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7. Abstract <p>This report documents ion exchange column sampling and Non Destructive Assay (NDA) results from activities in 1994, for the Plutonium Recycle Test Reactor (PRTR) ion exchange vault. The objective was to obtain sufficient information to prepare disposal documentation for the ion exchange columns found in the PRTR Ion exchange vault. This activity also allowed for the monitoring of the liquid level in the lower vault.</p> <p>The sampling activity contained five separate activities: (1) Sampling an ion exchange column and analyzing the ion exchange media for purpose of waste disposal; (2) Gamma and neutron NDA testing on ion exchange columns located in the upper vault; (3) Lower vault liquid level measurement; (4) Radiological survey of the upper vault; and (5) Secure the vault pending waste disposal.</p>		
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EXECUTIVE SUMMARY

The 309 Building (including the PRTR facility) has recently been declared surplus by the U. S. Department of Energy Nuclear Energy DOE-NE.) The 309 Building is tentatively scheduled for transfer to DOE-EM (Environmental Management) during FY 1996 for eventual decontamination and decommissioning. Facility transition planning includes removal of any remaining wastes and stabilization of nonsmearable contamination, including the PRTR Ion Exchange (IX) and Resin Disposal Vaults. The vaults were opened March 1994 to perform an initial investigation, the Westinghouse Hanford Company (WHC) recommendation was to dispose of the 12 ion exchange columns in the PRTR vault. During PRTR operation the vault housed four cleanup system ion exchangers. The ion exchange columns have the potential to contain corrosion products, fission products and small quantities of PuO_2 fuel.

Disposal requires a regulatory category determination and waste characterization. For this purpose one ion exchange column was sampled and assayed on July 19, 1994. The characterization approach used is balanced with ALARA considerations. The column selected for sampling was the primary coolant loop cleanup system column (IX-1). This column has the greatest potential to have corrosion and fission products and also transuranics (TRU), and minimizes the worker radiological exposure.

Radiation surveys in the upper vault show the general radiation field in the upper vault is 45 mR/hr, with a maximum contact dose of 1.5 R/hr on the IX column located in the south east vault corner. Maximum readings on other columns range from 35 mR/hr to 100 mR/hr. Smear samples in the upper vault were all below.

Laboratory analysis of digested primary coolant loop ion exchange resin shows the presence of: ^{137}Cs and ^{60}Co in the $6 \times 10^{-3} \mu\text{Ci/g}$ range, Total β content in the $1.7 \times 10^{-2} \mu\text{Ci/g}$ range, and Total α content in the $7.4 \times 10^{-5} \mu\text{Ci/g}$ range. Concentrations of CERCLA metals of concern are below regulatory limits, based upon a 40 CFR 261 Appendix II "totals analysis.

The IX-1 column Non-Destructive Assay (NDA) shows 31 mCi of ^{60}Co and 2.4 mCi of ^{137}Cs . Neutron counting was performed in an attempt to quantify transuranic materials, but was not successful. Laboratory sample analysis showed Pu and Am concentrations at below the laboratory detection limit, implying insufficient material to detect during NDA.

showed Pu and Am concentrations at below the laboratory detection limit, implying insufficient material to detect during NDA.

NDA of all upper vault columns was also attempted using a lead wrapped sodium iodide detector. The survey was not completed, due to detector saturation by the high background radiation.

No change was observed in the water level since the March 1994 measurement. March 3 and July 19 measurements, (a 131-day time span) were both 10 ft 3 inches. During that time 8.4 cm (3.3 in.) of rain was received in the 300 Area.

ACRONYMS

ASME	American Society of Mechanical Engineers
B&PV	Boiler and Pressure Vessel
BIX-5	Chemical Shim Removal Ion Exchanger
CAM	Continuous Air Monitor
D ₂ O	Heavy Water
DOE-EM	U. S. Department of Energy, Environmental Management
DOE-NE	U. S. Department of Energy, Nuclear Energy
HPGe	High Purity Germanium
IX	Ion Exchange
IX-1	Primary Coolant Cleanup Ion Exchanger
IX-2	Reflector Loop Cleanup Ion Exchanger
IX-3	Moderator Coolant Cleanup Ion Exchanger
IX-4	Top and Bottom Shield Cooling Cleanup Ion Exchanger
LEL	Lower explosive limit
NaI(Tl)	Sodium Iodide
NDA	Non-Destructive Assay
NDA	Non-Destructive Assay
PFP	Plutonium Finishing Plant
PRTR	Plutonium Recycle Test Reactor
T.O.C.	Total Organic Carbon
TCLP	Toxicity Characteristic Leaching Procedure
TRU	Transuranics
WHC	Westinghouse Hanford Company

PRTR ION EXCHANGE VAULT

COLUMN SAMPLING

1. INTRODUCTION

1.1 OBJECTIVE

This report documents ion exchange column sampling and Non-Destructive Assay (NDA) results for the Plutonium Recycle Test Reactor (PRTR) ion exchange vault. The objective of the sample was to obtain sufficient information to prepare disposal documentation for the ion exchange columns found in the PRTR ion exchange vault. Figure 1 depicts the vault configuration during the sampling activity. The NDA information with the analytic laboratory results will help to identify the contents of other ion exchange columns in the PRTR vault. This activity also allowed for the monitoring of the liquid level in the lower vault.

1.2 WORK SCOPE

The sampling activity contained five separate activities. (1) Obtain a sample from an ion exchange column and analyze the ion exchange media for purpose of waste disposal. (2) Perform gamma and neutron NDA testing on ion exchange columns located in the upper vault. (3) Obtain an additional lower vault liquid level measurement. (4) Perform a through upper vault radiological survey. (5) Secure the vault pending waste disposal.

1.3 BACKGROUND

The 309 Building (including the PRTR facility) has recently been declared surplus by the U. S. Department of Energy DOE-NE (Nuclear Energy.) The 309 Building is tentatively scheduled for transfer to DOE-EM (Environmental Management) during FY 1996 for eventual decontamination and decommissioning. Facility transition planning includes removal of any remaining wastes and stabilization of nonsmearable contamination, including the PRTR IX and Resin Disposal Vaults (reference WHC-SD-SP-SPP-001).

The PRTR Ion Exchange (IX) Main Vault and the PRTR Resin Disposal Vault (located below and accessed from the main vault) were opened March 1994 to perform an initial investigation on the vault stability and conditions. The WHC recommendation from the investigation was to remove the vault contents (reference WHC-SD-SP-ER-002.)

To dispose of the 12 ion exchange columns in the PRTR vault requires a regulatory category determination and waste characterization. The ion exchange columns have the potential to contain corrosion products, fission products and small quantities of fuel. During PRTR operation the vault housed

four cleanup system ion exchangers, (IX-1, primary coolant cleanup system; IX-2, reflector loop cleanup system; IX-3, moderator coolant cleanup system; the fourth position used initially as IX-4, top and bottom cooling cleanup system and after 1966 as BIX-5, boron removal exchanger.)

The characterization approach for the ion exchange media is balanced with ALARA considerations. The approach being, a laboratory characterization of one column and NDA analysis, and extrapolation of the results to other columns. The column selected for sampling was the primary coolant loop cleanup column in the IX-1 position. This IX-1 column has the greatest potential to have corrosion and fission products and also transuranics (TRU). (The initial investigation reported Co_{60} and Cs_{137} in the IX-1 column.) The column was also accessible with the initial vault investigation configuration. In addition, the potential radiological exposure met ALARA consideration. Obtaining the sample requires removal of a 5 cm. (2 in.) diameter seal weld on the fill plug. The maximum contact dose on the sampled column was expected to be less than 100 mRem/hr. Other columns have measured doses as high as 2.5 R/hr.

1.3.1 Ion Exchange Column Information.

Until its layaway in 1969, the PRTR ran with a variety of UO_2 - PuO_2 fuel combinations (levels of PuO_2 up to 4 wt%) with a heavy water (D_2O) moderator. Deionization of the heavy water was performed by top-loading IX columns, located in the main IX vault. After 1966, PRTR operation included the use of a chemical shim (boron) to control PRTR reactivity. A side loading column (BIX-5) was added to the PRTR main vault for boron concentration control. The BIX-5 column was installed in the position occupied by the IX-4 column, which was moved to inside the containment dome.

Cleanup system ion exchangers in the vault are of three designs, (see Figures 2 and 3.) The columns were built to American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code Section VIII. The IX columns are approximately 2.75 m. (9 ft.) tall (including support legs and entry and exit piping) and are either 45.7 or 76.2 cm (18 or 30 in.) in diameter. The outer shell on the large column is stainless steel, aluminum on the small columns. The columns contain both acidic cation resin and basic anion resin. They may also contain a variety of contaminants because of reactor operations. These contaminants may include TRUs, lithium hydroxide, cesium, cobalt-60, boron, tritium and fission products. Once an IX column was exhausted, the entrance and exit pipe were cut and the column placed into the lower disposal vault, for storage. In 1986 the IX column piping to the containment vessel was drained of heavy water, cut inside containment, and closed off.

The boron removal IX column (BIX-5) is approximately 86.4 cm. (34 in.) in diameter and 2.75 m. (9 ft.) tall (see Figure 4.) The column is built to ASME B&PV Section III, Class C. The outer shell is stainless steel. The boron removal column contains an anion resin. The column was designed for moderator system boron concentration control. The anticipated boron concentration was 0- to 30-ppm (ref Demmitt). In 1986 the IX column piping to the containment vessel was drained of heavy water, cut inside containment, and closed off.

Figure 2. Large Diameter Ion Exchange Column

Figure 3. Small Diameter Ion Exchange Column

Figure 4. Chemical Shim Ion Exchanger

1.3.2 Sampling Task Description

Sampling and NDA of the IX-1 column was performed on July 19, 1994. Work was controlled by Job Control System document number 3U-94-00507/0. The ion exchange sampling and NDA were performed using people from the Mobile Sampling Laboratory, PFP Analytical Laboratory, and 300 Area Health Physics. The sampling activity was conducted in four steps. First, gaining access to the IX-1 column for sampling; Second, sampling the column; Third, performing NDA; and Fourth, securing the sampled column and ion exchange vault.

Entering the column required the following. Removing the A and B level vault shield blocks, and erecting the confinement tent. Establishing confinement tent HEPA filtered exhaust air flow with continuous air monitoring (CAM). The vault was then opened by removing one C level cover block. Once the vault was open radiological and air quality surveys and a lower vault liquid level measurement were made. The access plug to lower vault was replaced, and a ladder placed into the upper vault. Upon entering the upper vault a detailed radiological survey was performed. Rigging was attached to the IX-1 column lifting point. Riggers worked from the C level shield blocks.

Fitters entered the upper vault, wearing respiratory protection. They cut the IX-1 column entrance and exit pipe with a three-wheel cutter (cutting took about five minutes.) Lower explosive limit (LEL) and total organic carbon (T.O.C.) measurements were made near the IX column, neither explosive gases or elevated total organic carbon concentration levels were found. The column was raised and rotated allowing for pipe plug installation on both column entrances and containment pipe stubs. The column was raised from the upper vault to the C level cover blocks to a height allowing Fitter access to fill plug, keeping the radiological dose ALARA. The column seal weld was ground out with an abrasive grinding wheel (grinding took about 20 minutes and used three abrasive grinding wheels.) Fitter work took about 1.5 hrs.

The column sample was taken by the Mobile Sampling Laboratory, using sampling procedure S4-037.001. An forty-eight inch long Arts Manufacturing Soil Auger Sampler¹ (catalog item no. 910) was used. The sampler obtained a core representing the top half of the resin column. The sample was transferred from the auger sampler into a sample container for storage and transport. The sampling activity took about one half hour. The sample was sent to 222-S Labs in a lead lined can for analysis, and a sample chain of custody control was used.

Personnel from the PFP Analytical Laboratory performed the column NDA. The steps used and set up were documented in an NDA log book. The NDA consisted of a segmented gamma scan and neutron counting of the column at various elevations. A crane was used to incremental raise and rotate the column to achieve uniform characterization. Following completion of the NDA, the column was placed back into the upper vault. The NDA took about 3 hrs.

¹ Arts Manufacturing & Supply, Inc. 105 Harrison, American Falls, ID. Phone 1-800-635-7330.

