Nov 20, 1998

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ENGINEERING DATA TRANSMITTAL

Page 1 of 1
1. EDT 625900

2. To: (Receiving Organization): Distribution		3. From: (Originating Organization) BWHC/300 Area Stabilization Proj.		4. Related EDT No.: NA	
5. Proj./Prog./Dept./Div.: 300 Area Stabilization Project		6. Design Authority/Design Agent/Cog. Engr.: JG Riddelle		7. Purchase Order No.: NA	
8. Originator Remarks: Release of HNF-3636, Rev. 0, "Technical Basis for Characterization of B-Cell Waste for Shipment in the 3-82B Shipping Cask"				9. Equip./Component No.: NA	
				10. System/Bldg./Facility: 324 Facility	
				12. Major Assm. Dwg. No.: NA	
11. Receiver Remarks: This supporting documents the historical technical basis for shipping 324 Facility B-Cell low-level waste.		11A. Design Baseline Document? ☐ Yes ☒ No		13. Permit/Permit Application No.: NA	
				14. Required Response Date: 11/18/98	

15. DATA TRANSMITTED					(F)	(G)	(H)	(I)
(A) Item No.	(B) Document/Drawing No.	(C) Sheet No.	(D) Rev. No.	(E) Title or Description of Data Transmitted	Approval Designator	Reason for Transmittal	Originator Disposition	Receiver Disposition
1	HNF-3636	All	0	Technical Basis for Char..	NA	2	1	1

16. KEY					
Approval Designator (F)		Reason for Transmittal (G)		Disposition (H) & (I)	
E, S, Q, D OR N/A (See WHC-CM-3-5, Sec. 12.7)		1. Approval 2. Release 3. Information 4. Review 5. Post-Review 6. Dist. (Receipt Acknow. Required)		1. Approved 2. Approved w/comment 3. Disapproved w/comment 4. Reviewed no/comment 5. Reviewed w/comment 6. Receipt acknowledged	

17. SIGNATURE/DISTRIBUTION (See Approval Designator for required signatures)											
(G) Reason	(H) Disp.	(J) Name	(K) Signature	(L) Date	(M) MSIN	(G) Reason	(H) Disp.	(J) Name	(K) Signature	(L) Date	(M) MSIN
		Design Authority				3		RL HILL		11/18/98	LI-08
		Design Agent									
		Cog. Eng. (Same as Cog. Mgr., below)									
1	I	Cog. Mgr. JG Riddelle		11/18/98	1-02						
		QA				3		Central Files			B1-07
		Safety				3		DOE Reading Room			H2-53
		Env.									

18. JG Riddelle Signature of EDT Originator Date 11/18/98		19. NA Authorized Representative for Receiving Organization Date		20. JG Riddelle Design Authority/Cognizant Manager Date 11/18/98		21. DOE APPROVAL (If required) Ctrl No. NA ☐ Approved ☐ Approved w/comments ☐ Disapproved w/comments	
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TECHNICAL BASIS FOR EXEMPTION FROM ALPHA SURVEYS FOR PERSONNEL, MATERIAL, AND EQUIPMENT IN THE 324 FACILITY

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U.S. Department of Energy Contract DE-AC06-96RL13200

EDT/ECN: 625900

UC: 2000

Org Code: 19300

Charge Code: 101031/GA00 HN990151

B&R Code: NA

Total Pages: 39 41

Key Words: B-Cell Waste; Alpha; Transuranic; Cesium; Strontium; Shipping Cask.

Abstract: This report documents the technical basis for the measurements and analyses used to characterize the 324Facility grouted B-Cell waste for disposal at the Hanford Burial Grounds using the 3-82B shipping cask.

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James Aardal 11-20-98
Release Approval Date

Release Stamp

Approved for Public Release

RELEASE AUTHORIZATION

Document Number: HNF-3636, Rev. 0

Document Title: Technical Basis for Exemption from Alpha Surveys for Personnel, Material, and Equipment in the 324 Facility

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11-20-78

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CONTENTS

I.	Purpose	1
II.	Background	1
III.	Applicable Limits for Disposal of B Cell Waste Packages	3
IV.	Methodology for Determining the Radionuclide Content in a B Cell Waste Container	4
	A. Dose Rate Measurements	4
	B. Estimating the Cs Activity of the Grout Container	4
	C. Determining ^{90}Sr and Alpha Activities	6
	D. Establishing the Waste Designation	9
	E. Establishing the Heat Generation Rate	10
	F. Loading Limits for Grout Containers	10
V.	Determination of the Maximum Activity of a LLW Package	10
VI.	Dose Rate Limits During Shipping Using the 3-82B Shipping Cask	11
VII.	Additional Source Term Characterization	11
VIII.	Conclusions	12
IX.	References	12
Appendix A.		14
Appendix B.		16
Appendix C.		19
Appendix D.		34

FIGURES AND TABLES

Figure 1.	Linear Regression Fit of the Alpha and ^{137}Cs Activity Data	7
Figure 2.	Linear Regression Fit of the ^{90}Sr and ^{137}Cs Activity Data for Both the 1993 and 1998 Smear Data	8
Figure 3.	Linear Regression Fit of the ^{90}Sr and ^{137}Cs Activity Data for the 1993 Smear Data	8
Table 1.	Results of Source Geometry Comparison	5
Table 2.	Results of Smears ($\mu\text{Ci/sample}$) Analyzed in 1993 and 1998	6
Table 3.	Results of Alpha Energy Analysis for 1998 Smear Samples	9

TECHNICAL BASIS FOR CHARACTERIZATION OF B-CELL WASTE FOR SHIPMENT IN THE 3-82B SHIPPING CASK

I Purpose

The purpose of this document is to establish the technical basis for characterizing grouted B-Cell waste for disposal at the Hanford Burial Grounds using the 3-82B shipping cask. The scope of this document includes establishing the technical basis for loading the shipping package, an HN-200 Grout Container, to ensure that: 1) the amount of material in the grout container does not exceed the 100 nCi alpha/g limit that would cause the waste to be designated as "greater than Category 3" (GC3) or transuranic (TRU) waste; 2) the amount of heat generated by the waste in the grout container does not exceed the 60 Watt heat generation limit established in the 3-82B shipping cask Safety Analysis Report (SAR); and 3) the dose rate on the surface of the shipping cask after loading does not exceed the 200 mrem/h limit established in the cask SAR. This document establishes the technical basis for performing measurements and analyses that will ensure that none of these three limits are exceeded.

II Background

Historical operations between 1965 and 1992 in the 324 Facility B Cell have generated an enormous amount of legacy radioactive waste. These operations included manufacturing radioisotopic heat sources for the Federal Republic of Germany, demonstration of liquid-fed melters and spray calciners for preparation of waste packages, and dissolution of spent nuclear fuel. During these operations, millions of Curies of radioactive material, primarily ^{137}Cs and ^{90}Sr (and their related radioactive progeny, ^{137}Ba and ^{90}Y), were processed in B Cell and some material was spilled or volatilized into the cell. No attempt was made to clean up or completely characterize the material remaining in B Cell after each operation was concluded. Consequently, many tons of highly contaminated material in B Cell require disposal. Millions of Ci of radioactive contamination, both in dispersible and fixed form, currently reside in B Cell.

