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Rockwell International
Energy Systems Group

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ABSTRACT

Data on exposures of Atomics International personnel at NRC-licensed facilities to radiation and/or radioactivity are presented for CY 1977. This summary, in conjunction with previous and subsequent annual reports, will be used to determine if (1) there have been any upward trends in either personnel exposures and/or effluent radioactivity, (2) the exposures and/or effluents can be further reduced under the ALARA concept, and (3) the equipment for effluent and personnel exposure control is performing properly. As for 1977, it is concluded that although there was a 40% increase in total man-rem exposure, primarily due to continuing D&D activities, all personnel exposures and effluent releases remained well below the prescribed limits. Moreover, in the long run, an overall net reduction in exposures are expected to result from D&D activities.

This report satisfies License Condition 23 of Special Nuclear Materials license SNM-21 for 1977.

RESERVED FOR PROPRIETARY/LEGAL NOTICES

CONTENTS

TABLES

FIGURES

INTRODUCTION

As set forth in the Atomics International special material license[1] as Condition 23: "A formal annual report shall be made to the Radioisotope Committee of the Nuclear Safeguards Review Panel (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 operation or effluent releases, (2) if exposures and effluent releases 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."*

These reports provide a 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 - Building 001 and supporting operations in Buildings 001 and 004, De Soto Facility, Canoga Park, California
2) Rockwell International Hot Laboratory - Building 020, Santa Susana Field Laboratories
3) Nuclear Material Development Facility - Building 055, Santa Susana Field Laboratories.

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*The most recent previous report is for the year 1976.[2]

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., bio-assays). 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 in Figure 1 using a log-probability scale. Data which have an essentially normal distribution will appear as a straight line when plotted on such a scale. Inspection of Figure 1 leads to the conclusion that most of the points can be grouped rather well on a straight line, with some deviation at either extreme (more than expected very low exposures, less than expected higher exposures). It also can be noted that all exposures are less than the 5 Rem annual occupational exposure limit, with the average being about 310 mRem or ~6% of this limit. An investigation into the source of the higher exposures (>500 mRem) shows that they are almost exclusively associated with D&D activities. However, even this highest group of annual exposures only amounts to 10-50% of the permissible amount. Compared to the previous year's data (1976), the total man-rem increased from ~48 to ~67, or about 40%.

TABLE 1

WHOLE BODY DOSES - 1977

Dose (mRem)	P (People)	CP (Cum. People)	CP%	Pop. Dose (man-Rem)
5	1	1	0.5	0.005
10	7	8	3.7	0.070
15	27	35	16	0.405
20	14	49	23	0.280
25	7	56	26	0.175
30	13	69	32	0.390
35	8	77	36	0.280
40	13	90	42	0.520
45	5	95	44	0.225
50	2	97	45	0.100
55	9	106	49	0.495
60	2	108	50	0.120
65	8	116	54	0.520
70	2	118	55	0.140
75	2	120	56	0.150
80	3	123	57	0.240
85	2	125	58	0.170
90	3	128	60	0.170
95	1	129	60	0.270
100	2	131	61	0.200
105	1	132	61	0.105
110	1	133	62	0.110
120	1	134	62	0.120
125	2	136	63	0.250
130	2	138	64	0.260
135	1	139	65	0.135
150	1	140	65	0.150
160	2	142	66	0.320
175	3	145	67	0.525
180	3	148	69	0.540
195	1	149	69	0.195
205	1	150	70	0.205
210	1	151	70	0.210
220	1	152	71	0.220
230	2	154	72	0.460
235	1	155	72	0.235
245	1	156	73	0.245
255	1	157	73	0.255
265	1	158	73	0.265
280	1	159	74	0.280
290	1	160	74	0.290
295	3	163	76	0.885

TABLE 1

WHOLE BODY DOSES - 1977
(Continued)

Dose (mRem)	P (People)	CP (Cum. People)	CP%	Pop. Dose (man-Rem)
300	1	164	76	0.300
305	1	165	77	0.305
310	3	168	78	0.930
315	2	170	79	0.630
325	1	171	80	0.325
350	1	172	80	0.350
355	1	173	80	0.355
360	1	174	81	0.360
365	3	177	82	1.095
405	2	179	83	0.810
410	1	180	84	0.410
460	1	181	84	0.460
490	1	182	85	0.490
525	1	183	85	0.525
555	1	184	86	0.555
585	1	185	86	0.585
640	1	186	87	0.640
680	1	187	87	0.680
710	1	188	87	0.710
760	1	189	88	0.710
785	1	190	88	0.785
845	1	191	88	0.845
920	1	192	89	0.920
930	1	193	90	0.930
1010	1	194	90	1.010
1045	1	195	91	1.045
1055	1	196	91	1.055
1065	1	197	92	1.065
1085	1	198	92	1.085
1135	1	199	93	1.135
1330	1	200	93	1.330
1445	1	201	93	1.445
1565	1	202	94	1.565
1570	1	203	94	1.570
1610	1	204	95	1.610
2015	1	205	95	2.015
2070	1	206	96	2.070
2210	1	207	96	2.210

TABLE 1

WHOLE BODY DOSES - 1977
(Continued)

Dose (mRem)	P (People)	CP (Cum. People)	CP%	Pop. Dose (man-Rem)
2250	1	208	97	2.250
2345	1	209	97	2.345
2410	1	210	98	2.410
2585	1	211	98	2.585
2665	1	212	99	2.740
2850	1	214	99	2.850
2995	1	215	100	2.995
Total				66.990