The final step was to secure the vault. The sampler was double bagged and placed in upper vault for later reuse. All plastic and wastes were bagged and other equipment (ladders) removed from upper vault. Replacing the C level cover block closed the upper vault. RCT surveyed the equipment and tent, and all equipment was given a release (ladders, rigging, and tent.) The CAM sample was sent to the laboratory for analysis. Approximately 1/2 bag (two cu ft) of radioactive waste (step off pad and materials from lower vault) was taken to 340 building for disposal with their wastes. The confinement tent was taken down, and all cover blocks surveyed and replaced.

1.3.3 Non-Destructive Assay Description

Characterization of the PRTR primary loop (IX-1) column identified and quantified potential radionuclides in the IX column using portable NDA techniques. Potential contaminants on the resin bed were corrosion products, fission products and plutonium. High resolution gamma spectroscopy was selected for both identifying and quantifying the radionuclides. In addition two other non-destructive assay methods were attempted. One method used gamma ray analysis with a sodium iodide [NaI(Tl)] detector. The passive neutron analysis for plutonium content was attempted using two slab neutron detectors each with 9 ³He tubes.

To analyze the ion exchange column, the high resolution gamma spectroscopy system was calibrated using characterized radionuclide sources in an assay mock up at the Plutonium Finishing Plant (PFP). Established methods for calculating the detector response to source activity at several energies were used. Because the IX-1 column was a unique assay situation, a method used to mathematically model the source activity within the column was developed for the detector calibration. The method used is similar to a 55-gallon drum segmented gamma-ray scan assay. Computer modeling was used to estimate the corrections for gamma-ray attenuation by the resin matrix and the column shell. The activity of each identified gamma-ray emitting radionuclide was calculated based on the calibration model. (Procedure details are located in Appendix C.)

2. RESULTS

2.1 RADIATION SURVEY

Radiation surveys in the upper vault show the following: (A copy of the radiation survey report is located in appendix A.)

- 1- The general radiation field in the upper vault is 45 mR/hr, with a maximum contact dose of 1500 MR/hr on the IX column located in the south east vault corner. Maximum readings on other columns are as follows: BIX-5, 35 mR/hr; IX-1, 100 mR/hr; IX-3; and IX-2, 80 mR/hr.
- 2- There was no smearable contamination found in the upper vault.
- 3- Equipment used for NDA and sampling was released. During release elevated counts of 500 dpm were found on a ladder rung. However, the ladder was decontaminated and released.

2.2 SAMPLE ANALYSIS

Laboratory analysis of the IX-1 resin were performed at the 222-S Analytical Laboratory. The laboratory identification number for the sample is R-5804. The laboratory analysis was performed on a digested resin, such that resin analysis is a CFR 261 Appendix II "totals analysis." The laboratory analysis includes metals' analysis, gross beta, gross alpha, gamma energy analysis, and alpha energy analysis. Concentrations of CERCLA metals of concern are below regulatory limits, when a 20 to 1 toxicity characteristic leaching procedure (TCLP) sample dilution is factored into the results. Appendix B. contains analysis summary information and laboratory data.

2.3 NON-DESTRUCTIVE ASSAY

Characterization of the PRTR primary loop column (IX-1), included portable non-destructive gamma-ray analysis using high purity germanium detector. It was used to identify and quantify the gamma-emitting radionuclides. Analysis of the PRTR primary loop column (IX-1) included calibration of a high purity germanium (HPGe) detector for a distributed source, computer modeling of the gamma-ray attenuation correction factor, final calculation of the total radionuclide activity within the column, and a comparison of the calculated activity with Westinghouse Hanford Health Physics measured dose rate.

In addition to the PRTR Primary Loop Column (IX-1) gamma-ray analysis two other non-destructive assay methods were attempted. One method used gamma ray analysis with a sodium iodide [NaI(Tl)] detector and the other method used passive neutron analysis. However, problems in the field prohibited the use of these techniques.

- 1- The primary isotopes detected in the PRTR primary loop column (IX-1) were ^{137}Cs and ^{60}Co . The total ^{60}Co activity in the IX column is 31 mCi with a range of 9.8 to 72 mCi, and the total ^{137}Cs activity in the IX column is 2.4 mCi with a range of 0.77 to 5.7 mCi. The activity range includes uncertainty estimates for the distribution of material within the column, for overall gamma-ray attenuation by materials within the column, for the mathematical models used, and random counting uncertainties. The high value for each isotope was calculated using a calculation model, which also provides calibration method verification. A detailed description of the calibration and analysis methods is given in Appendix C.
- 2- The NaI(Tl) data were to be used to compare the results from the PRTR Primary Loop column to other IX columns in the PRTR upper storage vault. However, the columns were not able to be assayed because of high count rates, which caused excessive electronics dead time in the detection system. The primary effect in a spectrum of this dead time is the loss of counts in gamma-ray peaks. Because of the loss of counts, identification of separate peaks energies was not possible with the NaI(Tl) detector. Therefore, NaI(Tl) analysis was not completed.
- 3- The passive neutron analysis for plutonium content was attempted using two slab neutron detectors each with 9 ^3He tubes. During analysis electronic problems with the pre-amplifier were encountered. Because the detectors could not be used to measure a known standard, it is not known whether the detectors were functioning properly during the IX column analysis. Even after the detectors were returned to the laboratory, the electronic problems persisted. Because the neutron detector could not be verified as operating properly, no neutron results can be reported for IX column plutonium content. (Note also that laboratory sample analysis showed Pu and Am concentrations below the detection limits, implying there was insufficient material to detect.)

2.4 WATER LEVEL

No water level change was observed since the March 1994 measurement. March 3 and July 19 measurements, (a 131-day time span) were both 10 ft 3 inches. Both measurements were taken with by the same person using the same equipment. The measurement supports the earlier conclusion that the vault structure is intact and stable, and that the water leak rates from the vault are very small or nonexistent. The 300 area weather station recorded 8.4 cm (3.3 in.) rain during this time.

3. REFERENCES

C. E. Graves, "309 Building Transition Plan," WHC-SD-SP-SPP-001 Revision 0, Westinghouse Hanford Company, Richland, Wa. August 31, 1994.

C. E. Graves, "Stability of the Plutonium Recycle Test Reactor Main In Exchange and Resin Disposal Vaults," WHC-SD-SP-ER-002 Revision 0, Westinghouse Hanford Company, Richland, Wa. April 5, 1994.

T.F. Demmitt et al, "Reactor Operation and Instrumentation and Parameter Measurements" Transaction of the American Nuclear Society Fifteenth Annual Meeting, June 15-19 1969 pg 307.

40 CFR 261 Appendix II, "Method 1311 Toxicity Characteristic Leaching Procedure (TCLP)," U.S. Environmental Protection Agency, Washington, D.C.

Appendix A

Radiation Survey Results

INFORMATION ONLY

Westinghouse Hanford Company		Date 4-19-94	Time From 0400 To 1700	Survey Number Nº 179136	F.C. R	Page 1 of 4												
		Bldg. 309	Area 300	Room Ion Exchange Vault														
RADIATION SURVEY REPORT		Description of Job <i>Entry into ion exchange vault to sample column # IX-1. To include detaching, raising, sampling and WDA sampling/injection counting. W/HC-IP-0718, 3.1.1, 4.2, 4.4</i>		RWP No. Q-011	Location Ion Exchange Vault													
Check if appropriate. When checked, do not place unrelated information on this record. <table border="0"> <tr> <td>☐ Personnel Contamination</td> <td>☐ High Radiation Level Work</td> </tr> <tr> <td>☐ CAM/Radiation Alarm</td> <td>☒ "Special Survey"</td> </tr> <tr> <td>☐ Establish Dose Rates</td> <td>☐ Property Release</td> </tr> <tr> <td>☐ Radiation/Contamination Incident</td> <td>☐ RAM Shipment</td> </tr> </table>							☐ Personnel Contamination	☐ High Radiation Level Work	☐ CAM/Radiation Alarm	☒ "Special Survey"	☐ Establish Dose Rates	☐ Property Release	☐ Radiation/Contamination Incident	☐ RAM Shipment				
☐ Personnel Contamination	☐ High Radiation Level Work																	
☐ CAM/Radiation Alarm	☒ "Special Survey"																	
☐ Establish Dose Rates	☐ Property Release																	
☐ Radiation/Contamination Incident	☐ RAM Shipment																	
Item No.	P E R (1)	Description of Work Performed, Radiation Controls, and Measurements	Meter Deflection		Dist.	C.F.	DOSE RATE			CONTAMINATION LEVELS								
			W/O	W/C			beta (non pen) mrad/hr	gamma (pen) mR/hr	neutron mrem/hr	Direct (dpm)		Smear 100 cm ²						
1		Column BIX-5 top	25	25	C	/		25			beta	alpha	beta (d/m)	alpha (d/m)	mrad/hr			
2		Column BIX-5 middle	25	25	C	/		35										
3		Column BIX-5 bottom	25	25	C	/		35										
4	✓	Column IX-1 pipping ^{Top} / _{Bottom}	40	40	C	/		40					1000	420				
5		Column IX-1 top	70	70	C	/		70										
6		Column IX-1 middle	100	100	C	/		100										
7		Column IX-1 bottom	90	90	C	/		90										
8		Column IX-3 top	50	50	C	/		50										
1. Check for personnel dose rate ☒ Continued on supplemental report form.																		
Instrument(s) Used		☒ BBEP		☒ CF=10 G-M/Pencake		☒ CF=7 PAM		☒ CF=10 G-M/P-11		☐		☐		Respiratory Protection Worn				
Serial No.(s)		7007		1226		2373		1930		392						☐ Supplied Air ☒ Filter <i>to detach/ent IX-1 pipping only.</i> ☐ Other _____ ☐ None		
ESTIMATED PERSONNEL DOSE RATES																		
Phase of Work		Based on Measurement(s)		Average Dose Rate		Limit Applying												
TO DETACH/CUT PIPING ON IX-1		# 4		40 mrem/hr		(WBP) S (E)												
TO SAMPLE IX-1 COLUMN		# 19		4 mrem/hr		(WBP) S E												
TO WDA IX-1 COLUMN		# 22		10 mrem/hr		(WBP) S E												
RPT Exposure 40 mrem				Work Location Code W/A				Signed <i>C.R. Meinecke</i> PR No. 95906				<i>HT Miller</i> <i>69681</i>						
Did you increase or reduce RWP requirements for this work? ☒ No ☐ Yes Explain on reverse side.				Did you attend a pre-job meeting for this work? ☐ N/A ☐ No ☒ Yes				Reviewed By				Date						

Appendix A - 2

WHC-SD-SP-ER-003 Rev. 0

INFORMATION ONLY

RADIATION SURVEY REPORT (cont.)		Survey Number N ^o 179136	F.C. R	Page 2 of 6
Further Descriptions, Date, and Comment				
<p>The water level of the lower vault measured the same as survey # 165688 3-3-94, 10' 3". A Beta CAM was running through out the day, no high airborne concentration were observed. Air samples taken prior to, during and at completion of work were ~ 10% of DAC. See page #5 for air sample results. An sample from CAM was taken to PNL analytical lab. Tattim samples were taken and ^{read at} to the 400 HPT office, see results on page #6. Dose rates for sampling vs NDA counting differ because of the position of column when that phase of work was being done. No problems were encountered. Hitter work for cutting/detaching & grinding sample plug took approx. 1 1/2 hours. Sampling of IX-1 took approx. 30 min. NDA counting of IX-1 took approx. 3 hours.</p>				
DIAGRAMS OR SKETCHES				
<p>see attachment, page # ⁰⁴⁻¹⁹⁻⁹⁴ 4</p>
ADDITIONAL REPORTS COMPLETED				
Radiological Problem Report Skin Contamination Survey Personnel Effects Contamination Report Radiation Survey Report		Log No. n/a		
Onsite Radioactive Shipment Offsite Radioactive Shipment Routine Radioactive Shipment Sample Counter Log		Log No. n/a		

BD-8000-010R (11/06/93)

