance criteria are not exceeded.

4.2 Recommendations

Based on the preceding analysis and commentary, the following recommendations are made for the development of the Deer Trail oil field pipe and equipment SOPs:

- The SOPs must account for shipments of small and large diameter pipe, and "junk" shipments of randomly sized and shaped mixtures of pipe and equipment.

- The SOPs must account for both multiple and single pipe or equipment exposure rate measurements as may be encountered in the field.

- For multiple small pipes less than four inches internal diameter, the average of nine exposure rate measurements in contact with the pipe evenly distributed across the shipment must be less than 170 μR/hour. The use of nine measurements would provide

95% confidence that the average radium concentrations for a shipment are less than 2,000 pCi/g.

- For multiple large pipes greater than four inches internal diameter, the average of nine exposure rate measurements in contact with the pipe evenly distributed across the shipment must be less than 230 μR/hour. The use of nine measurements would provide 95% confidence that the average radium concentrations for a shipment are less than 2,000 pCi/g.

- For shipments of junk (a mixture of randomly sized and shaped pipe and equipment), the average multiple small pipe exposure rate measurements in contact with the pipe must be less than 170 μR/hour. The use of nine measurements would provide 95% confidence that the average radium concentrations for a shipment are less than 2,000 pCi/g.

- For single small diameter pipes less than four inches internal diameter, the exposure rate in contact with the pipe must not exceed 120 μR/hour.

- For single large diameter pipes or equipment (valves, tanks, or other equipment) greater than for inches, the exposure rate in contact with the pipe must not exceed 150 μR/hour.

5. REFERENCES

American Petroleum Institute (API). 1997. Methods for Measuring Naturally Occurring Radioactive Materials (NORM) in Petroleum Production Equipment. Publication 7102, 1220 L Street, N.W., Washington, D.C.

Grove Engineering. 2009. *MicroShield, Version 8.0.* Rockville, Maryland.

National Council on Radiation Protection and Measurements (NCRP). 2009. *Ionizing Radiation Exposure of the Population of the United States.* NCRP Report No. 160. Bethesda, Maryland.

Smith, K. P., D. L. Blunt, G. P. Williams, and C. L. Tebes. 1996. Radiological Dose Assessment Related to Management of Naturally Occurring Radioactive Materials Generated by the Petroleum Industry. ANS/EAD-2. Environmental Assessment Division, Argonne National Laboratory, Argonne, Illinois.

U.S. Environmental Protection Agency (EPA). 1993. *DRAFT Diffuse NORM – Waste Characterization and Preliminary Risk Assessment.* Office of Radiation and Indoor Air, Washington, D.C.

U.S. Environmental Protection Agency (EPA). 2000. *Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM).* EPA 402-R-96-117695 (Rev. 1). Accessed at: http://www.epa.gov/radiation/marssim/obtain.html on October 28, 2009.

American National Standards Institute (ANSI). 2009. *Standard Steel Pipe Chart*. Accessed at: http://www.steel-pipes-tubes.com/ansi-chart-steel-pipes.html on October 9, 2009.

Wilson, W. F. 1994. *N O R M, A Guide to Naturally Occurring Radioactive Material.* PennWell Publishing Company, 1421 South Sheridan, P.O. Box 1260, Tulsa, Oklahoma.

APPENDIX A: DETERMINATION OF AVERAGE RADIUM CONCENTRATIONS

The estimated scale activity to equal 220 pCi/g total (scale plus pipe), for ^{226}Ra, and for selected pipe diameters, scale thicknesses, and pipe schedules is shown in Tables A-1 through A-3. The estimated scale activity to equal 200 pCi/g total for scale plus pipe, for ^{228}Ra, and for selected pipe diameters, scale thicknesses, and pipe schedules is shown in Tables A-4 through A-6.

Table A-1. ^{226}Ra activity to equal 220 pCi/g total for selected Schedule 40 pipe diameters.

Pipe Size (I.D. inches), Schedule 40	Scale Thickness = 0.01 cm (pCi/g in Scale)	Scale Thickness = 0.1 cm (pCi/g in Scale)	Scale Thickness = 0.5 cm (pCi/g in Scale)	Scale Thickness = 1.0 cm (pCi/g in Scale)	Scale Thickness = 2.0 cm (pCi/g in Scale)
2	2.91E+04	3.11E+03	7.98E+02	5.09E+02	3.64E+02
3	4.02E+04	4.21E+03	1.02E+03	6.19E+02	4.20E+02
6	5.02E+04	5.22E+03	1.22E+03	7.20E+02	4.70E+02
8	5.67E+04	5.86E+03	1.35E+03	7.84E+02	5.02E+02
10	6.42E+04	6.62E+03	1.50E+03	8.60E+02	5.40E+02
12	7.08E+04	7.27E+03	1.63E+03	9.25E+02	5.73E+02

Table A-2. ^{226}Ra activity to equal 220 pCi/g total for selected Schedule 80 pipe diameters.

Pipe Size (I.D. inches), Schedule 80	Scale Thickness = 0.01 cm (pCi/g in Scale)	Scale Thickness = 0.1 cm (pCi/g in Scale)	Scale Thickness = 0.5 cm (pCi/g in Scale)	Scale Thickness = 1.0 cm (pCi/g in Scale)	Scale Thickness = 2.0 cm (pCi/g in Scale)
2	3.99E+04	4.19E+03	1.01E+03	6.17E+02	4.19E+02
3	5.42E+04	5.62E+03	1.30E+03	7.60E+02	4.90E+02
6	7.55E+04	7.75E+03	1.73E+03	9.73E+02	5.96E+02
8	8.60E+04	8.80E+03	1.94E+03	1.08E+03	6.49E+02
10	1.02E+05	1.04E+04	2.25E+03	1.24E+03	7.29E+02
12	1.17E+05	1.19E+04	2.55E+03	1.39E+03	8.03E+02

Table A-3. ^{226}Ra scale activity to equal 220 pCi/g total for selected Schedule 160 pipe diameters.

Pipe Size (I.D. inches), Schedule 160	Scale Thickness = 0.01 cm (pCi/g in Scale)	Scale Thickness = 0.1 cm (pCi/g in Scale)	Scale Thickness = 0.5 cm (pCi/g in Scale)	Scale Thickness = 1.0 cm (pCi/g in Scale)	Scale Thickness = 2.0 cm (pCi/g in Scale)
2	5.92E+04	6.12E+03	1.40E+03	8.10E+02	5.15E+02
3	7.57E+04	7.77E+03	1.73E+03	9.75E+02	5.97E+02
6	1.20E+05	1.22E+04	2.61E+03	1.41E+03	8.17E+02
8	1.48E+05	1.50E+04	3.17E+03	1.70E+03	9.58E+02
10	1.83E+05	1.85E+04	3.88E+03	2.05E+03	1.13E+03
12	2.11E+05	2.13E+04	4.44E+03	2.33E+03	1.28E+03

Table A-4. ^{228}Ra activity to equal 200 pCi/g total for selected Schedule 40 pipe diameters.

Pipe Size (I.D. inches), Schedule 40	Scale Thickness = 0.01 cm (pCi/g in Scale)	Scale Thickness = 0.1 cm (pCi/g in Scale)	Scale Thickness = 0.5 cm (pCi/g in Scale)	Scale Thickness = 1.0 cm (pCi/g in Scale)	Scale Thickness = 2.0 cm (pCi/g in Scale)
2	2.62E+04	2.80E+03	7.18E+02	4.58E+02	3.28E+02
3	3.61E+04	3.79E+03	9.17E+02	5.57E+02	3.78E+02
6	4.52E+04	4.70E+03	1.10E+03	6.48E+02	4.23E+02
8	5.10E+04	5.28E+03	1.21E+03	7.06E+02	4.52E+02
10	5.78E+04	5.96E+03	1.35E+03	7.74E+02	4.86E+02
12	6.37E+04	6.55E+03	1.47E+03	8.33E+02	5.15E+02

Table A-5. ^{228}Ra activity to equal 200 pCi/g total for selected Schedule 80 pipe diameters.

Pipe Size (I.D. inches), Schedule 80	Scale Thickness = 0.01 cm (pCi/g in Scale)	Scale Thickness = 0.1 cm (pCi/g in Scale)	Scale Thickness = 0.5 cm (pCi/g in Scale)	Scale Thickness = 1.0 cm (pCi/g in Scale)	Scale Thickness = 2.0 cm (pCi/g in Scale)
2	3.59E+04	3.77E+03	9.13E+02	5.55E+02	3.77E+02
3	4.88E+04	5.06E+03	1.17E+03	6.84E+02	4.41E+02
6	6.80E+04	6.97E+03	1.55E+03	8.76E+02	5.37E+02
8	7.74E+04	7.92E+03	1.74E+03	9.70E+02	5.84E+02
10	9.18E+04	9.35E+03	2.03E+03	1.11E+03	6.56E+02
12	1.05E+05	1.07E+04	2.30E+03	1.25E+03	7.23E+02

Table A-6. ^{228}Ra activity to equal 200 pCi/g total for selected Schedule 160 pipe diameters.

Pipe Size (I.D. inches), Schedule 160	Scale Thickness = 0.01 cm (pCi/g in Scale)	Scale Thickness = 0.1 cm (pCi/g in Scale)	Scale Thickness = 0.5 cm (pCi/g in Scale)	Scale Thickness = 1.0 cm (pCi/g in Scale)	Scale Thickness = 2.0 cm (pCi/g in Scale)
2	5.33E+04	5.51E+03	1.26E+03	7.29E+02	4.64E+02
3	6.81E+04	6.99E+03	1.56E+03	8.77E+02	5.38E+02
6	1.08E+05	1.09E+04	2.35E+03	1.27E+03	7.35E+02
8	1.33E+05	1.35E+04	2.86E+03	1.53E+03	8.62E+02
10	1.65E+05	1.67E+04	3.49E+03	1.84E+03	1.02E+03
12	1.90E+05	1.92E+04	4.00E+03	2.10E+03	1.15E+03

APPENDIX B: EMPIRICAL ESTIMATES OF INSTRUMENT RESPONSE

Solving the API empirical equation discussed in Section 2.2.2 for the detector response (D in counts/minute), produces the results in Tables B-1 through B-3 for selected pipe diameters, scale thicknesses, and pipe schedules.

Table B-1. Analytical solution for counts per minute (D) for selected pipe diameters and scale thicknesses – Schedule 40.

Sch. 40 Pipe Diameter (I.D. inches)	D (counts/minute) for t_n=0.01 cm	D (counts/minute) for t_n=0.1 cm	D (counts/minute) for t_n=0.5 cm	D (counts/minute) for t_n=1.0 cm	D (counts/minute) for t_n=2.0 cm
2	1.37E+04	1.41E+04	1.59E+04	1.76E+04	1.98E+04
3	1.60E+04	1.64E+04	1.77E+04	1.89E+04	2.07E+04
6	1.74E+04	1.76E+04	1.87E+04	1.97E+04	2.12E+04
8	1.80E+04	1.82E+04	1.91E+04	2.01E+04	2.14E+04
10	1.89E+04	1.91E+04	1.98E+04	2.06E+04	2.18E+04
12	1.94E+04	1.95E+04	2.02E+04	2.09E+04	2.20E+04

Table B-2. Analytical solution for counts per minute (D) for selected pipe diameters and scale thicknesses – Schedule 80.

Sch. 80 Pipe Diameter (I.D. inches)	D (counts/minute) for t_n=0.01 cm	D (counts/minute) for t_n=0.1 cm	D (counts/minute) for t_n=0.5 cm	D (counts/minute) for t_n=1.0 cm	D (counts/minute) for t_n=2.0 cm
2	1.59E+04	1.62E+04	1.75E+04	1.88E+04	2.06E+04
3	1.80E+04	1.83E+04	1.92E+04	2.01E+04	2.15E+04
6	1.98E+04	2.00E+04	2.06E+04	2.13E+04	2.23E+04
8	2.04E+04	2.05E+04	2.10E+04	2.16E+04	2.25E+04
10	2.13E+04	2.14E+04	2.18E+04	2.23E+04	2.30E+04
12	2.18E+04	2.19E+04	2.23E+04	2.27E+04	2.33E+04

Table B-3. Analytical solution for counts per minute (D) for selected pipe diameters and scale thicknesses – Schedule 160.

Sch. 160 Pipe Diameter (I.D. inches)	D (counts/minute) for t_n=0.01 cm	D (counts/minute) for t_n=0.1 cm	D (counts/minute) for t_n=0.5 cm	D (counts/minute) for t_n=1.0 cm	D (counts/minute) for t_n=2.0 cm
2	1.81E+04	1.83E+04	1.91E+04	2.00E+04	2.13E+04
3	1.97E+04	1.99E+04	2.05E+04	2.12E+04	2.22E+04
6	2.16E+04	2.17E+04	2.20E+04	2.24E+04	2.30E+04
8	2.22E+04	2.22E+04	2.25E+04	2.28E+04	2.33E+04
10	2.29E+04	2.29E+04	2.32E+04	2.34E+04	2.38E+04
12	2.32E+04	2.32E+04	2.34E+04	2.36E+04	2.39E+04

APPENDIX C: STATISTICAL DETERMINATION OF THE NUMBER OF MEASUREMENTS REQUIRED TO CHARACTERIZE RADIONUCLIDE CONCENTRATIONS IN OIL FIELD PIPE AND EQUIPMENT

When the specific radionuclide activity is present in background (such as NORM), the Wilcoxon Rank Sum (WRS) test is used. The WRS evaluation is provided in Appendix C. The WRS test relies on managing two types of decision errors: Type I, or α (in this case, accepting waste that exceeds the waste acceptance criteria) and Type II, or β (in this case, rejecting waste that meets the waste acceptance criteria). First, Data Quality Objectives (DQOs) are set for this measurement situation. The DOQ process is a series of planning steps based on the scientific method for establishing criteria for data quality and developing appropriate survey designs. Early in the process, DQOs are qualitative and quantitative statements derived from the DQO process that clarify the objective, define the most appropriate type of data to collect, determine the most appropriate conditions for data collection, and specify limits on decision errors used as the basis for establishing the quantity and quality of data needed to support the decision. The simple decision rule is: if the mean concentration in the survey unit (a shipment of contaminated oil field pipe or equipment) is less than the identified exposure rate, then the shipment is in compliance with the waste acceptance criteria. The estimate of the mean concentration is the arithmetic average of a set of randomly selected locations. At this point, the "null hypothesis," or baseline condition that is assumed to be true in the absence of strong contradictory evidence, is established. For this problem, the null hypothesis is that the pipe shipment does not meet the waste acceptance criteria (i.e., an unacceptable exposure rate).

The decision based on the survey results can be simplified to a choice between "yes" and "no" as to whether the shipment meets the waste acceptance criteria. There are also two types of errors in this decision given the null hypothesis that the pipe shipment does not meet waste acceptance criteria: 1) incorrectly deciding that the answer is yes (deciding that the waste acceptance criteria is met), when the answer is actually no (a false positive or acceptance of contamination in excess of the waste acceptance criteria, or Type I decision error), and 2) incorrectly deciding that the answer is no (or deciding that the waste acceptance criteria is not met), when the answer is actually yes (a false negative or not accepting a shipment when in fact the levels meet the waste acceptance criteria, or Type II decision error). While the possibility of a decision error cannot be totally eliminated, it can be statistically controlled.

The number of total data points, N (reference plus survey area), for the WRS test is obtained for each reference area/survey unit pair (i.e., measurement/background pair) using the following equation:

$$N = (Z_{1-\alpha} + Z_{1-\beta})^2/3(P_r-0.5)^2$$

Where: $Z_{1-\alpha}$ = percentile represented by the selected decision error level α (Type I Error),
$Z_{1-\beta}$ = percentile represented by the selected decision error level β (Type II error), and
P_r = the probability that a measurement performed at a random location in the survey unit is larger than a random background measurement.

Inherent to this statistical determination is the grey region; or more specifically, the lower bound of the grey region, or LBGR. It is necessary to specify a grey region because variability in the parameter of interest and unavoidable imprecision in the measurement system combine to produce variability in the data so that a decision may be very difficult when the true but unknown value of the parameter is very near the waste acceptance criteria. For this situation, the LBGR is a exposure rate that is less than the exposure rate waste acceptance criteria, and is chosen to be easily distinguishable from the criteria. The grey region lies between the LBGR and the waste acceptance criteria in which only Type II decision errors occur (i.e., a false negative or not accepting a shipment when in fact the levels meet the waste acceptance criteria). The width of the grey region, or the difference between the waste acceptance criteria and the lower bound of the grey area of the decision, is referred to as the shift, or Δ. For this analysis the lower bound of the grey area is assumed to be half of the total waste acceptance criteria exposure rate, or 63 µR/hour; a reading that should be easily distinguished from the waste acceptance criteria. The shift and the estimated standard deviation of the measures of the exposure rate (σ) are used to calculate the relative shift (Δ/σ). For this analysis, the total contact exposure rate is 125 (source plus background) µR/hour, and background is 15 µR/hour. As indicated in Section 5.5.2.2 of MARSSIM (2000), when preliminary measurements are not available, it may be reasonable to assume a relative standard deviation on the order of 30%, based on experience. For this analysis, the estimated standard deviation of the survey measurement is taken as 30%. Thus,

$$\Delta/\sigma = (125\text{-}63)/(63 \times 0.3) = 62/19 = 3.28$$

From Table 5.1 of MARSSIM, when $\Delta/\sigma \sim 3.3$, $P_r \sim 0.99$. Solving for N, using $Z_{1-\alpha}$ ($\alpha = 0.05$; 5% decision error) = 1.645, and $Z_{1-\beta}$ ($\beta = 0.05$; 5% decision error) = 1.645, and assuming $P_r = 0.99$:

$$N = (1.645 + 1.645)^2/3(P_r\text{-}0.5)^2$$
$$N = (1.645+1.645)^2/3(0.99\text{-}0.5)^2 = 10.8/0.72 = 15$$

To assure sufficient data points to attain the desired power level with statistical tests and to allow for possible unusable data, MARSSIM recommends increasing the number of points by 20%, making the calculated value of N equal to 18. Since N equals the number of measurements of the shipment and background, the number for the shipment is one half of the total, or nine (rounding up to the nearest measurement). Thus, the average of nine exposure rate measurements, compared with the waste acceptance criteria exposure rate limit, will be used in the SOP for oilfield pipe and equipment.

Memorandum

Date: February 10, 2010

Subject: Technical Basis for Receipt Surveys for Radium-Scale Materials

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This memorandum serves to document the technical basis for exposure rate limits associated with the receipt of radium-scale containing wastes at the Clean Harbors Deer Trail (CHDT) facility.

A technical basis document was prepared by Dade Moeller and Associates (DMA) to describe the correlation between activity concentrations within contaminated piping and external exposure rates. However, the DMA guidance assumed that piping or other contaminated equipment would be transported without being containerized. In practice, this means of transport is unlikely; it is more reasonable to assume that contaminated piping and equipment would be placed in a roll-off container or end-dump truck for transport.

To prevent cross contamination and unnecessary handling of the contaminated materials, it is desirable for CHDT personnel to conduct verification surveys on the outside of shipping containers or received radium-scale wastes. Additional calculations are required to account for the attenuation of the gamma emissions from the radium scale by the shipping container, as well as from the buildup of scattered radiation. A range of measurements to account for the effects of shielding will be used by CHDT personnel to confirm that radiation levels on incoming radium-scale waste shipments are within CHDT waste acceptance criteria.

Calculations

Equation 10.17 of *Introduction to Health Physics* (Cember 1996) is as follows:

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

where:

I	=	gamma-ray intensity transmitted through an absorber of thickness x
B	=	buildup factor
I_O	=	gamma-ray intensity at zero absorber thickness
e	=	base of the natural logarithm system
μ	=	linear attenuation coefficient, cm^{-1}
x	=	thickness of absorber, cm

Both the buildup factor and the linear attenuation coefficient are functions of the absorber material and the incident gamma energy. Since the radium-containing materials evaluated using this the procedure are mostly in equilibrium with the gamma-emitting progeny, a weighted

average approach was used to determined the resulting beam intensity after interactions with two types of container materials – aluminum and steel (modeled as iron).

The gamma fluence modeling software Microshield was used to determine the energy composition of the gamma emissions from materials containing 75-percent radium-226 and 25-percent radium-228 following a one-year ingrowth period. The breakdown of gamma energies is shown in Table 1.

Table 1. Relative Contribution of Gamma Energies Associated with Radium Scale

Energy (MeV)	Percentage Contribution
0.04	0.003%
0.05	0.022%
0.06	0.007%
0.08	0.965%
0.1	0.165%
0.15	0.145%
0.2	1.794%
0.3	3.821%
0.4	5.794%
0.5	1.326%
0.6	12.148%
0.8	6.317%
1	25.242%
1.5	14.416%
2	19.562%
3	8.271%

As noted above, both the linear attenuation coefficient and buildup factors are functions of the incident energy. Published linear attenuation coefficients were found in Table 5.2 of Cember 1996, and additional values were found by graphing and extrapolating from the published values. The buildup factors aluminum and iron were obtained from Table 6.5.1, Exposure Buildup Factors, of the *Handbook of Health Physics and Radiological Health*, Third Edition (Shleien et al 1998). Thicknesses of 0.160 inches of aluminum (found to be a standard thickness for aluminum end-dump trailers) and 0.1345 inches of steel (equal to 10-gauge steel) were used in the calculation.

