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Rockwell International
Energy Systems Group

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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 1976.

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) exposure or effluent releases could be reduced under the ALARA concept, and (3) equipment for effluent and exposure control is performing properly.

A small increase in personnel radiation exposures over 1975 occurred, which is mainly attributable to increased D&D activities. However, all exposures remained well below established limits.

This report satisfies License Condition Number 23 of Special Nuclear Materials License No. SNM-21.

RESERVED FOR PROPRIETARY/LEGAL NOTICES

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INTRODUCTION

As set forth in the Energy Systems Group's special material license[1] as Condition 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) cannot be separated for each type of activity. When this occurs, the values are reported unmodified as measured and are conservatively attributed wholly to licensed activities.

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

1) Fuel Fabrications - Building 001 and supporting operations in Buildings 001 and 004, De Soto Facility, Canoga Park, California

*The initial report in this series was issued for the Calendar Year 1975[3]

2) Atomics International Hot Laboratory - Building 020, Santa Susana Field Laboratories, Santa Susana, California

3) Nuclear Material Development Facility - Building 055, Santa Susana Field Laboratories, Santa Susana, California

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 (e.g., urinalysis). 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 on a log-probability scale.* The average annual dose was approximately 200 mRem.

B. IN-VIVO LUNG SCANS

Measurements are made periodically of the total radioactive lung burden for specific radioisotopes of those employees who have been or potentially were 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, 54 lung scans were made for uranium

*Data which have a normal probability distribution appear as a straight line when plotted on this type of scale.

**Helgeson Nuclear Services, Inc., Pleasanton, California

TABLE 1
WHOLE BODY DOSES - 1976
Sheet 1 of 3

Dose (mRem)	P (people)	CP (cum. peo.)*	(cum. %)	Pop. Dose (man-Rem)
5	11	11	4.58	0.055
10	30	41	17.08	0.300
15	22	63	26.25	0.330
20	14	77	31.17	0.280
25	15	92	38.33	0.375
30	13	105	43.75	0.390
35	8	113	47.08	0.280
40	12	125	52.08	0.480
45	8	133	55.42	0.360
50	1	134	55.83	0.050
55	8	142	59.16	0.400
65	1	143	59.58	0.065
70	4	147	61.25	0.280
75	3	150	62.50	0.225
80	4	154	64.17	0.320
85	1	155	64.58	0.085
90	1	156	65.00	0.090
95	2	158	65.83	0.190
100	4	162	67.50	0.400
105	1	163	67.92	0.105
110	3	166	69.17	0.330
115	3	169	70.42	0.345
125	4	173	72.08	0.500
130	2	175	72.92	0.260
140	1	176	73.33	0.140
145	3	179	74.58	0.435
160	1	180	75.00	0.160
165	1	181	75.42	0.165

*Cumulative people, with annual dose ≤ Indicated value

TABLE 1
WHOLE BODY DOSES - 1976
Sheet 2 of 3

Dose (mRem)	P (people)	CP (cum. peo.)	(cum. %)	Pop. Dose (man-Rem)
175	1	182	75.83	0.175
185	1	183	76.25	0.185
190	1	184	76.67	0.190
195	1	185	77.08	0.195
200	1	186	77.50	0.200
205	2	188	78.33	0.410
220	1	189	78.75	0.220
230	1	190	79.17	0.230
250	1	191	79.58	0.250
255	1	192	80.00	0.255
270	1	193	80.42	0.270
275	2	195	81.25	0.550
280	2	197	82.08	0.560
290	1	198	82.50	0.290
300	2	200	83.33	0.600
310	1	201	83.75	0.310
315	1	202	84.17	0.315
320	1	203	84.58	0.320
325	1	204	85.00	0.325
350	1	205	85.42	0.350
390	2	207	86.25	0.780
415	1	208	86.67	0.415
470	1	209	87.08	0.470
485	1	210	87.50	0.485
510	1	211	87.92	0.510
515	1	212	88.33	0.515
530	1	213	88.75	0.530

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TABLE 1
WHOLE BODY DOSES - 1976
Sheet 3 of 3

Dose (mRem)	P (people)	CP (cum. peo.)	(cum. %)	Pop. Dose (man-Rem)
550	1	214	89.17	0.550
610	1	215	89.58	0.610
615	1	216	90.00	0.615
620	1	217	90.42	0.620
630	1	218	90.83	0.630
635	1	219	91.25	0.635
665	1	220	91.67	0.665
670	1	221	92.08	0.670
680	1	222	92.50	0.680
775	1	223	92.92	0.775
835	1	224	93.33	0.835
895	1	225	93.75	0.895
920	1	226	94.17	0.920
980	1	227	94.58	0.980
1015	1	228	95.00	1.015
1115	1	229	95.42	1.115
1200	1	230	95.83	1.200
1390	1	231	96.25	1.390
1430	1	232	96.67	1.430
1470	1	233	97.08	1.470
1600	1	234	97.50	1.600
1655	1	235	97.92	1.655
1675	1	236	98.33	1.675
1690	1	237	98.75	1.690
1830	1	238	99.17	1.830
2030	1	239	99.58	2.030
2055	1	240	100	2.055
		TOTAL Man-Rem		47.575