In 1992, the US Department of Energy, the U.S. Environmental Protection Agency, and the State Of Washington Department of Ecology established a Hanford Federal Facility Agreement and Consent Order, also known as the Tri-Party Agreement (TPA). TPA milestones M-89-02, "Complete Removal of 324 Building REC B-Cell MW Equipment," scheduled for completion on 5/31/1999, states that "actions under this milestone include containment and removal of all B Cell dispersible materials, equipment and debris" to reduce "risks to human health and the environment." In order to gain access to the dispersible material on the floor of the cell, the process racks and support equipment in B Cell must be size-reduced and placed into shipping containers, removed from the hot cell, and sent to disposal. The dispersible material on the exposed areas of the floor, which is known to contain hazardous components and has been designated as mixed waste, will be collected and placed into engineered containers. The engineered containers will be shipped to an interim storage location until a remote-handled mixed waste disposal facility is established. Cleanup activities to date have removed several racks and vessels from B-Cell and has allowed removal of dispersible mixed waste from approximately 75%

of the floor area. The material in B Cell that must be removed to allow access to the remaining floor area consists of three equipment racks: 1-A, 1-B, and 2-A. These racks contain structural material, process piping, and tanks.

Until 1997, the waste packages containing structural material from B Cell was designated as remote-handled low-level radioactive waste (LLW). A B-Cell waste package consists of a grout container containing the waste and grout (to fill void spaces and lower the external dose rate) and a liner that reduces the amount of contamination introduced into the shipping cask from the grout container. Waste is removed from B Cell by size-reducing the racks, placing the rack material into grout containers, adding grout to the containers, loading the grout container (which becomes highly contaminated when it is introduced into B-Cell) into a liner in a relatively low-contamination area (the REC Airlock), loading the liner into a 3-82B shipping cask, and shipping the sealed cask to the Hanford burial grounds. Designation of B Cell waste to date has relied on process knowledge and limited data on the removable contamination on the structural material.

Before shipping the waste to the Hanford burial grounds, the amount of ^{137}Cs and ^{90}Sr is determined by measuring the dose rate on the surface of the grout container (after grouting) at 24 locations (6 equally-spaced measurements from top to bottom at each of the four compass locations on the grout container) and determining the ^{137}Cs activity from the average of these 24 measurements using a gamma dose computer code. In the past the assumption was made that the fixed alpha contamination level of the waste was 0. When the quantity of fixed contamination level was added to the low quantity of removable contamination that had been measured, the material would not be designated as "greater than Category 3" (GC3) or TRU waste.

The dose rates at the surface of waste packages originating from B Cell are sufficiently high that they must be handled remotely. Thus, all waste from B Cell is "remote handled." The term remote handled is not a waste designation, rather it is a descriptor that indicates how it is to be handled and stored. Throughout this document, the waste designation is independent of whether the waste packed is remote handled or contact handled.

It is important to note that designating waste as TRU or GC3 is based on the amount of alpha-emitting radionuclides and the weight of the waste. There is no generic allowable quantity of alpha-emitting radionuclides for a given quantity of ^{137}Cs . According to WHC-EP-0063-4, waste packages that exceed 100 nCi of TRU radionuclides per g of waste are designated as TRU waste and those that exceed 100 nCi of alpha radionuclides per g of waste are designated as GC3. In B Cell, approximately 15% of the alpha activity is from ^{244}Cm , which is not a TRU radionuclide because the half-life of ^{244}Cm is less than 20 years. However, designation of the waste as GC3 is based on the total alpha activity; because there is no established disposal pathway for GC3 waste, B Cell waste must be less than the GC3 limit in order for the Hanford burial grounds to accept it.

In 1997, the basis for shipping B Cell waste to the Hanford burial grounds was reviewed. During the review, it was discovered that the waste designation was based on 3 smear samples obtained from rinsed surfaces of B Cell waste in 1993. These samples were used to establish the ratio of ^{90}Sr to ^{137}Cs so that the quantity of ^{90}Sr could be determined from a gamma dose computer code

calculation. The alpha content of waste shipped from B Cell to the Hanford burial grounds prior to 1997 was not explicitly determined. After reviewing the existing procedure for shipping waste to the burial grounds, it was decided that obtaining additional smear data was warranted.

Three additional smears were obtained for rinsed surfaces of non-waste material in B Cell in 1998. After analyzing the data, a ratio of the alpha activity to the ^{137}Cs activity was developed and shipments of waste to the Hanford burial grounds were resumed. However, a formal technical basis was never completed; a technical basis for maximizing the amount of waste in a B Cell waste package is needed to ensure that the minimum amount of resources are used without compromising safety.

This document is intended to establish the technical basis for shipping B Cell waste in HN-200 grout containers to the Hanford burial site for permanent disposal. To meet this objective, this document will include:

- 1) A review of the applicable requirements for shipping and waste acceptance,
- 2) A justification for a methodology to determine the activity of ^{137}Cs , ^{90}Sr , and alpha-emitting radionuclides in a grout container, the heat generation rate, and the waste designation based on measurable data,
- 3) A determination of the maximum activity in a B Cell waste package that can be designated as LLW,
- 4) A discussion of the dose rates to be encountered during shipping of LLW packages from B Cell, and
- 5) Recommendations for additional characterizations of the contamination in B Cell.

III Applicable Limits for Disposal of B Cell Waste Packages

Limits for B Cell waste packages (grout containers) are established by the Certificate of Compliance (COC)¹ and the Safety Analysis Report (SAR)² for the 3-82B shipping cask, Department of Transportation (DOT) shipping regulations³, and Hanford solid waste acceptance criteria, WHC-EP-0063-4⁴. The following limits are applicable to grout container waste packages:

- 1) The maximum alpha activity is 100 nCi per g of waste as described in the Hanford solid waste acceptance criteria
- 2) The maximum heat generation rate in a grout container is 60 Watts as defined in the SAR for the 3-82B Cask

- 3) The maximum dose rate of a loaded 3-82B shipping cask is 200 mrem/h at contact with the cask surface and 10 mrem/h at a distance of 2m from the cask surface as required by DOT shipping regulations

The only practical method for ensuring that these limits are met for each shipment is to measure the dose rate profile at the surface of the grouted grout container, determine the Cs activity in the grout container using a computer calculation, and estimate the Sr and alpha activity in the grout container from the ^{137}Cs activity based on established ratios of alpha and ^{90}Sr activity to ^{137}Cs activity. The heat generation rate can then be determined from the estimated quantities of ^{90}Sr and ^{137}Cs activities. The likelihood of exceeding the shipping dose rate limit can be minimized by evaluating the results of a pre-grout dose rate profile.

IV Methodology for Determining the Radionuclide Content in a B Cell Grout Container

The methodology for determining the radionuclide content in a B Cell Grout Container consists of three steps:

- 1) Measure the surface (contact) dose rate at 24 locations around the grout container and obtain the dose rate profile
- 2) Estimate the ^{137}Cs content of the grout container based on the average dose rate using conversion factors computed using the computer code WISE
- 3) Apply measured ratios of $^{90}\text{Sr}/^{137}\text{Cs}$ and alpha/ ^{137}Cs to determine the heat generation rate and to establish a waste designation.

A. Dose Rate Measurements

The dose rate in contact with the exterior of the grout container should be measured in a uniform pattern. Measurements should be made at six heights at each of four circumferential positions that correspond to the four major compass directions. From statistical purposes, 24 measurements is adequate to characterize the dose rate field around the grout container. The twenty four dose rate measurements should be added and divided by 24 to obtain the average dose rate at the surface of the grout container. An Excel spreadsheet is provided in Appendix A that, in addition to a number of other capabilities to be discussed later, calculates the average dose rate for 24 dose rate measurements for a single grout container.