Using the linear attenuation coefficients, buildup factors, and relative contributions of gamma energies to the overall exposure, an overall intensity reduction (I/I_O) value was obtained for both aluminum and steel containers. Using the shielding intensity reduction, as well as accounting for the buildup of scattered radiation, exposure rate limits for measurements on the outside of the shipping containers were calculated and are presented in Table 2. It is assumed that the

containers are full, and that the confirmation measurements will be collected on or near contact with the outside of the shipping container.

Table 2. Container Exposure Rate Limits

	Non-Containerized Limit (µR/hr) [1]	Aluminum Container (µR/hr) [2]	Aluminum Container w/ Buildup (µR/hr) [2]	Steel Container (µR/hr) [2]	Steel Container w/ Buildup (µR/hr) [2]
Multiple < 4 inches	170 [3]	160	230	140	200
Multiple > 4 inches	230 [3]	210	320	190	280
Multiple Debris	170 [3]	160	230	140	200
Single < 4 inches	120 [4]	110	160	100	140
Single > 4 inches	150 [4]	140	210	130	180
Single Debris	120 [4]	110	160	100	140

1. µR/hr = microroentgens per hour.
2. Values are rounded to the nearest multiple of ten.
3. The generator limit for multiple pipes/pieces of debris is the average of nine measurements.
4. The generator limit for single pipes/pieces of debris is the maximum of nine measurements.

For example, based on the calculation, a shipment of pipe less than four inches in diameter determined by the generator to be in compliance with the CHDT acceptance criteria in a multiple pipe configuration with an average of 170 uR/hr or less, would be expected to have an average exposure rate of 160 uR/hr to 230 uR/hr (when accounting for the buildup of scattered radiation). Acceptance at the CHDT facility requires the collection and documentation of measurements within the limits listed on Table 2.

Conclusions

The limits shown in Table 2 will be used by CHDT personnel to verify that shipments of radium scale wastes meet the CHDT waste acceptance criteria. CHDT confirmation measurements will be collected with a Ludlum Model 19 exposure rate meter. Any measurements exceeding the ranges listed in Table 2 may require the shipment to be unloaded from the shipping container for further surveys, or may result in rejection of the load.

References

Cember 1996. *Introduction to Health Physics*, Third Edition. McGraw-Hill Health Professions Division, New York.

Schleien et al 1998. *Handbook of Health Physics and Radiological Health*, Third Edition. Williams & Wilkins, Baltimore, MD.

Memorandum

Attachment 1 – Intensity Calculations for Aluminum Container

Energy (MeV) [1]	% Energy Activity	μ (cm^{-1}) [2]	x (cm) [3]	I/Io (without buildup)	I/Io - weighted average	Buildup Factor [4]	I/Io (with buildup)	I/Io with buildup-weighted average
0.04	0.00%	0.489	0.41	0.82	0.00	2.04	1.672	0.00
0.05	0.02%	0.467	0.41	0.83	0.00	2.00	1.655	0.00
0.06	0.01%	0.450	0.41	0.83	0.00	1.97	1.641	0.00
0.08	0.97%	0.422	0.41	0.84	0.01	1.92	1.617	0.02
0.1	0.17%	**0.435**	0.41	0.84	0.00	**1.91**	1.601	0.00
0.15	0.15%	**0.362**	0.41	0.86	0.00	1.81	1.564	0.00
0.2	1.79%	**0.324**	0.41	0.88	0.02	1.76	1.545	0.03
0.3	3.82%	**0.278**	0.41	0.89	0.03	1.69	1.511	0.06
0.4	5.79%	0.266	0.41	0.90	0.05	1.64	1.474	0.09
0.5	1.33%	**0.227**	0.41	0.91	0.01	**1.57**	1.432	0.02
0.6	12.15%	0.226	0.41	0.91	0.11	1.57	1.434	0.17
0.8	6.32%	**0.185**	0.41	0.93	0.06	1.52	1.413	0.09
1	25.24%	**0.166**	0.41	0.93	0.24	1.45	1.355	0.34
1.5	14.42%	**0.135**	0.41	0.95	0.14	1.41	1.339	0.19
2	19.56%	**0.117**	0.41	0.95	0.19	**1.37**	1.306	0.26
3	8.27%	**0.096**	0.41	0.96	0.08	**1.33**	1.279	0.11
					0.93			**1.37**

1. MeV = mega electron volts.
2. cm^{-1} = per centimeter. Bolded values are published values from Table 5.2 of Cember 1996. Non-bolded values were determined by extrapolation .
3. cm = centimeter. An aluminum thickness of 0.160 inches (0.41 centimeters) was assumed.
4. Buildup factors from Table 6.5.1 of Schleien et al 1998, for mean free path (equal to the product of μx) of 0.5. Bolded values are published values. Non-bolded values were determined by extrapolation.

Memorandum

Attachment 2 – Intensity Calculations for Steel Container

Energy (MeV) [1]	% Energy Activity	μ (cm^{-1}) [2]	x (cm) [3]	I/Io (without buildup)	I/Io - weighted average	Buildup Factor [4]	I/Io (with buildup)	I/Io with buildup-weighted average [5]
0.04	0.003%	3.330	0.34	0.32	0.00		0.000	0.00
0.05	0.022%	2.912	0.34	0.37	0.00		0.000	0.00
0.06	0.007%	2.610	0.34	0.41	0.00		0.000	0.00
0.08	0.965%	2.196	0.34	0.47	0.00		0.000	0.00
0.1	0.165%	**2.720**	0.34	0.39	0.00	**1.26**	0.498	0.00
0.15	0.145%	**1.445**	0.34	0.61	0.00		0.000	0.00
0.2	1.794%	**1.090**	0.34	0.69	0.01		0.000	0.00
0.3	3.821%	**0.838**	0.34	0.75	0.03		0.000	0.00
0.4	5.794%	0.836	0.34	0.75	0.04		0.000	0.00
0.5	1.326%	**0.655**	0.34	0.80	0.01	**1.48**	1.183	0.02
0.6	12.148%	0.655	0.34	0.80	0.10		0.000	0.00
0.8	6.317%	**0.525**	0.34	0.84	0.05		0.000	0.00
1	25.242%	**0.470**	0.34	0.85	0.21	**1.41**	1.201	0.30
1.5	14.416%	**0.383**	0.34	0.88	0.13		0.000	0.00
2	19.562%	**0.335**	0.34	0.89	0.17	**1.35**	1.204	0.24
3	8.271%	**0.285**	0.34	0.91	0.08	**1.32**	1.198	0.10
					0.84			**1.20**

1. MeV = mega electron volts.
2. cm^{-1} = per centimeter. Bolded values are published values from Table 5.2 of Cember 1996 for iron. Non-bolded values were determined by extrapolation .
3. cm = centimeter. A steel thickness of 0.1345 inches (0.34 centimeters) was assumed.
4. Buildup factors from Table 6.5.1 of Schleien et al 1998, for mean free path (equal to the product of μx) of 0.5. Bolded values are published values. Based on the distribution of published values, additional values could not be extrapolated.
5. Only the gamma energies with published buildup factors were used in the determination of the weighted average.

TECHNICAL MEMORANDUM

Alternate Analysis Technique for Radium-226 to Assure Compliance with Disposal Criteria at Clean Harbors Deer Trail Facility

To: Jack Kehoe, Facility General Manager
Clean Harbors Deer Trail

From: James M. Langsted, CHP
CB&I Federal Services

Date: March 12, 2015

1.0 Introduction

This document details the technical basis for using direct counting of the 186 keV photo peak from the decay of Ra-226 as a rapid counting technique for verification of meeting the Clean Harbors Deer Train (CHDT) waste acceptance criteria.

Colorado Department of Public Health and Environment (CDPHE) Radioactive Materials License Colo. 1102-01 (License Condition 10.A) requires that the summed activity of all radionuclides contained in waste materials received at the CHDT facility shall not exceed 2000 pCi/g, and additionally that the radium-226 (Ra-226) activity shall not exceed 222 pCi/g. To assure these requirements are met, CHDT requires pre-acceptance laboratory analysis of waste samples and periodic random confirmation sampling as the waste arrives at the facility (15.WAC.01).

2.0 Ra-226 Analysis by Gamma Spectroscopy of Decay Progeny

Radionuclides are analyzed by a number of acceptable test methods, specified in Table 1, Attachment D of 15.WAC.01. Ra-226 is typically analyzed by EPA Method 901.1, modified for a solid (soil) matrix. To perform this analysis, the sample is sealed in a standard-geometry container and decayed for 21 days, allowing the ingrowth of Ra-226 decay progeny to reach approximate secular equilibrium. Bismuth-214 (Bi-214) or lead-214 (Pb-214) is then quantified by gamma spectroscopy counting to determine the concentration of Ra-226 in the sample. Other gamma-emitting radionuclide concentrations are also quantified by the same gamma spectroscopy analysis.

The 21-day ingrowth requirement for this method results in a significant delay in determining the Ra-226 concentration and delays the determination if the waste meets the CHDT disposal

criteria. This delay places a burden on both the waste generator and the Deer Trail disposal facility, delaying appropriate disposition of the waste.

3.0 Alternate Analysis Method

There are a number of techniques that provide an alternative to Ra-226 analysis by gamma spectroscopy of decay progeny. 15.WAC.01 allows the use of alternate methods of analysis with prior written approval of both Adams County and the CDPHE (Section 6.b).

A useful alternative analysis technique is to directly quantify Ra-226 by analysis of the 186.1 keV peak. The technique is evaluated in a paper written by T. Romanko (2015). The technique has a number of advantages and disadvantages discussed here. This method provides timely analysis results, in many cases is able to verify that the Ra-226 concentration of received waste meets the radioactive materials license requirements, meets the facility waste acceptance criteria, and is appropriate for placement in the CHDT disposal cell. If the result from this analysis technique is inconclusive, it is possible to continue the 21-day Ra-226 decay without further preparation of the sample and then perform gamma spectroscopy in accordance with Method 901.1.

Romanko cites the utility of this method as a relatively quick screening technique if the potentially high-biased Ra-226 result doesn't exceed a pre-defined action level, the amount of Ra-226 compared to U-235 is relatively large, and the action level is well above background levels. These criteria are met for the oil and gas field waste streams accepted at CHDT.

Ra-226 emits a 186.1 keV photon during 3.59% of its decays. All other photons emitted from Ra-226 are of insignificant quantities. Unfortunately, one of the naturally occurring isotopes of uranium (U-235) emits a photon very close in energy to the Ra-226 photon. The U-235 emission at 185.7 keV occurs in 57.2% of its decays. These energies are close enough together that modern gamma spectroscopy systems cannot resolve these two peaks and will count them together. Thus, the presence of natural uranium in the sample will bias (high) the Ra-226 result. If the result of this analysis is below the waste acceptance criteria, the result verifies that the waste is acceptable (with respect to Ra-226 concentration) for disposal. If the result is above the waste acceptance criteria, completion of the 21-day decay and Method 901.1 analysis can be completed to remove the influence of U-235 on the result and determine if the Ra-226 concentration is in excess of the waste acceptance criteria.

Detector drift at the lower (186 keV) energy and the execution of the peak search and background subtraction routines on the gamma spectrometry data do result in a greater variability in the reported result. In most cases both the Ra-226 peak and the U-235 peak are counted together and credited to Ra-226, but infrequently these variations result in

underreporting of the Ra-226 concentration. Tucker, et. al. found an average error on the 186 keV peak of 45.4%.

This alternate method will avoid the U-235 interference when applied to waste materials with less natural uranium. This is typical of oil and gas field waste materials where because of the geochemistry of these materials and the radium content of these materials is significantly higher than the uranium content. The average activity concentration of U-235 in the oil and gas field wastes that have been received at CHDT is 1.2% of the Ra-226 activity concentration. Romanko documents a study in which twenty oil field-contaminated soil samples were analyzed by both the alternate method (186 keV peak analysis) and Method 901.1 (21-day decay). The results demonstrate that alternate method safely overestimates the Ra-226 concentration in all but two low-concentration samples that are 250 times below the CHDT waste acceptance criteria. The alternate method MDC is at worst case 33 times lower than the waste acceptance criteria. An analysis of 28 filter cake samples from the oil and gas industry shows a similar utility of this technique. The danger of a false negative mistake (determining that a sample is below the waste acceptance criteria when it in fact is above the criteria), is very low.

Another disadvantage of the alternative analysis method is that the minimum detectable concentration (MDC) for this analysis method is higher than the other analysis, but it is well below the license limit of 222 pCi/g.

4.0 Evaluation of Appropriateness

An evaluation of this alternate technique has been performed using data from the oil and gas field waste streams that have been accepted at CHDT. The interference from the U-235 emissions has been considered and demonstrates that that this will not be a concern.

The average Ra-226 concentration results, routinely measured by gamma spectroscopy using 21-day decay and the total uranium results measured by the submission of samples to an outside radiochemistry laboratory were evaluated. These averages include the pre-acceptance samples and random confirmation samples routinely collected as these waste streams were accepted at the facility. The results are shown in Table 1. The natural uranium ratio of U-238 to U-235 (99.2745% to 0.720%, by weight) was applied to estimate the concentration of U-235. The decay ratios for the U-235 and the Ra-226 186 keV photons (57.2% and 3.59%) were used to estimate the photon emission rate. It was assumed that both photons are counted at the “186 keV” count and all is attributed to the presence of Ra-226. Figure 1 shows a comparison of the estimated count, including the interference from U-235, and the Ra-226 that was measured using the 21-day decay method. In all cases, the overestimate caused by the U-235 interference is not significant at a level that would affect the determination of compliance.

Romanko indicates a Total Propagated Uncertainty (TPU) of 20% - 40% in the range of 3 - 11 pCi/g Ra-226 (111 – 407 Bq/kg) measured by this technique. There is no data that extends beyond 30 pCi/g (1100 Bq/kg), however. To maintain a reasonable margin of safety, if values in excess of 150 pCi/g are reported by this alternate method, analysis by other methods should be performed to validate the result.

5.0 Conclusion

Use of direct counting of the 186 keV photo peak from the decay of Ra-226 as a rapid alalysis technique to verify that waste meets the Clean Harbors Deer Train (CHDT) waste acceptance criteria is practical for wastes originating from oil and gas fields. This alternate analysis method provides a rapid analysis method which can be followed up, if necessary, by additional analysis to gain verify the analysis performed.

6.0 References

15.WAC.01, *Radioactive Materials Acceptance*, Revision 8, Clean Harbors Deer Trail LLC, March 31, 2015.

Colo. 1102-01, Radioactive Materials License, State of Colorado, Colorado Department of Health and Environment, Amendment Number 13, December 21, 2012.

EPA 901.1, *Gamma Emitting Radionuclides in Drinking Water*, Prescribed Procedures for Measurement of Radioactivity in Drinking Water, U.S. Environmental Protection Agency, EPA-600/4-80-032, August 1980.

Romanko, T., *^{226}Ra: An Inter-Method Comparison*, Romanko, Terry, TestAmerica Laboratories, Inc., Paper number 15447, WM2015 Conference, March 15-19, 2015, Phoenix, Arizona, USA.

Tucker, et. al., *Comparison of Activity Determination of Radium 226 in FUSRAP Soil using Various Energy Lines*, Brian Tucker, Jough Donakowski, and David Hays, Paper number 12299, WM2012 Conference, February 26 – March 1, 2012, Phoenix, Arizona, USA.

7.0 Attachments

Romanko, T., *^{226}Ra: An Inter-Method Comparison*, Romanko, Terry, TestAmerica Laboratories, Inc., Paper number 15447, WM2015 Conference, March 15-19, 2015, Phoenix, Arizona, USA

.

Figure

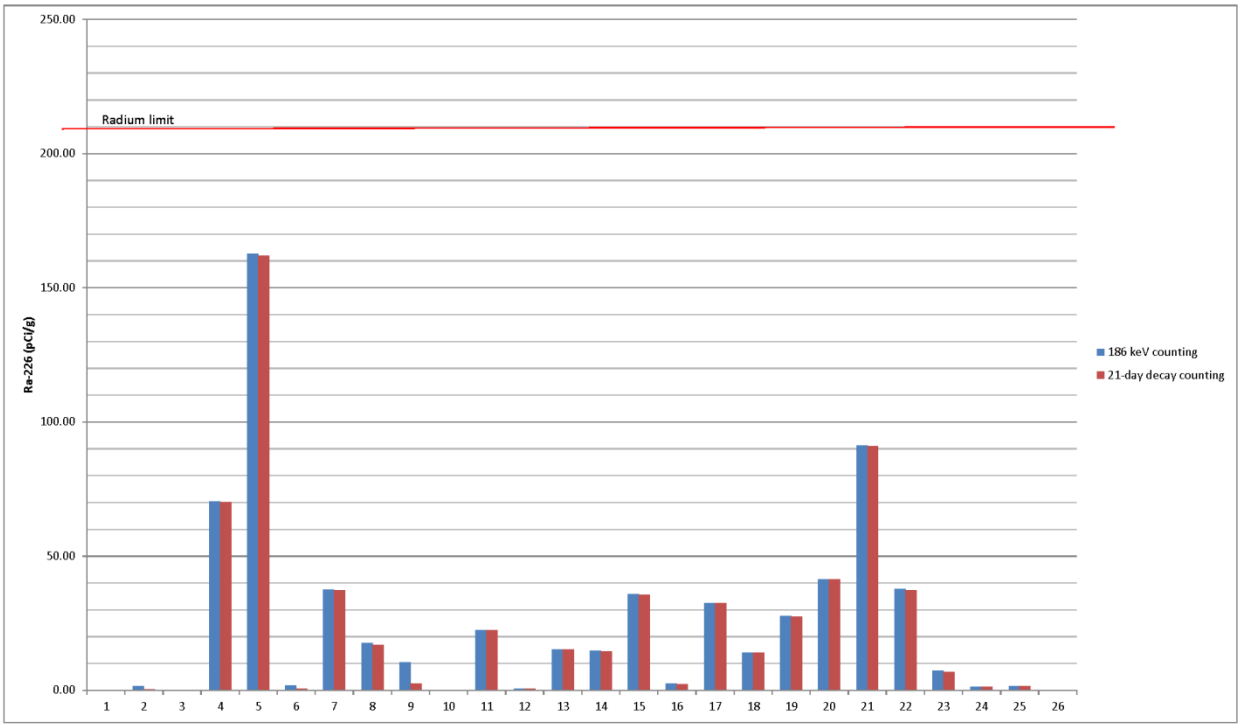

Figure 1: CHDT Waste Streams - Comparison of Analysis Methods

Table

Table 1: Estimated Ra-226 (including U-235 interference)

Waste Description	U-238 Avg. Conc. pCi/g	U-234 Avg. Conc. (pCi/g)	Th-230 Avg. Conc. (pCi/g)	Ra-226 Avg. Conc. (pCi/g)	Pb-210 Avg. Conc. (pCi/g)	Th-232 (Ra-228) Avg. Conc. (pCi/g)	U-235 estimate U-235 (pCi/g)	U-235 185.7 keV (γ/s)	Ra-226 186.1 keV (γ/s)	Ra-226+ U-235 "186" keV (γ/s)	Ra-226 estimate (pCi/g)
Filters w/ Pb-210	0.00	0.00	0.00	0.00	177.40	0.00	0.00	0.00	0.00	0.00	0.00
Pipeline sludge and water	1.50	2.50	0.30	0.40	266.00	0.50	0.07	5.14	1.91	7.05	1.48
Filters w/ Pb-210	0.00	0.00	0.00	0.00	228.90	0.00	0.00	0.00	0.00	0.00	0.00
Oil exploration tank sludge	0.50	0.50	0.50	70.00	70.00	0.20	0.02	1.71	334.73	336.45	70.36
Sludge from oil/gas processing	1.00	1.00	1.00	162.00	162.00	1.00	0.04	3.43	774.66	778.09	162.72
Pigging Sludge With Arsenic and Lead 210 (NORM)	1.90	1.10	3.50	0.50	152.00	1.70	0.09	6.51	2.39	8.90	1.86
TENORM from Oil Processing	0.51	0.51	0.51	37.23	37.23	15.29	0.02	1.75	178.03	179.78	37.60
LBU Tank TENORM from Oil Production	1.00	1.00	1.00	17.00	17.00	0.80	0.04	3.43	81.29	84.72	17.72
NORM Sludge	11.02	25.60	0.14	2.51	480.00	1.76	0.50	37.77	12.00	49.78	10.41
NORM PPE	0.05	0.09	0.01	0.03	339.00	0.01	0.00	0.17	0.14	0.31	0.07
TENORM from Oil Processing	0.22	0.22	0.22	22.30	22.30	9.28	0.01	0.75	106.64	107.39	22.46
Spent Natural Gas Descant	0.08	0.08	0.08	0.51	46.53	0.64	0.00	0.27	2.44	2.71	0.57
TENORM Filtercake from Oil Processing	0.30	0.30	0.30	15.10	16.16	7.18	0.01	1.03	72.21	73.23	15.32
TENORM Filter Socks	0.04	0.04	0.04	14.60	14.60	10.15	0.00	0.14	69.82	69.95	14.63
NORM/TENORM Desorber Solids	0.39	0.39	0.39	35.58	35.58	13.63	0.02	1.34	170.14	171.48	35.86
TENORM Wastewater Filters	0.25	0.25	0.25	2.28	13.20	0.43	0.01	0.86	10.90	11.76	2.46
TENORM Filter Socks from Petroleum Production	0.02	0.02	0.02	32.50	32.50	19.25	0.00	0.07	155.41	155.48	32.51

Waste Description	U-238 Avg. Conc. pCi/g	U-234 Avg. Conc. (pCi/g)	Th-230 Avg. Conc. (pCi/g)	Ra-226 Avg. Conc. (pCi/g)	Pb-210 Avg. Conc. (pCi/g)	Th-232 (Ra-228) Avg. Conc. (pCi/g)	U-235 estimate U-235 (pCi/g)	U-235 185.7 keV (γ/s)	Ra-226 186.1 keV (γ/s)	Ra-226+ U-235 "186" keV (γ/s)	Ra-226 estimate (pCi/g)
TENORM filter cake from oil processing containing Ra^{226} and Ra^{228}	0.25	0.25	0.25	13.92	13.92	6.57	0.01	0.86	66.56	67.42	14.10
NORM impacted filter cake	0.43	0.43	0.43	27.39	27.39	12.07	0.02	1.47	130.98	132.45	27.70
NORM impacted filter socks	0.27	0.27	0.27	41.25	41.25	21.75	0.01	0.93	197.25	198.18	41.44
NORM/TENORM waste from drilling mud solidification; contains Ra^{226}	0.16	0.16	0.16	91.00	91.00	22.20	0.01	0.55	435.15	435.70	91.11
Desorber Solids	0.45	0.45	0.45	37.35	37.35	11.50	0.02	1.54	178.60	180.15	37.67
Desorber Solids	0.83	0.83	0.83	6.75	6.75	2.85	0.04	2.85	32.28	35.12	7.34
NORM filter socks	0.00	0.00	0.00	1.26	5.16	1.06	0.00	0.00	6.03	6.03	1.26
NORM Filter Socks	0.11	0.11	0.11	1.43	1.43	1.23	0.00	0.38	6.84	7.22	1.51

Attachment

Ra-226: An Inter-Method Comparison - 15447

Terry Romanko
TestAmerica Laboratories, Inc.
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ABSTRACT

Radium is an element which can be found naturally at low levels in the environment (water, rock, soil) as well as due to impact from man. It is known to cause cancer, primarily through ingestion and inhalation, and is considered one of the main radioactive environmental concerns. Sources of ingestion include food and drinking water, while sources of inhalation include emissions from the burning of coal or other fuels as well as dust generated from the uranium or phosphate mining industries. Radium is generally present in relatively low levels in the environment, but certain geographic formations containing elevated concentrations of radium exist throughout the world. In addition, there are sites which are considered contaminated by prior industrial activity.

