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EVALUATION OF A GAMMA MONITOR FOR SURVEY OF WASTE FOR SHALLOW LAND BURIAL

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SAVANNAH RIVER LABORATORY
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PREPARED FOR THE U.S. DEPARTMENT OF ENERGY UNDER CONTRACT AT(07-2)-1

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ABSTRACT

During an 8 month evaluation of the Gamma Waste Monitor, 54 reactor-area scrap casks and 8 shipments from the separations areas containing fission product activities were measured. It was concluded that a more accurate and detailed record of radionuclide burials can be obtained by using this monitor than by present methods of estimation. The monitor will significantly improve records and will provide a more reliable data base for evaluating long-term effects on the environment. Although the monitor can directly detect only gamma-emitting radionuclides, a method is proposed in this report for estimating the total radionuclide distribution by association with radionuclides that are detected. The monitor cannot assay transuranic waste and tritium. In many cases where one radionuclide emits two gamma rays of different energy, corrections for attenuation at other photon energies can be empirically derived from the spectrum. Complete details are included in the Appendix for operating the facility. Experimental data gathered during the evaluation period are contained in the body of this report.

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EVALUATION OF A GAMMA MONITOR FOR SURVEY OF WASTE FOR SHALLOW LAND BURIAL

INTRODUCTION

The Gamma Waste Monitor (GWM) is a computer based spectrometer capable of detecting and quantifying specific gamma-emitting radionuclides in waste (Figure 1). The radionuclides emitting gamma rays of sufficient energy to penetrate waste shipping containers can be identified and measured directly. Other radionuclides which are either pure beta emitters or have gamma energies too low to escape from the package, can often be estimated by association with those photopeaks that are detected.

The total curie quantity and radionuclide composition of all significant levels of solid waste shipped to the burial ground must be estimated to compare additions to the annual limits specified in our Technical Standard. These limits are:

Radionuclide	Annual Limit (Ci)
^{137}Cs	5×10^2
^{90}Sr	5×10^2
^{60}Co	3×10^5
^{3}H	4×10^5
Others with $T_{1/2}$ >10 yr	1×10^3
Others with $T_{1/2}$ <10 yr	5×10^5

Current estimates for burial records are based on the best available information. In the past these included portable survey instrument readings, and detailed assays of typical waste shipments.

By using the Gamma Waste Monitor, the improved inventory records will be useful for evaluating the long-range impact of burial ground operations on the environment and will assist in the formulation of a practical decommissioning plan for the burial site. The information needed to make long-range environmental impact evaluations includes identification of specific radionuclides and the quantities of each committed to burial.

FIGURE 1. Gamma Waste Monitor

Present inventory records of beta-gamma nuclides are based on ionization chamber measurements of the total radiation field (mR/hr) at a fixed distance from the waste container. These measurements are made at the facility generating the waste. No measurement for inventory purposes is being made currently at the burial site. Corrections are made for shielding due to the type of shipping container being measured, and conversion factors are determined to relate the exposure rate observed (mR/hr) to curies of activity present. The result is accurate only if the distribution of radionuclides causing the measured external field remains the same as the distribution that was used to calibrate the instrument. Realizing the limitations of this method for showing compliance with annual limits of radionuclides specified in our Standard, a project was initiated to improve our solid waste assay capabilities. Estimated project cost was $175,000 for construction, plus $52,000 for instrumentation.

Preliminary experiments with portable gamma spectrometers and solid waste forms common to plant operations established the feasibility of this monitoring approach. The Gamma Waste Monitoring Facility was constructed to further evaluate the capabilities and limitations of this technique.

DESCRIPTION OF OPERATION

Overview

Solid waste generated in any plant facility must be packaged to prevent the spread of contamination during transfer, and shielded to prevent the exposure of operating personnel. This waste is sent to the burial ground by either rail or truck. A radioactivity shipping record is provided by the originator which includes an identification number, survey results, type of waste, estimated quantity of the principle radionuclide, point of origin, volume, date, and other pertinent data. The identification number of this record becomes part of a computer record (the COBRA System) which includes map coordinates of the point where the waste was buried.

The GWM is located near the main entrance road to the burial ground. The detector is mounted on a movable cross-arm which can be positioned over a graveled road bed or over a spur of the railroad track. The detector can be raised up to about 95 ft above the ground. This is necessary to adjust the radiation intensity incident on the detector and to simplify geometric corrections for different package sizes and shapes.

Waste arriving at the burial ground is routed to the monitor, where the burial shipment number and other data are entered into the record. The waste is positioned under the detector and counted (Figure 2) before being deposited in its assigned burial location. A background count is sometimes required on heavily shielded casks that are reused.

Data collected by counting the waste package are analyzed by a mini-computer (See Appendix). This analysis includes the identification of detected photopeaks with a particular radionuclide, corrections for attenuation of the package, and geometric corrections for the height of the detector. Results are tabulated in the form of curies of each radionuclide detected, which can be included in the large computer record system.

Facilities

The GWM consists of a trailer, earthen shield and a vertical tower (Figure 1). The trailer provides an office for the operator and a temperature-controlled environment for the electronics (Figure 3). The earthen shield between the trailer and the tower is to protect the operator whenever highly radioactive waste packages are being monitored. Controls on the tower permit the operator to raise or lower the detector with an electric winch. The detector must be lowered to ground level about every five days in order to be filled with liquid nitrogen. If all waste shipments estimated to contain >2 Ci are monitored, the anticipated monitoring load is about one shipment per day.

Time Requirements

It requires 30 min to ready the facility in the morning for routine monitoring, and an additional 30 min twice a week to fill the Dewar on the detector with liquid nitrogen. Most lightly shielded waste packages can be counted in 5 to 10 min, while heavily shielded packages may require up to 30 min. Analysis of the data can usually be accomplished in 30 to 40 min depending on the number of isotopes and whether the isotopes detected correspond with one of the 13 isotopes in the computer's library. Scrap metal casks from the 100 Area are assayed before dumping the cask and, if the radiation intensity is low, may require a second count on the empty cask to accurately establish contributions from low levels of exterior contamination.

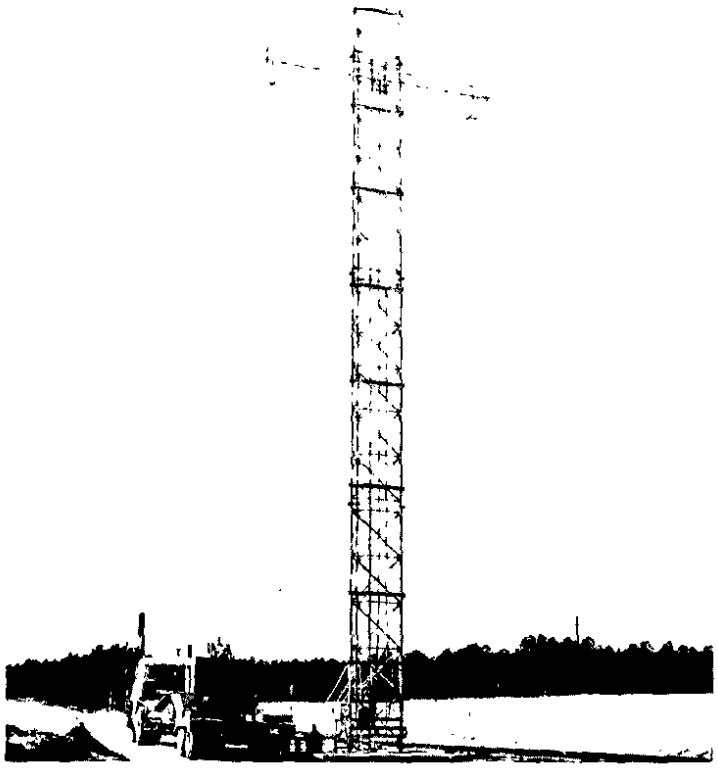


FIGURE 2. Monitor with Waste Positioned Under the Detector

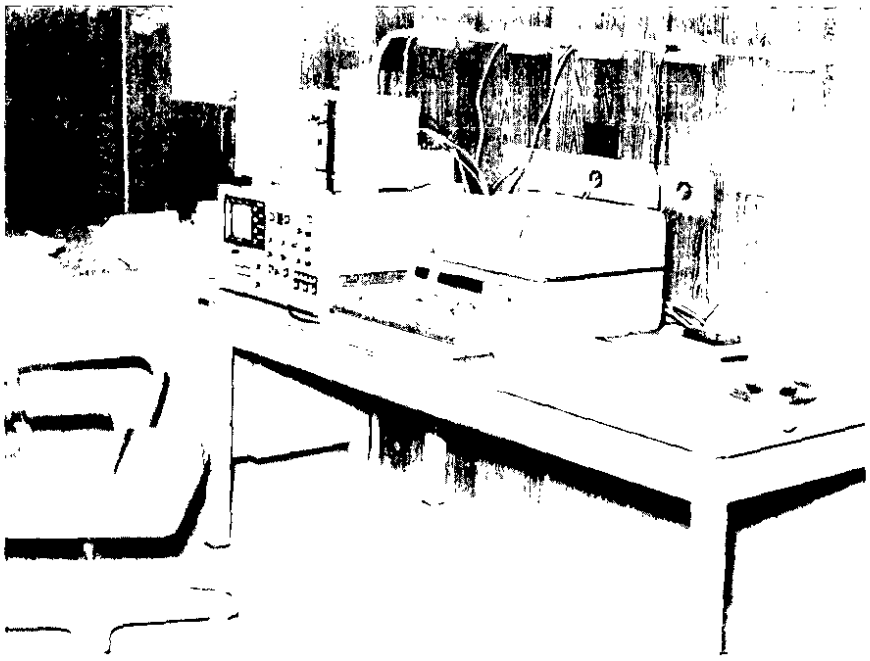


FIGURE 3. Temperature-Controlled Trailer for the Equipment

Known Source-to-Detector Distances

For waste packages that can be positioned beneath the detector. the source to detector spacing is read directly from a height indicator at the base of the tower. This indicator is accurate to the nearest 0.1 foot. If distances greater than 95 ft. are needed to reduce the radiation field at the detector, then the source can be moved away from the tower.

For example, the detector can be set at a height appropriate to measure many types of waste shipped by rail without using the spur line (Figure 4a). This has the advangage of avoiding many time-consuming problems of using the spur line to get the waste closer to the detector.

Figure 4b shows two points on the roadbed which have been marked by stakes to give a slant distance of 100 ft and 200 ft at a source height of 6 ft off the ground. The 100 ft position has been used once for a shipment of warm canyon jumpers, but we have not used the 200 ft position to date.

