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R. J. Tuttle	6/19/79
M. E. Remley	25 June 79

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ABSTRACT

Data on exposures of employees to internal and external radiation, releases of radioactivity in effluents, in-plant airborne radioactivity, and environmental radioactivity are reviewed for calendar year 1975.

This summary will be used in conjunction with subsequent years' results to determine if (1) there are any upward trends in exposures or effluent releases, (2) exposures and effluent releases could be reduced under the ALARA concept, and (3) equipment for effluent and exposure control is performing properly.

RESERVED FOR PROPRIETARY/LEGAL NOTICES

CONTENTS

INTRODUCTION

As set forth in the Energy Systems Group's special material license[1] as Condition No. 23, "A formal annual report shall be made to the radioisotope review committee of the NSRP reviewing employee exposures and effluent release data to determine (1) if there are any upward trends developing in personnel exposures for identifiable categories of workers or types of operations or effluent releases, (2) if exposures and effluents might be lowered under the concept of as low as reasonably achievable, and (3) if equipment for effluent and exposure control is being properly used, maintained and inspected. This report shall include review of other required audits and inspections performed during the past twelve months and review of the data from the following areas: employee exposures; bioassay results; effluent releases; in-plant airborne radioactivity and environmental monitoring."

This report presents an historical basis for the identification of trends. It should be noted that in some instances both NRC licensed and non-NRC-licensed activities take place in the same building. In these cases, certain measurements (e.g., ventilation air exhaust radioactivity) are not possible to be separated for each type of activity. When this occurs, the values are reported unmodified as measured and conservatively may be attributed wholly to licensed activities.

The following ESG facilities and operations are specifically covered in this report.

1. Fuel Fabrications — Bldg 001 and supporting operations in Bldgs 001 and 004, DeSoto Facility, Canoga Park, CA
2. Atomics International Hot Laboratory — Bldg 020, Santa Susana Field Laboratories, Santa Susana, CA
3. Nuclear Material Development Facility — Bldg 055, Santa Susana Field Laboratories, Santa Susana, CA

I. PERSONNEL DOSIMETRY

Personnel dosimetry techniques generally consist of two types; those which measure incident radiation on the body from external sources (e.g., film badges) and those which measure internal body organ accumulations of radioactivity via inhalation, ingestion, or possibly through cuts or puncture wounds. An attempt has been made to separate the exposure modes as much as possible along these lines to (1) permit an evaluation of the more significant exposure routes, and (2) to allow a differentiation between those exposure sources which are external and controllable in the future and those which may continue to irradiate the body for some time period, regardless of future efforts (i.e., internal body deposits).

A. FILM/TLD DATA

Personnel external radiation exposures for the pertinent activities for the year are presented in Table 1 where the number of individuals with a specific annual dose is shown along with the percentage of employees with each annual dose value or less and the man-mrem contributed by each dose value. These same data are plotted on Figure 1.

B. IN-VIVO LUNG SCANS

Measurements are periodically made of the total radioactive lung burden for specific radioisotopes of those employees who have been or were potentially exposed to radioactive aerosols in the respirably-sized particle range. These measurements are accomplished through the use of a whole body counter.* During the year, twenty-seven lung scans were made for uranium deposition. None of the scans showed any positive results.

C. BIOASSAYS

Bioassays normally consist of analysis of urine and occasionally, fecal samples. Personnel whose work assignments potentially expose them to

*Helgeson Nuclear Services, Inc., Pleasanton, CA

TABLE 1

WHOLE BODY DOSES — 1975
(Sheet 1 of 2)

Dose (mrem)	P (people)	CP (cumulative people)	Cumulative (%)	Man-rem
5	24	24	10.13	0.120
10	25	49	20.68	0.250
15	32	81	34.18	0.480
20	12	93	39.24	0.240
25	30	123	51.90	0.750
30	10	133	56.11	0.300
35	10	143	60.34	0.350
40	5	148	62.45	0.200
45	7	155	65.40	0.315
50	11	166	70.04	0.550
55	2	168	70.89	0.110
60	7	175	73.84	0.420
65	2	177	74.68	0.130
70	1	178	75.11	0.070
75	1	179	75.53	0.075
80	3	182	76.79	0.240
85	2	184	77.64	0.170
95	3	187	78.90	0.285
100	1	188	79.32	0.100
105	3	191	80.89	0.315
110	1	192	81.01	0.110
120	1	193	81.43	0.120
125	1	194	81.86	0.125
130	2	196	82.70	0.260
140	2	198	83.54	0.280
165	1	199	83.97	0.165
175	1	200	84.39	0.175
180	1	201	84.81	0.180
185	1	202	85.23	0.185
205	1	203	85.65	0.205
215	2	205	86.50	0.430
220	1	206	86.92	0.220
225	1	207	87.34	0.225
240	1	208	87.76	0.240
245	1	209	88.18	0.245
275	1	210	88.61	0.275
285	1	211	89.03	0.285

TABLE 1

WHOLE BODY DOSES – 1975
(Sheet 2 of 2)

Dose (mrem)	P (people)	CP (cumulative people)	Cumulative (%)	Man-rem
315	1	212	89.45	0.315
320	1	213	89.87	0.320
340	2	215	90.72	0.680
350	1	216	91.14	0.350
365	1	217	91.56	0.365
375	1	218	91.98	0.375
380	1	219	92.41	0.380
385	1	220	92.83	0.385
390	1	221	93.25	0.390
435	1	222	93.67	0.435
450	1	223	94.09	0.450
510	1	224	94.51	0.510
515	1	225	94.94	0.515
570	1	226	95.36	0.570
670	1	227	95.78	0.670
750	1	228	96.20	0.750
775	1	229	96.62	0.775
805	1	230	97.05	0.805
875	1	231	97.47	0.875
970	1	232	97.89	0.970
1190	1	233	98.31	1.190
1390	1	234	98.73	1.390
1395	1	235	99.16	1.395
1645	1	236	99.58	1.645
1900	1	237	100.00	1.900
			Total	26.700

