1 of 1

5

MAY 23 1994

ENGINEERING DATA TRANSMITTAL

Page 1 of 1

1. EDT 156449

2. To: (Receiving Organization) Eng Devel & Disposal Systems		3. From: (Originating Organization) Facility & Equip Engineering		4. Related EDT No.: N/A	
5. Proj./Prog./Dept./Div.: W7E850		6. Cog. Engr.: R. C. Brown, 376-0285		7. Purchase Order No.: N/A	
8. Originator Remarks: Please review for approval and provide written comments on the attached draft document, "Radiological Implications of Contact Maintenance: Impacts on Facility Design," WHC-SD-WM-TI-607. Please return comments by 2/25/94.				9. Equip./Component No.: N/A	
				10. System/Bldg./Facility: N/A	
11. Receiver Remarks:				12. Major Assm. Dwg. No.: N/A	
				13. Permit/Permit Application No.: N/A	
				14. Required Response Date: 2/25/94	

15. DATA TRANSMITTED					(F)	(G)	(H)	(I)
(A) Item No.	(B) Document/Drawing No.	(C) Sheet No.	(D) Rev. No.	(E) Title or Description of Data Transmitted	Impact Level	Reason for Trans- mittal	Origi- nator Dispo- sition	Receiv- er Dispo- sition
1.	WHC-EM-SD-WM-TI-607 RCB 5-23-94		Draft	Radiological Implications of Contact Maintenance	4	1	1	1

16. KEY					
Impact Level (F)	Reason for Transmittal (G)			Disposition (H) & (I)	
1, 2, 3, or 4 (see MRP 5.43)	1. Approval	4. Review		1. Approved	4. Reviewed no/comment
	2. Release	5. Post-Review		2. Approved w/comment	5. Reviewed w/comment
	3. Information	6. Dist. (Receipt Acknow. Required)		3. Disapproved w/comment	6. Receipt acknowledged

17. SIGNATURE/DISTRIBUTION (See Impact Level for required signatures)										(G)	(H)
Reason	Disp.	(J) Name	(K) Signature	(L) Date	(M) MSIN	(J) Name	(K) Signature	(L) Date	(M) MSIN	Reason	Disp.
1	1	Cog. Eng. R.C. Brown	<i>[Signature]</i>	02/11/94	G6-06	W. C. Miller	<i>[Signature]</i>	5/15/94	S4-55	4	
1	1	Cog. Mgr. J.G. Propson	<i>[Signature]</i>	5/15/94	S4-55	G. F. Williamson	<i>[Signature]</i>	5/15/94	R4-01	4	
1	1	QA Jack Weber	<i>[Signature]</i>	5/15/94	S1-57	P. Felice	<i>[Signature]</i>	5/15/94	G6-06	4	
1	1	Safety G. D. Wright	<i>[Signature]</i>	5/15/94	G6-04	J. Greenberg	<i>[Signature]</i>	5/15/94	H0-35	4	
		Env. <i>[Signature]</i>	<i>[Signature]</i>	5/15/94	G6-06	G. E. Stegen	<i>[Signature]</i>	5/15/94	G6-06	4	
1	1	TWR/UTO R. E. Raymond	<i>[Signature]</i>	5/15/94	G6-16	D. A. Smith	<i>[Signature]</i>	5/15/94	G6-16	4	
1	1	TWR/EDO R. C. Roal	<i>[Signature]</i>	5/15/94	G6-50						

18. <i>[Signature]</i> 5/23/94 Signature of EDT Originator		19. <i>[Signature]</i> 5-23-94 Authorized Representative Date for Receiving Organization		20. <i>[Signature]</i> 5-23-94 Cognizant/Project Engineer's Manager		21. DOE APPROVAL (if required) Ltr. No. ☐ Approved ☐ Approved w/comments ☐ Disapproved w/comments	
--	--	--	--	---	--	--	--

Date Received: 02/14/94	INFORMATION RELEASE REQUEST	Reference: WHC-CM-3-4	
<div style="display: flex; justify-content: space-between;"> <div>MAY 23 1994</div> <div>Complete for all Types of Release</div> </div>			
Purpose <div style="display: flex; justify-content: space-between;"> <div style="width: 45%;"> ☐ Speech or Presentation ☐ Full Paper (Check only one suffix) ☐ Summary ☐ Abstract ☐ Visual Aid ☐ Speakers Bureau ☐ Poster Session ☐ Videotape </div> <div style="width: 45%;"> ☐ Reference ☒ Technical Report ☐ Thesis or Dissertation ☐ Manual ☐ Brochure/Flier ☐ Software/Database ☐ Controlled Document ☐ Other </div> </div>		ID Number (include revision, volume, etc.) WHC-SD-WM-TI-607, DRAFT List attachments. Document attached Date Release Required <div style="text-align: center;">March 10, 1994</div>	
Title Radiological Implications of Contact Maintenance: Impacts on Facility Design		Unclassified Category UC- Impact Level <div style="text-align: right;">4</div>	
New or novel (patentable) subject matter? ☒ No ☐ Yes If "Yes", has disclosure been submitted by WHC or other company? ☐ No ☐ Yes Disclosure No(s).		Information received from others in confidence, such as proprietary data, trade secrets, and/or inventions? ☒ No ☐ Yes (Identify)	
Copyrights? ☒ No ☐ Yes If "Yes", has written permission been granted? ☐ No ☐ Yes (Attach Permission)		Trademarks? ☒ No ☐ Yes (Identify)	
Complete for Speech or Presentation			
Title of Conference or Meeting N/A		Group or Society Sponsoring N/A	
Date(s) of Conference or Meeting	City/State	Will proceedings be published? ☐ Yes ☐ No Will material be handed out? ☐ Yes ☐ No	
Title of Journal			
CHECKLIST FOR SIGNATORIES			
Review Required per WHC-CM-3-4	Yes	No	Reviewer - Signature Indicates Approval
			Name (printed) Signature Date
Classification/Unclassified Controlled Nuclear Information	☐	☒	<div style="border: 1px solid black; padding: 5px;"> <i>[Signature]</i> 2/23/94 </div>
Patent - General Counsel	☒	☐	
Legal - General Counsel	☒	☐	
Applied Technology/Export Controlled Information or International Program	☐	☒	<div style="border: 1px solid black; padding: 5px;"> G. F. Williamson 2-23-94 R4-01 </div>
WHC Program/Project	☒	☐	
Communications	☒	☐	
RL Program/Project	☒	☐	<div style="border: 1px solid black; padding: 5px;"> L. A. Huffman 2-23-94 R3-74 </div>
Publication Services	☒	☐	
Other Program/Project	☐	☐	
Information conforms to all applicable requirements. The above information is certified to be correct.			
References Available to Intended Audience ☒ Yes ☐ No Transmit to DOE-HQ/Office of Scientific and Technical Information ☐ Yes ☒ No Author/Requestor (Printed/Signature) Date R. C. Brown <i>[Signature]</i> 02/14/94		INFORMATION RELEASE ADMINISTRATION APPROVAL STAMP Stamp is required before release. Release is contingent upon resolution of mandatory comments. <div style="text-align: center;"> </div>	
Intended Audience ☒ Internal ☒ Sponsor ☐ External Responsible Manager (Printed/Signature) Date J. G. Propson <i>[Signature]</i> 2/16/94		Date Cancelled Date Disapproved	

SUPPORTING DOCUMENT

1. Total Pages **31**

2. Title

Radiological Implications of Contact Maintenance:
Impacts on Facility Design

3. Number

WHC-SD-WM-TI-607

4. Rev No.

0

5. Key Words

contact, maintenance, radiological, impacts,
facility, design

6. Author

Name: R. C. Brown

R. C. Brown
Signature

Organization/Charge Code 7E850/608371

7. Abstract

Concentrations of ⁹⁰Sr, ¹³⁷Cs, and transuranics that allow contact maintenance to be performed are derived and specified in this document. These concentrations are provided as "trigger level" values, above which design options must be used to offset the adverse effects to personnel.