INFORMATION ONLY

Westinghouse
Hanford Company

SUPPLEMENTAL REPORT FORM

page 3 of 6

RADIATION SURVEY REPORT

Date
July 19, 1994

Survey No
N2 179136

FC
R

Item No	P E R (1)	Description of Work Performed, Radiation Controls, and Measurements	Meter Deflection		Dist	CF	DOSE RATE			CONTAMINATION LEVELS				
			W/O	W/C			beta (non pen) mrad/hr	gamma (pen) mR/hr	neutron mrem/hr	Direct (dpm)		Smear 100 cm ²		
										beta	alpha	beta (d/m)	alpha (d/m)	mrad/hr
9		Column IX-3 middle	190	190	C	/		190						
10		Column IX-3 bottom	30	30	C	/		30						
11		Column IX-2 top	70	70	C	/		70						
12		Column IX-2 middle	80	80	C	/		80						
13		Column IX-2 bottom	90	90	C	/		90						
14		5th column 2' from bottom	1500	1500	C	/		1500						
15		max whole body to climb ladder	250	250	full	/		250						
16		General Area - 3' from West tank	45	45	full	/		45						
17		DOSE RATE AT RAIL WITH "C" COVER BLOCK REMOVED	2.5	2.5	full	/		2.5						
18		DOSE RATE AT OPENING ARM EXTENDED	10	10	full	/		10						
19	✓	TO SAMPLE IX-1 COLUMN	4	4	1'	/		4						
20		TO SAMPLE IX-1 COLUMN	20	20	C	/		20						
21		TO NDA IX-1 COLUMN	50	50	C	/		50						
22	✓	TO NDA IX-1 COLUMN	10	10	1'	/		10						
23		TO NDA IX-1 COLUMN	2	2	3'	/		2						
24		SPOT SURVEY OF FLOOR			Smears	/						<1000	<20	
25		SPOT SURVEY OF TANKS			Smears	/						<1000	<20	
26		Smears of "C" cover block bottom			Smears	/						2000	<20	
27		LATTERS BOTH (2)			C/S	/				1000	<20	1000	<20	
28		LATTERS AFTER DECON			C/S	/				<1000	<20	<1000	<20	
29		Smears OF WORK AREA			S	/						<1000	<20	
30		Personnel AT Exit			C	/				<D	<D			
31		Equipment			C/S	/				<5000	<500	<1000	<20	

Reviewed By

Date

Signed
PR No

C.R. Meinecke
C.R. Meinecke 9.5.94

INFORMATION ONLY

Survey, 11/10/66

Upper Vault Access Configuration Depicted

Cross Section Looking Towards North

Looking Towards Containment Dome

Column BIX-5	Loc. 1	25 mrem/hr EC	Loc. 2	35 mrem/hr EC	Loc. 3	36 mrem/hr EC	
Column IX-1	Loc. 4	40 mrem/hr EC of pipe, Loc. 5	70 mrem/hr EC	Loc. 6	100 mrem/hr EC	Loc. 7	80 mrem/hr EC
Column IX-3	Loc. 8	80 mrem/hr EC	Loc. 9	170 mrem/hr EC	Loc. 10	30 mrem/hr EC	
Column IX-2	Loc. 11	70 mrem/hr EC	Loc. 12	80 mrem/hr EC	Loc. 13	80 mrem/hr EC	
5th Column	Loc. 14	max at contact 1500 mrem/hr	Loc. 15	max whole body to use latter 250 mrem/hr field,	Loc. 16	general area 45 mrem/hr	

INFORMATION ONLY

HEALTH PHYSICS AIR SAMPLE COUNTER LOG

Building		370V		Counter Type MS-3		Detector Type PR-210		RD-13		Serial No.		45		Cal Date		9-30-94	
Date		Time		Date		Time		Date		Time		Date		Time		Date	
ON	Off	Flow	Total f1	Sample Location	Counted	MDA	Net DPM	Initial Count	Delay Count	HPT Print and Sign Name	C.E. Meinecke	C.E. Meinecke	C.E. Meinecke	C.E. Meinecke	C.E. Meinecke	C.E. Meinecke	C.E. Meinecke
7/18/94	0910	7/18/94	1400	580	309 Ion	7/21/94 0640	144	30	1.03E-12	7/30E-13	7/17/94 1515	1150	Exchange Valve	309 Ion	7/21/94 0645	144	42
7/17/94	0540	7/17/94	1515	1150	Exchange Valve	7/21/94 0645	144	42	7.30E-13	7/30E-13	7/17/94 1515	1150	Exchange Valve	309 Ion	7/21/94 0645	144	42

309/300 H₂ samplespage 4
Survey # 179136

Protocol #: 2 Name: H-3 DPM 19-Jul-94 13:04
 Region A: LL-UL= 0.0-18.6 Lcr= 0 Bkg= 0.00 %2 Sigma=0.00
 Region B: LL-UL= 2.0-18.6 Lcr= 0 Bkg= 0.00 %2 Sigma=0.00
 Region C: LL-UL= 0.0- 0.0 Lcr= 0 Bkg= 0.00 %2 Sigma=0.00
 Time = 4.00 QIP = tSIE/AEC ES Terminator = Count
 EPT William E. Parys Payroll No 83068
 Conventional DPM
 Nuclide 1 = 188548
 Luminescence Correction On

INFORMATION ONLY

Sample	Loc.	DPM	95% CL	FLAG tSIE
1 BKG		0.00	0.00	B 426.
2 hook #1		43.83	26.11	335.
3 floor		1406.03	91.36	245.
4 Hot tank		51.49	20.46	450.
5 floor		336.17	40.17	403.
6 Tank		162.70	37.70	310.
7 Blank		2.81	11.81	487.

Appendix B

Laboratory Analysis Results

Analysis Results of PRTR Vault IX-1 Resin

ION EXCHANGE RESIN ANALYSIS				CERCLA Metals of Concern (TCLP)		
Determination	Sample 1	Sample 2	Units	Listed Materials	Max Conc. /with Dilution mg/L (uG/G)	% TCLP Totals Conc.
pH	5.27					
pH	5.12					
ICP on Dissolved Resin (TOTALS ANALYSIS)*						
Ag		1.100E-01	uG/G	D011	5.0 / 100*	0.1%
Al	1.730E-00	1.790E 00	uG/G			
B	7.800E-01	8.190E-01	uG/G			
Ba	6.000E-01	6.300E-01	uG/G	D005	1.0 / 20*	0.03%
Ca	2.940E 01	3.020E 01	uG/G			
Cd	2.420E 00	2.500E 00	uG/G	D006	1.0 / 20*	10.0%
Co	1.590E 00	1.670E 00	uG/G			
Cr	1.710E 01	1.890E 01	uG/G	D007	5.0 / 100*	17.1%
Cu	4.410E 00	4.590E 00	uG/G			
Fe	7.190E 01	7.850E 01	uG/G			
K	6.070E 00	8.710E 00	uG/G			
Li	1.480E 00	1.520E 00	uG/G			
Mg	2.300E 01	2.360E 01	uG/G			
Mn	4.690E 01	4.860E 01	uG/G			
Mo	4.870E 00	5.010E 00	uG/G			
Na	2.120E 02	2.180E 02	uG/G			
Ni	3.990E 01	4.190E 01	uG/G			
Pb	8.120E 00	8.170E 00	uG/G	D008	5.0 / 100*	8.1%
S	2.230E 02	2.290E 02	uG/G			
Si	9.770E 00	1.010E 01	uG/G			
Sr	1.500E-01	1.600E-01	uG/G			
V	4.890E 00	5.060E 00	uG/G			
Zn	5.800E-01	5.800E-01	uG/G			
Zr	2.400E-01	2.300E-01	uG/G			
Uranium Analysis						
U	4.380E-06	3.900E-06	G/G			
Radioactive Analysis						
TOTAL BETA	1.710E-02	1.720E-02	uCi/G			
TOTAL ALPHA	7.390E-05	6.700E-05	uCi/G			
Cs-137	6.280E-03	6.380E-03	uCi/G			
Co-60	6.070E-03	6.120E-03	uCi/G			
Pu-239/240	< 5.920E-05	< 5.490E-05	uCi/G			
Am-241	< 6.540E-04	< 2.060E-03	uCi/G			

* For TCLP limit comparisons to ICP concentrations. The TCLP max. conc. is multiplied by 20 (accounting for dilution that occurs during a TCLP test.) Comparison is conservative because all material may not be leached during a TCLP test.

- 1- Concentration test data is based upon a dissolved resin (i.e. total concentration.)
 2- Sample represents a 33 in. vertical core, total depth of resin is approx 55 in.

SAMPLE STATUS REPORT FOR R 5802. 300 AREA STD TIME: 9/15/94 9:24
 DISPATCHED: 7/19/94 12: 4 SAMPLE HAS NOT BEEN SLURPED
 RECEIVED: 7/19/94 14:25

EXT.	DETER.	RESULTS OR STATUS	OUT OF GOOD RANGE? ANS?	
****	*****	*****	***	***
5515	pH	1.00200E 02 % RECOVERY	N	Y
5520	TB	OUT FOR RERUN		
5520	TB	8.99000E 01 % RECOVERY	N	Y
5525	AT	OUT FOR RERUN		
5525	AT	9.94000E 01 % RECOVERY	N	Y
5530	GEA	OUT FOR RERUN		
5530	GEA	1.07100E 02 % RECOVERY Cs-137	N	Y
5530	GEA	1.05100E 02 % RECOVERY Co-60	N	Y
5540	U	OUT FOR RERUN		
5540	U	OUT FOR RERUN		
5540	U	1.04400E 02 % RECOVERY	N	Y
5550	ICP	1.02620E 02 % RECOVERY Ag	N	Y
5550	ICP	1.02460E 02 % RECOVERY Al	N	Y
5550	ICP	1.01460E 02 % RECOVERY B	N	Y
5550	ICP	1.03960E 02 % RECOVERY Ba	N	Y
5550	ICP	1.06300E 02 % RECOVERY Ca	N	Y
5550	ICP	1.06240E 02 % RECOVERY Cd	N	Y
5550	ICP	1.06460E 02 % RECOVERY Co	N	Y
5550	ICP	1.06200E 02 % RECOVERY Cr	N	Y
5550	ICP	1.05660E 02 % RECOVERY Cu	N	Y
5550	ICP	1.06180E 02 % RECOVERY Fe	N	Y
5550	ICP	1.07000E 02 % RECOVERY K	N	Y
5550	ICP	1.01800E 02 % RECOVERY Li	N	Y
5550	ICP	1.01900E 02 % RECOVERY Mg	N	Y
5550	ICP	1.03480E 02 % RECOVERY Mn	N	Y
5550	ICP	1.06020E 02 % RECOVERY Mo	N	Y
5550	ICP	1.01680E 02 % RECOVERY Na	N	Y
5550	ICP	1.06180E 02 % RECOVERY Ni	N	Y
5550	ICP	1.05120E 02 % RECOVERY Pb	N	Y
5550	ICP	1.03060E 02 % RECOVERY S	N	Y
5550	ICP	1.00460E 02 % RECOVERY Si	N	Y
5550	ICP	1.03440E 02 % RECOVERY Sr	N	Y
5550	ICP	1.05960E 02 % RECOVERY V	N	Y
5550	ICP	1.04200E 02 % RECOVERY Zn	N	Y
5550	ICP	1.04380E 02 % RECOVERY Zr	N	Y
5581	Pu239/40	1.01400E 02 % RECOVERY	N	Y
5582	Am241	1.05500E 02 % RECOVERY	N	Y
8500	ACD DGST	D.F.=20		
8550	ICP	9.27400E 01 % RECOVERY Al	N	Y
8550	ICP	9.20200E 01 % RECOVERY B	N	Y
8550	ICP	9.35000E 01 % RECOVERY Ba	N	Y
8550	ICP	9.08200E 01 % RECOVERY Ca	N	Y
8550	ICP	9.14200E 01 % RECOVERY Cd	N	Y
8550	ICP	9.36600E 01 % RECOVERY Co	N	Y
8550	ICP	9.21200E 01 % RECOVERY Cr	N	Y
8550	ICP	9.44200E 01 % RECOVERY Cu	N	Y
8550	ICP	9.37600E 01 % RECOVERY Fe	N	Y
8550	ICP	8.62400E 01 % RECOVERY K	N	Y
8550	ICP	9.70800E 01 % RECOVERY Li	N	Y
8550	ICP	9.04600E 01 % RECOVERY Mg	N	Y
8550	ICP	9.01400E 01 % RECOVERY Mn	N	Y
8550	ICP	9.39800E 01 % RECOVERY Mo	N	Y