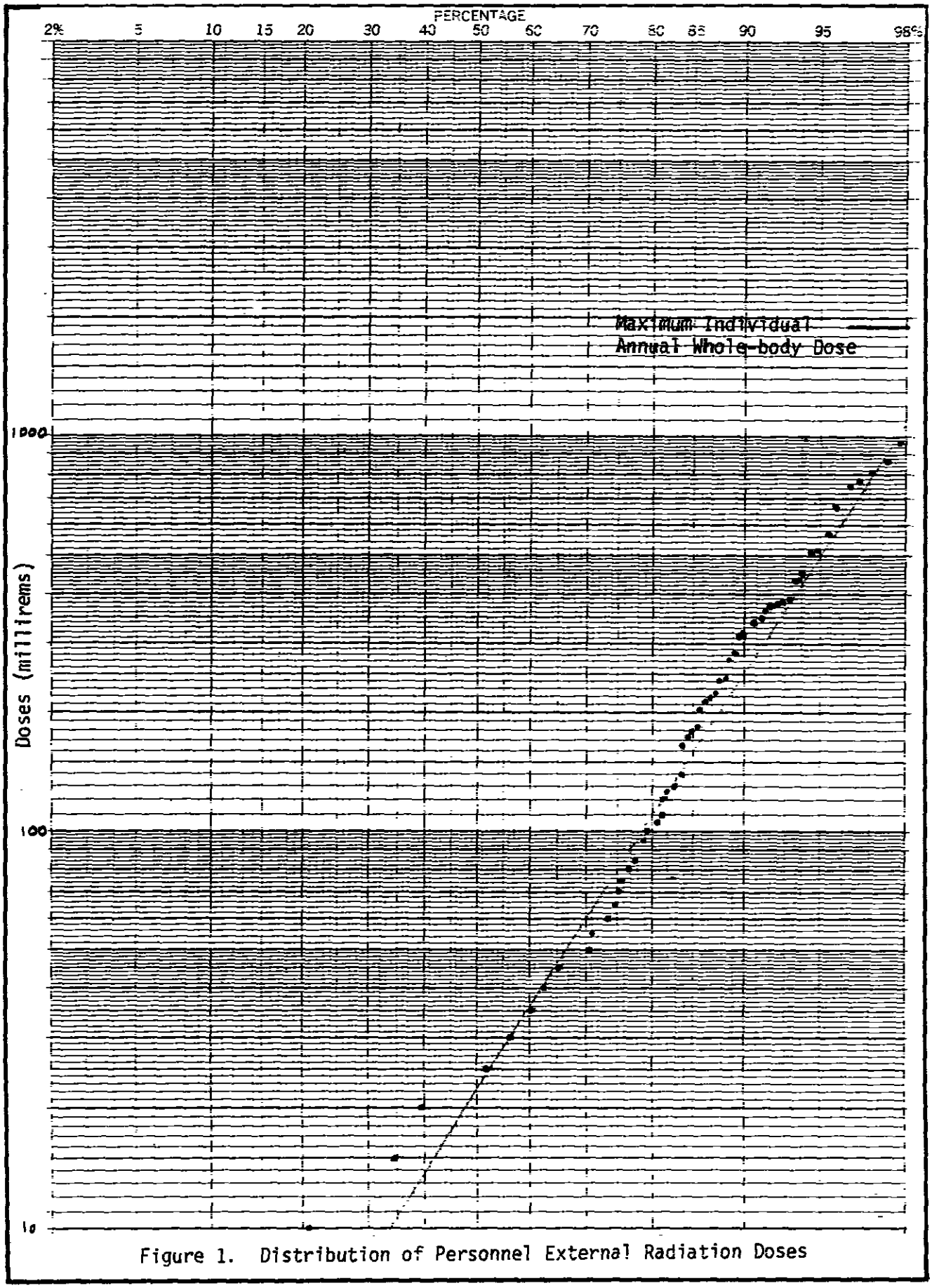

Figure 1. Distribution of Personnel External Radiation Doses

respirable-sized radioactive aerosols are routinely checked in this manner. Normally, urinalyses are performed quarterly and fecal analysis only when gross internal contamination is suspected. The techniques employed are described in Appendix A. A statistical summary of the results for 1975 appears in Table 2A, while a detailed listing of the positive results is shown in Table 2B.

respirable-sized radioactive aerosols are routinely checked in this manner. Normally, urinanalyses are performed quarterly and fecal analysis only when gross internal contamination is suspected. The techniques employed are described in Appendix A. A summary of the results for 1975 appears in Table 2.

TABLE 2A

BIOASSAYS — 1975

Measurement	Type**	Total Tests	Total Positive Results	Total Individuals With Positive Results*
U	UF	345	45	35
U	UR	346	1	1
GA	1A	4	0	0
GB	1B	187	0	0
GB	2B	1	0	0
GB	H	56	0	0
Pu	A	124	6	6
FP	1	9	0	0
FP	3A	194	4	4
FP	3B	190	1	1
U	IVLC	27	0	0

*Tests were repeated on individuals who showed positive results.

**UF = Uranium - fluorometric
UR = Uranium - radiometric
GA = Gross alpha
GB = Gross beta
Pu = Gross plutonium
FP = Fission products
For discussion of analytical techniques, see Appendix A
U-IVLC = Uranium in-vivo lung count

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TABLE 2B

POSITIVE BIOASSAY RESULT SUMMARY - 1975

H&S NUMBER	SAMPLE DATE	ANALYSIS + TYPE	RESULTS PER VOL. ANALYZED	RESULTS PER 1,500 ml-day*	SPECIFIC RADIONUCLIDE	EQUIVALENT MPBB %
0488	09/08/75	U-UF	0.0004 ug	0.6 ug	U	0.6
	11/24/75	U-UF	0.0001 ug		U	
1292	04/07/75	U-UF	0.0004 ug	0.6 ug	U	0.6
	07/28/75	U-UF	0.0 ug		U	
3631	07/08/75	U-UF	0.0005 ug	0.75 ug	U	0.8
	08/04/75	U-UF	0.0003 ug	0.45	U	0.45
3762	01/13/75	U-UF	0.0004 ug	0.6 ug	U	0.6
	05/19/75	U-UF	0.0001 ug		U	
	10/27/75	U-UF	0.0005 ug	0.75 ug	U	0.75
3884	04/20/75	U-PU	0.045 dpm	0.0675 dpm	Pu^{239}	0.0678
	06/08/75	U-PU	0.012 dpm		Pu^{239}	
3954	10/14/75	U-UF	0.0004 ug	0.6 ug	U	0.6
3742	03/18/75	U-PU	0.046 dpm	0.069 dpm	Pu^{239}	0.06
	04/20/75	U-PU	0.0 dpm		Pu^{239}	
3779	03/23/75	U-PU	0.039 dpm	0.062 dpm	Pu^{239}	0.05
	04/13/75	U-PU	0.0 dpm		Pu^{239}	
3969	10/13/75	U-UF	0.0007 ug	1.05 ug	U_{235}	1.1
	10/24/75	IVLC	0 ugm		U^{235}	
1473	08/18/75	U-UF	0.0004 ug	0.60 ug	U	0.6

TABLE 2B

POSITIVE BIOASSAY RESULT SUMMARY - 1975

(Continued)

H&S NUMBER	SAMPLE DATE	ANALYSIS TYPE +	RESULTS PER VOL. ANALYZED	RESULTS PER 1,500 ml-day*	SPECIFIC RADIONUCLIDE	EQUIVALENT MPBB %
3774	05/23/75	U-UF	0.0003 ug	0.45 ug	U	0.4
	10/27/75	U-UF	0.0 ug		U	
3286	09/08/75	U-UF	0.0004 ug	0.60 ug	U	0.6
	12/09/75	U-UF	0.0 ug		U	
2312	12/15/75	U-FP3B	10.2 dpm	76.5 dpm	Cs^{137}	0.01
3749	09/15/75	U-UF	0.0003 ug	0.45 ug	U_{235}	0.45
	10/24/75	IVLC		42 ug	U^{235}	11.7% MPLB
	11/17/75	U-UF	0.0001 ug		U	
3746	08/18/75	U-UF	0.0004 ug	0.60 ug	U	0.6
	08/25/75	U-UF	0.0 ug		U	
	09/15/75	U-UF	0.0004 ug	0.60 ug	U	0.6
	09/22/75	U-UF	0.0001 ug		U	
4018	10/27/75	U-UF	0.0009 ug	1.35 ug	U	1.35
	01/09/76	U-UF	0.0 ug		U	
0808	07/07/75	U-UF	0.0004 ug	0.60 ug	U	0.6
	08/11/75	U-UF	0.0004 ug	0.60 ug	U	0.6
1863	02/03/75	U-UF	0.0006 ug	0.9 ug	U	0.9
	04/07/75	U-UF	0.0 ug		U	
3945	06/30/75	U-UF	0.0011 ug	1.65 ug	U	1.65
	08/11/75	U-UF	0.0003 ug	0.45 ug	U	0.4
	11/03/75	U-UF	0.0002 ug		U	

