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Rockwell International
Energy Systems Group

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ABSTRACT

Data on exposures of Energy Systems Group personnel at NRC-licensed facilities to radiation and/or radioactivity are presented for CY 1979. 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 1979, it is concluded that the integrated personnel exposure decreased about 17% from 1978 (from 110 man-rem to 91 man-rem). This reverses the upward trend that existed between 1975-1978.

This report satisfies License Condition 23 of Special Nuclear Materials License SNM-21 for 1979.

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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, DeSoto 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.

*The most recent previous report is for the year 1978[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, injection, or possibly through cuts or puncture woulds (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 within a specific annual dose range is shown along with the percentage of employees within each annual dose range or less and the man-rem contributed by each dose range 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 the few points available can be grouped rather well on a straight line. It also should be noted (see Summary, Section VI) that all exposures were less than the 5 rem annual occupational limit, with the average being about 54 mrem or 1.1% of this value. Even the highest group of annual exposures only amounts to an average of ~1.5 rem or 30% of the permissible amount. Compared to the previous year's data (1978), the total man-rem decreased from 110 to 91, or about 17%.

*Doses listed as "no measureable exposure" were not used in the plot of Figure 1.

TABLE 1

PERSONNEL EXTERNAL EXPOSURES -- 1979

Dose Range (rem)	P (People)	CP (Cumulative People)	CP (%)	Pop.-Dose* (man-rem)
No Measurable Exposure	1225	1225	61.90	0
<0.100	347	1572	94.19	17.35
0.100-0.250	39	1572	96.53	6.83
0.250-0.500	19	1630	97.66	7.13
0.500-0.750	10	1640	98.26	6.25
0.750-1.00	4	1644	98.50	3.50
1.00-2.00	15	1659	99.40	22.50
2.00-3.00	8	1667	99.98	20.00
3.00-4.00	2	1669	100.00	7.00
>4.00-5.00	0			
>5.00	None			90.56

*The mid-point of each dose range was assumed for the average dose in calculating the Population-Dose (man-rem).

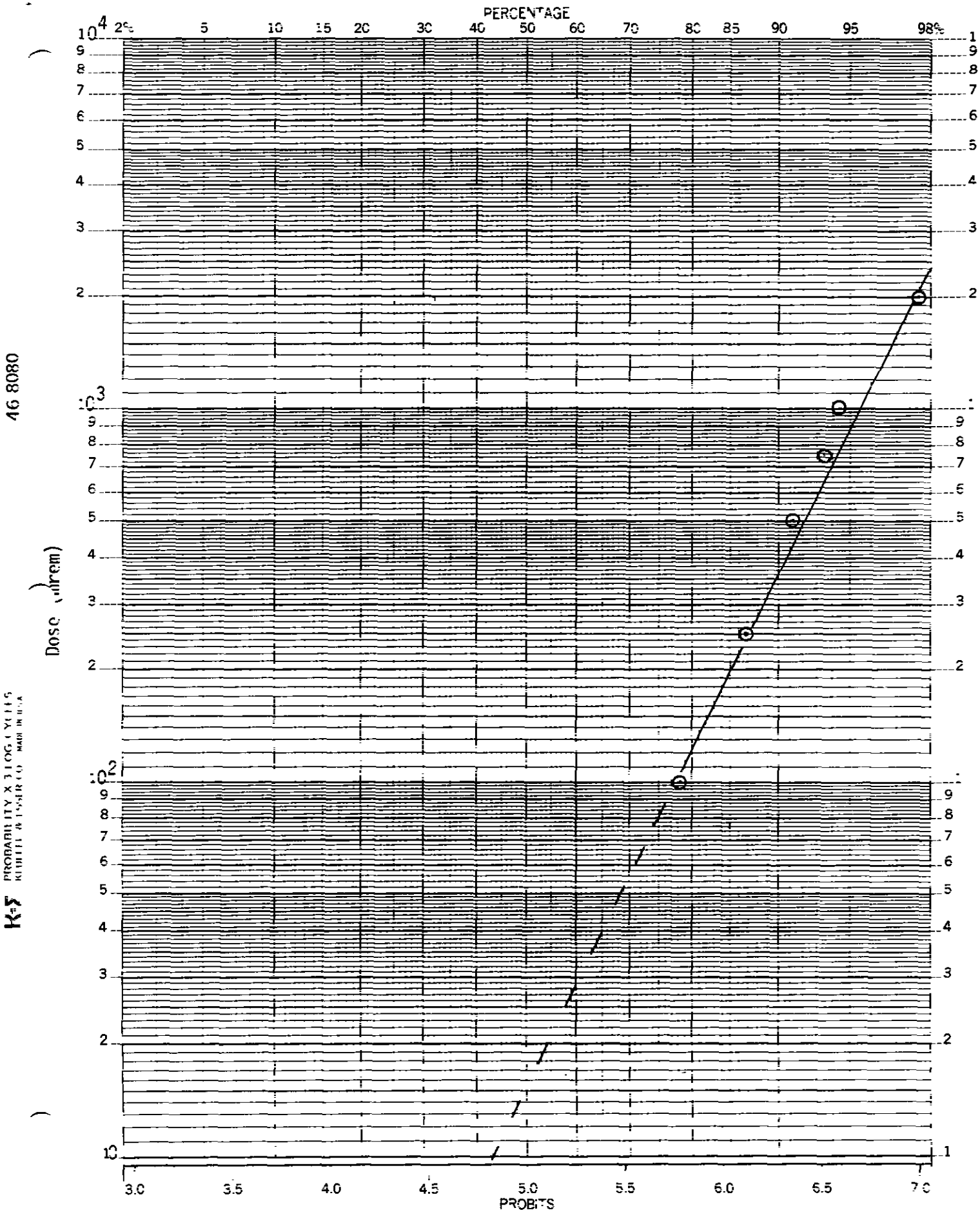

Figure 1. Distribution of Personnel External Radiation Exposure — 1979

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 respirable-sized particle range. These measurements are accomplished through the use of a whole body counter.* During 1979, forty-eight lung scans were made for uranium deposition. Eight of the scans (on eight different individuals) showed positive results. However, follow-up 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 the Appendix. A statistical summary of the results for 1979 appears in Table 2A, while a detailed listing of the positive results and follow-up measurements are shown in Table 2B. Data on the in-vivo lung scans performed in 1979 also appear in these tables.