8. PURPOSE AND USE OF DOCUMENT - This document was prepared for use within the U.S. Department of Energy and its contractors. It is to be used only to perform, direct, or integrate work under U.S. Department of Energy contracts. This document is not approved for public release until reviewed.

PATENT STATUS - This document copy, since it is transmitted in advance of patent clearance, is made available in confidence solely for use in performance of work under contracts with the U.S. Department of Energy. This document is not to be published nor its contents otherwise disseminated or used for purposes other than specified above before patent approval for such release or use has been secured, upon request, from the Patent Counsel, U.S. Department of Energy Field Office, Richland, WA.

✓ DISCLAIMER - This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, nor any of their employees, nor any of their contractors, subcontractors or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or any third party's use or the results of such use of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise, does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof or its contractors or subcontractors. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

10.

RELEASE STAMP

9. Impact Level **4**

APPROVED FOR
PUBLIC RELEASE
EW 3-22-94

MASTER

875

CONTENTS

1.0	INTRODUCTION	1
1.1	PURPOSE	1
1.2	SCOPE	1
1.3	BACKGROUND	2
2.0	EXPOSURE RATES FROM ¹³⁷ Cs AND ⁹⁰ Sr	4
2.1	CONTRIBUTIONS FROM ¹³⁷ Cs IN 6.1-m LENGTHS OF PIPE	4
2.2	BREMSSTRAHLUNG PRODUCTION	4
2.3	CONTRIBUTIONS FROM ⁹⁰ Sr IN 6.1-m LENGTHS OF PIPE	6
2.4	CONTRIBUTIONS FROM FULL PROCESS TANKS WITH ⁹⁰ Sr AND ¹³⁷ Cs	6
2.5	CONTRIBUTIONS FROM THE HEEL IN EMPTY PROCESS TANKS	9
3.0	DISCUSSION OF RESULTS	9
3.1	LIMITS BASED ON PENETRATING RADIATION	11
3.1.1	Piping Runs	11
3.1.2	Process Tanks	12
3.2	LIMITS BASED ON INTERNAL DEPOSITION	12
3.3	TRIGGER LEVEL CONCENTRATIONS	14
4.0	CONCLUSIONS AND RECOMMENDATIONS	15
5.0	REFERENCES	15
6.0	GLOSSARY	16
	APPENDIX: CRITERIA RELATED TO CONTACT MAINTENANCE	A-1

LIST OF FIGURES

1	Bremsstrahlung from 1.0 Ci/L ^{90}Sr 6.1-m of U.S. Nominal 2-in. Schedule 40 Pipe	7
---	--	---

LIST OF TABLES

1	Exposure Rates from 6.1-m Lengths of Specified Pipe Filled with 1.0 Ci/L of ^{137}Cs Solution with a Density of 1.05 g/mL	5
2	Effect of Z_{eff} on Bremsstrahlung Production	6
3	Exposure Rates from 6.1-m Lengths of Specified Pipe Filled with 1.0 Ci/L of ^{90}Sr Solution with a Density of 1.05 g/mL	8
4	Exposure Rates 91 cm from the Side and 91 cm from the Bottom of Full Process Tanks (2.4 m in Height) Containing 1.0 Ci/L of ^{90}Sr or ^{137}Cs	9
5	Exposure Rates 91 cm above the Bottom and at Specified Distances from the Side of an Empty Process Tank (2.4 m in Height) with a 1% Heel	10
6	Concentrations to Trigger Offsetting Design Features	14

RADIOLOGICAL IMPLICATIONS OF CONTACT MAINTENANCE: IMPACTS ON FACILITY DESIGN

1.0 INTRODUCTION

1.1 PURPOSE

The purpose of this document is to provide an initial, rough order of magnitude, evaluation of the relationship between the concentrations of selected radionuclides in process streams of a chemical process plant and the required degree of protective features for plant operations personnel. The evaluation provided in this document specifies "trigger level" concentrations of ^{90}Sr , ^{137}Cs , and transuranics (TRU) for "contact maintenance" activities. Concentrations above the "trigger levels" can be accommodated provided offsetting design features are incorporated (e.g., improved shielding partial to total confinement, etc.). This information will assist in defining the degree of reduction of concentrations of radionuclides required for the feed used in a low-level liquid waste processing plant to utilize a contact maintenance design concept. Offsetting design features that influence maintenance personnel exposures in a contact maintenance situation are identified. All concentrations are expressed in terms of actual process fluids before any flushing and decontamination steps.

Changes to equipment and facility design concepts, maintenance concepts, and confinement system concepts can affect the concentration limits for process liquids in a contact maintenance design concept. Thus, a more detailed analysis will be required to fully define acceptable concentration limits for all the radionuclides in process streams for a facility when the configuration can be specified. An iterative process is recommended to achieve a suitable balance between feed characteristics and facility design for the degree of "contact maintenance" required.

1.2 SCOPE

To properly evaluate the degree of compatibility between low-level liquid radioactive waste and contact maintenance limitations, the following items will be considered:

- Criteria and additional information related to contact maintenance (see the Appendix)
- Exposure rates from ^{137}Cs and ^{90}Sr , each in equilibrium with daughters, at unit concentrations of 1 Ci/L in 6.1-m runs of various sized pipes and nominal-sized tanks
- Estimates of upper concentration limits based on underlying assumptions about facility configuration and work conditions

- Design offset suggestions that can be employed to mitigate the effects of higher concentrations of radionuclides in process systems
- Conclusions and recommendations.

1.3 BACKGROUND

The newly revised *Hanford Federal Facility Agreement and Consent Order* (Ecology et al. 1994), also known as the Tri-Party Agreement, calls for vitrification of low-level liquid wastes to replace the previous technology, which was to incorporate the material in a grout matrix placed in appropriate vaults. Unlike the high-level vitrification plant that was to be remotely maintained and operated, the low-level vitrification plant may consider contact maintenance as a design option.

An operating vitrification plant presents a plethora of piping, pumps, agitators, vessels, valves, evaporators, and other process equipment. The glass melter may be physically large and could have a large inventory of glass contained within. To perform contact maintenance in such a facility each configuration would have to be analyzed for the following factors:

- Species and concentrations of radioactive materials in process vessels, piping, etc.
- Configuration of process equipment
- Personnel exposure limitation criteria.

Because the first two factors noted above require a detailed knowledge of the process and facility layout, only general advice for limited configurations is presented in this document. This advice is limited to the following conditions.