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TABLE 2B

POSITIVE BIOASSAY RESULT SUMMARY - 1975

(Continued)

H&S NUMBER	SAMPLE DATE	ANALYSIS TYPE +	RESULTS PER VOL. ANALYZED	RESULTS PER 1,500 ml-day*	SPECIFIC RADIONUCLIDE	EQUIVALENT MPBB %
4019	11/24/75	U-UF	0.0014 ug	2.10 ug	U	2.1
3711	03/13/75	U-UF	0.0010 ug	1.5 ug	U	1.5
	07/08/75	U-UF	0.0004 ug	0.60 ug	U	0.6
	08/11/75	U-UF	0.0004 ug	0.60 ug	U	0.6
3963	08/18/75	U-UF	0.0003 ug	0.45 ug	U	0.4
	10/27/75	U-UF	0.0 ug		U	
0607	08/18/75	U-UF	0.0004 ug	0.60 ug	U	0.6
3679	03/11/75	U-UF	0.0007 ug	1.05 ug	U	1.1
	07/14/75	U-UF	0.0 ug		U	
3578	04/27/75	U-PU	0.0410 dpm	0.0615 dpm	Pu^{239}	0.05
	06/08/75	U-PU	0.0 dpm		Pu^{239}	
0904	11/01/75	U-PU	0.05 dpm	0.077 dpm	Pu^{239}	0.06
	12/15/75	U-PU	0.010 dpm		Pu^{239}	
2040	11/24/75	U-UF	0.0004 ug	0.60 ug	U	0.6
1366	07/21/75	U-UF	0.0008 ug	1.2 ug	U	1.2
	10/27/75	U-UF	0.0007 ug	1.05 ug	U	1.1
1265	07/07/75	U-UF	0.0004 ug	0.60 ug	U	0.6
0896	08/05/75	U-UF	0.0004 ug	0.6 ug	U	0.6
4008	11/24/75	U-UF	0.0004 ug	0.60 ug	U	0.6

TABLE 2B

POSITIVE BIOASSAY RESULT SUMMARY - 1975

(Continued)

H&S NUMBER	SAMPLE DATE	ANALYSIS TYPE +	RESULTS PER VOL. ANALYZED	RESULTS PER 1,500 ml-day*	SPECIFIC RADIONUCLIDE	EQUIVALENT MPBB %
2930	03/28/75	U-UF	0.0014 ug	2.1 ug	U	2.1
3754	03/10/75	U-UF	0.0008 ug	1.2 ug	U	1.2
	07/15/75	U-UF	0.0 ug		U	
3946	08/18/75	U-UF	0.0004 ug	0.60 ug	U	0.6
	10/24/75	IVLC		84.0 ug	U^{235}	23.5% MPLB
	11/16/75	F-UF	44.0 ug	0.29 ug/gm	U	
	11/16/75	F-UR	1460 dpm	9.5 dpm/gm	U^{235}	
	01/15/76	F-UF	6.6 ug	0.023 ug/gm	U	
	01/15/76	F-UR	9.90 dpm	0.034 dpm/gm	U^{235}	
1298	03/10/75	U-UF	0.008 ug	1.2 ug	U	1.2
	07/14/75	U-UF	0.0001 ug		U	
3723	09/08/75	U-UF	0.0003 ug	0.45 ug	U	0.4
	10/21/75	U-UF	0.0019 ug	2.85 ug	U	2.8
	10/31/75	U-UF	0.0002 ug		U	
3998	09/29/75	U-FP3A	6.79 dpm	50.9 dpm	MFP	10.6 (Sr^{90})
	12/08/75	U-FP3A	0.47 dpm		MFP	
3970	09/29/75	U-UF	0.0080 ug	12.0 ug	U	12.0
3656	07/07/75	U-UF	0.0008 ug	1.2 ug	U	1.2
	08/04/75	U-UF	0.0003 ug	0.45 ug	U	0.4

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TABLE 2B

POSITIVE BIOASSAY RESULT SUMMARY - 1975

(Continued)

H&S NUMBER	SAMPLE DATE	ANALYSIS TYPE +	RESULTS PER VOL. ANALYZED	RESULTS PER 1,500 ml-day*	SPECIFIC RADIONUCLIDE	EQUIVALENT MPBB %
1547	02/17/75	U-FP3A	5.31 dpm	39.8 dpm	MFP	8.3 (Sr^{90})
	03/10/75	U-FP3A	2.59 dpm		MFP	
2757	07/21/75	U-UF	0.0005 ug	0.75 ug	U_{235}	0.8
	10/24/75	IVLC	45 ug		U^{235}	12.6% MPLB
	10/27/75	U-UF	0.0 ug		U	
3765	01/13/75	U-UF	0.0009	1.35 ug	U	1.4
	04/08/75	U-UF	0.0002 ug		U	
3768	03/12/75	U-FP3A	4.61 dpm	69.2 dpm	MFP	14.4 (Sr^{90})
	04/15/75	U-FP3A	1.67 dpm		MFP	
0544	03/09/75	U-UR	1.05 dpm	15.8 dpm	U^{235}	6.9
	03/09/75	U-UF	0.0060 ug	9.0 ug	U_{235}	9.0
	04/28/75	U-UR	0.104 dpm		U^{235}	
	04/28/75	U-UF	0.001 ug		U	
1876	10/24/75	IVLC	43 ug		U^{235}	12.0 % MPLB

* - Positive result

\+ - U-UF = Urine - Fluorometric Analysis for Uranium
U-UR = Urine - Radiometric Analysis for Uranium
F-UF = Fecal - Fluorometric Analysis for Uranium
F-UR = Fecal - Radiometric Analysis for Uranium
U-PU = Urine - Autoradiographic Analysis for Plutonium
U-FP3A-B = Urine - Radiometric Analysis for Fission Products
IVLC = In-Vivo Lung Count

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II. ON-SITE RADIATION/RADIOACTIVITY LEVELS

Radiation and/or radioactivity level measurements normally consist of three possible aspects, general area, air, and water.

A. AREA

Exposure rates in working and storage areas are periodically measured with portable survey instruments. The results of these surveys are used to establish the necessary posting requirements and for controlling personnel exposure. Film badges are mounted in selected locations to provide an additional record of exposure, integrated over quarterly periods.

B. AIR

Radioactivity in air is measured by fixed-location air samples and air monitors and by lapel (breathing zone) air samples.

C. SPECIAL AIR SAMPLING — Bldg 055

In Bldg 055, air samples are routinely taken near a number of points adjacent to the glove box train as well as other pertinent locations (stack, etc). These data are tabulated in Table 3 in descending order of magnitude and the date of each measurement is noted. In this manner, any unusual release is more apparent. For example, samples taken the week ending October 3 look suspiciously high, at least in a relative sense.

