

RECENT DEVELOPMENTS IN WASTE CHARACTERIZATION AT CHALK RIVER NUCLEAR LABORATORIES

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Abstract

The waste characterization program (WCP) at Chalk River Nuclear Laboratories (CRNL) was initiated in 1982 to determine the physical, chemical and radiological properties of wastes intended for disposal in IRUS (Intrusion Resistant Underground Structure), a belowground vault to be constructed at CRNL.

During the last year (1988), work on the WCP has centred on determining the radionuclide inventories in candidate wastes for IRUS by gamma-ray monitoring and destructive radiochemical analysis. For the gamma-ray monitoring work, a monitoring system was developed that is based upon an IBM-PC computer equipped with a multichannel analyser (MCA) card. This system has been used to determine the radionuclide inventories in a variety of packages (trash bags and bales, drums, pails, etc.) from a number of different sources. The radiochemical analysis part of the WCP has mainly consisted in developing the radioanalytical methods to determine the radionuclides listed in the US regulatory document 10CFR61 (^{90}Sr , ^{63}Ni , ^{99}Tc , ^{129}I , etc.). Radiochemical analysis has been used to characterize only incinerator ash to date.

This paper will present the technical problems associated with monitoring various waste forms (geometry considerations, shielding problems, operating environment, etc.) and will also present details of the destructive radiochemical analysis program at CRNL.

INTRODUCTION

From 1946 to the present, Chalk River Nuclear Laboratories (CRNL) has stored radioactive waste that it received from both on-site and off-site (universities, hospitals, etc.) producers. Information on the chemical and radionuclide inventory of these wastes meet existing storage regulations but will most likely fail to meet impending disposal criteria. In 1982, in support of its new waste disposal program, CRNL initiated a waste characterization program (WCP) to determine the physical, chemical and radiological properties of wastes. The near-term objective for the WCP is to develop the technologies and establish the procedures for characterizing candidate wastes destined for disposal in IRUS (Intrusion Resistant Underground Structure), a belowground disposal vault that is to be constructed at CRNL (1). The past distribution (1983-87) of IRUS candidate wastes, sorted according to volume and generator package type (IRUS will have standard package types), is shown in Fig. 1, which presents the data on concrete disposable (CD) wastes, and Fig. 2, which presents the data on non CD wastes.

Initially, the WCP consisted of a monitoring program and a radiochemical analysis program. The goal of the monitoring program was to develop a non-destructive gamma-monitor that was capable of determining the inventory of gamma-emitting radionuclides in wastes. The goals of the radiochemical analysis program were to develop the radioanalytical methods for the determination of the non gamma-emitting radionuclides, such as those listed in 10CFR61 (^{90}Sr , ^{99}Tc , ^{129}I , etc.), provide verification of the monitoring data, and provide the correlation factors between gamma-emitting and non gamma-emitting radionuclides to enable the monitor to estimate the inventories of non gamma-emitting radionuclides.

Initially, the two programs were under two separate groups at CRNL: one group responsible for the monitoring program and the second group for the radiochemical

analysis program. The monitor group developed a prototype monitor based on a multichannel analyser (MCA) interfaced to a DEC PDP micro-11 computer which controlled the data collection and processing and sent the processed data to the database on the CRNL mainframe computer. Unfortunately, this system proved difficult to automate for use in a process environment by an unskilled

OVERVIEW OF WIP INFORMATION: 1983 - 1987

VOLUMES OF WASTES IN AREA B (CD) - INCLUDES REMARKS DATA

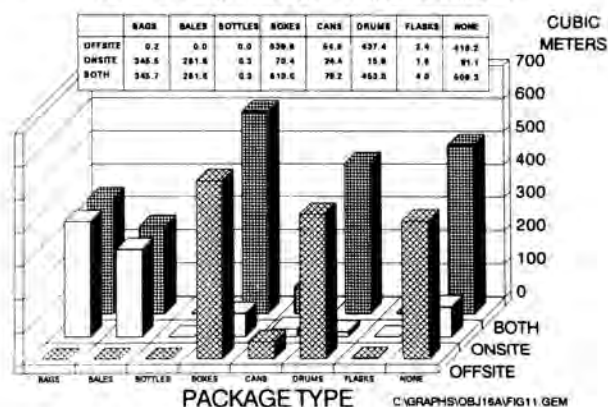


Fig. 1. Volumes of CD Wastes Sorted According to Package Type.