*Helgeson Nuclear Services, Inc., Pleasanton, California

TABLE 2A

SUMMARY OF BIOASSAYS — 1979

Measurement*	Type*	Total Tests	Total Positive Results	Total Individuals With Positive Results
U	UF	263	13	13
U	UR	228	1	1
GA	1A	8	0	0
GB	1B	2	0	0
GB	2B	0	0	0
GB	H	0	0	0
Pu	A	54	1	1
FP	1	0	0	0
FP	3A	233	14	11
FP	3B	233	27	4
U	IVLC	48	8	8
Sr-90	Sr-90	2	0	0
Cs-137	TBC	15	15	15
H-3	H-3	10	0	0
Th	Th	3	0	0

*UF = Uranium — Fluorometric
UR = Uranium — Radiometric
GA = Gross Alpha
GB = Gross Beta
Pu = Gross Plutonium
FP = Fission Products
(For a discussion of specific analytical techniques employed, see appendix)
U-IVLC = Uranium In-Vivo Lung Count
TBC = Total Body Count
H-3 = Tritium
Th = Thorium
H = High Level

TABLE 2B

POSITIVE BIOASSAY RESULT SUMMARY — 1979
(Sheet 1 of 2)

H&S Number	Sample Date	Analysis Type*	Results		Specific Radionuclide	Equivalent MPBB* (%)
			Per Vol. Anal. (dpm)	Per 1500 ml-day (dpm)		
4271	1-15-79	PuA	0.053	0.080	Pu-239	0.007
4271	3-7-79	PuA	0	0	Pu-239	—
1292	10-22-79	IVLC	51 µg		U-235	20.8**
1584	1-18-79	IVLC	55 µg		U-235	22.2**
4537	3-26-79	UF	0.0005 µg	0.075 µg	U	0.75
4185	1-18-79	IVLC	47 µg		U-235	19**
1073	3-19-79	UF	0.0003 µg	0.45 µg	U	0.45
1073	4-17-79	UF	0	0	U	—
4440	5-11-79	IVLC	55 µg		U-235	22.4**
4440	10-22-79	IVLC	0		U-235	—
4440	8-3-79	UF	0.0003 µg	0.45 µg	U	0.45
4440	8-6-79	UF	0		U	—
4572	10-22-79	IVLC	46 µg		U-235	18.8**
3774	3-19-79	UF	0.0010 µg	1.5 µg	U	1.5
3774	4-16-79	UF	0	0	U	—
2739	3-18-79	UF	0.0016 µg	2.4 µg	U	2.4
2739	4-17-79	UF	0	0	U	—
4513	3-26-79	UF	0.0004 µg	0.60 µg	U	0.60
4513	10-2-79	UF	0.0002 µg	<MDL	U	—
4442	3-19-79	UF	0.0006 µg	0.90 µg	U	0.90
4442	4-16-79	UF	0		U	
4369	8-20-79	UF	0.0016 µg	2.40 µg	U	2.4
4369	9-24-79	UF	0.0001 µg	<MDL	U	—
4345	10-22-79	IVLC	52 µgm		U-235	21.2**
4345	2-6-80	IVLC	0		U-235	—
4211	10-2-79	UF	0.0007 µg	1.05 µg	U	1.05
4125	1-18-79	IVLC	40 µg		U-235	16.1**
4125	5-11-79	IVLC	0		U-235	—
4125	3-1-79	UF	0.0004 µg	0.60 µg	U	0.60
4125	3-19-79	UF	0.001 µg	<MDL	U	—

TABLE 2B

POSITIVE BIOASSAY RESULT SUMMARY — 1979
(Sheet 2 of 2)

H&S Number	Sample Date	Analysis Type*	Results		Specific Radionuclide	Equivalent MPBB (%)
			Per Vol. Anal. (dpm)	Per 1500 ml-day (dpm)		
1374	2-10-79	FP-3B	1050	7875	Cs-137	1.2
1374	2-11-79	FP-3B	496	3720	Cs-137	0.56
1374	2-12-79	FP-3B	347	2603	Cs-137	0.40
1374	2-14-79	FP-3B	208	1560	Cs-137	0.24
1374	2-15-79	FP-3B	201	1508	Cs-137	0.23
1374	2-16-79	FP-3B	153	1148	Cs-137	0.17
1374	2-18-79	FP-3B	137	1028	Cs-137	0.15
1374	2-28-79	FP-3B	165	1240	Cs-137	0.19
1374	3-5-79	FP-3B	47.1	353	Cs-137	0.05
1374	3-19-79	FP-3B	117	878	Cs-137	0.13
1374	3-26-79	FP-3B	104	780	Cs-137	0.12
1374	4-2-79	FP-3B	27.1	204	Cs-137	0.03
1374	4-9-79	FP-3B	92	690	Cs-137	0.10
1374	4-16-79	FP-3B	69.3	520	CS-137	0.08
1374	4-23-79	FP-3B	79.2	594	Cs-137	0.09
1374	4-29-79	FP-3B	56.6	425	Cs-137	0.06
1374	5-11-79	FP-3B	109.3	820	Cs-137	0.12
1374	5-11-79	TBC	49.66 nCi	—	Cs-137	0.17
1374	6-15-79	FP-3B	13.9	104	Cs-137	0.02
1374	7-30-79	FP-3B	4.8	<MDL	Cs-137	—
2729	8-12-79	UF	0.0031 µg	4.65 µg	U	4.65
2729	9-18-79	UF	0	—		—
2729	8-12-79	UR	0.54	8.1	U-235	4.1
2729	9-18-79	UR	0	—	U-235	—
2729	10-22-79	IVLC	36 µg		U-235	14.7**
2729	2-6-80	IVLC	0		U-235	

*IVLC: In-Vivo Lung Count
UF: Uranium — Fluorometric
UR: Uranium — Radiometric
GA: Gross Alpha
GB: Gross Beta
Pu: Gross Plutonium
FP: Fission Products
MDL: Minimum Detectable Level
MPBB: Maximum Permissable Body Burden
**MPLB: Maximum Permissable Lung Burden
TBC: Total Body Count
(For a brief description of the specific analytical techniques, see appendix)

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 possibly could have existed for very limited periods in certain locations.

TABLE 3

RADIATION LEVELS — WORKING AREAS — 1979

Building/ Area	Average Dose Rate* (mRem/h)	Maximum Dose Rate (mRem/h)	Remarks
001-Fuel Fab	0.2	8	Final element inspection area
004	0.035	0.15	
020	0.1 0.5	0.2 200	Uncontrolled areas Controlled areas
055	2-3	~80	Waste storage area

*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 local air sampling is performed. A summary of these results for 1979 is given in Table 4.