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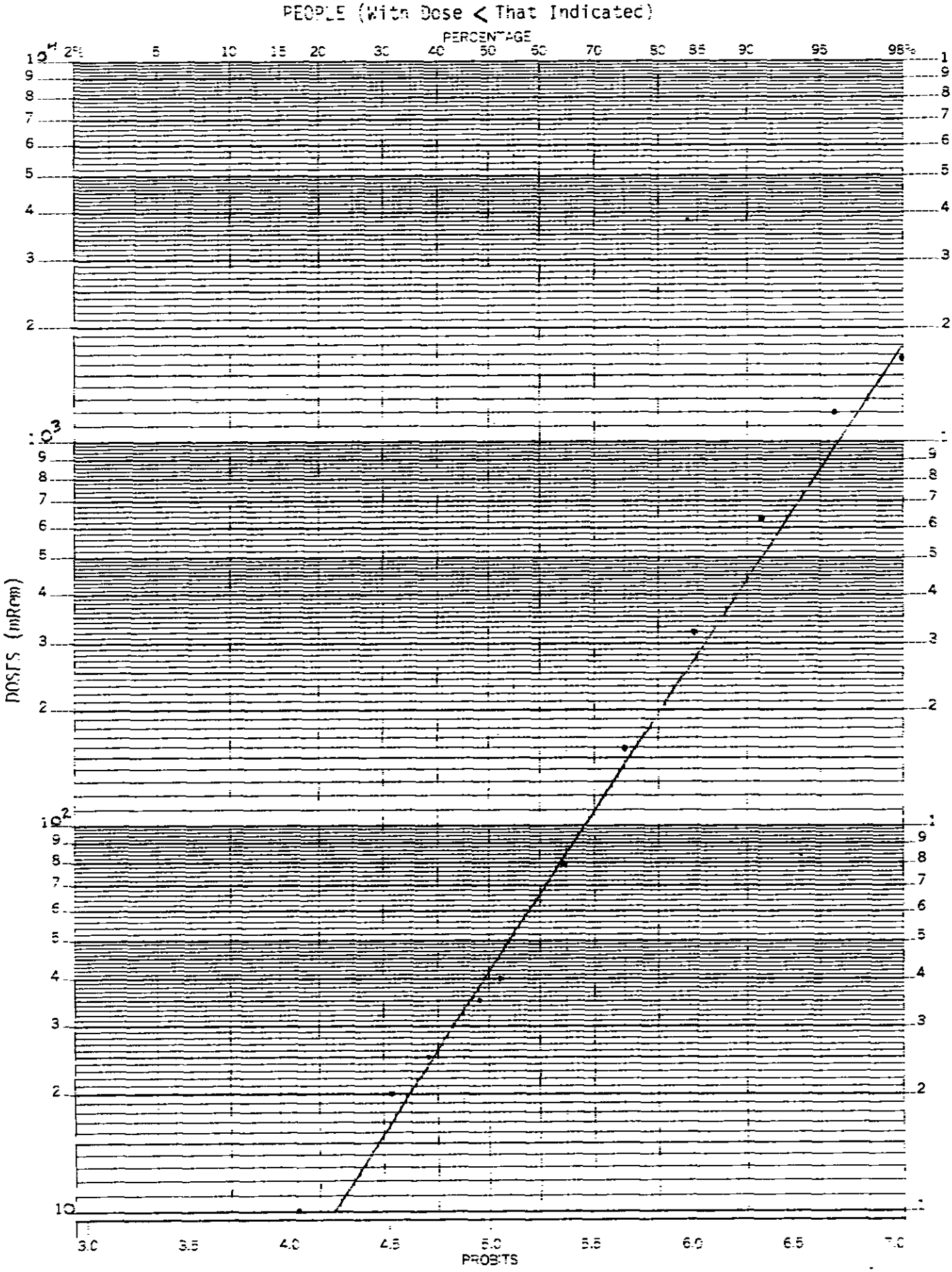

Figure 1 Personnel Annual Exposures (Whole Body) - 1976

deposition. Nine of the scans (on seven different individuals) showed positive results (Table 2A).

C. BIOASSAYS

Bioassays normally consist of analysis of urine and occasionally of fecal samples. Personnel whose work assignments potentially expose them to respirable-sized radioactive aerosols are routinely checked in this manner. Normally, urinalyses are performed quarterly and fecal analyses only when gross internal contamination is suspected. The techniques employed are described in the appendix. A statistical summary of the positive results for 1976 appears in Table 2A, while a detailed listing of the positive results is shown in Table 2B.

TABLE 2A

BIOASSAYS - STATISTICAL SUMMARY - 1976

Measurement**	Type**	Total Tests	Total Positive Results	Total Individuals With Positive Results*
U	UF	419	29	24
U	UR	415	7	6
GA	1A	0	0	0
GA	1B	167	0	0
GA	2B	2	0	0
GB	H	38	1	1
GB	G	1	0	0
Pu	A	38	1	1
FP	1	5	2	2
FP	3A	169	10	6
FP	3B	171	6	3
SR-90		2	2	2
U-235	IVLC	54	9	7

*Tests were repeated on some individuals who showed positive results

** U = uranium
UF = uranium - fluorometric
UR = uranium - radiometric
G = gross alpha
GB = gross beta
Pu = gross plutonium
FP = fission products
IVLC = uranium, in-vivo lung count

Note

See Appendix for description of various bioassay types

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TABLE 2B

POSITIVE BIOASSAY RESULT SUMMARY - 1976
(Sheet 1 of 4)