Radium's most prevalent isotopes are Ra-224, Ra-226, and Ra-228. Ra-226, a member of the U-238 decay series with a half-life of 1600 years, is the most common radium isotope. Several methods exist for analysis of this isotope, some targeting measurement of this radionuclide's alpha emission (or those of its decay progeny), others focused on the accompanying gamma-ray emissions. The choice of analytical method to utilize is dependent upon various factors. The purpose of this discussion is to identify various methods, highlighting strengths and weaknesses of each to assist in determining the most advantageous method to utilize, and to compare results obtained from each method.

INTRODUCTION

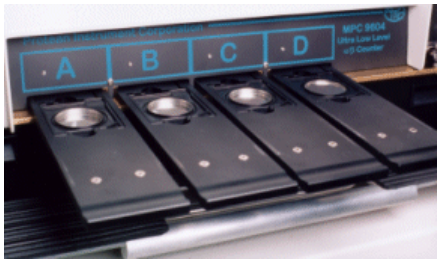
The most often requested radium isotope, Ra-226, appears in a variety of different projects. Whether the sample matrix is drinking water, groundwater, produced water, soil, sediment, drill cuttings, or waste the main analysis goal is typically to determine whether Ra-226 exists at or above a defined action limit or concentration. In many instances, the actual concentration is of importance (e.g. to help determine disposal, treatment or cleanup options).

While Ra-226 is primarily an alpha-emitting nuclide (94.45% branching ratio [BR] at 4784.4 keV, 5.55% BR at 4601.1 keV, and several very minor emissions), it does exhibit one gamma-ray of note (3.5% BR at 186.1 keV). Several methods exist for the determination of Ra-226, incorporating differing separation techniques and measurement instrumentation. The majority of analyses concentrate on the measurement of the alpha-particle emissions using gas-flow proportional counting (GFPC), Lucas cell counting, or alpha spectrometry (AS). In addition, several gamma spectrometry (GS) approaches may be utilized. While there are several other options [1], this paper will focus on the most commonly used methodology along with a laboratory-specific procedure.

Several compendial methods exploit the similarity in the chemistry of radium with that of barium to separate radium from the bulk sample. Such methods include EPA 903.0 (SW846 9315, SM 7500 Ra-B, ASTM D2460-90, and EPA Ra-03 employ the same approach) and EPA 903.1 (SM 7500 Ra-C, EPA Ra-04, ASTM D3454-91, and DOE Ra-05 employ the same approach). For this discussion we will focus on EPA methods 903.0 and 903.1, although the other methods listed are essentially equivalent. The laboratory-specific procedure also capitalizes, to a lesser extent, on this chemistry.

EPA 903.0 (GFPC)

The Summary of Method section of EPA 903.0 states the following: "The radium in the drinking water sample is collected by coprecipitation with barium and lead sulfate, and purified by reprecipitation from EDTA solution. Citric acid is added to the drinking water sample to assure that complete interchange occurs before the first precipitation step. The final $BaSO_4$ precipitate which includes Ra-226, Ra-224 and Ra-223 is alpha counted to determine the total disintegration rate of the radium isotopes." The alpha count is typically performed by GFPC. Note that as this method will include several alpha-emitting isotopes of radium (intended as a screening technique for Ra-226), the procedure is aptly titled "Alpha-Emitting Radium Isotopes in Drinking Water" and results obtained could be considered as potentially high-biased Ra-226. If the total alpha-emitting radium result is less than the action level, one knows the Ra-226 result also has to be lower than this level. The advantage to this is that no waiting time is needed, thus a 3-day turnaround time (TAT) is possible, although 14 days would be more standard for this method. The laboratory reports results from this method as Total Alpha Radium (or TAR)



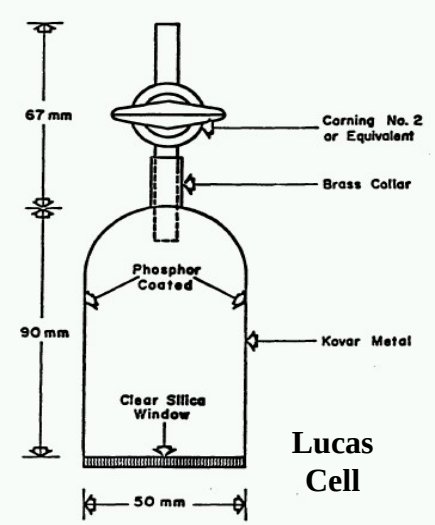
GFPC Instrumentation

TestAmerica's St. Louis laboratory utilizes a modification to this method to report results as Ra-226. The method mentions Ra-226, Ra-224, and Ra-223 as the alpha-emitting radium isotopes that may be present. Ra-223 (a daughter of Th-227 with a half-life of 11.4 days) is not normally present in environmental samples, although it might be found at isolated sites (mainly former uranium enrichment or radiopharmaceutical sites). Ra-224 is a short-lived (87.8 hours) daughter in the Th-232 decay chain. As it is short-lived, after the initial radium separation/coprecipitation time is allowed for the Ra-224 to decay out of the sample. Waiting 14 days allows for 93% of any Ra-224 present to decay out, while 21 days allows for 98% to decay out. The laboratory can report Ra-226 using 903.0, with a decay/waiting period of 14 or 21 days (depending upon client need/confidence). The 903.0 methodology provides one of the lowest cost/analysis of the available methods.

EPA 903.1 (Lucas Cell Counting)

The Summary of Method section of EPA 903.1 states: "The Ra-226 in the drinking water sample is concentrated and separated by coprecipitation on barium sulfate. The precipitate is dissolved in EDTA reagent, placed in a sealed bubbler and stored for ingrowth of Rn-222. After ingrowth, the gas is purged into a scintillation cell. When the short-lived Rn-222 daughters are in equilibrium with the parent (4h), the scintillation cell is counted for alpha activity." This technique is

commonly referred to as radon emanation. As only Ra-226 yields Rn-222 and progeny that have suitable characteristics for detection by such an emanation technique, this procedure is considered specific for Ra-226.



As can be seen from the method summaries, EPA 903.0 and 903.1 were designed for drinking water. These and other comparable methods listed find application to other liquid matrices such as groundwater or wastewater. In addition, laboratories often utilize these methods for solid matrices (e.g. soil, waste, filters, wipes) after solubilizing the radium in the sample by means of acid digestion or total dissolution.

Lab-Specific Method (AS)

In the laboratory-specific method, radium isotopes are initially separated from the sample matrix by coprecipitation with calcium carbonate. The radium is further purified using cation-exchange and extraction chromatography resins. The final eluant containing the radium is coprecipitated with a small, controlled amount of barium as the sulfate and collected on a 0.1 micron filter. Radiometric yield is determined by alpha spectrometry (AS) of traced Ra-225 (inferred from the At-217 daughter) during the analysis count along with the Ra-226 and Ra-224 (inferred from the Po-216 or Po-212 daughter).


AS Chambers

All of the aforementioned methods (EPA 903.0, 903.1, and lab-specific) are susceptible to potential matrix issues. For example, liquid samples from the hydrofracture process, whether hydrofracture fluid, flowback water, or produced water present challenges to the traditional water chemistry methods for radiochemical analysis. The high total dissolved solids (TDS) matrix creates substantial competition in the coprecipitation and chromatography chemical separations. Secondly, the high solids content exceeds some of the method-specific limits (e.g. mass on the final planchet for measurement). Thirdly, some of the individual elements typically present in very elevated amounts (e.g. Group II alkaline earth metals Ca, Sr, Ba) directly interfere with the chemistry. Matrix issues can also be experienced for solid matrices for similar reasons.

If there is stable barium native in the samples, it will add to the mass of the barium sulfate that is formed, thus biasing the apparent chemical recovery high. This will result in a low bias to the sample results. For typical water samples, barium is not normally observed in quantities significant enough to bias the recovery. Where the recovery of barium is elevated outside QC limits (>110%) for a water sample, an ICP or ICP-MS metals analysis should be run to determine the native barium and correct the recovery accordingly. All soil samples should be analyzed for native barium. Even with native analysis performed, recoveries often remain elevated due to other interferences (e.g. elevated native strontium content) which cannot be directly/easily corrected.

To alleviate such interferences, laboratories often analyze smaller aliquots to reduce the matrix effect. However, because the interference is often so great, substantially lower aliquots would need to be used, leading to elevated minimum detectable concentration (MDC) and frequently also not completely eliminating all the interference issues. Other means for determining the chemical recovery (e.g. Ba-133 tracer for 903.0 and 903.1 and Ra-225 tracer for the lab procedure) may provide relief for the recovery issue. However, they may not eliminate other problems (e.g. large amount of mass on planchet or column over-loading). EPA method 903.1 has the greatest tolerance for sample barium content and other such issues when using Ba-133 tracer.

EPA 901.1 (GS)

One cost-effective approach for samples known or suspected to contain high levels of interferences (e.g. hydrofracture projects) has been to perform gamma spectrometry (GS) (e.g. EPA 901.1) for Ra-226 on water samples, unless it is specifically known the samples are "clean" (e.g. a neighboring water well the client is testing as baseline). While the MDC by GS is much higher (~1 Bq/L) than by the drinking water methods (<0.037 Bq/L), GS avoids the chemistry issues of the 903.0/904.0 (Ra-228) methodology. And, given the high levels of activity seen in most of the samples (will vary based upon shale play, but well above 37 Bq/L in Marcellus Shale), it turns out the elevated MDC does not impact the data usability.

Ra-226 by GS is typically determined by inference from daughters (e.g. Bi-214) after sealing the sample in an appropriate counting geometry/container and waiting 21 days to allow the Ra-226 decay chain through Rn-222 to reach secular equilibrium. Such an approach is considered to be the most reliable and representative means for establishing the true Ra-226 concentration in the sample by GS. The ingrowth method is typically employed because Ra-226 analysis/quantitation by GS using its own 186.1 keV gamma-ray emission is subject to interference and potential high bias due to the 185.7 keV U-235 gamma ray. Compounding this issue is the relative peak abundances: ~3% for Ra-226 vs ~54% for U-235. This means that one count from U-235 can be mistaken as the equivalent of 18 counts of Ra-226. Experience also suggests gamma spectrometry software does not consistently assign accurate peak areas to Ra-226 (186.1 keV) and U-235 (185.7 keV), with the problem compounded by even slight drift of the instrumentation. So, while it would be expected that the 186.1 keV Ra-226 peak would normally produce a potential high bias in the presence of U-235, the laboratory has infrequently seen the Ra-226 peak area to be under-reported.

GS Instrumentation

Use of the 186 keV peak approach for background-level determination of Ra-226 in soil samples is generally not advised, as uranium is found at some level in nearly every soil type. Further, use of the 186 keV peak produces a MDC approximately 3-10 times higher than that using the Bi-214 inference approach. And, use of gamma spectrometry for analysis of Ra-226 in water samples is not normally recommended, even with the 21-day ingrowth approach, as the achievable MDC is generally too high to be of use (>1 Bq/L).

Employing the 186 keV peak, which allows for a screening method that is able to be performed relatively quickly to determine if the potentially high-biased Ra-226 doesn't exceed a predefined action level, may be warranted when:

- It is known that U-235 is not expected to be present in the sample,
- The amount of Ra-226 compared to U-235 is relatively large, or
- An action level well above background levels is targeted.

The 186 keV peak approach has been used successfully in water samples (e.g. hydrofracture flowback/produce water) for which uranium is not expected to be present in any significant amount (uranium does not normally follow the matrix) and in which the Ra-226 activity is relatively high (many hydrofracture waters exhibit Ra-226 concentrations well above the elevated MDC produced using the 186 keV peak). This approach also lends itself to verification that waste material falls below a regulatory screening limit. Thus, while the 21-day ingrowth approach is the most reliable and representative means for establishing the true Ra-226 concentration in the sample, the 186 keV approach may provide data appropriate for other uses. In such cases the data user should confirm to the laboratory their awareness of the limitations using the 186 keV peak and the approach used should be noted in the report case narrative.

Another approach is the use of an "interim" count interval (e.g. 10-day ingrowth). Results reported based upon the Bi-214 daughter with anything shorter than 21-day ingrowth are potentially biased low due to the fact the decay chain may not have reached secular equilibrium. The shorter the ingrowth, the greater the potential bias. Rn-222 is gaseous by nature, and therefore mobile/potentially volatile in a sample matrix. There are many factors which may affect the starting equilibrium, including (but not necessarily limited to):

- Soil makeup – samples with natural barite (barium sulfate) are more likely to trap Rn-222 in the matrix than other soil types
- Moisture content
- How samples are handled/collected in the field – the more the sample is disturbed, the more likely Rn-222 is to be released from the matrix
- How long samples sit in the collection container before being shipped to the laboratory for analysis
- How samples are handled at the laboratory (for example, whether samples are dried/disaggregated before placing in a geometry to count). Less Rn-222 loss is likely for samples analyzed "as-received" with reduced handling.

It is possible to apply a correction factor based upon the amount of ingrowth from sample preparation to analysis. However, to accurately apply such an ingrowth correction factor to a Ra-226 result obtained by gamma spectrometry, the laboratory must know the starting equilibrium level of the Rn-222 daughter in relation to the Ra-226 parent, information neither the

lab nor the sampler is likely to possess. In the simplest scenario, for which the Bi-214 result could be divided directly by the fractional ingrowth factor, the starting Rn-222 concentration is zero. However, the only way a laboratory could ensure a zero concentration of the Rn-222 daughter would be to perform a chemical separation, which is not typical for gamma spectrometry, especially when there are other nuclides of interest. Applying an ingrowth factor directly to a sample which has some native amount of ingrowth would actually result in a high bias to the result. The shorter the ingrowth period and the higher the native Rn-222 concentration, the larger this bias would be.

One way to estimate a starting Rn-222 equilibrium factor is for the client to set up a study of the soil type(s) involved. This would require analysis of a population of a particular soil type, both at the ingrowth interval desired (e.g. 10 days) and again after equilibrium is reached (21-days). If the data user expects results for dried/disaggregated soil (as is typically applied to ensure homogeneity, etc) the samples should be handled in this way for the study. From the resultant data, an "average" ingrowth factor for the soil can be derived. Note that this probably should be performed for each different soil type at the site, as each type may exhibit a substantially different native equilibrium factor. Short of a study such as this, the laboratory cannot know what the actual starting native amount of Rn-222 may be (and thus application of an ingrowth factor would produce potentially aberrant results).

It is preferable to report Bi-214 from analyses without ingrowth as-is (as Bi-214, not Ra-226), without any corrections, and leave the assumptions and calculations to the data user. If the laboratory does report Ra-226 with no ingrowth and no adjustment for ingrowth, a statement in the case narrative should be made to document this fact and the potential low bias. Similarly, if the laboratory reports ingrowth adjusted results when the starting equilibrium is not known, a statement in the case narrative should be included to note the potential high bias. Both of these approaches (one with a potential low bias, the other a potential high bias) come with significant risk, and the data user should confirm to the laboratory their awareness of the possible bias. It is not unusual for the results to become separated from the full report (including the narrative) in the form of a spreadsheet, and the results assumed to be "actual" Ra-226 data by others.

DATA COMPARISON

To illustrate potential differences in results generated by the various methodology the laboratory generated several sets of data that can be utilized to compare the previously mentioned methods. One such set includes 60 groundwater samples analyzed by both EPA 903.0 and EPA 903.1. A second set consists of 20 soil samples analyzed by each of EPA 903.0 (as both TAR and Ra-226), EPA 901.1 (using both 186 keV and 21-day ingrowth approaches), and the lab-specific alpha spectrometry methods. A third set of data contains results for 28 filter cake samples from the oil & gas industry.

Data Set 1 – 60 Groundwater Samples

Groundwater samples analyzed by both EPA 903.0 and EPA 903.1 demonstrated excellent correlation between the methods. The samples were analyzed by EPA 903.0 at the St. Louis, MO facility and by EPA 903.1 at the Richland, WA facility. Neither of the two methods

exhibited a significant bias compared to the other method (24 samples exhibited higher results by 903.0, 36 by 903.1). In addition, evaluating the sample results using a common QC parameter for replicates, the replicate error ratio (RER), yielded only three samples for which the results fall significantly outside statistical similarity (RER of 1.35, 1.35, and 1.57). The RER is calculated as the absolute value of the difference in the sample results (sample and replicate) divided by the sum of the 2-sigma total propagated uncertainties (TPU). This standard convention is utilized throughout this document. Figure 1 summarizes the results in graph form.

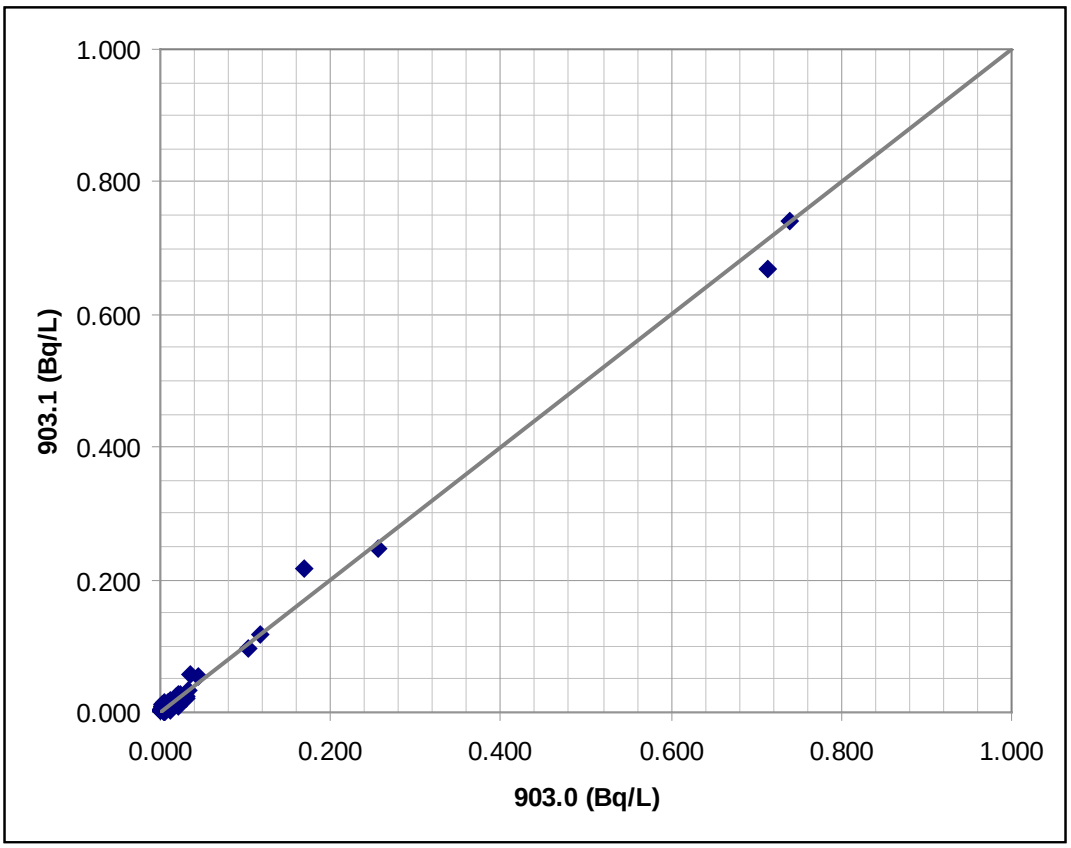


Figure 1: Comparison of Methods 903.0 vs 903.1 on groundwater samples.