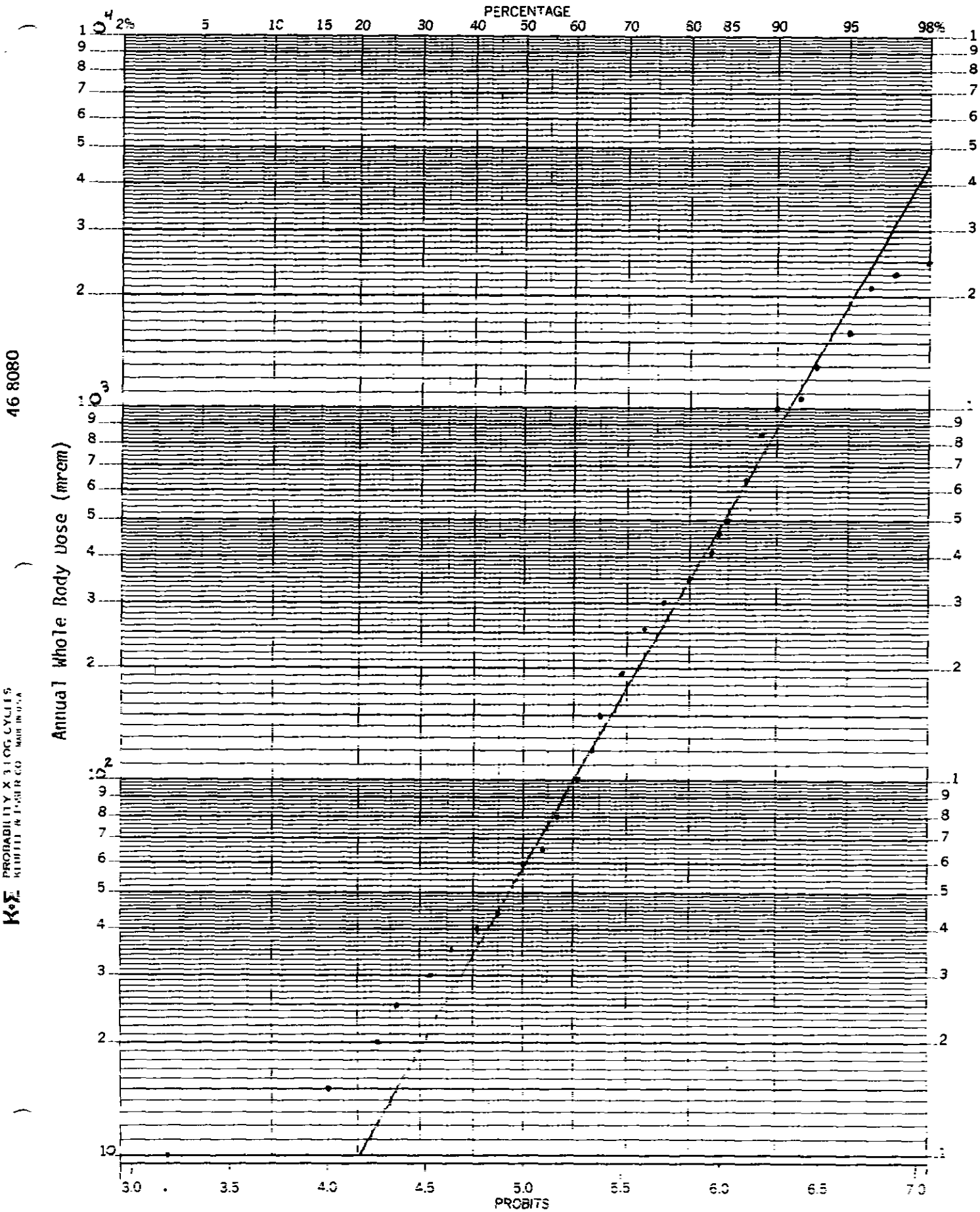

Figure 1. Distribution of Personnel External Radiation Exposure - 1977

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 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, fifty-four lung scans were made for uranium deposition. Twenty of the scans (on fifteen different individuals) showed positive results. However, followup scans showed a steady decrease of these lung burdens (see Table 2B).

C. BIOASSAYS

Bioassays normally consist of analysis of urine and occasionally, 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 analysis only when gross internal contamination is suspected. The techniques employed are described in Appendix A. A statistical summary of the results for 1977 appears in Table 2A, while a detailed listing of the positive results is shown in Table 2B. Data on the in-vivo lung scans performed in 1977 also appear in these tables.

*Helgeson Nuclear Services, Inc., Pleasanton, California

TABLE 2A

SUMMARY OF BIOASSAYS - 1977

Measurement*	Type*	Total Tests	Total Positive Results	Total Individual With Positive Results
U	UF	205	18	18
U	UR	226	2	2
GA	1A	0	0	0
GB	1B	132	0	0
GB	2B	1	0	0
GB	H	14	0	0
Pu	A	21	2	1
FP	1	0	0	0
FP	3A	298	62	18
FP	3B	298	43	9
U	IVLC	54	20	15
Sr-90	Sr-90	13	1	1
Cs-137	TBC	10	10	10

*UF = Uranium - Fluorometric
UR = Uranium - Radiometric
GA = Gross Alpha
GB = Gross Beta
Pu = Gross Plutonium
FP = Fisson Products
(For a discussion of specific analytical techniques employed, see appendix)
U-IVLC = Uranium In-Vivo Lung Count
TBC = Total Body Count

TABLE 2B

POSITIVE BIOASSAY RESULT SUMMARY - 1977

H&S Number	Sample Date	Analysis Type	Results Per Volume Analyzed	Results Per 1500 ml-day	Specific Radionuclide	Equivalent MPBB (%)
4124	05/09/77	UF	0.0004 μgm	0.60 μgm	U	0.6%
3889	02/03/77	IVLC	60 μgm		U^{235}	16.8% MPLB
3889	05/10/77	IVLC	0 μgm		U^{235}	
4015	02/03/77	IVLC	70 μgm		U^{235}	19.6% MPLB
4015	02/03/77	IVLC	35 μgm		U^{235}	8.5% MPLB
4015	02/03/77	IVLC	0 μgm		U^{235}	
4177	11/01/77	UF	0.0005 μgm	0.75 μgm	U	0.75%
4177	12/06/77	UF	0 μgm		U	
4218	06/20/77	UF	0.0011 μgm	1.65 μgm	U	1.65%
4218	09/12/77	UF	0 μgm		U	
1584	11/21/77	UF	0.0012 μgm	1.80 μgm	U	1.8%
1584	12/14/77	UF	0 μgm		U	
1584	11/21/77	UR	2.14 dpm	32.1 dpm	U^{235}	16.1%
1584	12/14/77	UR	0.09 dpm	<MDL	U^{235}	
4295	09/23/77	IVLC	141 μgm		U^{235}	34.4% MPLB
4295	01/11/78	IVLC	0 μgm		U^{235}	
2040	03/28/77	UF	0.0005 μgm	0.75 μgm	U	0.75%
2040	04/19/77	UF	0 μgm		U	
2040	09/23/77	IVLC	66 μgm		U^{235}	16.1% MPLB
3746	02/03/77	IVLC	65 μgm		U^{235}	18.2% MPLB
3746	05/10/77	IVLC	0 μgm		U^{235}	

TABLE 2B

POSITIVE BIOASSAY RESULT SUMMARY - 1977
(Continued)

H&S Number	Sample Date	Analysis Type	Results Per: Volume Analyzed	Results Per: 1500 mℓ-day	Specific Radionuclide	Equivalent MPBB (%)
1863	09/23/77	IVLC	62 μgm		U^{235}	15.1% MPLB
1863	01/11/78	IVLC	0 μgm		U^{235}	
3893	03/12/77	PuA	0.495 dpm	0.782 dpm	Pu^{239}	0.65%
3893	03/30/77	PuA	0.065 dpm	0.097 dpm	Pu^{239}	0.08%
3893	04/25/77	PuA	0.008 dpm	<MDL	Pu^{239}	
3893	07/31/77	UF	0.0004 μgm	0.60 μgm	U	0.6%
3893	11/21/77	UF	0 μgm		U	
3739	07/26/77	UF	0.0012 μgm	1.80 μgm	U	1.8%
3739	09/08/77	UF	0 μgm		U	
4154	05/10/77	IVLC	30 μgm		U^{235}	7.3% MPLB
4154	09/23/77	IVLC	31 μgm		U^{235}	7.6% MPLB
0307	05/02/77	UF	0.0005 μgm	0.75 μgm	U	0.75%
1731	02/03/77	IVLC	45 μgm		U^{235}	12.6% MPLB
4173	02/03/77	IVLC	82 μgm		U^{235}	22.9% MPLB
4173	05/10/77	IVLC	0 μgm		U^{235}	
3986	02/03/77	IVLC	67 μgm		U^{235}	18.7% MPLB
3986	05/10/77	IVLC	56 μgm		U^{235}	13.7% MPLB
3986	09/23/77	IVLC	0 μgm		U^{235}	
3986	09/12/77	UF	0.0004 μgm	0.60 μgm	U	0.6%
4206	09/23/77	IVLC	69 μgm		U^{235}	16.8% MPLB

TABLE 2B

POSITIVE BIOASSAY RESULT SUMMARY - 1977
(Continued)