OVERVIEW OF WIP INFORMATION: 1983 - 1987

VOLUMES OF WASTES IN AREA B (NON CD) - INCLUDES REMARKS DATA

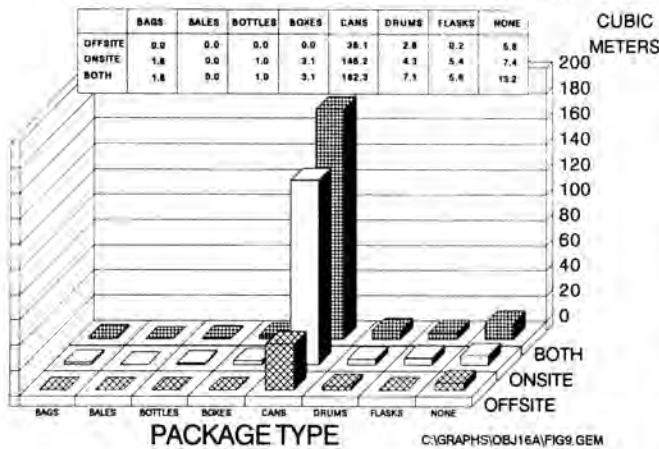


Fig. 2. Volumes of Non CD Wastes Sorted According to Package Type.

operator, and work on the automation of this system terminated in 1987. However, the system continues to monitor waste but under the control of a skilled technician.

In 1988, the two programs were placed under one group responsible for the monitoring and radiochemical characterization work. Under this combined program, a new monitor based upon a PC-AT computer using an MCA card was assembled along with gamma-spectrometry hardware (HPGe gamma-spectrometer and associated electronics). A commercial peak search and nuclide-identification software package was purchased from the supplier of the MCA card and a menu-driven control program was written to permit an unskilled operator to control the monitor. Work on the radiochemical characterization program was essentially complete except for actinide elements, and a detailed sampling and analysis of monitored ash drums has commenced.

This paper describes the technical problems associated with the development of a monitor to determine the inventory of gamma-emitting radionuclides in the diverse waste forms supplied by generators. The paper also describes the radiochemical analysis program and presents some of the preliminary data resulting from the sampling and analysis

TABLE I

Ratio of Composite Bale Inventory to Total Bag Inventory for Various Nuclides in 13 Bales as Determined by PDP-11 Based Prototype Monitor

RADIONUCLIDE	CE-144	CS-137	CS-134	ZR-95	CO-60
ENERGY keV	133.5 80.12	661.60	795.80 604.70	756.87 724.24	1173.20 1332.50
BALE #					
1	0.37	1.53	1.13	3.24	2.27
2		1.78			1.78
3	0.56	1.49			1.62
4		1.62		1.21	2.53
5		2.05		5.06	1.65
6		1.12		1.58	1.26
7		0.95	1.21	1.19	1.64
8	1.20	4.81	4.10	1.36	2.16
9		1.68		1.24	1.17
10		0.96		1.75	1.94
11		1.00		1.54	2.91
12	0.91	1.30	1.13	3.10	1.98
13		1.45	1.73	1.09	1.38
MEAN	0.76	1.67	1.86	2.03	1.87
STD. DEV.	0.32	0.96	1.14	1.19	0.48

of ash drums that have been assayed with the PC-based MCA system.