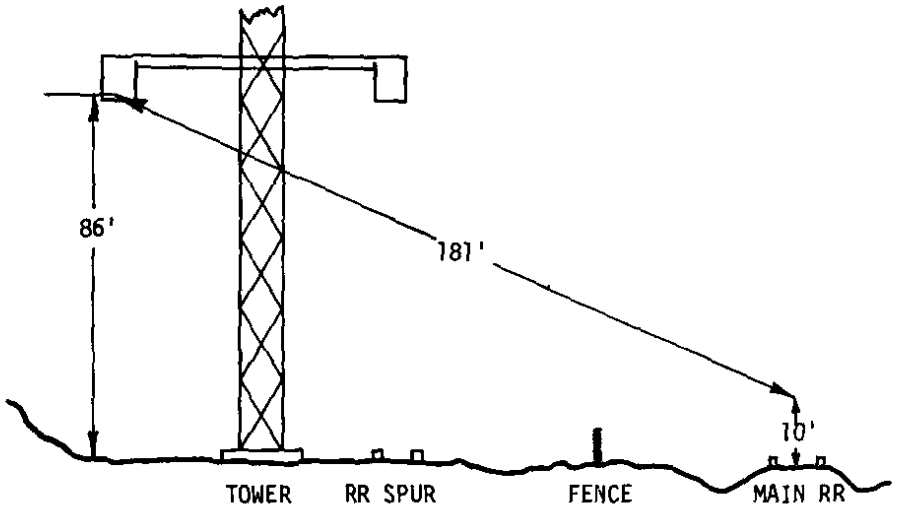
OPERATING EXPERIENCE

Measurement of Induced Activities in 100-Area Scrap Metal Casks

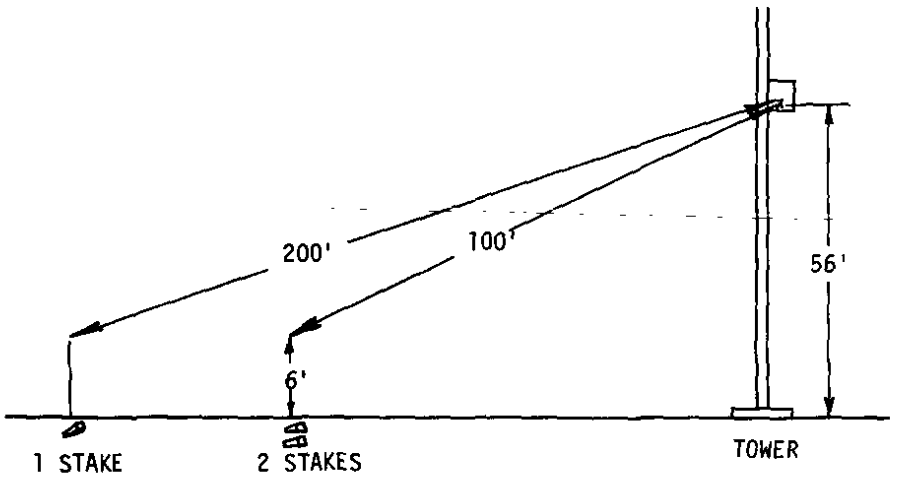
Scrap metal shipments to the burial ground from the 100 Areas average about 2×10^5 Ci annually, according to COBRA records based on the old estimating methods. About 98% of the activity was believed to be ^{60}Co which emits two gamma rays per nuclear transformation (beta decay). One gamma ray occurs in 99% of the transformations and has an energy of 1,173.2 keV. The other occurs 100% of the time and has an energy of 1,332.5 keV. The intensity of these two gamma rays are generally equal in intensity if little or no shielding is present. But when these two gamma rays pass through thick layers of steel or lead before reaching the detector, the lower energy ray is more strongly absorbed than the higher energy ray. The effective thickness of the cask can be calculated once the relative intensity of these two photopeaks have been measured. The calculation, however, is sensitive to the linear absorption coefficient of the material.

Scrap metal is transported to the burial ground in steel-lined lead-filled casks. The outer wall is 0.5 in. of steel followed by 3 in. of lead with an inner wall of 0.5 in. steel. There are three such casks in current use, all of similar design. The attenuation of gamma rays is so great that only photons with energies higher than 1 MeV can be detected through the cask. Typically, the monitor can detect ^{65}Zn (1115 keV) and the two ^{60}Co gamma rays (1173 and 1332 keV) when they are present in sufficient amounts. The radiations associated with all other

radionuclides are generally absorbed by the walls of the cask. The monitor is, therefore, incapable of directly verifying the kinds and quantities of other radionuclides that may be present inside the cask. These other induced activities can be estimated indirectly by using a ratio method that will be discussed later.



a. Distance to Mail Rail Line (Looking West)



b. Marked Distance Past Monitor (Looking North)

FIGURE 4. Diagram Showing Source to Detector Distances

Shield Thickness Determination

A short program was added at the end of the "Analyze Program" to calculate attenuation factors for the scrap metal cask lid (See Appendix). This was necessary since the program computes the curies of radionuclides detected at the cask surface uncorrected for attenuation. These data must be corrected in two ways: 1) to subtract the quantity of external contamination from the result before applying shielding factors, and 2) to apply the shielding factor to any residual in order to estimate the quantity of radionuclide inside the cask. After the Analyze Program is run, a second program to accomplish these corrections is accessed by the command "CONT 8820, Execute." It will ask the operator to input three values in this order from the Analyze Program: 1) curies printed for the 1115 keV peak of ^{65}Zn, 2) curies printed for the 1173 keV peak of ^{60}Co, and 3) curies printed for the 1332 keV peak of ^{60}Co. The program will take these data and print the estimated curies of ^{60}Co on the outside of the cask, the effective thickness of lead used for calculating the attenuation factors, and the curies of ^{65}Zn and ^{60}Co inside the cask.

This program was derived from four empirical measurements of scrap casks that were monitored first with the lid on, then with the lid off. A second set of measurements was made with the lid on and off after the cask was emptied. The results of these measurements are given in Table 1 for the three photopeaks from ^{65}Zn and ^{60}Co. Subtracting the background remaining on the cask after it was emptied and dividing by the activity inside the cask, we can calculate the attenuation coefficient of the cask for these energies. These attenuation factors, numerically equal to I/I_o, are shown in Table 2. Since the cask lid is mostly lead, the linear adsorption coefficient for lead was used to calculate the effective thickness of the lid which averaged about 12 cm.

Once this value is known, it establishes the ratio of the two ^{60}Co peaks as:

$$\text{Ratio} = \frac{e^{-\mu_L x}}{e^{-\mu_H x}} \qquad (1)$$

x = Thickness (cm)
μ = Linear adsorption coefficient (cm^{-1})

for all ^{60}Co radiations coming from inside the cask. A ratio greater than this indicates exterior contamination which can be calculated from the measured data. In simplified form, the logic is as follows:

TABLE 1

Lid-on and Lid-off for 100-Area Scrap Casks

Test No.	Energy (keV)	Nuclide	Cask Full		Cask Empty	
			Lid On (Ci)	Lid Off (Ci)	Lid On (Ci)	Lid Off (Ci)
1	1115.5	Zn	1.0418×10^{-3}	4.9712	(Not detected)	(Not detected)
	1173.2	Co	2.478×10^{-3}	1.6287	2.095×10^{-3}	6.4912×10^{-2}
	1332.5	Co	2.679×10^{-3}	1.6871	1.955×10^{-3}	6.600×10^{-2}
2	1115.5	Zn	5.5397×10^{-4}	4.0458	(Not detected)	5.3012×10^{-3}
	1173.2	Co	1.121×10^{-3}	1.3624	9.1296×10^{-4}	5.529×10^{-2}
	1332.5	Co	1.267×10^{-3}	1.4109	9.8774×10^{-4}	5.7398×10^{-2}
3	1115.5	Zn	7.463×10^{-4}	4.4690	(Not detected)	(Not detected)
	1173.2	Co	3.615×10^{-3}	1.5335	2.944×10^{-3}	6.515×10^{-2}
	1332.5	Co	4.111×10^{-3}	1.5319	3.098×10^{-3}	6.595×10^{-2}
4	1115.5	Zn	1.254×10^{-3}	6.0045	(Not detected)	(Not detected)
	1173.2	Co	1.628×10^{-3}	1.8196	9.065×10^{-4}	5.603×10^{-2}
	1332.5	Co	1.996×10^{-3}	1.8951	9.722×10^{-4}	5.755×10^{-2}

TABLE 2

Attenuation Coefficients for Scrap Casks

Energy	$\mu(Cm^{-1})$	Measured Attenuation Factors (I/I_o)				Average	Effective Lid Thickness (Cm, Lead)
		Test 1	Test 2	Test 3	Test 4		
1115.5	0.735	2.10×10^{-4}	1.369×10^{-4}	1.67×10^{-4}	2.08×10^{-4}	1.80×10^{-4}	11.73
1173.2	0.677	2.88×10^{-4}	1.53×10^{-4}	4.37×10^{-4}	2.40×10^{-4}	3.97×10^{-4}	11.56
1332.5	0.610	4.29×10^{-4}	1.98×10^{-4}	6.61×10^{-4}	5.68×10^{-4}	5.37×10^{-4}	12.34
						Average =	11.87

$$\frac{I_L + C}{I_H + C} = \frac{I_o e^{-\mu_L x}}{I_o e^{-\mu_H x}} + C \qquad (2)$$

where: L refers to the low energy ^{60}Co peak (1173.2)
H refers to the high energy ^{60}Co peak (1332.5)
C is the Ci of external ^{60}Co contamination.

In most waste shipments, the C on the right-hand side of the equation is small compared to the intensity penetrating the cask. I_o is the same for both energies giving:

$$I_L + C = 0.4475\ (I_H + C)$$

$$C = \frac{0.4475 I_H - I_L}{0.5525} \qquad (3)$$

C can be subtracted from both the low and high energy peaks and the residuals can be divided by its respective attenuation coefficient to obtain the estimated curie content of the cask. In the actual case used in the program, equation 3 was solved without dropping C from the equation.

Monitoring Results of 100-Area Scrap Casks

In the first half of 1978, the GWM was used to monitor 54 shipments of scrap metal coming to the burial ground. These measurements were made to obtain experience with the monitor and to determine if it could provide useful information not available using present techniques for inventory record purposes. Table 3 summarizes monitoring results of scrap casks shipped to the burial ground during this period. Generally, the estimates taken originally by the area surveyor are low and agreement is poor compared to the values detected by the monitor. The measured values were 3 times higher than the estimated values for ^{65}Zn and ^{60}Co.