H&S Number	Sample Date	Analysis Type	Results Per Volume Analyzed	Results Per 1500 ml-day	Specific Radionuclide	Equivalent MPBB (%)
4059	05/10/77	IVLC	82 μgm		U^{235}	20.0% MPLB
4059	09/23/77	IVLC	79 μgm		U^{235}	19.3% MPLB
4059	01/11/78	IVLC	0 gm		U^{235}	
4211	07/25/77	UF	0.0004 μgm	0.60 μgm	U	0.60%
4211	09/06/77	UF	0.0001 μgm	<MDL	U	
4125	09/23/77	IVLC	34 μgm		U^{235}	8.3% MPLB
4125	05/03/78	IVLC	0 μgm		U^{235}	
0606	11/21/77	UR	0.75 dpm	11.25 dpm	U^{235}	5.6%
0606	12/14/77	UR	0.21 dpm	<MDL	U^{235}	
0606	11/21/77	UF	0.0012 μgm	1.95 μgm	U	1.95%
0606	12/14/77	UF	0 μgm		U	
4007	05/02/77	UF	0.0004 μgm	0.60 μgm	U	0.60%
4007	02/23/78	UF	0 μgm		U	
4219	09/23/77	IVLC	40 μgm		U^{235}	9.8% MPLB
4219	01/11/78	IVLC	0 μgm		U^{235}	
4005	04/19/77	UF	0.0004 μgm	0.60 μgm	U	0.60%
4005	04/27/77	UF	0 μgm		U	
0533	03/14/77	UF	0.0003 μgm	0.45 μgm	U	0.45%
0533	07/25/77	UF	0 μgm		U	
1298	09/12/77	UF	0.0004 μgm	0.60 μgm	U	0.60%
1298	05/10/78	UF	0.0001 μgm	<MDL		

TABLE 2B

POSITIVE BIOASSAY RESULT SUMMARY - 1977
(Continued)

H&S Number	Sample Date	Analysis Type	Results Per Volume Analyzed	Results Per 1500 mℓ-day	Specific Radionuclide	Equivalent MPBB (%)
4208	07/27/77	UF	0.0004 μgm	0.60 μgm	U	0.60%
4208	12/06/77	UF	0.0001 μgm	<MDL		
1876	02/03/77	IVLC	65 μgm		U^{235}	18.2% MPLB
1876	09/23/77	IVLC	70 μgm		U^{235}	19.6% MPLB
1876	01/11/78	IVLC	0 μgm		U^{235}	
3949	04/01/77	UF	0.0003 μgm	0.45 μgm	U	0.45%

II. RADIATION/RADIOACTIVITY MEASUREMENTS

A. AREA RADIATION LEVELS

To roughly characterize the general external levels of penetration radiation which existed at each facility during the year, the data presented in Table 3 were compiled based on survey measurements made by the assigned HS&RS representative(s) during the year. It should be noted that while these data are approximately correct, somewhat higher levels could have possibly existed for limited periods in certain locations.

TABLE 3
RADIATION LEVELS - WORKING AREAS - 1977

Building/ Area	Average Dose Rate* (mRem/h)	Maximum Dose Rate (mRem/h)	Remarks
001-Fuel Fab	~0.03	~8.0	
004	~0.06	~2.0	
020	~0.02	~0.1	Uncontrolled areas
	~2.0	~400.0	Controlled areas
055	<1.0	~20.0	Max in glove box 3 (weight/measure box)

*Estimated

B. INTERIOR AIR SAMPLES - WORKING AREAS

In those working areas where the nature of the tasks being performed and of the materials in use potentially might lead to the generation of respirably-sized radioactive aerosols, periodic air sampling is performed. A summary of these results for 1977 is given in Table 4.

TABLE 4

INTERIOR AIR SAMPLE SUMMARY - 1977

Building/Area	Maximum μCi/cc	Average* μCi/cc
001-Fuel Fab	1.1×10^{-9} (α)	5.0×10^{-10} (α)
004	(Not Required)	—
020 Controlled Areas	2×10^{-11} (β,γ)	5×10^{-13} (β,γ)
Uncontrolled Areas	1.6×10^{-12} (β,γ)	2×10^{-13} (β,γ)
055	7.9×10^{-12} (α)	3×10^{-14} (α)

*Estimated

C. SPECIAL AIR SAMPLES - BUILDING 055

In Building 055, air samples were taken routinely at about 15 locations adjacent to the glove box train as well as several other locations. The results of these samples for 1977 are tabulated in Table 5 in a descending order of magnitude with the date (week) of each measurement noted. In this manner, any unusual activity release is more readily apparent. For example, the week ending February 18 and possible that ending December 16 seem to occur disproportionately often high on the tabulation. Because of the nature of the radioactive material in use at this facility, only an assessment of the quantity of alpha-emitting radioisotopes collected by the air samples is normally made. It may be noted from Table 5 that highest weekly concentration value occurred near glove box 24NE for the week ending February 18, when the cumulative air concentration was $7.0 \times 10^{-11} \frac{\mu Ci\text{-}h}{cc}$. This value is about 40% above the weekly integrated MPC for the material in use (Pu-239), but it is not viewed as overly significant when the annual weekly averaged exposure is considered ($\sim 7 \times 10^{-12} \frac{\mu Ci\text{-}h}{cc}$), which is about 14% of the annual average weekly concentration MPC.

TABLE 5

AIR SAMPLES
NMDF - BUILDING 055
1977

Sampling Location	Maximum Cumulative Weekly Exposure μCi-h/cc (α)	Week Ending
GB-24NE	7.0-11	2/18
GB-3AN	1.5-11	2/18
GB-15N	1.2-11	12/16
GB-19S	1.1-11	11/18
GB-26S	9.0-12	2/18
GB-15S	7.2-12	12/16
GB-24SW	5.7-12	2/18
GB-20S	5.6-12	12/16
GB-3S	4.6-12	6/27
GB-21S	3.4-12	10/21
GB-27S	3.3-12	2/18
Support Lab	2.6-12	7/8
GB-4N	1.8-12	10/14
GB-5S	1.6-12	12/16
Vault	1.1-12	7/22
GB-6N	1.0-12	11/11, 12/9
GB-"A"	9.6-13	12/16
Chem Lab	8.9-13	4/29
Radeco "B"	5.9-13	2/25

*Highest weekly measurement at each sampling location for the year. Allowable weekly exposure (average) is 5×10^{-11} μCi-h/cc, including the activity of the undetected beta-emitter, Pu-241.

III. EFFLUENT MONITORING

Effluents which may contain radioactive material are generated at certain ESG facilities as the result of operations performed under contract to DOE, under NRC Special Nuclear Materials License SNM-21, and under State of California Radioactive Material License 0015-59. The specific facilities are identified as Buildings 001 and 004 at the Headquarters site, and Buildings 020 and 055 at the Santa Susana site, SSFL.

A. TREATMENT AND HANDLING

Waste streams released to unrestricted areas are limited, in all cases, to gaseous effluents. No contaminated liquids are discharged to unrestricted areas.

The level of radioactivity contained in all atmospherically discharged effluents is reduced to the lowest practicable values by passing the effluents through certified, high-efficiency particulate air (HEPA) filters. These effluents are sampled for particulate radioactive materials by means of continuous stack exhaust samplers at the point of release. In addition, stack monitors installed at Buildings 020 and 055 provide automatic alarm capability in the event of the release of excessive gaseous activity from Building 020 or particulate activity from Building 055. The HEPA filters used for filtering gaseous effluents are 99.97% efficient for particles of 0.3-μm diameter. Particle filtration efficiency increases above and below this size.

The average concentration and total radioactivity in gaseous effluents released in 1977 to unrestricted areas are shown in Table 6. The effectiveness of the air cleaning systems is evident from the fact that, in most cases, the gaseous effluent released contains less radioactivity than the ambient air, which is indicative that there are not any measurable radioactivity releases during normal facility operations.