H&S Number	Sample Date	Analysis Type +	Results Per Volume Analyzed	Results Per 1500 mR-day	Specific Radionuclide	Equivalent MPBB %
1292	02/12/76	IVLC	38 ug		U^{235}	10.6% MPLB
1292	03/29/76	U-UF	0.0005 ug	0.75 ug	U	0.75
1292	06/21/76	U-UF	0.0001 ug		U	14.8% MPBB
1292	06/29/76	IVLC	53 ug		U^{235}	
3999	08/30/76	U-UF	0.0003 ug	0.45 ug	U	0.45
3999	02/04/77	IVLC	0 ug		U^{235}	
3897	10/25/76	U-FP3A	13.9 dpm	104 dpm	MFP (Sr^{90})	21.7
3897	11/08/76	U-FP3A	3.06 dpm		MFP (Sr^{90})	
3897	11/22/76	U-FP3B	15.4 dpm	132 dpm	MFP (Cs^{137})	0.02
3897	12/16/76	U-FP3B	3.63 dpm		MFP (Cs^{137})	
4015	02/10/76	IVLC	52 ug		U^{235}	14.5% MPLB
4015	06/30/76	IVLC	0 ug		U^{235}	
1215	05/17/76	U-UF	0.0004 ug	0.75 ug	U	0.75
1215	08/23/76	U-UF	0 ug		U	
4148	07/19/76	U-UF	0.0005 ug	0.75 ug	U	0.75
4148	11/22/76	U-UF	0.0005 ug	0.75 ug	U	0.75
4148	01/13/77	U-UF	0 ug		U	
3969	02/09/76	IVLC	96 ug		U^{235}	26.8% MPLB
3969	06/29/76	IVLC	26 ug		U^{235}	7.2% MPLB
3969	02/03/77	IVLC	0 ug		U^{235}	
1584	08/25/76	U-UR	2.24 dpm	33.6 dpm	U^{235}	14.6
1584	11/22/76	U-UR	0 dpm		U^{235}	
1584	08/25/76	U-UF	0.0006 ug	0.90 ug	U	0.9
1584	11/22/76	U-UF	0.0002 ug		U	

TABLE 2B

POSITIVE BIOASSAY RESULT SUMMARY - 1976
(Sheet 2 of 4)

H&S Number	Sample Date	Analysis Type +	Results Per Volume Analyzed	Results Per 1500 mR-day	Specific Radionuclide	Equivalent MPBB %
4025	03/15/76	U-UF	0.0004 ug	0.60 ug	U	0.6
4025	06/02/76	U-UF	0.0002		U	
2040	06/29/76	IVLC	16 ug		U^{235}	6.4% MPLB
2040	05/03/78	IVLC	0 ug		U^{235}	
3757	05/17/76	U-UF	0.0006 ug	0.90 ug	U	0.9
0863	03/08/76	U-UF	0.0004 ug	0.60 ug	U	0.6
0863	04/30/76	U-UF	0.0001 ug		U	
0863	06/21/76	U-UF	0.0004 ug	0.60 ug	U	0.6
0863	09/20/76	U-UF	0 ug		U	
1863	05/04/76	U-UF	0.0005 ug	0.75 ug	U	0.75
1863	08/16/76	U-UF	0 ug		U	
1863	05/04/76	U-UR	1.25 dpm	18.8 dpm	U^{235}	8.2
1863	09/28/76	U-UR	0.4 dpm		U^{235}	
1863	08/16/76	U-UR	1.0 dpm	15.0 dpm	U^{235}	6.5
1863	09/28/76	U-UR	0.4 dpm		U^{235}	
0178	01/26/76	U-UF	0.0003 ug	0.45 ug	U	0.45
0178	08/23/76	U-UF	0.0001 ug		U	
4154	09/20/76	U-UR	1.40 dpm	30 dpm	U^{235}	13.0
4154	10/26/76	U-UR	0 dpm		U^{235}	
4154	09/20/76	U-UF	0.0003 ug	0.45 ug	U	0.45
4154	10/26/76	U-UF	0 ug		U	

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TABLE 2B

POSITIVE BIOASSAY RESULT SUMMARY - 1976
(Sheet 3 of 4)

H&S Number	Sample Date	Analysis Type	Results Per Volume Analyzed	Results Per 1500 mR-day	Specific Radionuclide	Equivalent MPBB %
1731	01/19/76	U-UF	0.0013 ug	1.95 ug	U	1.95
1731	08/23/76	U-UF	0.0001 ug		U	
3698	08/23/76	U-UF	0.0005 ug	0.75 ug	U	0.75
3698	09/27/76	U-UF	0.0002 ug		U	
3986	02/10/76	IVLC	52 ug		U^{235}	14.5% MPLB
3986	09/23/76	IVLC	0 ug		U^{235}	
3888	04/23/76	U-UR	0.0007 ug	1.05 ug	U	1.05
4059	12/10/76	U-UR	1.11 dpm	16.7 dpm	U^{235}	7.3
4059	01/03/77	U-UR	0.24 dpm		U^{235}	
4059	12/10/76	U-UF	0.0013 ug	1.95 ug	U	1.95
4059	01/03/77	U-UF	0.0001 ug		U	
3981	05/18/76	U-UF	0.0003 ug	0.45 ug	U	0.45
3981	08/17/76	U-UF	0.0009 ug	1.35 ug	U	1.35
3981	09/20/76	U-UF	0 ug		U	
4016	02/07/76	IVLC	52 ug		U^{235}	14.5% MPLB
0606	05/24/76	U-UR	1.04 dpm	15.7 dpm	U^{235}	6.8
0606	08/16/76	U-UR	0.11 dpm		U^{235}	
3955	05/18/76	U-UR	0.53 dpm	7.95 dpm	U^{235}	3.4
3955	08/20/76	U-UR	0.22 dpm		U^{235}	
4007	05/02/76	U-UF	0.0005 ug	0.75 ug	U	0.75
4007	08/23/76	U-UF	0.0002 ug		U	