TABLE 4

INTERIOR AIR SAMPLE SUMMARY — 1979

Building	Maximum μCi/cc	Average* μCi/cc
001-Fuel Fab (Lapel)	(α) 1.5×10^{-10}	(α) 5.0×10^{-11}
004	(Not Required)	—
020 Controlled Areas Uncontrolled Areas	(β,γ) 8.5×10^{-12} (β,γ) 5.5×10^{-13}	$(\beta,\gamma)2 \times 15^{-12}$ $(\beta,\gamma)1 \times 10^{-13}$
055	(α) See Table 5	

*Estimated

C. SPECIAL AIR SAMPLES — BUILDING 055

In Building 055 (NMDF), local air samples were taken routinely at about 20 different locations adjacent to the glove box train and at several other room locations (e.g., vault, fan room). Upon inspection of the values given in Table 5, the week of 9-14 and, possibly, that of 10-5, would appear to occur more often than would be expected from pure chance. Also, the Fan Room and Chem Lab sampling locations usually have placed much lower, comparatively, on the list (1975-1978). Whether or not these general observations have any real significance is problematical. In any case, the maximum weekly integrated measurement of 2.3×10^{-11} $\frac{\mu Ci\text{-}hr}{cc}$ at the Chem Lab location, for the week ending September 14, is only 28% of the maximum permissable weekly integrated exposure (40 hr, occup.) for Pu-239 of 8×10^{-11} $\frac{\mu Ci\text{-}hr}{cc}$.

TABLE 5

INTERIOR AIR SAMPLES – BLDG 005 – 1979

Sampling Location	Maximum Cumulative Weekly Exposure (μCi-hr/ml)	Date of Maximum Exposure – Week(s) Ending
Chem Lab	2.3-11*	9-14
Fan Room	8.7-12	11-30
4 N	8.7-12	9-14
24 NE	8.1-12	11-30
3 AN	7.8-12	6-29
1 A	7.5-12	9-14
18 S	7.5-12	10-12
8 N	7.5-12	8-17
27 S	7.5-12	10-5
9 S	7.2-12	9-7
3 S	6.9-12	9-14
5 S	6.9-12	6-29
11 N	6.9-12	10-12
19 S	6.9-12	10-12
20 S	6.3-12	9-14
6 N	6.0-12	10-5, 10-19
1 W	5.7-12	10-5
15 S (fume hood)	5.7-12	10-12
24 SW	5.7-12	10-14
Support Area	3.3-12	3-16

*2.3-11 = 2.3×10^{-11}. To obtain average weekly concentration in μCi/ml, divide values shown in table by 40. The occupational MPC (40 hr/week) for the most restrictive radioisotope that could be present (Pu-239) is 2×10^{-12} μCi/ml or an integrated exposure of 8×10^{-11} $\frac{\mu Ci\text{-}hr.}{ml}$

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 after filtration and at the point of release. In addition, the continuous 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 99.9% efficient for particles of 0.3-μm diameter. Particle filtration efficiency increases above and below this size.

B. SUMMARY OF RESULTS

The average concentration and total radioactivity in gaseous effluents released in 1979 to unrestricted areas are shown in Table 6.

*A separate and comprehensive report on facility effluents and environmental monitoring is prepared annually. The data presented in Sections III and IV of this report were almost wholly abstracted from that report for 1979.[3]

TABLE 6

RADIOACTIVITY CONTENT OF ATMOSPHERICALLY DISCHARGED EFFLUENT RELEASED TO UNRESTRICTED AREAS 1979

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 ($\mu Ci/m\ell$)	Radioactivity Released (Ci)
001	Stack Exit	2.5×10^{10}	α	1.7×10^{-16}	$<2.9 \times 10^{-14}$	2.8×10^{-13}	$<2.1 \times 10^{-5}$
			β,γ	5.4×10^{-16}	$<8.1 \times 10^{-15}$	5.8×10^{-14}	$<5.8 \times 10^{-6}$
004	Stack Exit	3.3×10^{10}	α	3.5×10^{-16}	$<1.2 \times 10^{-15}$	5.2×10^{-15}	$<1.1 \times 10^{-6}$
			β,γ	10.7×10^{-16}	$<6.1 \times 10^{-15}$	1.3×10^{-13}	$<5.7 \times 10^{-6}$
020	Stack Exit	1.5×10^{10}	α	0.9×10^{-16}	$<4.2 \times 10^{-16}$	1.1×10^{-15}	$<5.3 \times 10^{-8}$
			β,γ	2.9×10^{-16}	1.0×10^{-13}	4.0×10^{-13}	2.1×10^{-7}
055	Stack Exit	6.1×10^{9}	α	2.8×10^{-16}	$<3.1 \times 10^{-16}$	1.1×10^{-15}	$<2.2 \times 10^{-5}$
			β,γ	8.2×10^{-16}	$<1.2 \times 10^{-16}$	9.2×10^{-15}	$<5.6 \times 10^{-5}$

Annual average ambient air radioactivity concentrations 1979

α $<2.4 \times 10^{-15}$ $\mu Ci/m\ell$

β,γ $<2.0 \times 10^{-14}$ $\mu Ci/m\ell$

Totals (all facilities) α $<2.2 \times 10^{-5}$ Ci

β,γ $<5.6 \times 10^{-5}$ Ci

*Facility effluent radioactivity is generally less than ambient air radioactivity because of filtration. However, during 1979, if facility released activity concentrations were at the ambient air average, a total of 1.4×10^{-5} (α) curies and 4.5×10^{-5} (β) curies would have been discharged. The measured discharge values were about 60% greater (α) and 24% greater (β,γ), but still well below the permissable limits.

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 1979 are shown in Table 7.