TABLE 6

ATMOSPHERICALLY DISCHARGED EFFLUENT RELEASED
TO UNRESTRICTED AREAS - 1977

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 Radio-Activity Released (Ci)
001	Stack Exit	2.4×10^{10}	α	1.7×10^{-16}	$<1.1 \times 10^{-14}$	8.4×10^{-14}	$<1.0 \times 10^{-5}$
			β	5.3×10^{-16}	$<4.4 \times 10^{-15}$	2.4×10^{-14}	$<4.1 \times 10^{-6}$
004	Stack Exit	5.0×10^{10}	α	4.6×10^{-16}	$<6.1 \times 10^{-16}$	2.1×10^{-15}	$<8.8 \times 10^{-7}$
			β	1.6×10^{-15}	$<5.2 \times 10^{-15}$	4.7×10^{-14}	$<7.5 \times 10^{-6}$
020	Stack Exit	2.1×10^{10}	α	9.8×10^{-17}	$<1.7 \times 10^{-16}$	4.2×10^{-16}	$<1.0 \times 10^{-7}$
			β	3.1×10^{-16}	2.3×10^{-14}	1.6×10^{-13}	1.3×10^{-5}
055	Stack Exit	1.6×10^{10}	α	2.5×10^{-16}	$<3.3 \times 10^{-16}$	6.3×10^{-16}	$<1.5 \times 10^{-7}$
						Total	$<3.9 \times 10^{-5}$

Annual average ambient air radioactivity concentration - 1977

α	$<6.6 \times 10^{-15}$	$\mu Ci/m\ell$
β	$<1.7 \times 10^{-13}$	$\mu Ci/m\ell$

*Effluent radioactivity is generally less than ambient air radioactivity.

Liquid wastes released to sanitary sewers, a controlled area as provided for by 10 CFR 20, are generated at the Headquarters Site only. Liquid wastes are discharged from Building 001 only following holdup and analysis on a volume batch basis. There is no continuous flow. Building 004 liquid chemical wastes are released to a proportional sampler installation which retains an aliquot each time a fixed volume is released to the sanitary sewers. No liquid effluents are released from the Santa Susana Buildings 020 or 055, except as controlled liquid radioactive waste solidified for land burial. The average concentration and total radioactivity in liquid effluents discharged during 1977 are shown in Table 7.

TABLE 7

LIQUID EFFLUENT DISCHARGED TO SANITARY SEWER - 1977

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 Radio-activity Released (Ci)
001	Retention Tank	1,500	α	1.2×10^{-9}	5.5×10^{-8}	5.5×10^{-8}	3.1×10^{-7}
			β	4.1×10^{-9}	6.5×10^{-8}	6.5×10^{-8}	3.7×10^{-7}
004	Proportional Sampler	1,408,200	α	1.2×10^{-9}	$<1.8 \times 10^{-8}$	1.9×10^{-7}	$<9.8 \times 10^{-5}$
			β	4.1×10^{-9}	$<1.0 \times 10^{-7}$	5.8×10^{-6}	$<5.4 \times 10^{-4}$
020*	-	0	-	-	-	-	0
055*	-	0	-	-	-	-	0

*All liquid radioactive wastes from these facilities are solidified and land buried as dry waste.

IV. ENVIRONMENTAL MONITORING PROGRAM*

A. GENERAL DESCRIPTION

Environmental soil and vegetation sample collection and analysis for radioactivity were initiated in 1952, in the Downey, California, area, where the AI Division initially was located. Environmental sampling subsequently was extended to the proposed Sodium Reactor Experiment (SRE) site in the Simi Hills in May of 1954. In addition, sampling was begun in the Burro Flats area, southwest of SRE, where other nuclear installations were planned, some of which are currently in operation. The Downey area survey was terminated when the Division relocated to Canoga Park in 1955. The primary purpose of the environmental monitoring program is to survey environmental radioactivity adequately to assure that AI operations do not contribute significantly to environmental radioactivity.

Environmental radioactivity monitoring at the Energy Systems Group 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 on-site and to a distance of 10 mi (Figures 2, 3, and 4, Table 8). Continuous ambient air sampling and thermoluminescent dosimetry is performed on-site for monitoring airborne radioactivity and site ambient radiation levels, respectively. 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 warrant additional engineering safeguards to minimize the radioactivity levels in such effluents.

The sampling and analytic methods used in the environmental monitoring program for radioactive materials are described in Reference 3.

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*A separate and comprehensive report on environmental monitoring in the vicinity of Energy Systems Group facilities is issued annually. The material presented here was almost wholly abstracted directly from this report for 1977.(3)