B. Estimating the Cs Activity of the Grout Container

The activity of Cs in the grout container should be estimated using the spreadsheet provided in Appendix A. The conversion factor from R/h to Ci in the Excel Spreadsheet was derived from calculations completed using the computer code WISE⁵. The calculated

conversion factor from average dose to Ci of ^{137}Cs is 0.219 R/h/Ci.

WISE, which was written by Pacific Northwest National Laboratory (PNNL) for the Electric Power Research Institute (EPRI), is a shielding code that calculates the dose rate from a known source. By calculating a dose rate for a 1 Ci source uniformly distributed throughout the interior of the grout container, a conversion factor in Ci/dose rate can be obtained that applies to the B Cell waste.

The HN-200 Grout Container was modeled as a right circular cylinder with an inner radius of 58 cm, a wall thickness of 0.34 cm, and a height of 132 cm. The density of the grout used in B Cell is 1.6 g/cm³. Dose rates were calculated in contact with the surface of the grout container. The conversion factor for a uniformly distributed source in an HN-200 grout container is 0.219 R/h/Ci. Results of the calculations performed to establish the conversion factors for HN-200 grout containers are given in Appendix B.

In order to verify the methodology for estimating the Cs activity in a grout container, the computer code WISE was used to perform a more detailed estimate of the source strength. Using WISE, the grout container was "sliced" into 6 stacked disks, which corresponds to the number of measurements taken at each compass location during a dose rate profile measurement. A different source strength was assigned to each slice of the grout container, and the dose rate profile obtained from actual measurements was modeled. Equations for the source strength in each slice were derived as a function of six dose rates that were obtained by averaging the 4 dose rates measured in contact with the surface of each slice at each major compass direction. Determining the six source strengths in terms of six dose rate measurements required solving six dependent equations containing six unknowns. The algebraic equations used to determine the six source strengths are provided in Appendix C. In addition, the same grout container was modeled as a single homogeneous source and all 24 of the dose rate profile measurements were averaged.

The purpose of slicing the grout container into six discrete sources was to determine the sensitivity of the activity calculation to the level of complexity of the model. Table 1 shows the results of the two calculations. For the dose rate profile modeled, the difference in total ^{137}Cs activity determined using the single source method and using the six slice method was only 4.5%. This exercise verifies that the method of modeling the HN-200 grout container as a single, homogeneous source to determine the ^{137}Cs activity is valid. The results of the two calculations are provided in Appendix C.

Source Geometry	Cs Curie Content
Single Homogeneous Source	1133 Ci
6 Independent Sources	1186 Ci

Table 1. Results of Source Geometry Comparison

WISE uses buildup factors when calculating gamma dose. Buildup factors, if used when modeling detailed, heterogeneous sources, can lead to overestimates of calculated radiation doses because buildup is the greatest in homogeneous sources and is reduced in heterogeneous sources. When shielding codes are used to determine source strengths from dose rates, buildup factors can underestimate calculated source strengths. However, in this application the source is modeled as homogeneous and the effect of the buildup factor is appropriately modeled.

C. Determining ^{90}Sr and Alpha Activities

^{90}Sr and alpha activities are determined from measured ratios of ^{90}Sr activity to ^{137}Cs activity and alpha activity to ^{137}Cs activity. The ^{137}Cs activity determined based on the WISE calculations is multiplied by the two ratios to obtain the ^{90}Sr and alpha activities, respectively. The ^{137}Cs and ^{90}Sr activities are used to compute the heat generation rate and the alpha activity, divided by the weight of the grouted container, is used to designate the waste as either GC3, TRU, or LLW. It is the goal of the B Cell Cleanup Safety Project to load the grout containers so that they meet the LLW designation.

Waste material that is loaded into grout containers is rinsed prior to loading the container. Limited data exist on the ^{90}Sr and alpha activity to ^{137}Cs activity ratios for rinsed waste material. The only existing data of this type was obtained in 1993 and is shown in Table 2. The analytical counting laboratory report for these smears is included in Appendix D.

As stated in the introduction, additional smears were obtained in 1998. However, the 1998 smears were obtained from rinsed surfaces of non-waste material and not from surfaces of waste material. Although these smears are not representative of smears obtained from rinsed waste material surfaces, the data are indicative of the B Cell environment and may be useful in developing the activity ratios. The smear data from the 1998 smears are also shown in Table 2. The analytical data laboratory reports associated with the data in Table 2 are provided in Appendix D.

Sample	Cs	Alpha	Sr	Alpha/Cs	Sr/Cs
1993-1	0.135	5.92E-05	0.365	0.000439	2.70
1993-2	0.0505	2.29E-05	0.113	0.000453	2.24
1993-3	0.0437	2.22E-05	0.074	0.000508	1.69
1993 Average				0.000467	2.21
1998-1	0.00973	5.45E-5	0.00128	0.00560	0.132
1998-2	0.229	1.00E-4	0.0769	0.000437	0.336
1998-3	8.93	3.65E-3	0.622	0.000409	0.0697
1998 Average				0.00215	0.179

Table 2. Results of Smears ($\mu\text{Ci/sample}$) Analyzed in 1993 and 1998.

Note: The averages shown in the table are not weighted.

Because of the similarity of the ratio of alpha activity to ^{137}Cs activity in Table 2 from both 1993 and 1998, a ratio of alpha activity to ^{137}Cs activity based on these data can be established. To establish the ratio, a plot of the alpha activity as a function of ^{137}Cs activity can be made and linear regression used to determine the slope of the line. The slope of the line, by definition, is the ratio of alpha activity to ^{137}Cs activity. The data in Table 2 are plotted in Figure 1, which also shows the results of the linear regression and the correlation coefficient. The nearness of the correlation coefficient ($R^2 = 0.9998$) to 1 indicates a very high level of confidence associated with the regression. Note that the regression was forced through the point (0,0). The linear regression was performed using functions included with Microsoft Excel⁶. Based on a linear regression of the data for alpha and ^{137}Cs in Table 2, the ratio of alpha activity to ^{137}Cs activity is 0.000408.

The data in Table 2 for the ratio of ^{90}Sr activity to ^{137}Cs activity are not consistent because in the 1993 data the ratio of ^{90}Sr activity to ^{137}Cs activity is greater than 1, while in the 1998 data the ratio is less than 1. A linear regression of both the 1993 and 1998 data is plotted in Figure 2. The correlation coefficient (R^2) of 0.4800 indicates an unacceptably low confidence in the fitted curve. Because the 1998 data results in a higher predicted ^{90}Sr activity, which leads to a conservative estimate of the heat generation rate in the waste package, the data from 1993 was used in a regression analysis to determine the ratio of ^{90}Sr activity to ^{137}Cs activity. This regression is shown in Figure 3 and the correlation coefficient, although not indicating a very high degree of confidence, is adequate for the available data. Based on the analysis, the ratio of ^{90}Sr activity to ^{137}Cs activity to be used for B Cell waste should be 2.57. Additional studies are warranted to verify and refine this ratio, however.

Figure 1. Linear Regression Fit of the alpha and ^{137}Cs activity data.

Figure 2. Linear Regression Fit of the ^{90}Sr and ^{137}Cs activity data for both the 1993 and 1998 smear data.

Figure 3. Linear Regression Fit of the ^{90}Sr and ^{137}Cs activity data for the 1993 smear data.