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TABLE 7

LIQUID EFFLUENT DISCHARGED TO SANITARY SEWER – 1979

Building	Point of Release	Approximate Effluent Volume (gal)	Activity Monitored	Approximate MDL (μCi/mℓ)	Annual Average Concentration (μCi/mℓ)	Sample Maximum Observed Concentration (μCi/mℓ)	Total Radioactivity Released (Ci)
001	Retention Tank	54,000	α	1.2×10^{-9}	1.5×10^{-7}	1.2×10^{-6}	3.0×10^{-5}
			β	3.7×10^{-9}	1.1×10^{-7}	6.5×10^{-7}	2.2×10^{-5}
004	Proportional Sampler	1,641,000	α	1.2×10^{-9}	$<1.1 \times 10^{-8}$	7.0×10^{-8}	$<6.9 \times 10^{-5}$
			β	3.7×10^{-9}	3.1×10^{-8}	1.4×10^{-7}	1.9×10^{-4}
020*	--	0	--	--	--	--	--
055*	--	0	--	--	--	--	--

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

IV. ENVIRONMENTAL MONITORING PROGRAM

The basic policy for control of radiological and toxicological hazards at ESG requires that through engineering controls adequate containment of such materials be provided, and through rigid operational controls, that facility effluent releases and external radiation levels are reduced to a minimum. The environmental monitoring program provides a measure of the effectiveness of the Group safety procedures and of the engineering safeguards incorporated into facility designs. Specific radionuclides in facility effluent or environmental samples, although not routinely identified due to the extremely low radioactivity levels normally detected, may be identified by analytical or radiochemistry techniques if significantly increased radioactivity levels are observed.

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).

