Savannah River Site Special Exposure Cohort Petition Evaluation Report

Thorium at the Savannah River Site (post 1972)

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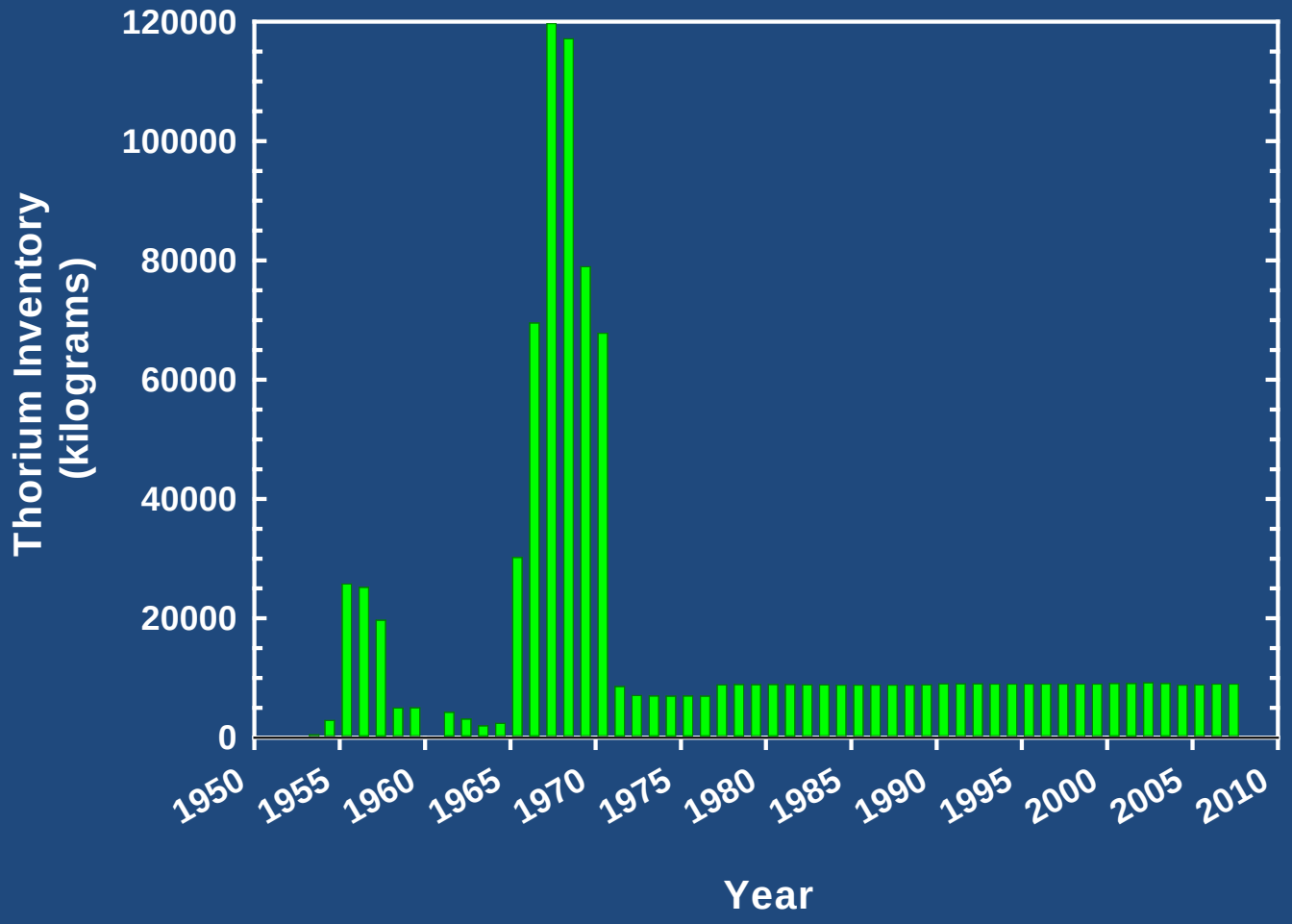
Overview

- Thorium Inventory
- Processes involving thorium
- Radiological Controls 1972-1990
  - Alternate Bioassay
  - Bioassay Control
- Am/Cm/Cf/Th Comparison
- Thorium 1990-2007

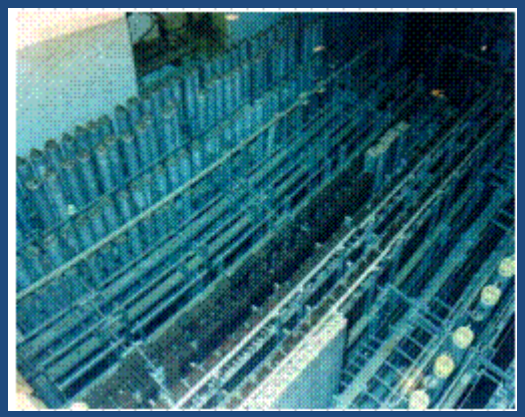
Thorium Inventory at SRS



Receiving Basin for Offsite Fuels (RBOF)

- Encapsulated spent nuclear fuels
- Fuels repackaged underwater
- Fuels stored underwater

RBOF Building



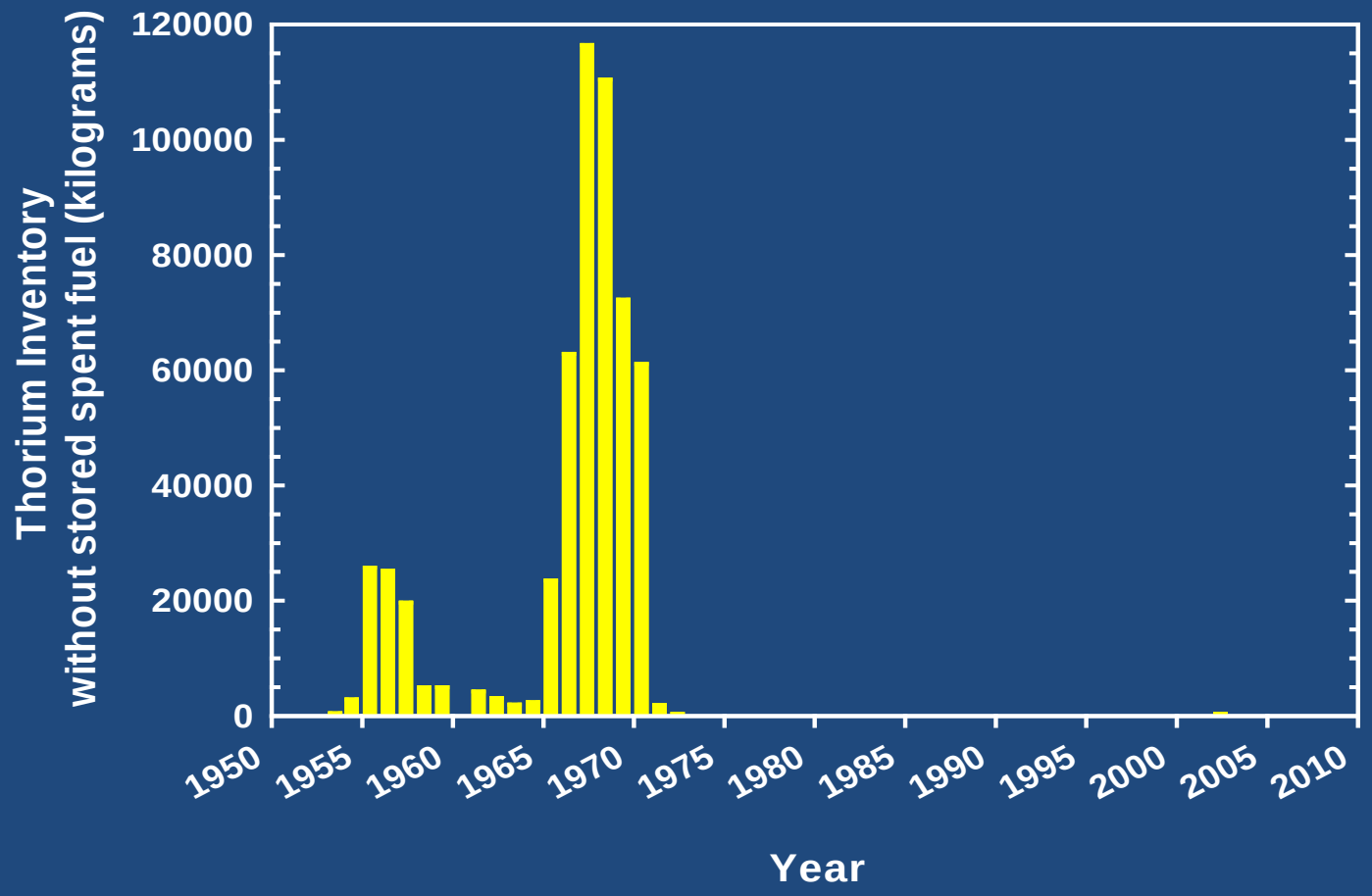
Stored spent fuel

Repackaging basin

Photos from WSRC-MS-99-00678 accessed via public internet at http://sti.srs.gov/fulltext/ms9900678/ms9900678.html