MONITORING PROGRAM

The first PDP-based prototype monitor, mentioned above, included a Nuclear Data model 65 MCA, an HPGe gamma-spectrometer, two NaI(Tl) gamma-detectors, associated electronics and shielding (2). In 1988, the system was used by a technician experienced in gamma-spectrometry to determine the radionuclide inventory in non-incinerable wastes processed at the CRNL waste treatment centre. To date, this system has monitored approximately 700 bags of non-incinerable waste and the 18 composite bales produced by compacting these bags. The comparison of the radionuclide inventory of 13 of the monitored bales and the inventory obtained by summing the individual inventories of the monitored bags is presented in Table I. The agreement between the two inventories is usually within a factor of two or better and this agreement can be improved upon by correcting for attenuation within the waste and providing a more controlled geometry for the bags which had a tendency to slump.

In 1987, a number of suppliers of gamma-spectrometry hardware and software released IBM-PC based MCA cards which took advantage of the ease of programming, increased memory and low cost of the PC computers. Of the four MCA cards available at the time, only the ORTEC system met all the requirements of having the card and software controlled by an external batch file under the control of a menu-driven program that was customized for the monitor. A detailed description of the hardware and software of the monitor follows.

The monitor hardware consists of a CANBERRA HPGe (13% relative efficiency, resolution of 1.9 keV FWHM at 1332.5 keV) horizontally mounted gamma-spectrometer with pre-amplifier, Canberra spectroscopy amplifier, Canberra bias supply and NIM bin. The computer is an IBM PC-AT clone with an ORTEC "ACE" MCA card with 4096 channels and an analog to digital converter (ADC) with 25 microsecond fixed dead time. The peak search and nuclide-identification software package was GELIGAM and was purchased from EG&G ORTEC. The control program and menu program were written at CRNL to permit simple and user-friendly operation of the monitor.

After assembling and testing the hardware and software, the detector was calibrated for efficiency with standard radionuclide sources purchased from Amersham and from the National Bureau of Standards. As the wastes were expected to vary considerably in radionuclide content, the detector was calibrated for efficiency at fixed positions of 0.5 m, 1.5 m, 3.0 m, 4.6 m, 6.1 m, 7.6 m, 9.1 m, 10.7 m and 12.2 m. This large dynamic range enabled the monitor to determine the radionuclide inventory in unshielded wastes containing up to 3.7×10^9 Bq of ^{60}Co or its equivalent with minimal dead time losses and short counting periods (typically 600-1000 s). Because the system was intended to be reasonably portable, the lead shielding for the detector consisted of an open-ended rectangular

array of lead bricks placed around the detector. The detector was positioned in the shield such that the shielding provided the detector with a $\pi/16$ steradian solid-angle view of the waste form.

In order to simulate field operating conditions, an old building located on one of the waste management storage sites was selected for monitoring the incoming wastes. The building did not have any climate control so the system operated at ambient temperature throughout the summer and fall months. Wastes from trucks were brought to the monitoring building via crane or forklift. A manually operated forklift was used to transport the wastes to the desired distance from the monitor as indicated by markings on the floor. Under these simple monitoring conditions, approximately 20 items could be monitored per day with most of the time spent transporting the waste to and from the monitoring site.

During the above sessions, the monitor was controlled by the menu-driven program which offered the operator a choice of performing an energy calibration of the card, collecting a background spectrum, collecting a sample spectrum, analyzing the collected spectra or exiting the program. Once all the items had been monitored, the sample spectra were corrected for the background contribution, a peak search and nuclide-identification analysis of the background-corrected spectra were performed, and a report of the absolute radionuclide inventories, decay corrected to the appropriate sample date, were saved on disk or printed out.

To date, the monitor has been used to determine the radionuclide inventories of approximately 200 items received from an external radioisotope producer who sells radionuclides to the international community. In addition, the monitor has also been used to assay the radionuclide content of drums of ash from the CRNL low-level waste incinerator. The data for the external generator is presented in Table II (sorted according to product stream identified by the producer), and the data for the ash drum monitoring is presented in Table III. The former data are essentially the first that CRNL has acquired on the radionuclide content of wastes in storage from the radioisotope generator.