D. Establishing the Waste Designation

The waste designation is established by determining the activity per gram of alpha-emitting radionuclides. This is accomplished by multiplying the calculated ^{137}Cs activity by the ratio of alpha activity to ^{137}Cs activity and dividing by the weight of the grouted container. If this value is greater than 100 nCi/g, then the waste is designated as GC3. In B Cell, the only non-TRU radioisotope is ^{244}Cm , which has a half life of less than 20 years. Alpha energy analysis of the smears obtain in 1998, the results of which are tabulated in Table 3, indicate that 26% of the alpha activity in B Cell is from ^{244}Cm . The contributions from other alpha-emitting radionuclides are as shown in the analytical counting laboratory report in Appendix D. Therefore, if the activity per gram of alpha-emitting radionuclides is greater than 135 nCi/g, then the waste is designated as TRU waste. The calculation of alpha activity per gram, and thus the designation of the waste type, is performed in the Excel spreadsheet in Appendix A.

Sample	Total Alpha	^{244}Cm	Fraction Non-TRU	^{137}Cs
1998-1	5.45E-5	1.21E-5	2.22E-1	9.73E-3
1998-2	1.00E-4	4.12E-5	4.12E-1	2.29E-1
1998-3	3.65E-3	5.72E-4	1.57E-1	8.93E+0
Average			2.63E-1	

Table 3. Results of Alpha Energy Analysis for 1998 Smear Samples

It should be noted that the value determined in the Excel spreadsheet can be directly compared to the 100 nCi/g limit. Whenever measurements are made and compared to a limit, there are four possible outcomes of the comparison, two of which are "good" and two of which are "bad." The two good outcomes is that the measurement is correct, that is, when the dose rate profile indicates that the waste is non-TRU and it is actually non-TRU (true negative) and when the measurement indicates that the waste is TRU and it is actually TRU (true positive). Problems arise when the dose rate measurement indicates that the waste is non-TRU when it is actually TRU (false negative) and when the measurement indicates that the waste is TRU when it is actually non-TRU. When comparing any measurement to an established limit, all four of these outcomes are always possible, and the possibility increases when the measurements are within 20% of the limits. It is common to establish confidence intervals that change the probability of either a false positive or a false negative; however, decreasing one form of error necessarily increases the other form of error proportionally. In other words, decreasing the probability of a false negative by 50% necessarily increases the probability of a false positive by 50%.

At first thought, it may seem that a false negative result is worse than a false positive result. However, one must balance the cost and risk associated with each outcome for the situation to which it is applied. A false negative outcome results in the disposal of waste containing a slightly elevated quantity of TRU radionuclides. A false positive outcome results in the generation of a waste package for which no final disposal pathway exists.

Given the size of the waste package (roughly 1.4 m³) and the low activity of the transuranic radionuclides (0.272 Ci [see calculation below]), the risk of a slightly elevated activity of TRU radionuclides appears to be no more than equivalent to the incremental cost of storing, retrieving, and potentially repackaging the waste once a disposal pathway has been identified. Therefore, it is recommended that the measurements of alpha activity be directly compared to the limit for TRU waste without adding the value associated with a confidence interval to the measurement.

E. Establishing the Heat Generation Rate

The heat generation rate for a grouted container is based on the ⁹⁰Sr and ¹³⁷Cs activities. ⁹⁰Sr has a heat generation rate of 0.0067 W/Ci ⁹⁰Sr and ¹³⁷Cs has a heat generation rate of 0.0048 W/Ci ¹³⁷Cs⁷. The heat generation rate is calculated in the Excel spreadsheet in Appendix A.

F. Loading Limits for Grout Containers

The heat generation rates and the ratio of ⁹⁰Sr activity to ¹³⁷Cs activity can be used to establish a maximum activity loading for a B Cell grout container. The maximum heat generation rate for a grout container is 60 W (205 BTU)². Based on this limit, the maximum activities of the waste are 2725 Ci ¹³⁷Cs and 7003 Ci ⁹⁰Sr. The activity of alpha-emitting radionuclides associated with 2725 Ci ¹³⁷Cs is 1.11 Ci, and the weight of a typical grout container is 2,600 kg (5,700 lb). In this case the specific alpha activity is 427 nCi/g and the waste would be designated as remote-handled TRU. Thus, at the maximum heat generation rate of the grout container, the associated quantity of alpha-emitting radionuclides would require that the waste package be designated as TRU. Therefore, the heat generation rate of any waste package that is designated as LLW will be significantly lower than the limit of 60 W.

Note that in the previous discussion the weight of the grout container was assumed to be 2,600 kg (5,700 lb). If the actual weight is greater, more activity may be placed into the grout container. If the weight of the grouted container is less, however, the limit is more restrictive. In either case, it is not conceivable that the heat generation limit would be exceeded for a waste package that is designated as LLW using the data included in this report.

V Determination of the Maximum Activity of a LLW Package

The maximum activity of alpha-emitting radionuclides in a waste package that is designated as LLW can be determined from the 100 nCi/g alpha activity limit, the weight of a typical waste package (6,000 lb [2720 kg]), the ratio of alpha activity to ¹³⁷Cs activity (0.000408), and the ratio of ⁹⁰Sr activity to ¹³⁷Cs activity (2.57). The maximum alpha activity of a 2720 kg LLW package is 100 nCi/g * 2720 kg = 0.272 Ci. The ¹³⁷Cs activity associated with this alpha activity is 0.272 Ci / 0.000408 = 667 Ci ¹³⁷Cs. The ⁹⁰Sr activity for

this package is $667 * 2.57 = 1713 \text{ Ci } ^{90}\text{Sr}$.

The heat generation rate for this waste package, based on a ^{90}Sr heat generation rate of $0.0067 \text{ W/Ci } ^{90}\text{Sr}$ and a ^{137}Cs heat generation rate of $0.0048 \text{ W/Ci } ^{137}\text{Cs}$, is $667 * 0.0048 + 1713 * 0.0067 = 14.7 \text{ W}$. Thus, the heat generation rate is less than the limit of 60 W .

VI Dose Rate Limits During Shipping Using the 3-82B Shipping Cask

Calculations of the quantity of ^{137}Cs that could be loaded into a grout container, based on the external dose rate of a loaded 3-82B shipping cask, have been previously determined⁸ for a ^{90}Sr activity to ^{137}Cs activity of 2.1. The report concluded "The 3-82B cask will meet the transportation dose rate limits of 0.002 Sv/h (200 mrem/h) on the cask surface, and 0.0001 Sv/h (10 mrem/h) at 2 m from cask surface for a mixture containing 3600 Ci of ^{137}Cs and 7560 Ci of ^{90}Sr . The limits will be met for a radioactive source that is uniformly distributed throughout the grout container as well as a concentrated radioactive source located near the periphery of the grout container." Based on the ratio established in this document, a mixture containing 3600 Ci of ^{137}Cs and $9252 \text{ Ci } ^{90}\text{Sr}$ would not exceed the shipping dose rate limits. A ^{137}Cs activity of 667 Ci , which is the maximum ^{137}Cs activity allowed for the B Cell waste to be designated as LLW, is well below the activity of ^{137}Cs that will cause the dose rate to exceed 200 mR/h at contact or 10 mR/h at a distance of 2 m . Because the dose rate for shipping cannot be exceeded for a B Cell LLW package, it can be concluded that the limiting parameter for loading LLW packages is the activity of alpha-emitting radionuclides.