- The radionuclides are limited to ^{90}Sr , ^{137}Cs , and TRUs.
- Piping configurations are limited to 6.1 m (20 ft) of piping of specified sizes and distances.
- Tank configurations are limited to full tanks and empty tanks with a specified heel.

Considerations for performing contact maintenance on the melter are not considered in this document. There is not sufficient information available at this time to estimate the type of melter, configuration, and other details necessary to estimate personnel exposure rates.

Selection of ^{90}Sr and ^{137}Cs for determining upper limits of concentrations of process streams is based on the ubiquitous nature of these radionuclides in high-level waste. Many low-level wastes (LLW) contain these radionuclides as well. Those LLWs that contain ^{90}Sr and ^{137}Cs in concentrations so low as to not be of concern as compared with the majority of radionuclides will have to be evaluated separately.

The TRU elements, particularly plutonium and americium, require special consideration for contact maintenance. The TRUs are particularly radiotoxic, and engineered barriers are necessary to prevent them from becoming airborne with a resultant inhalation hazard to maintenance and operations personnel. Many LLW streams may contain TRUs in quantities that pose potential personnel hazards if they become unconfined.

Operations involving contact handling of radioactive materials or equipment contaminated with radioactive materials contribute to personnel exposure. A study conducted by Science Applications International Corporation for the Hanford Waste Vitrification Plant Project addressed contact maintenance activities. One of the "lessons learned" from this study was that consideration of personnel exposure from these activities is essential during the conceptual design of a facility. Estimates of personnel exposures from contact maintenance and operations tasks can be evaluated in conjunction with design modifications that would reduce these exposures. During this phase of the design, changes are easily accommodated. After the design is fixed (i.e., detailed design), changes are difficult to incorporate and usually very expensive. Thus, an iterative process in which design options are tested against personnel exposure estimates is recommended during the conceptual design phase of a project.

Several regulatory and health physics concerns arise from the consideration of contact maintenance. Among these are the following.

- The new Radiological Control Manual attached to DOE N 5480.6, *Radiological Control* (DOE 1992), and the Westinghouse Hanford Company implementation of that document via WHC-CM-1-6, *Westinghouse Hanford Company Radiological Control Manual*, impose lower occupational exposure limits of 500 mrem/year. These documents also impose a 500-mrem/year design criterion for new facilities, which is 50% of the design criterion in DOE Order 6430.1A, *General Design Criteria* (DOE 1989).
- Skin and extremity maximum annual dose equivalent is 15,000 mrem (WHC-CM-1-6).
- The above annual dose limits place restrictions on exposure rates, both from external penetrating radiation and sources of radioactive material ingested into the body (primarily via inhalation), that can be tolerated in areas of a facility occupied by personnel.
- Maintenance operations often involve breaking open confinement systems. This results in an increased potential for internal depositions of radioactive materials by maintenance personnel (primarily by inhalation of airborne contaminants). As a result the amount of radioactive material that can be safely released outside of confinement is limited. Engineered controls are preferred over personal respiratory protective devices in this situation (DOE 1989, WHC-CM-1-6).

These factors will be utilized to estimate "trigger level" concentrations of ⁹⁰Sr and ¹³⁷Cs in liquid process systems.

2.0 EXPOSURE RATES FROM ^{137}Cs AND ^{90}Sr

This section contains a series of calculated exposure rates from ^{137}Cs and ^{90}Sr in specified configurations. These configurations were selected to represent conditions that may be present in a LLW processing facility.

2.1 CONTRIBUTIONS FROM ^{137}Cs IN 6.1-m LENGTHS OF PIPE

Exposure rate contributions at specified distances from the center of 6.1-m lengths of various sized pipes filled with 1.0 Ci/L of ^{137}Cs are presented in Table 1.

2.2 BREMSSTRAHLUNG PRODUCTION

The electromagnetic radiation (x rays) produced by slowing down electrons in a medium is called bremsstrahlung. Beta emitters can be sources of bremsstrahlung. To minimize the production of bremsstrahlung, the effective atomic number (Z_{eff}) should be kept as low as possible.

The manner in which Z_{eff} is calculated is as follows (Jaeger et al. 1968):

$$Z_{\text{eff}} = \frac{\sum_{i=1}^k \frac{w_i Z_i}{(MW)_i}}{\sum_{i=1}^k \frac{w_i}{(MW)_i}}$$

where

Z_{eff} = The effective atomic number of the mixture

w_i = The weight fraction of the " i^{th} " component

Z_i = Atomic number of the " i^{th} " component

MW_i = Molecular weight of the " i^{th} " component

k = Number of components in the mixture.

In process piping systems, the effect of the piping wall material is negligible but Z_{eff} of the solution is controlling in the production of bremsstrahlung; Table 2 shows this effect. The data from Table 2 are plotted in Figure 1. For purposes of this study, bremsstrahlung production will be calculated with $Z_{\text{eff}} = 10$.

Table 1. Exposure Rates from 6.1-m Lengths of Specified Pipe Filled with 1.0 Ci/L of ^{137}Cs Solution with a Density of 1.05 g/mL.

Nominal U.S. pipe size (in.)	ID ^a (in.)	Wall thickness ^a (in.)	Exposure rate at specified distance ^b			
			Contact (mR/h)	5 cm (mR/h)	30 cm (mR/h)	91 cm (mR/h)
Schedule 40 pipe						
0.5	0.622	0.109	16,000	2,600	490	150
1.0	1.049	0.133	29,000	6,500	1,300	400
1.5	1.610	0.145	46,000	13,200	2,900	900
2	2.067	0.154	59,000	19,500	4,500	1,400
4	4.026	0.237	96,000	44,800	12,800	4,400
6	6.065	0.280	120,000	68,000	22,600	8,300
Schedule 80 pipe						
0.5	0.546	0.147	12,000	2,000	370	110
1.0	0.957	0.179	22,000	5,200	1,100	320
1.5	1.500	0.200	37,000	10,900	2,400	760
2	1.939	0.218	48,000	16,100	3,700	1,200
4	3.826	0.337	75,000	31,900	9,300	3,300
6	5.761	0.432	88,000	51,100	17,200	6,400

^aBaumeister, T., editor, 1958, *Mechanical Engineers' Handbook*, McGraw-Hill Book Company, Inc., New York, New York.

^bExposure rates were determined with the Microshield shielding program. (Microshield is a trademark of Grove Engineering, Inc.)

To convert inches to centimeters, multiply by 2.54.

ID = Inner diameter

Table 2. Effect of Z_{eff} on Bremsstrahlung Production.^a

Z_{eff}	Material	Exposure rate	
		At contact (mR/h)	At 5 cm (mR/h)
4	Water	580 ^b	200 ^b
6	Carbon	590 ^b	200 ^b
7	Air	590 ^b	200 ^b
10	Ordinary concrete	600	200
12	Magnetite concrete	670	230
13	Aluminum	730	250
22	Titanium	1,100	390
26	Iron	1,300	440

^aExposure rate at contact and 5 cm from a 6.1-m length of nominal 2-in. diameter Schedule 40 pipe filled with 1.0 Ci/L of material with a density of 1.05 g/mL and Z_{eff} as indicated. Exposure rates calculated with ISOSHLD (see Laurant, I. O., 1990, *Validation of ISOSHLD-PL on IBM PC CPS-2/AT/XT*, WHC-SD-SQA-LSWD-303, Westinghouse Hanford Company, Richland, Washington).