TABLE 3

AIR SAMPLES
NMDF — BUILDING 055
1975

Sampling Location	Maximum Cumulative Weekly Exposure * $\frac{uCi\text{-}h}{cc}$	Week Ending
GBR 26S	3.8×10^{-12}	10/03
GBR 24SW	3.5×10^{-12}	10/03
GBR Waste Storage	2.4×10^{-12}	10/03
GBR 27N	1.8×10^{-12}	10/03
RADECO "B" airmonitor	1.8×10^{-12}	10/03
GBR 24N/E	1.5×10^{-12}	10/03
GBR 27S	1.5×10^{-12}	10/03
GBR 9S	1.5×10^{-12}	09/12
Filter Room	1.5×10^{-12}	05/05
Chem Lab	1.2×10^{-12}	02/07
Support Area	1.2×10^{-12}	09/19
GBR 9W	1.2×10^{-12}	01/31
GBR 1A	1.2×10^{-12}	05/16
GBR 4N	1.2×10^{-12}	11/21
GBR 13N	1.2×10^{-12}	01/03, 01/10, 01/17
GBR 18S	1.2×10^{-12}	06/27, 10/03, 10/10
GBR 20S	1.2×10^{-12}	09/12
GBR 5S	8.9×10^{-13}	12/12
GBR 11N	8.9×10^{-13}	05/16
GBR 11S	8.9×10^{-13}	03/28, 05/16, 05/23
GBR 19S	8.9×10^{-13}	09/19 and 10/10
GBR 21S	8.9×10^{-13}	10/17, 11/07, 12/12
GBR 3S	5.9×10^{-13}	06/06, 06/13, 12/19
GBR 6N	5.9×10^{-13}	12/05
SS Vault	5.9×10^{-13}	01/24, 01/31, 04/25, 08/22
GBR 8N	$\sim 3 \times 10^{-13}$	12/12 and 12/19
Eberline "A: air monitor	$\sim 3 \times 10^{-13}$	Many
Stack	$\sim 3 \times 10^{-13}$	Many

*Allowable exposure is 5×10^{-11} μC-h/cc

III. EFFLUENT MONITORING

A. FACILITY VENTILATION

Ventilation discharge air is continuously sampled at points downstream of the filters provided and prior to exiting the facility system. Thus, these measurements provide a record of radioactive aerosols that are actually discharged from the facility via this route. A portion of the discharged air is continuously drawn through a high efficiency filter medium which collects any particulate aerosols entrained in the sample. Table 4 presents a summary of average and maximum radioactivity concentrations measured in these systems during 1975.

TABLE 4

BUILDING VENTILATION DISCHARGE AIR RADIOACTIVITY — 1975

Building No.	Concentration		Total Activity Released (Ci)
	Ave. (μCi/cc)	Max. (μCi/cc)	
001 α	$<5.2 \times 10^{-15}$	3.3×10^{-14}	$<3.7 \times 10^{-6}$
β	$<3.6 \times 10^{-15}$	2.4×10^{-14}	$<2.6 \times 10^{-6}$
004 α	$<2.8 \times 10^{-15}$	3.5×10^{-14}	$<5.4 \times 10^{-6}$
β	$<6.2 \times 10^{-15}$	1.3×10^{-13}	$<1.2 \times 10^{-5}$
020 α	$<2.6 \times 10^{-16}$	5.3×10^{-16}	$<1.5 \times 10^{-7}$
β	1.1×10^{-11}	2.7×10^{-10}	6.7×10^{-3}
055 α	$<4.2 \times 10^{-16}$	2.8×10^{-15}	$<1.9 \times 10^{-7}$
β	(Not measured)		

B. LIQUID WASTES

All liquid wastes from buildings which may contain radioactivity are drained to hold-up tanks where they are sampled and the contents assessed for radioactivity prior to disposition. If the activities found are within prescribed limits (10 CFR 20), it is then discharged to the sanitary sewers. If however, these limits are exceeded, the waste material is either sufficiently diluted with clean water to meet the limits or pumped out and disposed through

the use of a commercial radioactive material disposal service. During the period in question, the following liquid wastes were drained to the sewer system:

TABLE 5

R/A LIQUID WASTE DISCHARGED TO THE SEWER SYSTEM — 1975

Building	Point of Release	Approximate Effluent Volume (gal)	Activity Monitored	Approximate MDL (μCi/mℓ)	Annual Average Concentration (μCi/mℓ)	Sample Maximum Observed Concentration (μCi/mℓ)	Total Radioactivity Released (Ci)
001	Retention Tank	17100	α	1.2×10^{-9}	3.6×10^{-7}	1.9×10^{-6}	2.3×10^{-5}
			β	4.1×10^{-9}	3.6×10^{-7}	6.9×10^{-6}	5.6×10^{-5}
004	Proportional Sampler	1694200	α	1.2×10^{-9}	7.2×10^{-9}	5.2×10^{-8}	4.6×10^{-5}
			β	4.1×10^{-9}	4.2×10^{-8}	7.6×10^{-7}	2.7×10^{-4}

The maximum permissible concentrations for discharge to the sewer are 8×10^{-4} μCi/mℓα and 1×10^{-3} μCi/mℓβ.

IV. ENVIRONMENTAL MONITORING*

A. INTRODUCTION

Environmental and facility effluent radioactivity monitoring at Atomics International is performed by the Radiation and Nuclear Safety Group of the Health, Safety, and Radiation Services Department. Soil, vegetation, and surface water are routinely sampled to a distance of 10 miles from Atomics International sites. Continuous ambient air sampling and thermoluminescent dosimetry is performed on site for monitoring airborne radioactivity and site ambient radiation levels. Radioactivity in effluents discharged to the atmosphere from Atomics International facilities is continuously sampled and monitored, to assure that the amounts and concentrations released to unrestricted areas are within appropriate limits, and to identify processes which may require additional engineering safeguards to minimize radioactivity levels in such effluents. In addition, selected nonradioactive constituents in surface water discharged to unrestricted areas are determined.

B. SUMMARY

The random variations observed in the environmental monitoring data indicate that no local source of artificial radioactive material existed in the environs. Additionally, the similarity between on-site and off-site results further substantiate that the contribution to general environmental radioactivity due to operations at Atomics International is essentially zero.

The environmental radioactivity measured and reported herein is attributed to natural sources and to continued fallout of radioactive material from foreign atmospheric testing of nuclear devices.

*A separate and comprehensive environmental radioactivity and facility effluent report is issued annually. The material in this section was largely abstracted from the report for 1975.[2]

The sampling and analytic methods used in the environmental monitoring program for radioactive materials are described in Reference 2.

The average radioactivity concentrations in local soil, vegetation, surface water, and ambient air for 1975 are presented in Tables 7 through 10. In calculating the averaged concentration value for the tables, those individual samples having radioactivity levels less than their minimum detection levels (MDL) are assumed to have a concentration equal to the MDL. This method of data averaging, required by ERDA Manual Chapter 0513, affords a significant level of conservatism in the data, as evident in the tables, in that most radioactivity concentrations are reported as "less than" (<) values. Thus, for measurements in which some apparent radioactivity concentrations are below the MDL, the true averaged value is somewhat less than the value reported.

The maximum level of radioactivity detected for a single sample is reported because of its significance in indicating the existence of a major episode or area-wide location of radioactive material deposition. None of the maximum observed values, which occurred randomly during this year, show a great increase over the average values beyond natural variability. This indicates that no significant event occurred, locally or worldwide, which resulted in a marked increase in local radioactive material deposition.

The results reported in Tables 7 and 8 show no significant difference between on-site and off-site samples. The detected activity is due to a variety of naturally occurring radionuclides, and to radioactive fallout resulting from dispersal of nuclear weapons materials and fission products by atmospheric testing. Naturally occurring radionuclides include Be^{7}, K^{40}, Rb^{87}, Sm^{147}, and the uranium and thorium series (including the inert gas radon and its radioactive daughters). Radioactivity from fallout consists primarily of the fission products Sr^{90}-Y^{90}, Cs^{137}, and Pm^{147}, and also U^{235} and Pu^{239}.