TABLE 2B

POSITIVE BIOASSAY RESULT SUMMARY - 1976
(Sheet 4 of 4)

H&S Number	Sample Date	Analysis Type +	Results Per Volume Analyzed	Results Per 1500 mR-day	Specific Radionuclide	Equivalent MPBB %
4028	01/25/76	U-Pu	0.120 dpm	0.180 dpm	Pu^{239}	0.15
4028	02/29/76	U-Pu	0.126 dpm		Pu^{239}	
4028	11/07/76	U-Pu	0 dpm		Pu^{239}	
3946	05/03/76	U-UF	0.0004 ug	0.60 ug	U	0.60
3946	05/28/76	U-UF	0 ug		U	
4020	05/17/76	U-UF	0.0005 ug	0.75 ug	U	0.75
4020	06/18/76	U-UF	0.0001 ug		U	
3656	01/19/76	U-UF	0.001 ug	1.5 ug	U	1.5
3656	09/14/76	U-UF	0.0007 ug	1.05 ug	U	1.05
3656	10/14/76	U-UF	0.0004 ug	0.60 ug	U	0.60
3656	10/14/76	U-UR	0.37 dpm		U^{235}	
2757	05/03/76	U-UF	0.0004 ug	0.60 ug	U	0.60
2757	11/01/76	U-UF	0.0001 ug		U	
4134	06/28/76	U-UF	0.0007 ug	1/05 ug	U	1.05
4134	08/20/76	U-UF	0 ug		U	
3949	02/09/76	IVLC	38 ug		U^{235}	10.6% MPLB
3949	02/03/77	IVLC	0 ug		U^{235}	

* - 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

II. ON-SITE RADIATION/RADIOACTIVITY LEVELS

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

A. AREA

Radiation 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. Table 3 presents a summary of the fixed sampler results.

TABLE 3

INTERIOR (WORKING AREA) AIR SAMPLING SUMMARY - 1976

Building/Area	Alpha (μCi/cc) Max.	Alpha (μCi/cc) Av.	Beta, gamma (μCi/cc) Max.	Beta, gamma (μCi/cc) Av.
01-fuel fab.	5×10^{-11}	9×10^{-12}	**	**
04-chem. labs	NS	NS	NS	NS
020-AIHL	*	*	6×10^{-12}	3×10^{-12}
055-NMDF	8.9×10^{-13}	$\sim 3 \times 10^{-13}$	**	**

*Because of the nature of materials in use, alpha activity is not normally assessed in Building 020

**Because of the nature of materials in use, beta activity is not normally assessed in Buildings 01 and 055

Notes: Based on materials in use, the most restrictive MPC's in each facility would be:

Bldg. 001: $1 \times 10^{-10} \frac{\mu Ci}{cc}$ (U-235) α

Bldg. 020: $1 \times 10^{-9} \frac{\mu Ci}{cc}$ (Sr-90) β,γ

Bldg. 050: $2 \times 10^{-12} \frac{\mu Ci}{cc}$ (Pu-239) α

NS = Not Sampled

In Building 020, the measurements shown were made in a restricted access area.

In the ATR fuel fabrication area of Building 001, about 50 individuals wore personal (lapel) air samplers during the year. The average reading of these samplers was $\sim 5.0 \times 10^{-10}$ μCi/hr-cc for a forty-hr work week. The maximum weekly reading was 3.4×10^{-9} μCi/hr-cc. The permissible limit is 4×10^{-9} μCi/hr-cc for the material in use. Fixed location air monitors in the area (13 locations) averaged 3.5×10^{-10} μCi/hr-cc, with the highest single sample being 2.1×10^{-9} μCi/hr-cc.

1. Special Air Sampling - Building 055

In Building 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 4 in descending order of magnitude and the date (week)of each measurement is noted. In this manner, any unusual release is more apparent. However, an inspection of the data in Table 4 leads to the conclusion that the values are distributed in an essentially random manner, with no one week standing out. Thus, it is concluded that no unusual releases occurred during the year.

TABLE 4

AIR SAMPLES, NMDF - BUILDING 055, 1976

Sampling Location	Max. Cum. Week Exp.* μCi-hr/cc	Week Ending
GBR-3A	1.8-12	10/8
GBR-15A	1.8-12	9/3
GBR-18S	1.5-12	7/30
GBR-24NE	1.5-12	7/12
GBR-1A	1.2-12	10/12, 11/19
GBR-3S	1.2-12	9/17, 10/18
GBR-5S	1.2-12	5/28, 9/17
GBR-6N	1.2-12	5/14, 7/23, 7/30, 8/27
GBR-15S	1.2-12	12/10
GBR-21S	1.2-12	5/7, 7/30
GBR-24SW	1.2-12	1/9
GBR-26S	1.2-12	4/23, 10/18
GBR-27N	1.2-12	8/13
Filter Room	1.2-12	8/13
Eberline "A"	1.2-12	12/23
Chem. Lab.	8.9-13	6/11, 8/13, 10/22 11/22, 12/23
GBR-4N	8.9-13	7/23
GBR-8N	8.9-13	9/24
GBR-19S	8.9-13	3/26, 10/22
GBR-20S	8.9-13	2/20, 9/3, 12/17
GBR-27S	8.9-13	5/28, 7/30, 11/19, 12/10
SS Vault	5.9-13	5/21, 9/24, 10/8, 10/22
Support Area	5.9-13	11/19, 11/24
Stack	5.9-13	4/30, 10/17
Radeco "B"	3-13	1/9, 1/30, 2/20, 9/17, 11/5, 12/28