For the external generator, in many cases the identification of the major radionuclide contaminant and the estimate of its inventory was seriously in error due to the decay of this nuclide resulting from the storage time before shipment. In general, the long-lived radionuclides of interest for storage and disposal conditions were not estimated or even identified in the shipping form, although they were observed in the monitored wastes. The wastes varied considerably in radionuclide content indicating little or no segregation of waste according to product stream. Until required, external waste producers have little incentive to segregate or characterize their wastes, thus putting the onus on the waste receiver to characterize the waste at his own expense. High-activity wastes (10^{11} Bq) containing short-lived radionuclides (eg ^{131}I , ^{99}Mo) could not be monitored under the existing conditions (manual handling, limited shielding), and these

TABLE II
Range of Inventories of Major Radionuclides Determined
by the Gamma Monitor in Four Waste Streams from a Radioisotope Producer

Radionuclide	Product Waste Stream Inventory (Bq)			
	¹⁴ C Product	¹³¹ I Product	⁶⁰ Co Product	⁹⁹ Mo Product
¹³⁴ Cs	7x10 ⁴ -3x10 ⁶	1x10 ⁴ -1x10 ⁵	---	---
¹³⁷ Cs	1x10 ⁴ -3x10 ⁶	1x10 ⁵ -4x10 ⁸	1x10 ⁴ -1x10 ⁵	---
¹³¹ I	1x10 ⁴ -2x10 ⁵	1x10 ⁵ -4x10 ⁸	---	---
¹²³ Te	1x10 ⁴ -8x10 ⁴	4x10 ⁴ -9x10 ⁷	---	---
¹²⁹ Te	---	1x10 ⁵ -1x10 ⁶	---	---
⁹⁹ Mo	1x10 ⁴ -1x10 ⁶	1x10 ⁶ -2x10 ⁷	---	8x10 ⁵ -2x10 ⁷
⁵⁷ Co	6x10 ³ -5x10 ⁵	---	1x10 ⁴ -5x10 ⁴	---
⁶⁰ Co	2x10 ⁵ -7x10 ⁵	7x10 ⁴ -3x10 ⁵	8x10 ⁵ -1x10 ⁷	1x10 ⁴ -5x10 ⁵
¹⁰⁶ Ru/Rh	2x10 ³ -3x10 ⁶	1x10 ⁵ -2x10 ⁶	---	---
²⁰¹ Tl	---	---	---	1x10 ⁴ -6x10 ⁷
⁹⁵ Zr	2x10 ⁵ -4x10 ⁵	1x10 ⁵ -2x10 ⁶	2x10 ⁵ -2x10 ⁷	1x10 ⁴ -9x10 ⁵

TABLE III
Range of Inventories of Radionuclides
Determined in Incinerator Ash Drums
by the Gamma Monitor

Radionuclide	Inventory Range (Bq)
¹³⁷ Cs	3.0x10 ⁶ - 5.8x10 ⁸
¹³⁴ Cs	3.0x10 ⁶ - 5.5x10 ⁷
⁶⁰ Co	2.0x10 ⁶ - 2.5x10 ⁸
¹⁴⁴ Ce	2.0x10 ⁷ - 3.8x10 ⁷
¹¹⁰ Ag	2.0x10 ⁶ - 8.4x10 ⁷
¹²⁵ Sb	5.3x10 ⁶ - 3.2x10 ⁷
¹⁵⁴ Eu	2.0x10 ⁶ - 9.2x10 ⁶
⁶⁵ Zn	5.0x10 ⁶ - 1.8x10 ⁸
⁹⁵ Zr	4.5x10 ⁶ - 5.6x10 ⁷
¹⁰⁶ Ru/Rh	2.5x10 ⁷ - 1.4x10 ⁸

wastes went directly to storage. It is planned to retrieve some of these high-activity wastes after adequate decay of the principle short-lived radionuclide to determine the inventory of the long-lived contaminants.