^bThis appears to be an artifact in ISOSHLD. Theory predicts that bremsstrahlung is directly proportional to Z .

2.3 CONTRIBUTIONS FROM ⁹⁰Sr IN 6.1-m LENGTHS OF PIPE

Bremsstrahlung exposure rate contributions at specified distances from the center of 6.1-m lengths of various sized pipes filled with 1.0 Ci/L of ⁹⁰Sr are presented in Table 3. It should be noted that exposures are approximately 1% of those contained in Table 1 and that contact exposure rates are approximately a factor of 3 higher than those at 5 cm. As long as contact dose rates are less than a factor of 30 of whole body dose rates, extremity exposure limits will not be exceeded.

2.4 CONTRIBUTIONS FROM FULL PROCESS TANKS WITH ⁹⁰Sr AND ¹³⁷Cs

Process tanks full of solution provide sources of radiation exposure. Exposure rates from several different diameter tanks full of ⁹⁰Sr and ¹³⁷Cs solutions are presented in Table 4.

Figure 1. Bremsstrahlung from 1.0 Ci/L ^{90}Sr
6.1-m of U.S. Nominal 2-in. Schedule 40 Pipe.

Table 3. Exposure Rates from 6.1-m Lengths of Specified Pipe Filled with 1.0 Ci/L of ^{90}Sr Solution with a Density of 1.05 g/mL.

Nominal U.S. pipe size (in.)	ID ^a (in.)	Wall thickness ^a (in.)	Exposure rate at specified distance ^b			
			Contact (mR/h)	5 cm (mR/h)	30 cm (mR/h)	91 cm (mR/h)
Schedule 40 pipe						
0.5	0.622	0.109	180	30	6	2
1.0	1.049	0.133	300	70	14	5
1.5	1.610	0.145	450	140	31	10
2	2.067	0.154	600	200	47	16
4	4.026	0.237	850	410	120	42
6	6.065	0.280	1,000	580	190	74
Schedule 80 pipe						
0.5	0.546	0.147	170	21	4	1
1.0	0.957	0.179	230	55	11	4
1.5	1.500	0.200	350	110	24	8
2	1.939	0.218	440	160	37	13
4	3.826	0.337	620	310	89	33
6	5.761	0.432	690	400	140	52

^aBaumeister, T., editor, 1958, *Mechanical Engineers' Handbook*, McGraw-Hill Book Company, Inc., New York, New York.

^bExposure rates were determined with the ISOSHLD (see Laurant, I. O., 1990, *Validation of ISOSHLD-PL on IBM PC CPS-2/AT/XT*, WHC-SD-SQA-LSWD-303, Westinghouse Hanford Company, Richland, Washington) shielding program with bremsstrahlung production only. The effective atomic number of the solution was set equal to that of ordinary concrete, $Z_{\text{eff}} = 10$.

To convert inches to centimeters, multiply by 2.54.

ID = Inner diameter

Table 4. Exposure Rates 91 cm from the Side and 91 cm from the Bottom of Full Process Tanks (2.4 m in Height) Containing 1.0 Ci/L of ^{90}Sr or ^{137}Cs .^a

OD (cm)	Capacity (L)	Exposure rate ^{90}Sr tank (mR/h)	Exposure rate ^{137}Cs tank (mR/h)
61	720	200	32,900
122	2,800	340	58,000
183	6,400	420	75,500
244	11,000	490	87,200
305	18,000	540	96,300
366	26,000	580	105,000

^aNOTE: Tanks are assumed to have 1.27-cm steel sides and are filled with a material of effective atomic number (Z_{eff}) of 10 with a density of 1.05 g/mL. Exposure rates are calculated with ISOSHL (see Laurant, I. O., 1990, *Validation of ISOSHL-PL on IBM PC CPS-2/AT/XT*, WHC-SD-SQA-LSWD-303, Westinghouse Hanford Company, Richland, Washington) and Microshield for the ^{90}Sr and ^{137}Cs , respectively. (Microshield is a trademark of Grove Engineering, Inc.)

OD = Outer diameter

2.5 CONTRIBUTIONS FROM THE HEEL IN EMPTY PROCESS TANKS

When full process tanks are emptied, some heel remains in the bottom. For purposes of these calculations, this heel is assumed to contain 1% of the total activity contained in the full tank. This activity is assumed to be uniformly distributed across the bottom of the tank. Exposure rates under these conditions for ^{90}Sr and ^{137}Cs heels are presented in Table 5.

3.0 DISCUSSION OF RESULTS

Concentration limits of radionuclides for process liquids that allow contact maintenance with minimum engineering controls can be estimated based on the material developed in Section 2.0 and the Appendix. These limits are based on assumptions as to the type of personnel protection provided. Preparation of systems (i.e., flushing, decontamination, etc.) for maintenance activities can greatly reduce the radiological hazards presented to maintenance personnel. The greater the degree of engineered barriers and flushing/decontamination activities for personnel protection, the greater the concentration limit for process liquids in the facility. Limits based on penetrating (external) radiation levels and internal deposition hazards will be developed.

Table 5. Exposure Rates 91 cm above the Bottom and at Specified Distances from the Side of an Empty Process Tank (2.4 m in Height) with a 1% Heel.

Tank OD (cm)	Nominal activity in heel ^a (Ci)	Exposure rate at specified distance from side of tank ^b	
		5 cm (mR/h)	91 cm (mR/h)
Heel contains ⁹⁰ Sr spread over bottom			
61	7	6	4
122	28	32	13
183	63	57	21
244	112	80	29
305	175	96	36
366	252	110	40
Heel contains ¹³⁷ Cs spread over bottom			
61	7	920	600
122	28	4,300	1,800
183	63	7,500	3,100
244	112	10,400	4,400
305	175	12,800	5,700
366	252	14,800	6,900

^aThe heel contains approximately 1% of the total activity in the tank when filled with 1.0 Ci/L of the specified radionuclide.

^bExposure rates approximated with equivalent geometries from ISOSHLD (see Laurant, I. O., 1990, *Validation of ISOSHLD-PL on IBM PC CPS-2/AT/XT*, WHC-SD-SQA-LSWD-303, Westinghouse Hanford Company, Richland, Washington) and Microshield computer programs for ⁹⁰Sr and ¹³⁷Cs, respectively. (Microshield is a trademark of Grove Engineering, Inc.)

OD = Outer diameter

Incorporation of appropriate design/operating features can offset deleterious aspects of maintenance operations. These features should be specified and incorporated into the design wherever possible.

3.1 LIMITS BASED ON PENETRATING RADIATION

3.1.1 Piping Runs

The exposure limits contained in WHC-CM-4-9, *Radiological Design*, are based on a 1,000 mrem/year design goal. Because the new DOE N 5480.6, *Radiological Control* (DOE 1992) design limit is 500 mrem/year, the values listed in WHC-CM-4-9 for external exposure (shielding) should be reduced to 50% of the values presently promulgated. This reduction to approximately 250 mrem/year provides a margin that allows for contact maintenance and accumulation of contamination within the facility and resultant increase in exposure rates.