Should further analysis of the B Cell environment indicate that the alpha to ^{137}Cs ratio is significantly lower than the data in this document indicates, it is possible that a different constraint would exist on the activity of a waste package. The heat generation rate for a grout container loaded with the maximum activity that can still be below the shipping dose rate limit, 3600 Ci of ^{137}Cs and $9252 \text{ Ci } ^{90}\text{Sr}$, would have a heat generation rate of $3600 * 0.0048 + 9252 * 0.0067 = 79 \text{ W}$, which is greater than the heat generation rate established by the 3-82B COC. The maximum activity of a loaded grout container that will not exceed the heat generation rate limit is 2725 Ci of ^{137}Cs and $7003 \text{ Ci } ^{90}\text{Sr}$. If further analysis of the B Cell environment reduces the ratio of ^{90}Sr activity to ^{137}Cs activity, then the allowed activity of ^{137}Cs and ^{90}Sr would increase. In fact, if the ratio were to drop to below 1.77, then the heat generation rate would no longer limit the activity and the shipping dose rate would be the limiting factor, providing the alpha contamination level is negligible.

VII Additional Source Term Characterization

Characterization of the source term in B Cell is necessary to resolve the discrepancy in the ratio of ^{90}Sr activity to ^{137}Cs activity between the 1993 smear data and the 1998 smear data. Should this ratio change significantly, the maximum cask loading could be increased if the alpha contamination issue is resolved. One method of resolving this issue is to

pursue additional characterization of the ratio of alpha activity to ^{137}Cs activity in B Cell. Another method is to investigate methods of removing the contamination from the waste material using a nonhazardous decontamination foam or soap before introducing it into the grout container. This would reduce the ^{137}Cs and alpha activity in the waste package while increasing the amount of waste material in the container. The removed activity could then be collected with the dispersible contamination on the B Cell floor.

VIII Conclusions

This document has established the technical basis for performing dose rate profile measurements and, based on the results of these measurements, has established the maximum activity loading of a grout container that will allow the waste package to be accepted at the Hanford burial grounds as remote-handled low level waste. In addition, this document provides the technical basis for determining the isotopic activity, the heat generation rate, and the waste designation of the waste package. The limit that currently restricts the activity that can be shipped to the Hanford burial grounds is the 100 nCi alpha activity per gram of waste. Because of this limit, the maximum activity that can be grouted in an HN-200 Grout Container is 667 Ci ^{137}Cs based on a waste package weight of 2,720 kg (6,000 lb). For grouted HN-200 Grout Containers that weigh more than 2,720 kg, the activity is greater; for grout containers that weigh less, the activity will be less. Regardless of the weight of the container, the heat generation rate of the contents will not exceed the SAR for the cask nor DOT shipping regulations provided that the waste package can be designated as LLW. The ratio of alpha activity to ^{137}Cs activity should be 0.000408, and the ratio of ^{90}Sr activity to ^{137}Cs activity should be 2.57. The conversion from average dose rate to ^{137}Cs activity should be 0.219 R/h/Ci. Additional characterization of the waste is needed to refine the ratio of alpha and ^{90}Sr activity to ^{137}Cs activity in order to allow more material to be buried.

IX References

1. Nuclear Regulatory Commission Certificate of Compliance. MMT-TN 3-82B Certificate of Compliance #6574.
2. *Safety Analysis Report for the 3-82B Radwaste Shipping Cask*, STD-R-02-014, Scientific Ecology Group, Inc., Oak Ridge, TN. 1991.
3. DOT Shipping Regulations.
4. WHC-EP-0063-4. Hanford Site Solid Waste Acceptance Criteria.
5. *WISE*. Electric Power Research Institute, 3412 Hillview Avenue, Palo Alto, California. 1987.
6. Excel 97 SR-1. Microsoft Corporation, 1997.

7. 85600-Battelle-bf222673. *Characteristics of Radioisotopic Heat Sources*. 1973.
8. EBU-RCAL-002. *3-82B Shipping Cask Shielding Analysis*. Waste Management Northwest, 1998.

Appendix A. Excel Spreadsheet for Use in Calculating the Average Dose Rate, ^{137}Cs Activity, ^{90}Sr Activity, and Alpha Activity in a Grouted HN-200 Liner

To use the spreadsheet, enter the Grout Container Identification Number in field B-2, enter the dose rate profile measurements in fields B-3 through E-8, and enter the weight of the grouted liner in field B-12. The results of the spreadsheet are the average dose rate (field F-10), the ^{137}Cs activity (field G-6), the ^{90}Sr activity (field G-8), and the nCi/g of alpha activity (field B-16).

ID number	0 degrees	90 degrees	180 degrees	270 degrees	Average (Rad/h)	
Dose (rad/h)	50	40	40	40	42.5	1 Ci Cs = 0.219 rad
Dose (rad/h)	40	40	40	30	37.5	
Dose (rad/h)	50	120	20	30	55	Cs Activity (Ci)
Dose (rad/h)	40	70	40	30	45	306
Dose (rad/h)	50	400	30	40	130	Sr Activity (Ci)
Dose (rad/h)	60	100	60	150	92.5	787
Average Dose Rate (rad/h)	48	128	38	53	67.1	
Weight of Waste (lb)	5700					
Weight of Waste (kg)	2585					
alpha/Cs Fraction	0.000408					
Ci of alpha in grout container	0.1250					
nCi/g alpha in grout container	48.3					

**Appendix B. Results of the Calculations Performed to Establish the Conversion Factors for
HN-200 Grout Containers**

**** WELCOME TO WISE/SIMPLE ****
DEVELOPED FOR EPRI
BY BATTELLE PACIFIC NORTHWEST LABORATORIES
WRITTEN AND TESTED BY WD REECE LAST ON 15 JULY 1987

FOR THIS RUN:

NUMBER OF SHAPES IS 1

ISOTOPE LIBRARY

1 Cs-137

MAX BETA ENERGY = .512 AVE BETA ENERGY = .157
 BETA YIELD = .946

5.000E-03 1.040E-02
 3.200E-02 2.070E-02
 3.200E-02 3.820E-02
 3.600E-02 1.390E-02
 6.620E-01 8.998E-01

SHIELDING MATERIALS LIBRARY

FOR MATERIAL 1 --grout

ATOMIC NUMBER (Z) IS 1.000E+01

FOR ENERGY (IN MEV)

.200 .300 .400 .500 .600 .800 1.000 1.500 2.000 3.000 4.000 6.000

MU (IN CM(-1)) IS

.203 .173 .154 .141 .130 .114 .102 .083 .072 .058 .051 .043

FOR MATERIAL 2 --steel

ATOMIC NUMBER (Z) IS 2.600E+01

FOR ENERGY (IN MEV)

.200 .300 .400 .500 .600 .800 1.000 1.500 2.000 3.000 4.000 6.000

MU (IN CM(-1)) IS

1.148 .865 .739 .660 .605 .527 .472 .385 .338 .283 .259 .244

SOURCE LIBRARY

SOURCE KEY 1

FOR Cs-137 THE CONCENTRATION = 7.190E-01 (MICROCURI/GEOMETRIC UNIT)

ESTIMATED BREMSSTRAHLUNG CONTRIBUTIONS

BREMSTRAHLUNG SHOULD CONTRIBUTE LESS THAN 20% TO ANY DOSE

FOR SHAPE 1 (CYLINDER)

(X1,Y1,Z1)= .000E+00 .000E+00 .000E+00

(X2,Y2,Z2)= .000E+00 .000E+00 1.320E+02

RADIUS= 5.794E+01

WALL THICKNESS IS 3.250E-01

FILL MATERIAL IS grout

WALL MATERIAL IS steel

SOURCE KEY IS 1.