Data Set 2 – 20 Soil Samples

The soil samples analyzed represent native soil (assumed to contain some amount of native U-235) with varying amounts of precipitated barium-radium sulfate, $Ba(Ra)SO_4$ (barite), from oil & gas produced water mixed/weathered into it. The majority of samples were selected based upon expected Ra-226 concentration < 370 Bq/kg, although a few samples were chosen with higher concentrations (up to approximately 1000 Bq/kg).

The soil samples were originally counted a single time by gamma spectrometry (EPA 901.1) in a tuna can geometry with the 21-day ingrowth approach (reporting Ra-226 based upon the Bi-214 daughter) well before this study was conceived. As a separate library is utilized by the laboratory when reporting Ra-226 with 21-day ingrowth versus based upon the 186 keV peak, the spectra were reprocessed using the 186 keV library. While the 186 keV library contains the

U-235 185.7 keV peak, as stated above the Ra-226 186.1 keV peak area may include some U-235 counts. Reprocessing the spectra (as opposed to recounting the sample) eliminates potential sources of error (such as positioning of the sample, detector calibration, etc). Table I summarizes the results, presented in Bq/kg with TPU at 2 sigma. Figure 2 presents the same data in graph form with error bars.

Table I: Gamma - 186 keV vs 21-day

	Gamma - 186 keV			Gamma - 21-day Ingrowth				
Sample ID	Activity	TPU	MDC	Activity	TPU	MDC	>	RER
490-40346-2	1092	159.8	248.8	786.7	29.3	86.9	186	1.61
490-40346-4	16.0	40.5	40.6	31.2	6.4	7.1	21day	0.32
490-40346-6	64.8	32.4	34.3	28.3	5.8	6.5	186	0.95
490-40346-8	201.1	65.7	74.5	145.9	12.1	19.4	186	0.71
490-40346-14	255.6	82.9	94.2	143.2	11.6	18.8	186	1.19
490-40346-15	19.5	29.2	29.4	26.1	5.8	6.4	21day	0.19
490-40346-16	52.8	33.9	35.2	15.6	4.9	5.2	186	0.96
490-40346-20	310.5	79.0	95.8	245.7	16.3	30.3	186	0.68
490-40346-22	167.2	63.8	70.1	91.0	9.8	13.6	186	1.04
490-40346-64	269.4	72.3	86.3	175.5	12.8	22.3	186	1.10
490-40346-66	190.9	69.7	77.3	116.2	10.2	15.8	186	0.93
490-40346-70	324.6	73.3	92.6	154.4	15.0	22.0	186	1.93
490-40346-96	378.3	130.3	146.1	287.5	18.8	35.3	186	0.61
490-40346-110	144.6	48.1	54.4	59.8	10.1	11.9	186	1.45
490-40346-119	689.8	146.9	190.0	520.6	24.5	59.4	186	0.99
490-40346-124	210.2	55.6	66.6	84.1	11.5	14.4	186	1.88
490-40346-125	213.0	51.0	63.1	47.1	6.9	8.5	186	2.86
490-40346-128	91.9	43.5	46.4	47.4	7.0	8.6	186	0.88
490-40346-131	98.8	53.4	56.1	58.4	9.4	11.2	186	0.64
490-40346-135	354.7	94.4	112.9	281.9	18.5	34.6	186	0.64

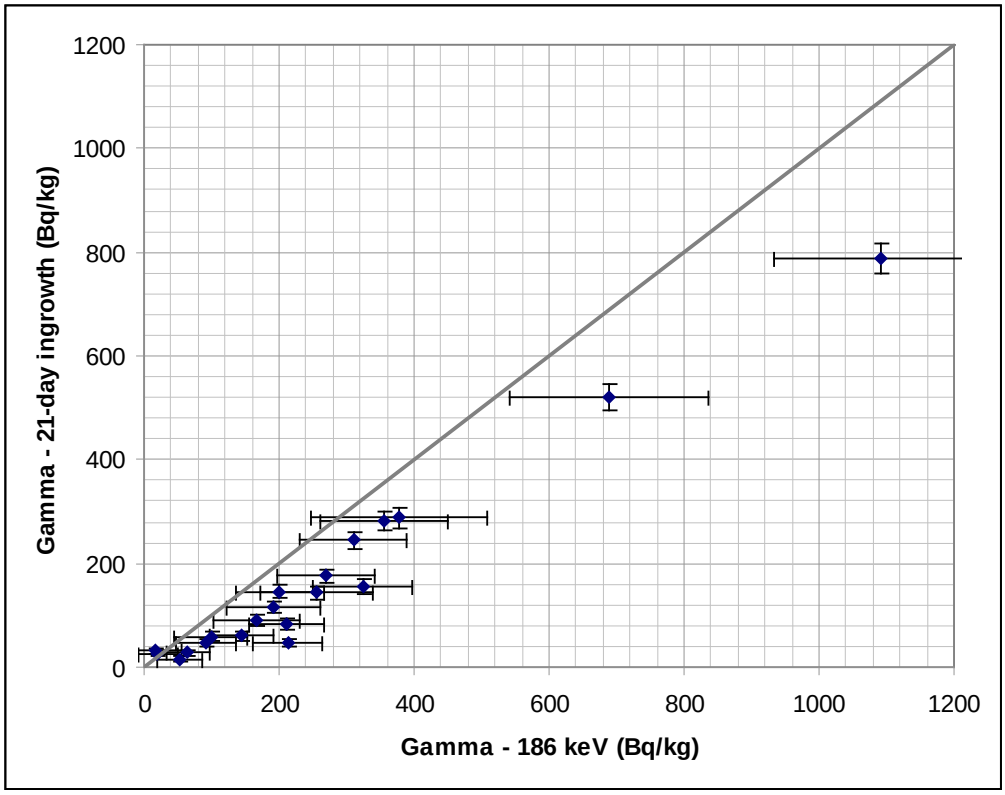


Figure 2: 20 Soil samples – Gamma Spec 186 keV vs 21-day ingrowth

As expected, the results in general demonstrate the high bias of the 186 keV approach, as all but two sample results are higher, and for those two samples the results between the 186 keV and 21-day techniques are not significantly different (RER <<1, thus the right-hand error bars overlap the control line). Analogous to the GFPC TAR vs 21-day (decay), if the 186 keV result is less than the action limit, it is typically assumed the 21-day ingrowth result must be as well. Note the significant difference in MDC between the two techniques, with the 186 keV producing a much higher MDC (as previously discussed).

The samples were prepared by EPA 903.0 (modified) with a 1 gram aliquot removed from the tuna can geometry using the laboratory standard dissolution technique employing mixed acids (nitric, hydrochloric, and hydrofluoric) before proceeding with the initial sulfate coprecipitation. After preparation, the samples were delivered to the count room to be alpha counted using GFPC. After this initial analysis, which represents the TAR, the planchets were set aside to allow for 21 days of decay. After this decay period, the samples were recounted resulting in the Ra-226 values. Results are summarized in Table II, presented in Bq/kg with TPU at 2 sigma. Figure 3 displays the data in graph form with error bars included.

Table II: GFPC - TAR vs 21-day

	GFPC - TAR			GFPC - 21-day decay				
Sample ID	Activity	TPU	MDC	Activity	TPU	MDC	>	RER
490-40346-2	954.6	95.5	9.14	851.0	79.6	2.62	TAR	0.59
490-40346-4	81.0	15.1	9.21	33.0	5.2	2.12	TAR	2.37
490-40346-6	37.4	9.5	7.03	10.0	2.8	2.40	TAR	2.22
490-40346-8	115.1	17.9	6.48	45.1	6.4	2.32	TAR	2.88
490-40346-14	166.9	23.2	7.10	106.2	12.2	2.42	TAR	1.71
490-40346-15	42.9	10.5	7.92	18.6	3.7	2.28	TAR	1.71
490-40346-16	35.2	9.4	7.47	8.3	2.5	2.28	TAR	2.25
490-40346-20	769.6	78.8	7.77	521.7	49.6	2.18	TAR	1.93
490-40346-22	229.8	28.9	7.55	142.1	15.4	2.62	TAR	1.98
490-40346-64	207.6	26.8	8.47	172.8	18.2	2.39	TAR	0.77
490-40346-66	136.2	19.8	6.92	112.9	12.7	2.22	TAR	0.72
490-40346-70	135.1	20.1	9.03	106.9	12.2	2.53	TAR	0.87
490-40346-96	256.4	31.4	7.22	276.8	27.6	2.27	21day	0.35
490-40346-110	64.8	12.5	6.88	41.1	5.9	2.29	TAR	1.29
490-40346-119	436.6	48.1	8.10	418.1	40.3	2.35	TAR	0.21
490-40346-124	122.8	18.6	6.66	69.6	8.7	2.55	TAR	1.95
490-40346-125	62.5	12.4	7.88	36.5	5.5	2.19	TAR	1.45
490-40346-128	81.4	14.4	6.81	37.7	5.7	2.13	TAR	2.18
490-40346-131	106.9	17.1	7.84	51.8	7.0	2.28	TAR	2.29
490-40346-135	271.6	32.8	7.47	203.5	20.9	3.26	TAR	1.27

As expected, the TAR results in general are greater than the Ra-226 values (after 21-day decay). Only one result exhibited a Ra-226 result greater than the TAR. For that sample, the RER was less than 1, suggesting the results are not statistically different. More than half of the results exhibit a significant statistical difference (RER >> 1). Only six of the samples would be considered acceptable replicates (RER < 1) from a QC standpoint. However, as stated previously, if the TAR result is less than the client's action limit, the Ra-226 would be considered to be as well. Note the lower MDC for the 21-day decay data set. This is due to the ingrowth of three (3) alpha-emitting daughters in the Ra-226 decay chain contributing to a larger ingrowth factor in the denominator of the calculation equations.

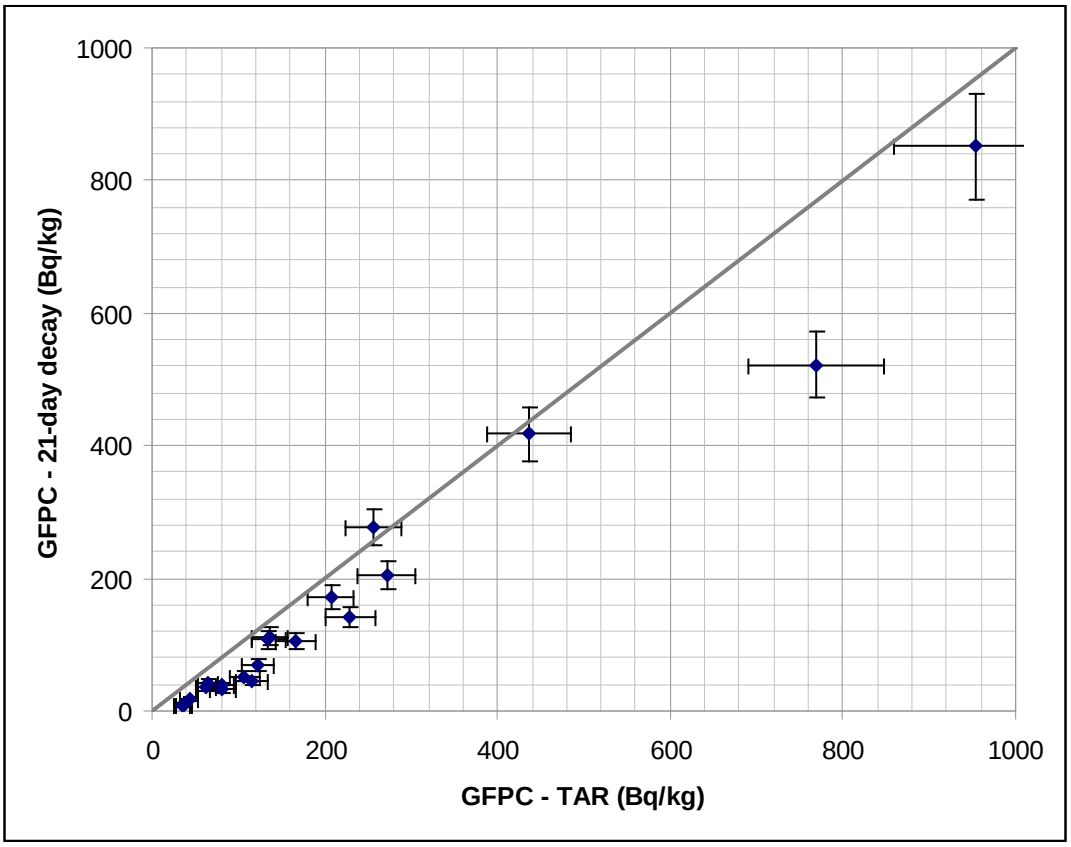


Figure 3: 20 Soil samples – GFPC TAR vs 21-day decay

The soil samples were also analyzed using the laboratory AS method via an aliquot removed from the GS tuna can. The 1 gram aliquot was put into solution employing a sodium hydroxide fusion. The data gathered from the AS method are compared in Table III to the 21-day decay (GFPC) and 21-day ingrowth (GS) results (as these are expected to best represent the true Ra-226 concentration). Results are presented in Bq/kg with TPU at 2 sigma. The same data is shown in Figure 4 in graph form with error bars. Using a RER comparison between pairs of values for each of AS:GFPC, AS:GS, and GFPC:GS we note an overall agreement between the methods (the RER for 12 of the 20 samples was <1 for all three sets). The AS:GFPC set exhibited reasonable agreement on 17 of the 20 samples. Of the three not in agreement (highlighted), two were significantly different, including one with RER = 5.65. The AS:GS data demonstrated similar consistency with 17 of the 20 samples. Of the remaining three, two displayed RER > 3. The GFPC:GS results displayed the most dissimilarity. The RER for 8 of the 20 samples exhibited significant difference. 3 of these exhibited RER similar to that of the AS:GFPC set and 3 (possibly 5) others comparable to the AS:GS set. There were no samples for which reasonable agreement did not exist for at least one of the set pairs.

Table III: Alpha Spec, GFPC 21-day, and Gamma 21-day (kBq/kg)

	Alpha Spectrometry			GFPC - 21-day decay			Gamma - 21-day Ingrowth			RER		
Sample ID	Act	TPU	MDC	Act	TPU	MDC	Act	TPU	MDC	AS/ GFPC	AS/ GS	GFPC /GS
490-40346-2	921	79.9	1.13	851	79.6	2.62	787	29.3	86.9	0.44	1.23	0.59
490-40346-4	43.3	5.1	0.85	33.0	5.2	2.12	31.2	6.4	7.1	1.00	1.05	0.15
490-40346-6	27.6	3.7	0.93	10.0	2.8	2.40	28.3	5.8	6.5	2.72	0.08	2.15
490-40346-8	174	16.3	1.22	45.1	6.4	2.32	146	12.1	19.4	5.65	0.97	5.45
490-40346-14	117	11.4	0.79	106	12.2	2.42	143	11.6	18.8	0.47	1.13	1.56
490-40346-15	28.0	3.9	1.01	18.6	3.7	2.28	26.1	5.8	6.4	1.23	0.19	0.79
490-40346-16	13.8	2.5	1.25	8.3	2.5	2.28	15.6	4.9	5.2	1.10	0.25	0.98
490-40346-20	466	41.1	1.30	522	49.6	2.18	246	16.3	30.3	0.61	3.85	4.19
490-40346-22	175	16.5	1.35	142	15.4	2.62	91.0	9.8	13.6	1.02	3.18	2.02
490-40346-64	172	16.2	1.02	173	18.2	2.39	176	12.8	22.3	0.01	0.11	0.09
490-40346-66	105	10.4	1.05	113	12.7	2.22	116	10.2	15.8	0.34	0.54	0.15
490-40346-70	155	15.1	1.24	107	12.2	2.53	154	15.0	22.0	1.76	0.02	1.74
490-40346-96	330	29.5	0.23	277	27.6	2.27	287	18.8	35.3	0.93	0.88	0.23
490-40346-110	48.5	5.6	1.02	41.1	5.9	2.29	59.8	10.1	11.9	0.64	0.72	1.17
490-40346-119	444	38.9	0.88	418	40.3	2.35	521	24.5	59.4	0.33	1.21	1.58
490-40346-124	80.7	8.3	0.98	69.6	8.7	2.55	84.1	11.5	14.4	0.65	0.17	0.72
490-40346-125	48.1	5.7	1.30	36.5	5.5	2.19	47.1	6.9	8.5	1.04	0.08	0.86
490-40346-128	42.9	5.3	1.27	37.7	5.7	2.13	47.4	7.0	8.6	0.47	0.36	0.76
490-40346-131	54.4	6.0	1.11	51.8	7.0	2.28	58.4	9.4	11.2	0.20	0.26	0.40
490-40346-135	196	18.1	1.08	204	20.9	3.26	282	18.5	34.6	0.18	2.34	1.99

The author suspects some amount of non-homogenaity of these samples. In retrospect, while the samples were originally dried and disaggregated, the samples in the GS tuna can geometry (upon opening to obtain the aliquots for the AS and GFPC analyses) showed evidence for such non-homogenaity, including small rocks that were included as part of the original analysis. These small rocks were not included in the AS or GFPC aliquots, which are relatively small (1 g) in comparison to the GS aliquot (~350 g). This is a plausible explanation for the disagreement of portions of the results while there is apparent overall correspondence in results amongst the three methods.

Of possible importance to the end user of data produced from these three methods is the achievable MDC. The lowest MDC achieved was by AS, with the highest by GS. The 21-decay GFPC MDC, while much lower than that by GS, was still a factor of about 2-3 times that of AS. AS presents a couple of other advantages over the GFPC and GS methods. First is the spectral confirmation for identification (see Figure 5 for example) as well as the ability to easily determine Ra-224 (if desired). Second is the potential for a more rapid TAT – while there is a short ingrowth period necessary for the At-217 daughter of the Ra-225 tracer, results may still be available in as little as 3 days.

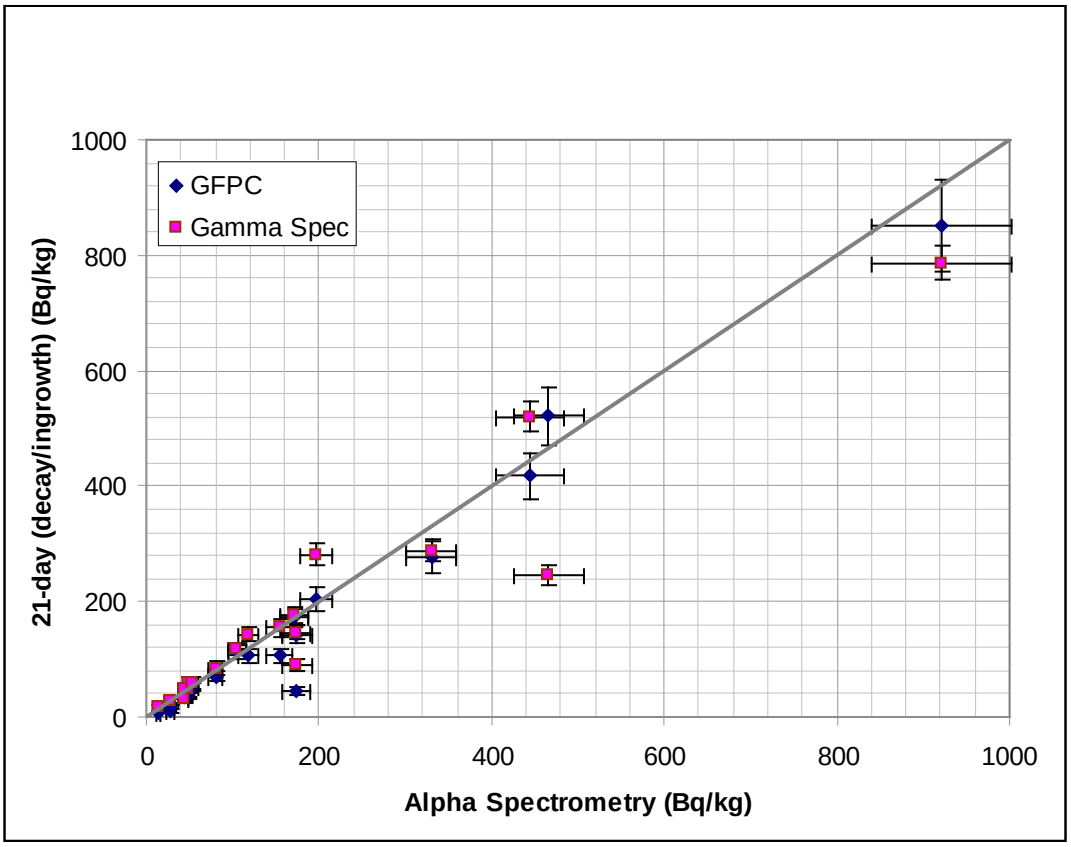


Figure 4: 20 Soil samples – Alpha Spec vs GFPC and Gamma Spec.