Based on the above, the following assumption is invoked: the design limit for continuous occupancy in a controlled access area is 0.1 mrem/h. Thus, the exposure rate at 91 cm from a piping run can be calculated from a given concentration of process fluid within the pipe. Using a U.S. nominal 2-in. Schedule 80 pipe as a typical piping run for process fluids, and the data from Tables 1 and 3, results in the following limits:

- $^{90}\text{Sr}--8 \times 10^{-3} \text{ Ci/L}$
- $^{137}\text{Cs}--8 \times 10^{-5} \text{ Ci/L}$.

Higher limits can be accommodated provided personnel access is restricted, distance from areas normally occupied by personnel is increased, or shielding is employed.

Contact dose rates should not exceed 1,500 mrem/h or 50 mrem/h at 5 cm based on criteria in WHC-CM-1-6, *Westinghouse Hanford Company Radiological Control Manual*, and WHC-CM-4-9 with process fluid in the pipe. Concentrations based on the data in Tables 1 and 3 for U.S. nominal 2-in. Schedule 80 pipe results in the following limits:

- $^{90}\text{Sr}--3 \times 10^{-1} \text{ Ci/L}$
- $^{137}\text{Cs}--3 \times 10^{-3} \text{ Ci/L}$.

This indicates that the concentration, based on the continuous occupancy limit at 91 cm, would be controlling. In the event that continuous occupancy is not required, the limits could be raised in proportion to the reduction in access time (i.e., 1 hour/day occupancy → increase the exposure rate criterion to 1 mR/h). Shielding could be provided to reduce exposure rates; increased distance between personnel and piping locations also will reduce exposure rates. These design factors would allow proportionate increase in the concentration of these radionuclides in the process fluid.

If more than one pipe is present, it would be necessary to reduce the concentration limits roughly in an inverse proportion to the number of pipes present (i.e., three parallel piping runs with similar fluid → decrease the

exposure rate to 0.03 mR/h from each pipe). Also, the presence of ^{90}Sr and ^{137}Cs in the process fluid would further reduce the limits for each in accordance with the following "unity" rule:

$$\frac{(\text{Conc}^{90}\text{Sr})}{(8 \times 10^{-3})} + \frac{(\text{Conc}^{137}\text{Cs})}{(8 \times 10^{-5})} \leq 1$$

3.1.2 Process Tanks

If continuous personnel access to within 91 cm of a nominally full process tank is required, the following concentration limits (based on 0.1 mrem/h and the exposure rates in Table 4) would apply:

- ^{90}Sr $5 \times 10^{-6} \rightarrow 1 \times 10^{-6}$ Ci/L for 720 $\rightarrow$ 26,000-L (60-370 cm outer diameter and 2.4 m in height) tanks, respectively
- ^{137}Cs $3 \times 10^{-6} \rightarrow 1 \times 10^{-6}$ Ci/L for 720 $\rightarrow$ 26,000-L (60-370 cm outer diameter and 2.4 m in height) tanks, respectively.

NOTE: This evaluation does not imply or recommend contact with full process tanks; any operations involving contact with a tank should be performed with the tank empty and flushed. Shielding should be provided to protect personnel who require continuous access in the vicinity of full process tanks. Increasing the distance from the tanks to areas where personnel require continuous access also is recommended. Each of these steps would allow higher concentrations of process fluids to be contained in the tanks.

Two assumptions are invoked for contact operations involving process tanks:

- Tanks normally would be emptied and flushed before any contact by Operations or Maintenance personnel would be permitted
- A heel uniformly distributed on the bottom of the tank with a nominal activity of 1% of the total tank activity (when filled) would exist.

The following concentration limits of the normal tank contents (before flushing) are based on the more restrictive 0.1 mrem/h at 91 cm (i.e., the contact dose rate is <50 mrem/h under the conditions assumed):

- ^{90}Sr $2 \times 10^{-2} \rightarrow 2 \times 10^{-3}$ Ci/L for 190 $\rightarrow$ 26,000-L tanks, respectively.
- ^{137}Cs $1 \times 10^{-4} \rightarrow 1 \times 10^{-5}$ Ci/L for 190 $\rightarrow$ 26,000-L tanks, respectively.

3.2 LIMITS BASED ON INTERNAL DEPOSITION

Internal deposition limits apply when process fluids become unconfined (i.e., when process lines are intentionally opened or leakage of process

fluids occurs exposing them to the operating environment). There are two primary mechanisms by which the radioactive materials contained in the process liquid become airborne: (1) the liquid is directly atomized and the resulting aerosol is inhaled, and (2) the liquid evaporates leaving a film, which when disturbed, may cause small particles to become airborne and then inhaled. This latter process is called resuspension.

The governing limits for inhalation are the annual limit on intake (ALI) from which the derived air concentration (DAC) is calculated. For example, the ALIs for ^{90}Sr , ^{137}Cs , and ^{239}Pu are approximately 2×10^{-5} Ci, 2×10^{-4} Ci, and 5×10^{-9} Ci, respectively. Thus, surface contamination can lead to airborne radioactive materials through the resuspension process.

Conservative recommendations are provided in WHC-CM-4-9, *Radiological Design*, for the limits total activity presents as an unconfined source before airborne concentrations of radioactive materials become high enough to be of concern for an inhalation hazard. The WHC-CM-4-9 limits will be used to provide "trigger-level" action levels rather than trying to estimate airborne concentrations directly. Two assumptions are invoked to determine these limits.

- Before any process lines are opened for maintenance, they will be flushed and decontaminated. Any fluid remaining in the line would have a concentration of 1% of the concentration of the original process fluid.

NOTE: Any scale or crud remaining after this flushing and decontamination may contribute to penetrating exposure rates and internal deposition. No estimates of this component are provided here. Such an estimate is beyond the scope of this document.

- The volume of fluid released from the process system is assumed to be ≤ 1.0 L.

The activity limits in WHC-CM-4-9, *Radiological Design*, are $\leq 10^{-6}$, 10^{-5} , and 10^{-8} Ci for ^{90}Sr , ^{137}Cs , and TRUs, respectively, when the material is unconfined. These activity limits result in the following concentration limits when 1 L of liquid at a concentration of 1% of the process fluid is released to an unconfined environment, i.e., no glove box or hood:

- 1×10^{-4} Ci/L for ^{90}Sr
- 1×10^{-3} Ci/L for ^{137}Cs
- 1×10^{-6} Ci/L for TRUs.

Again, the unity rule would apply for liquids containing both of the radionuclides. Greater volumes of liquids and/or different decontamination factors can be applied if better data concerning these factors are available.

*NOTE: The concentration of the remaining fluid is assumed to be 1% of the original process fluid for purposes of this example. Proven or generally accepted values should be used if available.

Offsetting features that would accommodate higher concentrations in the process fluids include partial confinement devices (air flow velocities of 30 ± 8 m/min), total confinement (glove boxes), better decontamination factors in the process system, and washdown systems that would flush and thereby dilute any spilled liquids to radioactive drains.

3.3 TRIGGER LEVEL CONCENTRATIONS

From the above considerations, the overall "trigger level" concentration limit for process fluids as determined by the penetrating and internal deposition limits for ^{90}Sr and ^{137}Cs are provided in Table 6.