ENTER THE COORDINATES FOR THE DOSE POINT

MEASUREMENTS ARE IN CENTIMETERS

EXAMPLE - 0.0,10.0,0.0

.0000000 58.3000000 66.0000000

RESULTS FROM THIS INPUT FOLLOW

FOR SHAPE # 1 RESULT IS 218.9430000

FOR THE POINT

WHOSE COORDINATES ARE... .000E+00 5.830E+01 6.600E+01

TOTAL DOSE (MREM/HR) IS.....

2.189E+02

WOULD YOU LIKE ANOTHER DOSE POINT? (Y/N)

n

**Appendix C. Results of Hand Calculations and Computer Models Used to Model the HN-200
Grout Container as Six Source Slices (14 pages)**

Dose Points, $\frac{C_{\text{root}} + 6}{D} \text{ (R/h)}$ $\frac{D}{\text{DIST (miles)}}$ ①

58.5, 0, 11	195	Ⓐ	7.76
58.5, 0, 33	220	Ⓑ	8.38
58.5, 0, 55	222	↓	11.6
58.5, 0, 77	222		7.81
58.5, 0, 99	220		22.0
58.5, 0, 121	195	Ⓐ	141

$$D_1 = .87 S_1 + .130 S_2$$
~~$$D_1 = .130 S_2 + .87 S_1$$~~

$$D_2 = .115 S_1 + .77 S_2 + .115 S_3 \quad 25.03$$

$$D_3 = .115 S_2 + .77 S_3 + .115 S_4 \quad 167.75$$

$$D_4 = .115 S_3 + .77 S_4 + .115 S_5 \quad 25.03$$

$$D_5 = .115 S_4 + .77 S_5 + .115 S_6 \quad A_1: .130(S_2) + .87(S_1)$$

$$D_6 = .13 S_5 + .87 S_6$$

$$B_2: .115(S_1) + .77(S_2) + .115(S_3)$$

BEST AVAILABLE COPY

Calculate Source Strengths.

(2)

$$S_1 = \frac{D_1}{.87} - \frac{.13}{.87} S_2$$

$$C_1 = 1.15 D_1$$

$$S_1 = C_1 - .15 S_2$$

$$D_2 = .115 (C_1 - .15 S_2) + .77 S_2 + .115 S_3$$

$$D_2 = .115 C_1 - .017 S_2 + .77 S_2 + .115 S_3$$

$$D_2 = .115 C_1 + .753 S_2 + .115 S_3$$

$$S_2 = \frac{D_2 - .115 C_1}{.753} - .15 S_3$$

$$C_2 = \frac{D_2 - .115 C_1}{.753}, C_2 = 1.33 D_2 - .15 S_3$$

$$S_2 = C_2 - .15 S_3$$

$$D_3 = .115 S_2 + .77 S_3 + .115 S_4$$

$$D_3 = .115 (C_2 - .15 S_3) + .77 S_3 + .115 S_4$$

$$D_3 = .115 C_2 - .017 S_3 + .77 S_3 + .115 S_4$$

$$D_3 = .115 C_2 + .75 S_3 + .115 S_4$$

$$S_3 = \frac{D_3 - .115 C_2 - .115 S_4}{.75}$$

$$C_3 = 1.33 D_3 - .15 S_4$$

$$S_3 = C_3 - .15 S_4$$

$$D_4 = .115 S_3 + .77 S_4 + .115 S_5$$

$$D_4 = .115 (C_3 - .15 S_4) + .77 S_4 + .115 S_5$$

$$D_4 = .115 C_3 - .017 S_4 + .77 S_4 + .115 S_5$$

$$S_4 = \frac{D_4 - .115 C_3 - .115 S_5}{.75}$$

$$C_4 = 1.33 D_4 - .15 C_3$$

$$S_4 = C_4 - .15 S_5$$

$$D_5 = .115 S_4 + .77 S_5 + .115 S_6$$

$$D_5 = .115 (C_4 - .15 S_5) + .77 S_5 + .115 S_6$$

$$D_5 = .115 C_4 - .017 S_5 + .77 S_5 + .115 S_6$$

$$S_5 = \frac{D_5 - .115 C_4 - .115 S_6}{.75}$$

$$C_5 = 1.33 D_5 - .15 C_4$$

$$S_5 = C_5 - .15 S_6$$

$$D_6 = .13 S_5 + .87 S_6$$

$$D_6 = .13 (C_5 - .15 S_6) + .87 S_6$$

$$D_6 = .13 C_5 - .0195 S_6 + .87 S_6$$

$$D_6 = .13 C_5 + .85 S_6$$

(1) (2)

$$S_6 = \frac{D_6 - .13 C_5}{.85}$$

$$C_6 = 1.18 D_6$$

$$S_6 = 1.18 D_6 - .15 C_5$$

BEST AVAILABLE COPY

** WELCOME TO WISE/SIMPLE **
 DEVELOPED FOR EPRI
 BY BATTELLE PACIFIC NORTHWEST LABORATORIES
 WRITTEN AND TESTED BY WD REECE LAST ON 15 JULY 1987

FOR THIS RUN:
 NUMBER OF SHAPES IS 6

ISOTOPE LIBRARY

1 Cs-137

MAX BETA ENERGY = .512 AVE BETA ENERGY = .157
 BETA YIELD = .946

5.000E-03 1.040E-02
 3.200E-02 2.070E-02
 3.200E-02 3.820E-02
 3.600E-02 1.390E-02
 6.620E-01 8.998E-01

SHIELDING MATERIALS LIBRARY

FOR MATERIAL 1 --Steel
 ATOMIC NUMBER (Z) IS 2.600E+01
 FOR ENERGY (IN MEV)
 .200 .300 .400 .500 .600 .800 1.000 1.500 2.000 3.000 4.000 6.000
 MU (IN CM(-1)) IS
 1.148 .865 .739 .660 .605 .527 .472 .385 .338 .283 .259 .244
 FOR MATERIAL 2 --grout
 ATOMIC NUMBER (Z) IS 1.000E+01
 FOR ENERGY (IN MEV)
 .200 .300 .400 .500 .600 .800 1.000 1.500 2.000 3.000 4.000 6.000
 MU (IN CM(-1)) IS
 .203 .173 .154 .141 .130 .114 .102 .083 .072 .058 .051 .043
 FOR MATERIAL 3 --lead
 ATOMIC NUMBER (Z) IS 8.200E+01
 FOR ENERGY (IN MEV)
 .200 .300 .400 .500 .600 .800 1.000 1.500 2.000 3.000 4.000 6.000
 MU (IN CM(-1)) IS
 11.248 4.585 2.622 1.827 1.419 1.010 .806 .590 .511 .477 .477 .488
 FOR MATERIAL 4 --air

ATOMIC NUMBER (Z) IS 7.000E+00
 FOR ENERGY (IN MEV)
 .200 .300 .400 .500 .600 .800 1.000 1.500 2.000 3.000 4.000 6.000
 MU (IN CM(-1)) IS
 .000 .000 .000 .000 .000 .000 .000 .000 .000 .000 .000 .000

SOURCE LIBRARY

SOURCE KEY 1

FOR Cs-137 THE CONCENTRATION = 7.180E-01
 (MICROCURI/GEOMETRIC UNIT)