*Maximum permissible exposure = 5×10^{-11} μCi-hr/cc, alpha

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TABLE 5

RADIOACTIVITY CONTENT OF ATMOSPHERICALLY DISCHARGED EFFLUENT RELEASED TO UNRESTRICTED AREAS - 1976

Building	Point of Release	Approximate Effluent Volume (ft^3)	Activity Monitored	Approximate Minimum Detection Limit ($\mu Ci/m\ell$)	Annual* Average Concentration ($\mu Ci/m\ell$)	Sampling Period Maximum Observed Concentration ($\mu Ci/m\ell$)	Total Radioactivity Released (Ci)
001	Stack Exit	2.4×10^{10}	α	1.7×10^{-16}	$<9.4 \times 10^{-14}$	8.3×10^{-13}	$<6.4 \times 10^{-5}$
			β	5.5×10^{-16}	$<2.5 \times 10^{-14}$	2.3×10^{-13}	$<1.7 \times 10^{-5}$
004	Stack Exit	6.9×10^{10}	α	4.5×10^{-16}	$<4.1 \times 10^{-15}$	9.3×10^{-14}	$<8.1 \times 10^{-6}$
			β	1.6×10^{-15}	$<4.5 \times 10^{-15}$	2.5×10^{-14}	$<8.9 \times 10^{-6}$
020	Stack Exit	2.1×10^{10}	α	9.7×10^{-17}	$<2.6 \times 10^{-16}$	1.6×10^{-15}	$<1.5 \times 10^{-7}$
			β	3.1×10^{-16}	9.8×10^{-15}	3.2×10^{-14}	5.8×10^{-6}
022	Stack Exit	1.0×10^{10}	α	3.0×10^{-16}	$<8.1 \times 10^{-16}$	8.3×10^{-15}	$<2.3 \times 10^{-7}$
			β	9.7×10^{-16}	$<3.6 \times 10^{-15}$	1.9×10^{-14}	$<1.1 \times 10^{-6}$
055	Stack Exit	1.6×10^{10}	α	2.6×10^{-16}	$<3.4 \times 10^{-16}$	7.1×10^{-16}	$<1.5 \times 10^{-7}$
						Total	$<1.1 \times 10^{-4}$

Annual average ambient air radioactivity concentration - 1976	α	$<6.6 \times 10^{-15}$
	β	$<1.0 \times 10^{-13}$

*Effluent radioactivity is generally less than ambient air radioactivity.

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III. EFFLUENT MONITORING

A. FACILITY VENTILATION

Ventilation discharge air is continuously sampled at points downstream of the exhaust filters provided, prior to exiting the facility system. Thus, these measurements provide a record of radioactive aerosols that were actually discharged from the facility by this route during the year. A portion of the discharged air is continuously drawn via a side-stream through a high efficiency filter medium which collects any particulate aerosols entrained in the sample. Table 5 presents a summary of average and maximum radioactivity concentrations measured in these systems during 1976.

B. LIQUID WASTES

All liquid wastes from buildings which may contain significant amounts of radioactivity are drained to hold-up tanks where they are sampled and the contents assessed for radioactivity prior to disposition. If the activity is below the prescribed limits (10 CFR 20), it is then discharged to the sanitation sewers. If, however, these limits are exceeded, the waste material is pumped out and disposed through the use of a commercial radioactive material disposal service. During the period in question, the liquid wastes which were discharged to the sewer system are summarized in Table 5.

TABLE 6

LIQUID EFFLUENT RADIOACTIVITY DISCHARGED TO SANITARY SEWER - 1976

Building	Point of Release	Approximate Effluent Volume (gal)	Activity Monitored	Approximate Minimum Detection Limit (μCi/mℓ)	Annual Average Concentration (μCi/mℓ)	Sample Maximum Observed Concentration (μCi/mℓ)	Total Radioactivity Released (Ci)
001	Retention Tank	13,500	α	1.2×10^{-9}	1.4×10^{-7}	4.4×10^{-7}	7.3×10^{-6}
			β	4.1×10^{-9}	1.1×10^{-7}	2.3×10^{-7}	5.7×10^{-6}
004	Proportional Sampler	1,253,000	α	1.2×10^{-9}	$<3.2 \times 10^{-8}$	1.4×10^{-7}	$<1.5 \times 10^{-4}$
			β	4.1×10^{-9}	$<7.3 \times 10^{-8}$	3.7×10^{-7}	$<3.3 \times 10^{-4}$
020*	---	0		---	---	---	---
055*	---	0		---	---	---	---

*All liquid radioactive wastes from these facilities are solidified and land buried as dry waste at an appropriate disposal site.
The maximum permissible concentrations for discharge to the sewer are 8×10^{-4} μCi/m$\ell\alpha$ and 1×10^{-3} μCi/m$\ell\beta$.

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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 mi 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 significant 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.

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*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 1976.[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 1976 are presented in Tables 7 through 11. 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 noble 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}.