TABLE 3

Observed Isotopes in 100-Area Scrap Casks
Includes Only Those Monitored at Building 643-12G
(Data and Program in SRL. Edit 2139 - File Called Datacask)

File No.	Estimated Ci*	Area	Observed with GWM ^{65}Zn	Observed with GWM ^{60}Co
99559	11.5	105-P	27.0	9.0
99586	6.9	105-P	0.0	3.5
99587	6.9	105-P	0.0	1.4
99588	6.9	105-P	16.0	11.1
99580	4.6	105-P	22.0	3.0
99591	80.5	105-P	118.4	81.5
99593	6.9	105-P	7.7	3.0
99596	5.7	105-P	58.6	24.2
97200	23.0	105-P	56.1	56.0
99607	6.9	105-P	66.5	47.7
102296	6.9	105-P	34.0	5.4
102764	6.9	105-P	96.1	25.1
91189	23.0	105-C	10.0	2.0
91190	23.0	105-C	12.0	1.6
91191	23.0	105-C	0.0	91.0
91193	23.0	105-C	0.0	5.0
91194	23.0	105-C	17.0	7.5
91196	23.0	105-C	0.0	161.0
91197	161.0	105-C	0.0	1048.0
93794	17.5	105-C	0.0	645.0
93805	69.0	105-C	13.0	272.0
394229	2.1	105-C	2.0	2.0
93814	340.0	105-C	0.0	1883.1
93815	69.0	105-C	131.7	428.9
93816	340.0	105-C	0.0	331.2
93818	338.0	105-C	145.0	541.1
93822	13.8	105-C	127.4	301.5
93824	6.9	105-C	205.5	72.9
93825	34.0	105-C	143.2	219.2
103116	230.0	105-C	236.5	354.1
99526	9.2	105-C	33.9	4.9
101012	138.0	105-C	266.0	453.1
101018	690.0	105-C	0.0	1421.0
101021	11.5	105-C	72.6	207.2
101022	11.5	105-C	133.5	417.2
101029	184.0	105-C	108.2	400.0
101041	69.0	105-C	110.2	199.7
103106	34.5	105-C	134.0	254.2
101749	11.5	105-C	65.4	11.0
101048	230.0	105-C	103.7	337.3
~~103102~~	~~460.0~~	105-C	42.9	84.0
101047	115.0	105-C	32.2	230.7
97377	690.0	105-C	61.5	641.5
97812	12.5	105-K	0.0	19.0
97814	11.5	105-K	0.0	67.0
97821	11.5	105-K	16.6	3.1
97819	11.3	105-K	0.0	1.0
97823	11.5	105-K	26.0	7.5
97824	11.5	105-K	22.0	6.0
97825	11.5	105-K	24.0	12.0
97826	13.8	105-K	35.0	7.0
97827	11.5	105-K	18.0	15.0
97820	2.3	105-K	40.9	7.4
97807	2.3	105-K	13.8	5.4
504470	2.3	105-K	18.9	5.6
504445	2.3	105-K	13.2	2.7
504446	2.3	105-K	33.3	5.4
504814	27.6	105-K	0.0	5.2
101745	11.5	105-K	48.1	13.2
Totals	4744.3		3019.6	11482.3

* Recorded as 98% ^{60}Co.

Other Induced Activities in Scrap Metal

It is known that many other radionuclides are formed whenever metals are exposed to a high flux of thermal neutrons over an extended period of time. The monitor could detect more of these induced activities if the lid of the cask were removed during assay. There are two problems with removing the lid which make this approach impractical: 1) the radiation intensity saturates the electronics if the cask contains more than 5 Ci of activity, and 2) the lid weighs several tons and requires a crane, a crew, and Health Physics coverage whenever it is removed. There is also an increased chance of contaminating the monitoring area. Even if the lid is removed, that are still many radionuclides that do not emit gamma radiation. They decay by beta or alpha, internal conversion, or K-capture. None of these transformations can be detected by the monitor unless they are accompanied by penetrating gamma emission.

Currently, these other radionuclides must be estimated from the quantities of ^{65}Zn or ^{60}Co that can be directly measured. This technique is inexact because some of the major radionuclides formed are due to impurities in the metal which probably vary widely according to the source of supply. For example, Co is not listed as an alloy element for either stainless steel or aluminum. Yet, ^{60}Co is definitely present in both metals when they are discharged from our reactors. In addition, we must use the abundance of ^{60}Co, even though it is an impurity, as a basis for estimating the quantities of other induced radionuclides.

Current standards require annual inventory limits for ^{60}Co and other isotopes, so we are obligated to make an attempt at apportionment even if the technique used may be approximate.

Because of the limited memory and programming capabilities at the GWM, the apportionment of radionuclides not directly measured by the system can best be accomplished on a larger computer as part of a data interpretation program.

The estimation of radionuclides associated with ^{65}Zn and ^{60}Co was originally thought to be simple. The reactor flux, the metallurgy, and the decay periods are all rigidly controlled. However, upon performing activation calculations and comparing these with experimental results, the apportionment of activity in irradiated scrap metal was found to be extremely complex. Activation of trace impurities in the metal often produced more radioactivity than could be accounted for from the known chemical compositions of the alloy. For these reasons, the calculations of induced activities presented here should be viewed as tentative. Many variables exist that were either not considered or were assumed to be constant in order to simplify the calculations. This one aspect of the waste monitor, the apportionment of radionuclides, could be made a separate study in itself.

The calculations involving activation products produced in our reactors were based on the chemical compositions given in Table 4.

Using the elemental compositions of these metals and assuming an average flux of 2×10^{14} n/cm^2-sec., exposure time of 100 days, and a decay time of 0.8 year, the quantities of radionuclides present at the end of that time were calculated. For simple neutron capture (ignoring burnup), the quantity of an isotope formed per unit weight of element is given by the following equation:

$$Ci/g = \phi\sigma NKA\ (1-e^{-\lambda t_1})e^{-\lambda t_2}/3.7 \times 10^{10}\ d/sec \cdot Ci$$

Where ϕ = flux (n/cm^2/sec)

σ = cross section (10^{-24} cm^2/barn)

N = Atoms = $\dfrac{(g)(6.023 \times 10^{23})}{At.\ Wt.}$

K = Fractional natural abundance of isotope

A = Fraction of element present in alloy

λ = Decay constant = $0.693/T_{\frac{1}{2}}$

t_1 = Exposure time (same units as $1/\lambda$)

t_2 = Decay time (same units as $1/\lambda$)

Induced Activities in 304L Stainless Steel

From the elemental composition of stainless steel listed in Table 4, we would expect the nuclear reactions listed in Table 5 to occur. Results of these calculations are listed in the next to last column of Table 5. The last column is the ratio of each radionuclide formed relative to the quantity of ^{60}Co formed. Assuming these ratios are correct, the apportionment of all other radionuclides in stainless steel scrap can be estimated once the curies of ^{60}Co in stainless steel is measured.

Unfortunately, the entire apportionment is dependent on measured quantities of ^{60}Co. As expalined in the footnote of Table 5, the amount of ^{60}Co produced from activation of ^{58}Fe gives an unbelievably low value. If true, it would make the quantities of the other activation products extremely high. If, however, ^{59}Co is assumed to be a trace impurity in stainless steel (possibly an impurity of Ni) then the ratios change in proportion to the quantity of this impurity. Also, unless the ^{59}Co impurity is relatively constant, the apportionment based on measured amounts of ^{60}Co will always be in error. It is not known how much the ^{59}Co varies in stainless steel but for the present, it will be assumed to be a constant of 0.05%.

TABLE 4

Elemental Composition of Common Stainless Steel and Aluminum Alloys

Stainless Steel		*Aluminum*		
Element	*Type 304L % by Wt.*	*Element*	*Type 6061 ~% by Wt.*	*Type 8001 % by Wt.*
C	0.08	Si	0.60	-
Mn	2.00	Fe	0.70	0.70
Si	1.00	Cu	0.30	-
Cr	19.00	Mn	0.15	0.05
Ni	9.50	Mg	1.00	-
Fe	68.5	Cr	0.25	-
		Zn	0.25	0.10
		Ti	0.15	-
		Ni	-	0.80 - 1.35
		Al	96.60	98.40

TABLE 5

Calculated Activation Products from Irradiation of Stainless Steel Type 304L

Probable Reaction	*Abundance*		$T_{½}$	$\sigma(10^{-24}$ *Cm²/Atom)*	$\lambda(y^{-1})$	*Estimated Quantity Formed (Ci/g)*	*Ratio (Isotope/ ^{60}Co)*
	Natural	*Stainless Steel*					
$^{13}C(n,\gamma)^{14}C$	1.1×10^{-3}	8×10^{-4}	5,760y	1.2×10^{-4}	1.2×10^{-4}	7.3×10^{-9}	2×10^{-7}
$^{30}Si(n,\gamma)^{31}Si$	3.09×10^{-2}	0.01	2.6 hr.	0.11	2.3×10^{3}	(All decays)	0.0
$^{31}Si(n,\gamma)^{32}Si$	($\sim9.6\times10^{-11}$)		700y	0.47	9.9×10^{-4}	1.3×10^{-12}	3.7×10^{-11}
$^{50}Cr(n,\gamma)^{51}Cr$	4.3×10^{-2}	0.19	27.8d	17	9.10	5.7×10^{-3}	0.16
$^{58}Ni(n,\gamma)^{59}Ni$	6.78×10^{-1}	0.095	7.5×10^{4}y	4.6	9.2×10^{-6}	4.2×10^{-5}	1.2×10^{-3}
$^{62}Ni(n,\gamma)^{63}Ni$	3.66×10^{-2}	0.095	100y	15	$.693\times10^{-2}$	5.2×10^{-3}	0.15
$^{54}Fe(n,\gamma)^{55}Fe$	5.84×10^{-2}	0.685	2.7y	2.9	.2567	0.475	13.3
$^{58}Fe(n,\gamma)^{59}Fe$	3.3×10^{-3}	0.685	45d	1.1	5.62	1.21×10^{-3}	3.4×10^{-2}
$^{59}Fe(n,\gamma)^{60}Fe$	($\sim3.5\times10^{-8}$)		1×10^{5}y	~1.0	6.9×10^{-6}	3.6×10^{-12}	1×10^{-10}
↓β-							
$^{59}Co(n,\gamma)^{60}Co$	(Beta decay of ^{59}Fe)		5.26y	37	0.132	1.4×10^{-4}	(not used)*
$^{59}Co(n,\gamma)^{60}Co$	1.0	$\sim5\times10^{-4}$	5.26y	37	0.132	3.56×10^{-2}	1.00*
$^{54}Fe(n,p)^{54}Mn$	5.82×10^{-1}	0.685	278d	0.25	0.91	5.1×10^{-3}	0.14

*^{60}Co formed from the complex activation of ^{58}Fe, the beta decay of ^{59}Fe, then activation of ^{59}Co is too low to account for the quantity of ^{60}Co measured. An impurity of 0.05% ^{59}Co in stainless steel is indicated from samples exposed to reactor neutrons. Therefore, this empirical value was used to calculate ratios.