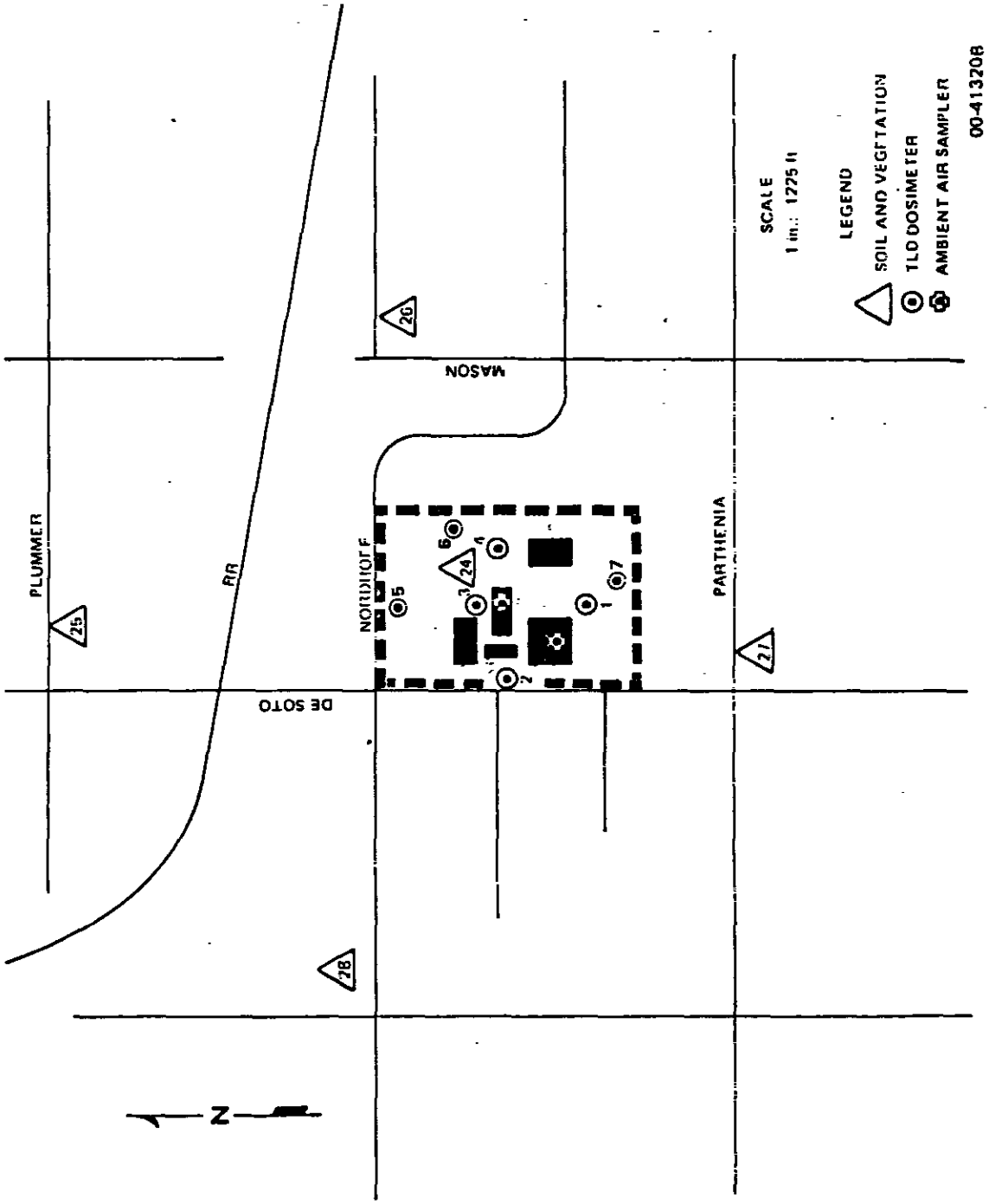

Figure 2. Map of DeSoto Site and Vicinity Sampling Stations

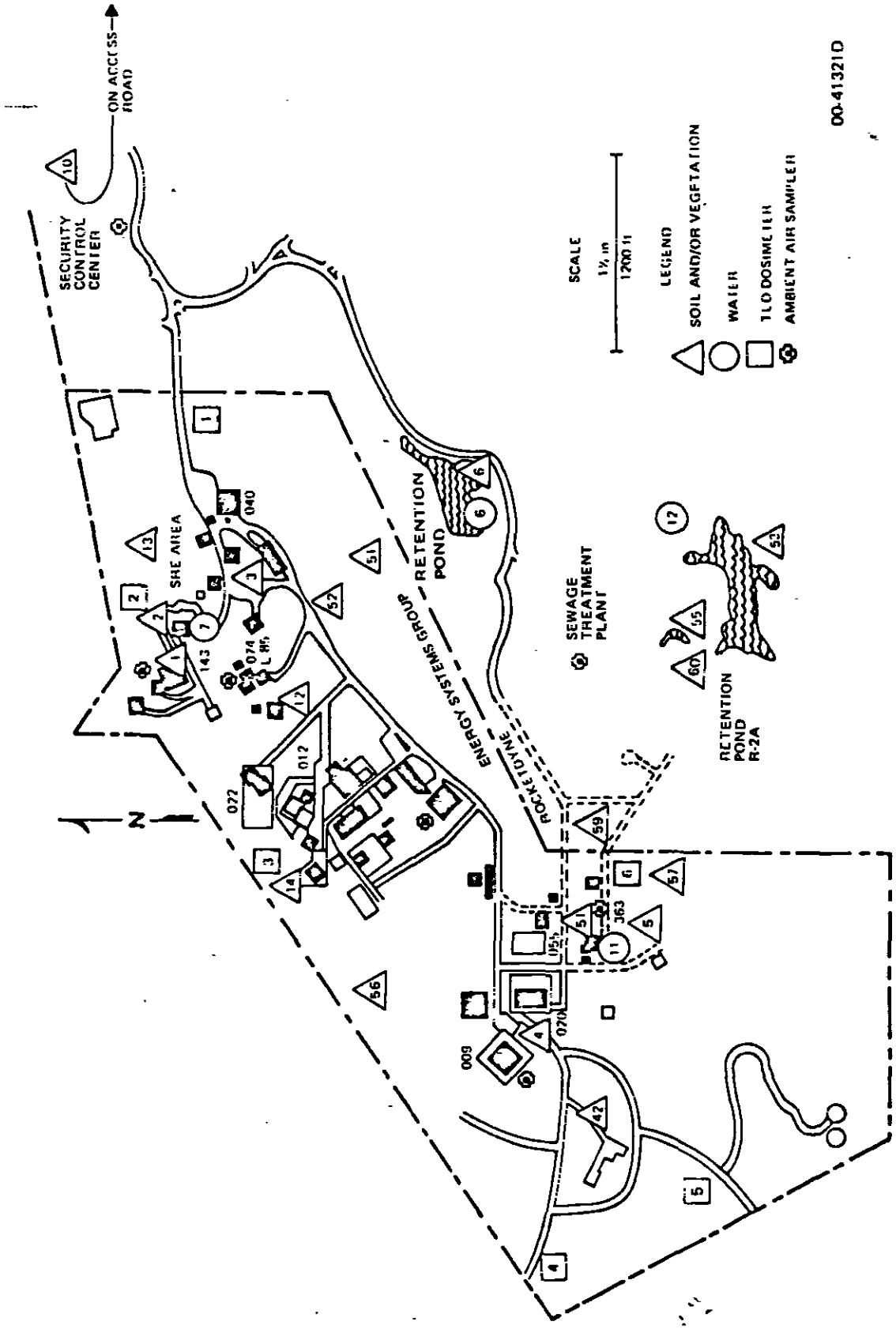

Figure 3. Map of Santa Susana Field Laboratories Site Sampling Stations

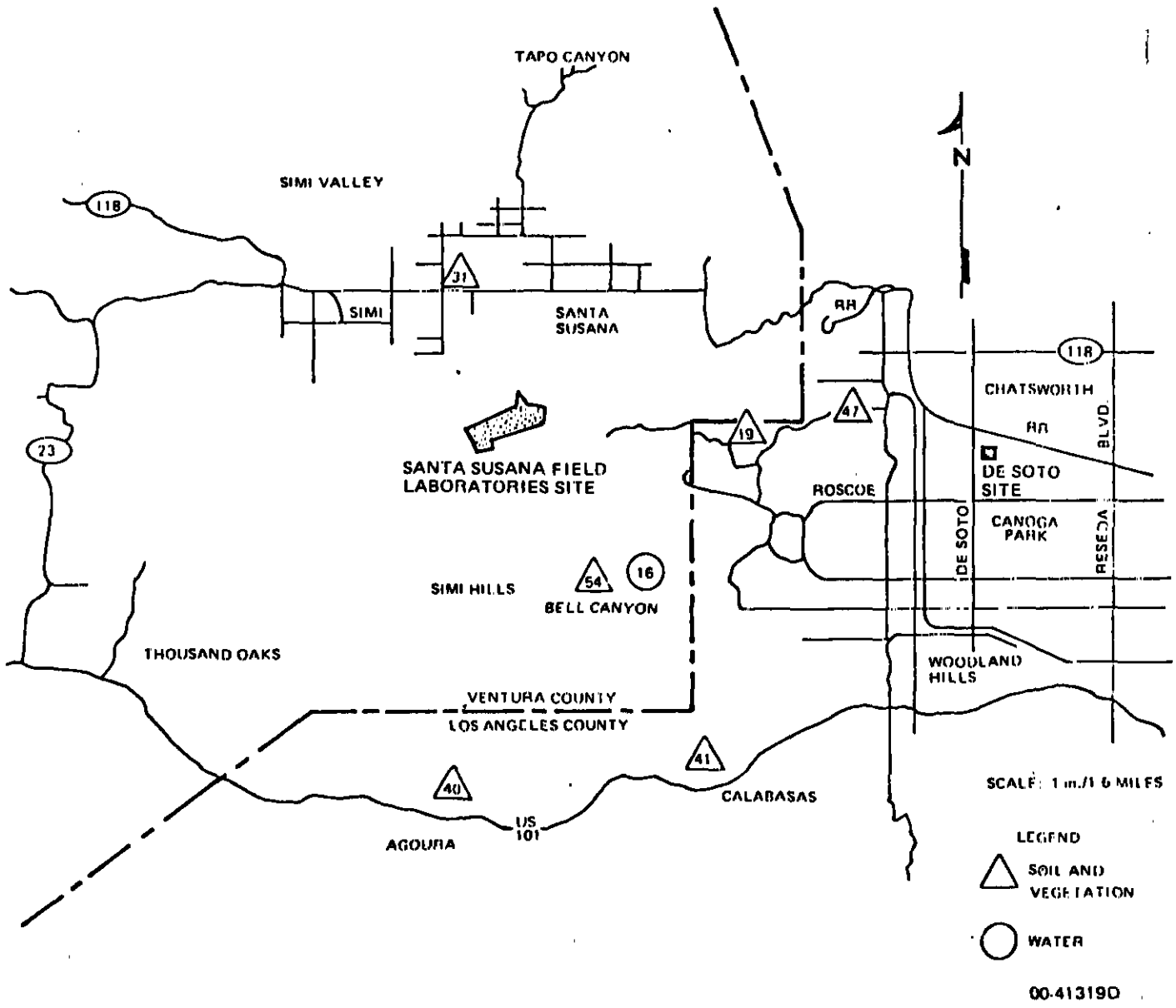

Figure 4. Map of Canoga Park, Simi Valley, Agoura, and Calabasas Sampling Stations

TABLE 8

SAMPLE STATION LOCATIONS
(Sheet 1 of 3)