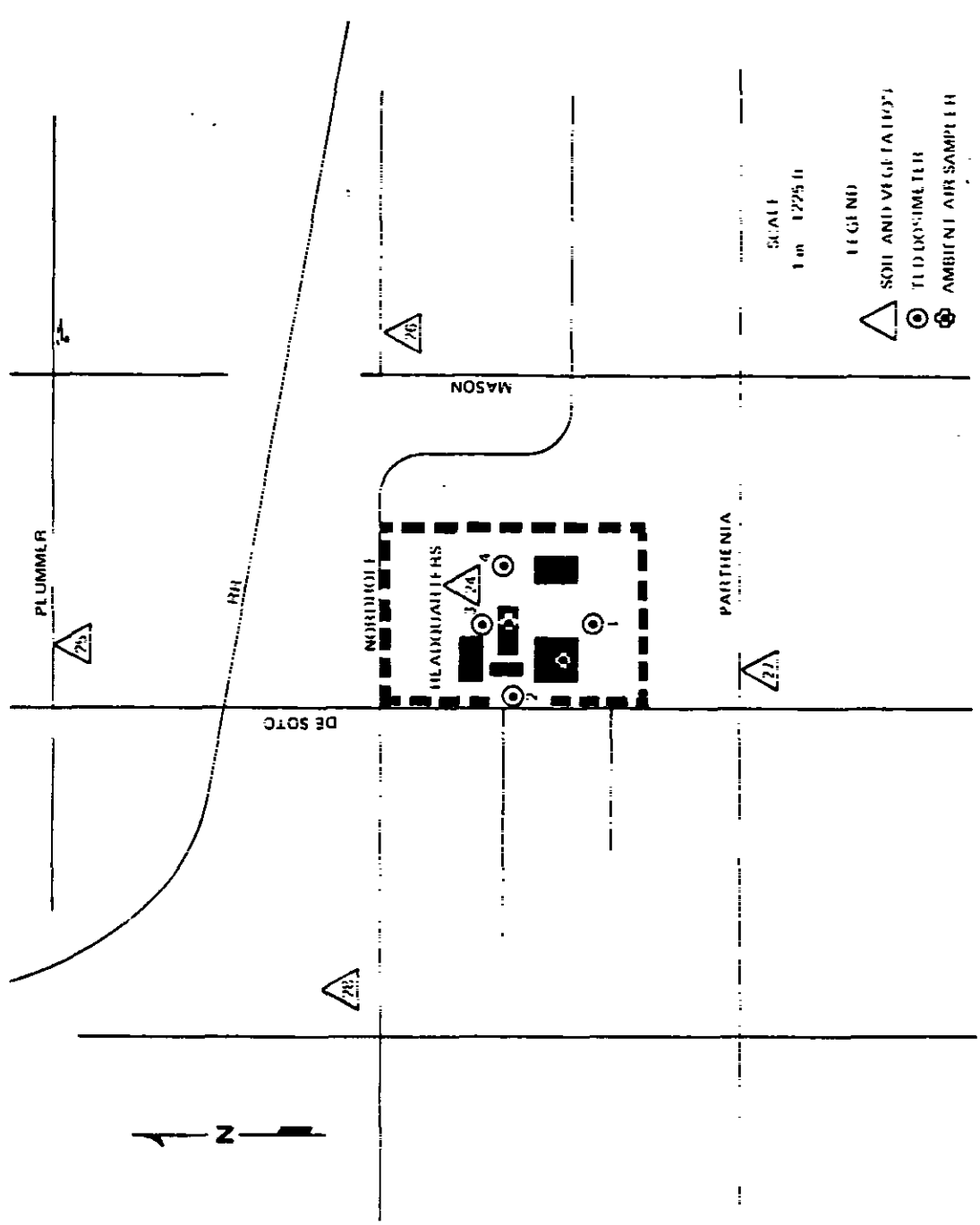

Figure 2. Map of Headquarters Vicinity Sampling Stations

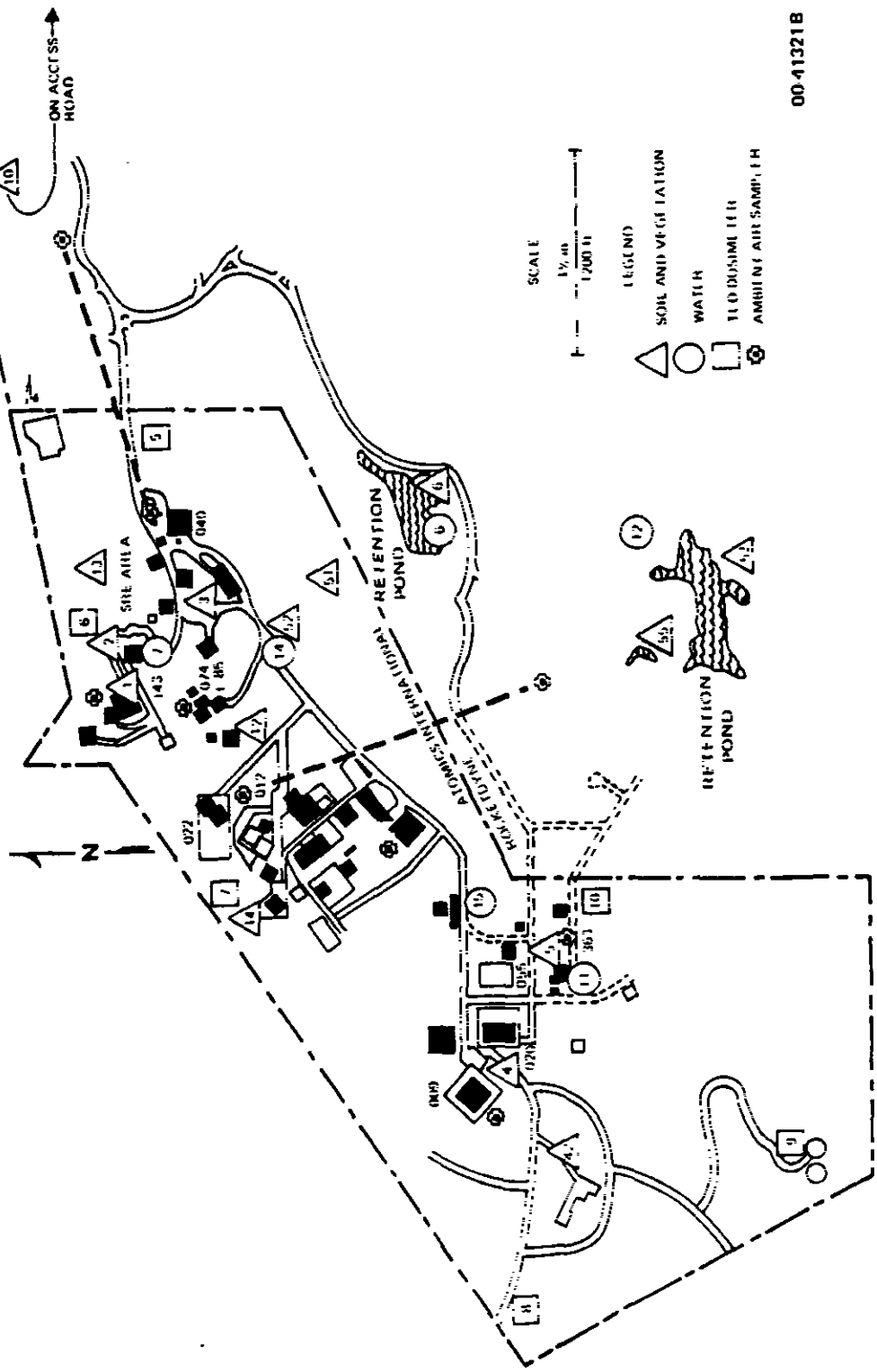

Figure 3. Map of SSFL Sampling Stations

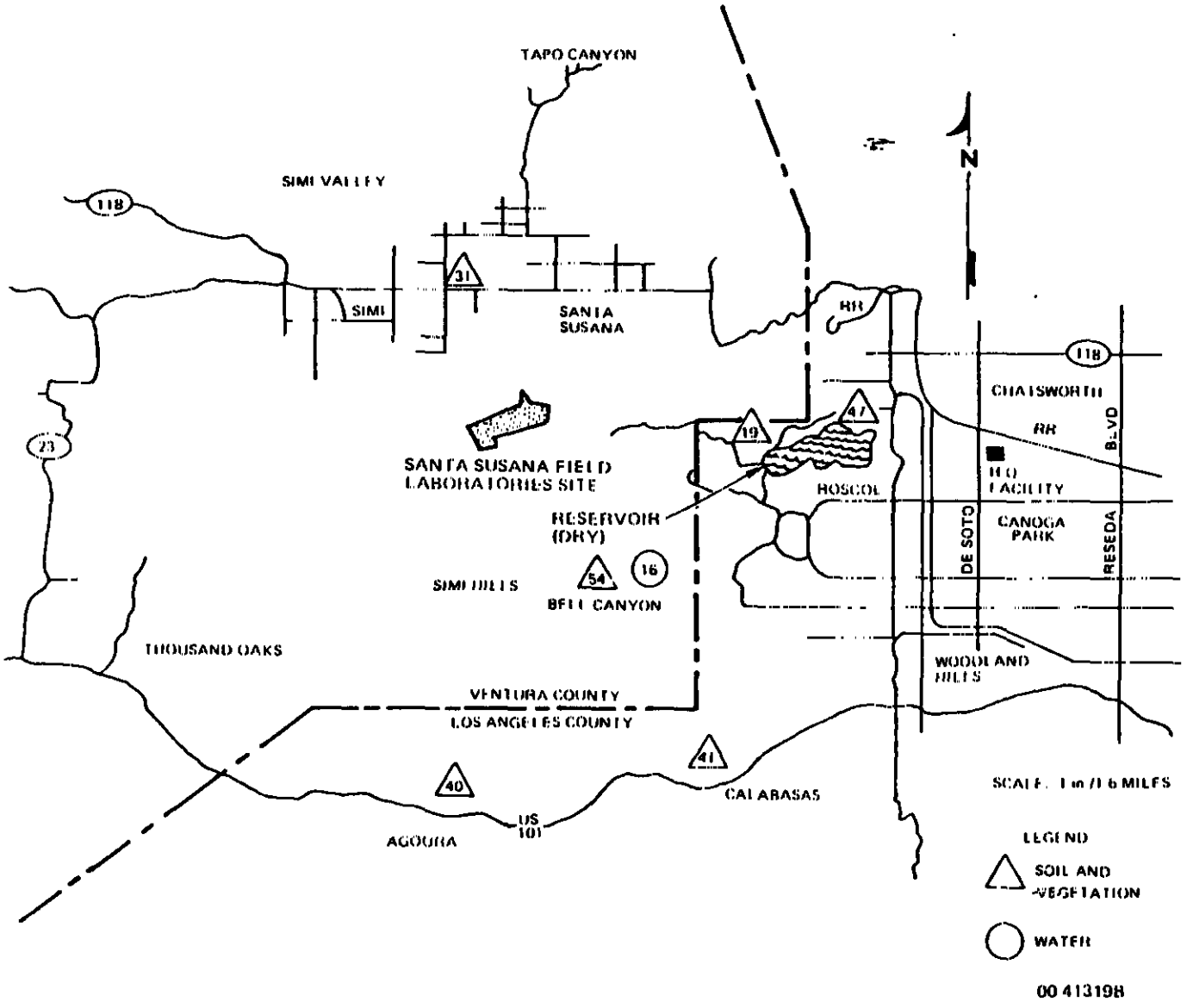

Figure 4. Map of Canoga Park, Simi Valley, Agoura, and Calabasas Sampling Stations