Process solutions at concentrations above these "trigger levels" can be accommodated by providing offsetting design features. Limits imposed by penetrating (external) radiation considerations can be offset by decreasing occupancy time, increasing the distance between personnel and the source(s), and providing shielding. Limits imposed by internal dose considerations can be offset by improving confinement (improved air flow characteristics or glove boxes), increasing the efficacy of the decontamination process used in cleaning out systems before opening the system, and reducing the amount of radioactive material present after opening the system (e.g., flushing away the "spilled" fluids).

Table 6. Concentrations to Trigger Offsetting Design Features.

Radioactive species	Trigger level (Ci/L)	Basis	Offsetting design features
^{90}Sr	$>1 \times 10^{-4}$	Penetrating (full tank)	Penetrating → time, distance, and shielding
	$>1 \times 10^{-4}$	Internal	Internal → partial to total confinement, improved DFs, flushing area after opening system
^{137}Cs	$>1 \times 10^{-6}$	Penetrating (full tank)	Penetrating → time, distance, and shielding
	$>1 \times 10^{-3}$	Internal	Internal → partial to total confinement, improved DFs, flushing area after opening system
TRU	$>1 \times 10^{-6}$	Internal	Partial to total confinement, improved DFs, flushing area after opening system

DF = Decontamination factor
TRU = Transuranic

4.0 CONCLUSIONS AND RECOMMENDATIONS

The radiological consequences from operations with low-level liquid waste have been reviewed with respect to external and internal radiation hazards from ^{90}Sr , ^{137}Cs , and TRU. Applicable design guidance has been reviewed to identify key requirements and factors that affect estimates of "trigger level" concentrations of ^{90}Sr , ^{137}Cs , and TRU that could be accommodated in a contact maintained facility without offsetting design features. Implementation of the following design recommendations would result in a contact maintained facility that would have a high probability of operating within the new administrative dose limits of 500 mrem/year. Recommendations that should be considered are as follows.

1. Respiratory protection cannot be considered as part of a design for contact maintained systems (i.e., to meet routine and upset operational maintenance tasks).
2. The effective atomic number, Z_{eff} , of any contaminated process solutions should be kept as low as practical (preferably below $Z_{\text{eff}} \leq 13$) to minimize bremsstrahlung production from beta emitters.
3. The shielding design criterion should be 50% of those presently recommended in WHC-CM-4-9, *Radiological Design*, i.e., 200-250 mrem/year. The more stringent guidance provided in WHC-CM-1-6, *Westinghouse Hanford Company Radiological Control Manual*, and DOE N 5480.6, *Radiological Control* (DOE 1992) for design and design evaluation is the basis of this recommendation.
4. The DOE Order 6430.1A, *General Design Criteria* (DOE 1989), cited in the Appendix, should be followed.
5. Trigger level concentration levels of process fluids contaminated with ^{90}Sr , ^{137}Cs , and TRU provided in Table 6 should be used to determine when to consider offsetting design features.
6. A detailed evaluation of the facility and equipment configuration, maintenance operations, and anticipated process fluid concentrations should be performed during conceptual design to verify compliance with the personnel exposure limits in DOE N 5480.6.

These recommendations are intended to provide an initial basis for consideration of contact maintenance in a low-level liquid waste treatment facility where ^{90}Sr and ^{137}Cs are the radionuclides of concern. Other radionuclides may impact contact maintenance operations. The impact of these other radionuclides should be considered during conceptual design.

5.0 REFERENCES

Baumeister, T., editor, 1958, *Mechanical Engineers' Handbook*, McGraw-Hill Book Company, Inc., New York, New York.

- Ecology, EPA, and DOE, 1994, *Hanford Federal Facility Agreement and Consent Order*, as amended, Washington State Department of Ecology, U.S. Environmental Protection Agency, and U.S. Department of Energy, Olympia, Washington.
- DOE, 1989, *General Design Criteria*, DOE Order 6430.1A, U.S. Department of Energy, Washington, D.C.
- DOE, 1992, *Radiological Control*, DOE N 5480.6, U.S. Department of Energy, Washington, D.C.
- Jaeger, R. G., E. P. Blizard, A. B. Chilton, M. Grotenhuis, A. Hönig, T. A. Jaeger, and H. H. Eisenlohr, editors, 1968, *Engineering Compendium on Radiation Shielding, Volume 1, Shielding Fundamentals and Methods*, Springer-Verlag, New York, New York.
- Lourant, I. O., 1990, *Validation of ISOSHLD-PL on IBM PC CPS-2/AT/XT*, WHC-SD-SQA-LSWD-303, Westinghouse Hanford Company, Richland, Washington.
- WHC-CM-1-6, *Westinghouse Hanford Company Radiological Control Manual*, Westinghouse Hanford Company, Richland, Washington.
- WHC-CM-4-9, *Radiological Design*, Westinghouse Hanford Company, Richland, Washington.

6.0 GLOSSARY

ABBREVIATIONS AND ACRONYMS

ALI	annual limit on intake
DAC	derived air concentration
DF	decontamination factor
ID	inner diameter
LLW	low-level waste
OD	outer diameter
TRU	transuranic

APPENDIX

CRITERIA RELATED TO CONTACT MAINTENANCE

This page intentionally left blank.

CONTENTS

A1.0	RELATED CRITERIA	A-5
A1.1	RADIOLOGICAL CONTROL	A-5
A1.1.1	Section 128, Radiological Design	A-5
A1.1.2	Section 531, Respiratory Protection	A-5
A1.2	GENERAL DESIGN CRITERIA	A-6
A1.2.1	General Requirements	A-6
A1.2.2	Equipment	A-6
A1.2.3	Special Facilities	A-7
A1.2.4	Mechanical	A-9
A1.2.5	Comments	A-9
A1.3	WESTINGHOUSE HANFORD COMPANY CRITERIA	A-9
A1.3.1	External Exposure Rate Limitations	A-9
A1.3.2	Confinement Device Recommendations	A-10
A1.3.3	Summary	A-10
A1.4	RELATED INFORMATION ON CONTACT MAINTENANCE	A-11
A2.0	REFERENCES	A-11
A3.0	GLOSSARY	A-12

This page intentionally left blank.

APPENDIX**CRITERIA RELATED TO CONTACT MAINTENANCE****A1.0 RELATED CRITERIA****A1.1 RADIOLOGICAL CONTROL (DOE N 5480.6)**

The U.S. Department of Energy (DOE) states in Section 111 of DOE N 5480.6, *Radiological Control* (DOE 1992), "The Department of Energy is firmly committed to having a Radiological Control Program of the highest quality. This applies to those DOE activities that manage radiation and radioactive materials and that may potentially result in radiation exposure to workers, the public and environment."

Several requirements from this manual that are pertinent to radiological work conditions and contact maintenance in particular are quoted below.

A1.1.1 Section 128, Radiological Design

- Individual worker dose should be as low as reasonably achievable (ALARA) and <500 mrem/year.
- Control of contamination should be achieved by containment (*sic*) of radioactive material.**
- Efficiency of maintenance, decontamination, and operations should be maximized.
- Components should be selected to minimize the buildup of radioactivity.

A1.1.2 Section 531, Respiratory Protection

- Use of respiratory protection shall be reduced to the minimum practicable by implementing engineering controls and work practices to contain radioactivity at the source.
- Facility safety analyses should not be credited for the use of respiratory protection for routine work involving potential exposure to airborne radioactive materials. Engineering controls should be

**Containment should be achieved by use of a confinement system, i.e., the barrier and its associated systems (including ventilation), between areas containing hazardous materials and the environment or other areas in the facility that are normally expected to have levels of hazardous materials lower than allowable concentration limits.

designed to control radioactive materials at the source so that respiratory protection can be reduced.