The ability of the monitor to do this characterization in the original waste yields an uncertainty of a factor of 2-

3 in the inventories of gamma-emitting radionuclides. For the present time, this is more than adequate for sorting wastes and estimating waste inventories. The monitor has functioned without any serious failures under field operating conditions, under less than ideal conditions and at extreme ranges of temperatures. The MCA card has operated without failure and the commercial software available for the MCA card is extensive, well documented, easy to learn, and analyses complex spectra quickly and accurately. In only six months of operation with this new system, the monitor has been tested, calibrated, commissioned for use in the field, a menu program written to permit routine operation by an unskilled operator, and over 300 items monitored for radionuclide content. The data for these items has been processed and added to the CRNL waste management database. Work is continuing on automating the monitor to permit automatic transfer of the data to the database through a modem, to interface it to a remote label reader for sample identification, and to automate sample positioning via a remote-control crane or automatic conveyor-belt system.

RADIOCHEMICAL ANALYSIS PROGRAM

In 1982 the radiochemical analysis program was started to determine the long-lived non gamma-emitting radionuclides such as those listed in the US regulatory document 10CFR61. The radionuclides of interest were ⁹⁰Sr, ⁹⁹Tc, ⁶³Ni, ¹²⁹I, ¹⁴C, ³H, ⁹³Zr and Pu. For the incinerator ash, radiochemical methods have been developed for the first four radionuclides. ¹⁴C and ³H are not expected to be present in the ash, and methods are still being developed for the last two radioelements.

The objectives of the radiochemical analysis program were to 1) develop the radioanalytical methods capable of determining these radionuclides in the various waste forms destined for disposal, and 2) provide the correlation factors between the gamma- and non gamma-emitting radionuclides to enable the monitor to estimate the non gamma-emitting radionuclides in wastes.

Radioactive ash from the CRNL low-level radioactive waste incinerator was selected as the first waste form for characterization by the radiochemical analysis program. This waste was selected for the following reasons: 1) to compare the inventories determined by the monitor of the waste bags before incineration with that determined by radiochemical analysis in the ash product, 2) the ash contained low levels of radionuclides and could be easily mixed and sampled for analysis, and 3) the radiochemical methods developed for ash could easily be adapted for analysis of other wastes.

The radiochemical methods for the analysis of incinerator ash were developed over a period of three years. To obtain the ash samples, the 0.2 m³ ash drums weighing approximately 150 kg, were mounted in a drum handler and rotated end over end 50-75 times. The drum lid was then replaced with a sampling lid containing 5 polyethylene (1000 mL) sample bottles. The drum was rotated three times and the bottles containing approximately 75 g of ash were removed, labelled and transferred to the laboratory for analysis. A total of 14 ash drums that had been monitored by the gamma-monitor were sampled according to this procedure, providing a total of 70 ash samples for analysis.

A flow sheet for the radiochemical analysis of the ash is presented in Fig 3. Each 75 g ash sample is ground to 200-400 mesh in a SPEX shatter box equipped with a tungsten carbide grinding head. Three 5 g sub-samples from each 75 g sample are weighed out into 60 mL polyethylene bottles, and the homogeneity of the sample checked by determining the inventory of gamma-emitting radionuclides on a gamma-spectrometer calibrated for the ash geometry. If the sub-samples indicate the 75 g sample is homogeneous, a 0.5 g sample is taken for destructive radiochemical analysis. The results from the gamma-spectrum analysis are used to check the monitor data as well as determine the correlation factors between the gamma- and non gamma-emitting radionuclides in the ash.