Process water used at the NDFL is obtained from Ventura County Water District No. 8, which also supplies nearby communities, and is distributed on-site by the same piping system previously used when process water was

obtained from on-site wells. Conversion was completed during 1969. Pressure is provided by elevated storage tanks and storage reservoirs.

Water from the pipe system is sampled monthly at two locations. The average process water radioactivity concentration is presented in Table 6.

TABLE 6

NDFL PROCESS WATER RADIOACTIVITY DATA — 1975

Area	Activity	No. Samples	Gross Radioactivity (μCi/mℓ)	
			Average Value (95% Confidence Level)	Maximum* Observed Value
NDFL	α	24	$(<2.4 \pm 2.7)\ 10^{-10}$	5.5×10^{-10}
	β	24	$(2.3 \pm 0.7)\ 10^{-9}$	3.2×10^{-9}

*Maximum value observed for single sample

Surface waters discharged from NDFL facilities and the sewage plant effluent drain southward into a retention pond on Rocketdyne property. When full, the pond may be drained into Bell Creek, a tributary of the Los Angeles River in the San Fernando Valley, Los Angeles County. Pursuant to the requirements of Los Angeles Regional Water Quality Control Board Resolution 66-49 of September 21, 1966, a sampling station for evaluating environmental radioactivity in Bell Canyon was established in 1966 approximately 2.5 miles downstream from the southern Rockwell International Corporation boundary. Samples, obtained and analyzed monthly, include stream bed mud, vegetation, and water. Average radioactivity concentrations in Rocketdyne ponds and Bell Creek samples are presented in Table 9 .

Comparison of the radioactivity concentrations in water from the ponds and from Bell Creek with that of the supply water shows no significant variation in either alpha or beta activity.

TABLE 7

SOIL RADIOACTIVITY DATA — 1975

Area	Activities	No. Sample	Gross Radioactivity (μCi/gm)	
			Annual Average Value (95% Confidence Level)	Maximum Observed Value*
On Site	α	144	$(5.7 \pm 1.4)\ 10^{-7}$	1.0×10^{-6}
	β	144	$(2.5 \pm 0.1)\ 10^{-5}$	3.5×10^{-5}
Off Site	α	48	$(5.8 \pm 1.5)\ 10^{-7}$	1.0×10^{-6}
	β	48	$(2.4 \pm 0.1)\ 10^{-5}$	2.7×10^{-5}

*Maximum value observed for single sample.

TABLE 8

VEGETATION RADIOACTIVITY DATA — 1975

Area	Activity	No. Samples	Gross Radioactivity (μCi/gm)		
			Dry Weight Annual Average Value	Ash	
				Annual Average Value (95% Confidence Level)	Maximum Observed Value*
On Site	α	144	$(<3.9 \pm 3.0)\ 10^{-8}$	$(<2.1 \pm 1.6)\ 10^{-7}$	8.4×10^{-7}
	β	144	$(2.6 \pm 0.1)\ 10^{-5}$	$(1.55 \pm 0.03)\ 10^{-4}$	2.61×10^{-4}
Off Site	α	48	$(<4.5 \pm 3.4)\ 10^{-8}$	$(<2.1 \pm 1.6)\ 10^{-7}$	8.9×10^{-7}
	β	48	$(3.0 \pm 0.1)\ 10^{-5}$	$(1.41 \pm 0.03)\ 10^{-4}$	2.40×10^{-4}

*Maximum value observed for single sample.

TABLE 9

BELL CREEK AND ROCKETDYNE SSFL RETENTION POND
RADIOACTIVITY DATA — 1975

Area	Activity	No. Samples	Gross Radioactivity		
			Average Value (95% Confidence Level)	Maximum* Observed Value	% of Guide÷
Bell Creek Mud No. 54 (μCi/gm)	α	12	$(2.9 \pm 1.1)\ 10^{-7}$	4.5×10^{-7}	NA
	β	12	$(2.2 \pm 0.1)\ 10^{-5}$	2.3×10^{-5}	NA
SSFL Pond§ Mud No. 55 (μCi/gm)	α	6	$(6.2 \pm 1.5)\ 10^{-7}$	8.5×10^{-7}	NA
	β	6	$(2.6 \pm 0.1)\ 10^{-5}$	2.7×10^{-5}	NA
Bell Creek Vegetation No. 54 (μCi/gm ash)	α	12	$(<1.9 \pm 1.5)\ 10^{-7}$	3.6×10^{-7}	NA
	β	12	$(1.23 \pm 0.02)\ 10^{-4}$	1.79×10^{-4}	NA
Bell Creek Vegetation No. 54 (μCi/gm dry weight)	α	12	$(<5.0 \pm 3.9)\ 10^{-8}$	1.2×10^{-7}	NA
	β	12	$(3.0 \pm 0.1)\ 10^{-5}$	3.9×10^{-5}	NA
Bell Creek Water No. 16 (μCi/m)	α	12	$(<2.2 = 2.7)\ 10^{-10}$	2.8×10^{-10}	<0.006
	β	12	$(2.4 \pm 0.8)\ 10^{-9}$	3.4×10^{-9}	0.8
SSFL Pond Water No. 6 (μCi/m)	α	12	$(<2.4 \pm 2.7)\ 10^{-10}$	5.5×10^{-10}	<0.006
	β	12	$(4.2 \pm 0.8)\ 10^{-9}$	5.5×10^{-9}	1.4
SSFL Pond Water No. 12 (μCi/m)	α	12	$(<3.1 \pm 2.9)\ 10^{-10}$	1.2×10^{-9}	<0.008
	β	12	$(4.5 = 0.8)\ 10^{-9}$	5.4×10^{-9}	1.5

*Maximum value observed for single sample.
÷Guide: 4×10^{-6} μCi/mℓα, 3×10^{-7} μCi/mℓβ; 10 CFR 20 Appendix B.
§Established July 1975.
NA = Not applicable, no Guide value having been established.

The surface water (NDFL) and the ambient air radioactivity Guide values selected for each site are the most restrictive limits for those radionuclides currently in use at AI facilities. The identity of all such radionuclides is known, irrespective of concentration. Accordingly, for NDFL surface water, the Guide values of 4×10^{-6} μCi/mℓα and 3×10^{-7} μCi/mℓβ, for Pu^{239} and for Sr^{90}, respectively, are appropriate. The correspondingly most restrictive Guide value for Headquarters wastewater radioactivity discharged to the sanitary sewage system, a controlled area, is 8×10^{-4} μCi/mℓα and 1×10^{-3} μCi/mℓβ, for U^{235} and Co^{60}, respectively. The Guide value of 6×10^{-14} μCi/mℓα for NDFL ambient air radioactivity is due to work with unencapsulated plutonium at this site. The Guide value of 3×10^{-11} μCi/mℓβ, for Sr^{90}, is due to the presence of fission products in irradiated nuclear fuel at the site. The Guide value of 3×10^{-11} μCi/mℓβ, for Sr^{90}, is due to the presence of fission products in irradiated nuclear fuel at the site. The Guide value of 3×10^{-12} μCi/mℓα for Headquarters ambient air radioactivity is due to work with unencapsulated uranium (including depleted uranium) at this facility. The Guide value of 3×10^{-10} μCi/mℓβ for Co^{60}, for Headquarters ambient air radioactivity is appropriate since it is the most restrictive limit for beta emitting radionuclides present at the facility. Guide value percentages are not presented for soil or vegetation data since no concentration Guide values have been established.