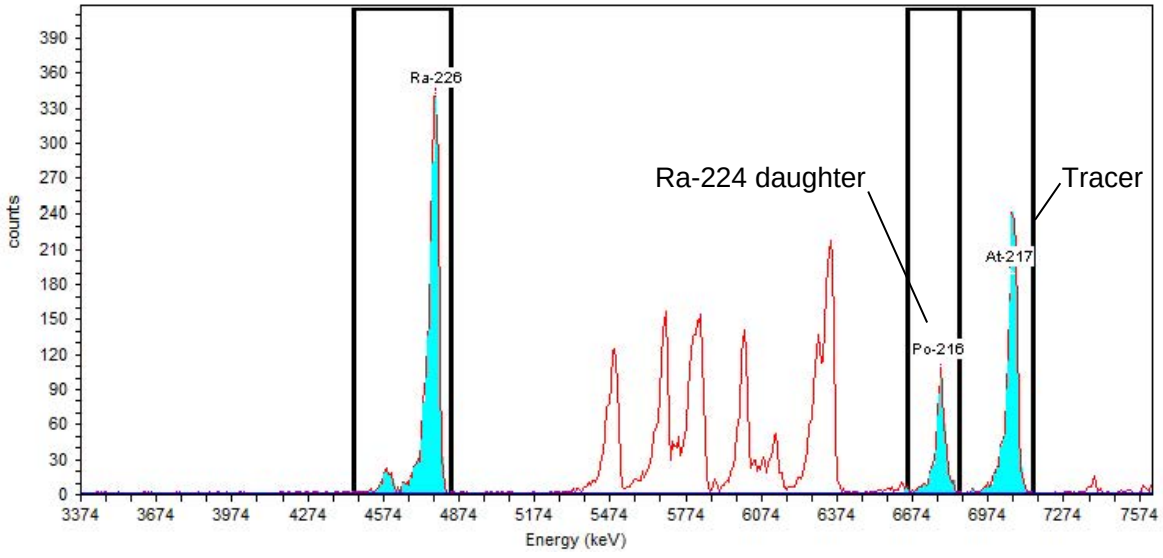


Figure 5: Alpha Spectrum from laboratory Ra-226 method

Data Set 3 – 28 Filter Cake Samples

The third set of data, 28 filter cake samples (plus 3 sample duplicates) from the oil & gas industry is included to highlight a legitimate instance for the use of the 186 keV GS approach. These solids originate from flocculated/precipitated produced water, and are expected to contain elevated radium isotopes but no significant uranium (U-235). These samples were initially dried/disaggregated and then sealed into a tuna can geometry. The samples were counted soon after canning and processed using the 186 keV library. The samples were subsequently held for 21-day ingrowth, counted, and evaluated by means of the Bi-214 daughter. The Ra-226 activity ranged from 10 kBq/kg to 25 kBq/kg. The 4.1% average relative percent difference (RPD, which provides a better evaluation of replication at elevated activity) between the 186 keV and 21-day approaches indicates very good agreement between the methods. It is interesting to note that the 21-day ingrowth resulted in the higher average concentration. Figure 6 shows the results in graph form.

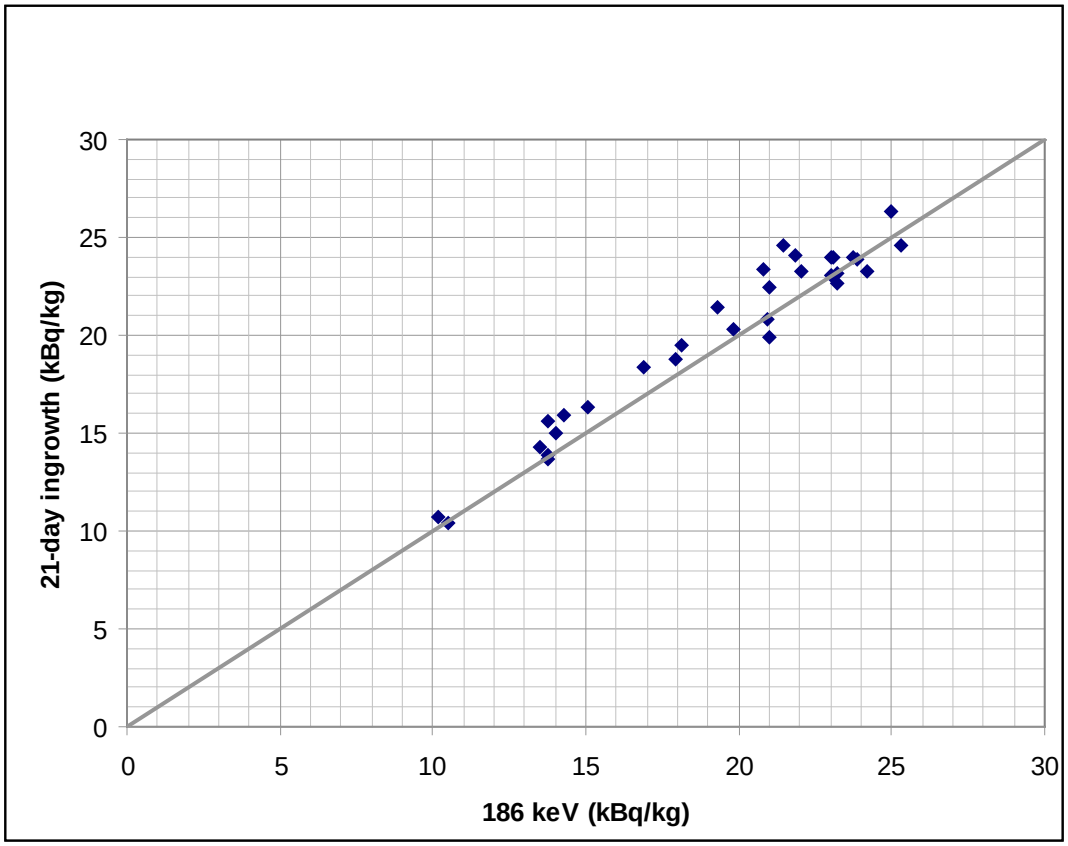


Figure 6: Filter Cake Samples – Gamma Spec 186 keV vs 21-day ingrowth

CONCLUSION

It is apparent from the above discussion and data comparison that sufficient consideration to methodology should be made when Ra-226 analysis is a necessary part of project objectives. EPA Methods 903.0 and 903.1 have been the method of choice for many projects in the past.

However, given the current needs in a variety of different industries they may not be the best choice for all projects. Suitability of the approach for detection limit, required TAT, tolerance for bias, and even cost play a part in the decision of which method to choose.

The 21-day GFPC, 21-day GS, and AS procedures appear to produce reasonably comparable results. Approaches using GFPC and GS may offer the overall lowest cost/analysis, but AS can produce the lowest MDC. EPA 903.1 demonstrates the most tolerance to native barium and other interferences (although this benefit diminishes if Ra-228 is also required). GS offers the ability to obtain more than just Ra-226 (can report many other gamma-emitting isotopes, including Ra-228), provides some defense against non-homogenaity due to the large aliquot volume used, and eliminates the interference concerns. The TAR and 186 keV GS approaches offer the potential of rapid TAT, but at the price of a potential bias to results.

Table IV provides a quick reference for method, instrumentation, and TAT. It is strongly suggested you contact your analytical laboratory representative to discuss options in regard to your Ra-226 analysis needs to help determine the method to best fit your objectives.

Table IV: Methods, Instrumentation, and TAT

Method Reference	Analyte Reported	Instrument Type	Method Defined TAT	Standard TAT
EPA 903.0	Ra-226	Gas Flow Proportional Counter	23 days	28 days
EPA 903.0	Ra-226 as Total Alpha Radium	Gas Flow Proportional Counter	3 days	14 days
EPA 901.1	Ra-226 (from Bi-214)	Gamma Spectrometry	22 days	28 days
EPA 901.1	Ra-226 (from 186 kev)	Gamma Spectrometry	4 hours	14 days
ST-RC-0301	Ra-226	Alpha Spectrometry	3 days	14 days
EPA 903.1	Ra-226	Lucas Cell Alpha Scintillation	7 days	28 days

Method defined turnaround times do not reflect any pre-preparation techniques such as drying, disaggregating, or compositing.

REFERENCES

1. Nelson, Andrew W.; May, Dustin; Knight, Andrew W.; Eitrheim, Eric S.; Mehrhoff, Marinea; Shannon, Robert; Litman, Robert; Schultz, Michael K. Matrix Complications in the Determination of Radium Levels in Hydraulic Fracturing Flowback Water from Marcellus Shale. *Environ. Sci.Technol. Lett.* **2014**, 1, 204-208.

TECHNICAL MEMORANDUM

Deer Trail Exposure Rate Acceptance Criteria

To: Jack Kehoe, Facility General Manager
Clean Harbors Deer Trail

From: Mark O. Somerville Ph.D., CHP
CB&I Federal Services

Date: March 30, 2015

1.0 Executive Summary

Naturally Occurring or Technologically Enhanced Radioactive Materials (NORM and TENORM) are by-products associated with the oil and natural gas industry as well as other industrial sectors. The concentration of naturally occurring radioactive materials in wastes, process filters, process equipment, produced waters, and other derived-from materials is a function of natural, in-situ concentrations, concentrating processes, and contaminant load capacity.

Standards and regulations exist in the various States for the handling and disposition of NORM and TENORM. Individual locations may be granted or issued specific limits that are deemed protective to worker and members of the public. These limits are set above the concentrations typically found in nature.

For the purposes of this document, the primary radionuclide of concern is Radium-226 in equilibrium with its daughter products. Radium-228 is present at an assumed concentration of 50% of that of Ra-226. Natural uranium is not a significant contributor, meaning that it accounts for less than 10% of the activity. Additionally, the limits used as a basis for action are conservative with respect to the absolute acceptance limits.

This is a companion document to previously performed work and, therefore, will not repeat the detailed decay schemes, operational descriptions, or industry history.

2.0 Introduction

The purpose of this report is to provide the basis for field level acceptance criteria for NORM/TENORM related material surveyed for acceptance at the Clean Harbors, Deer trail

Facility. The subject material is described as loose material, from tank bottoms and sock filters with entrained contaminants.

The material is transported in covered trucks, roll offs, drums, and/or smaller packages containing filters.

Oil field wastes and related equipment and materials typically contain and concentrate NORM. The background, nature, and specific decay modes for the most commonly encountered radionuclides have been previously described for this facility (Kennedy Jr., Winslow, & Ikenberry, 2009).

A key difference between the assumptions in previous reports and the assumptions here is the form and location of the material. Pipe scale adherent to the inner wall of piping has been described with a density ranging from 1.7 to 3.5 g/cm^3 (Kennedy Jr. et al., 2009, p. 2-1). This report assumes the material has been freed from the pipe surface and is containerized in a form approximated by loose soil. Loose, moist soil has a density of 1200 kg/m^3 or 1.2 g/cm^3. Compact soil has a density of 1.6 g/cm^3 (The Engineering Toolbox, n.d., Table 1).

Truck box sides are assumed to be of standard double wall design with the inner layer consisting of 0.19 inch steel and the outer wall of 10 gauge steel, equivalent to 0.1406 inch USG.

3.0 Method

The point-kernel software MicroShield version 9.07 was used to model radionuclide concentration to exposure rate relationships. Dump truck boxes and roll offs were set to the same dimensions of 20 feet in length, 8 feet in width, and 3.38 feet in height. The value for height corresponds to the depth of material in the truck or roll off and the overall dimensions allow for the modeling of less than full containers. The individual dimensions are not critically important because when detectors of small volume are placed in close proximity to relatively large, planer sources, the angle viewed by the detector is a very small portion of the total source.

Because roll off construction is typically less robust than dump truck construction, roll off conversion factors will be conservative due to a higher modeled shielding component than actually exists.

The truck and roll off model is also acceptable for use with smaller steel boxes, such as B-25 or equivalent containers. As above, measurements taken close to the container sides diminishes the importance of any specific dimension as the presenting plane is much larger than the detector volume.

Drums were modeled as 1A2, 55-gallon open head drums with a 35-inch height and a 22.5-inch diameter. Such a drum has an 18 gauge shell and lid. This is a thickness of 0.0478 inches.

A one-cubic-foot unshielded item was also modeled. This element is intended to characterize small packages or individual sock-filters and similar items or elements.

The source term was defined as Radium-226 with an initial concentration of 222 pCi/gram of material. This is the derived acceptance concentration used in other facility related technical reports (Clean Harbors Deer Trail, LLC [Clean Harbors], 2005).

The source was decayed through an interval of 1 year. This is a short time period with respect to the 1,600-year half-life of Radium-226 but forces the in-growth of decay daughters into the activity to exposure rate calculation.

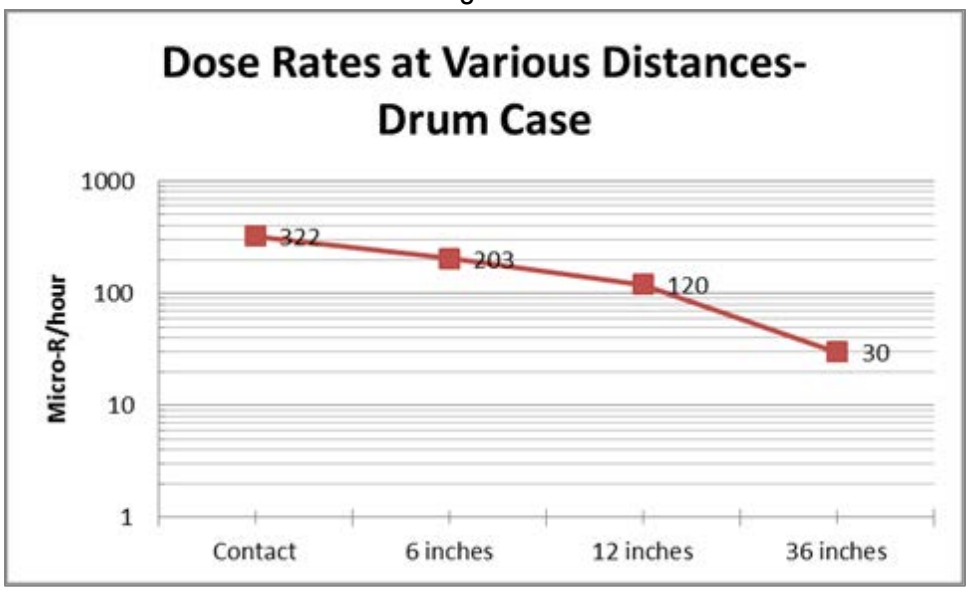
Exposure rates were calculated for 4 source-to-detector distances. The distances are 2 inches or contact with the container side, 6 inches (15 cm), 12 inches (30 cm), and 36 inches.

The results of the MicroShield case runs are attached in Appendix A.

4.0 Results

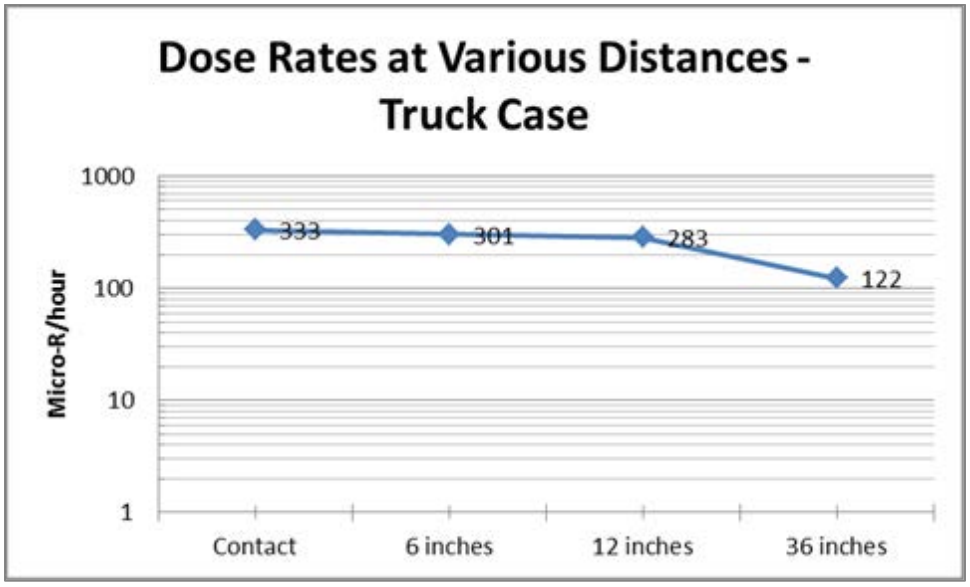
The results for the drum case are shown in Figure 1. The exposure rate is in micro-R/hour and is above background.

Figure 1



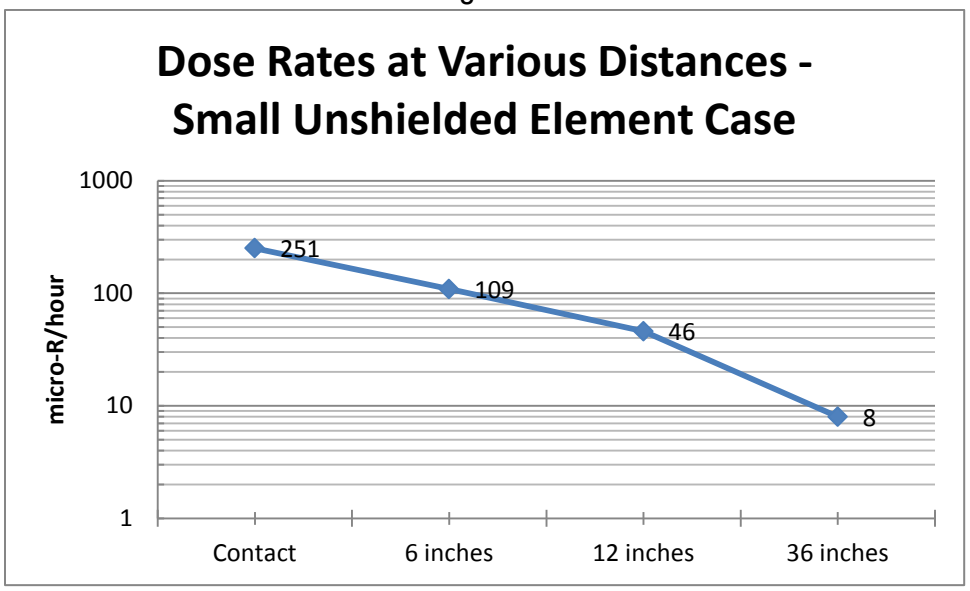
The results for the truck case, which includes roll off-type containers and metal boxes is shown in Figure 2.

Figure 2



Finally, the results for the 1 cubic-foot unshielded item are shown in Figure 3.

Figure 3



5.0 Discussion

There is consistency in the derived exposure rate values in the near-field as would be expected when the radionuclide concentration is held constant and the source to detector distance is small with respect to the size of the defined source. The main source of variability in this simplified model is the shielding and, ultimately, the total volume of source.

Because of the consistency in results, a simple and efficient approach will be to utilize a **CONTACT** exposure rate reading of 3**00 micro-R/hour above background** (0.3 mR/hr) as the acceptance screening value for trucks, drums, roll offs, and boxes.

Small, individual and unshielded items should be assessed at **one foot (30 cm)** for contamination control purposes and use an acceptance screening value **of 45 micro-R/hour above background.**

Surveys should be performed in a manner suitable for the exposure rate instrument in use and background in the survey area should be known or assessed prior to beginning any survey. Particular attention should be paid to the contribution to ambient, general area exposure rates by vehicles, containers, or objects near the survey area. The acceptance values are most useful when the only source of non-background radiation readings is from the container or vehicle of immediate interest.

For Trucks, Roll Offs, Boxes, and Drums:

Calculations and models such as this necessarily assume homogeneity of the source. It is possible that an array of exposure rates will be detected upon the performance of an acceptance survey. If exposure rates in excess of 1.7 times the acceptance screening value are discovered (500 micro-R/hour or 0.5 mrem/hour) supervision should be notified. The 500 micro-R/hour value is equivalent to the limit for exposure rate on contact with a limited quantity package and, while most shipments will be excepted, may represent an issue that requires follow up. If multiple locations are found by survey that reach 500 micro-R/hour, consider rejecting the shipment.

When surveys show most or all locations assessed in excess of the **300 micro-R/hour** screening criterion, but no points exceed 330 micro-R/hour, the shipment should be accepted.

If less than 10% of points surveyed exceed the 300 micro-R/hour screening criterion, the shipment can likely be accepted based on average concentrations and the notion that the radioactive material is “essentially evenly distributed.”

The license or regulatory-based requirements for concentration determination and averaging should be factored into the use of any screening criteria.

For Small, Unshielded Items:

When exposure rates exceed the values in Table 3, place the object in a secure location to preclude the spread of contamination and to control the object. Consider rejecting the individual item or combining in a permissible manner with lower activity concentration items for characterization and acceptance.

Best Use Procedure:

- To properly focus on the potential radiological impact of the material being evaluated, radiation reading should be made with a detector and meter that reads out in micro-R/hour or has a scale that allows the values listed in this document to be easily read and distinguished from background fluctuations.

- Surveys should be performed essentially in contact with the vehicle or package except, as mention, in the case of un-packaged items.

- The speed of the survey must be in accordance with the ability of the meter to resolve and process signal. Use of detectors and meters by qualified, experienced personnel will typically demonstrate understanding of proper use protocols.

- Because of the variability in natural and technologically enhanced sources of radiation, there should be an expectation of variability in survey data.

- It should be expected that sources, concentrations, quantities, and forms of radioactive materials will vary widely. If a particular condition creates uncertainty, the best course of action is to reject the material or to request assistance in making an acceptance or rejection decision.

- The radionuclide mixture should be validated periodically. A representative composite sample or an individual sample the is representative of the waste streams entering the site should be sent for off-site analysis and the results used to validate the exposure rate acceptance criteria. Typically, periodically means annually but other definitions can be supported.

6.0 References

Clean Harbors Deer Trail, LLC. (2005). *Radioactive materials Application* [Radioactive Materials License Application]. Deer Trails, CO: Author.