Station	Location
SV-1	SSFL Site, Bldg. 143
SV-2	SSFL Site, Bldg. 143 Perimeter Drainage System
SV-3	SSFL Site, Bldg. 064
SV-4	SSFL Site, Bldg. 020
SV-5	SSFL Site, Bldg. 363
SV-6	Rocketdyne Site Interim Retention Pond
SV-10	SSFL Site Access Road
SV-12	SSFL Site, Bldg. 093 (L-85 Reactor)
SV-13	SSFL Site, Below Sodium Cleaning Facility at SRE Pond
SV-14	SSFL Site, Bldg. 028
SV-19	SSFL Site Entrance, Woolsey Canyon
SV-24	De Soto Site, Bldg. 004
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	SSFL Site, Bldg. 886
SV-47	Chatsworth Reservoir North Boundary
SV-51	SSFL Site, Bldg. 029
SV-52	SSFL Site, Burro Flats Drainage Control Pond, G Street and 17th Street
SV-53	Rocketdyne Site Pond R-2A Spillway, Head of Bell Canyon
SV-54	Bell Creek
S-55	Rocketdyne Site Retention Pond R-2A (Pond Bottom Mud)
S-56	SSFL Site, F Street and 24th Street

SV — Soil and Vegetation Sample Station
S — Soil Sample Station

TABLE 8

SAMPLE STATION LOCATIONS
(Sheet 2 of 3)

Station	Location
S-57	SSFL Site, J Street at Bldg. 055
S-58	SSFL Site, Bldg. 353
S-59	Rocketdyne Site Test Area CTL 4
S-60	Rocketdyne Site Retention Pond R-2A
W-6	Rocketdyne Site Interim Retention Pond (drains to Pond R-2A)
W-7	SSFL Site Domestic Water, Bldg. 003
W-11	SSFL Site Domestic Water, Bldg. 363
W-12	Rocketdyne Site Area II Final Retention Pond R-2A
W-16	Bell Creek
A-1	De Soto Site, Bldg. 001 Roof
A-2	De Soto Site, Bldg. 004 Roof
A-3	SSFL Site, Bldg. 009, West Side
A-4	SSFL Site, Bldg. 011, West Side
A-5	Rocketdyne Site, Bldg. 600, North Side
A-6	Rocketdyne Site, Bldg. 207, North Side
A-7	SSFL Site, Bldg. 074, South Side
A-8	SSFL Site, Bldg. 143, West Side
A-9	SSFL Site, Bldg. 363, West Side
TLD-1	De Soto Site, South of Bldg. 102
TLD-2	De Soto Site, West Boundary
TLD-3	De Soto Site, Guard Post No. 1, Bldg. 201
TLD-4	De Soto Site, East Fence
TLD-5	De Soto Site, North Boundary
TLD-6	De Soto Site, East Boundary
TLD-7	De Soto Site, South Boundary
TLD-1	SSFL Site, Bldg. 114

S — Soil Sample Station
W — Water Sample Station
A — Air Sampler Station
TLD — Thermoluminescent Dosimeter Location

TABLE 8

SAMPLE STATION LOCATIONS
(Sheet 3 of 3)

Station	Location
TLD-2	SSFL Site, SRE Water Retention Pond
TLD-3	SSFL Site, Electric Substation No. 719
TLD-4	SSFL Site, West Boundary on H Street
TLD-5	SSFL Site, Water Tank No. 701
TLD-6	SSFL Site, Bldg. 854
TLD-1	Off Site, Northridge
TLD-2	Off Site, Simi Valley
TLD-3	Off Site, Northridge

TLD – Thermoluminescent Dosimeter Location

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 the Energy Systems Group facilities is continuously sampled and monitored, to assure that the amounts and concentrations released are within appropriate limits, and to identify those processes which may warrant additional engineering safeguards to minimize the radioactivity levels in such effluents. These measurements provide a direct indication of the effectiveness of engineering controls and allow for remedial action to be taken before a significant release of hazardous material can occur.

Environmental sampling stations that are located within the boundaries of ESG sites are referred to as "on-site" stations; those located within a 10-mile radius of the sites are referred to as "off-site" stations. The on-site environs of the DeSoto and SSFL sites are sampled monthly to determine the concentration of radioactivity in typical surface soil, vegetation, and water. Soil is also sampled on-site semiannually, for plutonium analysis. Similar off-site environmental samples, except for plutonium analysis, are obtained quarterly. Continuous on-site and off-site ambient air sampling provides information concerning long-lived airborne particulate radioactivity. A site ambient radiation monitoring program, utilizing thermoluminescent dosimetry (TLD), begun in 1971, measures radiation levels in the environs of both the DeSoto and SSFL sites.

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

B. SUMMARY RESULTS — 1979

The average radioactivity concentrations in local soil, vegetation, supply water, surface water, and in ambient air for 1979 are presented in Tables 9 through 14. 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 DOE 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 actually 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 the year, show a great increase over the average values beyond natural variability. The ambient air sampling data show no greatly increasing or decreasing trends for the year and can be described as generally constant levels with only very minor transient increases in local airborne radioactivity levels.

The results reported in Tables 9 and 11 show no significant difference between on-site and off-site samples. Table 10 shows no significant variations in soil plutonium concentrations for the 1979 sample sets. The detected activity is due to a variety of naturally occurring radionuclides, and to radioactive fallout resulting from dispersal of nuclear weapons materials and fission products by atmospheric testing. Naturally occurring radionuclides include Be^{7}, K^{40}, Rb^{87}, Sm^{147}, and the uranium and thorium series (including the inert gas radon and its radioactive daughters). Radioactivity from fallout consists primarily of the fission products Sr^{90} - Y^{90}, Cs^{137}, and Pm^{147}, and also U^{235} and Pu^{239}.

Domestic water used at the SSFL site is obtained from Ventura County Water District No. 17, which also supplies nearby communities, and is distributed on site by the same piping system previously used when all facility process water was obtained from on-site wells. Two on-site water wells were operated during 1979 to reduce consumption of Ventura County domestic water. The well water proportion in the blend averaged about 69% for the year for a total well water consumption of approximately 7.9×10^{7} gal. Pressure for the water system is provided by elevated storage tanks.