The intent of these requirements is clear: routine operations that require respiratory protection for personnel are to be avoided as a design option.

A1.2 GENERAL DESIGN CRITERIA (DOE ORDER 6430.1A)

A review of DOE Order 6430.1A, *General Design Criteria* (DOE 1989), to identify requirements on maintenance revealed the following, which have been screened for their possible applicability to a low-level liquid radioactive waste treatment facility.

A1.2.1 General Requirements

- Section 0110-9, Operation, Maintenance, Repair, and Replacement. "Planning and design of buildings and other structures, including their operating components and systems, shall take into account all aspects of maintenance including: Personnel safety. . . ."
- Section 0110-99.0, Special Facilities, Nonreactor Nuclear Facilities - General

Subsection 0110-99.0.2. Building Services and Distribution (paragraph 6). "Special facilities services shall extend from horizontal service headers. Services shall be located to avoid penetration of adjacent facility walls and floors in those cases where routine maintenance or alterations of these services would result in undesirable curtailment or interruptions of operations in the adjacent special or other facilities. All piping shall be located outside hazardous areas whenever possible to reduce personnel exposure during maintenance."

Subsection 0110-99.0.4. Building Layout. ". . . The arrangement and location of process equipment and its maintenance provisions shall ensure that exposure to radiation and other hazardous material is within the requirements of DOE 5480.11 and DOE 5480.10, respectively. In addition, exposures shall be maintained ALARA. The design shall protect workers sufficiently from hazards to ensure that workers can perform actions required during normal operations, anticipated operational occurrences, and postulated DBAs. . . ."

A1.2.2 Equipment

- Section 1161, Enclosures

Subsection 1161-1. General Considerations. "Where practical and without penetrating the enclosure, all equipment components not functionally required to operate directly in the presence of radioactive materials shall be located outside the enclosure."

All equipment that must be located within the enclosure shall be designed to allow for in-place maintenance and/or replacement."

Subsection 1161-2. Construction. "Glove ports shall be located to facilitate both operations and maintenance work inside the enclosure. They shall have flexible gloves attached to allow operating personnel access to all interior surfaces and equipment. They shall be designed to allow replacement of gloves without losing contamination control and with minimum exposure to the operator. When gloves are not in place, a noncombustible shield or cover for each glove port shall be provided."

A1.2.3 Special Facilities

- Section 1300-3, Safety Class Criteria

Subsection 1300-3.5. Maintenance (paragraph 1). "The design shall consider the maintainability factors peculiar to the specific equipment to be used in the facility. Facility design shall provide for routine maintenance, repair, or replacement of equipment subject to failure."

Subsection 1300-3.5. Maintenance (paragraph 3). "The design of equipment that must be located within confinement systems shall allow for in-place maintenance or replacement."

Subsection 1300-3.5. Maintenance (paragraph 4). "The capability shall be provided for the maintenance of contaminated equipment that cannot be repaired in place. This capability shall include the necessary provisions for confinement, ventilation, and waste control."

Subsection 1300-3.5. Maintenance (paragraph 5). "The design of all process equipment shall include features to minimize self-contamination of the equipment, piping, and confinement areas. The design of process equipment shall also include features to minimize the spread of contamination out of the local areas."

- Section 1300-6, Radiation Protection

Subsection 1300-6.2. Shielding Design. "Shielding and other radiation protection measures shall be provided for areas requiring intermittent access, such as for preventive maintenance, component changes, adjustment of systems and equipment, and so forth. The projected dose rates based on occupancy, time, and frequency of exposure shall not exceed 1 rem/y."

Subsection 1300-6.9. Breathing Air System. "Operation and maintenance of special facilities may lead to situations (e.g., accidents, special maintenance, spill recovery) where air-supplied

*NOTE: DOE N 5480.6 now requires a reduction to 500 mrem/year.

respiratory protection is required. Breathing air supply systems shall comply with ANSI Z88.2 and 29 CFR 1910.134."

- Section 1300-7, Confinement Systems

Subsection 1300-7.2. General (paragraph 7). "Design of confinement areas shall provide adequate means for decontamination of the areas prior to entry or breaching for maintenance and repair purposes."

- Section 1300-12, Human Factors Engineering

Subsection 1300-12.1. Coverage. "It is DOE policy to ensure that appropriate human factors technology is considered in the design, operation, and maintenance of Departmental nonreactor nuclear facilities. The criteria and requirements provided in this section are applicable to the design of the work environment and human-machine systems at DOE facilities. These criteria shall apply to new construction and to retrofitting of existing facilities. These criteria shall be considered for upgrading existing facilities where cost-benefit or risk-tradeoff analyses indicate justification for such expenditures."

Subsection 1300-12.2. Objectives (paragraph 2). "Enhancements to the work environment and human-machine interfaces will reduce human error and its consequences and lead to increased productivity, lower costs, better product quality, decreased equipment and property damage, improved program schedules, personal job satisfaction, and, perhaps more important, to further improvements in the safe operation and maintenance of DOE facilities."

- Section 1304, Plutonium Processing and Handling Facilities

Subsection 1304-4. Radiation Protection (paragraph 2). "Facility design shall provide for the continuous monitoring of external radiation exposure levels in process areas such as hot cells and canyons during entries required for maintenance or repair operations."

Subsection 1304-6.2. Primary Confinement System (paragraph 5). "Occasional breaches that are required for anticipated maintenance shall be made only under carefully controlled conditions. Provisions shall be made for storage of in-process material elsewhere, for temporary alternative barriers, and for adequate inflow of air to ensure contamination control. The exhaust ventilation system shall be sized to ensure radiological doses are maintained at ALARA levels in the event of the largest credible breach."

- Section 1326, Tritium Facilities

Subsection 1326-5.1. Design Features (paragraph 1). "Design considerations shall evaluate all maintenance operations offering a potential for significant exposure and where practical they should be designed for remote repair or service. In those instances in

which remote or enclosed maintenance cannot be achieved, a compressed-air breathing air system shall be provided to allow the use of supplied-air suits by maintenance personnel. However, every effort shall be made to allow routine maintenance activities to be conducted without the need for supplied breathing air."

A1.2.4 Mechanical

- Section 1550, Heating, Ventilating and Air-Conditioning Systems--Special Facilities

Subsection 1550-99.0.2. Confinement Ventilation Systems (paragraph 12). "Air filtration units shall be located and provided with appropriate radiation shielding to maintain occupational doses ALARA during operations and maintenance."

Subsection 1550-99.0.2. Confinement Ventilation Systems (paragraph 47). "Airborne contaminant cleaning systems shall be designed for convenient maintenance and the ability to decontaminate components and replace components in the supply, exhaust, and cleanup systems without exposure of maintenance or service personnel to hazardous materials. Filtration systems shall be designed so that a bank of filters can be completely isolated from the ventilation systems during filter element replacement."

A1.2.5 Comments

From these DOE Order 6430.1A, *General Design Criteria* (DOE 1989) requirements, selected out of context, there is a general concern about maintenance activities. These requirements do not prohibit contact maintenance where radioactive materials are involved, but compliance with appropriate limits must be assured and the design must be ALARA. Remote repair and maintenance capabilities are preferred.