Kennedy Jr., W. E., Winslow, R. C., & Ikenberry, T. A. (2009). *Technical basis for the acceptance of oil production piping and equipment containing radioactive pipe scale at the Deer Trail landfill* (DMA-TR-42, Rev 0). Richland, WA: Dade Moeller & Associates, Inc.

The Engineering Toolbox. (n.d.). www.engineeringtoolbox.com/dirt-mud-densities-d_1727.html

Appendix A

MicroShield 9.07
CB&I Federal Services-Mark O. Somerville (9.07-0000)

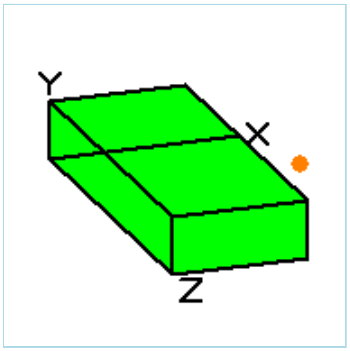
Date	By	Checked

Filename	Run Date	Run Time	Duration
DUmp and Rolloff Case deer trail 222 pCi Ra226 and 111 pCi Ra228.msd	March 30, 2015	3:21:57 PM	00:00:01

Project Info	
Case Title	TENORM Rev1
Description	Colorado Rolloff Case Includes Ra-226/228 decayed 1y
Geometry	13 - Rectangular Volume

Source Dimensions	
Length	243.84 cm (8 ft)
Width	609.6 cm (20 ft 0.0 in)
Height	103.022 cm (3 ft 4.6 in)

Dose Points			
A	X	Y	Z
#1	250.027 cm (8 ft 2.4 in)	51.511 cm (1 ft 8.3 in)	304.8 cm (10 ft 0.0 in)
#2	260.177 cm (8 ft 6.4 in)	51.511 cm (1 ft 8.3 in)	304.8 cm (10 ft 0.0 in)
#3	265.267 cm (8 ft 8.4 in)	51.511 cm (1 ft 8.3 in)	304.8 cm (10 ft 0.0 in)
#4	341.467 cm (11 ft 2.4 in)	51.511 cm (1 ft 8.3 in)	304.8 cm (10 ft 0.0 in)



Shields			
Shield N	Dimension	Material	Density
Source	1.53e+07 cm³	Compact Soil	1.6
Shield 1	.482 cm	Iron	7.86
Shield 2	.254 cm	Air	0.00122
Shield 3	.357 cm	Iron	7.86
Air Gap		Air	0.00122

Source Input: Grouping Method - Standard Indices
Number of Groups: 25
Lower Energy Cutoff: 0.015
Photons < 0.015: Included
Library: Grove

Nuclide	Ci	Bq	µCi/cm³	Bq/cm³
Ac-228	2.4112e-003	8.9213e+007	1.5745e-004	5.8257e+000
Bi-210	1.6090e-004	5.9535e+006	1.0507e-005	3.8877e-001
Bi-212	7.6560e-004	2.8327e+007	4.9994e-005	1.8498e+000
Bi-214	5.4359e-003	2.0113e+008	3.5497e-004	1.3134e+001
Pb-210	1.6416e-004	6.0738e+006	1.0720e-005	3.9662e-001
Pb-212	7.6570e-004	2.8331e+007	5.0001e-005	1.8500e+000
Pb-214	5.4359e-003	2.0113e+008	3.5497e-004	1.3134e+001
Po-210	8.5697e-005	3.1708e+006	5.5961e-006	2.0706e-001
Po-212	4.9052e-004	1.8149e+007	3.2031e-005	1.1852e+000
Po-214	5.4347e-003	2.0109e+008	3.5489e-004	1.3131e+001
Po-216	7.6675e-004	2.8370e+007	5.0069e-005	1.8526e+000
Po-218	5.4370e-003	2.0117e+008	3.5504e-004	1.3136e+001
Ra-224	7.6675e-004	2.8370e+007	5.0069e-005	1.8526e+000
Ra-226	5.4369e-003	2.0117e+008	3.5504e-004	1.3136e+001

Ra-228	2.4109e-003	8.9202e+007	1.5743e-004	5.8249e+000
Rn-220	7.6675e-004	2.8370e+007	5.0069e-005	1.8526e+000
Rn-222	5.4370e-003	2.0117e+008	3.5504e-004	1.3136e+001
Th-228	7.7529e-004	2.8686e+007	5.0627e-005	1.8732e+000
Tl-208	2.7508e-004	1.0178e+007	1.7963e-005	6.6463e-001

Buildup: The material reference is Source Integration Parameters	
X Direction	10
Y Direction	20
Z Direction	20

Results - Dose Point # 1 - (2.50e+02,51.5112,304.8) cm

Energy (MeV)	Activity (Photons/sec)	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup	Absorbed Dose Rate mrad/hr No Buildup	Absorbed Dose Rate mrad/hr With Buildup	Absorbed Dose Rate mGy/hr No Buildup	Absorbed Dose Rate mGy/hr With Buildup
0.015	7.592e+07	3.520e-203	4.215e-26	3.019e-204	3.615e-27	2.636e-204	3.156e-27	2.636e-206	3.156e-29
0.04	2.896e+05	1.297e-16	6.455e-16	5.735e-19	2.855e-18	5.006e-19	2.492e-18	5.006e-21	2.492e-20
0.05	2.469e+06	9.642e-10	7.951e-09	2.569e-12	2.118e-11	2.242e-12	1.849e-11	2.242e-14	1.849e-13
0.06	4.473e+05	9.048e-08	9.892e-07	1.797e-10	1.965e-09	1.569e-10	1.715e-09	1.569e-12	1.715e-11
0.08	5.990e+07	2.387e-03	2.858e-02	3.778e-06	4.522e-05	3.298e-06	3.948e-05	3.298e-08	3.948e-07
0.1	6.179e+06	2.041e-03	2.237e-02	3.123e-06	3.423e-05	2.726e-06	2.988e-05	2.726e-08	2.988e-07
0.15	3.568e+06	8.769e-03	7.398e-02	1.444e-05	1.218e-04	1.261e-05	1.064e-04	1.261e-07	1.064e-06
0.2	4.026e+07	2.260e-01	1.608e+00	3.988e-04	2.837e-03	3.482e-04	2.477e-03	3.482e-06	2.477e-05
0.3	6.076e+07	8.236e-01	4.660e+00	1.562e-03	8.839e-03	1.364e-03	7.716e-03	1.364e-05	7.716e-05
0.4	7.902e+07	1.865e+00	8.994e+00	3.634e-03	1.753e-02	3.173e-03	1.530e-02	3.173e-05	1.530e-04
0.5	1.097e+07	3.925e-01	1.674e+00	7.704e-04	3.287e-03	6.725e-04	2.869e-03	6.725e-06	2.869e-05
0.6	1.069e+08	5.349e+00	2.065e+01	1.044e-02	4.031e-02	9.114e-03	3.519e-02	9.114e-05	3.519e-04
0.8	3.543e+07	3.003e+00	1.001e+01	5.712e-03	1.903e-02	4.986e-03	1.662e-02	4.986e-05	1.662e-04
1.0	1.140e+08	1.454e+01	4.345e+01	2.681e-02	8.009e-02	2.340e-02	6.992e-02	2.340e-04	6.992e-04
1.5	4.817e+07	1.286e+01	3.210e+01	2.164e-02	5.400e-02	1.889e-02	4.715e-02	1.889e-04	4.715e-04
2.0	5.398e+07	2.392e+01	5.379e+01	3.699e-02	8.317e-02	3.229e-02	7.261e-02	3.229e-04	7.261e-04
3.0	1.016e+07	8.880e+00	1.751e+01	1.205e-02	2.375e-02	1.052e-02	2.074e-02	1.052e-04	2.074e-04
Totals	**7.084e+08**	**7.188e+01**	**1.946e+02**	**1.200e-01**	**3.331e-01**	**1.048e-01**	**2.908e-01**	**1.048e-03**	**2.908e-03**

Results - Dose Point # 2 - (2.60e+02,51.5112,304.8) cm

Energy (MeV)	Activity (Photons/sec)	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup	Absorbed Dose Rate mrad/hr No Buildup	Absorbed Dose Rate mrad/hr With Buildup	Absorbed Dose Rate mGy/hr No Buildup	Absorbed Dose Rate mGy/hr With Buildup
0.015	7.592e+07	2.724e-191	3.489e-26	2.337e-192	2.992e-27	2.040e-192	2.612e-27	2.040e-194	2.612e-29
0.04	2.896e+05	3.989e-16	1.936e-15	1.764e-18	8.561e-18	1.540e-18	7.474e-18	1.540e-20	7.474e-20
0.05	2.469e+06	1.312e-09	1.041e-08	3.494e-12	2.772e-11	3.051e-12	2.420e-11	3.051e-14	2.420e-13
0.06	4.473e+05	8.826e-08	9.260e-07	1.753e-10	1.839e-09	1.530e-10	1.606e-09	1.530e-12	1.606e-11
0.08	5.990e+07	2.005e-03	2.404e-02	3.172e-06	3.804e-05	2.770e-06	3.321e-05	2.770e-08	3.321e-07

0.1	6.179e+06	1.762e-03	2.011e-02	2.696e-06	3.076e-05	2.354e-06	2.686e-05	2.354e-08	2.686e-07
0.15	3.568e+06	8.037e-03	7.079e-02	1.323e-05	1.166e-04	1.155e-05	1.018e-04	1.155e-07	1.018e-06
0.2	4.026e+07	2.108e-01	1.537e+00	3.720e-04	2.713e-03	3.248e-04	2.369e-03	3.248e-06	2.369e-05
0.3	6.076e+07	7.759e-01	4.404e+00	1.472e-03	8.353e-03	1.285e-03	7.292e-03	1.285e-05	7.292e-05
0.4	7.902e+07	1.761e+00	8.426e+00	3.431e-03	1.642e-02	2.995e-03	1.433e-02	2.995e-05	1.433e-04
0.5	1.097e+07	3.704e-01	1.558e+00	7.270e-04	3.057e-03	6.347e-04	2.669e-03	6.347e-06	2.669e-05
0.6	1.069e+08	5.041e+00	1.910e+01	9.839e-03	3.727e-02	8.589e-03	3.254e-02	8.589e-05	3.254e-04
0.8	3.543e+07	2.819e+00	9.165e+00	5.361e-03	1.743e-02	4.680e-03	1.522e-02	4.680e-05	1.522e-04
1.0	1.140e+08	1.359e+01	3.949e+01	2.504e-02	7.279e-02	2.186e-02	6.355e-02	2.186e-04	6.355e-04
1.5	4.817e+07	1.188e+01	2.876e+01	1.998e-02	4.838e-02	1.744e-02	4.224e-02	1.744e-04	4.224e-04
2.0	5.398e+07	2.188e+01	4.774e+01	3.383e-02	7.383e-02	2.954e-02	6.446e-02	2.954e-04	6.446e-04
3.0	1.016e+07	8.018e+00	1.538e+01	1.088e-02	2.087e-02	9.496e-03	1.822e-02	9.496e-05	1.822e-04
Totals	**7.084e+08**	**6.635e+01**	**1.757e+02**	**1.110e-01**	**3.013e-01**	**9.687e-02**	**2.630e-01**	**9.687e-04**	**2.630e-03**

Results - Dose Point # 3 - (2.65e+02,51.5112,304.8) cm

Energy (MeV)	Activity (Photons/sec)	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup	Absorbed Dose Rate mrad/hr No Buildup	Absorbed Dose Rate mrad/hr With Buildup	Absorbed Dose Rate mGy/hr No Buildup	Absorbed Dose Rate mGy/hr With Buildup
0.015	7.592e+07	4.881e-190	3.219e-26	4.187e-191	2.761e-27	3.655e-191	2.410e-27	3.655e-193	2.410e-29
0.04	2.896e+05	3.908e-16	1.891e-15	1.728e-18	8.363e-18	1.509e-18	7.301e-18	1.509e-20	7.301e-20
0.05	2.469e+06	1.196e-09	9.477e-09	3.187e-12	2.525e-11	2.782e-12	2.204e-11	2.782e-14	2.204e-13
0.06	4.473e+05	8.199e-08	8.673e-07	1.629e-10	1.723e-09	1.422e-10	1.504e-09	1.422e-12	1.504e-11
0.08	5.990e+07	1.981e-03	2.401e-02	3.135e-06	3.799e-05	2.737e-06	3.317e-05	2.737e-08	3.317e-07
0.1	6.179e+06	1.763e-03	2.012e-02	2.697e-06	3.078e-05	2.354e-06	2.687e-05	2.354e-08	2.687e-07
0.15	3.568e+06	7.986e-03	6.937e-02	1.315e-05	1.142e-04	1.148e-05	9.972e-05	1.148e-07	9.972e-07
0.2	4.026e+07	2.080e-01	1.488e+00	3.672e-04	2.627e-03	3.206e-04	2.293e-03	3.206e-06	2.293e-05
0.3	6.076e+07	7.597e-01	4.218e+00	1.441e-03	8.002e-03	1.258e-03	6.986e-03	1.258e-05	6.986e-05
0.4	7.902e+07	1.715e+00	8.028e+00	3.342e-03	1.564e-02	2.917e-03	1.366e-02	2.917e-05	1.366e-04
0.5	1.097e+07	3.594e-01	1.479e+00	7.054e-04	2.903e-03	6.158e-04	2.534e-03	6.158e-06	2.534e-05
0.6	1.069e+08	4.875e+00	1.809e+01	9.516e-03	3.530e-02	8.308e-03	3.082e-02	8.308e-05	3.082e-04
0.8	3.543e+07	2.712e+00	8.650e+00	5.159e-03	1.645e-02	4.504e-03	1.436e-02	4.504e-05	1.436e-04
1.0	1.140e+08	1.302e+01	3.717e+01	2.400e-02	6.852e-02	2.095e-02	5.982e-02	2.095e-04	5.982e-04
1.5	4.817e+07	1.130e+01	2.695e+01	1.901e-02	4.535e-02	1.660e-02	3.959e-02	1.660e-04	3.959e-04
2.0	5.398e+07	2.072e+01	4.463e+01	3.204e-02	6.902e-02	2.797e-02	6.025e-02	2.797e-04	6.025e-04
3.0	1.016e+07	7.554e+00	1.434e+01	1.025e-02	1.946e-02	8.946e-03	1.699e-02	8.946e-05	1.699e-04
Totals	**7.084e+08**	**6.324e+01**	**1.652e+02**	**1.059e-01**	**2.835e-01**	**9.241e-02**	**2.475e-01**	**9.241e-04**	**2.475e-03**

Results - Dose Point # 4 - (3.41e+02,51.5112,304.8) cm

Energy (MeV)	Activity (Photons/sec)	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup	Absorbed Dose Rate mrad/hr No Buildup	Absorbed Dose Rate mrad/hr With Buildup	Absorbed Dose Rate mGy/hr No Buildup	Absorbed Dose Rate mGy/hr With Buildup
0.015	7.592e+07	1.850e-188	1.426e-26	1.587e-189	1.223e-27	1.385e-189	1.068e-27	1.385e-191	1.068e-29
0.04	2.896e+05	3.167e-16	1.535e-15	1.401e-18	6.789e-18	1.223e-18	5.927e-18	1.223e-20	5.927e-20
0.05	2.469e+06	1.040e-09	8.273e-09	2.771e-12	2.204e-11	2.419e-12	1.924e-11	2.419e-14	1.924e-13

0.06	4.473e+05	7.090e-08	7.437e-07	1.408e-10	1.477e-09	1.229e-10	1.290e-09	1.229e-12	1.290e-11
0.08	5.990e+07	1.481e-03	1.702e-02	2.344e-06	2.694e-05	2.047e-06	2.352e-05	2.047e-08	2.352e-07
0.1	6.179e+06	1.163e-03	1.215e-02	1.779e-06	1.859e-05	1.553e-06	1.623e-05	1.553e-08	1.623e-07
0.15	3.568e+06	4.523e-03	3.487e-02	7.448e-06	5.743e-05	6.502e-06	5.013e-05	6.502e-08	5.013e-07
0.2	4.026e+07	1.110e-01	7.037e-01	1.959e-04	1.242e-03	1.710e-04	1.084e-03	1.710e-06	1.084e-05
0.3	6.076e+07	3.845e-01	1.910e+00	7.294e-04	3.623e-03	6.368e-04	3.163e-03	6.368e-06	3.163e-05
0.4	7.902e+07	8.445e-01	3.568e+00	1.645e-03	6.953e-03	1.436e-03	6.070e-03	1.436e-05	6.070e-05
0.5	1.097e+07	1.736e-01	6.509e-01	3.408e-04	1.278e-03	2.975e-04	1.115e-03	2.975e-06	1.115e-05
0.6	1.069e+08	2.322e+00	7.902e+00	4.532e-03	1.542e-02	3.956e-03	1.346e-02	3.956e-05	1.346e-04
0.8	3.543e+07	1.264e+00	3.743e+00	2.405e-03	7.119e-03	2.099e-03	6.215e-03	2.099e-05	6.215e-05
1.0	1.140e+08	5.975e+00	1.598e+01	1.101e-02	2.945e-02	9.616e-03	2.571e-02	9.616e-05	2.571e-04
1.5	4.817e+07	5.056e+00	1.148e+01	8.506e-03	1.932e-02	7.426e-03	1.686e-02	7.426e-05	1.686e-04
2.0	5.398e+07	9.141e+00	1.893e+01	1.414e-02	2.927e-02	1.234e-02	2.555e-02	1.234e-04	2.555e-04
3.0	1.016e+07	3.286e+00	6.066e+00	4.458e-03	8.230e-03	3.892e-03	7.185e-03	3.892e-05	7.185e-05
Totals	**7.084e+08**	**2.857e+01**	**7.100e+01**	**4.797e-02**	**1.220e-01**	**4.188e-02**	**1.065e-01**	**4.188e-04**	**1.065e-03**

MicroShield 9.07
CB&I Federal Services-Mark O. Somerville (9.07-0000)

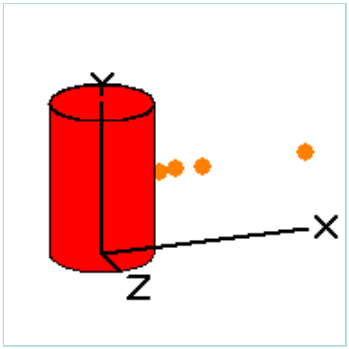
Date	By	Checked

Filename	Run Date	Run Time	Duration
Deer trails drum.msd	March 30, 2015	3:18:54 PM	00:00:01

Project Info	
Case Title	Deer Trails Drum
Description	222 pCi p gram Ra226 and 111 pCi p gram Ra228 drum
Geometry	7 - Cylinder Volume - Side Shields

Source Dimensions	
Height	88.9 cm (2 ft 11.0 in)
Radius	28.575 cm (11.3 in)

Dose Points			
A	X	Y	Z
#1	33.655 cm (1 ft 1.3 in)	44.45 cm (1 ft 5.5 in)	0.0 cm (0 in)
#2	43.815 cm (1 ft 5.3 in)	44.45 cm (1 ft 5.5 in)	0.0 cm (0 in)
#3	59.055 cm (1 ft 11.3 in)	44.45 cm (1 ft 5.5 in)	0.0 cm (0 in)
#4	120.015 cm (3 ft 11.3 in)	44.45 cm (1 ft 5.5 in)	0.0 cm (0 in)



Shields			
Shield N	Dimension	Material	Density
Source	2.28e+05 cm³	Compact Soil	1.6
Transition		Air	0.00122
Air Gap		Air	0.00122
Wall Clad	.122 cm	Iron	7.86
Top Clad	.122 cm	Iron	7.86

Source Input: Grouping Method - Standard Indices
Number of Groups: 25
Lower Energy Cutoff: 0.015
Photons < 0.015: Included
Library: Grove

Nuclide	Ci	Bq	µCi/cm³	Bq/cm³
Ac-228	3.5906e-005	1.3285e+006	1.5745e-004	5.8257e+000
Bi-210	2.3961e-006	8.8657e+004	1.0507e-005	3.8877e-001
Bi-212	1.1401e-005	4.2184e+005	4.9994e-005	1.8498e+000
Bi-214	8.0949e-005	2.9951e+006	3.5497e-004	1.3134e+001
Pb-210	2.4446e-006	9.0449e+004	1.0720e-005	3.9662e-001
Pb-212	1.1403e-005	4.2189e+005	5.0001e-005	1.8500e+000
Pb-214	8.0949e-005	2.9951e+006	3.5497e-004	1.3134e+001
Po-210	1.2762e-006	4.7219e+004	5.5961e-006	2.0706e-001
Po-212	7.3047e-006	2.7027e+005	3.2031e-005	1.1852e+000
Po-214	8.0932e-005	2.9945e+006	3.5489e-004	1.3131e+001
Po-216	1.1418e-005	4.2247e+005	5.0069e-005	1.8526e+000
Po-218	8.0965e-005	2.9957e+006	3.5504e-004	1.3136e+001
Ra-224	1.1418e-005	4.2247e+005	5.0069e-005	1.8526e+000
Ra-226	8.0965e-005	2.9957e+006	3.5504e-004	1.3136e+001
Ra-228	3.5902e-005	1.3284e+006	1.5743e-004	5.8249e+000

Rn-220	1.1418e-005	4.2247e+005	5.0069e-005	1.8526e+000
Rn-222	8.0965e-005	2.9957e+006	3.5504e-004	1.3136e+001
Th-228	1.1545e-005	4.2718e+005	5.0627e-005	1.8732e+000
Tl-208	4.0964e-006	1.5157e+005	1.7963e-005	6.6463e-001