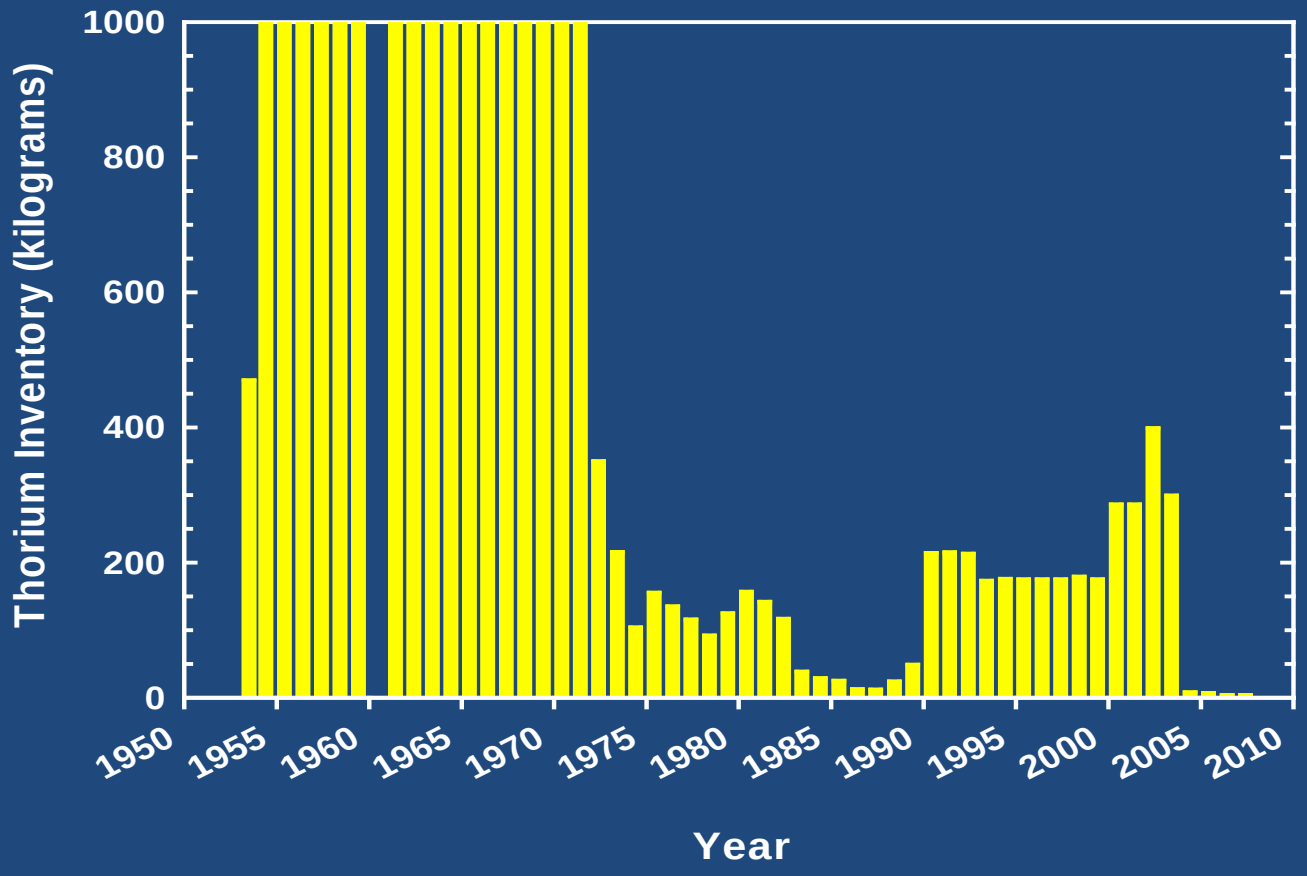
Thorium inventory

w/o water stored, encapsulated spent fuel



Expanded Scale of Previous Graph

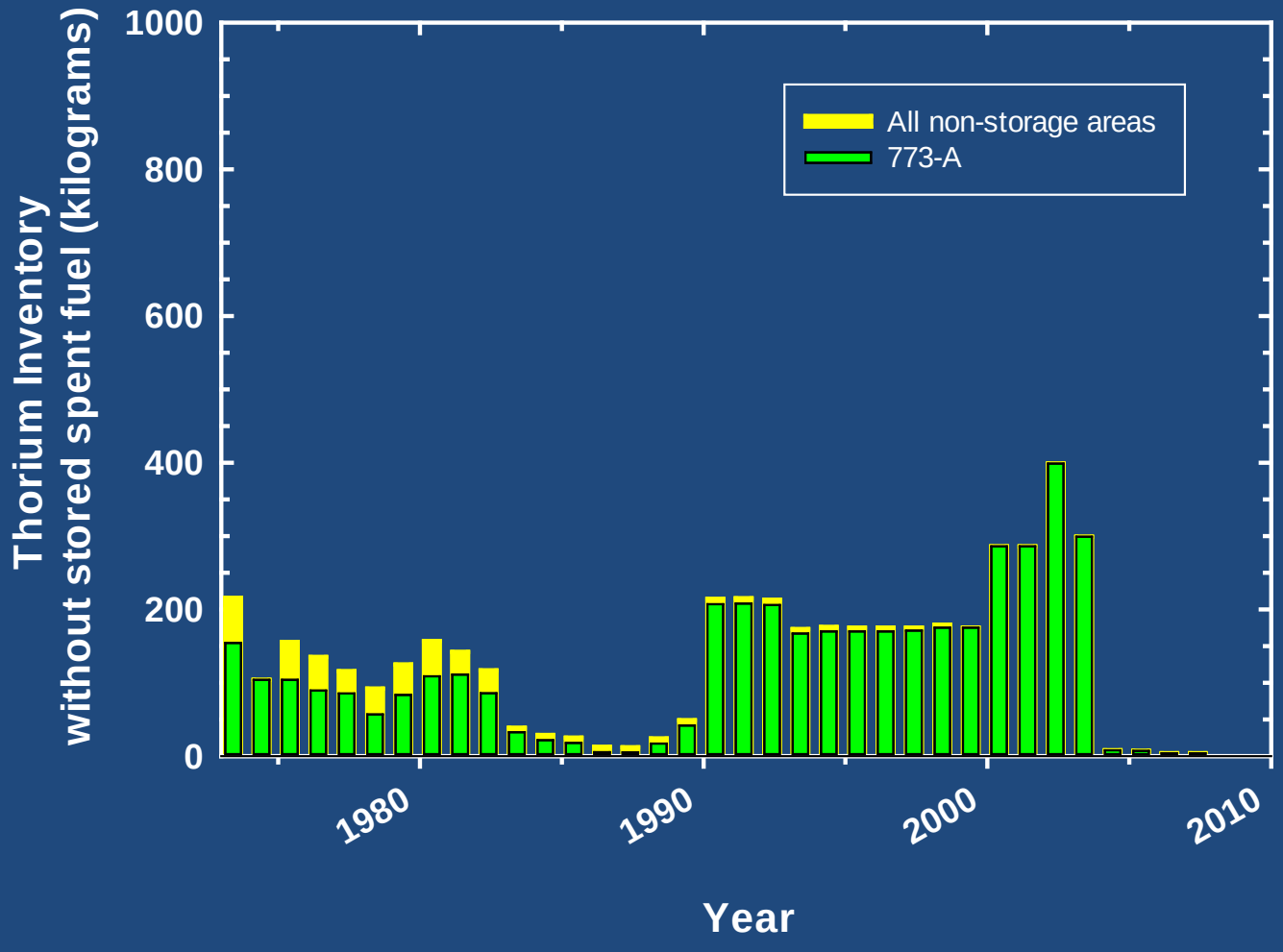
- Inventories indicate very small thorium inventory 1972 - 2007



Low Inventory @ Minimal Locations

SRS Thorium Inventory, 1972-2007 (kg) (partial)										
Year	773A	723A	235-F	772-F Lab	M Area	777 M	217-A Storage	100-K Basin	100-L Basin	RBOF
1973	154	---	0	---	57	6	0	52	---	6679
1974	104	---	---	---	---	---	---	52	---	
1975	104	1	1	1	43	5	---	52	---	6757
1976	89	1	1	1	41	2	0	52	---	6757
1977	85	1	1	1	25	2	0	52	---	8329
1978	57	---	4	4	25	2	0	52	---	8729
1979	83	---	4	4	31	2	0	52	---	8729
1980	109	---	5	5	31	---	8.0	52	3	8726
1981	111	---	4	4	23	---	0	55	3	8726
1982	86	---	4	4	23	---	0	55	3	8726
1983	33	---	4		2	---	0	55	3	8726
1984	22	---	4	1	2	---	---	55	3	8726
1985	18	---	4	1	2	---	---	55	3	8726
1986	5	---	4	1	2	---	---	52	3	8726
1987	5	---	4	1	2	---	---	52	3	8726

Minimal Locations



Process Knowledge
Savannah River Laboratory 773A

- 1972 - Alpha Material Laboratory used thorium oxide as a surrogate for Pu-238 testing in glove boxes
- 1973 – Gram quantities of thorium dioxide shards were used in 773A hot cells to test vapor deposition
- 1977-1980 Alternate Fuel Cycle Technology Program (AFCT) and Thorium Fuel Cycle Technology Program (TFCT) – several research projects

Process Knowledge
Savannah River Laboratory 773A-cont.

- Multiple AFCT/TFCT studies
  - Mechanical grinding of ThO_2 in high level caves
  - Study on effects of heat treatment on thorium oxide
  - Testing on conceptual THOREX flowsheets of Elk River fuel in high level caves
  - Analysis of off gassing of spent thorium fuel (Elk River Fuel – high level caves)
  - Hanford prepared (encapsulated) 30 fuel rods with 80%ThO_2 – 20%UO_2 for irradiation at SRS. SRS received rods in 1979 and stored them in a cage in 773A. The program was cancelled in May 1980 before they could be irradiated.

Process Knowledge - Other

- Pu-238 Fuel Form Facility (1980)
  - Thorium used as a surrogate for some of the work performed in the hot cells of the PuFF. Also used as a doping agent of the iridium welding agents
- Galileo Project (1987)
  - Thorium used as a surrogate for plutonium during process testing
- Defense Waste Stabilization (1990-2010)
  - Thorium used as a surrogate for plutonium and other radionuclides to test methods for defense waste stabilization and immobilization

Radiological Controls 1972-1989

- **Special Hazards Bulletins and DPSOP-40 Savannah River Plant Radiation and Contamination Control and DPST-RH Radiation Hazards Technical Standards covered:**

  1. Work in regulated areas
  2. Investigating radiation and contamination incidents
  3. Protective clothing
  4. Injury in regulated areas
  5. Disposal of contaminated waste
  6. Fires in regulated areas
  7. Radiation exposure control

Alternate Bioassay data

- A large number of workers in 773A were monitored for Am, Cm, Cf
- Review of the bioassay method during development of the co-worker model for Am, Cm, Cf revealed that thorium would come through in the analysis and the alpha emissions would be counted as if it were Am, Cf, Cm

Alternate Bioassay data – cont.

FE Butler and RM Hall, Analytical Chemistry Vol 42, No 9 pp. 1073-1076 (1970)

A procedure was developed for sequential extraction of plutonium, neptunium, and uranium with tri-isooctylamine (TIOA), followed by extraction of thorium, americium, curium, berkelium, californium, and einsteinium with bidentate. Compared with previous methods, the new procedure is simpler, requires less analysis time, and gives better recovery. The recovery of Am-Cm-Cf from 250 ml of urine or 20 grams of feces was 90%.

Alternate Bioassay data – cont.

All alpha emitting actinides from thorium through einsteinium extract, indicating an excellent gross alpha analytical procedure. The data show that in analysis of americium, curium, and californium any contaminating plutonium, neptunium, or uranium must be removed. At this laboratory, thorium, berkelium, and einsteinium are not present in biological samples in sufficient quantities to require separation or routine identification by alpha spectrometry.

Alternate Bioassay data

- No effort was made by the lab to remove the thorium contaminant from the urine sample
- Why?
  - Activities were much lower
  - Not viewed as a significant contaminant
  - Thorium used as a surrogate because it was less hazardous than plutonium (i.e. safer to use)
- Effectively we have alpha urine bioassay sample that doesn't contain plutonium, uranium, or neptunium, but does contain Th, Am, Cm, Cf, Es, Bk

Bioassay Control

- DPSOL 193-302
  - Rev. 5 (1971)

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DPSOL 193-302 REV 5 PAGE 2 OF

3. ROUTINE BIOASSAY SAMPLING FREQUENCIES
(Excluding Construction Division)

CAT	PERSONNEL	SAMPLES PER YEAR[a] URINE ^{3}H	Pu	FP	EU	U	Am Cm Cf	CHEST EU	Pu Am Cm Cf
A	**Minimum Potential (Except HTO).** Personnel assigned to 284-F & -H, 704-F & -H, 706-F & -H, 717-F, and nonprocess sections of other facilities; patrolmen.		b						
B	**221-F & -H Fourth Level.** Separations supervision; all Sep Tech personnel, control room operators, janitors, and Clerical personnel.		1	1					
C	**221-H & H-Area Outside Facilities.** All operators (except control room and sample aisle), HP personnel, and selected Power, E & I, and Maintenance personnel assigned to 221-H process areas; all personnel assigned to H-Area outside facilities.	2	1	2	1				
D	**221-H Sample Aisle.** All 221-H sample aisle operators.		2	2	2				1
E	**221-F Sample Aisle.** All 221-F sample aisle operators; selected 772-F personnel.		2	2			2		1
F	**221-F, 723-F, & 643-G.** All operators (except control room and sample aisle), HP personnel, and selected Power, E & I, and Maintenance personnel assigned to 221-F process areas; all personnel assigned to 723-F and 643-G.		1	2					
G	**221-H B-Line, 221-F B-Line, JB-Line, & 235-F.** All personnel assigned to process sections in building 235-F, and all assigned personnel in other facilities.		2	2					1
H	**F-Area Outside Facilities.** All assigned personnel.		b	2		4[c]			
J	**772-F (Excluding UO_3 Section).** All assigned personnel.		2	2	1	1			1[g]
K	**313-M.** All assigned personnel.					4			
L	**322-M.** All assigned personnel (excluding personnel processing samples from field). **320-M.** All laboratory and selected RM personnel. **773-A.** Reactor Engineering group and 777-M assigned personnel.		b		1	4			
M	**322-M.** Personnel processing samples from field. **772-F, UO_3 Section.** All assigned personnel.		b	1	1	4			
	321-M. All assigned personnel.		1		4[d]			2[f]	
T	**100 Areas, 105 Building.** Reactor Department personnel from C&D crews, Purification, and pump room observation; control room and monitor operators; all 100-Area HP, Maintenance, and T & T personnel; all E & I personnel assigned to 105 Buildings; T & T personnel in Central Shops; and selected Reactor Tech and 400-Area personnel.	h		1[e]					
V	**773-A.** Analytical Chemistry, High Level Caves, Building Services, Radiation Control, and Maintenance personnel.		b	1			2		1[g]
W	**773-A.** Selected Clerical, supervisory personnel, and selected 100-Area personnel.		b				1		
X	**232-H, 234-H, 237-H, & 238-H.** All assigned personnel. **241-H & 244-H.** Selected personnel.	h	b						
	700-Area shop personnel provide samples as considered advisable by Health Physics.								

NOTE: Neptunium analysis is performed when requested by area Health Physics. Neptunium has never been detected without at least an equal amount of Pu.

Bioassay Control - Construction

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DPSOL 193-302
REV 5 PAGE 3 OF 14

4. BIOASSAY SAMPLING FREQUENCIES — Construction Division

a. Routine Urine Samples

- Fission Products and/or Induced Activity — one sample per year and when terminating.
- Tritium — sample frequency is outlined in Radiation and Contamination Control DPSOP 40-1 or Construction Job Plans.
- Plutonium — one sample every 3 years and when terminating.
- Other Nuclides — as specified by area Health Physics in Construction Job Plans.