There is also a recognition that breathing air and respiratory protection may be required in cases where design considerations do not permit adequate control and confinement of radioactive materials during maintenance activities. However, engineered controls are considered to be a better alternative than respiratory protection.

A1.3 WESTINGHOUSE HANFORD COMPANY CRITERIA (WHC-CM-4-9)

Radiological Design, WHC-CM-4-9, provides radiological design guidance for external radiation and prevention of internal depositions of radio-nuclides. Aspects that relate to contact maintenance are discussed below.

A1.3.1 External Exposure Rate Limitations

External exposure rate design guidance is provided in Section 8 of WHC-CM-4-9. Operations areas in which full-time access is required should

provide an initial design level of 0.2 mrem/h (400-500 mrem/year). Proportionally higher rates are recommended for lower allowed access times. These rates are purposely set at 50% of design guidance provided in DOE Order 5480.11, *Radiation Protection for Occupational Workers* (DOE 1988) and DOE Order 6430.1A, *General Design Criteria* (DOE 1989). This provides a "reality check" to ensure that achieved facility operational experience, after contact maintenance and operations activities and the buildup of contamination within the facility are taken into consideration, will be less than the 1,000 mrem/year design basis provided in these DOE Orders.

A1.3.2 Confinement Device Recommendations

To maintain internal doses to the ALARA objective of no internal depositions, prudent design recommendations are provided in Section 5 of WHC-CM-4-9. For moderately dispersible radioactive materials, such as those that would occur in routine maintenance of process systems, the following limits are recommended when working with ^{90}Sr and ^{147}Pm :

- Total confinement device (glove box or hot cell)--Activity $\geq 10^{-6}$ Ci
- Partial confinement device (hood)--Activity $< 10^{-6}$ Ci
- No confinement-- $< 2.0 \times 10^{-9}$ Ci/g on a dry weight basis.

Because ^{90}Sr represents the worst case internal deposition hazard of beta emitting radionuclides, this limit is appropriate. This assumes that transuranics (TRU) are not present at the same time in levels $\geq 10^{-8}$ Ci.

For ^{137}Cs the limit is a factor of 10 greater, i.e., $\leq 10^{-5}$ Ci for partial confinement.

Uranium and thorium as dilute liquids (< 100 g/L) do not require special consideration.

The TRUs require partial or total confinement. Hoods (with air flow velocities of approximately 33 ± 8 m/min) are acceptable if liquid forms will be used and the operation will not generate airborne contamination. Highly dispersible forms, or dry dusty operations, require total confinement for activities $> 10^{-8}$ Ci. The U.S. Atomic Energy Commission's *Regulatory Guide 3.10, "Liquid Waste Treatment System Design Guide for Plutonium Processing and Fuel Fabrication Plants"* (AEC 1973) provides guidance for work areas with wet processing operation of plutonium. Milligram quantities ($\sim 1 \times 10^{-3}$ Ci) of plutonium undergoing wet processing can be conducted in a suitable closed system or glove box.

A1.3.3 Summary

Radiological Design, WHC-CM-4-9, guidance takes an operational based view of personnel exposure in newly designed facilities. Recognition of the deterioration of "cold clean" design basis exposure rates and contamination control as a facility is operated for a period of time are accomplished through conservative recommendations. To provide a similar "margin of safety" with the new 500 mrem/year guidance in WHC-CM-1-6, *Westinghouse Hanford Company Radiological Control Manual*, the external exposure design criteria

provided in WHC-CM-4-9 should be reduced to 50% of the stated values. Recommendations for contamination control are believed to be adequate as promulgated.

A1.4 RELATED INFORMATION ON CONTACT MAINTENANCE

An ancillary task exposure study prepared for the Hanford Waste Vittrification Plant (HWVP) Project by Science Applications International Corporation (SAIC) projected occupational radiation doses associated with operations of that facility. Personnel exposures were estimated for the following categories: operations, maintenance, management, engineering, administrative, and health physics technicians.

Based on contact maintenance tasks scheduled for the high-level vittrification plant, an annual collective effective dose equivalent of 217 person-rem was projected for a total of 348 day shift workers and 235 rotating shift workers. Operations and Maintenance personnel incurred the majority of the collective dose equivalent (71%). Health physics personnel with 27% of the collective dose equivalent had average doses exceeding the administrative dose limits of 500 mrem/year in WHC-CM-1-6. While Operations and Maintenance personnel had average doses <500 mrem/year, a number of these personnel would be expected to exceed that limit also.

These estimates were based on source terms thought to be characteristic of the high-level vittrification plant feed. To extrapolate these estimates to different conditions, a comparison of the respective source terms would have to be accomplished.

The salient point of the SAIC study was that with the design objective of 1,000 mrem/year employed in the HWVP, there may be difficulty in operating the planned facility to a new, decreased administrative control limit of 500 mrem/year. Contact maintenance and operations activities are far more widespread in contributing to collective personnel exposures than originally anticipated. Therefore, new facility designs should use the new 500 mrem/year personnel exposure guidance.

This study also pointed out the value of overlaying planned operations and anticipated "feed" onto the intended design. At this point, evaluations of potential success at achieving and/or staying within personnel exposure limits to plant personnel can be conducted. Results from these evaluations can then be used to modify design parameters for acceptable outcomes. This iterative process is most effectively used during conceptual design when required changes are most easily accommodated.

A2.0 REFERENCES

29 CFR 1910.134, "Occupational Safety and Health Standards," *Code of Federal Regulations*, as amended.

AEC, 1973, *Regulatory Guide 3.10*, "Liquid Waste Treatment System Design Guide for Plutonium Processing and Fuel Fabrication Plants," U.S. Atomic Energy Commission, Washington, D.C.

ANSI, 1992, *Respiratory Protection*, ANSI Z88.2, American National Standards Institute, New York, New York.

DOE, 1985, *Contractor Industrial Hygiene Program*, DOE Order 5480.10, U.S. Department of Energy, Washington, D.C.

DOE, 1988, *Radiation Protection for Occupational Workers*, DOE Order 5480.11, U.S. Department of Energy, Washington, D.C.

DOE, 1989, *General Design Criteria*, DOE Order 6430.1A, U.S. Department of Energy, Washington, D.C.

DOE, 1992, *Radiological Control*, DOE N 5480.6, U.S. Department of Energy, Washington, D.C.

WHC-CM-1-6, *Westinghouse Hanford Company Radiological Control Manual*, Westinghouse Hanford Company, Richland, Washington.

WHC-CM-4-9, *Radiological Design*, Westinghouse Hanford Company, Richland, Washington.

A3.0 GLOSSARY

ABBREVIATIONS AND ACRONYMS

ALARA	as low as reasonably achievable
DBA	design basis accident
DOE	U.S. Department of Energy
HWVP	Hanford Waste Vitrification Plant
SAIC	Science Applications International Corporation
TRU	transuranic

AIM

Association for Information and Image Management

1100 Wayne Avenue, Suite 1100
Silver Spring, Maryland 20910

301/587-8202

Centimeter

Inches

MANUFACTURED TO AIM STANDARDS
BY APPLIED IMAGE, INC.

DATE

FILMED

12/ 8 /94

END