In an effort to explore this theory further, 2 one-g samples of 304 stainless steel exposed for 1 hr to a thermal neutron fluence of 1×10^{15} n/cm^2 were analyzed spectrometrically for gamma-emitting radionuclides. Both samples gave essentially the same results, and showed the presence of activation products that were produced by neutron capture of unlisted trace elements in the sample. These results are shown in Table 6. It should be pointed out that radionuclides that decay by beta emission (no gamma) would not be detected by this spectrometric analysis. Also, the ratios given in the last column have been corrected for equilibrium, decay, and other conditions to make these numbers compatible with data listed in Table 5. An error that could exist in the analysis of these irradiated samples involve the incorrect identification of an isotope with respect to its photopeak energy detected. Many of the isotopes, however, emit two gamma rays which add to the confidence of the identification of a particular radionuclide.

Comparing the calculated ratios in Table 5 with observed ratios in Table 6 (for those isotopes that emit gamma rays) a reasonable agreement is found for ^{51}Cr, ^{59}Fe, and ^{54}Mn. Additional studies are needed to prove or disprove the consistency of ^{60}Co and the quantities of some of the other trace elements found in this study. Such an undertaking is beyond the intent of this project, i.e., setting up a waste monitor. But based on the information obtained, it appears that the apportionment given in Table 7 represents the best approximation at this time.

TABLE 6

Observed Activation Products from the Irradiation of Stainless Steel Type 304

Radionuclide ($T_{½}$)	*Photopeak Energy (keV)*	*Activity (pCi/g)*	*Ratio (Isotope/^{60}Co)*
^{59}Fe (46.6d)-	1291.6	4×10^6	0.02
^{59}Fe (46.6d)	1099.0	4×10^6	0.02
^{60}Co (5.26y)	1332.0	1.9×10^8	1.00
^{60}Co (5.26y)	1172.0	1.9×10^8	1.00
^{54}Mn (312d)	834.8	2×10^7	0.11
^{99}Mo (66.6 hr)	739.6	Decays to ^{99}Tc	-
^{99}Tc (2.1×10^5y)	739.6*	5.4	2.8×10^{-8}
^{122}Sb (2.72d)	686.0	Decays	-
^{59}Fe (46.6d)	192.2	4×10^6	0.02
^{99}Mo (66.6 hr)	140.5	Decays to ^{99}Tc	-
^{51}Cr (27.7d)	320.0	3.2×10^7	0.16
^{75}Se (120d)	136.0	2.7×10^7	0.14

*Decayed from ^{99}Mo.

TABLE 7

Apportionment of Induced Activities per Ci of ^{60}Co in Stainless Steel

Radionuclides with $T_{1/2}$ >10 years		*Radionuclides with $T_{1/2}$ <10 years*	
Isotope ($T_{1/2}$)	*Ci of Isotope/ Ci of ^{60}Co*	*Isotope ($T_{1/2}$)*	*Ci of Isotope/ Ci of ^{60}Co*
^{14}C (5,760y)	$2x10^{-7}$	^{51}Cr (27.7d)	0.16
^{32}Si (700y)	$3.7x10^{-11}$	^{55}Fe (2.7y)	13.30
^{59}Ni ($7.5x10^{4}$y)	$1.2x10^{-3}$	^{59}Fe (46.6d)	0.02
^{63}Ni (100y)	0.15	^{60}Co (5.26y)	1.00
^{60}Fe ($1x10^{5}$y)	$1x10^{-10}$	^{54}Mn (312d)	0.11
^{99}Tc ($2.1x10^{5}$y)	$5x10^{-12}$	^{75}Se (120d)	0.14

This table (or a corrected table similar to the above) can be used to calculate and maintain a continuous inventory of these isotopes with respect to our Standard.

Apportionment of Induced Activities in Aluminum Scrap

Other than stainless steel, aluminum is the other most prominent metal associated with irradiated scrap. There are at least three major aluminum alloys used: series 1000, 6000, and 8000. The radionuclide common to all three of these alloys is ^{65}Zn. It has a half-life of 245 days, and a photopeak energy of 1115 keV. Zn-65 also decays by positron emission 1.7% of the time giving rise to annihilation peak at 511 keV, and a photon yield of 0.5075. All three alloys of aluminum contain zinc (0.1 to 0.25%), but stainless steel does not. Therefore, since the waste monitor can detect ^{65}Zn through the cask, this isotope can serve to indicate the presence of aluminum as well as provide a basis for estimating the abundance of other radionuclides associated with aluminum.

The monitor cannot detect which alloy of aluminum is present. Therefore, one type must be assumed for the purpose of apportionment. Since series 1000 aluminum is nearly pure (containing only 0.2% copper, 0.05% manganese, and 0.10% zinc), an apportionment based on this alloy would probably underestimate the radionuclide content of an average scrap metal cask. At the other extreme, series 8000 aluminum is similar in elemental composition to 6000 series (see Table 4) except for its nickel content (0.8 to 1.35%). Using this alloy as a basis for apportionment would result in a high inventory of ^{60}Co being attributed to the aluminum scrap and thus reduce the apportionment of radionuclides attributed to stainless steel which might be in the same cask.

It appears that an apportionment based on 6000 series aluminum is a middle-of-the-road approach. And, for the purpose of the monitoring program at this stage, that series represents a reasonable basis for estimating the inventory of other radionuclides associated with irradiated aluminum scrap.

In order to estimate the activity and isotopic abundance characteristic of 6000 series aluminum, a 400-g end section of a Mark 53A fuel housing tube was obtained from 105-P Area. The aluminum type was 6063-T833, decayed for ∿4½ months. The results of a gamma spectrometric analysis on March 7, 1978, is given in Table 8.

Activation calculations, similar to those made for stainless steel, were made to estimate the quantities of other radionuclides produced in 6000 series aluminum during a reactor cycle. Table 9 lists the quantities of nongamma-emitting radionuclides that should also be present in the irradiated aluminum relative to ^{65}Zn.

The best estimate of apportionment of these radionuclides per Ci of ^{65}Zn in accordance with the Plant Standard is given in Table 10.

TABLE 8

Gamma Spectrometric Analysis of a 6063 Aluminum Housing (Decayed 4½ Months)

Isotopes ($T_{½}$)	*Energy (keV)*	*Measured Activity (Ci/g)*	*Ratio (Isotope/^{65}Zn)*
^{181}Hf (42.5d)	133.42	4.7×10^{-5}	0.25
^{51}Cr (27.8d)	320.17	1.0×10^{-4}	0.53
^{175}Hf (70.0d)	343.50	3.8×10^{-6}	2.0×10^{-2}
^{181}Hf (42.5d)	482.21	3.6×10^{-5}	0.20
^{95}Zr (65d)	723.42	7.9×10^{-6}	4.2×10^{-2}
^{95}Nb (35.1d)	765.36	1.1×10^{-5}	5.8×10^{-2}
^{54}Mn (278d)	834.72	1.49×10^{-5}	7.8×10^{-2}
^{46}Sc (83.8d)	889.03	3.3×10^{-5}	0.17
^{59}Fe (45d)	1099.02	5.4×10^{-5}	0.28
^{65}Zn (245d)	1115.34	1.9×10^{-4}	1.00
^{60}Co (5.26y)	1173.14	1.1×10^{-5}	5.8×10^{-2}
^{59}Fe (45d)	1291.65	5.4×10^{-5}	0.28
^{60}Co (5.26y)	1332.47	1.1×10^{-5}	5.8×10^{-2}
^{46}Sc (83.8d)	1120.55	3.3×10^{-5}	0.17

TABLE 9

Nongamma-Emitting Radionuclides in 6000 Series Aluminum

Isotopes ($T_{½}$)	*Calculated Activity (Ci/g)*	*Ratio (Isotope/^{65}Zn)*
^{55}Fe (2.7y)	4.8×10^{-4}	2.5
^{60}Fe (1×10^{5}y)	2.5×10^{-14}	1.3×10^{-10}
^{32}Si (700y)	1.3×10^{-12}	6.8×10^{-9}
^{59}Ni (7.5×10^{4}y)	4.2×10^{-6}	2.2×10^{-2}
^{63}Ni (100y)	5.2×10^{-4}	2.7

TABLE 10

Apportionment of Induced Activities per Ci of ^{65}Zn in 6064 Aluminum

Radionuclides with $T_{½}$ >10 years

Isotope ($T_{½}$)	*Ci of Isotope/ Ci of ^{65}Zn*
^{60}Fe (1x10y)	1.3×10^{-10}
^{32}Si (700y)	6.8×10^{-9}
^{59}Ni (7.5x10y)	2.2×10^{-2}
^{63}Ni (100y)	2.7

Radionuclides with $T_{½}$ <10 years

Isotope ($T_{½}$)	*Ci of Isotope/ Ci of ^{65}Zn*
^{181}Hf (42.5d)	0.25
^{51}Cr (27.8d)	0.53
^{95}Zr (65d)	4.2×10^{-2}
^{95}Nb (35.1d)	5.8×10^{-2}
^{54}Mn (278d)	7.8×10^{-2}
^{46}Sc (83.8d)	0.17
^{59}Fe (45d)	0.28
^{65}Zn (245d)	1.00
^{60}Co (5.26y)	5.8×10^{-2}
^{55}Fe (2.7y)	2.5

Fission Product Waste From Separation Areas

Most of the waste from the HM and Purex processes ends up as a liquid fraction that is stored in waste tanks. Only a small fraction is associated with solid waste to be disposed of in the burial ground. The solid waste from the separation areas generally accounts for less than 10% of the annual burial ground radionuclide inventory, but is the source of most of the Cs and Sr which have the lowest limits in our Standard.

There is a steady flow of low-level waste from the separation areas to the burial ground. Except for canyon equipment, laboratory sample vials, filters, and waste tank components, most of the solid waste is less than 2 Ci per shipment. Seldom is the waste heavily shielded as is the case with induced activities in 100-Area scrap metal or spent melts from the tritium facilities. Therefore, most of the gamma-emitting radionuclides, typically 8 to 10 radionuclides, can be directly measured by the waste monitor. Nuclides that emit more than one gamma ray per nuclear transformation can be used to determine if the effective attenuation of the package is significant. If determined to be significant, corrections can be made for attenuation at all energies. The following example will make this clearer.

Table 11 shows the results of monitoring a dumpster pan from 772-F at a source-to-detector distance of 30 ft. These data are presented to demonstrate the detail of the spectral information available from the monitor. The analyzer will search the recorded spectrum and print the peak energy and total counts. The operator must identify these peaks with specific isotopes and list their gamma yields. The second half of the Analyze Program searches a library automatically to match peak energy with isotope, but all of these isotopes are not in the library and usually only one energy is listed for a particular isotope. A second general program has been written and is stored in File #25. This program permits the operator to identify the isotope and calculate the curies of activity present. The result is corrected for height, air absorption, and detector efficiency, but not for any shielding that may be associated with the waste package. Table 11 is the result of this program.