Ambient air sampling for long-lived particulate alpha and beta radioactivity is performed continuously with automatic sequential samplers at both the Headquarters and NDFL sites. Air is drawn through Type HV-70 filter media which are analyzed, after a minimum 120-h decay period to eliminate the naturally-occurring radon particulate daughters, for long-lived radioactivity. The average concentrations of alpha and beta ambient air radioactivity are presented separately in Table 10.

Radioactivity levels observed in environmental samples for 1975, reported in Tables 7 through 10, compare closely with levels reported for recent years. Local environmental radioactivity levels, which result primarily from beta-emitting radionuclides and had shown the effect of fallout during past extensive

TABLE 10

AMBIENT AIR RADIOACTIVITY DATA — 1975

Site Location	Activity	No. Samples	Average Value (95% Confidence Level)	Maximum* Observed Value (daily)	% of Guide†
Headquarters	α§	709	$(<6.3 \pm 6.8)\ 10^{-15}$	6.0×10^{-14}	<0.21
(μCi/mℓ)	β**	709	$(<7.6 \pm 1.6)\ 10^{-14}$	4.6×10^{-13}	<0.025
NDFL	α§	2477	$(<6.0 \pm 6.3)\ 10^{-15}$	8.8×10^{-14}	<10.0
(μCi/mℓ)	β**	2477	$(<7.3 \pm 1.5\ 10^{-14}$	7.3×10^{-13}	<0.24

*Maximum value observed for single sample.
†Guide: Headquarters - 3×10^{-12} μCi/mℓα, 3×10^{-10} μCi/mℓβ; 10 CFR 20 Appendix B
NDFL - 6×10^{-14} μCi/mℓα, 3×10^{-11} μCi/mℓβ; 10 CFR 20 Appendix B
§MDL = 5.6×10^{-15} Ci/m — Individual daily samples with activity levels of 0 to 5.6×10^{-15} μCi/mℓ are recorded and averaged as 5.6×10^{-15} μCi/mℓ.
**MDL - 1.2×10^{-14} μCi/mℓ — Individual daily samples with activity levels of 0 to 1.2×10^{-14} μCi/mℓ are recorded and averaged as 1.2×10^{-14} μCi/mℓ. Indicated average values are upper limits, since some data were below the minimum detection levels.

atmospheric testing of nuclear devices, have decreased, and have been generally constant during the past several years. The effects of recent, although infrequent, foreign atmospheric nuclear tests continue to be occasionally observable in daily ambient air radioactivity levels; however, the long-term effects on surface sample radioactivity levels are not discernible. The continued relative constancy in current environmental radioactivity is due primarily to the dominance of naturally occurring radionuclides in the environment and to the longer-lived fission product radioactivity from fallout.

Site ambient radiation monitoring is performed with several types of TLD's. Each dosimeter packet includes a single calcium fluoride (CaF_2:Mn) low background bulb-type chip dosimeter which produced the data used in this report, a single calcium fluoride (CaF_2:Mn) bare chip dosimeter, and two calcium sulfate ($CaSO_4$:Dy) low background powder-type dosimeters. The additional chip dosimeter is used for continued development of TLD dosimetry programs at AI. The

powder dosimeters, supplied and evaluated by a commercial radiation dosimetry laboratory, are used as backup to the low background bulb-type dosimeter. The dosimeter sets are placed at selected locations on or near the perimeters of the Headquarters and NDFL sites. Each dosimeter, sealed in a light-proof energy compensation shield, is installed in a polyethylene container mounted at each location. The dosimeters are exchanged and evaluated quarterly. There are 10 on-site TLD monitoring locations. Three additional dosimeter sets are located off-site at locations up to 10 miles from the sites and similarly evaluated to determine the local area ambient radiation level, which averaged 0.011 mrem/h for 1975. The average radiation dose rate and equivalent annual dose monitored at each dosimeter location are presented in Table 11.

TABLE 11

SITE AMBIENT RADIATION DOSIMETRY DATA — 1975

Dosimeter Location	Average Dose Rate (mrem/h)	Equivalent Annual Dose (mrem)
TLD-1 Headquarters	0.012	105
TLD-2 Headquarters	0.009	79
TLD-3 Headquarters	0.011	96
TLD-4 Headquarters	0.012	105
TLD-5 NDFL	0.011	96
TLD-6 NDFL	0.014	123
TLD-7 NDFL	0.013	114
TLD-8 NDFL	0.013	114
TLD-9 NDFL	0.009	79
TLD-10 NDFL	0.011	96
TLD-11 Off-Site	0.010	88
TLD-12 Off-Site	0.012	105
TLD-13 Off-Site	0.010	88

The table shows that radiation dose rates and equivalent annual doses monitored on-site are nearly identical to levels monitored at three widely separated off-site locations. These data include the natural background radiation component, which exists as a consequence of cosmic radiation, radionuclides in the soil, and radon and thoron in the atmosphere, in addition to radioactive fallout from nuclear weapons tests. Locally, this is approximately 100 mrem/year. The small variability observed in the data is attributed to differences in elevation and geologic conditions at the various dosimeter locations. Since the data for the on-site and off-site locations are nearly identical, no measurable radiation dose to the general population or to individuals in uncontrolled areas resulted from AI operations.

V. UNUSUAL EVENTS

On July 29 at the AIHL, a small uranium slug caught fire while being heated under a heat lamp in Cell 2. A high volume nitrogen purge was used to extinguish the fire. The increased inflow of nitrogen briefly over-pressurized the cell and some airborne activity was released to the operating gallery. Approximately 660 μCi of krypton-85 was released by way of the exhaust stack. No exposure or release limits were exceeded.

VI. SUMMARY/TRENDS — EXPOSURES, EFFLUENTS

All personnel and environmental exposures, and effluent releases to uncontrolled areas, were below the allowable limits for 1975. This report establishes baseline data for identification of trends in succeeding years.

VII. ANTICIPATED ACTIVITIES DURING NEXT REPORTING PERIOD

A. Building 001 (Fuel Fabrication Facility)

Fabrication of fuel pins for EBR-II using 67% enriched uranium was completed. Fabrication of ATR fuel using highly enriched UAl_x was begun.

B. Building 004 (Hot Analytical Chemistry)

Analytical chemistry was performed on small samples of EBR-II and ATR fuel materials.

C. Building 020 (Hot Laboratory)

Decladding SRE Core I fuel was completed.

D. Building 055 (Nuclear Materials Development Facility)

The facility was prepared for fabrication of (U, Pu) C fuel elements for irradiation in EBR-II.

REFERENCES

1. U. S. Nuclear Regulatory Commission — Special Nuclear Materials License No. SNM-21, USNRC (September 15, 1977)

2. "Atomics International Environmental Monitoring and Facility Effluent Annual Report — 1975" J. D. Moore, AI-76-14, Atomics International (1976)

APPENDIX A
PERSONNEL MONITORING PROGRAM

Film badges are furnished by a vendor service, the Radiation Detection Company. Kodak type H personnel monitoring film is used. The film badge holder is equipped with plastic, aluminum, cadmium, and lead shields, as well as an "open window" behind which the film is unshielded. Evaluation of radiation dose on the basis of film density requires an interpretation of the type and energy of the radiation involved. This interpretation is made by the differences in the film densities behind these shields.