REPORT CONTINUED ON NEXT PAGE

SAMPLE STATUS REPORT FOR R 5802. 300 AREA STD TIME: 9/15/94 9:24
 DISPATCHED: 7/19/94 12: 4 SAMPLE HAS NOT BEEN SLURPED

EXT.	DETER.	RESULTS OR STATUS	OUT OF GOOD RANGE? ANS?	
****	*****	*****	***	***
8550	ICP	9.42800E 01 % RECOVERY Na	N	Y
8550	ICP	9.22400E 01 % RECOVERY Ni	N	Y
8550	ICP	9.00000E 01 % RECOVERY Pb	N	Y
8550	ICP	8.97800E 01 % RECOVERY S	N	Y
8550	ICP	8.97800E 01 % RECOVERY Si	N	Y
8550	ICP	9.45200E 01 % RECOVERY Sr	N	Y
8550	ICP	9.28800E 01 % RECOVERY V	N	Y
8550	ICP	8.93000E 01 % RECOVERY Zn	N	Y
8550	ICP	9.40200E 01 % RECOVERY Zr	N	Y
8550	ICP	8.51400E 01 % RECOVERY Ag	N	Y

END OF REPORT

SAMPLE STATUS REPORT FOR R 5803. 300 AREA BLANK TIME: 9/15/94 9:24
 DISPATCHED: 7/19/94 12: 5 SAMPLE HAS NOT BEEN SLURPED
 RECEIVED: 7/19/94 14:26

EXT.	DETER.	RESULTS OR STATUS	OUT OF GOOD RANGE?	ANS?
****	*****	*****	***	***
8600	ACD DGST	COMPLETED		
8620	TB	OUT FOR RERUN		
8620	TB	5.49000E-05 uCi/G	N	Y
8625	AT	OUT FOR RERUN		
8625	AT	< 1.60000E-05 uCi/G Rec	N	Y
8630	GEA	OUT FOR RERUN		
8630	GEA	< 2.55000E-04 uCi/G Rec Cs-137	N	Y
8630	GEA	< 1.88000E-04 uCi/G Rec Co-60	N	Y
8640	U	OUT FOR RERUN		
8640	U	OUT FOR RERUN		
8640	U	< 3.67000E-01 uG/ML	N	Y
8650	ICP	< 5.00000E-02 uG/ML Al	N	Y
8650	ICP	< 1.00000E-02 uG/ML Ag	N	Y
8650	ICP	< 5.00000E-02 uG/ML B	N	Y
8650	ICP	< 5.00000E-02 uG/ML Ba	N	Y
8650	ICP	< 1.00000E-01 uG/ML Ca	N	Y
8650	ICP	< 1.00000E-02 uG/ML Cd	N	Y
8650	ICP	< 2.00000E-02 uG/ML Co	N	Y
8650	ICP	< 1.00000E-02 uG/ML Cr	N	Y
8650	ICP	< 1.00000E-02 uG/ML Cu	N	Y
8650	ICP	< 5.00000E-02 uG/ML Fe	N	Y
8650	ICP	< 3.00000E-01 uG/ML K	N	Y
8650	ICP	< 1.00000E-02 uG/ML Li	N	Y
8650	ICP	< 1.00000E-01 uG/ML Mg	N	Y
8650	ICP	< 1.00000E-02 uG/ML Mn	N	Y
8650	ICP	< 5.00000E-02 uG/ML Mo	N	Y
8650	ICP	< 1.00000E-01 uG/ML Na	N	Y
8650	ICP	< 2.00000E-02 uG/ML Ni	N	Y
8650	ICP	< 1.00000E-01 uG/ML Pb	N	Y
8650	ICP	< 5.00000E-02 uG/ML S	N	Y
8650	ICP	< 5.00000E-02 uG/ML Si	N	Y
8650	ICP	< 1.00000E-02 uG/ML Sr	N	Y
8650	ICP	< 5.00000E-02 uG/ML V	N	Y
8650	ICP	< 1.00000E-02 uG/ML Zn	N	Y
8650	ICP	< 1.00000E-02 uG/ML Zr	N	Y
8681	Pu239/40	< 5.52000E-05 uCi/G Rec PU239/240	N	Y
8682	AM241	< 2.06000E-03 uCi/G Rec	N	Y

END OF REPORT

SAMPLE STATUS REPORT FOR R 5804. 300 AREA S4047-01 TIME: 9/15/94 9:25
 DISPATCHED: 7/19/94 12: 6 SAMPLE HAS NOT BEEN SLURPED
 RECEIVED: 7/19/94 14:26

EXT.	DETER.	RESULTS OR STATUS	OUT OF GOOD RANGE?	ANS?
****	*****	*****	***	***
5715	pH	5.27000E 00 NONE	N	Y
5815	pH	5.12000E 00 NONE	N	Y
8700	ACD DGST	1.00054E 02 G/L	N	Y
8720	TB	OUT FOR RERUN		
8720	TB	1.71000E-02 uCi/G	N	Y
8725	AT	OUT FOR RERUN		
8725	AT	7.39000E-05 uCi/G Rec	N	Y
8730	GEA	OUT FOR RERUN		
8730	GEA	6.28000E-03 uCi/G Rec Cs-137	N	Y
8730	GEA	6.07000E-03 uCi/G Rec Co-60	N	Y
8740	U	OUT FOR RERUN		
8740	U	OUT FOR RERUN		
8740	U	4.38000E-06 G/G Rec	N	Y
8755	ICP-SOL	1.73000E 00 uG/G Rec Al	N	Y
8755	ICP-SOL	7.80000E-01 uG/G Rec B	N	Y
8755	ICP-SOL	6.00000E-01 uG/G Rec Ba	N	Y
8755	ICP-SOL	2.94000E 01 uG/G Rec Ca	N	Y
8755	ICP-SOL	2.42000E 00 uG/G Rec Cd	N	Y
8755	ICP-SOL	1.59000E 00 uG/G Rec Co	N	Y
8755	ICP-SOL	1.71000E 01 uG/G Rec Cr	N	Y
8755	ICP-SOL	4.41000E 00 uG/G Rec Cu	N	Y
8755	ICP-SOL	7.19000E 01 uG/G Rec Fe	N	Y
8755	ICP-SOL	6.07000E 00 uG/G Rec K	N	Y
8755	ICP-SOL	1.48000E 00 uG/G Rec Li	N	Y
8755	ICP-SOL	2.30000E 01 uG/G Rec Mg	N	Y
8755	ICP-SOL	4.69000E 01 uG/G Rec Mn	N	Y
8755	ICP-SOL	4.87000E 00 uG/G Rec Mo	N	Y
8755	ICP-SOL	2.12000E 02 uG/G Rec Na	N	Y
8755	ICP-SOL	3.99000E 01 uG/G Rec Ni	N	Y
8755	ICP-SOL	8.12000E 00 uG/G Rec Pb	N	Y
8755	ICP-SOL	2.23000E 02 uG/G Rec S	N	Y
8755	ICP-SOL	9.77000E 00 uG/G Rec Si	N	Y
8755	ICP-SOL	1.50000E-01 uG/G Rec Sr	N	Y
8755	ICP-SOL	4.89000E 00 uG/G Rec V	N	Y
8755	ICP-SOL	5.80000E-01 uG/G Rec Zn	N	Y
8755	ICP-SOL	2.40000E-01 uG/G Rec Zr	N	Y
8781	Pu239/40	< 5.92000E-05 uCi/G Rec PU239/240	N	Y
8782	Am241	< 6.54000E-04 uCi/G Rec	N	Y
8800	ACD DGST	1.00068E 02 G/L	N	Y
8820	TB	OUT FOR RERUN		
8820	TB	1.72000E-02 uCi/G	N	Y
8825	AT	OUT FOR RERUN		
8825	AT	6.70000E-05 uCi/G Rec	N	Y
8830	GEA	OUT FOR RERUN		
8830	GEA	6.38000E-03 uCi/G Rec Cs-137	N	Y
8830	GEA	6.12000E-03 uCi/G Rec Co-60	N	Y
8840	U	OUT FOR RERUN		
8840	U	OUT FOR RERUN		
8840	U	3.90000E-06 G/G Rec	N	Y
8855	ICP-SOL	1.10000E-01 uG/G Rec Ag	N	Y
8855	ICP-SOL	1.79000E 00 uG/G Rec Al	N	Y
8855	ICP-SOL	8.19000E-01 uG/G Rec B	N	Y

REPORT CONTINUED ON NEXT PAGE

SAMPLE STATUS REPORT FOR R 5804. 300 AREA S4047-01 TIME: 9/15/94 9:25
 DISPATCHED: 7/19/94 12: 6 SAMPLE HAS NOT BEEN SLURPED

EXT.	DETER.	RESULTS OR STATUS	OUT OF GOOD RANGE?	ANS?
****	*****	*****	***	***
8855	ICP-SOL	6.30000E-01 uG/G Rec Ba	N	Y
8855	ICP-SOL	3.02000E 01 uG/G Rec Ca	N	Y
8855	ICP-SOL	2.50000E 00 uG/G Rec Cd	N	Y
8855	ICP-SOL	1.67000E 00 uG/G Rec Co	N	Y
8855	ICP-SOL	1.89000E 01 uG/G Rec Cr	N	Y
8855	ICP-SOL	4.59000E 00 uG/G Rec Cu	N	Y
8855	ICP-SOL	7.85000E 01 uG/G Rec Fe	N	Y
8855	ICP-SOL	8.71000E 00 uG/G Rec K	N	Y
8855	ICP-SOL	1.52000E 00 uG/G Rec Li	N	Y
8855	ICP-SOL	2.36000E 01 uG/G Rec Mg	N	Y
8855	ICP-SOL	4.86000E 01 uG/G Rec Mn	N	Y
8855	ICP-SOL	5.01000E 00 uG/G Rec Mo	N	Y
8855	ICP-SOL	2.18000E 02 uG/G Rec Na	N	Y
8855	ICP-SOL	4.19000E 01 uG/G Rec Ni	N	Y
8855	ICP-SOL	8.17000E 00 uG/G Rec Pb	N	Y
8855	ICP-SOL	2.29000E 02 uG/G Rec S	N	Y
8855	ICP-SOL	1.01000E 01 uG/G Rec Si	N	Y
8855	ICP-SOL	1.60000E-01 uG/G Rec Sr	N	Y
8855	ICP-SOL	5.06000E 00 uG/G Rec V	N	Y
8855	ICP-SOL	5.80000E-01 uG/G Rec Zn	N	Y
8855	ICP-SOL	2.30000E-01 uG/G Rec Zr	N	Y
8881	Pu239/40	< 5.49000E-05 uCi/G Rec PU239/240	N	Y
8882	Am241	< 2.06000E-03 uCi/G Rec	N	Y
8900	ACD DGST	100.0210G/L, WHC-1A&2 DF=20		
8920	TB	OUT FOR RERUN		
8920	TB	1.01700E 02 % RECOVERY	N	Y
8925	AT	OUT FOR RERUN		
8925	AT	1.28000E 02 % RECOVERY	N	Y
8940	U	OUT FOR RERUN		
8940	U	OUT FOR RERUN		
8940	U	8.28000E 01 % RECOVERY	N	Y
8950	ICP	9.79400E 01 % RECOVERY Al	N	Y
8950	ICP	1.01400E 02 % RECOVERY Ba	N	Y
8950	ICP	9.64400E 01 % RECOVERY B	N	Y
8950	ICP	9.82000E 01 % RECOVERY Ca	N	Y
8950	ICP	9.93600E 01 % RECOVERY Cd	N	Y
8950	ICP	1.01220E 02 % RECOVERY Co	N	Y
8950	ICP	1.09400E 02 % RECOVERY Cr	N	Y
8950	ICP	9.77800E 01 % RECOVERY Cu	N	Y
8950	ICP	1.44200E 02 % RECOVERY Fe	N	Y
8950	ICP	1.06060E 02 % RECOVERY K	N	Y
8950	ICP	9.98400E 01 % RECOVERY Li	N	Y
8950	ICP	9.50000E 01 % RECOVERY Mg	N	Y
8950	ICP	8.84000E 01 % RECOVERY Mn	N	Y
8950	ICP	1.04460E 02 % RECOVERY Mo	N	Y
8950	ICP	9.20000E 01 % RECOVERY Na	N	Y
8950	ICP	8.22000E 01 % RECOVERY Ni	N	Y
8950	ICP	9.69600E 01 % RECOVERY Pb	N	Y
8950	ICP	7.20000E 01 % RECOVERY S	N	Y
8950	ICP	9.64600E 01 % RECOVERY Si	N	Y
8950	ICP	1.01700E 02 % RECOVERY Sr	N	Y
8950	ICP	1.00620E 02 % RECOVERY V	N	Y