TABLE 8

ENVIRONMENTAL SAMPLE STATION LOCATIONS

Station	Location
SV-1	SRE Reactor, SSFL
SV-2	SRE Perimeter Drainage Ditch, SSFL
SV-3	Bldg. 064 Parking Lot, SSFL
SV-4	Bldg. 020, SSFL
SV-5	Bldg. 363, SSFL
SV-6	Rocketdyne Retention Pond, SSFL
SV-10	Santa Susana Site Access Road
SV-12	L-85 Reactor, SSFL
SV-13	Sodium Cleaning Pad, SSFL
SV-14	Below Bldg. 021-022, SSFL
SV-19	Santa Susana Site Entrance, Woolsey Canyon
SV-24	Atomics International Headquarters
SV-25	De Soto Avenue and Plummer Street
SV-26	Mason Avenue and Nordhoff Street
SV-27	De Soto Avenue and Parthenia Street
SV-28	Canoga Avenue and Nordhoff Street
SV-31	Simi Valley, Alamo Avenue and Sycamore Road
SV-40	Agoura - Kanan Road and Ventura Freeway
SV-41	Calabasas - Parkway Calabasas and Ventura Freeway
SV-42	Nonradioactive Materials Disposal Area, SSFL
SV-47	Chatsworth Reservoir North Boundary
SV-51	Bldg. 029, SSFL
SV-52	Burro Flats Drainage Control Pond, G Street and 17th Street, SSFL
SV-53	Top of Bell Canyon Below Rocketdyne Delta Pond Spillway
SV-54	Bell Creek
S-55	Rocketdyne Retention Pond

SV - Soil and Vegetation Sample Station
S - Soil Sample Station

TABLE 8

ENVIRONMENTAL SAMPLE STATION LOCATIONS
(Continued)

Station	Location
W-6	Rocketdyne Retention Pond
W-7	Process Water from Bldg. 003, SSFL
W-11	Process Water from Bldg. 363, SSFL
W-12	Rocketdyne Retention Pond R2A
W-16	Bell Creek
A-1	Atomics International Headquarters, Bldg. 001 Roof
A-2	Atomics International Headquarters, Bldg. 004 Roof
A-3	Bldg. 009, SSFL, Grade Level, West Side
A-4	Bldg. 011, SSFL, Grade Level, West Side
A-5	Bleg. 012, SSFL, Grade Level, West Side (Relocated to Bldg. 600 Rocketdyne on August 1, 1977)
A-6	Bldg. 040, SSFL, Grade Level, North Side (Relocated to Bldg. 207 Rocketdyne on August 1, 1977)
A-7	Bldg. 074, SSFL, Grade Level, South Side
A-8	Bldg. 143, SSFL, Grade Level, North Side
A-9	Bldg. 363, SSFL, Grade Level, South Side
TLD-1	Atomics International Headquarters, South of Bldg. 102 on Fence
TLD-2	Atomics International Headquarters, West of Bldg. 001 on Gate to Plant Water Supply Enclosure
TLD-3	ATomics Interantional Headquarters, Guard Post No. 1, Bldg. 201
TLD-4	Atomics International Headquarters, East Fence Gate
TLD-5	Bldg. 113, SSFL
TLD-6	SRE Retention Pond, SSFL
TLD-7	Electric Substation No. 719, SSFL
TLD-8	Property Line Gate, West End of H Street, SSFL
TLD-9	Water Tank No. 701, SSFL
TLD-10	Bldg. 854, SSFL
TLD-11	Offsite, Northridge
TLD-12	Offsite, Simi Valley
TLD-13	Offsite, Northridge

W - Water Sample Station
A - Air Sample Station
TLD - Thermoluminescent Dosimeter Location

The average radioactivity concentrations in local soil, vegetation, and ambient air for 1977 are presented in Tables 9 through 11. In calculating the average concentration value for these 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 significant 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 soil and vegetation radioactivity results reported in Tables 9 and 10 show no significant difference between onsite and offsite 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 remaining 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 SSFL is obtained from Ventura County Water District 8, which supplies nearby communities, and is distributed onsite by the same piping system previously used when process water was obtained from onsite wells. Conversion was completed during 1969. One onsite well was reactivated during June to

TABLE 9

SOIL RADIOACTIVITY DATA - 1977

Area	Activity	Number Samples	Gross Radioactivity (μCi/g) Annual Average Value (95% Confidence Level)	Maximum Observed Value*
Onsite	α	144	$(5.6 \pm 1.5)\ 10^{-7}$	1.1×10^{-6}
	β	144	$(2.4 \pm 0.09)\ 10^{-5}$	3.1×10^{-5}
Offsite	α	48	$(5.3 \pm 1.5)\ 10^{-7}$	8.5×10^{-7}
	β	48	$(2.3 \pm 0.08)\ 10^{-5}$	2.7×10^{-5}

*Maximum value observed for single sample

TABLE 10

VEGETATION RADIOACTIVITY DATA - 1977

Area	Activity	Number Samples	Gross Radioactivity (μCi/g) Dry Weight Annual Average Value	Ash Annual Average Value (95% Confidence Level)	Ash Maximum Observed Value*
Onsite	α	144	$(<3.7 \pm 2.9)\ 10^{-8}$	$(<2.2 \pm 1.7)\ 10^{-7}$	1.1×10^{-6}
	β	144	$(2.3 \pm 0.04)\ 10^{-5}$	$(1.62 \pm 0.03)\ 10^{-4}$	5.87×10^{-4}
Offsite	α	48	$(<5.4 \pm 3.3)\ 10^{-8}$	$(<2.1 \pm 1.6)\ 10^{-7}$	1.0×10^{-6}
	β	48	$(3.3 \pm 0.07)\ 10^{-5}$	$(1.42 \pm 0.03)\ 10^{-4}$	2.57×10^{-4}

*Maximum value observed for single sample

TABLE 11

AMBIENT AIR RADIOACTIVITY DATA - 1977

Site Location	Activity	Number Samples	Average Value (95% Confidence Level)	Maximum* Observed Value (daily)	% of Guide†
Headquarters	α§	729	$(<6.6 \pm 7.7)\ 10^{-15}$	3.9×10^{-14}	<0.22
(μCi/mℓ)	β**	729	$(<1.7 \pm 0.2)\ 10^{-13}$	3.0×10^{-12}	<0.057
SSFL	α§	2438	$(<6.6 \pm 7.5)\ 10^{-15}$	3.5×10^{-14}	<0.22
(μCi/mℓ)	β**	2438	$(<1.7 \pm 0.2)\ 10^{-13}$	2.8×10^{-12}	<0.057

*Maximum value observed for single sample.

†Guide: Headquarters, 3×10^{-12} μCi/mℓ , 3×10^{-10} μCi/mℓ ; 10 CFR 20 Appendix B, SSFL, 6×10^{-14} μCi/mℓ , 3×10^{-11} μCi/mℓ ; 10 CFR 20 Appendix B, CAC 17, and DOE Manual Chapter 0524

§MDL = 6.2×10^{-15} μCi/mℓ-Individual daily samples with activity levels of 0 to 6.2×10^{-15} μCi/mℓ are recorded and averaged as 6.2×10^{-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.

reduce Ventura County domestic water consumption as a water conservation measure due to the local drought conditions. The well water proportion in the blend averaged about 56% for the 6-month period ending in November at which time 100% county water was used again. Pressure is provided by elevated storage tanks.