TABLE 9

SOIL GROSS RADIOACTIVITY DATA — 1979

Area	Activity	No. Samples	Gross Radioactivity (μCi/g)	
			Annual Average Value (95% Confidence Level)	Maximum Observed Value*
On Site	α	144	$(6.4 \pm 1.5)\ 10^{-7}$	10.9×10^{-7}
	β	144	$(2.5 \pm 0.1)\ 10^{-5}$	9.7×10^{-5}
Off Site	α	48	$(5.0 \pm 1.4)\ 10^{-7}$	8.1×10^{-7}
	β	48	$(2.3 \pm 0.1)\ 10^{-5}$	2.9×10^{-5}

*Maximum value observed for single sample

TABLE 10

SOIL PLUTONIUM RADIOACTIVITY DATA — 1979

Sample Location	June 23, 1979 Survey Results		December 17, 1979 Survey Results	
	Pu^{238} (μCi/g)	$Pu^{239}+Pu^{240}$ (μCi/g)	Pu^{238} (μCi/g)	$Pu^{239}+Pu^{240}$ (μCi/g)
S-56	$(1.4 \pm 2.7)10^{-9}$	$(9.9 \pm 3.9)\ 10^{-9}$	$(-0.6 \pm 2.2)\ 10^{-9}$	$(6.0 \pm 3.5)\ 10^{-9}$
S-57	$(1.5 \pm 3.0)10^{-9}$	$(2.5 \pm 2.4)\ 10^{-9}$	$(-1.9 \pm 2.0)\ 10^{-9}$	$(4.5 \pm 3.0)\ 10^{-9}$
S-58	$(-1.1 \pm 2.1)10^{-9}$	$(2.2 \pm 2.2)\ 10^{-9}$	$(3.3 \pm 5.3)\ 10^{-9}$	$(18.9 \pm 8.7)\ 10^{-9}$
S-59	$(-2.0 \pm 1.9)10^{-9}$	$(4.2 \pm 3.1)\ 10^{-9}$	$(2.3 \pm 4.5)\ 10^{-9}$	$(18.6 \pm 8.0)\ 10^{-9}$
S-60	$(-2.8 \pm 1.9)10^{-9}$	$(0.3 \pm 1.6)\ 10^{-9}$	$(0.2 \pm 2.9)\ 10^{-9}$	$(3.3 \pm 3.1)\ 10^{-9}$

Note: Minus (-) indicates sample value less than reagent blank.

TABLE 11

VEGETATION RADIOACTIVITY DATA — 1979

Area	Activity	No. Samples	Gross Radioactivity (μCi/g)		
			Dry Weight Annual Average Value	Ash	
				Annual Average Value (95% Confidence Level)	Maximum Observed Value*
On Site	α	144	$(<5.2 \pm 3.5)\ 10^{-8}$	$(<2.4 \pm 1.6)\ 10^{-7}$	1.4×10^{-6}
	β	144	$(2.6 \pm 0.04)\ 10^{-5}$	$(1.39 \pm 0.02)\ 10^{-4}$	2.48×10^{-4}
Off Site	α	48	$(<6.3 \pm 4.4)\ 10^{-8}$	$(<2.3 \pm 1.6)\ 10^{-7}$	8.6×10^{-7}
	β	48	$(3.0 \pm 0.04)\ 10^{-5}$	$(1.34 \pm 0.02)\ 10^{-4}$	2.30×10^{-4}

*Maximum value observed for single sample

Water from the system is sampled monthly at two widely separated SSFL site locations. The average domestic water radioactivity concentration is presented in Table 12.

TABLE 12

SSFL SITE — DOMESTIC WATER RADIOACTIVITY DATA — 1979

Area	Activity	No. Samples	Gross Radioactivity (μCi/mℓ)	
			Average Value (95% Confidence Level)	Maximum* Observed Value
ESG-SSFL	α	24	$(<2.3 \pm 2.7)\ 10^{-10}$	$<2.3 \times 10^{-10}$
	β	24	$(2.8 \pm 0.7)\ 10^{-9}$	3.9×10^{-9}

*Maximum value observed for single sample

Surface waters discharged from SSFL 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. 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 and Bell Creek samples are presented in Table 13.

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

The SSFL site surface water and the ambient air radioactivity concentration Guide values selected for each site are the most restrictive limits for those radionuclides currently in use at ESG facilities. The identity of all such radionuclides is known, irrespective of concentration. Accordingly, for SSFL site surface water, the Guide value of 5×10^{-6} $\mu Ci/m\ell\alpha$ and 3×10^{-7} $\mu Ci/m\ell\beta$, for Pu^{239} and for Sr^{90}, respectively, is appropriate. The correspondingly most restrictive Guide value for DeSoto site wastewater radioactivity discharged to the sanitary sewage system, a controlled area, is 8×10^{-4} $\mu Ci/m\ell\alpha$ and 1×10^{-3} $\mu Ci/m\ell\beta$ for U^{235} and Co^{60}, respectively. These values are established in 10 CFR 20, California Administrative Code Title 17, and DOE Manual Chapter 0524.

The Guide value of 6×10^{-14} $\mu Ci/m\ell\alpha$ for SSFL site ambient air radioactivity is due to work with unencapsulated plutonium. The Guide value of 3×10^{-11} $\mu Ci/m\ell\beta$ for Sr^{90} is due to the presence of fission products in irradiated nuclear fuel at the SSFL site. The Guide value of 3×10^{-12} $\mu Ci/m\ell\alpha$ for De Soto site ambient air radioactivity is due to work with unencapsulated uranium (including

TABLE 13

BELL CREEK AND ROCKETDYNE SITE RETENTION POND
RADIOACTIVITY DATA – 1979

Area	Activity	No. Samples	Gross Radioactivity Concentration		
			Average Value (95% Confidence Level)	Maximum* Observed Value	% of Guide†
Bell Creek Mud No. 54 (μCi/g)	α	12	$(4.6 \pm 1.3)\ 10^{-7}$	6.2×10^{-7}	NA
	β	12	$(2.3 \pm 0.1)\ 10^{-5}$	2.7×10^{-5}	NA
Pond R-2A Mud No. 55 (μCi/g)	α	12	$(7.1 \pm 1.6)\ 10^{-7}$	1.1×10^{-6}	NA
	β	12	$(2.5 \pm 0.1)\ 10^{-5}$	3.3×10^{-5}	NA
Bell Creek Vegetation No. 54 (μCi/g ash)	α	12	$(<2.6 \pm 1.7)\ 10^{-7}$	7.6×10^{-7}	NA
	β	12	$(1.36 \pm 0.02)\ 10^{-4}$	2.20×10^{-4}	NA
Bell Creek Vegetation No. 54 (μCi/g) dry weight)	α	12	$(<7.5 \pm 4.8)\ 10^{-8}$	3.3×10^{-7}	NA
	β	12	$(3.0 \pm 0.1)\ 10^{-5}$	7.6×10^{-5}	NA
Bell Creek Water No. 16 (μCi/mℓ)	α	12	$(<2.3 \pm 2.7)\ 10^{-10}$	2.4×10^{-10}	<0.005
	β	12	$(3.2 \pm 0.9)\ 10^{-9}$	8.2×10^{-9}	1.1
Pond Water No. 6 (μCi/mℓ)	α	12	$(<2.5 \pm 2.8)\ 10^{-10}$	5.5×10^{-10}	<0.005
	β	12	$(3.1 \pm 0.8)\ 10^{-9}$	4.7×10^{-9}	1.0
SSFL Pond R-2A Water No. 12 (μCi/mℓ)	α	12	$(<2.3 \pm 2.7)\ 10^{-10}$	2.5×10^{-10}	<0.005
	β	12	$(4.5 \pm 0.8)\ 10^{-9}$	1.0×10^{-8}	1.5