REPORT CONTINUED ON NEXT PAGE

SAMPLE STATUS REPORT FOR R 5804. 300 AREA S4047-01 TIME: 9/15/94 9:25
DISPATCHED: 7/19/94 12: 6 SAMPLE HAS NOT BEEN SLURPED

EXT.	DETER.	RESULTS OR STATUS	OUT OF GOOD RANGE?	ANS?
****	*****	*****	***	***
8950	ICP	9.50400E 01 % RECOVERY Zn	N	Y
8950	ICP	1.02520E 02 % RECOVERY Zr	N	Y
8950	ICP	9.25000E 01 % RECOVERY Ag	N	Y
8981	Pu239/40	1.04100E 02 % RECOVERY	N	Y
8982	Am241	1.11100E 02 % RECOVERY	N	Y

END OF REPORT

SAMPLE STATUS REPORT FOR R 5805. 300 AREA STD TIME: 9/15/94 9:26
DISPATCHED: 7/19/94 12: 7 SAMPLE HAS NOT BEEN SLURPED
RECEIVED: 7/19/94 14:26

EXT.	DETER.	RESULTS OR STATUS	OUT OF GOOD RANGE?	ANS?
****	*****	*****	***	***
5515	pH	1.00200E 02 % RECOVERY	N	Y
5520	TB	OUT FOR RERUN		
5520	TB	9.10000E 01 % RECOVERY	N	Y
5525	AT	OUT FOR RERUN		
5525	AT	9.15000E 01 % RECOVERY	N	Y
5530	GEA	OUT FOR RERUN		
5530	GEA	1.04500E 02 % RECOVERY Cs-137	N	Y
5530	GEA	1.03600E 02 % RECOVERY Co-60	N	Y
5540	U	OUT FOR RERUN		
5540	U	OUT FOR RERUN		
5540	U	9.59000E 01 % RECOVERY	N	Y
5550	ICP	1.02100E 02 % RECOVERY Ag	N	Y
5550	ICP	1.01420E 02 % RECOVERY Al	N	Y
5550	ICP	9.94400E 01 % RECOVERY B	N	Y
5550	ICP	1.00880E 02 % RECOVERY Ba	N	Y
5550	ICP	1.05960E 02 % RECOVERY Ca	N	Y
5550	ICP	1.04760E 02 % RECOVERY Cd	N	Y
5550	ICP	1.05420E 02 % RECOVERY Co	N	Y
5550	ICP	1.05120E 02 % RECOVERY Cr	N	Y
5550	ICP	1.03120E 02 % RECOVERY Cu	N	Y
5550	ICP	1.04500E 02 % RECOVERY Fe	N	Y
5550	ICP	1.00240E 02 % RECOVERY K	N	Y
5550	ICP	9.93200E 01 % RECOVERY Li	N	Y
5550	ICP	1.01060E 02 % RECOVERY Mg	N	Y
5550	ICP	1.02120E 02 % RECOVERY Mn	N	Y
5550	ICP	1.04640E 02 % RECOVERY Mo	N	Y
5550	ICP	9.89600E 01 % RECOVERY Na	N	Y
5550	ICP	1.04800E 02 % RECOVERY Ni	N	Y
5550	ICP	1.05540E 02 % RECOVERY Pb	N	Y
5550	ICP	1.02480E 02 % RECOVERY S	N	Y
5550	ICP	9.95200E 01 % RECOVERY Si	N	Y
5550	ICP	1.00620E 02 % RECOVERY Sr	N	Y
5550	ICP	1.04400E 02 % RECOVERY V	N	Y
5550	ICP	1.03780E 02 % RECOVERY Zn	N	Y
5550	ICP	1.02340E 02 % RECOVERY Zr	N	Y
5581	Pu239/40	1.03500E 02 % RECOVERY	N	Y
5582	Am241	1.12100E 02 % RECOVERY	N	Y

END OF REPORT

Appendix C

Non-Destructive Assay Results and Methods

Westinghouse
Hanford CompanyInternal
Memo

From: PFP Analytical Laboratory
Phone: 373-4654 T5-05
Date: December 27, 1994
Subject: Gamma-Ray Analysis of PRTR Primary Loop Column (IX-1)

15400-94-264

To: ~~B. G. Cornwell~~ L4-73 (With Attachment)

cc: E. J. Bitten L4-73
G. A. Westsik T5-53
C. R. Stallbaum T5-05
NDA File
ELW:LB File

As part of the campaign to characterize the PRTR primary loop column (IX-1), portable non-destructive gamma-ray analysis was used to identify and quantify the gamma-emitting radionuclides. Analysis of the PRTR primary loop column (IX-1) included calibration of a high purity germanium (HPGe) detector for a distributed source, computer modeling using MicroShield 4.0 for the gamma-ray attenuation correction factor, final calculation of the total radionuclide activity within the column, and a comparison of the calculated activity with Westinghouse Hanford Health Physics measured dose rate.

The primary isotopes detected in the PRTR primary loop column (IX-1) were ^{137}Cs and ^{60}Co . The total ^{60}Co activity in the ion exchange (IX) column is 31 mCi with a range of 9.8 to 72 mCi, and the total ^{137}Cs activity in the IX column is 2.4 mCi with a range of 0.77 to 5.7 mCi. The activity range includes uncertainty estimates for the distribution of material within the column, for overall gamma-ray attenuation by materials within the column, for the mathematical models used, and random counting uncertainties. The high value for each isotope was calculated using a calculation model, which also provided verification of the calibration method. A detailed description of the calibration and analysis methods is given in the Attachment.

Should you have any questions regarding this assay, please contact Cheryl Stallbaum on 373-2562 or myself on 373-2628.

E. W. Curfman
Manager, PFP Analytical Laboratory

Concurrence:

Roger A. Jones, Senior Scientist
222-S Laboratory

1-25-95

Date

crs

Attachment

Appendix C - 2

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PRTR Primary Loop Column Calibration and Analysis Method

INTRODUCTION

Part of the characterization of the PRTR primary loop (IX-1) column was to identify and quantify potential radionuclides in the ion exchange (IX) column using portable non-destructive assay (NDA) techniques. The IX-1 column was filled with a mixture of anion and cation resins that were used to remove contaminants from the reactor primary cooling water. Potential contaminants on the resin bed were fission products and plutonium. Because the IX column was constructed of stainless steel, the only radionuclides that could be identified and quantified were those with gamma-ray emitting isotopes.

High resolution gamma spectroscopy was selected for both identifying and quantifying the radionuclides. To analyze the ion exchange column, the high resolution gamma spectroscopy system was calibrated using characterized radionuclide sources in a mock up of the assay situation at the Plutonium Finishing Plant (PFP). Established methods for calculating the detector response to source activity at several energies were used (see Ref. 1 and 2). Because the IX-1 column was a unique assay situation, a method used to mathematically model the source activity within the column was developed at PFP for the detector calibration. The detector calibration was subjected to verification by an alternative method.

The method used for the assay of 55-gallon drums with segmented gamma-ray scanning assayers was used to assay the IX-1 column. Computer modeling was used to estimate the corrections for the attenuation of the radionuclide gamma-rays by the resin matrix and the column shell. The activity of each identified gamma-ray emitting radionuclide was calculated based on the calibration model developed at PFP for the IX-1 column. A comparison with Westinghouse Hanford Health Physics measured dose rate was used to verify the calculated activity. The results generated by this method were reviewed by a source outside the PFP Analytical Laboratory.

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DETECTOR CALIBRATION

Methodology

The IX-1 column potentially had several radionuclides present within an IX resin bed nominally 30 inches in diameter and 5 feet high. Illustrations of the IX-1 column inner structures showed the cooling water was uniformly distributed radially across the IX-1 resin bed by means of a perforated plate. The IX resins within the column were used to remove contaminants from the primary cooling water which presumably resulted in an uneven distribution of material from top to bottom of the column. To measure the IX-1 column, a detector system was needed to analyze for isotopes with gamma-ray energies close together and to view a 2-3" inch vertical slice of a 30" diameter cylinder with uniform internal distribution of radionuclides.

Mimicking a segmented gamma-ray drum assayer, a high purity germanium detector with a lead collimator 8" deep, 7" wide, and 2" high was selected for the analysis. To analyze the IX-1 column, a detector calibration which related detector response at any energy to isotopic activity was calculated. The relationship of the detector response to isotopic activity was referred to as the detector efficiency. Using characterized sources at PFP, the detector efficiency at different energies was measured at several locations within a mock 30" diameter IX column setup. The mock IX column was setup to portray the IX-1 column potential gamma-ray sources without any matrix materials present. Each source measurement location is listed in Table I and illustrated in Figure 1. Because the detector efficiencies at each location within the mock IX column were not equal, a mathematical correction which correlated the detector efficiency to the source measurement location was developed. This correction was referred to as the geometry factor. The corrected detector efficiency was referred to as the absolute detector efficiency. The absolute detector efficiency was used to calculate a full energy efficiency curve.

Table I: Calibration Source Measurement Location

Point Identification	Radius	Angle	Point Identification	Radius	Angle
D0/S0	0"	0°	D90/S2	8.56"	90°
D0/S1	8.56"	0°	D90/S3	14.88"	90°
D0/S2	12.13"	0°	D135/S1	8.56"	135°
D0/S3	14.88"	0°	D135/S2	12.13"	135°
D45/S1	8.56"	45°	D135/S3	14.88"	135°
D45/S2	12.13"	45°	D180/S1	8.56"	180°
D45/S3	14.88"	45°	D180/S2	12.13"	180°
D90/S1	8.56"	90°	D180/S3	14.88"	180°

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Figure 1: Illustration of Detector/Source Configuration
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Calculations

To calculate the full energy efficiency curve for the detector system shown in Figure 1, the isotopic activity for each source and the absolute efficiency at for each gamma-ray energy was calculated. The absolute efficiency for each energy was calculated from the detector efficiency at each source measurement location and the geometry factor. The isotopic activity, detector efficiency, and the full energy efficiency curve were calculated using established methods outlined in Ref. 1 and Ref 2. The geometry factor and the absolute efficiency at each energy were calculated using a method developed for the assay of the IX-1 column.

Isotopic Activity

The isotope activity listed on the certificate in Appendix A was corrected for nuclear decay to the date of calibration. The following formula was used to calculate the activity of each source at the time of analysis. (See Ref. 3.)

$$A_t^I = A_0^I e^{\frac{\ln 2}{T_{1/2}} \times (t_t - t_0)}$$

Where:

A_t^I =	Isotopic Activity in μCi on Analysis Date for Isotope (I);
A_0^I =	Isotopic Activity in μCi on Reference Date for Isotope (I) (see Appendix A);
$T_{1/2}$ =	The Isotopic Halflife in days (see Appendix B);
t_t =	The Analysis Date Number in days; and
t_0 =	The Reference Date Number in days (see Appendix A).

Detector Efficiency

The detector efficiency was calculated at each location measured in the mock IX column at every peak energy detected. To calculate the detector efficiency, the source isotopic activity at the time of analysis, the gamma-ray fractional intensity (see Appendix B), and the background corrected net count rate had to be known. The background corrected net count rate was calculated from source net count rate minus the background net count rate. The net count rates for the source and the background were obtained from the analysis software Aptec PC MCA/WIN Software Version 5.30, Release 3F. The detector efficiency at energy (E) for each source measurement location within the mock IX column was calculated using the following equation:

$$KE_E(\text{SL}\#) = \frac{A_t^I \times FI_E^I}{\text{CPS}_E}$$

Where:

$KE_E(\text{SL}\#)$ =	The Detector Efficiency in $\mu\text{Ci}/\text{cps}$ at Energy (E) for the Source Measurement Location (SL#);
A_t^I =	The Isotopic Activity in μCi on analysis date for Isotope (I);
FI_E^I =	The Gamma Ray Fractional Intensity at Energy (E) for Isotope (I) (see Appendix B.); and
CPS_E =	The Background Corrected Net Peak Count Rate for Energy (E).