[NOTE] Construction Division Medical Department annually provides each employee with a sample bottle and label and instructs the employee to submit a one-liter urine sample. Samples are also obtained from new employees who worked in Radiation Zones at another installation where radioactive materials were handled. Personnel Monitoring will forward requests for resamples through Construction Medical.

b. Special Sampling (See Division B and Construction Division Safety Procedure 58)

c. Whole Body or Chest Counting

1) New employees, who worked in Radiation Zones at another installation where radioactive materials were handled, will be required to take a whole body and chest count. This count should preferably be made on the same day as the entry physical examination.

2) A whole body and/or chest count shall be made whenever an employee's bioassay samples (except tritium) indicate he has a confirmed uptake or when he has been involved in a contamination incident and a count is considered necessary by Health Physics supervision.

3) A count (chest or 40 cm arc) will be required when terminating for those employees who have had a previous whole body or chest count at SRP.

DPSOL 193-302 (Rev 5. 1971)

Bioassay Control Procedure Revision

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HEALTH PROTECTION DEPARTMENT
DPSOP Ref 193

DPSOL 193-302T
Revision 0
Approval Date 2/25/85
Page 1 of 18

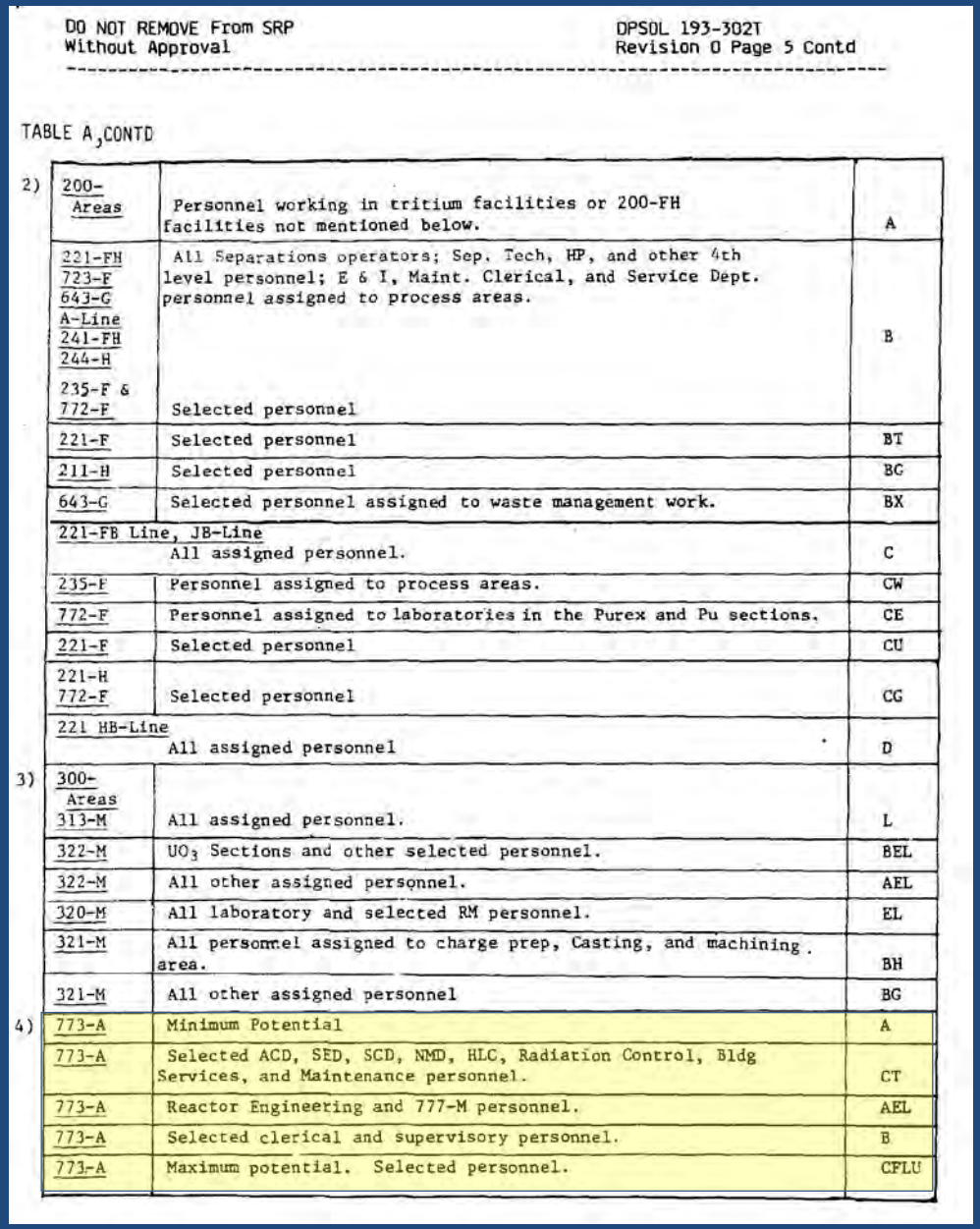
BIOASSAY CONTROL (TEMPORARY)

[NOTE: This DPSOL is a duplicate of DPSOL 193-302, Rev 8, approved 1/78 from the old manual. It is to be used until new bioassay procedures 193-211, -212, and -213 are issued.]

PURPOSE: To establish operating guides, bioassay sampling and in-vivo counting frequencies, and related administrative controls.

Bioassay Control

- DPSOL 193-302
  - Rev. 8 (1978)
- DPSOL 193-302T
  - Rev. 0 (1985)
- DPSOL 193-211
  - Rev. 0 (1989)



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DPSOL 193-302T
Revision 0 Page 5 Contd

TABLE A, CONTD

2)	200-Areas	Personnel working in tritium facilities or 200-FH facilities not mentioned below.	A
	221-FH 723-F 643-G A-Line 241-FH 244-H	All Separations operators; Sep. Tech, HP, and other 4th level personnel; E & I, Maint. Clerical, and Service Dept. personnel assigned to process areas.	B
	235-F & 772-F	Selected personnel	
	221-F	Selected personnel	BT
	211-H	Selected personnel	BG
	643-G	Selected personnel assigned to waste management work.	BX
	221-FB Line, JB-Line	All assigned personnel.	C
	235-F	Personnel assigned to process areas.	CW
	772-F	Personnel assigned to laboratories in the Purex and Pu sections.	CE
	221-F	Selected personnel	CU
	221-H 772-F	Selected personnel	CG
	221 HB-Line	All assigned personnel	D
3)	300-Areas 313-M	All assigned personnel.	L
	322-M	UO_3 Sections and other selected personnel.	BEL
	322-M	All other assigned personnel.	AEL
	320-M	All laboratory and selected RM personnel.	EL
	321-M	All personnel assigned to charge prep, Casting, and machining area.	BH
	321-M	All other assigned personnel	BG
4)	773-A	Minimum Potential	A
	773-A	Selected ACD, SED, SCD, NMD, HLC, Radiation Control, Bldg Services, and Maintenance personnel.	CT
	773-A	Reactor Engineering and 777-M personnel.	AEL
	773-A	Selected clerical and supervisory personnel.	B
	773-A	Maximum potential. Selected personnel.	CFLU

Bioassay Control – 773A

- Minimum Potential – A
- Analytical Chemistry, High Level Caves, Radiation Control, Building Services, Maintenance - CT
- Reactor Eng. - AEL
- Clerical and Supervisory Personnel - B
- Maximum Potential - CFLU

TABLE B

Nuclide	Description of Code Samples/year	Code
Plutonium	0.3	A
	1	B
	2	C
	4	D
Enriched Uranium	1	E
	2	F
	4	G
	12	H
Natural Uranium	1	J
	2	K
	4	L
	12	M
Fission Product Induced Activities	0	N
	1	P
	2	R
	4	S
Americium Curium and Californium	1	T
	2	U
	4	V
Neptunium	1	W
Strontium	1	X
	2	Y
	4	Z

DPSOL 193-302T (Rev 0 1985)

Bioassay Control - Construction

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DPSOL 193-302T
Revision 0 Page 7 Contd

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2. Construction Division

2.1 Routine Urine Samples

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NOTE: Construction Division Medical Department annually provides each employee with a sample bottle and label and instructs the employee to submit a one-liter urine sample. Samples are also obtained from new employees who worked in Radiation Zones at another installation where radioactive materials were handled. Personnel Monitoring will forward requests for resamples through Construction Medical.

2.2 Special Sampling.
See division C and Construction Division Safety Procedure 58.

DPSOL 193-302T (Rev 0 1985)

Bioassay Control Summary

- Monitoring prescribed by work area
- Monitoring frequency is based on potential for exposure
- Construction Workers covered under Job Plans

Dose Reconstruction Methodology

- NIOSH proposed to use the Am/Cm/Cf/Th bioassay results to reconstruct thorium doses

- Given a particular cancer, NIOSH will use the radionuclide (Am, Cm, Cf, or Th) that results in the highest dose to the organ of interest

Example Dose Reconstruction

Summary of Thorium and Trivalent Actinides Doses (1985–1989)
The table below outlines the doses for each organ of concern. The highest dose of the five (americium, curium, californium, thorium Type M, and thorium Type S) was used for dose assignment. Thorium M (including thorium-228 and thorium-232) resulted in the highest dose to the bone, kidney, prostate, and skin. Thorium S (including thorium-228, thorium-232, and californium-252) resulted in the highest dose to the lung.

Summary of Am/Cm/Cf/Th Dose

Radionuclide	Bone	Kidney	Lung	Prostate	Skin
Am	18.960	0.146	0.814	0.021	0.021
Cm	15.240	0.132	0.971	0.016	0.016
Cf	36.580	0.005	3.688	0.009	0.002
Th M (total)	37.67	**0.755**	3.852	**0.077**	**0.077**
Th-228	3.071	0.080	1.390	0.008	0.008
Th-232	34.590	0.674	0.729	0.069	0.069
Th S (total)	**37.944**	0.279	**16.390**	0.034	0.033
Th-228	0.312*	0.009*	5.282*	0.001*	0.001*
Th-232	6.416*	0.147*	7.961*	0.015*	0.014*
Am	16.180	0.123*	0.695	0.018*	0.018*
Cm	13.000	0.111	0.823	0.014	0.014
Cf	31.22*	0.003	3.147*	0.007	0.002

Note: **Bold** characters indicate the dose used in the dose assignment (either Th M total *or* Th S total).
*Indication of which options were used in the Th S dose total (highest dose of Am or Cm or Cf).

Am/Cm/Cf/Th Bioassay Comparison

- ORAUT-RPRT-0055: *A Comparison of Exotic Trivalent Radionuclide Co-worker Models at the Savannah River Site*
- Comparison is being reviewed by SEC Workgroup
- Three Co-worker Models Developed:
  - Construction Trades Workers (CTW) Coworker Model
  - Non-Construction Trades Workers (nCTW) Coworker Model
  - Non-Construction Trades + unknowns (nCTW+unk) Co-worker Model

Am/Cm/Cf/Th Bioassay Results

Period	AMW		CTW		nCTW		nCTW +unk	
	Total	OPOS	Total	OPOS	Total	OPOS	Total	OPOS
1970	1,955	567	328	124	1,593	451	1,627	461
1971	1,856	663	292	107	1,545	550	1,564	559
1972	1,565	650	208	109	1,312	525	1,357	541
1973	1,249	644	243	115	969	509	1,006	530
1974	1,067	456	162	86	876	357	905	371
1975	831	467	173	94	628	356	658	375
1976	695	450	148	90	523	346	547	360
1977	478	383	87	68	368	292	391	315
1978	306	228	66	49	232	171	240	179
1979	441	322	79	67	337	234	362	255
1980	253	230	44	42	198	178	209	188
1981	341	267	80	44	524	379	586	422
1982	325	307						
1983	330	303	41	39	255	232	289	264
1984	347	275	63	20	234	210	284	255
1985	340	259	42	24	266	214	298	235
1986	399	273	101	26	253	219	298	247
1987	379	305	65	25	598	336	656	371
1988–1989	342	288						