Two separate calibration energies are used to determine x-rays and gamma doses on the basis of film densities: (1) Co^{60} gamma rays, and (2) 35 keV x-rays obtained from 80 kVp x-rays filtered with 2 mm Al. The effective energy of x-ray or gamma radiation is determined on the basis of the ratios of open window film density to film densities under the different filters as indicated under Appendix 1. If the effective energy of the radiation is determined as <70 keV, the 35 keV x-ray calibration data are used. In this case, the film density of the open window area is converted to dose by means of the 35 keV calibration curve. A correction factor is then applied as determined from Appendix II. For example, if the effective energy is 30-50 kV, the correction factor is 1.0. If the effective energy is 60 keV, the correction factor is 1.1, etc. If the effective energy of the radiation is above 70 keV, the Co^{60} data are used and the density of the film behind the Pb filter is converted to dose by means of the Co^{60} calibration curve.

Beta dose calculations are made by subtracting the density of the film located behind the plastic shield from the density of the film behind the open window, multiplying the remainder by a beta factor, and converting to dose by means of the Co^{60} calibration curve. Each beta factor is specific to a single, known radionuclide. If the radionuclide is unknown, a factor of 1.3 is applied.

Eastman type NTA track plate film is used for neutron monitoring. The film is calibrated with a polonium-beryllium source. High energy neutron exposures are interpreted by counting the number of proton tracks in 25 fields under high-power microscopy and assigning a dose on the basis of the total number of tracks observed.

Thermal neutron exposure is determined to be present when the film density under the cadmium filter is >1.25 times the film density under the lead filter. When such is the case, both density readings are converted to dose from the Co^{60} calibration curve and the dose from the lead filter density is then subtracted from the dose obtained from the cadmium filter density. Half of the remainder is converted directly to dose in rem.

All personal film badges are processed routinely by the AI film badge vendor (Radiation Detection Company) according to the methods described above.

Certain operations, such as hot cell entries, which pose a high exposure potential, require the use of special badges, which are badges worn for a single operation in place of personal badges. When special badges are required, two badges are worn by each individual. Special badges are evaluated according to the method previously described; however, the average reading of the two badges is recorded on the dose. All special badges are processed at AI by the Radiation and Nuclear Safety Group.

In the event of an accidental criticality incident, the film badge holder also contains additional components for the measuring of high level gamma and neutron exposures generally associated in this type incident. Excessive film blackening prevents the microscopic identification of proton tracks. Therefore, neutron exposures above 10 rad are determined by means of sulfur pellets, gold and indium foils, and a copper washer which are incorporated into the film holder.

HIGH LEVEL NEUTRON DETECTORS

Material	Dimensions	Energy Detected	Maximum Sensitivity-n/cm^2
Indium	0.70 in. x 0.70 in. x 0.005 in.	Thermal to 2.0 ev	Approximately 10^4
Sulfur	(Four pills of 9/32-in. diameter) 0.25 gm total	2.9 MeV and above	5×10^7
Copper	Circular Washer	2.0 eV to 1.0 MeV	
Gold (bare)	0.25 in. x 0.25 in. x 0.005 in.	1.0 MeV to 2.9 MeV	2×10^6

The very high thermal neutron sensitivity of indium makes it extremely useful as an exposure indicator. In the event of an accidental criticality the high energy neutrons will be moderated and reflected by the body, thereby producing thermal and intermediate energy neutrons that will activate the indium. By using a G.M. survey instrument, those exposed can be detected for five hours following an incident.

Maximum sensitivity of the film is about 900 R. Since the gamma dose in a criticality incident is liable to be much greater, a LiF TLD (Thermoluminescent Dosimeter) in capsule is also incorporated into the holder. TLD material can measure up to 10^5 R.

In the Film Badge Dosimetry Report, X-ray, gamma, and neutron doses are listed as penetrating radiation, and beta exposure is listed as non-penetrating radiation.

Type of Radiation	Reporting Range	Energy (MeV)
X-Ray	3.5 mR to 900 R	0.020 to 0.250
Gamma	10 mR to 900 R	0.250 to 3.0
Beta	45 mrad to 900 rad	Above 1.0
Fast Neutrons	10 mrem to 50 rem	0.300 to 14.0
Thermal Neutrons	10 mrem to 50 rem	Thermal

The Film Badge Dosimetry report also contains the following information on monitored personnel:

(1) Social Security Number

(2) Name

(3) Date of Birth

(4) Badge Number

(5) Current Dose
X + Gamma, Neutron, Beta

(6) Calendar Quarter Dose
Penetrating, Nonpenetrating

(7) Calendar Year Dose
Penetrating, Nonpenetrating

(8) Lifetime Dose
Penetrating, Nonpenetrating

At the end of the year, Radiation Detection also sends an individual ERDA Form-5 on each person on the film badge roster with a summary of the above information.

779-A.2/jbd/jca

TABLE I

FILTER RATIOS AS A FUNCTION OF EFFECTIVE
X-RAY ENERGY FOR R-D PLASTIC BADGE

keV	Ratios			
	Open Window to Al	Open Window to Plastic	Open Window to Cd	Open Window to Pb
11	15	1.8	-	-
16	2.5	1.2	-	-
21	2.2	1.1	-	-
23	1.9	1.05	-	-
25	1.6	1.05	40	-
30	1.5	1.05	31	-
35	1.25	1.0	8.0	-
44	1.10	1.0	7.0	23
72	1.05	1.0	3.3	10
93	1.0	1.0	2.1	6.5
115	1.0	1.0	2.0	5.4

Note: Filter ratios apply only to linear portion of characteristic curve which is up to about a net density of 1.0. If higher densities are encountered, then the ratio of apparent doses as determined from the characteristic curve must be used.

TABLE II

keV	Factor
11	6.0
16	4.4
21	2.75
26	1.06
30	1.0
44	0.95
72	1.2
93	1.6
115	2.2

keV Energy Range	Factor
30 - 50	1.0
60	1.1
70	1.2
80	1.3

ANALYTICAL PROCEDURE SUMMARY FOR BIOASSAY BY URINALYSIS

The following summary of analytical procedures is limited to the most frequently performed urinalyses for radioactive material.

Uranium-Radiometric and Fluorometric (UR, UF)

Uranium is extracted from an acidic solution of ashed urine using aluminum nitrate, tetrapropyl ammonium hydroxide, and methyl isoburyl ketone. The uranium is recovered by back extracting into water by evaporating to ketone. The water solution is planchetted for alpha counting for the UR analysis. Fluorometric analysis requires that an appropriate aliquot of the water solution be removed prior to planchetting for pelletizing with NaF-LiF. The pellet is then analyzed for uranium with a fluorometer.

Mixed Fission Products (FP1)

Mixed fission products will precipitate from a basic oxalate media. By adjustment of pH and oxalate concentrations, those elements which are amphoteric or which form oxalate complexes in the form of excess oxalate, will also precipitate. Alkali metals such as Cs^{137} will not precipitate. Also, volatile fission products such as I^{131} will be lost.

The precipitate is washed with NaOH and water and planchetted for counting.

Mixed Fission Products (FP2)

Same extraction procedure as FP1, however, the soluble oxalate precipitates are gamma counted for Cs^{137} and other gamma emitters. The results from its FP1 analysis and the FP2 analysis are summed and reported as a single value.

Mixed Fission Products (FP3)

Same as FP2 except that the oxalate insoluble results will be reported separately as FP3a and the oxalate soluble results will be reported separately as FP3b.

Plutonium (PUA, PUB)

After reduction to plutonium (III) and (IV) with hydroxylamine hydrochloride, plutonium is precipitated with lanthanum fluoride. This isolates the plutonium from most elements, including uranium, except thorium, the rare earths and actinides.