TABLE 7

SOIL RADIOACTIVITY DATA - 1976

Area	Activities	Number Sample	Gross Radioactivity (μCi/gm) Annual Average Value (95% Confidence Level)	Gross Radioactivity (μCi/gm) Maximum Observed Value*
On-Site	α	144	$(5.6 \pm 1.5)\ 10^{-7}$	8.4×10^{-7}
	β	144	$(2.5 \pm 0.1)\ 10^{-5}$	3.2×10^{-5}
Off-Site	α	48	$(5.6 \pm 1.5)\ 10^{-7}$	1.0×10^{-6}
	β	48	$(2.4 \pm 0.1)\ 10^{-5}$	3.0×10^{-5}

*Maximum value observed for single sample.

TABLE 8

VEGETATION RADIOACTIVITY DATA - 1976

Area	Activity	Number Sample	Gross Radioactivity (μCi/gm) Dry Weight Annual Average Value	Gross Radioactivity (μCi/gm) Ash Annual Average Value (95% Confidence Level)	Gross Radioactivity (μCi/gm) Ash Maximum Observed Value*
On-Site	α	144	$(<3.1 \pm 2.6)\ 10^{-8}$	$(<1.9 \pm 1.6)\ 10^{-7}$	1.2×10^{-6}
	β	144	$(2.5 \pm 0.1)\ 10^{-5}$	$(1.70 \pm 0.03)\ 10^{-4}$	2.99×10^{-4}
Off-Site	α	48	$(<4.4 \pm 3.3)\ 10^{-8}$	$(<2.2 \pm 1.7)\ 10^{-7}$	8.4×10^{-7}
	β	48	$(3.0 \pm 0.0)\ 10^{-5}$	$(1.41 \pm 0.03)\ 10^{-4}$	2.74×10^{-4}

*Maximum value observed for single sample.

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TABLE 9

NDFL PROCESS WATER RADIOACTIVITY DATA - 1976

Area	Activity	Number Samples	Gross Radioactivity (μCi/mℓ) Average Value (95% Confidence Level)	Gross Radioactivity (μCi/mℓ) Maximum* Observed Value
NDFL	α	24	$(<2.5 = 2.9)\ 10^{-10}$	4.2×10^{-10}
	β	24	$(2.0 \pm 0.7)\ 10^{-9}$	2.5×10^{-9}

*Maximum value observed for single sample.

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TABLE 10

BELL CREEK AND ROCKETDYNE SSFL RETENTION POND
RADIOACTIVITY DATA - 1976

Area	Activity	Number Samples	Gross Radioactivity: Average Value (95% Confidence Level)	Gross Radioactivity: Maximum Observed Value*	% of Guide**
Bell Creek Mud No. 54 (μCi/gm)	α	12	$(3.8 \pm 1.3)\ 10^{-7}$	9.1×10^{-7}	NA
	β	12	$(2.3 \pm 0.1)\ 10^{-5}$	3.0×10^{-5}	NA
SSFL Pond Mud No. 55 (μCi/gm)	α	12	$(5.9 \pm 1.5)\ 10^{-7}$	7.7×10^{-7}	NA
	β	12	$(2.5 \pm 0.1)\ 10^{-5}$	2.6×10^{-5}	NA
Bell Creek Vegetation No. 54 (μCi/gm ash)	α	12	$(<1.7 \pm 1.6)\ 10^{-7}$	3.2×10^{-7}	NA
	β	12	$(1.64 \pm 0.03)\ 10^{-4}$	2.51×10^{-4}	NA
Bell Creek Vegetation No. 54 (μCi/gm dry weight)	α	12	$(<3.4 \pm 3.1)\ 10^{-8}$	7.0×10^{-8}	NA
	β	12	$(3.2 \pm 0.1)\ 10^{-5}$	4.6×10^{-5}	NA
Bell Creek Water No. 16 (μCi/mℓ)	α	12	$(<2.5 \pm 2.9)\ 10^{-10}$	2.8×10^{-10}	<0.006
	β	12	$(2.2 \pm 0.8)\ 10^{-9}$	2.9×10^{-9}	0.7
SSFL Pond Water No. 6 (μCi/mℓ)	α	12	$(<2.4 \pm 2.9)\ 10^{-10}$	$<2.4 \times 10^{-10}$	0.006
	β	12	$(4.3 \pm 0.8)\ 10^{-9}$	5.5×10^{-9}	1.4
SSFL Pond Water No. 12 (μCi/mℓ)	α	12	$(<2.8 \pm 3.0)\ 10^{-10}$	5.3×10^{-9}	<0.007
	β	12	$(4.4 \pm 0.8)\ 10^{-9}$	7.0×10^{-9}	1.5

*Maximum value observed for single sample.

**Guide: 4×10^{-6} μCi/m$\ell\alpha$, 3×10^{-7} μCi/m$\ell\beta$; 10 CFR 20 Appendix B

NA = Not applicable, no Guide value having been established.

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TABLE 11

AMBIENT AIR RADIOACTIVITY DATA - 1976

Area	Activity	Number Samples	Average Value (95% Confidence Level)	Maximum Observed Value*	% of Guide**
Headquarters	α^{Ψ}	719	$(<6.7 \pm 8.4)\ 10^{-15}$	1.4×10^{-13}	0.22
(μCi/mℓ)	$\beta\gamma$ ***	719	$(<9.6 \pm 1.8)\ 10^{-14}$	3.7×10^{-12}	0.032
NDFL	α	2520	$(<6.5 \pm 7.2)\ 10^{-15}$	5.3×10^{-14}	11.0
(μCi/mℓ)	$\beta\gamma$ ***	2520	$(<1.1 \pm 0.2)\ 10^{-13}$	3.4×10^{-12}	3.7

*Maximum value observed for single sample.