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Geometry Factor

The geometry factor related the detector efficiency at the center of the mock IX column to the detector efficiency at every location within the mock IX column by a mathematical model. The calculation of the geometry factor included determination of the total detector efficiency and the ideal detector efficiency. The total detector efficiency was a mathematical representation of the detector efficiency as a function of the mock IX column source measurement location. The ideal detector efficiency was a mathematical representation of the detector efficiency at the center of the mock IX column as a function of the mock IX column source measurement location. Figure 2 illustrates the total detector efficiency and the ideal detector efficiency.

Figure 2: Graphical Representation of the Detector Efficiency for ^{137}Cs

Where:

Total Detector Efficiency; and

Ideal Detector Efficiency.

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The first step in determining the geometry factor was developing a mathematical model to describe the total detector efficiency (see Figure 2). The total detector efficiency related the detector efficiency to the source measurement location. Lotus® 1-2-3® Data Regression routine (see. Ref. 4) was used to fit the detector efficiency at each measurement location to the following equation:

$$\ln[KE_E(SL\#)] = a_E N^2 + b_E N + c_E$$

Where: $KE_E(SL\#)$ = The Detector Efficiency in $\mu\text{Ci/cps}$ at Energy (E) for the Source Measurement Location (SL#);
 N = The Radius in inches times the cosine of the radial angle where the radial angle was equal to the angle divided 180° and multiplied by the constant π ($\pi = 3.141592$);
 a_E = The Regression Coefficient at Energy (E) from the fit of $\ln(\text{Efficiency})$ vs. Radius x $\cos(\text{Radial Angle})$;
 b_E = The Regression Coefficient at Energy (E) from the fit of $\ln(\text{Efficiency})$ vs Radius x $\cos(\text{Radial Angle})$; and
 c_E = The Regression Coefficient at Energy (E) from the fit of $\ln(\text{Efficiency})$ vs Radius x $\cos(\text{Radial Angle})$.

Because some of the data indicated that the source measurement locations were close to the edge of the detector view (see Figure 1), the following two criteria were applied to the data to determine which locations would be used to calculate the final regression equation:

- The Regression Output "R Squared" value had to be greater than 0.9506 ($R=0.975$) where R was used as an indicator to determine how well the regression model fit the data (the closer R was to 1 the better the fit); and
- The experimental data had to fall within the 99.5% CI of the Regression calculated efficiency data (the 99.5% CI was calculated from the uncertainty in the regression data).

The following source measurement locations passed the above criteria and were used to calculate the regression data: D0/S0, D0/S1, D0/S2, D0/S3, D45/S1, D45/S2, D90/S1, D90/S2, D135/S1, D135/S2, D135/S3, D180/S1, D180/S2, and D180/S3.

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The purpose of the regression fit was to determine the coefficients a_E , b_E , and c_E . The total detector efficiency for the mock IX column was calculated by taking the integral of $\ln[KE_E(SL\#)]$ with respect to N where $N = -14.88$ to 14.88 . The formula for the integral was:

$$\int_{-14.88}^{14.88} \ln[KE_E(SL\#)] dN = \frac{a_E}{3} N^3 + \frac{b_E}{2} N^2 + c_E N$$

The ideal detector efficiency was calculated by taking the integral of $\ln[KE_E(SL\#_0)]$ with respect to N where $N = -14.88$ to 14.88 . The formula for the integral was:

$$\int_{-14.88}^{14.88} \ln[KE_E(SL\#_0)] dN = c_E N$$

Where: $KE_E(SL\#_0) =$ The efficiency at the center of the mock IX column where
 $KE_E(SL\#_0) = c_E$.

The geometry factor was calculated by taking the ratio of the total detector efficiency to the ideal detector efficiency. The formula for the geometry factor was:

$$GF_E = \frac{\int_{-14.88}^{14.88} \ln[KE_E(SL\#)] dN}{\int_{-14.88}^{14.88} \ln[KE_E(SL\#_0)] dN}$$

Absolute Detector Efficiency

The absolute detector efficiency corrected the detector efficiency at the center of the mock IX column to the detector efficiency at every location within the mock IX column. The absolute detector efficiency was calculated using the following equation.

$$Eff_E = e^{c_E \times GF_E}$$

Where: $Eff_E =$ The absolute efficiency for a 30" IX column at Energy (E) in $\mu Ci/cps$;
 $c_E =$ The Regression Coefficient at Energy (E) from the fit of Efficiency vs Radius x Cos(Radial Angle); and
 $GF_E =$ The Geometry Factor for a 30" IX column at Energy (E).

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Full Energy Efficiency Curve

Because the detector efficiency was energy dependent, a relationship between the absolute detector efficiency ($\mu\text{Ci}/\text{cps}$) versus energy (keV) had to be determined. The relationship of the absolute detector efficiency versus energy was known as the full energy efficiency curve. The full energy efficiency curve was fitted using the Lotus® 1-2-3® Data Regression routine (see Figure 3). The equation used as the regression model was:

$$\ln(\text{Eff}_E) = a \ln(E)^4 + b \ln(E)^3 + c \ln(E)^2 + d \ln(E) + e$$

Where:

<p>Eff_E =</p> <p>E =</p> <p>a =</p> <p>b =</p> <p>c =</p> <p>d =</p> <p>e =</p>	<p>The absolute detector efficiency for a 30" diameter IX column at Energy (E) in $\mu\text{Ci}/\text{cps}$;</p> <p>The gamma-ray energy in keV;</p> <p>Regression Coefficient from the fit of $\ln(\text{Eff}_E)$ vs $\ln(E)$;</p> <p>Regression Coefficient from the fit of $\ln(\text{Eff}_E)$ vs $\ln(E)$;</p> <p>Regression Coefficient from the fit of $\ln(\text{Eff}_E)$ vs $\ln(E)$;</p> <p>Regression Coefficient from the fit of $\ln(\text{Eff}_E)$ vs $\ln(E)$;</p> <p>and</p> <p>Regression Coefficient from the fit of $\ln(\text{Eff}_E)$ vs $\ln(E)$.</p>
--	---

The full energy efficiency curve was used to calculate the absolute detector efficiency at any energy for the detector system illustrated in Figure 1. The full energy efficiency curve just described was specific for a 30" diameter IX column whose center was located 30 inches from the detector face.

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Figure 3: Graphical Representation of Full Energy Efficiency Curve

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INSTRUMENTATION

The instrumentation used in the analysis of the IX-1 column was a high purity germanium (HPGe) detector (Canberra Industries Serial #8912843) with a multichannel analyzer. The multichannel analyzer was an Ortec Spectrum Master™ 920-16 with Ortec Maestro™ Software Version 2.03 (for Windows). Aptec PC MCA/WIN® Software Version 5.30, Release 3F was used to analyze the spectra collected with the Ortec Spectrum Master™ 920-16.

SOURCES

The following sources were used to calibrate the gamma-ray detector:

Am-241	Source No. CE 885;
Co-57	Source No. CE 884;
Ba-133	Source No. 1R726;
Cs-137	Source No. 1S594; and
Co-60	Source No. 1U548.

Appendix A contains copies of the source certificates. Appendix B summarizes, from the literature, radiological information on the sources used during the calibration and assay.

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QUALITY CONTROL

Because many conditions can affect the performance of a detector system, a method to confirm the detector was operating as it was at the time of calibration had to be developed. To verify the detector was operating within the 95% CI interval of the method, the sources used for calibration were measured immediately after calibration and before any sample analysis. The activity for each source was calculated using a full energy efficiency curve developed for point sources. The calculated activities were then compared to the activity corrected for decay to the date of analysis. If the decay corrected reference activities were within the 95% CI of the calculated activities, the detector system was considered to be operating within the limitations of the method.

Point Source Calibration

To calculate the source activities, a full energy efficiency curve that was specific to point sources was developed. The full energy efficiency curve for point sources was calculated from the regression fitted detector efficiency at the center of the mock IX column. Because the detector response for a point source was dependent on the distance from the detector to the source, the detector efficiency was corrected for distance using the $1/R^2$ rule (R being the detector to source distance). (See Ref. 3.) The detector efficiency distance correction at each energy was calculated using the following formula:

$$PEff_E = \frac{KE_E(SL\#_0)}{D^2}$$

Where: $PEff_E$ = The Efficiency for a Point Source at Energy (E) in $\mu Ci/(cps \cdot in^2)$
 $KE_E(SL\#_0)$ = The Detector Efficiency at the center of the mock diameter IX column at Energy (E) in $\mu Ci/cps$.
 $KE_E(SL\#_0) = e^{c_E}$

Where: c_E = The Regression Coefficient at Energy (E) from the fit of Efficiency vs Radius x Cos(Radial Angle).
 D = The distance from the detector face to the mock IX column center at time of setup (D = 30 inches).

The full energy efficiency curve for the point source was calculated using the same method outline for the calculation of the 30" diameter IX column full energy efficiency curve.

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Source Activity

To calculate the measured activity for each source, the detector efficiency for a point source, the gamma-ray fractional intensity, and the background corrected net count rate at each gamma-ray energy were required. The activity of each the source was calculated using the following equation:

$$SA_E^I = \frac{PEff_E \times CPS_E \times D^2}{FI_E^I}$$

Where:

SA_E^I =	The source activity in μCi at Energy (E) for Isotope (I);
$PEff_E$ =	The Efficiency for a Point Source at Energy (E) in $\mu\text{Ci}/(\text{cps} \cdot \text{in}^2)$;
FI_E^I =	The Gamma Ray Fractional Intensity at Energy (E) for Isotope (I) (see Appendix B.);
CPS_E =	The Background Corrected Net Peak Count Rate for Energy (E); and
D =	The distance from the detector face to the source.

To confirm the instrument operation, the measured source activities were compared to the decay corrected reference activities. The sources labeled as SCHK 002 to SCHK 004 were analyzed immediately after the detector system calibration, and the sources labeled as PRTR 001 were analyzed immediately before the measurement of the IX-1 column. Table II shows that the source decay corrected reference activities for ^{60}Co and ^{137}Cs were within 95% CI uncertainty range of the measured activity. The measured activity uncertainty ranges for all source runs did overlap. Therefore, the detector system used to analyze the IX-1 column was okay. Because ^{60}Co and ^{137}Cs were the only isotopes detected in the IX-1 column, the source activities for these isotopes were only reported in this document (see Table II).

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Table II: Source Activities

⁶⁰Co Activity			
Identification Number	Activity (μCi)	95% CI Uncertainty Range (μCi)	Decay Corrected Reference Activity (μCi)
SCHK 002	2.161	1.255 to 3.067	1.924
SCHK 003	2.117	1.229 to 3.005	1.924
SCHK 004	2.206	1.275 to 3.137	1.924
SCHK 005	2.125	1.227 to 3.023	1.924
PRTR 001	1.674	0.9753 to 2.373	1.921

¹³⁷Cs Activity			
Identification Number	Activity (μCi)	95% CI Uncertainty Range (μCi)	Decay Corrected Reference Activity (μCi)
SCHK 002	9.297	5.422 to 13.17	8.849
SCHK 003	9.332	5.448 to 13.22	8.849
SCHK 004	9.248	5.395 to 13.10	8.849
SCHK 005	9.298	5.417 to 13.18	8.849
PRTR 006	7.395	4.313 to 10.48	8.847

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SAMPLE ANALYSIS

For the analysis of the IX-1 primary loop column, the detector system was configured in the same manner as was shown in Figure 1. The detector system was place north of the PRTR upper vault opening pointing south.