Worker and Bioassay Proportions

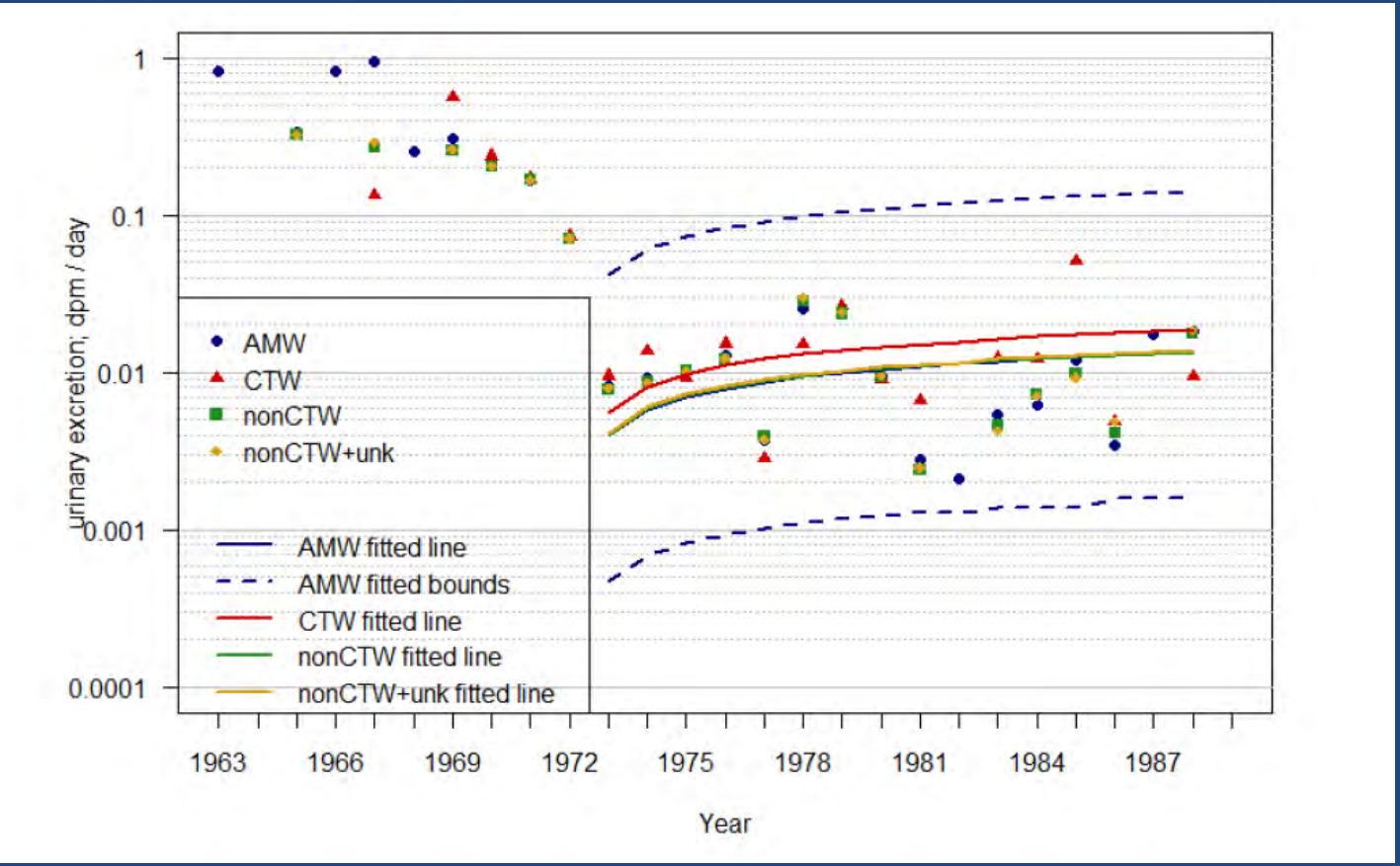
Year	# nonCTW+unks	# CTWs	Population ratio	Bioassay ratio
1973	5,255	500	0.10	0.22
1974	5,205	600	0.12	0.23
1975	5,140	973	0.19	0.25
1976	5,407	995	0.18	0.25
1977	5,598	1,344	0.24	0.22
1978	5,944	1,973	0.33	0.27
1979	5,709	1,958	0.34	0.26
1980	6,050	1,991	0.33	0.22
1981	6,593	2,159	0.33	0.10

Am/Cm/Cf/Th - OPOS Test Results

Period	CTW:nCTW		CTW:nCTW+unknowns	
	Peto-Prentice	Holm cutoff	Peto-Prentice	Holm cutoff
1970	0.9815	0.05	0.8923	0.025
1971	0.806	0.0125	0.813	0.0125
1972	0.7692	0.01	0.7053	0.01
1973	0.3314	0.005	0.3383	0.005
1974	0.8536	0.0167	0.8684	0.0167
1975	0.5771	0.0071	0.5465	0.0071
1976	0.2562	0.0042	0.3021	0.0045
1977	0.3587	0.0062	0.4499	0.0062
1978	0.2401	0.0038	0.2186	0.0038
1979	0.9295	0.025	0.9082	0.05
1980	0.276	0.0045	0.2646	0.0042
1981–1982	0.0243	0.0029	0.0315	0.0029
1983	0.0129	0.0028	0.0074	0.0028
1984	0.1263	0.0031	0.0930	0.0031
1985	0.0006	0.0026	0.0005	0.0026
1986	0.1268	0.0033	0.1782	0.0033
1987–1989	0.6796	0.0083	0.6796	0.0083

Significance $p<0.05$ and Peto-Prentice < Holm cutoff

Am/Cm/Cf/Th Coworker Model



Construction Trades Workers

- Am/Cm/Cf/Th Bioassay Samples (1972-1989)
  - Construction Trades Workers – 1602
  - Non-Construction Trades Workers – 7573
  - Unknown occupation – 422
- Recall construction trades workers were monitored based on Job Plans
- If Construction trades workers were never monitored then we would not have 1602 bioassay samples

Thorium (1990-2007)

- Originally proposed use of Whole Body Counts to bound thorium Exposures
- Although bounding, the assignment of WBC missed dose would result in some significant doses in the modern era that we just don't believe occurred given the radiological controls in place at the time
- NIOSH proposes using an air sample concentration of 2e-13 uCi/cc as a maximum potential exposure
  - 10% of the Plutonium DAC that was used as the cut point for respiratory protection

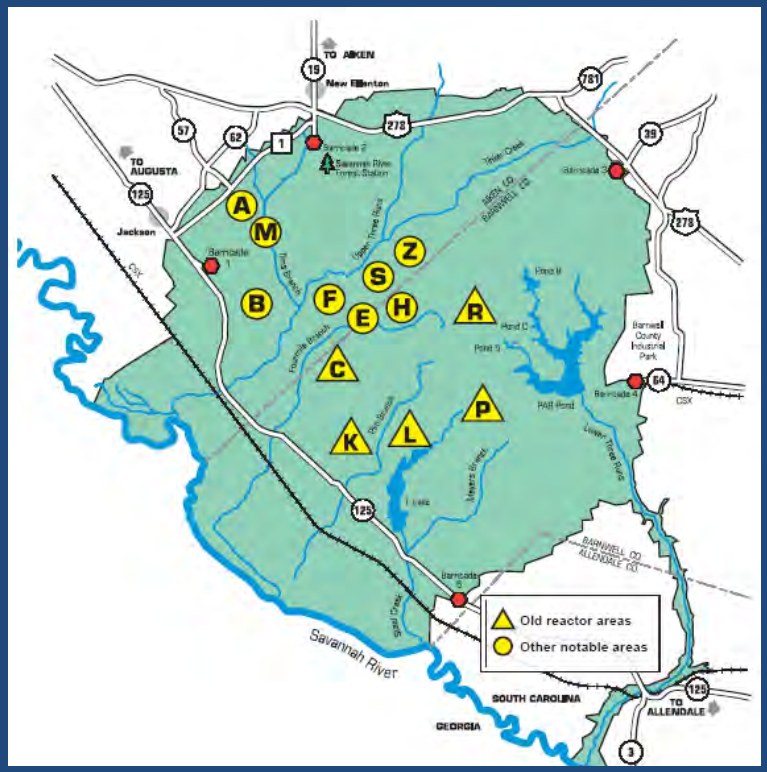
Thorium Inventory post - 1986

Year	773A	235-F	772-F	M Area	100-K Basin	100-L Basin	RBOF
1987	5	4	1	2	52	3	8726
1988	17	4	1	2	52	3	8726
1989	42	4	1	2	52	3	8726
1990	207	4	1	2	52	3	8726
1991	208	4	1	2	52	3	8726
1992	206	4	1	2	52	3	8726
1993	167	5	1	---	52	3	8727
1994	170	5	1	---	52	3	8727
1995	170	5	---	---	52	3	8730
1996	170	5	---	---	52	3	8730
1997	171	4	---	---	52	3	8730
1998	175	4	---	---	52	3	8730
1999	175	---	---	---	52	3	8730
2000	286	---	---	---	52	3	8730
2001	286	---	---	---	52	3	8730
2002	399	---	---	---	52	3	8730
2003	299	---	---	---	52	3	8730
2004	8	---	---	---	52	3	8730
2005	7	---	---	---	52	8785	---
2006	4	---	---	---	---	8968	---
2007	4	---	---	---	---	8968	---

Kilograms of thorium

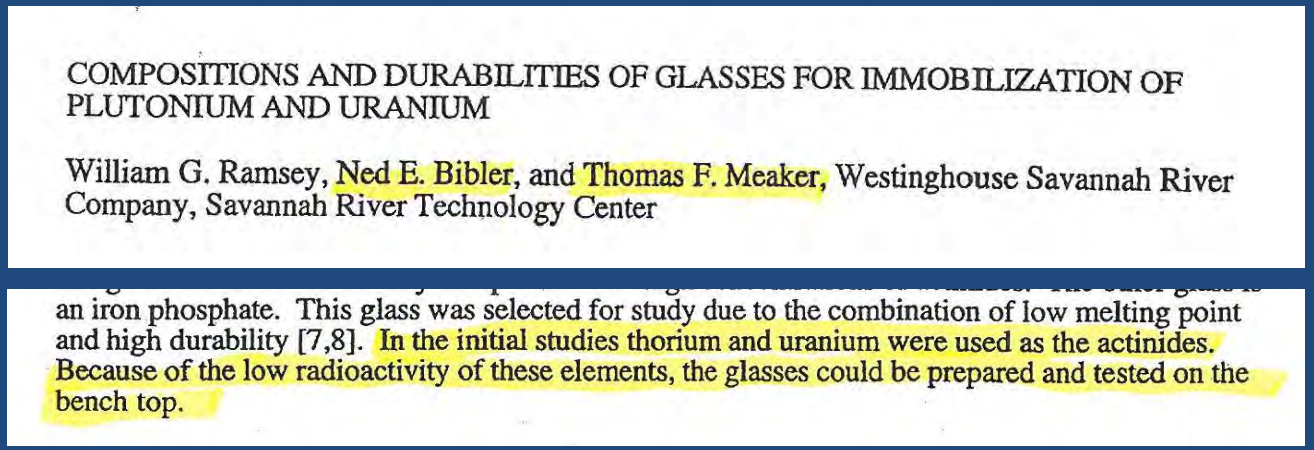
Thorium Volume and Activity

- Mass
  - 200 kg = 20.0 mCi
- Small Volume
  - 200 kg = approximately ten 2L bottles of ThO_2
- 773-A is a fairly large building
  - Small volumetric source term



Thorium (post 1990)

- Based on NIOSH's review, the primary work involving thorium during the 1990s was defense waste processing research
- Thorium was used as a surrogate in place of plutonium and other actinides for vitrification research



COMPOSITIONS AND DURABILITIES OF GLASSES FOR IMMOBILIZATION OF PLUTONIUM AND URANIUM

William G. Ramsey, Ned E. Bibler, and Thomas F. Meaker, Westinghouse Savannah River Company, Savannah River Technology Center

an iron phosphate. This glass was selected for study due to the combination of low melting point and high durability [7,8]. In the initial studies thorium and uranium were used as the actinides. Because of the low radioactivity of these elements, the glasses could be prepared and tested on the bench top.

Ramsey et al. (1994)

Research Examples (post 1990)

WESTINGHOUSE SAVANNAH RIVER COMPANY
SAVANNAH RIVER TECHNOLOGY CENTER

WSRC-TR-96-0323 Rev. 0

Keywords: Plutonium surrogate, ternary diagram, processing region.