**Guide: Headquarters - 3×10^{-12} μCi/mℓ (α); 3×10^{-10} μCi/mℓ; (β,γ)[4]
NDFL - 6×10^{-14} μCi/mℓ (α); 3×10^{-11} μCi/mℓ (β,γ)

Ψ Miniminum Detectable level (MDL); 5.6×10^{-15} μCi/m^3 - Individual samples with activity levels of 0 to 5.6×10^{-15} μCi/mℓ are recorded and averaged as 5.6×10^{-15} μCi/mℓ.

*** Minimum Detectable Level (MDL) 1.2×10^{-14} μCi/mℓ - Individual daily samples with activity levels of 0 to 1.2×10^{-14} μCi/mℓ. Indicated average values are upper limits, since some data were below the minimum detection limits.

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 9.

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 mi 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 10.

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.

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/m$\ell\alpha$ and 3×10^{-7} μCi/m$\ell\beta$, for Pu^{239} and for Sr^{90}, respectively, are appropriate.

The correspondingly most restrictive Guide value for Headquarters wastewater radioactivity discharged to the sanitary sewers, a controlled area, is 8×10^{-4} μCi/m$\ell\alpha$ and 1×10^{-3} μCi/m$\ell\beta$, for U-235 and Co^{60}, respectively. For NDFL ambient air radioactivity is due to work with unencapsulated plutonium at this site. The Guide value of 3×10^{-11} μCi/m$\ell\beta$, 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/m$\ell\alpha$ 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/m$\ell\beta$ 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 for these media.

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 periodically 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 11.

Radioactivity levels observed in environmental samples for 1976, reported in Tables 7 through 11, compare closely with levels reported for recent years. Local environmental radioactivity levels, which result primarily from beta-emitting radionuclides, and have shown the effect of fallout during past extensive 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 occasionally continue to be observable in daily ambient air radioactivity levels; however, the long-term effects on surface air 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 NLDF 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 mi from the sites and similarly evaluated to determine the local area ambient radiation level, which averaged 0.0095 mRem/h for 1976. The average radiation dose rate and equivalent annual dose monitored at each dosimeter location are presented in Table 12.

The table shows that radiation dose rates and equivalent annual doses monitored on-site (average = 95.5 mRem) are nearly identical to levels monitored at three widely separated off-site locations (average = 93.6 mRem). 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

TABLE 12

SITE AMBIENT RADIATION DOSIMETRY DATA - 1976

Dosimeter Location	Average Dose Rate (mRem/h)	Equivalent Annual Dose (mRem)
TLD-1 Headquarters	0.010	88
TLD-2 Headquarters	0.008	70
TLD-3 Headquarters	0.011	96
TLD-4 Headquarters	0.011	96
TLD-5 NDFL	0.012	105
TLD-6 NDFL	0.013	114
TLD-7 NDFL	0.014	123
TLD-8 NDFL	0.012	105
TLD-9 NDFL	0.008	70
TLD-10 NDFL	0.010	88
TLD-11 Off-Site	0.010	88
TLD-12 Off-Site	0.012	105
TLD-13 Off-Site	0.010	88

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100 mRem/yr or ~0.011 mRem/h. 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, it is considered that no measurable radiation exposure to the general population or to individuals in uncontrolled areas resulted from AI operations.

V. UNUSUAL EVENTS

A. BUILDING 001 - ATR X-RAY BOOTH

On May 26, 1976, some men were discovered doing work atop the roof of the ATR radiographic room while X-ray operations were in progress. In reconstructing the incident, it was learned that the dose rate in the area where the men were working was generally less than 2 mR/h with the highest spot being about 4 mR/h. Since the machine had only been on a total of 6 minutes, the maximum possible dose was 0.4 mR, which is negligible.

Steps to establish closer liaison between personnel working on/around radiation producing devices and personnel for whom the work is being performed will be established to avoid such future incidents.(5)

B. BUILDING 001 - ATR POWDER ROOM

On June 17, 1976, a can of ~3 kg of recycled UAl_x briquets was dropped and the briquets momentarily ignited, and then extinguished themselves. However, some amount of the uranium oxide was dispersed as an aerosol. Nasal swipes were obtained from all who had been in the room. Only one measurable swipe was obtained (~20 dpm/100 cm^2). Some protective clothing contamination occurred. Involved personnel submitted urinalysis samples. Additional instruction was given on this operation which should help to prevent future such incidents.(6)

C. BUILDING 001

On August 26, 1976, a sealed transfer can containing uranium fines was inadvertently dropped to the floor while being transferred from the SNM Vault Weigh Room to the Head-In Process Room. The drop resulted in the autoignition of the contained fines. A very limited amount of uranium dispersal occurred before an inverted 5-gal can was placed over the transfer

can. A floor smear survey revealed 200 dpm/100 cm^2 present in an area immediately adjacent to the incident. Nasal swipes were taken which showed a maximum of 11 dpm α. A small finger cut also occurred during the incident. A wound smear revealed ~40 dpm, alpha and was successfully decontaminated to 0. The five personnel involved were requested to submit bioassay specimens.(7)