SOURCE KEY 2

FOR Cs-137 THE CONCENTRATION = 7.180E-01
 (MICROCURI/GEOMETRIC UNIT)

SOURCE KEY 3

FOR Cs-137 THE CONCENTRATION = 7.180E-01
 (MICROCURI/GEOMETRIC UNIT)

SOURCE KEY 4

FOR Cs-137 THE CONCENTRATION = 7.180E-01
 (MICROCURI/GEOMETRIC UNIT)

SOURCE KEY 5

FOR Cs-137 THE CONCENTRATION = 7.180E-01
 (MICROCURI/GEOMETRIC UNIT)

SOURCE KEY 6

FOR Cs-137 THE CONCENTRATION = 7.180E-01
 (MICROCURI/GEOMETRIC UNIT)

ESTIMATED BREMSSTRAHLUNG CONTRIBUTIONS

BREMSTRAHLUNG SHOULD CONTRIBUTE LESS THAN 20% TO ANY DOSE

FOR SHAPE 1 (CYLINDER)

(X1,Y1,Z1)= .000E+00 .000E+00 .000E+00

(X2,Y2,Z2)= .000E+00 .000E+00 2.200E+01

RADIUS= 5.794E+01

WALL THICKNESS IS 3.250E-01

FILL MATERIAL IS grout

WALL MATERIAL IS Steel

SOURCE KEY IS 1.

FOR SHAPE 2 (CYLINDER)

(X1,Y1,Z1)= .000E+00 .000E+00 2.200E+01

(X2,Y2,Z2)= .000E+00 .000E+00 4.400E+01

RADIUS= 5.794E+01

WALL THICKNESS IS 3.250E-01

FILL MATERIAL IS grout

WALL MATERIAL IS Steel

SOURCE KEY IS 2.

FOR SHAPE 3 (CYLINDER)

(X1,Y1,Z1)= .000E+00 .000E+00 4.400E+01

(X2,Y2,Z2)= .000E+00 .000E+00 6.600E+01

RADIUS= 5.794E+01

WALL THICKNESS IS 3.250E-01

FILL MATERIAL IS grout

WALL MATERIAL IS Steel

SOURCE KEY IS 3.

FOR SHAPE 4 (CYLINDER)

(X1,Y1,Z1)= .000E+00 .000E+00 6.600E+01

(X2,Y2,Z2)= .000E+00 .000E+00 8.800E+01

RADIUS= 5.794E+01

WALL THICKNESS IS 3.250E-01

FILL MATERIAL IS grout

WALL MATERIAL IS Steel

SOURCE KEY IS 4.

FOR SHAPE 5 (CYLINDER)

(X1,Y1,Z1)= .000E+00 .000E+00 8.800E+01

(X2,Y2,Z2)= .000E+00 .000E+00 1.100E+02

RADIUS= 5.794E+01

WALL THICKNESS IS 3.250E-01

FILL MATERIAL IS grout

WALL MATERIAL IS Steel
SOURCE KEY IS 5.

FOR SHAPE 6 (CYLINDER)

(X1,Y1,Z1)= .000E+00 .000E+00 1.100E+02

(X2,Y2,Z2)= .000E+00 .000E+00 1.320E+02

RADIUS= 5.794E+01

WALL THICKNESS IS 3.250E-01

FILL MATERIAL IS grout

WALL MATERIAL IS Steel

SOURCE KEY IS 6.

ENTER THE COORDINATES FOR THE DOSE POINT
MEASUREMENTS ARE IN CENTIMETERS

EXAMPLE - 0.0,10.0,0.0

58.5000000 .0000000 11.0000000

RESULTS FROM THIS INPUT FOLLOW

FOR SHAPE # 1 RESULT IS 167.7474000

RESULTS FROM THIS INPUT FOLLOW

FOR SHAPE # 2 RESULT IS 25.0342000

RESULTS FROM THIS INPUT FOLLOW

FOR SHAPE # 3 RESULT IS 1.8639540

RESULTS FROM THIS INPUT FOLLOW

FOR SHAPE # 4 RESULT IS 1.316853E-001

RESULTS FROM THIS INPUT FOLLOW

FOR SHAPE # 5 RESULT IS 9.286723E-003

RESULTS FROM THIS INPUT FOLLOW

FOR SHAPE # 6 RESULT IS 6.294497E-004

FOR THE POINT
WHOSE COORDINATES ARE... 5.850E+01 .000E+00 1.100E+01
TOTAL DOSE (MREM/HR) IS.....
1.948E+02

WOULD YOU LIKE ANOTHER DOSE POINT? (Y/N)

y

ENTER THE COORDINATES FOR THE DOSE POINT
MEASUREMENTS ARE IN CENTIMETERS

EXAMPLE - 0.0,10.0,0.0
58.5000000 .0000000 33.0000000

RESULTS FROM THIS INPUT FOLLOW

FOR SHAPE # 1 RESULT IS 25.0342000

RESULTS FROM THIS INPUT FOLLOW

FOR SHAPE # 2 RESULT IS 167.7474000

RESULTS FROM THIS INPUT FOLLOW

FOR SHAPE # 3 RESULT IS 25.0342000

RESULTS FROM THIS INPUT FOLLOW

FOR SHAPE # 4 RESULT IS 1.8639540

RESULTS FROM THIS INPUT FOLLOW

FOR SHAPE # 5 RESULT IS 1.316853E-001

RESULTS FROM THIS INPUT FOLLOW

FOR SHAPE # 6 RESULT IS 9.286723E-003

FOR THE POINT

WHOSE COORDINATES ARE... 5.850E+01 .000E+00 3.300E+01

TOTAL DOSE (MREM/HR) IS.....

2.198E+02

WOULD YOU LIKE ANOTHER DOSE POINT? (Y/N)

y

ENTER THE COORDINATES FOR THE DOSE POINT
MEASUREMENTS ARE IN CENTIMETERS

EXAMPLE - 0.0,10.0,0.0

58.5000000 .0000000 55.0000000

RESULTS FROM THIS INPUT FOLLOW

FOR SHAPE # 1 RESULT IS 1.8639540

RESULTS FROM THIS INPUT FOLLOW

FOR SHAPE # 2 RESULT IS 25.0342000

RESULTS FROM THIS INPUT FOLLOW

FOR SHAPE # 3 RESULT IS 167.7474000

RESULTS FROM THIS INPUT FOLLOW

FOR SHAPE # 4 RESULT IS 25.0342000

RESULTS FROM THIS INPUT FOLLOW

FOR SHAPE # 5 RESULT IS 1.8639540

RESULTS FROM THIS INPUT FOLLOW

FOR SHAPE # 6 RESULT IS 1.316853E-001

FOR THE POINT

WHOSE COORDINATES ARE... 5.850E+01 .000E+00 5.500E+01

TOTAL DOSE (MREM/HR) IS.....