It should be noted that ^{144}Ce, ^{106}Ru, ^{134}Cs, ^{154}Eu, and ^{60}Co all have two or more energies that were detected. In general, the curie quantity as derived from the lower energy peak is less than the curie quantity seen at higher energies. Therefore, ignoring some error due to counting statistics, we would conclude the radiations detected during this measurement had passed through a waste matrix or a container wall that absorbed or scattered the original photons. In a dumpster we would assume the package wall to be iron with an unknown waste matrix. The equation to calculate

shield thickness knowing the ratio of photon intensities (I_L/I_H) and the linear absorption coefficient of the material at these two energies is:

$$Cm = \frac{-\ln(I_L/I_H)}{(\mu_L - \mu_H)}$$

Where: I_L = Intensity of low-energy peak

I_H = Intensity of high-energy peak

μ_L = Linear absorption coefficient of low-energy photons

μ_H = Linear absorption coefficient of high-energy photons

If we select iron as the attenuating material and the 621.7 and 1050.1 keV peaks of ^{106}Ru as the energies we have:

$$Cm \text{ iron equivalent} = \frac{-\ln(0.428/0.460)}{(0.585 - 0.460)} = \frac{0.062}{0.125} = 0.576 \text{ cm}$$

or about 1/4-in. of steel. If we had selected the two photopeaks of ^{144}Ce at 132 and 1489 keV, the equation would give an iron equivalent of 0.492 cm (0.2-in. of steel). All data in Table 11 should therefore be corrected for the attenuation of 1/4-in. of steel.

TABLE 11

Analysis of a Dumpster Pan From 772-F
(Count Time = 300 sec, Detector Height = 30 ft)

Isotope	*Peak Energy*	*γ-Yield*	*Detector Efficiency*	*Total Counts*	*Measured Ci*
^{152}Eu	123.11	0.39	3.5×10^{-8}	7,653	0.018
^{144}Ce	132.34	0.111	3.2×10^{-8}	50,816	0.446
^{103}Ru	496.9	0.864	9.1×10^{-9}	6,000	0.024
^{106}Ru	511.5	0.205	8.9×10^{-9}	25,398	0.452
^{134}Cs	604.4	0.978	7.6×10^{-9}	9.293	0.041
^{106}Ru	621.7	0.098	7.3×10^{-9}	9,508	0.428
^{137}Cs	661.3	0.852	7.2×10^{-9}	49,511	0.273
^{144}Ce	696.3	0.013	6.5×10^{-9}	2,129	0.808
^{95}Zr	723.8	0.43	6.3×10^{-9}	24,181	0.288
^{95}Nb	763.5	0.998	6.1×10^{-9}	121,542	0.658
^{134}Cs	795.4	0.854	5.7×10^{-9}	6,558	0.043
^{154}Eu	873.1	0.117	5.2×10^{-9}	1,038	0.055
^{154}Eu	995.8	0.099	4.7×10^{-9}	491	0.035
^{154}Eu	1004.5	0.17	4.6×10^{-9}	1,007	0.042
^{106}Ru	1050.1	0.015	4.4×10^{-9}	933	0.460
^{60}Co	1173.2	0.999	4.1×10^{-9}	611	0.005
^{154}Eu	1274.2	0.336	3.6×10^{-9}	1,385	0.037
^{60}Co	1332.4	1.00	3.5×10^{-9}	592	0.006
^{134}Cs	1365.2	0.03	3.4×10^{-9}	149	0.048
^{144}Ce	1489.4	0.0029	3.1×10^{-9}	239	0.859

These kinds of corrections involve considerable operator time. They can best be done on a larger computer where memory storage space is available. The point here is that the information to make these corrections is available from the monitor.

During the evaluation of this waste monitor, several types of fission product waste packages were monitored. Table 12 shows a summary of the results obtained. The monitor appears capable of resolving a complex array of isotopes into a meaningful distribution. However, the radioactive shipment record does not presently have space for more than two isotopes. In addition, the isotopes listed do not necessarily represent all that may be present. If these data are to be saved, the record system will have to be modified and enlarged.

TABLE 12

Results of Monitoring some Typical Kinds of Waste Containing Fission Products

RSR No.:	95276	95285	98522	95177	Special	102676	Special	95264
Date:	1/16/78	1/31/78	1/9/78	1/9/78	11/17/77	2/28/78	2/6/78	2/28/78
Area:	772-F	772-F	772-F	772-F	200-F	241-H	221-F	772-F
Description:	Dumpster	Dumpster	Dumpster	Dumpster	9.1 Evap.	Ann. Jet	Tea Pot Filter	Dumpster
Estimated Ci:	0.08	0.80	0.0	0.0	-	-	-	-
Isotope	Ci	Ci	Ci	Ci	Ci	Ci	Ci	Ci
^{144}Ce	0.44	8.92	0.01	0.86	-	-	-	0.33
^{103}Ru	0.01	0.02	-	0.02	0.01	-	-	0.05
^{106}Ru	0.22	4.35	0.01	0.46	1.33	-	-	0.17
^{137}Cs	0.03	0.09	0.01	0.27	0.18	3.0	9.4	0.10
^{95}Zr	0.26	1.20	-	0.29	0.23	-	-	0.17
^{95}Nb	0.24	0.35	-	0.66	0.73	-	-	0.32
^{134}Cs	0.003	0.62	-	0.04	-	0.03	0.2	0.01
^{154}Eu	0.01	0.05	-	0.04	0.04	-	-	-
^{125}Sb	-	0.08	-	-	0.06	-	-	-
^{60}Co	-	0.05	-	-	-	-	-	0.001
^{152}Eu	-	-	-	0.02	-	-	-	-
Total Ci Measured	1.21	15.73	0.03	2.66	2.58	3.03	9.6	1.15

As for apportionment of activity with respect to the Technical Standard, a considerable effort was made to correlate other fission product activities with the quantity of ^{137}Cs measured. But because of the wide range of half lives and fission yields as well as the separations that may favor the solubility of one chemical form over another, no consistant relationship was found. For example, the "Annular Jet" and "Tea Pot Filter" in Table 12 (next to last - 2 columns) show only $^{137-134}Cs$. These items originated from the waste tanks and probably do not contain the

general mixture of fission products existing in the tank proper. Similar difficulties involving decay or isotopic depletion make apportionment based on ^{137}Cs inaccurate with respect to the short-lived isotopes. And where known chemical separations have favored one element over another, it is inaccurate for long-lived species as well. However, the decay of isotopes with half lives >10 years should not introduce large errors in the time frame we are dealing with.

It is recommended that for apportionment, the following procedure be followed:

For Isotopes with $T_{1/2}$ <10 years; record only those isotopes detected by the monitor and record them in the quantities measured (corrected for shielding if significant).

For Isotopes with $T_{1/2}$ >10 years; use the relationship between fission yield and half-life relative to that for ^{137}Cs. This relationship permits the derivation of a factor relative to ^{137}Cs that can be used to approximate the quantity of other long-lived radionuclides that may be present (ignoring chemical isolation).

The factor is obtained as follows:

$$\text{Factor} = \frac{\text{Ci of Unknown}}{\text{Ci of } ^{137}\text{Cs}} = \frac{\text{Fission Yield of Unknown}}{\text{Fission Yield of } ^{137}\text{Cs}} \times \frac{T_{1/2}\ ^{137}\text{Cs}}{T_{1/2} \text{ of Unknown}}$$

$$= \frac{(\text{Ci of Unknown})(4.715)}{T_{1/2} \text{ of Unknown, yr}}$$

The apportionment for isotopes >10 years then becomes:

Isotope ($T_{1/2}$)	*Fission Yield, %*	*Factor* $\left[\frac{Ci(Isotope)}{Ci(^{137}Cs)}\right]$
^{87}Rb (6×10^{10}y)	2.49	2.0×10^{-10}
^{90}Sr (28y)	5.8	0.98
^{93}Zr (1.1×10^{6}y)	6.45	2.8×10^{-5}
^{129}I (1.7×10^{7}y)	0.9	2.5×10^{-7}
^{135}Cs (2.6×10^{6}y)	6.41	1.2×10^{-5}
^{137}Cs (29y)	6.15	1.0
^{144}Nd (5×10^{15}y)	5.67	5.3×10^{-15}
^{147}Sm (1.3×10^{11}y)	2.38	8.6×10^{-11}
^{151}Sm (80y)	0.45	2.6×10^{-2}

Spent Melts from 232-H

Spent melts are shipped to the burial ground in specially designed casks. This cask tapers in thickness from 1 to 5-in.-thick lead with the thicker end covering the bottom of the crucible.

During this evaluation, only one Line 3 melt was measured. It showed only ^{65}Zn and ^{60}Co photopeaks. The observed curies uncorrected for shielding was:

Isotope	*Energy (keV)*	*Curies*
^{65}Zn	1115.5	2.8955×10^{-2}
^{60}Co	1173.2	9.5309×10^{-4}
^{60}Co	1332.5	1.3913×10^{-3}

Using the two ^{60}Co photopeaks to calculate effective shield thickness we have

$$\frac{-\ln(9.5308\times10^{-4}/1.3913\times10^{-3})}{0.067} = 5.64 \text{ cm lead}$$

Correcting the above for attenuation, we have

^{95}Zn = 1.62 Ci

^{60}Co = 0.04 Ci

If additional measurements of spent melts show similar low levels of induced activities, we might elect not to assay spent melts.

In the event that future measurements of spent melts show higher levels of induced activities, the following apportionment of other radionuclides should be considered relative to ^{65}Zn measured:

Isotope ($T_{1/2}$)	*Fraction Relative to ^{65}Zn*
^{65}Zn (245d)	1.00
^{60}Co (5.26y)	Report quantity measured
^{59}Fe (45.1d)	0.12
^{124}Sb (60d)	0.14
^{55}Fe (2.7y)	2.5
^{60}Fe (1×10^{5}y)	1.3×10^{-10}
^{32}Si (700y)	6.8×10^{-9}
^{59}Ni (5.7×10^{4}y)	2.2×10^{-2}
^{63}Ni (100y)	2.7

APPENDIX

TECHNICAL INFORMATION

Instrumentation

The electronics used in the Gamma Waste Monitor are listed below:

Item	*Tag No.*	*EP No.*	*Cost*
MCA CANBERRA (8100-4096 Channels)	L-24350	436WT-1	$10,500.00
Interface-Aston (AST-100)	L-24351	436WT-2	2,500.00
Read/Write Memory/Hewlett Packard (OP-T276)	L-24352	436WT-3	3,760.00
Printer-Thermal/Hewlett Packard (9866A)	L-24353	460BC-2	3,145.00
Prog. Calculator/Hewlett Packard (9830A)	S-19258	460BC	6,800.00
Power Supply/Mechtronics (152)	T-22348	429NP	630.00
Amplifier/Princ. V-Tech. (340)	T-22334	-	725.00
Bias Voltage Supply/Princ. V-Tech. (AEC-5000)	T-22333	-	450.00
Ge Hi Res. Det./Princ. γ-Tech. (55 cm^3)	S/N 252	-	14,080.00
Ge Coaxial Det./Princ. γ-Tech. (500mm^2×5mm)	S/N 253	-	9,120.00
Monitor/Victoreen (808D)	L-24736	436ABS	
Software	-	-	2,500.00
Total Cost of Instrumentation			$51,710.00

Detector Cooling and Bias Voltage Criteria

Both detectors are intrinsic (pure germanium, not lithium-drifted). The loss of liquid nitrogen and warm-up of the detector to room temperature will not affect the detector as long as the bias voltage is off. It requires 5 hrs to recool the detector once it has lost its liquid nitrogen. The operator should not

apply the bias voltage before the detector is cooled because it may damage the pre-amplifier. When the bias voltage is applied, it should be raised slowly (~100 V/sec).