The IX-1 column was a 30 inch diameter by 6 feet 5 inches tall stainless steel column filled with Amberlite® IRN-77 and IRN-78 IX resin. The IX resin within the column was 4 feet 11 inches tall and started 9" from the column top. Water from the primary loop entered the top of the column and was uniformly distributed radially across the resin bed by means of a perforated plate (see Figure 4, Section A-A). Figure 4 is a diagram of the IX-1 column.

A crane was used to remove the IX-1 column from the PRTR upper vault, support the column in free space during analysis, and move the column to each analysis segment. The IX-1 column was analyzed for a counting time of 100 seconds per segment and rotated during each analysis. Because the radionuclide activity was assumed to be located only in the IX resin, the analysis of the column started at the top of the resin bed which was marked by a seam at the top of the IX-1 column. Each segment was measured in 3 inch increments from the previous measurement location. Because of time constraints and field limitations, only 9, 3" segments were analyzed on the column.

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Figure 4: PRTR Primary Loop Column (IX-1)
(Drawing Reference: H-3-13990)

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Gamma-Ray Analysis

The gamma-ray spectrums collected in the field were analyzed using Aptec PC MCA/WIN Software 5.30, Release 3F. Figure 5 shows a typical gamma-ray spectrum collected from the analysis of the IX-1 column. The radionuclides present in the gamma-ray spectrum were ^{137}Cs and ^{60}Co .

Figure 5: Segment PRTR_010 Gamma-Ray Spectrum

The Aptec PC/MCA Analysis Software was setup to look for any gamma-ray peaks above the spectral background. The software peak search parameters are listed in Table V. All detected peaks were then fitted to a gaussian curve because of low energy tailing observed in the gamma-ray spectrum. The gaussian multifit parameters are listed in Table VI. Figure 6 shows a typical gaussian fit for the ^{60}Co 1332.49 keV peak.

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Table V: IX-1 Primary Loop Peak Fit Parameters

Peak Calculation Setup for IX Primary Loop Column	
ROI Bkgnd Line ☐ Straight Line Least Square Fit Degree ☒ 1 ☐ 2 ☐ 3 ☐ 4 ☐ 5 With <input type="text" value="5"/> End Channels ☐ Use Method 2	Centroid Calculated ☐ Between Gross Half Max ☒ Between Net Half Max ☐ Between Net Tenth Max ☐ Over Whole ROI
Confidence MDA Sigma <input type="text" value="1.645"/> Error Sigma <input type="text" value="1"/>	Centroid Weighting ☐ Net * Chan ☒ Net * abs(Net) * Chan
Peak Search ☒ All Start <input type="text"/> ☐ Partial Stop <input type="text"/>	
<div> <input type="button" value="Defaults"/> <input type="button" value="Recall"/> <input type="button" value="Save"/> <input type="button" value="OK"/> <input type="button" value="Cancel"/> </div>	

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Table VI: IX-1 Primary Loop Multifit Parameters

Fitting Parameters for PRTR primary loop column (IX-1) Data	
Spectrum Type ☒ HPGe/Ge(Li) ☐ Na(Tl)	
Optimize Centroid Search for ☒ Separate Overlaps ☐ Close Overlaps	
Peaks to be Fitted ☐ Multiplets Only ☒ Multiplets and Singlets Whose Amplitude was $\geq$ Minimum Detectable Amplitude times <input type="text" value="1.645"/> Sigma	
Fit with Peak Width ☐ Varying ☒ Fixed	Fit with Background ☐ Varying ☒ Fixed
Accuracy <div>Maximum Iterations <input type="text" value="50"/></div> <div>Percent Change <input type="text" value="0.25"/></div> <div>Minimum Reduced Chi Square <input type="text" value="0.5"/></div>	
<input type="button" value="Defaults"/> <input type="button" value="Recall"/> <input type="button" value="Save"/> <input type="button" value="OK"/> <input type="button" value="Cancel"/>	

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Figure 6: Typical Multifit for Co-60 1332.49 keV Peak (PRTR_010)

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The isotopic activity for each IX-1 column segment analyzed was calculated from the results generated from the analysis software. The activity in mCi was calculated using the following formula:

$$AS_E^I = \frac{CPS_E \times Eff_E}{FI_E^I} \times \frac{1 \text{ mCi}}{1000 \mu\text{Ci}} \times CF_E$$

Where:

- AS_E^I = The Segment Activity in mCi at Energy (E) for Isotope (I);
- CPS_E = The Background Corrected Net Count Rate in counts per second from the analysis software at Energy (E);
- Eff_E = The absolute efficiency at Energy (E) in $\mu\text{Ci}/\text{cps}$ from the 30" IX Column Full Energy Efficiency Curve;
- FI_E^I = The Gamma Ray Fractional Intensity at Energy (E) for Isotope (I) (see Ref. 3); and
- CF_E = The Gamma-Ray Attenuation Correction Factor at Energy (E) (see IX-1 Gamma-Ray Attenuation).

The activity for isotopes with 2 or more peaks was averaged. The total activity for the IX-1 column was calculated by assuming the reduction in activity from the maximum activity segment to 24" below the top seam of the column was linear. The maximum activity segment for ^{137}Cs was 6 inches below the top seam and for ^{60}Co was 9 inches below the top seam. The data from the maximum activity segment to 24" below the top seam of the column was fitted to the following regression model equation:

$$AS_E^I = a D_s + b$$

Where:

- AS_E^I = The Segment Activity in mCi at Energy (E) for Isotope (I);
- D_s = The Distance in inches from the IX-1 column top seam to the analyzed segment center;
- a = Regression Coefficient from the fit of Segment Activity vs. Distance from the IX-1 column top seam; and
- b = Regression Coefficient from the fit of Segment Activity vs. Distance from the IX-1 column top seam.

The isotopic activity for each 3" segment was calculated from the maximum activity segment to the bottom of the IX column (~60 inches) using the regression model equation. The total column activity was calculated as the sum of ALL the 3" segments.

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IX-1 GAMMA-RAY ATTENUATION

One parameter required for the analysis of the IX-1 column was the overall gamma-ray attenuation correction factor by matrix materials. The gamma-ray attenuation correction factor defined the amount of gamma-ray absorption within the sample and other materials located between the source and the detector. The attenuation was a function of the gamma-ray transmission. The gamma-ray transmission was defined as the ratio of the gamma-ray intensity with shielding between the source and the detector to the gamma-ray intensity without shielding between the source and the detector. Two common methods of determining the gamma-ray transmission were direct measurement and computer modeling. The direct measurement of the gamma-ray transmission required the use of a source with an activity high enough to be detected by the detector system when placed on the sample side opposite of the detector. Because a high activity source presented personnel dose rate problems and the sample configuration caused source positioning difficulties, the direct measurement method for determining the gamma-ray transmission was not selected. The other method to determine the gamma-ray transmission was computer modeling. To determine the gamma-ray transmission with the computer modeling, the sample attenuating materials and the source distribution within the sample had to be known.

Transmission

Computer Modeling

Because the attenuating materials within the column were well known and the column was designed to deliver a radially uniform distribution of liquid onto the column, computer modeling was selected as the method to calculate the sample gamma-ray transmission. MicroShield 4.0 was selected for the computer modeling code. The following assumptions were made about the transmission model:

- The material was homogeneously distributed through out the segment being analyzed; and
- The IX resin contained no material that was more attenuating than the IX resin itself.

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Two runs of the model code were required to calculate the gamma-ray transmission. The two runs varied only in the shielding materials used. One run was used to calculate the gamma-ray intensity with the shielding materials between the source and the dose point, and the other run was used to calculate the gamma-ray intensity without shielding material between the source and the dose point. The dose point represented the front face of the gamma-ray detector. The MicroShield 4.0 modeling parameters were as follows (see Figure 7):

Source Geometry: Cylinder Volume/Side Shields

Geometry Dimensions: X = 30 inches Distance from Source Center to Dose Point
Y = 3½ inches Arbitrarily chosen equals ½H
R = 14¾ inches Radius of Source (Resin Bed)
H = 7 inches Arbitrarily chosen

Shield Dimensions: CY1 = ¾ inches Thickness of the stainless steel column

Shield Material: See Table III.

Table III: MicroShield 4.0 Shielding Materials

Shield Description	Air (No Shielding)		Resin/Column (Shielding)	
	Material	Density (g/cm ³)	Material	Density (g/cm ³)
Source	Air	0.00122	50% Mixture of: IRN-77 Cation Resin* IRN-78 Anion Resin*	0.625 0.600
Shield 1	Air	0.00122	Iron (Column Material)	7.86

* MicroShield Custom Material (see Appendix C.)

Source(s): See Table IV.

Table IV: MicroShield 4.0 Source Energy and Activity

Energy (MeV)	Activity (photons/sec)
0.662	2.834E+10
1.173	3.700E+10
1.332	3.700E+10

Build-Up Reference: NO BUILDUP MODEL

Integration Parameters: Radial 16
Circumferential 16
Axial (Along Z) 16

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The MicroShield 4.0 modeling code calculated the gamma-ray intensity as a fluence rate when the above parameters were entered. The gamma-ray fluence rate was the number of photons/cm²/sec emitted by the sample at the gamma-ray energy of interest. This fluence rate was then used to calculate the overall gamma-ray transmission for each energy modeled.

Figure 7: Illustration of MicroShield 4.0 Computer Model Setup

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Calculations

The fluence rate from the MicroShield model for Air and for the Resin/Iron was used to calculate the gamma-ray transmission at the dose point. Because the gamma-ray transmission is energy dependant, the transmission had to be calculated for each gamma-ray emitted by the sample. Since the sample only had ^{137}Cs and ^{60}Co , the transmission model was created only for these isotopes and their associated gamma-ray energies. The following formula was used to calculate the transmission:

$$T_E = \frac{FR_E^R}{FR_E^A}$$

Where: T_E = The Gamma-Ray Transmission at Energy (E);
 FR_E^R = The Fluence Rate from MicroShield 4.0 for the Resin/Iron Shielding at Energy (E); and
 FR_E^A = The Fluence Rate from MicroShield 4.0 for Air at Energy (E).

The gamma-ray attenuation correction factor was calculated using the following formula:

$$CF_E = \frac{1}{T_E}$$

Where: CF_E = The Gamma-Ray Attenuation Correction Factor at Energy (E); and
 T_E = The Gamma-Ray Transmission at Energy (E).

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METHODOLOGY CHECK

Calibration Separate Verification

The calibration of the system was verified with a method outlined by Los Alamos National Laboratory (LANL) for plutonium measurements (Ref. 5). This method assumed a line source that was parallel with the detector and located at the center of the column. The efficiency was calculated by determining the radial response of the detector or the effective length. After the effective length was calculated a detector efficiency constant which related the detector response in counts per second per inch squared to activity was calculated. Using the diameter of the 30" diameter IX column and the distance from the detector to the center of the column, a detector efficiency at each peak energy was calculated specifically for the 30" diameter IX column. The sample total activity was then calculated with this efficiency versus energy relationship.

Independent Verification of Portable Non-Destructive Assay Results

To verify the portable non-destructive assay results, an independent check was completed. The verification compared the Westinghouse Hanford (WHC) Health Physics measured dose rate of the PRTR primary loop column (IX-1) with a computer modeled dose rate. The computer modeled dose rate was calculated using the portable non-destructive assay results. MicroShield 4.0 was used to calculate the computer modeled dose rate. The following parameters were used for MicroShield 4.0 dose rate model:

Source Geometry:	Cylinder Volume/Side Shields	
Geometry Dimensions:	The dose was modeled at three different locations from the column: On Contact, 1 foot, and 3 feet.	
Model #1	X = 15½ inches	Distance from Source Center to Dose Point, MODELS ON CONTACT
Model #2	X = 27½ inches	MODELS AT 1 FOOT
Model #3	X = 51½ inches	MODELS AT 3 FEET
	Y = 29½ inches	½ the Column Resin Bed Height
	R = 14¾ inches	Radius of Source (Resin Bed)
	H = 59 inches	Column Resin Bed Height
Shield Dimensions:	Cyl = ¾ inches	Thickness of Stainless Steel Column
Shield Material:		
Source:	50% Mixture of	
	IRN-77 Cation Resin (Density=0.625 g/cm ³)	
	IRN-78 Anion Resin (Density=0.6 g/cm ³)	
	(See Appendix C.)	

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Source(s): See Table VII.