September 30, 1996

To: Distribution, 773-A

From: T.F. Meaker, 773-A

Homogeneous Glass Processing Region Defined for a Lanthanide Borosilicate Glass Composition for the Immobilization of Plutonium Using Thorium as a Surrogate (U)

CC: E.W. HOLTZSCHEITER, 773-A
J.M. PAREIZS, 773-A
N.E. BIBLER, 773-A
D.K. PEELER, 773-A
J.C. MARRA, 773-43A
M.J. PLODINEC, 773-A
E.F. DUHN, 773-A
D.A. CROWLEY, 773-43A
T.Y. BAXTER, 773-A
STI (4)

Experimental

Glass compositions were batched in platinum crucibles with reagent chemicals in the oxide form with the exception of boron. The boron content was added as boric acid. Glasses were melted at 1475° C for 4-6 hours with manual stirring after ~2 hours at temperature. The ramp rate was 560° C/hr. As the melts were removed from the furnace, the crucibles were placed in a pan of water to 'shock' the glass from the crucible surface which allowed for total recovery of the processed glass. The compositional envelope was defined by varying the percentages of:

(1) Frit (Table I),
(2) Rare Earth (lanthanide) oxides, and
(3) ThO_2 (PuO_2 surrogate).

Table II. Batched compositons of glasses that were tested for durability in weight percent oxide.

Glass Id	Base Frit	REE	ThO_2	Homogeneous?
B-24-17	59	24	17	YES
Base	100	0	0	$AlBO_3$
B-5-0	95	5	0	$AlBO_3$
B-5-10	85	5	10	$AlBO_3$
B-15-0	85	15	0	$AlBO_3$
B-25-0	75	25	0	YES
B-45-0	55	45	0	YES
B-55-0	45	55	0	YES
B-65-0	35	65	0	YES
B-75-0	25	75	0	Sr-Nd-SiO_4
B-70-0	30	70	0	Sr-Nd-SiO_4
B-15-10	75	15	10	$ThSiO_4$
B-20-5	75	20	5	$ALBO_3$
B-25-10 (PCT)	65	25	10	YES
B-35-10	55	35	10	YES
B-45-10	45	45	10	YES
B-55-10 (PCT)	35	55	10	YES
B-65-10	25	65	10	Sr-Nd-SiO_4†
B-60-10	30	60	10	Sr-Nd-SiO_4†
B-20-20	60	20	20	YES
B-25-20	55	25	20	YES
B-30-20	50	30	20	YES
B-45-20	35	45	20	ThO_2
B-40-20	40	40	20	ThO_2
B-35-20	45	35	20	YES
B-20-25	55	20	25	YES
B-25-25 (PCT)	50	25	25	YES
B-30-25	45	30	25	ThO_2
B-20-30	50	20	30	A.P.S.††
B-25-30	45	25	30	ThO_2
B-35-30	35	35	30	ThO_2
B-15-30	55	15	30	A.P.S.††
B-15-25	60	15	25	A.P.S.††
B-15-20	65	15	20	A.P.S.††
B-17.5-22.5	60	17.5	22.5	YES
B-20-10	70	20	10	YES
B-15-15	70	15	15	YES
B-20-15	65	20	15	YES
B-45-15	30	45	15	YES
B-50-15	35	50	15	ThO_2

† Also contained a ThO_2 layer on the bottom of the melt pool.
†† Amorphous Phase Separation.

Research Examples (post 1990)

WESTINGHOUSE SAVANNAH RIVER COMPANY
SAVANNAH RIVER TECHNOLOGY CENTER

WSRC-TR-96-0380 Rev. 1

Keywords: Neptunium surrogate, dissolution, surface area.

January 16, 1997

To: Distribution

From: T.F. Meaker, 773-A

NEPTUNIUM IMMOBILIZATION AND RECOVERY USING PHASE SEPARATED GLASSES

CC: E.W. HOLTZSCHEITER, 773-A
J.M. PAREIZS, 773-A
N.E. BIBLER, 773-A
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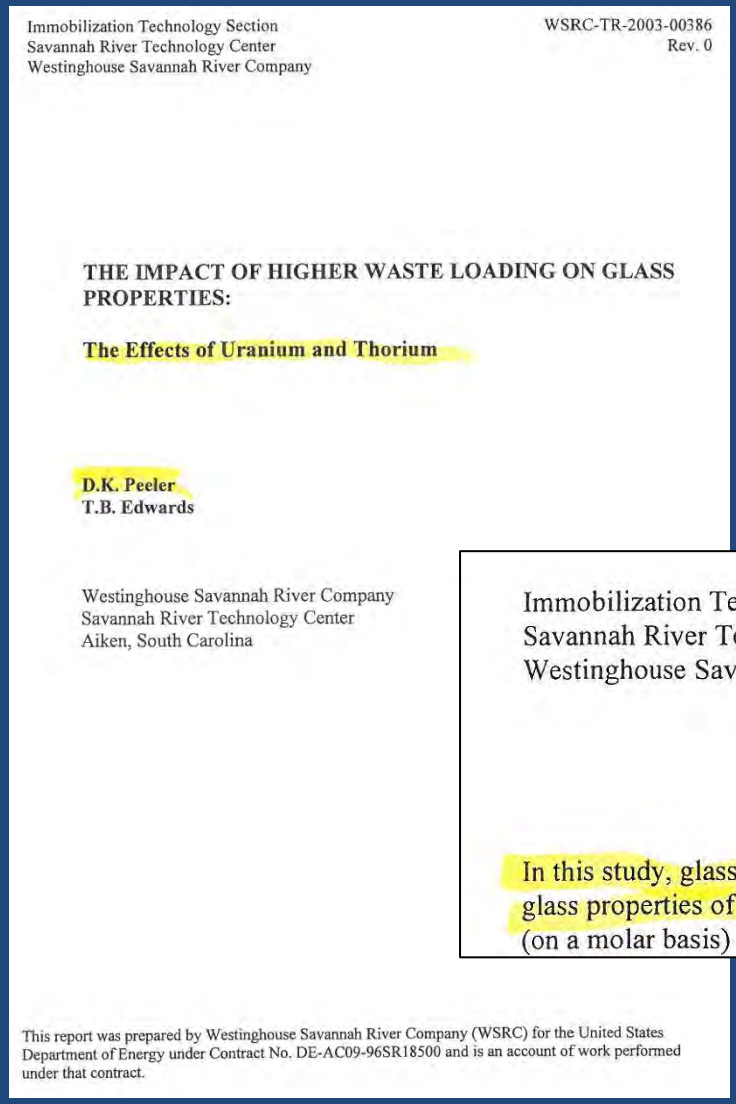
T.Y. BAXTER, 773-A
STI (4)

Summary

A phase separated (amorphous) glass has been developed which allows very effecient recovery of +4 valence actinides. The total amount of crystal formation in a heat treated vycor-type glass can be controlled with time, temperature and loading. Heat treatments at lower temperatures and for less time inhibit crystal formation while still allowing significant phase separation. If the Thorium loading exceeds 10 weight percent oxide, crystal formation during heat treatment may not be avoided. The total amount of crystal growth has a direct affect on thorium leachability. An increase in crystal formation limits the Th recovery significantly. High thorium loaded glasses (15 weight percent) with heat treatments (increased crystal formation) leach at approximately the same rate as non-heat treated glasses.

Research Examples (post 1990)



Immobilization Technology Section
Savannah River Technology Center
Westinghouse Savannah River Company

WSRC-TR-2003-00386
Rev. 0

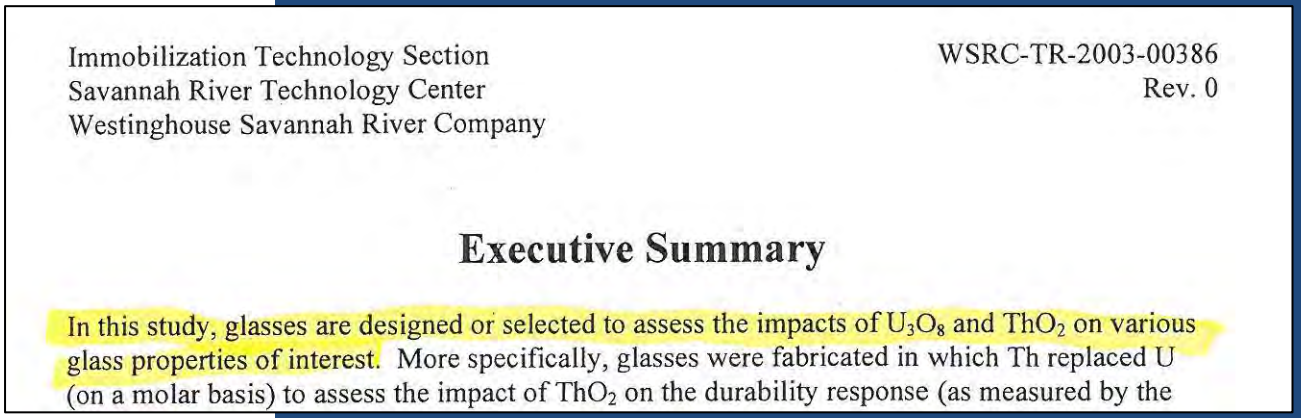
THE IMPACT OF HIGHER WASTE LOADING ON GLASS PROPERTIES:

The Effects of Uranium and Thorium

D.K. Peeler
T.B. Edwards

Westinghouse Savannah River Company
Savannah River Technology Center
Aiken, South Carolina

This report was prepared by Westinghouse Savannah River Company (WSRC) for the United States Department of Energy under Contract No. DE-AC09-96SR18500 and is an account of work performed under that contract.



Immobilization Technology Section
Savannah River Technology Center
Westinghouse Savannah River Company

WSRC-TR-2003-00386
Rev. 0

Executive Summary

In this study, glasses are designed or selected to assess the impacts of U_3O_8 and ThO_2 on various glass properties of interest. More specifically, glasses were fabricated in which Th replaced U (on a molar basis) to assess the impact of ThO_2 on the durability response (as measured by the

Thorium Handling Procedure

APPROVALS: Name Redacted
Group Manager
4/20/98
Date

Prepared By: Name Redacted
Date: 4/19/98

HANDLING DEPLETED URANIUM, NATURAL URANIUM, AND THORIUM

1.0 PURPOSE

The purpose of this procedure is to provide instructions for handling depleted uranium, natural uranium, and thorium in the Immobilization Technology Section.

2.0 SCOPE

This procedure applies to all Immobilization Technology Section (ITS) personnel who handle depleted or natural uranium and/or thorium.

3.0 PRECAUTIONS/LIMITATIONS

All established safety and work practices set forth by the SRS Safety Manual, 8Q, will be followed.

- Make a deliberate effort not to spread contamination to adjacent areas or to other personnel.
- Notify co-workers, supervision, and HPO immediately.

5.3 Handling

5.3.1 Depleted uranium, natural uranium and/or thorium may be used in a chemical hood (minimum face velocity of 60 fpm) or on a bench top if these elements are in the form of a hydrated nitrate compound, in a solution, or a solid block, e.g. a depleted uranium and/or thorium containing glass. It is recommended that the defined work area on a bench top be covered with paper before work is initiated.

5.3.2 Size reduction of depleted uranium and/or thorium glass for subsequent durability testing should be carried out inside double heavy duty plastic bags. The glass should be placed inside the double plastic bags and hit with a hammer several blows until the glass block is reduced to adequate non-respirable size. Non-respirable size reduction should be carried out in a chemical hood if possible, (minimum face velocity of 60 fpm) or on a bench top. It is recommended that the defined work areas on a bench top be covered with paper before work is initiated.

5.3.3 Depleted uranium, natural uranium, or thorium in a powdered (oxide or finely crushed glass) form may be used in a chemical hood (minimum face velocity of 150 fpm). After use, the chemical hood should be smeared to check for contamination. If contamination is found, contact HPO and decontaminate before further use.

5.3.4 Depleted uranium, natural uranium and depleted thorium shall be handled in the same manner as depleted uranium and thorium from Technical/Chemical Stores.

5.3.5 The indicated handling of depleted uranium, natural uranium and thorium given in steps 5.3.1 to 5.3.4 agrees with the definitions set forth in Table 2.2 of Manual 5Q, "Radiological Control".

5.3.6 If the source of depleted uranium/thorium is a mixed waste (both hazardous and radioactive, and is classified as a RCRA characteristic or listed waste, Procedure 6.01, "Waste Handling Guidelines" in Manual L1 should be followed.