There are two detectors used with the system. Only one can be operated at a time. The normal detector used (S/N 252 - active vol. 55 cm^3) has a high efficiency for photon energies >100 keV and is the more sensitive of the two. It can be identified by its stainless steel Dewar. Operate this detector at a bias of +2,000 volts.

The other detector is highly efficient for low energy photons <100 keV. Its efficiency is about 10 times less than the other detector for high-energy photons and can be useful whenever the radiation intensity is too high for the 55 cm^3 detector. This detector (S/N 253) can be identified by its white Dewar. Its bias voltage is -1400 V. The polarity of the bias voltage is changed by turning a screw-switch on the back of the high voltage Power Supply Chassis. The unit must be unplugged to change the polarity.

The beryllium windows which provide a hermetic seal for the germanium detectors are extremely thin and should never be touched with the fingers or any other object. When filling the Dewars with liquid nitrogen, spilling should be avoided because a rapid temperature change of the pre-amplifier may damage seals.

Cable Connections

There are eight instrument lines (two sets of four) connecting the electronics in the trailer with the detector used at the tower. Four of these go to the "near-pod" for monitoring waste shipments being transported by truck. The other four are identical, but go to the "far-pod" for monitoring shipments arriving by rail. These cables are numbered 1-4 for "near-pod" connectors, and 5-8 for "far-pod" connections. Their use is summarized in the following table:

Cable No.	*Description*
1 & 5	Pre-amplifier connector: connects the 9-pin pre-amp supply on back of amplifier in trailer to the 9-pin connector on the pre-amp of the detector. (Identical for both detectors.)
2 & 6	High voltage bias supply: From connector J1 on rear of power supply in trailer to H.V. - connector on pre-amplifier of detector (identical for both detectors).
3 & 7	Signal cable: connects to "Sig. Input" on amplifier in trailer and "Sig. Out" on pre-amp of detector (identical for both detectors).

Cable No.	*Description*
4 & 8	Inhibit cable: for detectors S/N 253 only - attach to connector marked "Inhibit" on detector pre-amp and to the "Gate IN" connector on the front (bottom row - right side of the multi-channel analyzer). The white toggle switch directly above this connector should be in the "Anti" position or "down."

Other cable connections inside the trailer remain the same regardless of which detector is used. These are as follows:

Device	*Description*
AST-100 Calculator Interface	Small multi-pin connector on rear connects to Aux. #1105 on rear of MCA.
	Large multi-pin connector connects to slot on rear of Hewlett-Packard calculator.
Princeton Gamma-Tech. Amplifier	BNC connector on rear labeled "93 ohm output-UNI" to "ADC IN" connector on front of MCA. Switch above MCA connector to "ADC-IN" or "up" position.
Hewlett-Packard Calculator	A power cable and a signal cable on rear of unit to connect one to the other. Cannot be connected wrong.

Instrument Control Settings

The settings described utilize the 4096 memory storage units of the multichannel analyzer (MCA) as two separate units of 2048 channels each. This was done so that two different spectra could be stored at one time in case two waste packages had to be counted one after the other. Another arbitrary choice was to adjust the gain of the system so that each memory unit corresponds to about 1 keV. Therefore, at 1 keV/ch, the range of energies that can be detected in each half of the memory ranges between 0 and 2.05 MeV. This can be changed at the discretion of the operator in order to increase resolution or to include higher photon energies.

Multichannel Analyzer (MCA) Settings.

An instruction manual for the Canberra Model 8100 MCA is on file in the trailer. This manual describes the functions of all control knobs and switches for the general operation of this instrument. The specific settings important for the operation of the MCA as a waste monitor are as follows:

1) Power Switch - "On."

2) Control Buttons - Use as needed.

3) Scope Control Knobs - Adjust as needed to view display.

4) Preset Count - "Off."

5) Preset Time - Set by operator: 1st digit (N) = sec, 2nd digit (M) = power of 10. (A setting of 32 is equal to 300 sec.) 00 will continue to count until stopped by operator. Switch to "Live."

6) Function - PHA.

7) I/O Device - Two positions used: TTY Out (normally used) and TTY In (used in test program).

8) Data - Add.

9) I/O Cycle - Manual.

10) Memory Control - Set knob to 1/2 to store data in first 2048 channels of memory.
    - Set knob to 2/2 to store data in second half of memory.
    - Push down on both toggle switches at once to clear memory.

11) Collect Button - When lit, data will be collected and stored in memory. Will go off when preset time is reached.

12) I/O Button - Is called for in program to transfer data from MCA to calculator or vice versa.

13) ADC Dead Time % (Meter) - Will indicate percent of time instrument is not counting. Problems in pulse pile-up began to occur when radiation field caused dead time above 50%.

14) Amp - Not used. External amplifier is used.

15) SCA - LLD - Lower Level Discriminator 0.05
    ULD - Upper Level Discriminator 10.0

16) ADC - Conv. Gain - 2048 (normally)
    - Baseline - 4.56

17) Digital Offset - All switches "down" except "ADC In," which should be "up."

Mech-Tronics (NIM)

Power supply for modules. Power switch "On."

AST-100 Interface

All switches "Up."

Power Design H.V. Power Source

- Polarity and voltage dependent on detector used. Polarity light on front indicates which polarity is being output.
- For detector No. 252 with stainless steel Dewar: Polarity +, Voltage 2,000.
- For detector No. 253 - White Dewar: Polarity -, Voltage 1,400.
- H.V. switch "Up."
- Adjust knob no faster than 100 V/sec.
- Turn off overnight and whenever there is danger of loss of liquid N^2 before returning to fill Dewar.
- Polarity changed by turning off, removing unit from NIM, and turning slotted screw on rear of chassis.

Princeton Gamma Tech. - Amplifier

- Detector No. 252

  Coarse Gain - 100

  Fine Gain - 100

  Input Polarity - Negative

- Detector No. 253

  Coarse Gain - To be determined.

  Fine Gain - To be determined.

  Input Polarity - Positive

- Shaping 0 4 μSec
- Polarity Output - Polarity "Positive"
  Range - 10V, Restorer - Low

Hewlett-Packard

- Calculator "On."
- Printer "On."

Operation of Hoist

Three locking switches (up, down, and stop) control the movement of the crossarm which carries the detector. The switch marked "Up" engages the drive motor and will cause the crossarm to raise. It will continue to rise until the upper limit switch is reached or the "Stop" switch is depressed. Similarly, the "Down" switch will cause the crossarm to lower until the bottom limit switch is engaged* or the "Stop" switch is depressed. There is one set of these control switches at the base of the tower and one set inside the trailer for remotely operating the tower.

A height indicating device which is driven by the cable has a digital readout at the tower base and in the trailer. This device may be set to read height above the ground or height above a waste package.

Care must be observed when changing the height of the detector so that the instrument cables do not become entangled in the hoist mechanism or fouled on the tower structure.

Filling the Dewar with Liquid Nitrogen

The detector's Dewar flask hold ∿15 L of liquid nitrogen - a sufficient supply to last from 5 to 7 days depending on the outside temperature. The liquid nitrogen level should be checked before turning on the bias voltage if there is any doubt that it may have evaporated since the last filling. For this reason, a check sheet of filling dates is posted on the wall inside the trailer. The list gives the last date the Dewar was filled.

Liquid nitrogen is brought into the burial ground in a 30 liter transfer Dewar equipped with a special type of top housing. A reserve supply of nitrogen is kept in this large storage flask. This reserve flask is filled at any convenient source by the following procedure:

1) Follow all safety rules (gloves, face shield, and cover on arms and legs) before filling.
2) Before removing the spring hooks holding the top plug on the flask, vent any internal pressure by means of the small petcock valve.

*Lower limit switch does not always stop drive if water has condensed inside the box. Test its operation by manually depressing the lever before depending on the switch to stop the drive.

3) When vented, remove the spring latches and remove the rubber stopper.

4) Insert nozzle from supply into transfer flask and fill 1/2 to 2/3 full.

5) Re-install rubber stopper, spring clamps and close all vent valves.

The flask is now ready to be taken to the tower.

To fill the Dewar, which is inside one of the pods in the tower, first lower the tower to its lowest position and proceed as follows:

1) Unlatch and remove top cover of pad.

2) Position safety platform ladder beside the pad and, using gloves, lift off the vent plug on top of the Dewar.

3) Carry the large flask to platform and insert the nitrogen hose into the detector's Dewar.

4) Open transfer valve to fill.

5) If pressure has not built up in the large flask sufficiently to produce a flow of nitrogen, then pressurize the flask by the use of a hand pump until a good flow occurs.

6) When full, shut valve on transfer flask and remove nozzle.

7) Carry flask down ladder and set on pad.

8) Replace the vent on the Dewar and the top on the pad.

9) Raise detector to its appropriate height.

10) The reserve in the transfer flask can be replenished as needed.

Using Programs on Cassette Tape

Because of the limited memory capacity of the Hewlett-Packard calculator plus the repetitive nature of the waste monitoring operation, several basic operations have been programmed as an aid to the operator. These programs can be called by use of "Special Function Keys" once the tape cassette has been loaded into memory. Other programs may be called from the tape by file number. All programs are in BASIC. Instruction manuals for operating the HP-9830A are on file in the trailer. The tape marked "E. I. duPont-Savannah River Laboratory" is the cassette tape discussed below.

Loading Cassette Tape into Calculator

The above burial ground sotfware tape is inserted into the calculator with the "name-side" toward the operation as follows:

1) Open cassette door in calculator by pulling forward on door button (uper right-hand side of keyboard).
2) Insert cassette tape and close door by pushing door inward until latched.
3) Turn calculator "On" and printer "On."
4) Press "Load" then "Execute."
5) When tape stops, press "CONT, Execute."
6) The calculator will display "Select Special Function Key, Execute."