The 0.5 g ash sample is placed in a teflon bomb, ⁸⁵Sr and ⁹⁹Tcm tracers and 20 mg of iodine carrier (as iodide) are added and evaporated with the ash. Approximately 2 g KHF₂ flux are added to the spiked ash, and the sample fused at 250C for 30 minutes to break down acid insoluble silicates. The fused sample is then transferred to a platinum crucible and fused at 900C for 10 minutes to break down refractory oxides and minimize losses of volatile radionuclides. The flux is dissolved in dilute HCl, and the insoluble fluoride precipitate is centrifuged off. The precipitate is washed and the washing transferred

with the supernate and diluted to 50.0 mL in a volumetric flask. The precipitate is dissolved in saturated boric acid containing concentrated HNO₃, and transferred quantitatively to a 10.0 mL volumetric flask. Details of the radiochemical separations are presented below.

In the case of ⁹⁰Sr, 20 mg each of inactive strontium and iron carriers are added to a 2.0 mL aliquot of the boric acid solution, the solution made basic with NH₄OH and the insoluble hydroxides centrifuged off and discarded. Ammonium oxalate solution is added to the supernate to precipitate strontium oxalate which is centrifuged off, washed and redissolved in HNO₃. Strontium nitrate is then precipitated by the addition of 90% HNO₃, the precipitate washed with 90% HNO₃, and then dissolved in water. The pH of the strontium nitrate solution is adjusted to near neutral, diluted to 10.0 mL with citric acid and the strontium yield determined from the ⁸⁵Sr activity. The strontium is then adsorbed on a cation exchange column, the column washed with citric acid to remove yttrium and the rare earths and the time noted as time zero for ⁹⁰Y growth. The column is set aside for 15 days after which the ⁹⁰Y is eluted quantitatively with citric acid and its activity determined by LSC. The ⁹⁰Sr activity is determined from the 90Y activity after correcting for the efficiency of the LSC, the growth factor for ⁹⁰Y from time zero, the yield from the ⁸⁵Sr activity, and normalized to the sample weight. The precision of the method is typically 8% (95% confidence limits) at the 10 Bq.g⁻¹ level in ash.

The method for ¹²⁹I has been reported previously (3); however, for completeness it is briefly summarized below. A 5.0 mL aliquot from the acid soluble fluorides is made basic with NaOH and any insoluble hydroxides centrifuged off. The iodide carrier is oxidized to periodate by addition of NaOCl and heating. The solution is then made acidic with HNO₃, and periodate is then reduced to iodide by the addition of Na₂SO₃. These steps ensure equilibration of the carrier with the ¹²⁹I in the sample. The sample is then passed through a cation exchange column, the column is washed and the eluents combined. The iodide is then oxidized to iodine by addition of KNO₂ and the iodine extracted into CCl₄. The CCl₄ is washed and the iodine adsorbed onto activated charcoal which is filtered, washed and air dried before neutron activation analysis. The sample, along with a known amount of inactive iodine containing a known quantity of ¹²⁹I, is irradiated for a period of 10 minutes in a flux of approximately 1.0x10¹⁴ neutrons per second per square centimeter to activate the ¹²⁷I to ¹²⁸I and ¹²⁹I to ¹³⁰I. The chemical yield of the iodine is determined by comparing the ¹²⁸I activity of the sample to that of the standard, while the ¹²⁹I activity in the sample is determined by comparing the ¹³⁰I activity of the standard and sample, correcting for chemical yield and normalizing to sample weight. The precision of the method is 8% (95% confidence limit) at the 0.1 Bq/g level in ash.