Buildup: The material reference is Source Integration Parameters	
Radial	10
Circumferential	10
Y Direction (axial)	20

Results - Dose Point # 1 - (33.655,44.45,0) cm

Energy (MeV)	Activity (Photons/sec)	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup	Absorbed Dose Rate mrad/hr No Buildup	Absorbed Dose Rate mrad/hr With Buildup	Absorbed Dose Rate mGy/hr No Buildup	Absorbed Dose Rate mGy/hr With Buildup
0.015	1.131e+06	1.691e-32	9.721e-27	1.450e-33	8.338e-28	1.266e-33	7.279e-28	1.266e-35	7.279e-30
0.04	4.313e+03	2.321e-06	5.852e-06	1.026e-08	2.588e-08	8.960e-09	2.259e-08	8.960e-11	2.259e-10
0.05	3.677e+04	2.674e-04	8.645e-04	7.124e-07	2.303e-06	6.219e-07	2.010e-06	6.219e-09	2.010e-08
0.06	6.661e+03	1.929e-04	7.314e-04	3.831e-07	1.453e-06	3.345e-07	1.268e-06	3.345e-09	1.268e-08
0.08	8.920e+05	1.094e-01	4.755e-01	1.731e-04	7.524e-04	1.511e-04	6.569e-04	1.511e-06	6.569e-06
0.1	9.201e+04	2.375e-02	1.058e-01	3.634e-05	1.619e-04	3.172e-05	1.413e-04	3.172e-07	1.413e-06
0.15	5.313e+04	3.426e-02	1.409e-01	5.641e-05	2.320e-04	4.925e-05	2.025e-04	4.925e-07	2.025e-06
0.2	5.995e+05	6.324e-01	2.379e+00	1.116e-03	4.200e-03	9.744e-04	3.666e-03	9.744e-06	3.666e-05
0.3	9.048e+05	1.778e+00	5.814e+00	3.373e-03	1.103e-02	2.944e-03	9.628e-03	2.944e-05	9.628e-05
0.4	1.177e+06	3.546e+00	1.045e+01	6.910e-03	2.037e-02	6.032e-03	1.778e-02	6.032e-05	1.778e-04
0.5	1.634e+05	6.852e-01	1.861e+00	1.345e-03	3.654e-03	1.174e-03	3.190e-03	1.174e-05	3.190e-05
0.6	1.592e+06	8.752e+00	2.220e+01	1.708e-02	4.334e-02	1.491e-02	3.783e-02	1.491e-04	3.783e-04
0.8	5.276e+05	4.455e+00	1.022e+01	8.474e-03	1.944e-02	7.398e-03	1.697e-02	7.398e-05	1.697e-04
1.0	1.697e+06	2.004e+01	4.263e+01	3.694e-02	7.857e-02	3.225e-02	6.859e-02	3.225e-04	6.859e-04
1.5	7.173e+05	1.559e+01	2.919e+01	2.623e-02	4.912e-02	2.290e-02	4.288e-02	2.290e-04	4.288e-04
2.0	8.038e+05	2.673e+01	4.636e+01	4.134e-02	7.169e-02	3.609e-02	6.259e-02	3.609e-04	6.259e-04
3.0	1.513e+05	8.988e+00	1.412e+01	1.219e-02	1.915e-02	1.065e-02	1.672e-02	1.065e-04	1.672e-04
Totals	**1.055e+07**	**9.137e+01**	**1.860e+02**	**1.553e-01**	**3.217e-01**	**1.356e-01**	**2.809e-01**	**1.356e-03**	**2.809e-03**

Results - Dose Point # 2 - (43.815,44.45,0) cm

Energy (MeV)	Activity (Photons/sec)	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup	Absorbed Dose Rate mrad/hr No Buildup	Absorbed Dose Rate mrad/hr With Buildup	Absorbed Dose Rate mGy/hr No Buildup	Absorbed Dose Rate mGy/hr With Buildup
0.015	1.131e+06	1.742e-30	5.893e-27	1.494e-31	5.055e-28	1.304e-31	4.413e-28	1.304e-33	4.413e-30
0.04	4.313e+03	1.405e-06	3.506e-06	6.213e-09	1.550e-08	5.424e-09	1.354e-08	5.424e-11	1.354e-10
0.05	3.677e+04	1.669e-04	5.535e-04	4.446e-07	1.474e-06	3.881e-07	1.287e-06	3.881e-09	1.287e-08
0.06	6.661e+03	1.305e-04	5.159e-04	2.592e-07	1.025e-06	2.263e-07	8.945e-07	2.263e-09	8.945e-09
0.08	8.920e+05	7.761e-02	3.381e-01	1.228e-04	5.350e-04	1.072e-04	4.671e-04	1.072e-06	4.671e-06
0.1	9.201e+04	1.663e-02	7.173e-02	2.544e-05	1.097e-04	2.221e-05	9.580e-05	2.221e-07	9.580e-07
0.15	5.313e+04	2.317e-02	9.138e-02	3.816e-05	1.505e-04	3.331e-05	1.314e-04	3.331e-07	1.314e-06
0.2	5.995e+05	4.216e-01	1.529e+00	7.440e-04	2.699e-03	6.495e-04	2.356e-03	6.495e-06	2.356e-05

0.3	9.048e+05	1.171e+00	3.716e+00	2.221e-03	7.049e-03	1.939e-03	6.154e-03	1.939e-05	6.154e-05
0.4	1.177e+06	2.322e+00	6.668e+00	4.524e-03	1.299e-02	3.950e-03	1.134e-02	3.950e-05	1.134e-04
0.5	1.634e+05	4.469e-01	1.185e+00	8.772e-04	2.326e-03	7.658e-04	2.030e-03	7.658e-06	2.030e-05
0.6	1.592e+06	5.691e+00	1.411e+01	1.111e-02	2.755e-02	9.697e-03	2.405e-02	9.697e-05	2.405e-04
0.8	5.276e+05	2.884e+00	6.480e+00	5.485e-03	1.232e-02	4.788e-03	1.076e-02	4.788e-05	1.076e-04
1.0	1.697e+06	1.293e+01	2.696e+01	2.383e-02	4.969e-02	2.080e-02	4.338e-02	2.080e-04	4.338e-04
1.5	7.173e+05	9.988e+00	1.838e+01	1.680e-02	3.092e-02	1.467e-02	2.700e-02	1.467e-04	2.700e-04
2.0	8.038e+05	1.705e+01	2.909e+01	2.636e-02	4.499e-02	2.301e-02	3.928e-02	2.301e-04	3.928e-04
3.0	1.513e+05	5.698e+00	8.824e+00	7.730e-03	1.197e-02	6.748e-03	1.045e-02	6.748e-05	1.045e-04
Totals	**1.055e+07**	**5.871e+01**	**1.174e+02**	**9.987e-02**	**2.033e-01**	**8.718e-02**	**1.775e-01**	**8.718e-04**	**1.775e-03**

Results - Dose Point # 3 - (59.055,44.45,0) cm

Energy (MeV)	Activity (Photons/sec)	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup	Absorbed Dose Rate mrad/hr No Buildup	Absorbed Dose Rate mrad/hr With Buildup	Absorbed Dose Rate mGy/hr No Buildup	Absorbed Dose Rate mGy/hr With Buildup
0.015	1.131e+06	2.000e-30	3.387e-27	1.715e-31	2.905e-28	1.498e-31	2.536e-28	1.498e-33	2.536e-30
0.04	4.313e+03	1.022e-06	2.577e-06	4.519e-09	1.140e-08	3.945e-09	9.951e-09	3.945e-11	9.951e-11
0.05	3.677e+04	1.230e-04	4.040e-04	3.278e-07	1.076e-06	2.861e-07	9.395e-07	2.861e-09	9.395e-09
0.06	6.661e+03	9.079e-05	3.467e-04	1.803e-07	6.886e-07	1.574e-07	6.012e-07	1.574e-09	6.012e-09
0.08	8.920e+05	4.964e-02	2.082e-01	7.855e-05	3.294e-04	6.857e-05	2.876e-04	6.857e-07	2.876e-06
0.1	9.201e+04	1.032e-02	4.364e-02	1.579e-05	6.677e-05	1.379e-05	5.829e-05	1.379e-07	5.829e-07
0.15	5.313e+04	1.417e-02	5.538e-02	2.333e-05	9.120e-05	2.036e-05	7.962e-05	2.036e-07	7.962e-07
0.2	5.995e+05	2.570e-01	9.221e-01	4.535e-04	1.627e-03	3.959e-04	1.421e-03	3.959e-06	1.421e-05
0.3	9.048e+05	7.121e-01	2.225e+00	1.351e-03	4.221e-03	1.179e-03	3.685e-03	1.179e-05	3.685e-05
0.4	1.177e+06	1.409e+00	3.976e+00	2.746e-03	7.746e-03	2.397e-03	6.762e-03	2.397e-05	6.762e-05
0.5	1.634e+05	2.707e-01	7.043e-01	5.313e-04	1.382e-03	4.638e-04	1.207e-03	4.638e-06	1.207e-05
0.6	1.592e+06	3.441e+00	8.370e+00	6.716e-03	1.634e-02	5.863e-03	1.426e-02	5.863e-05	1.426e-04
0.8	5.276e+05	1.737e+00	3.831e+00	3.305e-03	7.287e-03	2.885e-03	6.361e-03	2.885e-05	6.361e-05
1.0	1.697e+06	7.765e+00	1.590e+01	1.431e-02	2.932e-02	1.249e-02	2.559e-02	1.249e-04	2.559e-04
1.5	7.173e+05	5.965e+00	1.080e+01	1.004e-02	1.818e-02	8.761e-03	1.587e-02	8.761e-05	1.587e-04
2.0	8.038e+05	1.014e+01	1.706e+01	1.568e-02	2.639e-02	1.369e-02	2.304e-02	1.369e-04	2.304e-04
3.0	1.513e+05	3.372e+00	5.162e+00	4.574e-03	7.004e-03	3.993e-03	6.114e-03	3.993e-05	6.114e-05
Totals	**1.055e+07**	**3.514e+01**	**6.927e+01**	**5.982e-02**	**1.200e-01**	**5.222e-02**	**1.047e-01**	**5.222e-04**	**1.047e-03**

Results - Dose Point # 4 - (120.015,44.45,0) cm

Energy (MeV)	Activity (Photons/sec)	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup	Absorbed Dose Rate mrad/hr No Buildup	Absorbed Dose Rate mrad/hr With Buildup	Absorbed Dose Rate mGy/hr No Buildup	Absorbed Dose Rate mGy/hr With Buildup
0.015	1.131e+06	8.894e-31	8.645e-28	7.629e-32	7.415e-29	6.660e-32	6.474e-29	6.660e-34	6.474e-31
0.04	4.313e+03	3.677e-07	9.092e-07	1.626e-09	4.021e-09	1.420e-09	3.510e-09	1.420e-11	3.510e-11
0.05	3.677e+04	3.715e-05	1.181e-04	9.896e-08	3.145e-07	8.639e-08	2.746e-07	8.639e-10	2.746e-09
0.06	6.661e+03	2.544e-05	9.525e-05	5.053e-08	1.892e-07	4.411e-08	1.652e-07	4.411e-10	1.652e-09
0.08	8.920e+05	1.345e-02	5.606e-02	2.129e-05	8.871e-05	1.859e-05	7.745e-05	1.859e-07	7.745e-07
0.1	9.201e+04	2.763e-03	1.148e-02	4.227e-06	1.756e-05	3.690e-06	1.533e-05	3.690e-08	1.533e-07
0.15	5.313e+04	3.714e-03	1.404e-02	6.116e-06	2.312e-05	5.339e-06	2.018e-05	5.339e-08	2.018e-07

0.2	5.995e+05	6.673e-02	2.319e-01	1.178e-04	4.093e-04	1.028e-04	3.573e-04	1.028e-06	3.573e-06
0.3	9.048e+05	1.832e-01	5.585e-01	3.476e-04	1.059e-03	3.034e-04	9.248e-04	3.034e-06	9.248e-06
0.4	1.177e+06	3.608e-01	9.982e-01	7.031e-04	1.945e-03	6.138e-04	1.698e-03	6.138e-06	1.698e-05
0.5	1.634e+05	6.910e-02	1.771e-01	1.356e-04	3.476e-04	1.184e-04	3.034e-04	1.184e-06	3.034e-06
0.6	1.592e+06	8.764e-01	2.107e+00	1.711e-03	4.112e-03	1.493e-03	3.590e-03	1.493e-05	3.590e-05
0.8	5.276e+05	4.414e-01	9.665e-01	8.395e-04	1.838e-03	7.329e-04	1.605e-03	7.329e-06	1.605e-05
1.0	1.697e+06	1.970e+00	4.019e+00	3.631e-03	7.408e-03	3.170e-03	6.467e-03	3.170e-05	6.467e-05
1.5	7.173e+05	1.511e+00	2.737e+00	2.542e-03	4.606e-03	2.220e-03	4.021e-03	2.220e-05	4.021e-05
2.0	8.038e+05	2.569e+00	4.328e+00	3.972e-03	6.692e-03	3.468e-03	5.843e-03	3.468e-05	5.843e-05
3.0	1.513e+05	8.550e-01	1.312e+00	1.160e-03	1.780e-03	1.013e-03	1.554e-03	1.013e-05	1.554e-05
Totals	**1.055e+07**	**8.922e+00**	**1.752e+01**	**1.519e-02**	**3.033e-02**	**1.326e-02**	**2.648e-02**	**1.326e-04**	**2.648e-04**

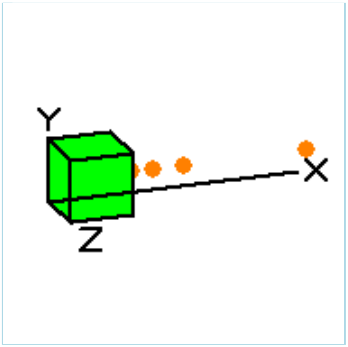
MicroShield 9.07
CB&I Federal Services-Mark O. Somerville (9.07-0000)

Date	By	Checked

Filename	Run Date	Run Time	Duration
Deer Trail s Blob.msd	March 30, 2015	3:21:01 PM	00:00:01

Project Info	
Case Title	Blob Rev2
Description	Now includes Ra-228
Geometry	13 - Rectangular Volume

Source Dimensions	
Length	30.48 cm (1 ft)
Width	30.48 cm (1 ft)
Height	30.48 cm (1 ft)



Dose Points			
A	X	Y	Z
#1	35.56 cm (1 ft 2.0 in)	15.24 cm (6.0 in)	15.24 cm (6.0 in)
#2	45.72 cm (1 ft 6.0 in)	15.24 cm (6.0 in)	15.24 cm (6.0 in)
#3	60.96 cm (2 ft)	15.24 cm (6.0 in)	15.24 cm (6.0 in)
#4	121.92 cm (4 ft)	15.24 cm (6.0 in)	15.24 cm (6.0 in)

Shields			
Shield N	Dimension	Material	Density
Source	2.83e+04 cm³	Loose Soil	1.2
Air Gap		Air	0.00122

Source Input: Grouping Method - Standard Indices
Number of Groups: 25
Lower Energy Cutoff: 0.015
Photons < 0.015: Included
Library: Grove

Nuclide	Ci	Bq	µCi/cm³	Bq/cm³
Ac-228	4.4585e-006	1.6496e+005	1.5745e-004	5.8257e+000
Bi-210	2.9753e-007	1.1009e+004	1.0507e-005	3.8877e-001
Bi-212	1.4157e-006	5.2380e+004	4.9994e-005	1.8498e+000
Bi-214	1.0052e-005	3.7191e+005	3.5497e-004	1.3134e+001
Pb-210	3.0354e-007	1.1231e+004	1.0720e-005	3.9662e-001
Pb-212	1.4159e-006	5.2387e+004	5.0001e-005	1.8500e+000
Pb-214	1.0052e-005	3.7191e+005	3.5497e-004	1.3134e+001
Po-210	1.5846e-007	5.8632e+003	5.5961e-006	2.0706e-001
Po-212	9.0703e-007	3.3560e+004	3.2031e-005	1.1852e+000
Po-214	1.0049e-005	3.7183e+005	3.5489e-004	1.3131e+001
Po-216	1.4178e-006	5.2459e+004	5.0069e-005	1.8526e+000
Po-218	1.0054e-005	3.7198e+005	3.5504e-004	1.3136e+001
Ra-224	1.4178e-006	5.2459e+004	5.0069e-005	1.8526e+000
Ra-226	1.0053e-005	3.7198e+005	3.5504e-004	1.3136e+001
Ra-228	4.4579e-006	1.6494e+005	1.5743e-004	5.8249e+000
Rn-220	1.4178e-006	5.2459e+004	5.0069e-005	1.8526e+000
Rn-222	1.0054e-005	3.7198e+005	3.5504e-004	1.3136e+001

Th-228	1.4336e-006	5.3043e+004	5.0627e-005	1.8732e+000
Tl-208	5.0865e-007	1.8820e+004	1.7963e-005	6.6463e-001

Buildup: The material reference is Source Integration Parameters	
X Direction	10
Y Direction	20
Z Direction	20

Results - Dose Point # 1 - (35.56,15.24,15.24) cm

Energy (MeV)	Activity (Photons/sec)	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup	Absorbed Dose Rate mrad/hr No Buildup	Absorbed Dose Rate mrad/hr With Buildup	Absorbed Dose Rate mGy/hr No Buildup	Absorbed Dose Rate mGy/hr With Buildup
0.015	1.404e+05	4.272e-05	4.607e-05	3.665e-06	3.952e-06	3.199e-06	3.450e-06	3.199e-08	3.450e-08
0.04	5.356e+02	3.018e-04	4.589e-04	1.335e-06	2.030e-06	1.165e-06	1.772e-06	1.165e-08	1.772e-08
0.05	4.566e+03	5.171e-03	9.527e-03	1.377e-05	2.538e-05	1.203e-05	2.216e-05	1.203e-07	2.216e-07
0.06	8.271e+02	1.524e-03	3.305e-03	3.027e-06	6.565e-06	2.642e-06	5.731e-06	2.642e-08	5.731e-08
0.08	1.108e+05	3.812e-01	9.780e-01	6.032e-04	1.548e-03	5.266e-04	1.351e-03	5.266e-06	1.351e-05
0.1	1.143e+04	5.806e-02	1.572e-01	8.883e-05	2.405e-04	7.755e-05	2.100e-04	7.755e-07	2.100e-06
0.15	6.597e+03	6.057e-02	1.588e-01	9.974e-05	2.615e-04	8.708e-05	2.283e-04	8.708e-07	2.283e-06
0.2	7.444e+04	9.995e-01	2.439e+00	1.764e-03	4.304e-03	1.540e-03	3.757e-03	1.540e-05	3.757e-05
0.3	1.124e+05	2.535e+00	5.472e+00	4.809e-03	1.038e-02	4.198e-03	9.061e-03	4.198e-05	9.061e-05
0.4	1.461e+05	4.754e+00	9.389e+00	9.263e-03	1.829e-02	8.087e-03	1.597e-02	8.087e-05	1.597e-04
0.5	2.029e+04	8.764e-01	1.617e+00	1.720e-03	3.174e-03	1.502e-03	2.771e-03	1.502e-05	2.771e-05
0.6	1.977e+05	1.076e+01	1.878e+01	2.101e-02	3.665e-02	1.834e-02	3.199e-02	1.834e-04	3.199e-04
0.8	6.551e+04	5.137e+00	8.281e+00	9.770e-03	1.575e-02	8.530e-03	1.375e-02	8.530e-05	1.375e-04
1.0	2.108e+05	2.192e+01	3.336e+01	4.040e-02	6.150e-02	3.527e-02	5.369e-02	3.527e-04	5.369e-04
1.5	8.907e+04	1.541e+01	2.143e+01	2.593e-02	3.605e-02	2.263e-02	3.147e-02	2.263e-04	3.147e-04
2.0	9.981e+04	2.460e+01	3.249e+01	3.805e-02	5.024e-02	3.322e-02	4.386e-02	3.322e-04	4.386e-04
3.0	1.878e+04	7.530e+00	9.350e+00	1.022e-02	1.268e-02	8.918e-03	1.107e-02	8.918e-05	1.107e-04
Totals	**1.310e+06**	**9.503e+01**	**1.439e+02**	**1.637e-01**	**2.511e-01**	**1.429e-01**	**2.192e-01**	**1.429e-03**	**2.192e-03**

Results - Dose Point # 2 - (45.72,15.24,15.24) cm

Energy (MeV)	Activity (Photons/sec)	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup	Absorbed Dose Rate mrad/hr No Buildup	Absorbed Dose Rate mrad/hr With Buildup	Absorbed Dose Rate mGy/hr No Buildup	Absorbed Dose Rate mGy/hr With Buildup
0.015	1.404e+05	3.893e-05	4.195e-05	3.339e-06	3.598e-06	2.915e-06	3.141e-06	2.915e-08	3.141e-08
0.04	5.356e+02	1.382e-04	2.049e-04	6.110e-07	9.064e-07	5.334e-07	7.913e-07	5.334e-09	7.913e-09
0.05	4.566e+03	2.314e-03	4.175e-03	6.164e-06	1.112e-05	5.382e-06	9.710e-06	5.382e-08	9.710e-08
0.06	8.271e+02	6.737e-04	1.432e-03	1.338e-06	2.844e-06	1.168e-06	2.483e-06	1.168e-08	2.483e-08
0.08	1.108e+05	1.664e-01	4.196e-01	2.634e-04	6.640e-04	2.299e-04	5.796e-04	2.299e-06	5.796e-06
0.1	1.143e+04	2.521e-02	6.729e-02	3.858e-05	1.029e-04	3.368e-05	8.987e-05	3.368e-07	8.987e-07
0.15	6.597e+03	2.618e-02	6.799e-02	4.312e-05	1.120e-04	3.764e-05	9.774e-05	3.764e-07	9.774e-07
0.2	7.444e+04	4.314e-01	1.047e+00	7.613e-04	1.848e-03	6.647e-04	1.613e-03	6.647e-06	1.613e-05
0.3	1.124e+05	1.093e+00	2.359e+00	2.073e-03	4.474e-03	1.810e-03	3.906e-03	1.810e-05	3.906e-05
0.4	1.461e+05	2.049e+00	4.058e+00	3.991e-03	7.906e-03	3.485e-03	6.902e-03	3.485e-05	6.902e-05