2.217E+02

WOULD YOU LIKE ANOTHER DOSE POINT? (Y/N)

y

ENTER THE COORDINATES FOR THE DOSE POINT

MEASUREMENTS ARE IN CENTIMETERS

EXAMPLE - 0.0,10.0,0.0

58.5000000 .0000000 77.0000000

RESULTS FROM THIS INPUT FOLLOW

FOR SHAPE # 1 RESULT IS 1.316853E-001

RESULTS FROM THIS INPUT FOLLOW

FOR SHAPE # 2 RESULT IS 1.8639540

RESULTS FROM THIS INPUT FOLLOW

FOR SHAPE # 3 RESULT IS 25.0342000

RESULTS FROM THIS INPUT FOLLOW

FOR SHAPE # 4 RESULT IS 167.7474000

RESULTS FROM THIS INPUT FOLLOW

FOR SHAPE # 5 RESULT IS 25.0342000

RESULTS FROM THIS INPUT FOLLOW

FOR SHAPE # 6 RESULT IS 1.8639540

FOR THE POINT

WHOSE COORDINATES ARE... 5.850E+01 .000E+00 7.700E+01

TOTAL DOSE (MREM/HR) IS.....

2.217E+02

WOULD YOU LIKE ANOTHER DOSE POINT? (Y/N)

y

ENTER THE COORDINATES FOR THE DOSE POINT
MEASUREMENTS ARE IN CENTIMETERS

EXAMPLE - 0.0,10.0,0.0

58.5000000 .0000000 99.0000000

RESULTS FROM THIS INPUT FOLLOW

FOR SHAPE # 1 RESULT IS 9.286723E-003

RESULTS FROM THIS INPUT FOLLOW

FOR SHAPE # 2 RESULT IS 1.316853E-001

RESULTS FROM THIS INPUT FOLLOW

FOR SHAPE # 3 RESULT IS 1.8639540

RESULTS FROM THIS INPUT FOLLOW

FOR SHAPE # 4 RESULT IS 25.0342000

RESULTS FROM THIS INPUT FOLLOW

FOR SHAPE # 5 RESULT IS 167.7474000

RESULTS FROM THIS INPUT FOLLOW

FOR SHAPE # 6 RESULT IS 25.0342000

FOR THE POINT
WHOSE COORDINATES ARE... 5.850E+01 .000E+00 9.900E+01
TOTAL DOSE (MREM/HR) IS.....
2.198E+02

WOULD YOU LIKE ANOTHER DOSE POINT? (Y/N)
y

ENTER THE COORDINATES FOR THE DOSE POINT
MEASUREMENTS ARE IN CENTIMETERS

EXAMPLE - 0.0,10.0,0.0
 58.5000000 .0000000 121.0000000

RESULTS FROM THIS INPUT FOLLOW

FOR SHAPE # 1 RESULT IS 6.294497E-004

RESULTS FROM THIS INPUT FOLLOW

FOR SHAPE # 2 RESULT IS 9.286723E-003

RESULTS FROM THIS INPUT FOLLOW

FOR SHAPE # 3 RESULT IS 1.316853E-001

RESULTS FROM THIS INPUT FOLLOW

FOR SHAPE # 4 RESULT IS 1.8639540

RESULTS FROM THIS INPUT FOLLOW

FOR SHAPE # 5 RESULT IS 25.0342000

RESULTS FROM THIS INPUT FOLLOW

FOR SHAPE # 6 RESULT IS 167.7474000

FOR THE POINT
 WHOSE COORDINATES ARE... 5.850E+01 .000E+00 1.210E+02
 TOTAL DOSE (MREM/HR) IS.....
 1.948E+02

Appendix D. Analytical Counting Laboratory Data Sheets *(4 pages)*

Cognizant scientist:

GO Hayward

Date:

7/28/73

Reviewer:

Richard A. Johnson

Date:

7/27/73

ALO #	Customer ID	Sr-90	Cs-137	Total Beta as SrY-90	Total Alpha
93-9076	71393 #1 WHITE FILTER CAN #2	3.65 E-1	1.35 E-1	7.47 E-1	5.92 E-5
93-9077	71393 #2 WHITE FILTER CAN #2	1.13 E-1	5.05 E-2	2.49 E-1	2.29 E-5
93-9078	71393 #3 WHITE FILTER CAN #2	7.40 E-2	4.37 E-2	1.69 E-1	2.22 E-5
93-9079	6993 #1 BLACK FILTER CAN #1	6.74 E-1	1.35 E0	2.66 E0	6.55 E-4
93-9080	6993 #2 BLACK FILTER CAN #1	3.89 E-2	2.34 E-2	9.15 E-2	4.84 E-6
93-9081	6993 #3 BLACK FILTER CAN #1	2.44 E-1	1.55 E-1	5.92 E-1	3.26 E-5

Per your instructions, each filter was leached and the leachate analyzed. The values listed are uCi/ml of the leach solution but no effort was made to quantitate the results, therefore the filters are not directly comparable. Note that the results for Strontium results are Sr only, not Sr+Y at equilibrium.

Battelle Pacific Northwest Laboratory
 Analytical Chemistry Laboratory
 Radioanalytical Group - 325 Bldg.

98-2812
 3/24/98

Client : S. Landsman Wp#: K82285

Cognizant Scientist: Richard T. RS Date: 3/24/98
 Concur: L R Gueanwood Date: 3/24/98

Measured Activities ($\mu\text{Ci/sample}$)

ALO ID Client ID	Measured Activities ($\mu\text{Ci/sample}$)					Analysis		
	Mn-54 Error %	Co-60 Error %	Sb-125 Error %	Cs-134 Error %	Energy Cs-137 Error %	Eu-154 Error %	Eu-155 Error %	Am-241 Error %
98-2812 B-Cell 1	<4.E-5	<7.E-5	<2.E-4	<3.E-5	9.73E-3 5%	<1.E-4	<1.E-4	<2.E-4
98-2813 B-Cell 2	<8.E-5	<2.E-4	<2.E-3	<6.E-5	2.29E-1 5%	<3.E-4	<7.E-4	<2.E-3
98-2814 B-Cell 3	<2.E-3	7.66E-2 5%	<2.E-2	<3.E-3	8.93E+0 5%	<4.E-3	<7.E-3	<2.E-2

Battelle Pacific Northwest Laboratory
Analytical Chemistry Laboratory
Radioanalytical Group - 325 Bldg.

98-2812
3/27/98

Client : S. Landsman

Wp#: K82285

Cognizant Scientist: JR Greenwald

Date: 3/27/98

Concur: T. Tranter

Date: 3/29/98

Measured Activities ($\mu\text{Ci/sample}$)

ALO ID Client ID	Sr-90 Error %
98-2812	1.28E-3
B-Cell 1	5%
98-2813	7.69E-2
B-Cell 2	6%
98-2814	6.22E-1
B-Cell 3	8%
Standard	96%
Sample Spike	93%
Blank	<9.E-5

Battelle Pacific Northwest Laboratory
Analytical Chemistry Laboratory
Radioanalytical Group - 325 Bldg.

98-2812
3/25/98

Client: S. Landsman

Wp#: K82285

Cognizant Scientist: Richard T. R

Date: 3/25/98

Concur: L.R. Freeman

Date: 3/25/98

Measured Activities ($\mu\text{Ci/sample}$)

ALO ID Client ID	Total Alpha Error %	Pu-239/240 Error %	Pu-238/Am-241 Error %	Cm-243/244 Error %	Cm-242 Error %
98-2812 B-Cell 1	5.45E-5 17%	<1.E-5	4.24E-5 33%	1.21E-5 62%	<5.E-6
98-2813 B-Cell 2	1.00E-4 14%	<2.E-5	5.88E-5 24%	4.12E-5 27%	<5.E-6
98-2814 B-Cell 3	3.65E-3 4%	8.00E-4 7%	2.28E-3 6%	5.72E-4 8%	<3.E-5
Standard	101%				
Sample Spike	101%				
Blank	<2.E-5				