Table VII: MicroShield 4.0 Source Activity

Isotope	Activity** (Ci)
Ba-137m*	2.07E-3
Cs-137	2.43E-3
Co-60	2.43E-3

* The ^{137}Cs decay product ^{137}Ba was included in the model because it is the isotope which emits the 662 keV gamma-ray associated with ^{137}Cs .

** The activities listed were calculated from the measured portable nondestructive assay data.

Build-Up Reference: Source

Integration Parameters: Radial 16
Circumferential 16
Axial (Along Z) 16

Table VIII lists the MicroShield 4.0 modeled dose rates along with the WHC Health Physics measured dose rates of the PRTR primary loop column (IX-1). The dose rates calculated using the computer model have very good comparison with the WHC Health Physics measured dose rates.

Table VIII: Measured Dose versus Modeled Dose

Measurement Location	Measured Dose Rate* (mR/hr)	Modeled Dose Rate (mR/hr)	Modeled Dose Rate** 95% CI Range (mR/hr)
On CONTACT	50	63	20 to 106
At 1 Foot	10	28	9 to 47
At 3 Feet	2	9	3 to 16

* See Ref. 6.

** The 95% CI Range of the modeled dose rates were calculated using the 95% CI activity range determined by the uncertainty in the measurement.

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APPENDIX A: SOURCE CERTIFICATES

Reproduction of Certificate

Amersham Corporation
2636 S. Clearbrook Drive
Arlington Heights, IL 60005

USA-Arlington Heights, IL

Certificate no. 494380-2

Type	AMR1151 to drawing VZ-1240/E
Source No	CE 885
Activity	402 kBq (10.9 μ Ci) Americium-241
Reference Date	1 September 1990
Overall uncertainty*	$\pm 5 \%$
Capsule	0.5 mm plastic (density 1.18 g/cm ³)
Integrity ISO/DIN Classification	C.34343
The source passed the tests for for leakage** and contamination**	L and A
Date of test	3 September 1990

* This estimate of uncertainty was calculated in accordance with the recommendation of the International Commission on Radiation Units and Measurements (ICRU Report 12). The limits of uncertainty were taken as the arithmetic sum of the uncertainty due to random variations, calculated at the 99.7% confidence level and the estimated maximum systematic uncertainty in the measurement.

** See overleaf for description of tests

Remarks Traceability: This product satisfies the quality assurance requirements of USNCR Regulatory Guide 4.15 Revision 1, February 19, 1979, for achieving NIST traceability, as defined in NCRP 58 (1985), through Amersham's participation in the USCEA/NIST Measurements Assurance Program of the Nuclear Power Industry.

Amersham Buchler

Braunschweig, 4 September 1990 Dor/AM

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Reproduction of Certificate

Amersham Corporation
2636 S. Clearbrook Drive
Arlington Heights, IL 60005

USA-Arlington Heights, IL

Certificate no. 494380-1

Type	CTR1151 to drawing VZ-1240/E
Source No	CE 885
Activity	433 kBq (11.7 μ Ci) Cobalt-57
Reference Date	1 September 1990
Overall uncertainty	$\pm 5 \%$
Capsule	0.5 mm plastic (density 1.18 g/cm ³)
Integrity ISO/DIN Classification	C.34343
The source passed the tests for for leakage" and contamination"	L and A
Date of test	3 September 1990

* This estimate of uncertainty was calculated in accordance with the recommendation of the International Commission on Radiation Units and Measurements (ICRU Report 12). The limits of uncertainty were taken as the arithmetic sum of the uncertainty due to random variations, calculated at the 99.7% confidence level and the estimated maximum systematic uncertainty in the measurement.

** See overleaf for description of tests

Remarks Traceability: This product satisfies the quality assurance requirements of USNCR Regulatory Guide 4.15 Revision 1, February 19, 1979, for achieving NIST traceability, as defined in NCRP 58 (1985), through Amersham's participation in the USCEA/NIST Measurements Assurance Program of the Nuclear Power Industry.

Amersham Buchler

Braunschweig, 4 September 1990 Dor/AM

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Reproduction of Certificate

Amersham International pic
Amersham UK

Description
BDR.151

Principal radionuclide: Barium-133

Product Code:

Source Number: 1R726

Measurement

Reference time: 1200 GMT on 1 December 1982

Activity of the principal nuclide: 10.38 microcuries

Recommended half life: 10.8 years

Method of measurement:

The source was measured using equipment calibrated directly or indirectly with similar sources prepared from a series of absolutely standardized solutions.

Accuracy

The OVERALL UNCERTAINTY in the activity quoted above for the principal radionuclide was estimated to be less than $\pm 4.8\%$

The estimate of uncertainty was calculated in accordance with the recommendations of the international Commission on Radiation Units and Measurements (ICRU Report 12). The limits of uncertainty were taken as the arithmetic sum of the uncertainty due to random variations, calculated at the 99.7% confidence level, and the estimate systematic uncertainties in the measurement.

Radionuclidic
Purity

The estimated activities of any radionuclide impurities found by high resolution gamma spectrometry are listed below expressed as percentages of the principal radionuclide at the reference time.

Remarks

Tests for leakage and surface contamination have been carried out with satisfactory results

Further information about this source including details of its construction, purity of the radionuclide, recommended nuclear data and half life is given in the data sheet accompanying the source.

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Reproduction of Certificate

Certificate of measurement of gamma reference sources

R 1250

Nuclide	²⁴¹ Am	¹³³ Ba	¹³⁷ Cs	⁶⁰ Co	⁸⁸ Y	²² Na	⁵⁴ Mn	²⁰³ Hg	⁵⁷ Co
Colour Code	Red	White	Yellow	Light Green	Black	Cream	Blue	Pink	Dark Green
Source No.	1Q522	1R44	1S594	1U548	1Y681	1X498	1V582	1W693	2T086
Activity, μ Ci	10.07	11.05	11.53	12.10	11.65	10.92	10.13	21.72	11.26
Accuracy %	5.0	4.8	3.7	1.9	5.0	3.7	3.7	3.0	4.4

Activity reference time 1200 GMT on 1 December 1980

Wipe test: each source has passed a wipe test with an acceptance limit of 0.005 μ Ci.

Accuracy is defined as the OVERALL UNCERTAINTY of the result.

Definitions of the terms used in this certificate and further details of the sources are given in the data sheet provided with each set or replacement source.

Set No 1316 Checked by Approved by
for Managing Director

The Radiochemical Centre Amersham England

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APPENDIX B: SOURCE LITERATURE INFORMATION

Table IX is a list of the literature values (see Ref. 7) for each isotope measured in the calibration.

Table IX:

Isotope	Half life	Gamma-Ray Energy (keV)	Gamma-Ray Percent Intensity	Reference Page # Decay Mode
Americium-241	432.2 years	59.537 keV	35.9%	Page 211 ^{241}Am α Decay (τ_{12})
Barium-133	10.5 years	80.997 keV	33.0%	Page 138 ^{133}Ba EC Decay (τ_3)
Cobalt-57	270.9 days	122.063 keV	85.51%	Page 76 ^{57}Co EC Decay (τ_2)
Barium-133	10.5 years	276.397 keV	6.9%	Page 138 ^{133}Ba EC Decay (τ_6)
Barium-133	10.5 years	302.839 keV	17.8%	Page 138 ^{133}Ba EC Decay (τ_7)
Barium-133	10.5 years	356.005 keV	60%	Page 138 ^{133}Ba EC Decay (τ_8)
Barium-133	10.5 years	383.851 keV	8.7%	Page 138 ^{133}Ba EC Decay (τ_9)
Cesium-137	30.17 years	661.649 keV From ^{137}Ba	^{137}Ba Feed 94.6% ^{137}Ba 89.98% • Overall: 85.12%	Page 143 ^{137}Cs β^- Decay Feeds to ^{137}Ba IT Decay
Cobalt-60	5.271 years	1173.216 keV	100%	Page 78 ^{60}Co EC Decay (τ_3)
Cobalt-60	5.271 years	1332.486 keV	100%	Page 78 ^{60}Co EC Decay (τ_4)

* The OVERALL Gamma-Ray Intensity of 85.12% for Cesium-137 is the fractional percent of Cesium-137 which feeds to Barium-137 (0.946) times the fractional percent gamma-ray intensity of Barium-137 661.649 keV gamma-ray 0.8998 times 100%.

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APPENDIX C: MICROSHIELD 4.0 CUSTOM MATERIALS

To model the attenuation within the PRTR primary loop column (IX-1), the resins within the column had to be specially modeled in MicroShield 4.0 as a custom material. The computer modeling of the ion exchange resins required the elemental knowledge for each resins within the column and the major constituents that each resin composes. Figure 7 shows the molecular formula of IRN-77, and Figure 8 shows the molecular formula of IRN-78 (see Ref. 8).

Figure 7: Cation Exchange Resin Amberlite® IRN-77 in the Lithium Form

Figure 8: Anion Exchange Resin Amberlite® IRN-78 in the Chloride Form

Table X lists the physical properties for each ion exchange resin needed for the MicroShield 4.0 custom materials model.

Table X: Exchange Resins Physical Properties

Exchange Resin	Chemical Form	Molecular Formula	Major Constituents (Ref. 6)	Density (g/cm ³)
IRN-77 (Anion)	Lithium	C ₈ H ₈ SO ₃ Li	48% Resin 52% Water	1.2*
IRN-78 (Cation)	Chloride	C ₁₂ H ₁₈ N ⁺ Cl ⁻	52% Resin 48% Water	1.25**

* See Ref. 9

** See Ref. 10

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Appendix D.

Ion Exchange Column Reference Information

PRTR Ion Exchange Vault					
Vault is located north east of 309 Building Containment Dome. Vault structural drawing is: H-3-11018 Piping arrangement drawing references: H-3-1125, H-3-11652, H-3-11653					
	Position 1 (North)	Position 2	Position 3	Position 4 (South)	
Cleanup System Identification	IX-2 Reflector Loop Cleanup	IX-3 Moderator Loop Cleanup	IX-1 Primary Loop cleanup	IX-4 Top and Bottom Shield Cooling loop (Used before 1966)	BIX-5 Boron Removal Exchanger (Use after 1966)
Mechanical Data Drawing Diameter Material	H-3-13989-1 1'-6" Dia. 1/4" wall ASTM 6061-T6	H-3-15266 H-3-13989-1 EPN 230.3 1'-6" Dia. 1/4" wall ASTM 6061-T6	H-3-13990-1 2'-6" Dia. 1/8" wall ASTM A240 Type 304	H-3-13990-1 2'-6" Dia. 1/8" wall ASTM A240 Type 304	H-3-15067 34 in Dia. 304 Stainless Steel ASME B&PV Sec III Class C
Resin Data Specification Resin Type Expected Life Resin Volume	HWS 8512 (1) Basic Anion Resin/2 parts Amberlite XE-78* Acidic Cation Resin/3 parts Amberlite XE-77** Expected Life 4 to 8 mo. 7.9 ft ³	HWS 8512 (1) Basic Anion Resin/2 parts Amberlite XE-78* Acidic Cation Resin/3 parts Amberlite XE-77** Expected Life 4 to 8 mo. 7.9 ft ³	Basic Anion Resin Amberlite XE-78* Acidic Cation Resin Amberlite XE-77** Expected Life 4 to 8 mo. 22.1 ft ³ Mix Ratio Equal by Volume	22.1 ft ³	Strong Base Anion Resin Volume approx. 25.8 ft ³
Cleanup System Flow Diagram	SK-3-9747	SK-3-9747	SK-1-6375	H-3-11567 SK-3-9732	H-3-15166
Potential Resin Contents	May contain corrosion products	May contain corrosion products, one column in lower vault expected to contain failed fuel.	May contain corrosion and fission products	IX-4 relocated to inside containment around 1965. May contain corrosion products.	May contain corrosion products

* Anion - Strong basic quaternary ammonium compound converted to hydroxide form

** Cation - Highly acidic sulfonic acid type converted to hydrogen form

- (1) HWS 8512 "Specification for the procurement of Resin, Mixed Bed, Moderator Type, For IX 2 and 3," August 31, 1965. Also, HWS 8511 "Specification for the Procurement of Resin, Mixed Bed. Lithium Based," Sept. 15, 1967.