A program function is now ready to be selected and read into the memory of the calculator. The following upper case (UC) and lower case (LC) functions are available:

Function Key	*Case*	*Name*	*Function*
f_0	UC	Scan-ID	Searches spectrum for peaks, prints peak channel, energy, FWHM, intensity, and error. SCAN-ID will search an internal library to match peak energy with isotope name. If found, it will list isotope and curies.
	LC	Scan	
f_1	UC	Bkgd.	Permits the storage of any photopeaks of the background spectra that corresponds to photopeaks in the library.
	LC	Analyzer	Same as SCAN-ID but includes extra options. This program is normally used routinely.
f_2	UC	FWHM	Using ^{60}Co spectra, this program calculates the full width at half maximum for the two ^{60}Co photopeaks.
	LC	EN Cal.	Stores current energy calibrations of the MCA into memory (keV/ch). Used daily.
f_3	UC	Eff. 2 Cal.	Calculates and stores the absolute photon efficiency of each detector relative to a known standard source. It must be calibrated at a source to detector distance of 50 ft. Since we have 2 detectors, the efficiency of each may be stored on tape.
	LC	Eff. 1 Cal.	

Function key	*Case*	*Name*	*Function*
f_4	UC	Not used.	Permits the operator to assign specific channels to use in calculating peak intensity. (Used when two peaks are close together.)
	LC	Design Ch.	
f_5	UC	Not used.	Permits the printout of spectrum store in memory of the MCA.
	LC	MCA Print	
f_6	UC	TP-MCA	Allows the operator to read a spectrum recorded on tape into the memory of the MCA.
	LC	MCA-TP	Transfers data from the MCA and permits storage on tape.
f_7	UC	Test In	Reads a signal from the calculator into MCA and displays pattern on MCA scope.
	LC	Test Out	Prints out channels of MCA.
f_8	UC	Lib. Clear	Clears the isotope library (not used unless a new library is to be typed into file).
	LC	Lib. Update	Changes, adds or deletes an isotope in the library.
F_9	-	-	Not used.

There is one additional program on the cassette tape that is used to calculate curies of any known isotope that has been identified by its photopeak during execution of the analysis program. This program is excessed by the command "LOAD 25, Execute."

Operating the Program

All of the programs have brief instructions that inform the operator what to do next. They all assume that the necessary data has been collected on the MCA. With a minimum of training, most of these programs can be utilized as needed. The programs most frequently used are:

1) f_2 (LC) - Energy Calibrate - This program is used once or twice daily to correct for drifts in amplifier gain. It requires the collection of a spectrum with two photopeaks of known energy, preferably sidely separated in energy. The 344.3 and 1408.0 keV photopeaks of the ^{252}Eu calibration

source* are convenient. The resultant data is stored on tape and is used to translate channel number into energy (keV/ch).

2) F_1 (UN) - Background - At the location of the monitor in the burial ground there is a significant ^{137}Cs photopeak in the background. When the background program is run, the background of this ^{137}Cs peak is stored in the isotope library and is subtracted out of the results of waste package analysis when the curies of activity is being calculated. (It is not subtracted from the peak intensity calculations performed in the first half of the analyze or scan program.) This program should be run whenever a new detector height of ^{137}Cs background changes.

3) f_1 (LC) - Analyze - After collecting a suitable spectrum of the gamma ray emission from a waste package, the "analyze" program will identify photopeaks and calculate their area. If a calibrated detector is used, the second half of this program will search the library of 27 photopeaks, identify the radionuclide (if in the library) and calculate curies of that radionuclide (without shielding correction applied). It is the program most routinely used by the operator.

4) f_3 (both UC and LC) Eff. 1 or Eff. 2 Calibrate - This program calculates and stores an equation which expresses detector (1 or 2) efficiency as a function of energy. It must be run at a source-to-detector distance of 50 ft.

   A source of known strength must be used. At least two photopeaks must exist (more if possible) in the collected spectrum. A 2 Ci ^{252}Eu source (2 Ci as of June 1977) is provided for this purpose. The efficiency calibration should remain relatively constant for each detector. It should be rerun whenever repairs or modifications to the detectors are made.

Using the ^{252}Eu Calibration Source

The ^{252}Eu source had an original strength of 2 Ci (June 7, 1977). Its half life is 13.2 years. At 6 in. above the source the dose rate is $\sim$5 R/hr. For this reason the source has been sealed in an aluminum capsule attached to the end of a 10-ft-long aluminum rod. The dose rate to the operator holding the opposite end of the rod is $\sim$10 mR/hr.

When not in use, the source is placed in a pipe housing that is buried horizontally into the earthen mound at the H-Area end of the trailer. The pipe is closed with a cap and the area roped off and tagged to notify the casual visitor that the pipe contains a radioactive source. The radiation field at the end of the closed pipe is <2 mR/hr.

*See section on "Using the ^{252}Eu Calibration Source."

Energy Calibration (Special key f_2, LC)

The ^{252}Eu source is a convenient way to provide known energies to calibrate the MCA. Remove the cap on the pipe and remove the source rod by slipping it out of the pipe. Lay the source rod across the top of the earthen mound near the source storage area and collect a 5 min spectrum.

Replace the source rod back into the pipe and close the end cap.

On the MCA, intensify a region around the peak corresponding to 344.3 keV photopeak and do the same for the 1408.0 keV peak. Run the "Energy Calibration" program. When complete, the data will be used to define the energy of peaks at any channel.

Efficiency Calibration

The efficiency of a germanium detector with respect to photon energy should not change rapidly, if at all. The efficiency will be different for each detector, however. There are reserved files on the tape to store two efficiencies (special function key f_3). Efficiency 1 is the efficiency data for detector No. 252 and Efficiency 2 will contain efficiency data for detector No. 253. To perform an efficiency calibration proceed as follows:

1) Set detector height at 50 ft above the point where the ^{252}Eu source is to be placed.
2) Remove source from storage pipe and place in position. (Note: If source is positioned in roadway, rope off road.)
3) Collect a spectrum - 300 to 600 sec. (Make sure "Present Time Switch" is on "Live.")
4) Insert program tape and load Eff. 1 or 2, depending on which detector is being calibrated.
5) Use the 344.3, 778.9, 964.0, and 1408 keV peaks for input.
6) Type input data as called for in the program. Date must be entered in sequence stated.
7) The input requirements for the ^{252}Eu source are:
   a) Source age in years - type decay time since June 1977.
   b) Peak energy (keV) - type lowest energy (344.3)*.

*See Table 13.

c) Gamma/sec of this energy - type 1.813 E10.

d) Half-life of source in years - type 13.2.

Repeat for next higher energy, starting at a) After each entry, depress the "Execute" key. Continue until all input requirements are satisfied (if the four peaks under 5 above are used, you must type in all information--a) through d)--for each peak. For the ^{252}Eu source, the decay time a) and half-life will be the same for all input sets.

8) The ^{252}Eu source emits a series of about 13 prominent photopeaks from 121.8 to 1408 keV. These peaks, their yield and photons/sec are listed in Table 13. (Peaks below 244 keV should not be used for calibrating detector efficiency since the equation used to calculate the detector response is in error in this region.)

9) When the program is completed, it will print out the new equation that will be used to calculate the detector efficiency vs. energy. Example: Eff. = Exp. [-12.4 + (-0.983×log (Energy)] The coefficients of this equation are derived from the input data and the spectrum collected.

TABLE 13

Photopeaks of ^{252}Eu Calibration Source, Calibrated on June 7, 1977
(Source Intensity when new = 2.0 Ci, $T_{\frac{1}{2}}$ = 13.2 years)

Peak Energy (keV)	*Yield*	*Gammas/sec When Calibrated*
121.8	.254	1.8796×10^{10}
244.7 (**)	.068	5.032×10^{9}
344.3* (**)	.245	1.813×10^{10}
411.0	.020	1.480×10^{9}
444.0	.029	2.146×10^{9}
778.9 (**)	.120	8.880×10^{9}
867.3	.038	2.812×10^{9}
964.0 (**)	.132	9.768×10^{9}
1085.8	.097	7.178×10^{9}
1112.0 (**)	.124	9.176×10^{9}
1212.9	.013	9.620×10^{8}
1299.2 (**)	.016	1.184×10^{9}
1408.0* (**)	.198	1.465×10^{10}

*Peaks used for energy calibration of the detector.
(**)Peaks used for efficiency calibration of detector at 50 ft above the source.

Height Measuring Device

A counter driven by the cable which raises the detector carriage was designed to indicate 1.0 count per foot of height. There is one counter at the base of the tower and a tracking counter inside the trailer for remote height indication. Zero height can be set to indicate detector distance above ground level, lower limit of cable drive, or source-to-detector height. Because different waste shipments vary greatly in height, it is more convenient to set the device to indicate the distance between the ground and the gamma detector. Therefore, if a package is 6 ft above the road bed, the detector can be set at χ+6 ft to yield a source-to-detector height of χ feet.

Figure 5 shows the color code and wiring diagram for this device. To adjust the indicator, leave the power on and:

1) Remove the 12 screws on the railroad side of the box at the tower base.
2) Remove cover and turn the cable driven shaft to the desired setting.
3) Replace cover and screws.
4) Check the counter inside the trailer to make sure it is in agreement with the setting at the lower base.

NOTE: There is a possibility that the drive pulley may slip on the cable. Tape on the cable at 10-ft intervals can be used to indicate slippage. At the bottom limit of the crossarm, the height indicated should be 4 ft.

Radiation Background Information

Prior to constructing the GWM at its present site, a background survey was made showing radiation intensity of 30 μR/hr. A lower background of 8 μR/hr was found at a site outside the burial ground on the east end toward H Area. However, because of the lack of power and roads, this lower background site was abandoned in favor of the present location of the experimental facility.

Once the monitor was constructed, a spectrum of the background was taken. Table 14 lists the isotopes, photopeak energies, and peak counts obtained with the detector 36 ft above the roadbed. The count lasted 50 min. With the exception of a large photopeak due to ^{137}Cs and two minor peaks due to ^{60}Co, all other radionuclides appear to be of natural origin.