In the case of ⁶³Ni, a 5.0 mL aliquot of the acid soluble fluoride fraction is sampled, diluted to 10 mL and

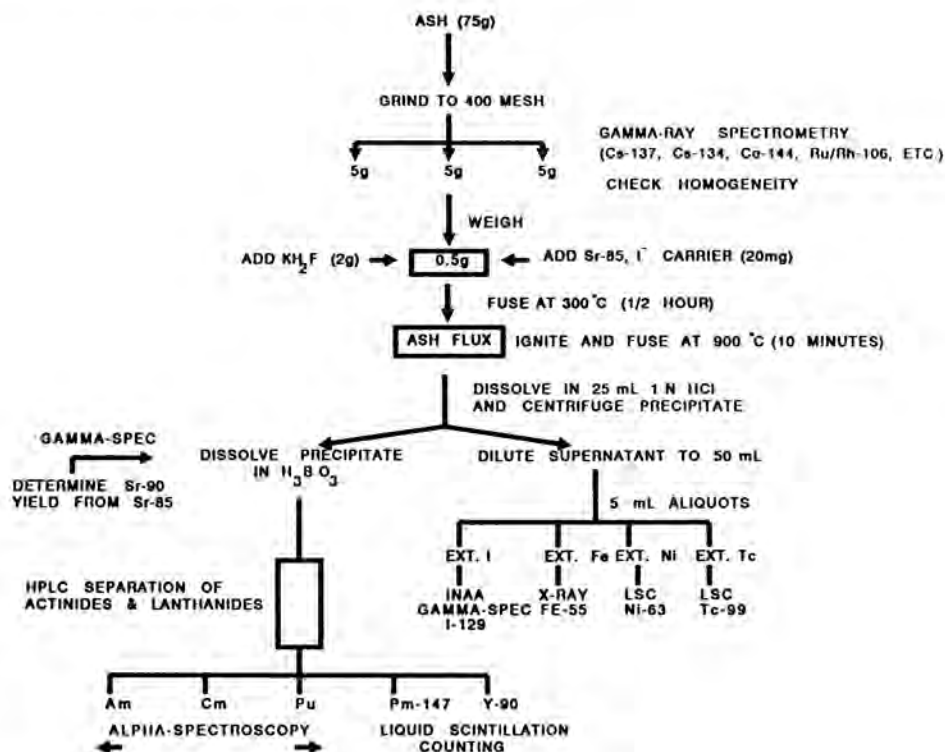


Fig. 3. Flow Sheet for the Radiochemical Analysis of Incinerator Ash.

sodium tartrate added to complex iron. The solution is made basic with NH_4OH , dimethylglyoxime added, and the nickel extracted into toluene. The toluene layer containing the nickel is washed with dilute NH_4OH , the toluene layer removed and mixed with an equal volume of LSC cocktail for the determination of the ^{63}Ni . The chemical yield for the solvent extraction is 98.5%. The ^{63}Ni in the sample is determined from the LSC counting data, corrected for the LSC efficiency and any quenching effects, corrected for yield, aliquot size, and normalized to sample weight. The precision of the method is 8% (95% confidence limits) at the 0.5 Bq.g^{-1} level for a 2000 second counting period. The decontamination factors for most radionuclides, including ^{60}Co , is greater than 10^6 .

For ^{99}Tc , a 5.0 mL aliquot of the acid soluble flux is sampled, ferric chloride added, and the solution made basic with NH_4OH to precipitate the insoluble hydroxides. The solution is centrifuged, the supernate removed and made acidic by the addition of HNO_3 . The solution is passed through a cation exchange column, the column washed, the eluents combined and made basic with NH_4OH . The Tc is separated by a repeated extraction with cyclohexanone, the cyclohexanone fractions combined, washed and evaporated to dryness in a LSC vial. The technetium is redissolved in water and the chemical yield of the technetium determined from the $^{99}\text{Tc}^m$ activity determined by gamma-spectrometry. The

sample is permitted to decay to eliminate the $^{99}\text{Tc}^m$ activity, after which the ^{99}Tc in the sample is determined by LSC. The ^{99}Tc in the sample is determined after correcting the LSC data for the contribution from the $^{99}\text{Tc}^m$ tracer, corrected for chemical yield and normalized to sample weight. The precision of the method is 8% (95% confidence limits) at the 0.5 Bq/g level in ash for a 2000 second counting period.