D. BUILDING 001 - ATR FLUOROSCOPE ROOM

On November 3, 1976, an inadvertent exposure occurred to the operator of a fluoroscope machine while performing routine operations on an ATR fuel plate. Apparently, the machine had been still operating during the time a plate was being removed, with the window-door open. With the machine shut down, a mock repeat of the incident was carried out. The time was noted as 12 seconds. Radiation readings were then taken with the machine on at the same settings with the window-door open. By duplicating the operator's motions as close as possible, the following doses were estimated:(8)

Conservative estimated head dose -	2.4 R
Conservative estimated eye dose -	1 R
Probable head dose	0.8 R
Probable eye dose	0.3 R

The automatic interlock switch that failed to function, causing the beam to stay on, was examined and found to have a faulty spring in the open position inside the switch. This was repaired, and two new micro shutoff switches and spare parts ordered. As an additional safety device, a radiation rate meter with loudspeaker volume control was installed with the probe positioned at the operator's head level. This will give the operator a sound alarm in the event he does not see the light.

E. BUILDING 004

(None)

F. BUILDING 020

(None)

G. BUILDING 055

(None)

VI. SUMMARY/TRENDS - EXPOSURES, EFFLUENTS

All personnel and environmental exposures, and effluent releases to uncontrolled areas for 1976, were below the allowable limits. The 1975 report[3] established baseline data for identification of trends in succeeding years. Comparing the 1976 measurements to those of 1975, the following is noted:

1) Total Man-Rem (whole body): 26.700 (1975); 47.575 (1976)
   Comment: An inspection of the source for the majority of this increase leads to the conclusion that it arose largely with an increased level of D&D activities.

2) Bioassay

	1483 (1975)	1442 (1976)
Total Assessments (all types)		
Total Positive Results	57	64
Total Individuals with Positive Results	47	58

   Comment: There would not appear to be a significant increase in the number of positive tests, although there were a few more such tests and individuals with positive results.

3) Building Discharge Air
   Radioactivity in building discharge air remained well below established limits. The largest change was from Building 020, where the total β, α activity discharged decreased from 6.7×10^{-3} Ci in 1975 to 5.8×10^{-6} Ci in 1976.

4) Environmental Monitoring
   A general comparison of the site environmental monitoring results for the current and previous years (soil, vegetation, process water, Bell Creek mud and water, and SSFL pond water)

reveals no significant differences. Also, ambient air radioactivity did not vary significantly between the two years. The general background radiation (13 stations, see Table 12) was 0.012 mRem/h (1976) and 0.011 mRem/h (1975).

The overall conclusion is that no increase in radiological impact of the environs occurred in 1976. Personnel exposures did increase somewhat, primarily due to D&D activities, although they still remained well within the prescribed limits.

VII. ANTICIPATED ACTIVITIES DURING NEXT REPORTING PERIOD (1977)

A. BUILDING 001 (FUEL FABRICATION FACILITY)

Continued fabrication of ATR fuel elements.

B. BUILDING 004 (HOT ANALYTICAL CHEMISTRY)

Continued routine analytical chemical processes and analytical chemistry of 93% U-235 enriched ATR fuel elements.

C. BUILDING 020 (HOT LABORATORY)

Continued preparation for decladding of irradiated fuel elements from HNPF. Continued decladding of SRE Core II fuel.

D. BUILDING 055 (NMDF)

It is currently planned to remove all plutonium from the facility early in the year and then to proceed with decontamination of the glove boxes.

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 - 1976," J. D. Moore, AI-77-14, Atomics International

3. "Annual Review of Radiological Controls - 1975," R. S. Hart, N001TI990002, Atomics International (July 2, 1979)

4. Code of Federal Regulations, 10 Energy, Part 20, as revised, January 1979

5. IL, "ATR X-Ray Booth Incident," R. J. Tuttle from R. R. Garcia, Rockwell International, July 3, 1976

6. IL, "ATR Powder Room Incident," R. J. Tuttle from R. R. Garcia, Rockwell International, June 28, 1976

7. IL, "Incident Report - SNM Transfer Can Drop C/001," R. J. Tuttle from F. E. Begley, Rockwell International, September 2, 1976

8. IL, "ATR Fluoroscope Incident," R. J. Tuttle from J. H. Wallace, Rockwell International, November 16, 1976

APPENDIX
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 evaluating the differences in the film densities behind these various shields.

Two separate calibration energies are used to determine X-ray 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 in Table A-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 Table A-2. 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

TABLE A-1

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

	Ratios			
keV	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.

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TABLE A-2
FILM CORRECTION FACTORS

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

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 nuclear track plate film is used for fast neutron monitoring. The film is calibrated with a polonium-beryllium neutron source. High energy neutron exposures are interpreted by counting the number of proton recoil tracks in 25 fields of view 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 personnel film badges are processed routinely by the AI film badge vendor (Radiation Detection Company) according the 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 as 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 the high level gamma and neutron exposures generally associated with this type of incident. Excessive film blackening (from gamma radiation) may prevent the microscopic enumeration of proton recoil tracks. Therefore, neutron exposures above 10 rad are determined by means of activation analysis 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 ~5 hr following an incident.

Maximum gamma sensitivity of the film is about 900 R. Since the gamma dose in a criticality incident could be much greater, a LiF TLD (Thermoluminescent Dosimeter) in capsule also is 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.

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. The natural radioactive

isotope of 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.

A summary of the bioassay services provided by the United States Testing Company, Inc. is shown in Table A-3.

TABLE A-3
(Sheet 1 of 3)

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.

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.