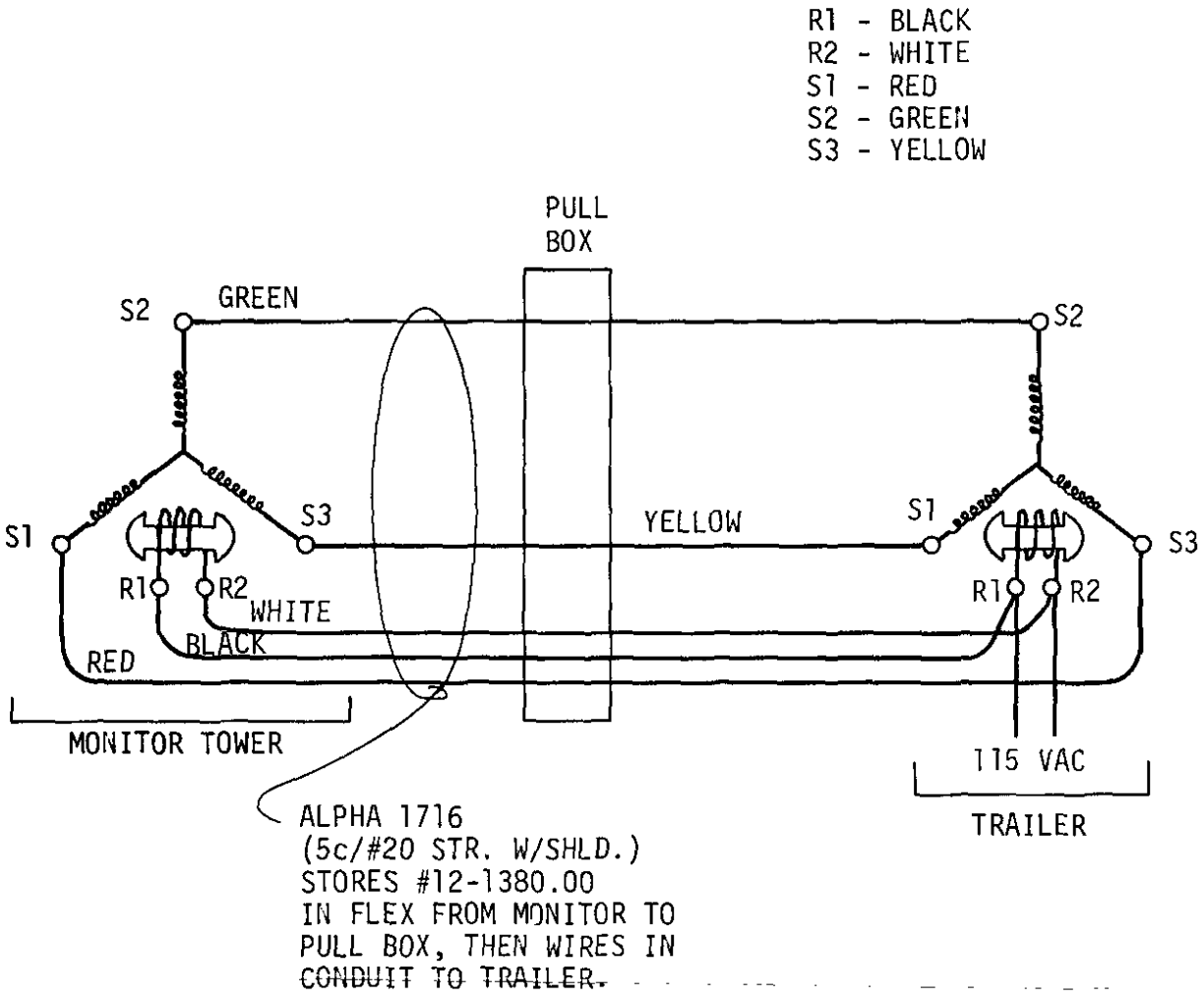


FIGURE 5. Schematic of Height Measuring Device

TABLE 14

Major Photopeaks in the Background Spectrum
(Detector Height - 36 ft; Lenght of Count - 50 min)

Isotope Identification	*Photopeak Energy (keV)*	*Peak Counts/ 50 min*
^{212}Pb	238.84	725
^{214}Pb	351.98	468
^{208}Tl	583.27	281
^{214}Bi	609.51	462
^{137}Cs	661.66	3,719
^{223}Ac	911.21	263
^{214}Bi	1,120.59	92
^{60}Co	1,173.30	79
^{60}Co	1,332.85	97
^{214}Bi	1,765.94	98

The computer program permits the operator to store a background count in memory. This background will automatically be subtracted when the second half of the "analyze" program is run. It is not subtracted in the first half of the program where peak energies and peak counts are printed. This procedure works well as long as the background does not change radically and if the waste package is lightly shielded. Under these conditions 1 c/sec in the ^{137}Cs peak will represent an error of about 2 mCi at a height of 30 ft above the package. However, if the package has a thickness of 12 cm lead (like a 100-Area scrap metal cask), 1 c/sec detected by the monitor is equivalent to about 600 Ci inside the package after correcting for the absorption of a 12-cm thick lead wall. Most packages containing ^{137}Cs are not heavily shielded and most of the ^{137}Cs detected when monitoring scrap casks is on the outer surfaces or in the drip pan of the cask. Therefore, counting errors of ^{137}Cs in the background is troublesome, but presents no great problem for the types of waste monitored during this test.

Table 15 shows the variation in ^{137}Cs background observed at two detector heights during four months. It varies around 1.25 c/sec at 36 ft and about 1.1 c/sec at 56 ft above the roadbed. Therefore, a new background for ^{137}Cs should be stored in memory whenever the detector height is changed significantly.

The radioactive background is slightly lower when the ground is wet. This is probably due to the additional absorption of gamma rays by soil moisture. However, the variation with wetness is only slight. Routine measurement of the background is useful to detect low levels of contamination which might occur during measurement of a waste shipment.

Brief Summary of Operating Procedure

To prepare the monitor for measurement of waste packages a daily startup routine is required. Briefly, the following procedure should be followed:

1) Check liquid nitrogen content of Dewar - fill if low. (Wait 5 hrs after filling before applying bias voltage if Dewar was found to be empty.)
2) Raise detector to the height to be used for monitoring waste shipments.
3) Set bias voltage for detector used: +2,000 volts (Steel Dewar), -1,400 volts (White Dewar).
4) Load program tape cassette into calculator.
   a) Run "Energy Calibrate" by placing ^{252}Eu source on top of mound and collecting a spectrum for 300 sec.
   b) Intensify the region around the 344.3 + 1408 keV photopeaks.
   c) Follow instruction of the program.

5) Put ^{252}Eu source back into its storage pipe.

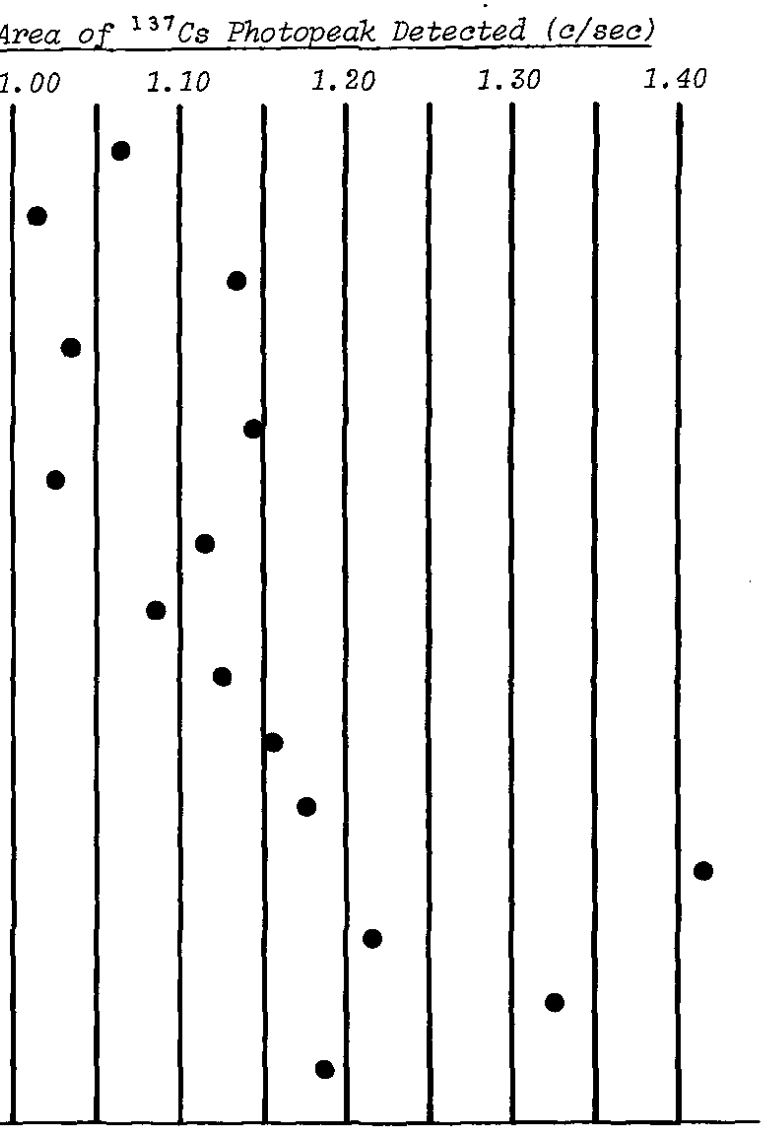
6) Collect a 600 sec background spectrum and intensify any peaks in the isotope library (usually only ^{137}Cs at 661.6 keV).

7) System is ready to analyze waste packages.

TABLE 15

Variation of ^{137}Cs Count in Background

Date	*Detector Height (ft)*
1/12/78	56
1/18/78	56
1/19/78	50
1/20/78	56
1/23/78	36
1/24/78	56
1/31/78	56
2/07/78	56
2/10/78	56
2/27/78	56 (Dry)
3/23/78	36 (Dry)
4/03/78	36 (Dry)
4/10/78	36 (Dry)
4/11/78	36 (Dry)
4/14/78	36 (Wet)



Program Cassette Files

A list of current files on the cassette tape may be obtained by rewinding the tape and typing the command "TLIST, Execute." Table 16 shows current files on the tape and identifies each file.

Copies of program files may be printed by first giving the command "Load 2, Execute" (where 2 is an example of a file number). Then, "LIST, Execute." Only program files can be printed out. Data files require a program to print out the file.

TABLE 16

Data Table of Program Tape

File No.	*Type File*	*Size (Words)*	*Used (Words)*	*Description*
0	Program	600	496	Boot strap - Loads File 1
1	Key	250	119	Identifies Key Functions
2	Program	2,500	2,263	Library Clear & Library Update
3	Data	600	600	Isotope Library
4*	Program	6,000	5,671	Analyze Program
5	Data	10	8	-
6	Not Used	10	0	-
7	Data	10	4	Efficiency No. 1
8	Data	10	4	Efficiency No. 2
9	Not Used	10	0	-
10	Program	2,000	1,574	Energy Calibrate
11	Program	2,000	1,715	Efficiency Calibrate
12	Program	2,000	1,635	FWHM for ^{60}Co Photopeaks
13	Program	2,000	1,406	MCA Print
14	Program	1,200	1,199	MCA to Tape
15	Program	1,200	1,038	Tape to MCA
16	Program	1,200	760	Test Input to MCA
~~17~~	~~Program~~	~~1,200~~	~~115~~	~~Used to Print Eff. Data~~
18	Not Used	20	0	-
19	Data	4,100	4,096	Store Spectrum on Tape
20	Not Used	20	0	-
21	Data	4,100	2,048	Store Spectrum
22	Not Used	20	0	-
23	Not Used	4,100	0	-
24	Not Used	20	0	-
25	Program	4,100	108	Calculate Ci from Raw Data
26	Not Used	20	0	-
27	Not Used	20	0	-

*A short program to calculate the attenuation of the lid of 100-Area scrap casks was attached to the end of the analyze program. When File 4 is in memory, it can be called by "Continue 8820, Execute." See Section for further details.