Radiological Control Program

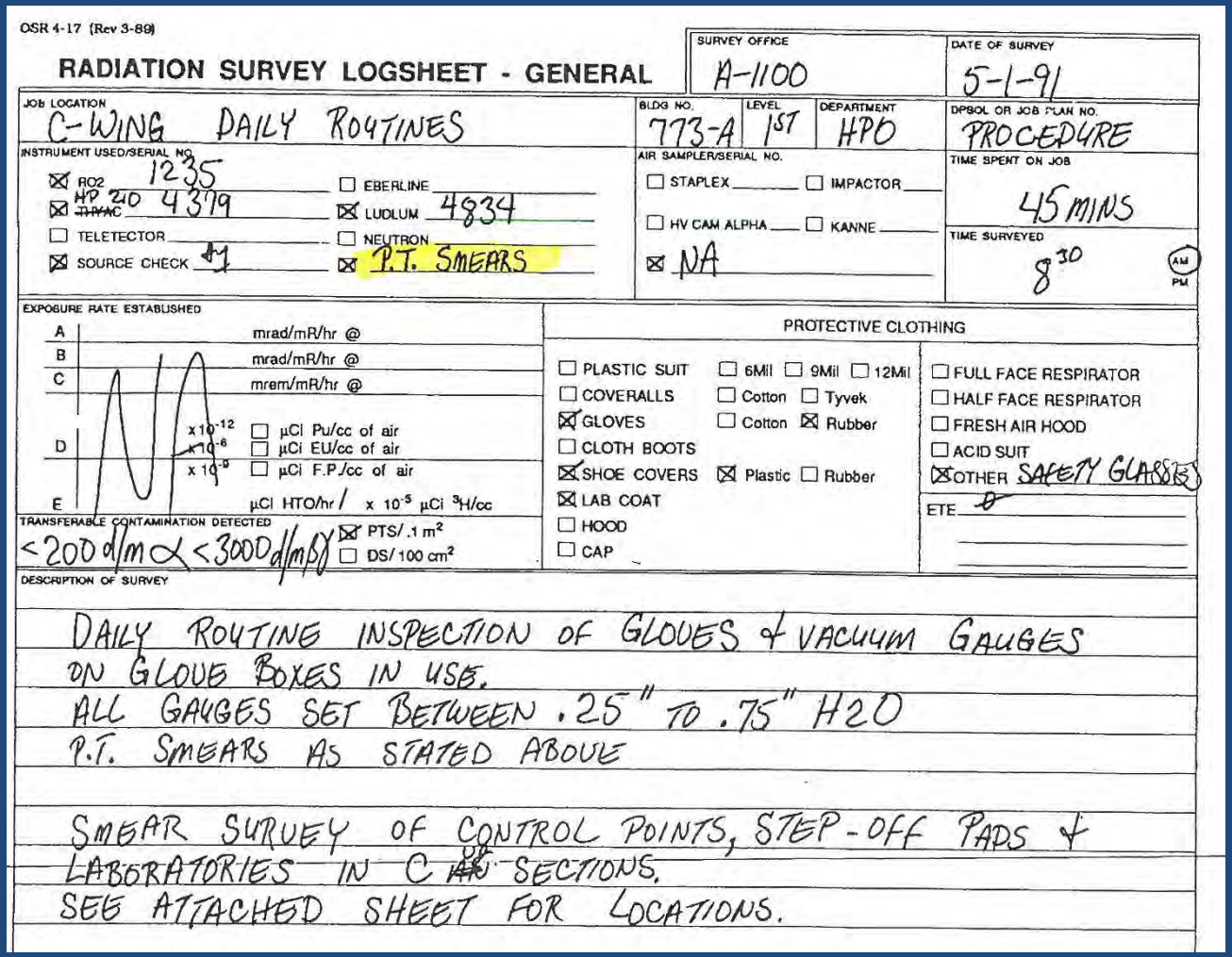
- Three Components
  - Radiation Survey Logsheets (RSL)
    - Daily, Weekly, Monthly, Quarterly
  - Air Sample Logsheets (ASL)
    - Daily, Weekly
    - <2% DAC
    - <10% DAC

- Routine workplace air sampling to <2% of a Derived Air Concentration (DAC). Performed in general areas to provide a reasonable assurance those non-radiation workers located in close proximity to a Radiological Buffer Area (RBA).
- Routine workplace air sampling to document radiological conditions/changes, detect gradual buildup of radioactive material, verify engineering controls and identify likely sources of airborne exposure to radioactive material. Routine workplace air sampling in non-Airborne Radioactivity Areas provides a general assurance that workers are not chronically exposed to airborne levels ≥10% of a DAC.

ESH-HPT-2000-00064

  - Radiological Work Permit (RWP) System

773A Radiation Survey Logsheets



OSR 4-17 (Rev 3-89)

RADIATION SURVEY LOGSHEET - GENERAL

SURVEY OFFICE: A-1100

DATE OF SURVEY: 5-1-91

JOB LOCATION: C-WING DAILY ROUTINES

BLDG NO.: 773-A | LEVEL: 1ST | DEPARTMENT: HPO

DPSOL OR JOB PLAN NO.: PROCEDURE

INSTRUMENT USED/SERIAL NO.

- ☒ RO2 1235
- ☒ ~~HP 210~~ TMAC 4379
- ☐ TELETECTOR
- ☒ SOURCE CHECK ~~4~~
- ☐ EBERLINE
- ☒ LUDLUM 4934
- ☐ NEUTRON
- ☒ P.T. SMEARS

AIR SAMPLER/SERIAL NO.

- ☐ STAPLEX ☐ IMPACTOR
- ☐ HV CAM ALPHA ☐ KANNE
- ☒ NA

TIME SPENT ON JOB: 45 MINS

TIME SURVEYED: 8:30 (AM / PM)

EXPOSURE RATE ESTABLISHED

A ___ mrad/mR/hr @
B ___ mrad/mR/hr @
C ___ mrem/mR/hr @
D ___ x 10^{-12} ☐ µCi Pu/cc of air; ~~x 10^{-6}~~ ☐ µCi EU/cc of air; x 10^{-9} ☐ µCi F.P./cc of air
E ___ µCi HTO/hr / ___ x 10^{-5} µCi ^{3}H/cc

TRANSFERABLE CONTAMINATION DETECTED

<200 d/m α <3000 d/m β ☒ PTS/.1 m² ☐ DS/100 cm²

PROTECTIVE CLOTHING

- ☐ PLASTIC SUIT ☐ 6Mil ☐ 9Mil ☐ 12Mil
- ☐ COVERALLS ☐ Cotton ☐ Tyvek
- ☒ GLOVES ☐ Cotton ☒ Rubber
- ☐ CLOTH BOOTS
- ☒ SHOE COVERS ☒ Plastic ☐ Rubber
- ☒ LAB COAT
- ☐ HOOD
- ☐ CAP
- ☐ FULL FACE RESPIRATOR
- ☐ HALF FACE RESPIRATOR
- ☐ FRESH AIR HOOD
- ☐ ACID SUIT
- ☒ OTHER SAFETY GLASSES