Water from the system is sampled monthly at two widely separated SSFL locations. The process water radioactivity concentrations are presented in Table 12 for 1977.

Surface waters discharged from SSFL facilities and the sewage plant effluent drain southward into a retention pond on Rockwell (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

TABLE 12

SSFL PROCESS WATER RADIOACTIVITY DATA - 1977

Area	Activity	No. Samples	Gross Radioactivity (μCi/mℓ) Average Value (95% Confidence Level)	Maximum* Observed Value
AI-	α	24	$(<2.5 \pm 2.0)\ 10^{-10}$	3.0×10^{-10}
SSFL	β	24	$(2.5 \pm 0.7)\ 10^{-9}$	3.6×10^{-9}

*Maximum value observed for single sample

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. It is located 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 for 1977 are presented in Table 13.

Comparison of the radioactivity concentrations in water from the ponds and from Bell Creek with that of the supply water does not show any significant variation in either alpha or beta radioactivity.

Figure 5 is a graph of the daily averaged long-lived alpha and beta ambient air radioactivity concentrations for the Headquarters and SSFL facilities during 1977. The average beta concentration for each month is indicated by horizontal bars. The graph shows few prominent peaks occurring during the first 9 months, followed by a large increase in concentration during late September and early October with subsequent decreasing levels through the year's end.

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

TABLE 13

BELL CREEK AND ROCKETDYNE SSFL RETENTION POND
RADIOACTIVITY DATA - 1977

Area	Activity	No. Samples	Gross Radioactivity: Average Value (95% Confidence Level)	Gross Radioactivity: Maximum* Observed Value	% of Guide†
Bell Creek Mud Creek No. 54 (μCi/g)	α	12	$(2.9 \pm 1.0)\ 10^{-7}$	4.5×10^{-7}	NA
	β	12	$(2.2 \pm 0.08)\ 10^{-5}$	2.4×10^{-5}	NA
SSFL Pond Mud No. 55 (μCi/g)	α	12	$(6.3 \pm 1.5)\ 10^{-7}$	8.9×10^{-7}	NA
	β	12	$(2.4 \pm 0.09)\ 10^{-5}$	2.6×10^{-5}	NA
Bell Creek Vegetation No. 54 (μCi/g ash)	α	12	$(<1.9 \pm 1.6)\ 10^{-7}$	3.2×10^{-7}	NA
	β	12	$(1.55 \pm 0.03)\ 10^{-4}$	2.05×10^{-4}	NA
Bell Creek Vegetation No. 54 (μCi/g dry weight)	α	12	$(<4.8 \pm 4.0)\ 10^{-8}$	1.3×10^{-7}	NA
	β	12	$(3.6 \pm 0.07)\ 10^{-5}$	5.4×10^{-5}	NA
Bell Creek Water No. 16 (μCi/mℓ)	α	12	$(<2.4 \pm 2.9)\ 10^{-10}$	$<2.4 \times 10^{-10}$	<0.006
	β	12	$(1.8 \pm 0.8)\ 10^{-9}$	2.6×10^{-9}	0.6
SSFL Pond Water No. 6 (μCi/mℓ)	α	12	$(<2.4 \pm 2.9)\ 10^{-10}$	$<2.5 \times 10^{-10}$	<0.006
	β	12	$(4.3 \pm 0.8)\ 10^{-9}$	6.4×10^{-9}	1.4
SSFL Water No. 12 (μCi/mℓ)	α	12	$(<2.5 \pm 2.9)\ 10^{-10}$	2.8×10^{-10}	<0.006
	β	12	$(5.2 \pm 0.9)\ 10^{-9}$	1.3×10^{-8}	1.7

*Maximum value observed for single sample
†Guide: 5×10^{-8} μCi/m$\ell\alpha$, 3×10^{-7} μCi/m$\ell\beta$, 10 CFR 20 Appendix B, CAC 17, DOE Manual Chapter 0524
NA - not applicable, no guide values having been established for these types of environmental material.

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 (Figures 2 and 3) on or near the perimeters of the AI Headquarters and SSFL sites. Each dosimeter, sealed in a light-proof, energy compensation shield, is installed in a polyethylene container mounted 1 m above ground at each location. The dosimeters are exchanged and evaluated quarterly. There were 10 onsite TLD monitoring locations used during the year with 3 additional perimeter locations added at Headquarters in November which indicated an equivalent annual dose of 134 mrem for their 9 weeks of exposure. Three additional dosimeter sets, located offsite at locations up to 10 miles from the sites, are similarly evaluated to determine the local area ambient radiation level, which averaged 0.012 mRem/h for 1977. The average radiation dose rates and equivalent annual dose monitored at each dosimeter location are presented in Table 14.

TABLE 14
SITE AMBIENT RADIATION DOSIMETRY DATA - 1977

Dosimeter Location	Average Dose Rate (mRem/h)	Equivalent Annual Dose (mRem)
TLD-1 Headquarters	0.014	125
TLD-2 Headquarters	0.013	114
TLD-3 Headquarters	0.012	108
TLD-4 Headquarters	0.013	118
TLD-5 SSFL	0.014	125
TLD-6 SSFL	0.016	137
TLD-7 SSFL	0.014	125
TLD-8 SSFL	0.016	138
TLD-9 SSFL	0.010	86
TLD-10 SSFL	0.013	117

TABLE 14
SITE AMBIENT RADIATION DOSIMETRY DATA - 1977
(Continued)

Dosimeter Location	Average Dose Rate (mRem/h)	Equivalent Annual Dose (mRem)
TLD-11 Offsite	0.012	105
TLD-12 Offsite	0.012	108
TLD-13 Offsite	0.012	106

The table shows that radiation dose rates and equivalent annual doses monitored onsite are nearly identical to levels monitored at three widely separated offsite 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 gases and their particulate radioactive fallout from nuclear weapons tests. Locally, this dose rate is approximately 100 mRem/year. The small variability observed in the data is attributed to differences in elevation and geologic conditions at the various dosimter locations. Since the data for the onsite and offsite locations are nearly identical, it is concluded that no measurable radiation dose to the general population or to individuals in uncontrolled areas resulted from AI operations.

B. SUMMARY OF RESULTS

The average radioactivity concentrations in local soil, vegetation, and ambient air for 1977 were presented in Tables 9, 10, and 11.

The maximum level of radioactivity detected for a single sample is reported because of its potential significance in indicating the existence of a major episode or area-wide location of radioactive material deposition. Except for ambient air radioactivity, none of the maximum observed values, which occurred randomly during this year, show a great increase over the average values beyond