The above methods have been tested on inactive ash samples spiked with known levels of the above radionuclides. The 70 ash samples generated from the sampling of the 14 monitored ash drums will be destructively analysed once they have been ground and tested for homogeneity. To date, 20 of the 70 ash samples (representing 4 of the 14 drums) have been ground and analysed for the gamma-emitting radionuclides. The comparison of the inventories determined in these samples with those from the monitor is presented in Table IV. In general, the agreement between the two data sets is within a factor of two or better, indicating the monitor can determine the inventory of gamma-emitting radionuclides with reasonable accuracy. Once the destructive analysis of the ash has been completed, if correlation factors for the non gamma-emitting radionuclides prove to be constant for a particular waste stream, then the monitor should be able to be used to determine the total radionuclide inventory in

TABLE IV
Comparison of Radionuclide Inventory ($\text{Bq}\cdot\text{g}^{-1}$) in Ash Drums as Determined from Ash Sampling and Monitoring of Four Ash Drums

Radionuclide	Drum Inventory ($\text{Bq}\cdot\text{g}^{-1}$)				Avg. Ratio Ash Sample:Monitor
	Drum 63	Drum 123	Drum 156	Drum 180	
^{137}Cs Ash Sample ^{137}Cs Monitor	1.26×10^4 6.2×10^3	1.33×10^3 3.2×10^2	1.63×10^3 8.5×10^3	1.45×10^3 4.2×10^2	2.9 ± 1.1
^{134}Cs Ash Sample ^{134}Cs Monitor	1.31×10^3 8.7×10^2	1.61×10^2 $\leq 7.0\times 10^1$	1.42×10^3 9.8×10^2	1.22×10^2 5.0×10^1	1.8 ± 0.6
^{60}Co Ash Sample ^{60}Co Monitor	1.31×10^2 8.6×10^1	6.36×10^3 2.7×10^3	1.34×10^2 1.1×10^2	2.30×10^3 1.6×10^3	1.6 ± 0.5

these waste streams with reasonable precision and accuracy.

SUMMARY AND CONCLUSIONS

An automated gamma-monitor system using a PC-AT compatible computer with MCA card, gamma-spectrometry hardware and software, has been developed to determine the inventory of gamma-emitting radionuclides in wastes from various sources (including external generators) and in various geometries. Comparison of the inventories determined by the monitor with those from sampling and analysis, as well as data from monitoring individual items used to prepare composite packages, indicates that the monitor can reliably determine the inventory of gamma-emitting radionuclides in various waste packages to within a factor of two or better. The monitor has functioned for approximately 6 months under field operating conditions without a failure and work is continuing to automate the system for uploading processed data to a database as well as interfacing to a label reader and sample handler.

Radiochemical methods have been developed to determine ^{90}Sr , ^{129}I , ^{99}Tc and ^{63}Ni in radioactive incinerator ash. Ash drums monitored with the above system have been sampled and inventories of the above radionuclides from destructive ash analysis will be used to determine the correlation factors between the gamma-emitting and non gamma-emitting nuclides in this waste. If these correlation factors prove constant for a particular waste stream, then the gamma-monitor can be used to ac

curately determine the total inventory of radionuclides in these wastes.

Data from monitoring external generator wastes indicate that wastes from initially separated waste streams are combined to reduce the cost of shipment. Although this practice is economical under the present regulations, future costs for waste characterization by the waste receiver will force the generator to either segregate his wastes according to its "hazardous" lifetime or do his own characterization. Once a generator has determined the appropriate scaling factors for non gamma-emitting radionuclides, a gamma-monitor similar to the one described above could be used to determine inventories prior to shipment. Clearly, segregation combined with an adequate monitoring program that is verified with periodical sampling and analysis should provide the most efficient and cost effective way to characterize waste prior to shipment and disposal. The initial cost of setting up this program will quickly be offset by savings in reduced shipping and receiver characterization and handling charges.

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