0.5	2.029e+04	3.776e-01	7.000e-01	7.413e-04	1.374e-03	6.471e-04	1.199e-03	6.471e-06	1.199e-05
0.6	1.977e+05	4.639e+00	8.139e+00	9.055e-03	1.589e-02	7.905e-03	1.387e-02	7.905e-05	1.387e-04
0.8	6.551e+04	2.215e+00	3.595e+00	4.213e-03	6.838e-03	3.678e-03	5.970e-03	3.678e-05	5.970e-05
1.0	2.108e+05	9.458e+00	1.450e+01	1.743e-02	2.672e-02	1.522e-02	2.333e-02	1.522e-04	2.333e-04
1.5	8.907e+04	6.661e+00	9.325e+00	1.121e-02	1.569e-02	9.783e-03	1.370e-02	9.783e-05	1.370e-04
2.0	9.981e+04	1.065e+01	1.416e+01	1.647e-02	2.189e-02	1.438e-02	1.911e-02	1.438e-04	1.911e-04
3.0	1.878e+04	3.266e+00	4.079e+00	4.431e-03	5.534e-03	3.868e-03	4.831e-03	3.868e-05	4.831e-05
Totals	**1.310e+06**	**4.106e+01**	**6.251e+01**	**7.073e-02**	**1.091e-01**	**6.175e-02**	**9.521e-02**	**6.175e-04**	**9.521e-04**

Results - Dose Point # 3 - (60.96,15.24,15.24) cm

Energy (MeV)	Activity (Photons/sec)	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup	Absorbed Dose Rate mrad/hr No Buildup	Absorbed Dose Rate mrad/hr With Buildup	Absorbed Dose Rate mGy/hr No Buildup	Absorbed Dose Rate mGy/hr With Buildup
0.015	1.404e+05	2.287e-05	2.460e-05	1.961e-06	2.110e-06	1.712e-06	1.842e-06	1.712e-08	1.842e-08
0.04	5.356e+02	5.317e-05	7.856e-05	2.351e-07	3.475e-07	2.053e-07	3.033e-07	2.053e-09	3.033e-09
0.05	4.566e+03	8.947e-04	1.624e-03	2.383e-06	4.327e-06	2.081e-06	3.778e-06	2.081e-08	3.778e-08
0.06	8.271e+02	2.619e-04	5.632e-04	5.201e-07	1.119e-06	4.541e-07	9.766e-07	4.541e-09	9.766e-09
0.08	1.108e+05	6.528e-02	1.688e-01	1.033e-04	2.671e-04	9.018e-05	2.332e-04	9.018e-07	2.332e-06
0.1	1.143e+04	9.953e-03	2.744e-02	1.523e-05	4.197e-05	1.329e-05	3.664e-05	1.329e-07	3.664e-07
0.15	6.597e+03	1.043e-02	2.822e-02	1.717e-05	4.647e-05	1.499e-05	4.057e-05	1.499e-07	4.057e-07
0.2	7.444e+04	1.727e-01	4.382e-01	3.048e-04	7.734e-04	2.661e-04	6.752e-04	2.661e-06	6.752e-06
0.3	1.124e+05	4.406e-01	9.941e-01	8.357e-04	1.886e-03	7.296e-04	1.646e-03	7.296e-06	1.646e-05
0.4	1.461e+05	8.303e-01	1.716e+00	1.618e-03	3.343e-03	1.412e-03	2.918e-03	1.412e-05	2.918e-05
0.5	2.029e+04	1.537e-01	2.965e-01	3.017e-04	5.820e-04	2.634e-04	5.081e-04	2.634e-06	5.081e-06
0.6	1.977e+05	1.895e+00	3.452e+00	3.699e-03	6.737e-03	3.229e-03	5.882e-03	3.229e-05	5.882e-05
0.8	6.551e+04	9.101e-01	1.527e+00	1.731e-03	2.904e-03	1.511e-03	2.536e-03	1.511e-05	2.536e-05
1.0	2.108e+05	3.904e+00	6.166e+00	7.196e-03	1.137e-02	6.282e-03	9.923e-03	6.282e-05	9.923e-05
1.5	8.907e+04	2.772e+00	3.978e+00	4.664e-03	6.693e-03	4.072e-03	5.843e-03	4.072e-05	5.843e-05
2.0	9.981e+04	4.457e+00	6.050e+00	6.892e-03	9.356e-03	6.016e-03	8.168e-03	6.016e-05	8.168e-05
3.0	1.878e+04	1.376e+00	1.747e+00	1.866e-03	2.370e-03	1.629e-03	2.069e-03	1.629e-05	2.069e-05
Totals	**1.310e+06**	**1.700e+01**	**2.659e+01**	**2.925e-02**	**4.637e-02**	**2.553e-02**	**4.049e-02**	**2.553e-04**	**4.049e-04**

Results - Dose Point # 4 - (121.92,15.24,15.24) cm

Energy (MeV)	Activity (Photons/sec)	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup	Absorbed Dose Rate mrad/hr No Buildup	Absorbed Dose Rate mrad/hr With Buildup	Absorbed Dose Rate mGy/hr No Buildup	Absorbed Dose Rate mGy/hr With Buildup
0.015	1.404e+05	3.517e-06	3.779e-06	3.016e-07	3.242e-07	2.633e-07	2.830e-07	2.633e-09	2.830e-09
0.04	5.356e+02	7.220e-06	1.079e-05	3.193e-08	4.770e-08	2.788e-08	4.164e-08	2.788e-10	4.164e-10
0.05	4.566e+03	1.238e-04	2.300e-04	3.299e-07	6.127e-07	2.880e-07	5.349e-07	2.880e-09	5.349e-09
0.06	8.271e+02	3.686e-05	8.218e-05	7.322e-08	1.632e-07	6.392e-08	1.425e-07	6.392e-10	1.425e-09
0.08	1.108e+05	9.426e-03	2.588e-02	1.492e-05	4.095e-05	1.302e-05	3.575e-05	1.302e-07	3.575e-07
0.1	1.143e+04	1.460e-03	4.341e-03	2.234e-06	6.642e-06	1.951e-06	5.798e-06	1.951e-08	5.798e-08
0.15	6.597e+03	1.563e-03	4.640e-03	2.574e-06	7.641e-06	2.247e-06	6.671e-06	2.247e-08	6.671e-08
0.2	7.444e+04	2.620e-02	7.312e-02	4.624e-05	1.291e-04	4.037e-05	1.127e-04	4.037e-07	1.127e-06
0.3	1.124e+05	6.791e-02	1.679e-01	1.288e-04	3.184e-04	1.125e-04	2.780e-04	1.125e-06	2.780e-06

0.4	1.461e+05	1.295e-01	2.912e-01	2.523e-04	5.675e-04	2.202e-04	4.954e-04	2.202e-06	4.954e-06
0.5	2.029e+04	2.419e-02	5.049e-02	4.748e-05	9.911e-05	4.145e-05	8.653e-05	4.145e-07	8.653e-07
0.6	1.977e+05	3.005e-01	5.891e-01	5.865e-04	1.150e-03	5.120e-04	1.004e-03	5.120e-06	1.004e-05
0.8	6.551e+04	1.461e-01	2.615e-01	2.778e-04	4.975e-04	2.425e-04	4.343e-04	2.425e-06	4.343e-06
1.0	2.108e+05	6.324e-01	1.059e+00	1.166e-03	1.952e-03	1.018e-03	1.704e-03	1.018e-05	1.704e-05
1.5	8.907e+04	4.566e-01	6.866e-01	7.682e-04	1.155e-03	6.706e-04	1.008e-03	6.706e-06	1.008e-05
2.0	9.981e+04	7.418e-01	1.048e+00	1.147e-03	1.620e-03	1.001e-03	1.414e-03	1.001e-05	1.414e-05
3.0	1.878e+04	2.320e-01	3.039e-01	3.147e-04	4.123e-04	2.747e-04	3.599e-04	2.747e-06	3.599e-06
Totals	**1.310e+06**	**2.770e+00**	**4.566e+00**	**4.755e-03**	**7.958e-03**	**4.151e-03**	**6.947e-03**	**4.151e-05**	**6.947e-05**

STANDARD OPERATING PROCEDURE

15.WAC.04

OIL AND GAS E&P NORM WASTE ACCEPTANCE

1.0 OBJECTIVE

To define alternate methods and procedures for characterization and acceptance of oil and gas production waste proposed for acceptance at the Clean Harbors Deer Trail (CHDT) facility comply with the Adams County Amended Certificate of Designation (Amended CD) and the Colorado Department of Public Health and Environment (CDPHE) Radioactive Materials License.

2.0 SCOPE

This SOP defines the survey process and exposure rate acceptance criteria for receipt of oil and gas production waste containing Naturally Occurring Radioactive Material (NORM) and Technologically Enhanced Naturally Occurring Radioactive Materials (TENORM). The acceptance criteria contained in this procedure will be used be provide assurance that materials accepted at the facility conform to the waste acceptance criteria. This procedure may be used for sludges and filters, and similar wastes but may not be used for pipes and process equipment.

3.0 POLICY

Oil and gas production wastes will be evaluated for acceptability at CHDT according to the procedures in this SOP. Radiation surveys of the waste shall be conducted to ensure that average contamination levels do not exceed the CHDT waste acceptance criteria. CHDT can accept oil and gas production waste containing NORM/TENORM with a total activity of less than 2,000 pCi/g (natural uranium and thorium decay chain products only), and with a maximum ^{226}Ra concentration of less than 222 pCi/g. Generators following this procedure are required to complete the standard CHDT Waste Profile and Supplemental NORM questionnaire provided in 15.WAC.01, Waste Acceptance, to certify that no man-made or other unacceptable radionuclides are present and that generating process and method of characterization is specified.

4.0 RESPONSIBILTITES OF CHDT

Responsibilities of the CHDT Radiation Safety Officer (“RSO”), management, and staff are defined in the CHDT Radiation Protection Plan (SOP 15.RPP.01). Surveys at generators sites may be performed by the generator, Clean Harbors on behalf of the generator, or an independent party, such as a licensed waste broker. Acceptance surveys at CHDT will be performed by trained CHDT personnel.

5.0 PRE-ACCEPTANCE REQUIREMENTS

Any waste generator proposing to send oil or gas production waste to CHDT shall document the radiological properties. The waste generator or authorized representative

shall provide pre-acceptance survey results and documentation in accordance with the following requirements:

5.1 Waste Documentation

5.1.1 Waste Material Profile Sheet

The waste generator shall submit a Waste Material Profile Sheet to CHDT. The waste generator may submit other forms or supporting documents as long as the data and information required in the Waste Profile are fully and completely provided in such alternate documents.

5.1.2 Supplemental NORM Questionnaire

The waste generator shall also submit a Supplemental NORM Questionnaire to CHDT. The waste generator shall complete all requested information and certify by signature of a responsible official that the information is complete. The waste generator shall provide supporting documents to support the Supplemental NORM Questionnaire. The waste generator shall document the characterization method and sampling/survey protocol for the wastes and the generating process. Other forms or documentation may be used as long as the required information is provided.

5.1.3 Oil and Gas NORM Field Survey Report

The waste generator or authorized representative shall attach to the Questionnaire a signed and complete Oil and Gas NORM Field Survey Report or equivalent. The Survey Report shall support and confirm the description of radionuclides set forth and described in the Waste Profile and the Questionnaire by demonstrating that exposure rates are within the acceptance criteria of this SOP. The Survey Report shall serve as a representative sample of the waste for acceptance purposes. Photo-documentation of survey meter readings, calibration dates and waste materials is strongly encouraged.

5.2 Pre-Acceptance Evaluation

5.2.1 CHDT Review of Characterization

Before any oil and gas production waste is approved for acceptance, the RSO shall ensure that such waste's radiological, physical, and chemical characteristics meet the sire acceptance criteria. CHDT shall use the Waste Material Profile Sheet, the Supplemental NORM Questionnaire, Survey Reports and any other supporting information received from the waste generator such as photographs of instrument readings, calibration stickers and the waste, to confirm that the oil and gas production waste is acceptable for disposal. Properly documented oil and gas production waste that has exposure rates meeting the criteria in Section 7 of this SOP are assumed to meet CHDT waste acceptance criteria.

5.2.2 Documentation of Acceptance

If the oil and gas production waste is confirmed to be appropriate for acceptance at the facility, the waste stream will be approved. CHDT shall maintain all documents, data, and information that it used to make the determination and include the information and documents with the Waste Profile. A copy of the approval, and the supporting documentation, will be kept on file and be available for review.

5.2.3 Annual Recertification

At the one year anniversary of the date of approval, the waste stream shall be reevaluated for compliance with this the Amended CD and the Radioactive Materials License ("Annual Recertification"). A generator may certify by re-signature of the Waste Materials Profile Sheet and Supplemental NORM Questionnaire that the original characterization is still complete and still applicable and the oil and gas production waste acceptance can continue. The waste generator's certification shall be evaluated by CHDT in the same manner as before. If CHDT re-confirms that the waste stream meets the requirements of the Amended CD and the Radioactive Materials License, the oil and gas production waste may be re-approved for acceptance. All documentation for any Annual Recertification of an oil and gas production waste or waste stream will be kept on file and be available for inspection.

6.0 CHDT ON-SITE ACCEPTANCE REQUIREMENTS

6.1.1 Gamma Isotopic Identification

Upon arrival at CHDT, each shipment of oil and gas waste will be analyzed for isotopic identification. Shipments of waste will be surveyed for radionuclide identification using an approved portable gamma spectroscopy unit per SOP 15.OPS.03, Operation of Portable Gamma Spectroscopy Unit. If man-made radionuclides are detected above background, the waste shipment shall be rejected and returned to the Generator or alternate facility.

6.1.2 Exposure Rate Survey

All shipments of oil and gas production wastes received at the Facility for acceptance and disposal shall be screened for radiation exposure rate at the outside of the shipment package. Exposure rate meters shall be operated in accordance with License SOP 15.OPS.07, Exposure/Dose Rate Meters. Criteria for acceptance or rejection using exposure rate surveys are detailed in Section 7.0 of this SOP. In addition, contamination surveys will be performed on the received waste materials per SOP 15.OPS.21 (Package Receipt Surveys) as required. If exposure rate surveys determine that the waste is above CHDT acceptance limits, the waste shipment shall be rejected and returned to the Generator or alternate facility. Alternatively, other means such as laboratory analysis may be used in order to confirm that the shipment is in compliance.

6.1.3 Documentation Review

All documents accompanying the oil production pipe waste shipment will be reviewed and kept on file to ensure that the requirements of CHDT are met.

6.1.4 On-Site Tracking

CHDT shall maintain data, information and other documentation that conforms to industry standards and prudent practices for on-site waste shipments and the disposal location of all wastes. Such information shall document and track the location and final disposition of each oil production pipe waste shipment.

7.0 PROCEDURE FOR SURVEYING AND APPLYING ACCEPTANCE CRITERIA

7.1 Overview

Oil and gas production waste shall be tested and analyzed as indicated below to assure the waste meets the Waste Acceptance Criteria and to ensure compliance with the requirements of the Amended CD and this SOP. If the oil and gas production waste shipment meets the measurement criteria stated below, the waste may be accepted for treatment, storage and disposal at CHDT.

7.2 Bulk Truck, Roll-off, or Steel Box Survey

Oil and gas production waste, while essentially homogenous in content will display variations in concentration. Concentration of radioactive materials will directly affect measured exposure rates. Distance from the source allows the detector to view a greater surface area and, therefore, tends to obscure local variations. For that reason, **survey** the truck, roll-off container, or box **at contact** using a meter capable of **detecting and displaying at least 500 micro-R/hour** (0.5 millirem/hour).

7.2.1 First Check

Check the status of the gate monitor to determine if it has alarmed. **IF** the gate monitor **ALARMED** as the material in question entered the site, **THEN** consider rejecting the material. This procedure may be exited. **IF** the gate alarm **DID NOT ALARM, PROCEED to the next step.**

7.2.2 Second Check

Survey the truck, roll-off, or steel box at contact. Trucks or roll-offs often have ribs. Scan each side, end, and bottom (if accessible). Collect exposure rates for acceptance determination from the non-ribbed sections of the truck or roll-off. IF no exposure rate is found to exceed **300 micro-R/hour, THEN** the material conforms to the acceptance criterion.

7.2.3 Additional Information for Second Check

IF exposure rates exceed 3**00 micro-R/hour *but* do not exceed 330 micro-R hour**, the material **MEETS** the acceptance criterion. Exposure rates in excess of 330 micro-R/hour should be identified to RSO for decision making about averaging higher and lower exposure rates and acceptance or rejection of the material.

7.3 Small or Individual Unshielded Items

Individual items should be surveyed at **1 foot** for contamination control purposes. **IF** individual, small, unshielded items present exposure rates less than **45 micro-R/hour** above the prevailing background, **THEN** the material meets the acceptance criterion.

8.0 ENFORCEMENT POLICY

If exposure rates exceed the acceptance levels shown in Section 7.0, then the oil and gas production waste shall be rejected. The waste may alternatively be analyzed and characterized using the procedures in 15.WAC.01, using laboratory analysis. The RSO may, based on actual survey data, average exposure rate readings to account for a small number (e.g., 10%) of elevated locations in an otherwise conforming shipment.

ATTACHMENT A

Supplemental NORM Questionnaire

Supplemental NORM Questionnaire

Clean Harbors Deer Trail Facility

General Information

Waste Name: _____

Profile Number: _____

Generator Name: _____

EPA ID: _____

Site Address: _____

Regulatory Information

Is the Waste Subject to an NRC, State or other Radioactive Materials License?	Yes ☐	No ☐
Does the Waste contain NORM (Naturally Occurring Radioactive Material)?	Yes ☐	No ☐
Does the Waste contain TENORM (Techologically Enhanced NORM)?	Yes ☐	No ☐
Does the Waste contain Source Material?	Yes ☐	No ☐
Does the Waste Contain Byproduct Material?	Yes ☐	No ☐
Does the Waste Contain Man Made or Artificial Radionuclides?	Yes ☐	No ☐
Does the Waste Contain Special Nuclear Material?	Yes ☐	No ☐
Please describe the process which produced the waste:		

Radiological Information

What is the maximum dose rate at the surface of the waste (microR/hour)? _____

Please list the isotope, average specific activity, and maximum specific activity for each radionuclide present

Isotope	Ave (pCi/g)	Max (pCi/g)	Isotope	Ave (pCi/g)	Max (pCi/g)

Generators Certification

I hereby certify that all information submitted in this and attached documents is correct to the best of my knowledge. I also certify that any samples submitted are representative of the actual waste. I also certify that the Specific Acitivity (radioactivity per gram) of waste prepared for shipment to the facility meets the acceptance criteria for Regulated Waste at the Facility.

Authorized Signature	Name (Print)	Title	Date
_____	_____	_____	_____

Generators Outside the Rocky Mtn. Compact Region (CO, NM, NV) Certification

I hereby certify that the generator is licensed or permitted to, and agrees to, receive back the waste and residues if they cannot be disposed of. (In the case of oil and gas waste collectors/processors, this certification may be signed by the processor/collector.) I agree to provide the Clean Harbors Deer Trail Facility written or e-mail notification at least five days in advance of the initial shipment of waste.

Authorized Signature	Name (Print)	Title	Date
_____	_____	_____	_____

Clean Harbors Deer Trail, LLC
108555 East Highway 36,
Deer Trail, CO 80105
Phone 970-386-2293, Fax 970-386-2262

ATTACHMENT B

Oil and Gas NORM Field Survey Report Form

Oil and Gas NORM Field Survey Report

Clean Harbors Deer Trail Facility

General Information

Waste Name: _____

Profile Number: _____

Generator Name: _____

EPA ID: _____

Site Address: _____

Container ID: _____

Did the gate monitor alarm? Yes ☐ No ☐ N/A ☐

Survey Instrument Information

µR instrument Make:	Model:	Serial No.	
Calibrated within the last year?		Yes ☐	No ☐
Battery/Instrument check OK?		Yes ☐	No ☐

µR instrument Make:	Model:	Serial No.	
Calibrated within the last year?		Yes ☐	No ☐
Battery/Instrument check OK?		Yes ☐	No ☐

Survey Results

What is the maximum dose rate at the surface of the container (µR/hour)? _____

Location	Average	Maximum

Location	Average	Maximum

Diagram

Does the container pass 15.WAC.04? Yes ☐ No ☐

Surveyor Signature | Name (Print) | Title | Date

Clean Harbors Deer Trail, LLC
108555 East Highway 36,
Deer Trail, CO 80105
Phone 970-386-2293, Fax 970-386-2262