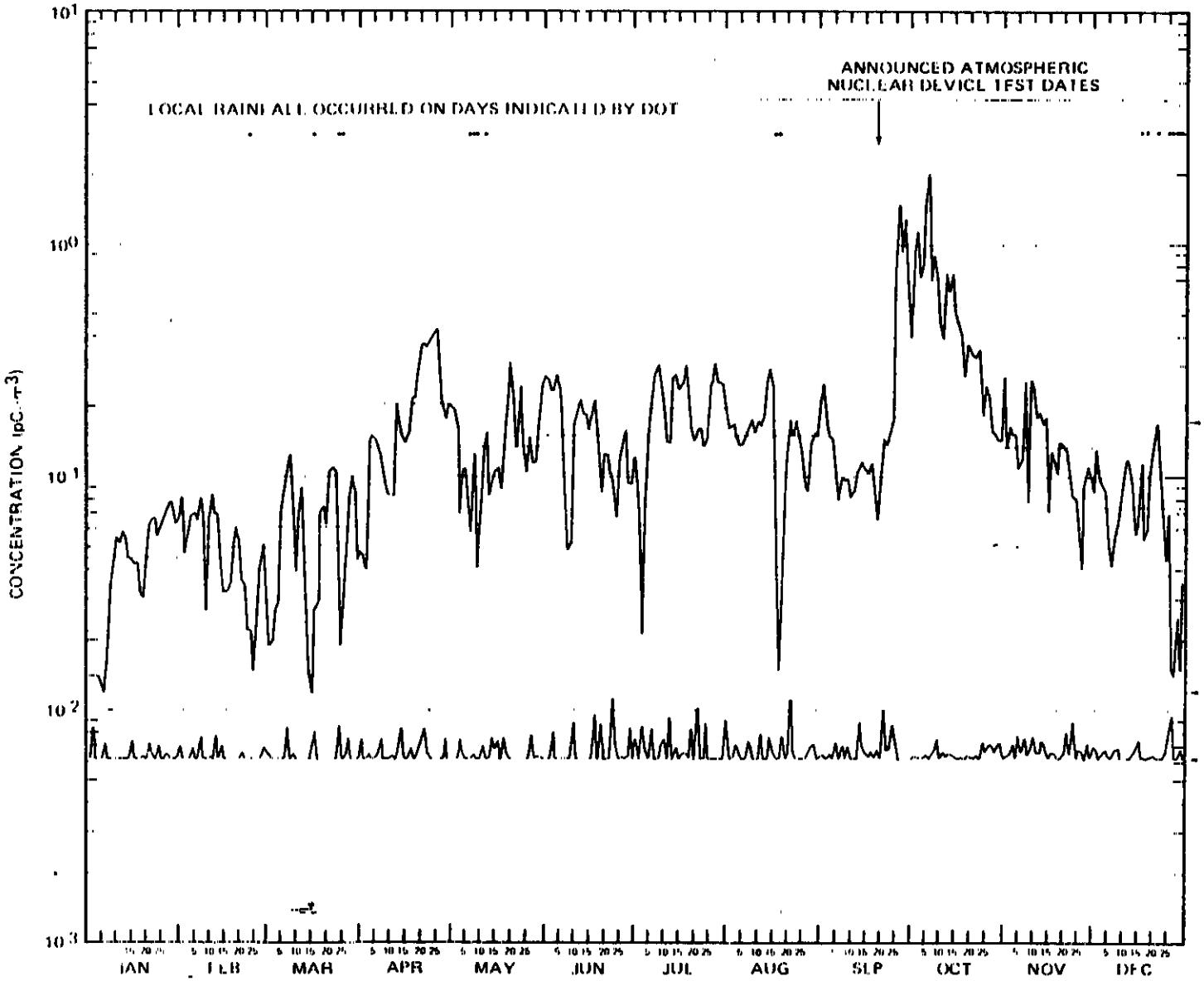

Figure 5. Daily Averaged Long-Lived Airborne Radioactivity at Headquarters and SSFL - 1977

that expected from natural variability. The air sample data (Figure 5) reflect a September atmospheric nuclear device detonation which resulted in a marked but transient increase in local atmospheric radioactivity levels.

The soil and vegetation results reported in Tables 9 and 10 show no significant difference between onsite and offsite 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 particulate daughters). Radioactivity from nuclear device testing fallout consists primarily of the long-lived fission proudcts Sr^{90}-Y^{90}, Cs^{137}, and Pm^{147}, and also U^{235} and Pu^{239}.

C. CONCLUSIONS

The random variations observed in the environmental monitoring data indicate that no significant local source of artificial radiation or radioactive material existed in the environs. Additionally, the similarity between onsite and offsite results further substantiate that the contribution to the general environmental radioactivity and radiation levels from operations at Atomics International was essentially zero.

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

V. UNUSUAL EVENTS

Building 001

On April 26, a small fire occurred in the ATR Quality Assurance Laboratory (Room 11-B-40). By the time the HS&RS personnel arrived, the fire had been extinguished. Nasal smears were obtained from all personnel (7) who had been in the area at the time. Smears were also taken of hands and shoes. All of these smears proved negative (<20 dpm). Smears were also taken of the immediate floor area; these also were negative. A 10-min sample from the high volume air sampler (15 cfm) was also negative (<20 dpm). About 20 min after the incident, the area was released for normal activity. No measurable external or internal exposures were incurred as a result of this incident.[4]

Building 004

None of note. The SNAP machine shop was deactivated. All the equipment used for the machining of depleted uranium was either decontaminated or disposed of as R/A waste. The room was subsequently declared free of contamination. The absolute filters in plenum EF 406 were removed. The hoods that are vented to exhaust 406 are no longer used to work with R/A materials. The fuel (liquid) in the L-77 reactor core was removed in drums and sent to the SNM vault for storage.

Building 020

None.

Building 055

None.

VI. SUMMARY/TRENDS - EXPOSURES, EFFLUENTS

A comparison of the 1977 data with that of the previous years reveals:

1) Although personnel radiation exposures continued to increase somewhat from 1976 and 1975 (67 man-Rem - 1977, 48 man-Rem - 1976, 27 man-Rem - 1975), all individual exposures remained well below the annual occupational limit of 5 Rem (see Table 1).
2) There continues to be an upward trend in total personnel exposure (man-Rem). An investigation into the source of this increase reveals that it is almost solely due to D&D (decontamination and decommissioning) activities. Hence, the somewhat increased exposures incurred now probably will be more than offset by a decrease in potential exposures in future years of utilization of the area. Moreover, such decontamination reduces the potential exposure from any accident-related release.

Consequently, it is concluded that while some increases in personnel exposures were necessarily incurred in D&D activities, they have been acceptable, and should provide for a net decrease in overall radiation exposures in the long run.

VII. ANTICIPATED ACTIVITIES DURING NEXT REPORTING PERIOD (1978)

A. BUILDING 001/04

Continued routine operations of fuel element fabrication with supportive chemistry, etc.

B. BUILDING 020

Continue decladding 150 HNPF fuel assemblies, remove sodium bond, package and ship the recovered fuel to Savannah River for eventual reprocessing. Package high-level R/A waste for shipment to Beatty, Nevada for burial.

C. BUILDING 055

All Pu was removed from this facility in the spring of 1978. A program using depleted uranium (as uranium carbide) was initiated.

REFERENCES

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

2. "Annual Review of Radiological Controls - 1976," R. S. Hart, N001TI99003, Energy Systems Group, Rockwell International, 1980

3. "Atomics International Environmental Monitoring and Facility Effluent Annual Report - 1977," J. D. Moore, AI-78-16, Rockwell International, Atomics International Division, April 1978

4. "ATR-QA Lab Fire/Incident Report," IL dated May 4, 1977, R. J. Tuttle from J. H. Wallace

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 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 I. 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

A-4

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.

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.

Some data pertinent to these bioassay services are shown in Table A-1.

TABLE I

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.

TABL) (Continued)

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
lutonium (B)	PU B	Plutonium	0.0495 dpm	± 75%	1000 ml	Double precipitations, washes and extractions are eliminated for faster analysis at reduced accuracy.
lutonium (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.
trontium-90	SR90	Strontium-90	30 dpm	± 50%	200 ml	
horium	TH	Thorium	0.99μg	± 50%	1000 ml	
ross Beta-High Level	GBH	All beta emitters except halogens	750 dpm	± 75%	50 ml	K^{40} corrected
ross Alpha (1a)	GA1A	Uranium and Plutonium	1.5 dpm	± 50%	100 ml	Sample electrodeposited on SS disc and autoradiographed.
ross Alpha (1b)	GA1B	Uranium and Plutonium	9 dpm	± 50%	100 ml	Sample planchetted and proportional counted for alpha.

TABLE A-1 (continued)

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.