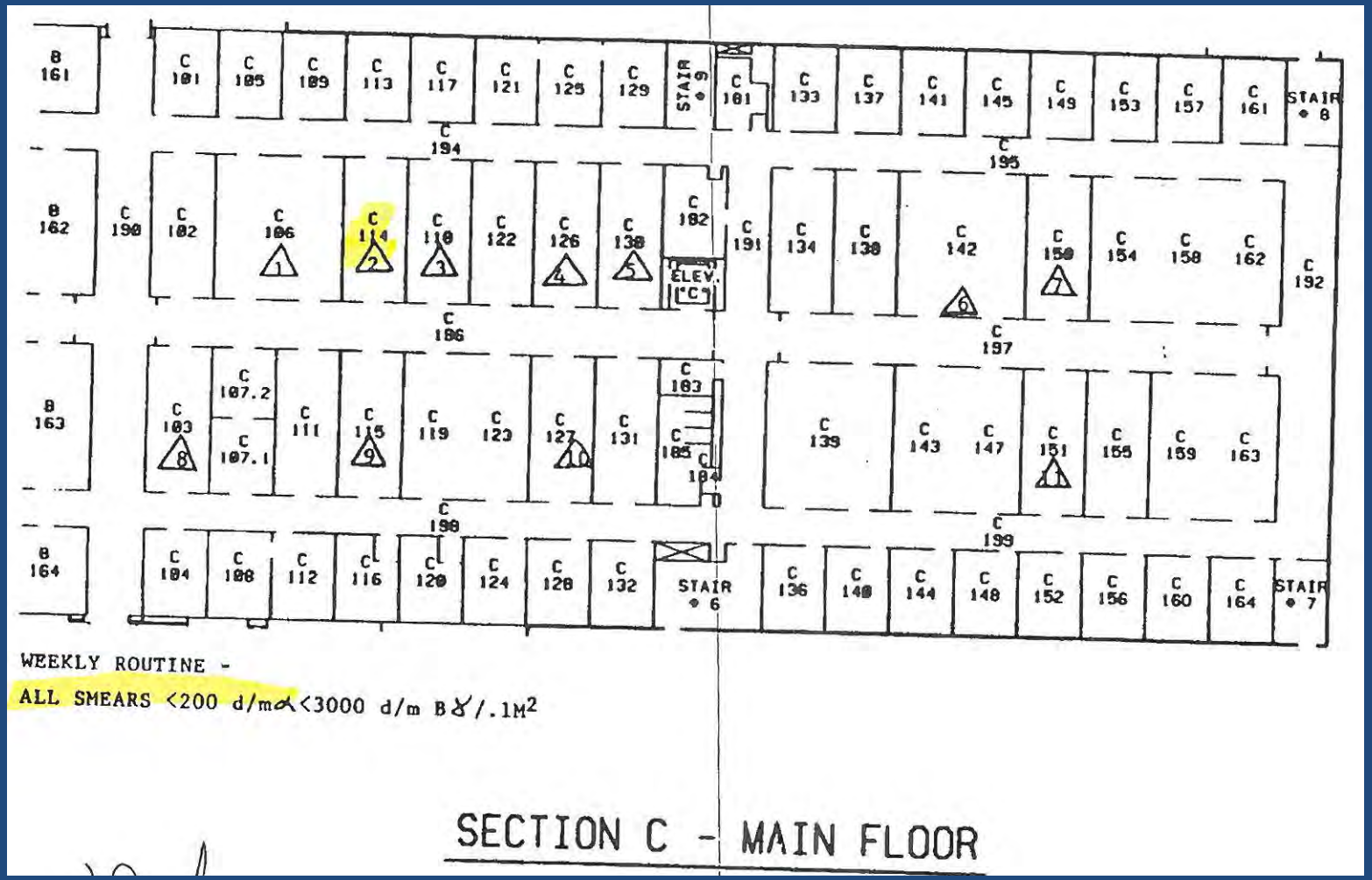
ETE: 0

DESCRIPTION OF SURVEY

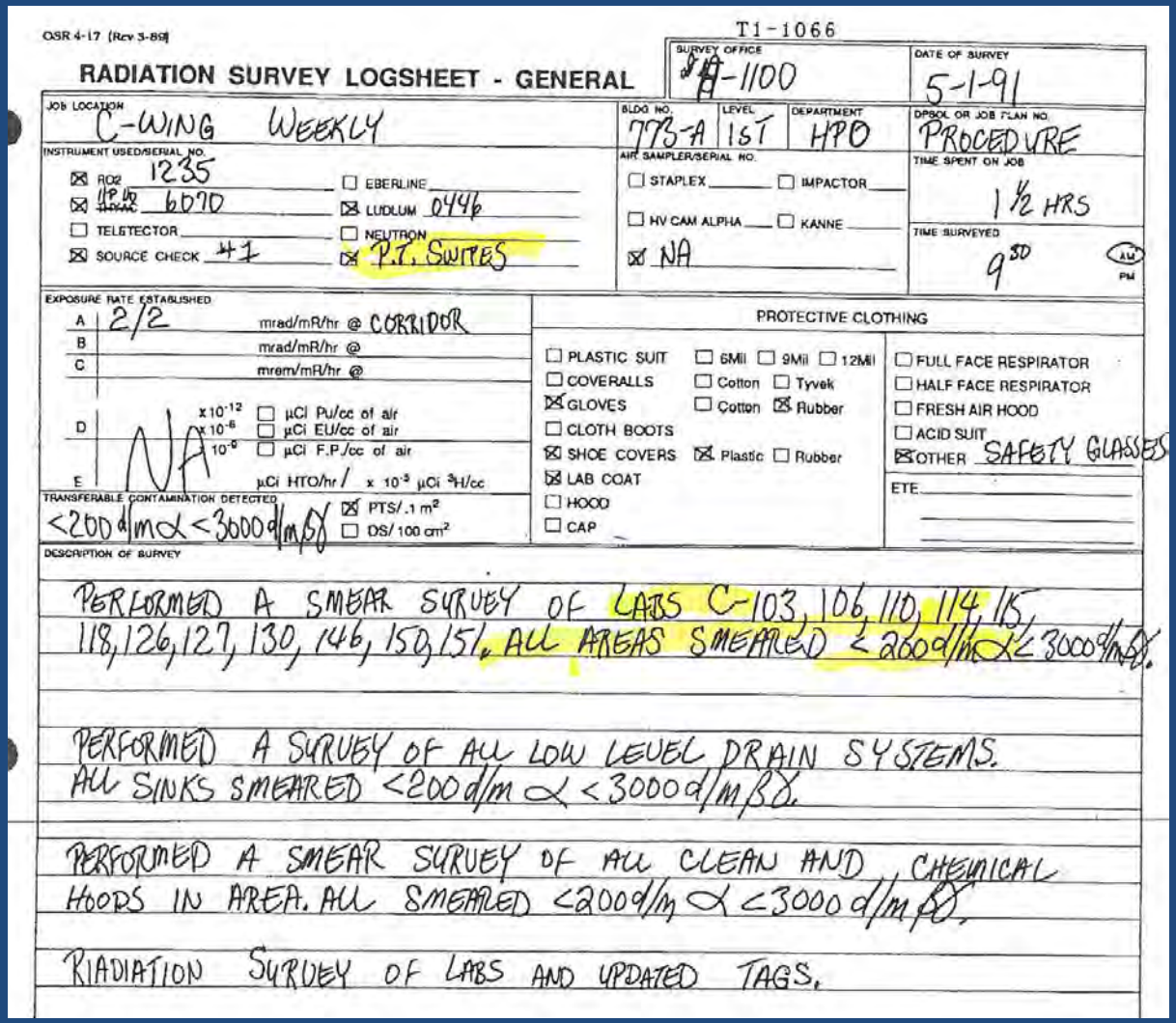
DAILY ROUTINE INSPECTION OF GLOVES & VACUUM GAUGES ON GLOVE BOXES IN USE.
ALL GAUGES SET BETWEEN .25" TO .75" H2O
P.T. SMEARS AS STATED ABOVE

SMEAR SURVEY OF CONTROL POINTS, STEP-OFF PADS & LABORATORIES IN C ~~AN~~ SECTIONS.
SEE ATTACHED SHEET FOR LOCATIONS.

773A - Contamination Surveys



773A Radiation Survey Logsheet



OSR 4-17 (Rev 3-89)

T1-1066

RADIATION SURVEY LOGSHEET - GENERAL

SURVEY OFFICE: #A-1100

DATE OF SURVEY: 5-1-91

JOB LOCATION: C-WING WEEKLY

BLDG NO.: 773-A | LEVEL: 1ST | DEPARTMENT: HPO

DPSOL OR JOB PLAN NO.: PROCEDURE

INSTRUMENT USED/SERIAL NO.

- ☒ RO2 1235
- ☒ ~~HP 49~~ 6070
- ☐ TELETECTOR
- ☒ SOURCE CHECK #1
- ☐ EBERLINE
- ☒ LUDLUM 044b
- ☐ NEUTRON
- ☒ P.T. SWIPES

AIR SAMPLER/SERIAL NO.

- ☐ STAPLEX ☐ IMPACTOR
- ☐ HV CAM ALPHA ☐ KANNE
- ☒ NA

TIME SPENT ON JOB: 1 ½ HRS

TIME SURVEYED: 9:30 AM

EXPOSURE RATE ESTABLISHED

A	2/2	mrad/mR/hr @ CORRIDOR
B		mrad/mR/hr @
C		mrem/mR/hr @
D	NA	x 10^{-12} ☐ µCi Pu/cc of air x 10^{-6} ☐ µCi EU/cc of air 10^{-9} ☐ µCi F.P./cc of air
E		µCi HTO/hr / x 10^{-3} µCi ^{3}H/cc

TRANSFERABLE CONTAMINATION DETECTED: <200 d/m α <3000 d/m β ☒ PTS/.1 m² ☐ DS/100 cm²

PROTECTIVE CLOTHING

- ☐ PLASTIC SUIT ☐ 6Mil ☐ 9Mil ☐ 12Mil
- ☐ COVERALLS ☐ Cotton ☐ Tyvek
- ☒ GLOVES ☐ Cotton ☒ Rubber
- ☐ CLOTH BOOTS
- ☒ SHOE COVERS ☒ Plastic ☐ Rubber
- ☒ LAB COAT
- ☐ HOOD
- ☐ CAP
- ☐ FULL FACE RESPIRATOR
- ☐ HALF FACE RESPIRATOR
- ☐ FRESH AIR HOOD
- ☐ ACID SUIT
- ☒ OTHER SAFETY GLASSES

ETE

DESCRIPTION OF SURVEY

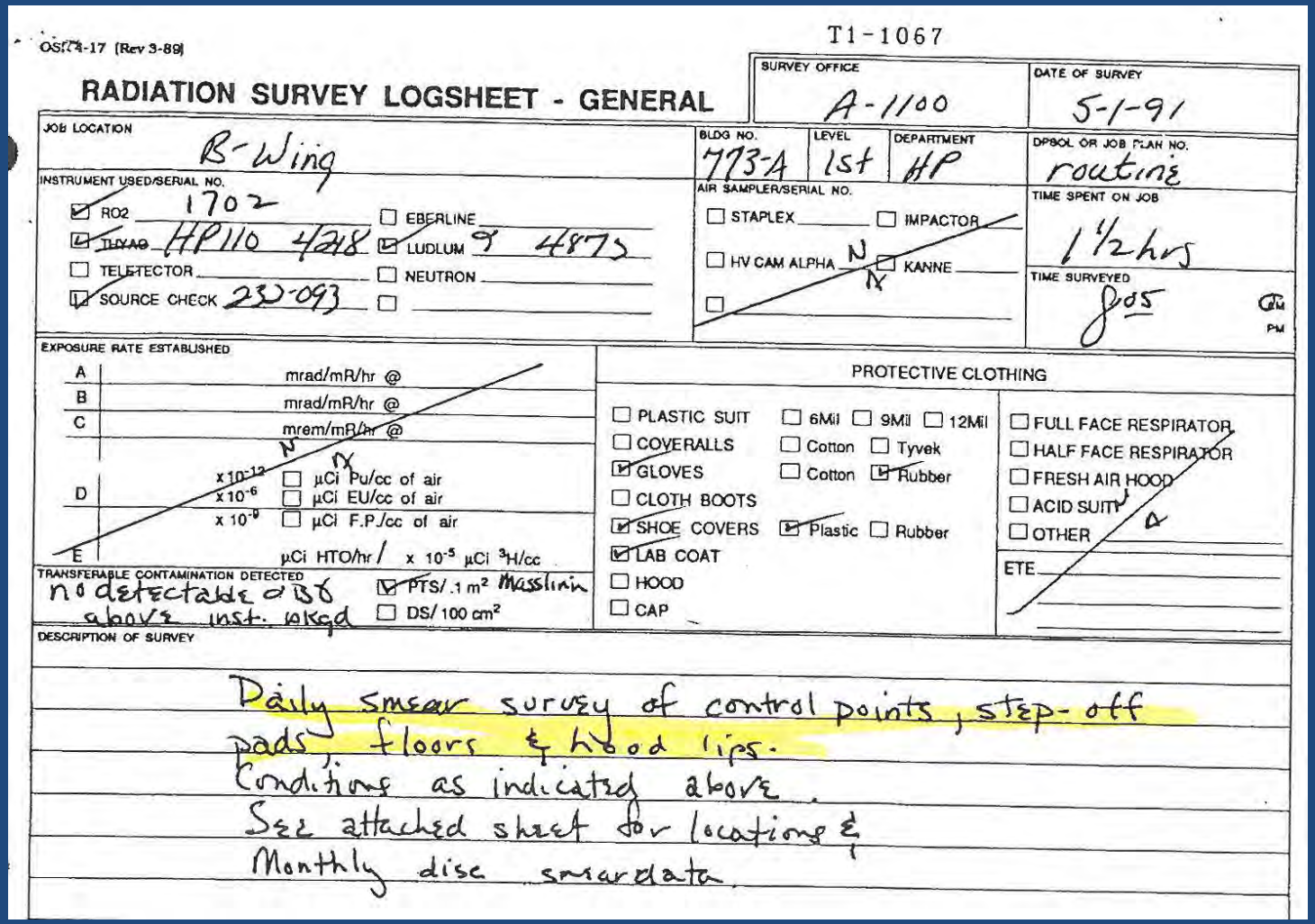
PERFORMED A SMEAR SURVEY OF LABS C-103, 106, 110, 114, 115, 118, 126, 127, 130, 146, 150, 151, ALL AREAS SMEARED < 200 d/m α < 3000 d/m β.

PERFORMED A SURVEY OF ALL LOW LEVEL DRAIN SYSTEMS. ALL SINKS SMEARED <200 d/m α < 3000 d/m β.

PERFORMED A SMEAR SURVEY OF ALL CLEAN AND CHEMICAL HOODS IN AREA. ALL SMEARED <200 d/m α < 3000 d/m β.

RIADIATION SURVEY OF LABS AND UPDATED TAGS.

773A Radiation Survey Logsheets



T1-1067

OSR 4-17 (Rev 3-89)

RADIATION SURVEY LOGSHEET - GENERAL

SURVEY OFFICE	DATE OF SURVEY
A-1100	5-1-91

JOB LOCATION: B-Wing

BLDG NO.	LEVEL	DEPARTMENT	DPSOL OR JOB PLAN NO.
773-A	1st	HP	routine

INSTRUMENT USED/SERIAL NO.

- ☑ RO2 1702
- ☑ ~~TLVAD~~ HP110 4218
- ☐ TELETECTOR
- ☑ SOURCE CHECK 23-093
- ☐ EBERLINE
- ☑ LUDLUM 9 4873
- ☐ NEUTRON
- ☐

AIR SAMPLER/SERIAL NO.

- ☐ STAPLEX
- ☐ IMPACTOR
- ☐ HV CAM ALPHA N N
- ☐ KANNE
- ☐

TIME SPENT ON JOB: 1½ hrs

TIME SURVEYED: 805 AM PM

EXPOSURE RATE ESTABLISHED

A	mrad/mR/hr @	
B	mrad/mR/hr @	
C	mrem/mR/hr @	
D	x 10^{-12} ☐ µCi Pu/cc of air	N N
	x 10^{-6} ☐ µCi EU/cc of air	
	x 10^{-9} ☐ µCi F.P./cc of air	
E	µCi HTO/hr / x 10^{-5} µCi ^{3}H/cc	

TRANSFERABLE CONTAMINATION DETECTED: no detectable α BG above inst. MDA'd

- ☑ ~~PTS~~/.1 m² Masslinn
- ☐ DS/100 cm²

PROTECTIVE CLOTHING

- ☐ PLASTIC SUIT ☐ 6Mil ☐ 9Mil ☐ 12Mil
- ☐ COVERALLS ☐ Cotton ☐ Tyvek
- ☑ GLOVES ☐ Cotton ☑ Rubber
- ☐ CLOTH BOOTS
- ☑ SHOE COVERS ☑ Plastic ☐ Rubber
- ☑ LAB COAT
- ☐ HOOD
- ☐ CAP