After oxidation of plutonium with sodium nitrate in acid media, extraction of plutonium is carried out with 0.5 M thenoyltrifluoro acetone in xylene. Following extraction the aqueous solution containing plutonium is neutralized and concentrated by heating. After oxidation of the plutonium in a basic media, it is electrodeposited on a stainless steel disc. The plutonium activity is determined by autoradiography (PUA) for greater sensitivity, or counted for alpha radiation with a proportional counter (PUB).

Gross Beta, High Level (GBH)

The gross sample is evaporated to dryness, followed by organic digestion by hydrogen peroxide and nitric acid. Natural potassium (K^{40}) correction is determined by diluting the ashed salts to a known volume, and removing an aliquot for flame spectrophotometry. The remaining solution is evaporated to near dryness, planchetted, and counted for beta radiation with a proportional counter. The radioactivity in the urine sample due to K^{40} is subtracted from the gross count.

Gross Alpha (GA1a)

Specific for uranium and/or plutonium which is extracted from ashed urine salts using aluminum nitrate, tetrapropylammonium hydroxide, and methyl isobutyl ketone. Transuranics do not extract to any appreciable extent. Uranium and/or plutonium are recovered by back extracting into water by evaporating the ketone. The uranium and/or plutonium are electrodeposited on a stainless steel disc and autoradiographed.

Gross Alpha (GA1b)

Same as GA1a except the extraction solution is planchetted and counted for alpha radiation with a proportional counter.

Gross Alpha (GA2)

Specific for all alpha emitters. Metabolized actinides are converted to states suitable for coprecipitation with alkaline earth phosphates by digesting the gross urine sample in 10% nitric acid. The actinides are coprecipitated with calcium phosphate by neutralizing the acid solution with ammonia. The precipitate is washed, planchetted, and counted for alpha radiation with a proportional counter.

SUMMARY OF BIOASSAY SERVICES AVAILABLE FROM UNITED STATES TESTING COMPANY, INC.

Analysis Type	Listing Code	Analysis Specific For	Sensitivity/ 1500 ml	Accuracy at Minimum Sensitivity	Minimum Volume Required	Remarks
Plutonium (B)	PU B	Plutonium	0.0495 dpm	± 75%	1000 ml	Double precipitations, washes and extractions are eliminated for faster analysis at reduced accuracy.
Plutonium (B) (Optional)	PU B	Plutonium	0.75 dpm	± 100% alpha counting	1000 ml	Sample proportional counted for Alpha - radiation for immediate result. Sample may be later autoradiographed.
Strontium-90	SR90	Strontium-90	30 dpm	± 50%	200 ml	
Thorium	TH	Thorium	0.99 μg	± 50%	1000 ml	
Gross Beta-High Level	GBH	All beta emitters except halogens	750 dpm	± 75%	50 ml	K^{40} corrected
Gross Alpha (1a)	GA1A	Uranium and Plutonium	1.5 dpm	± 50%	100 ml	Sample electrodeposited on SS disc and autoradiographed.
Gross Alpha (1b)	GA1B	Uranium and Plutonium	9 dpm	± 50%	100 ml	Sample planchetted and proportional counted for alpha.

SUMMARY OF BIOASSAY SERVICES AVAILABLE FROM UNITED STATES TESTING COMPANY, INC.

Analysis Type	Listing Code	Analysis Specific For	Sensitivity/ 1500 ml	Accuracy at Minimum Sensitivity	Minimum Volume Required	Remarks
Gross Alpha (2)	GA2	All other alpha emitters including Th, Pa, U, Np, Pu, Am, Cm, Po, and Ra	15 dpm	± 50%	100 ml	Sample planchetted and proportional counted for alpha
Iodine-131	I 131	Iodine-131	300 dpm	± 50%	250 ml	Decay corrected to sampling date.

Gross Alpha (GA2)

Specific for all alpha emitters. Metabolized actinides are converted to states suitable for coprecipitation with alkaline earth phosphates by digesting the gross urine sample in 10% nitric acid. The actinides are coprecipitated with calcium phosphate by neutralizing the acid solution with ammonia. The precipitate is washed, planchetted, and counted for alpha radiation with a proportional counter.

SUMMARY OF BIOASSAY SERVICES AVAILABLE FROM UNITED STATES TESTING COMPANY, INC.

Analysis Type	Listing Code	Analysis Specific For	Sensitivity/ 1500 ml	Accuracy at Minimum Sensitivity	Minimum Volume Required	Remarks
Fluorometric Uranium	UF	Normal or Depleted Uranium	0.3 μg	± 50%	10 ml	
Radiometric Uranium	UR	Enriched Uranium	7.5 dpm	± 50%	100 ml	
Fission Products (1)	FP 1	Insoluble oxalates including alkaline earths, transition elements, lanthanides, antimony, phosphates. Excludes soluble oxalates i.e. Cs 137	30 dpm	± 50%	200 ml	Volatile fission products lost.
Fission Products (2)	FP 2	Same as FP 1 plus gamma scan on soluble oxalates.	60 dpm	± 50%	300 ml	Results combined into single value for report. Volatile fission products lost.
Fission Products (3)	FP 3	Same as FP 2 with insoluble and soluble oxalate results reported separately as FP 3a and FP 3b respectively.	30 dpm FB3a 60 dpm FB3b	± 50%	300 ml	Volatile fission products lost.
Tritium	H3	Tritium	2.25×10^6 dpm	± 50%	10 ml	
Plutonium (A)	PU A	Plutonium	0.0495 dpm	± 50%	1000 ml	Greater accuracy than PU B analysis.

SUMMARY OF BIOASSAY SERVICES AVAILABLE FROM UNITED STATES TESTING COMPANY, INC.

Analysis Type	Listing Code	Analysis Specific For	Sensitivity/ 1500 ml	Accuracy at Minimum Sensitivity	Minimum Volume Required	Remarks
Plutonium (B)	PU B	Plutonium	0.0495 dpm	± 75%	1000 ml	Double precipitations, washes and extractions are eliminated for faster analysis at reduced accuracy.
Plutonium (B) (Optional)	PU B	Plutonium	0.75 dpm	± 100% alpha counting	1000 ml	Sample proportional counted for Alpha-radiation for immediate result. Sample may be later autoradiographed.
Strontium-90	SR90	Strontium-90	30 dpm	± 50%	200 ml	
Thorium	TH	Thorium	0.99μg	± 50%	1000 ml	
Gross Beta-High Level	GBH	All beta emitters except halogens	750 dpm	± 75%	50 ml	K^{40} corrected
Gross Alpha (1a)	GA1A	Uranium and Plutonium	1.5 dpm	± 50%	100 ml	Sample electrodeposited on SS disc and autoradiographed.
Gross Alpha (1b)	GA1B	Uranium and Plutonium	9 dpm	± 50%	100 ml	Sample planchetted and proportional counted for alpha.

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Analysis Type	Listing Code	Analysis Specific For	Sensitivity/ 1500 ml	Accuracy at Minimum Sensitivity	Minimum Volume Required	Remarks
Gross Alpha (2)	GA2	All other alpha emitters including Th, Pa, U, Np, Pu, Am, Cm, Po, and Ra	15 dpm	± 50%	100 ml	Sample planchetted and proportional counted for alpha
Iodine-131	I 131	Iodine-131	300 dpm	± 50%	250 ml	Decay corrected to sampling date.