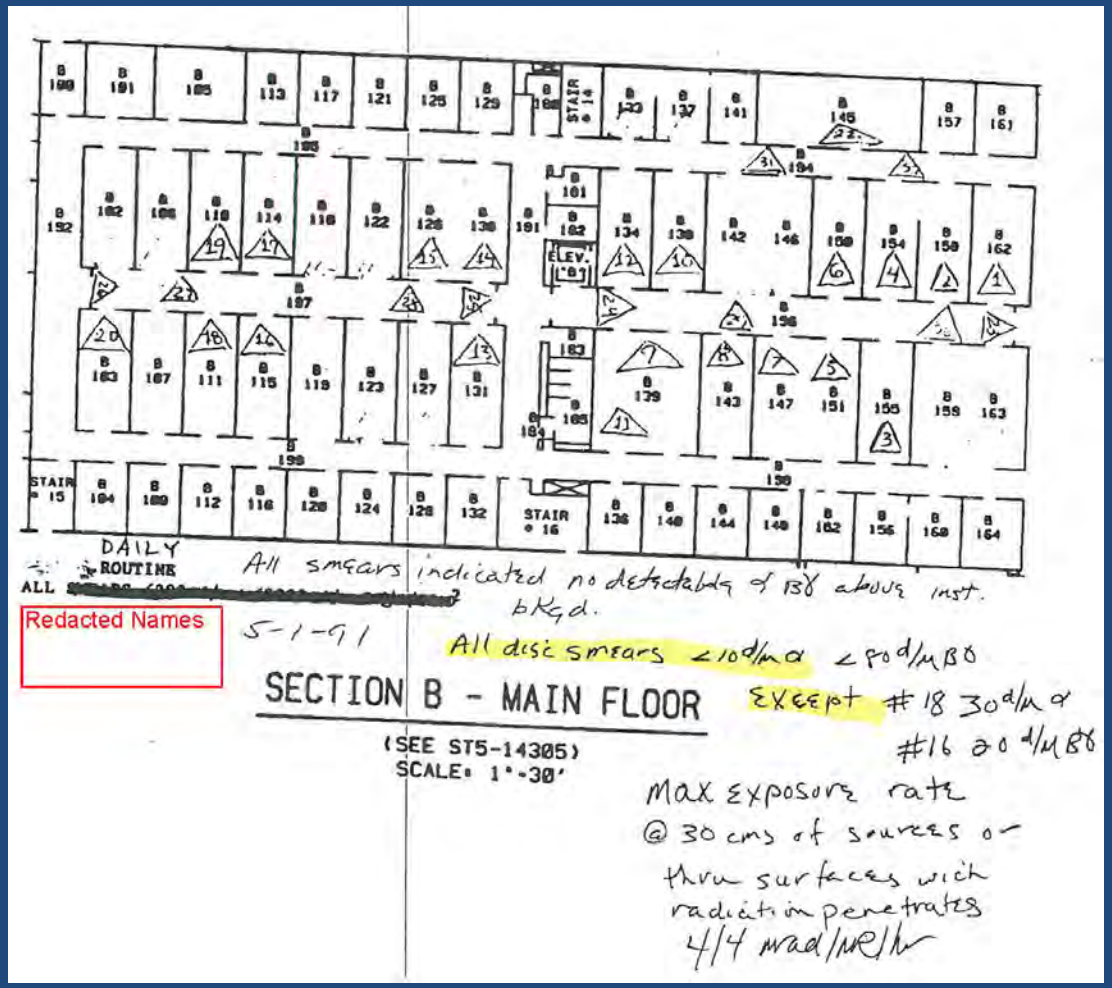
- ☐ FULL FACE RESPIRATOR
- ☐ HALF FACE RESPIRATOR
- ☐ FRESH AIR HOOD
- ☐ ACID SUIT
- ☐ OTHER

ETE

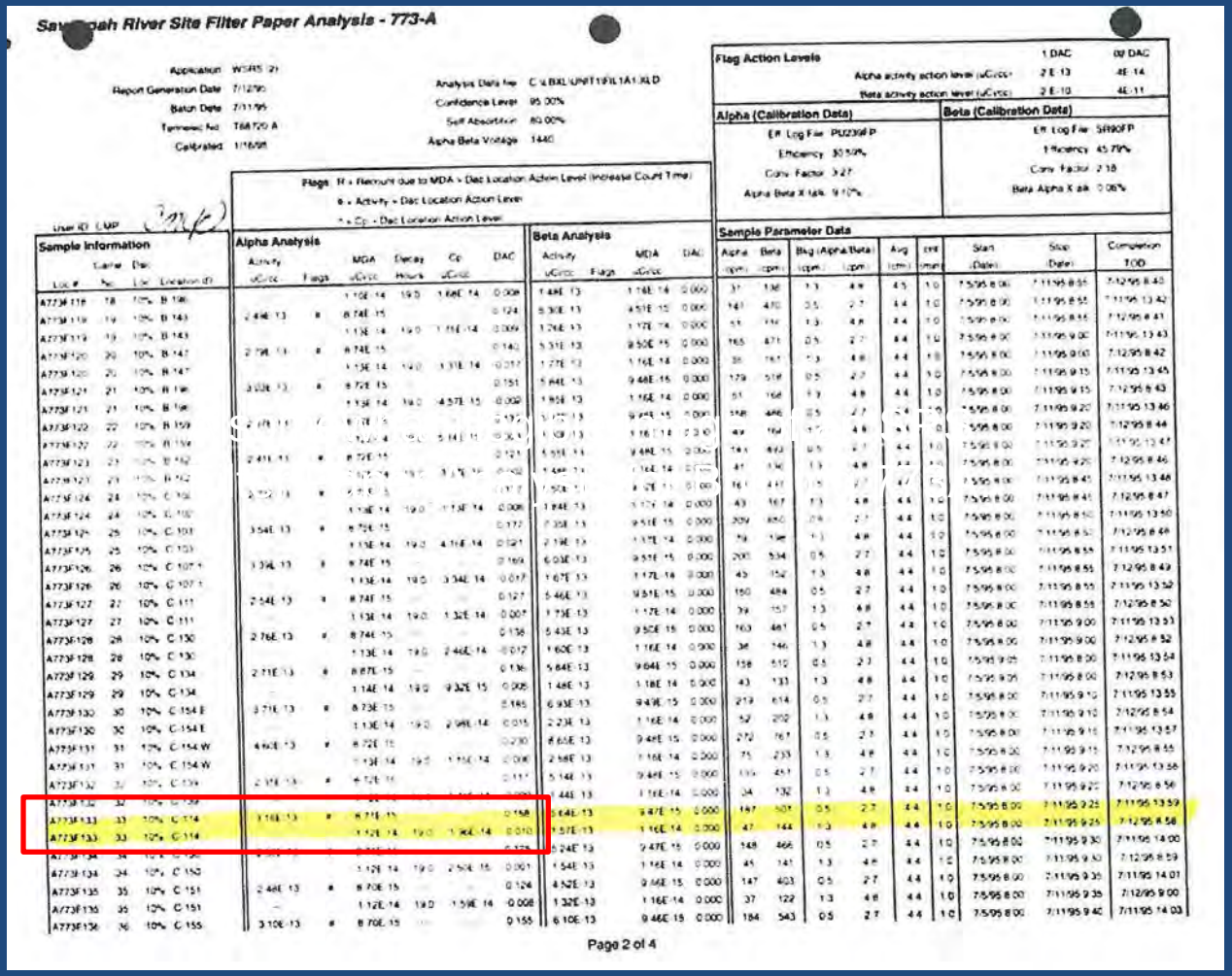
DESCRIPTION OF SURVEY

Daily smear survey of control points, step-off pads, floors & hood lips.
Conditions as indicated above.
See attached sheet for locations & Monthly disc smear data.

773A Radiation Survey Logsheets



773A Air Monitoring (Weekly)



Savannah River Site Filter Paper Analysis - 773-A

Page 2 of 4

773A - Air Monitoring (Weekly)

- **Sample Data**
  - **Start: 7/5/1995 8:00**
  - **Stop time: 7/11/95 9:25**
  - **Average (cfm): 4.4**

	Sample Information				Alpha Analysis						
	Loc #	Carrier No.	DAC Loc	Location ID	Activity uCi/cc	Flags	MDA uCi/cc	Decay Hours	Cp uCi/cc	DAC	Completion TOD
Initial	A773F133	33	10%	C-114	3.16E-13		8.71E-15	-	-	0.158	7/11/95 13:59
24 Hour	A773F133	33	10%	C-114	-		1.12E-14	19.0	1.96E-14	0.010	7/12/95 8:59

Initial
24 Hour

DSR 3-4A-W2 (Rev 1-89)

WESTINGHOUSE SAVANNAH RIVER COMPANY
INTER-OFFICE MEMORANDUM

RECORDS ADMINISTRATION
R0720913

November 15, 1996

ESH-HPT-96-0208
Page 1 of 2

To: **C.G. Brown, 773-A, A1113**
Manager, SRTC Radiological Control Operations

From: **C.R. Morgan, 773-61A**
Health Physics Technology
Technical Support Group

SRTC - BUILDING 773-A, LAB C-114 - GLASS FURNACES OUTSIDE RADIOHOODS HEALTH PHYSICS TECHNOLOGY COMMENTS (U)

As requested [1], we have evaluated the potential for volatilization of uranium and thorium compounds while making samples of glass in furnaces in Lab C-114 of Building 773-A. The furnaces are on bench tops in the lab and not in radiohoods. The ultimate concern is the potential for airborne radioactivity in the lab.

Summary

Airborne or even significant surface radioactive contamination has not been and should not be a problem in Lab C-114 in its current role for development work making surrogate glass samples.

Discussion

Preparation of surrogate glass samples using uranium and thorium oxides has been going on in this lab for some time with the furnaces on bench tops outside the radiohoods. From communications with D.K. Peeler, the researcher for the work, the uranium oxide and thorium oxide used in the glasses are high purity, reagent grade chemicals [2]. Therefore, radionuclide impurities are not a concern. Also, the oxides of uranium and thorium are very stable at the temperatures used in the furnaces [2]. Volatilization of uranium or thorium will not occur.

Historical radiological survey and air sample data for the lab were reviewed, and new smears were taken on upper, exterior surfaces of the furnaces where hot air escapes during their operation. All smear data meet clean area limits. Air entering the lab from the outside corridor moves past the furnaces on the way to the radiohoods where it is exhausted from the room. The air sampler in the lab is positioned next to one of the hoods and should detect airborne activity which might escape the furnaces. The past nine months of air sample data show only background activity levels. This is a clean, well run laboratory with no history of radiological problems [3].

The evidence noted above supports operation of the furnaces on benchtops to make these surrogate glass samples. HPT supports the continued use of the lab in its current mission with the furnaces outside the radiohoods.

References

1. cc:Mail from S.D. Hyman, 10/29/96.
2. cc:Mail from and phone conversations with D.K. Peeler, various dates.
3. Discussions with Radiological Control Operations personnel.

773A Lab C-114 Evaluation

Radiological Work Permits (RWP)

- RWP Program implemented in 1991
  - Official Use Only - (Personal Privacy Data)
  - Section 1 – Requestor
    - Task to be performed and location
  - Section 2 - Radiological Control Operations
    - Monitoring requirements
  - Section 3 - Approvals
  - RWP Sign in sheets contain:
    - RWP number, Name, SSN, Dept./Craft, Work location, Time in, Time out

RSL, ASL, RWPs Availability

- NIOSH has only collected a few examples
  - RSL, ASL, and RWP are typically filed together through 2003 time frame
- In EDWS two records sets have been identified
  - QH Series - 2695 records/boxes
  - QR Series - 7651 records/boxes
    - Within QR series QR600 pertains to SRTC (773A)
    - SRTC = 407 records/boxes
      - Some records/boxes contain 200 pages (folder)
      - Some records/boxes contain 2500 pages (box)

Example Dose Reconstruction

- **Assign intakes based on 2E-13 uCi/cc air concentration if the worker was assigned to 773A**

Thorium Dose (1990-1997)
The thorium exposure and intake for this period were based on 10% of the plutonium derived air concentration (DAC) of 2.0 E-12 microcuries per cubic centimeter, a breathing rate of 1.2 cubic meters per hour and a 2000 hour work year. The resultant annual inhalation intake rate for thorium was 1070 dpm per year. In addition, ingestion intake rate of 22.2 dpm per year was considered in accordance with Technical Information Bulletin: Estimation of Ingestion Intakes. The thorium dose was based on the assumption of 100% Th-232 as the most claimant-favorable radionuclide.

Missed Thorium Dose (Based on 10% Pu DAC)

Description	Thorium M	Thorium S
Bone	**5.617**	0.395
Kidney	**0.119**	0.009
Lung	0.173	**0.694**
Prostate	**0.010**	0.001
Skin	**0.010**	0.001

Bold characters indicate the dose used in the dose assignment (either Th M total *or* Th S total).

Summary

- Most thorium onsite was waste/storage (>95%)
  - Encapsulated
- Very low unencapsulated inventory (source term)
  - More thorium inventory in 1990s and 2000s than 1970s and 1980s
- Minimal use in certain defined locations
  - Mostly 773-A (especially post 1983)
- Knowledge of the processes
  - AFCT/TFCT (1977-1980) - interviews gram quantities
  - Mostly used as a surrogate in post 1990 for DWPF

Summary – cont.

- Radiological controls
  - Procedures in place, Routine monitoring (Daily, Weekly) of the workplace
  - Survey data and air monitoring data available electronically in pdf format
- 1972-1989 Alternate bioassay data
  - Am, Cm, Cf bioassay was effectively gross alpha analysis that included thorium
  - Doses are reasonable
- 1990-2007 Compliant Radiological Control Program
  - Air controlled to <2.0 E-13 uCi/cc
  - Doses are low