*Maximum value observed for single sample.
†Guide: 5×10^{-6} μCi/mℓα, 3×10^{-7} μCi/mℓβ; 10 CFR 20 Appendix B, CAC 17, DOE Manual Chapter 0524.
NA – not applicable, no Guide value having been established.

depleted uranium). The Guide value of 3×10^{-10} μCi/mℓ for Co^{60} for ambient air radioactivity is appropriate since it is the most restrictive limit for beta-emitting radionuclides present at the DeSoto site. Guide value percentages are not presented for soil or vegetation data since no concentration Guide values have been established.

Ambient air sampling for long-lived particulate alpha and beta radioactivity is performed continuously with automatic sequential samplers at both the DeSoto and SSFL sites. Air is drawn through Type HV-70 filter media which are analyzed for long-lived radioactivity, after a minimum 120-h decay period that eliminates the naturally occurring short lived particulate radioactivity. The average concentrations of ambient air alpha and beta radioactivity are presented separately in Table 14.

Radioactivity levels observed in environmental samples for 1979, reported in Tables 9 through 14, compare closely with levels reported for recent years. Local environmental radioactivity levels, which result primarily from beta-emitting radionuclides and had shown the effect of fallout during past extensive atmospheric testing of nuclear devices, have decreased, and have been generally constant during the past several years. The effects of foreign atmospheric nuclear tests continue to be occasionally observable in daily ambient radioactivity levels, although this effect was not readily discernible during 1979. The long-term effects of airborne radioactivity on surface sample radioactivity levels are also not discernible in recent years. The continuing relative constancy in environmental radioactivity levels is due primarily to the dominance of naturally occurring radionuclides in the environment and to the longer-life fission product radioactivity from fallout.

Site ambient radiation monitoring is performed with thermoluminescent dosimeters. Each dosimeter set contains two calcium fluoride (CaF_2:Mn) low background, bulb-type chip dosimeters. The dosimeter sets are placed at selected locations (Figures 2 and 3) on or near the perimeters of the DeSoto and SSFL sites. Each dosimeter, sealed in a light-proof energy compensation shield, is

TABLE 14

AMBIENT AIR RADIOACTIVITY DATA — 1979

Site Location	Activity	No. Samples	Average Value (95% Confidence Level)	Maximum* Observed Value (daily)	% of Guide†
De Soto On Site (μCi/mℓ)	α§	702	$(<6.6 \pm 7.8)\ 10^{-15}$	4.5×10^{-14}	<0.28
	β**	702	$(<2.1 \pm 1.3)\ 10^{-14}$	1.0×10^{-13}	<0.030
SSFL On Site (μCi/mℓ)	α§	1793	$(<6.5 \pm 7.6)\ 10^{-15}$	4.0×10^{-14}	<12.0
	β**	1793	$(<2.1 \pm 1.3)\ 10^{-14}$	1.1×10^{-13}	<0.29
SSFL Sewage Treatment Plant Off Site (μCi/mℓ)	α§	362	$(<6.2 \pm 7.4)\ 10^{-15}$	2.0×10^{-14}	<12.2
	β**		$(<2.0 \pm 1.3)\ 10^{-14}$	1.1×10^{-13}	<0.28
SSFL Control Center Off Site (μCi/mℓ)	α§	364	$(<6.2 \pm 8.4)\ 10^{-15}$	3.4×10^{-14}	<11.8
	β**		$(<1.8 \pm 1.5)\ 10^{-14}$	1.0×10^{-13}	<0.30

*Maximum value observed for single sample.

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

§MDL = 6.1×10^{-15} μCi/mℓ — Individual daily samples with activity levels of 0 to 6.1×10^{-15} μCi/mℓ are recorded and averaged as 6.1×10^{-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.

installed in a polyethylene container which is mounted ~1 meter above ground at each location. The dosimeters are exchanged and evaluated quarterly. There were 13 on-site TLD monitoring locations used during the year. Three additional dosimeter sets, placed at locations up to 10 miles from the ESG sites, are similarly evaluated to determine the local area off-site ambient radiation level, which averaged 0.015 mRem/h for 1979. The average radiation dose rate and equivalent annual dose monitored at each dosimeter location are presented in Table 15.

TABLE 15

DE SOTO AND SSFL SITES – AMBIENT RADIATION DOSIMETRY DATA – 1979

TLD Location*	Average Dose Rate (mRem/h)	Equivalent Annual Dose (mRem)
1 De Soto	0.016	140
2 De Soto	0.015	131
3 De Soto	0.014	123
4 De Soto	0.016	140
5 De Soto	0.016	140
6 De Soto	0.017†	149
7 De Soto	0.016§	140
1 SSFL	0.017	149
2 SSFL	0.018	158
3 SSFL	0.022	193
4 SSFL	0.021	184
5 SSFL	0.016	140
6 SSFL	0.016	140
1 Off-Site Control	0.014	123
2 Off-Site Control	0.016	140
3 Off-Site Control	0.015	131

*See Table 8 for specific location.
†Excludes first quarter data due to missing dosimeter.
§Excludes second quarter data due to missing dosimeter.

The table shows that radiation dose rates and equivalent annual doses monitored on site are essentially identical to those monitored at three widely separated off-site locations. These data include the natural background radiation component which exists as a consequence of cosmic radiation, radionuclides in the soil, and radon and thoron in the atmosphere, in addition to radioactive fallout from nuclear weapons tests. Locally, this total is approximately 125 mrem/year. The small variability observed in the data is attributed to differences in elevation and geologic conditions at the various dosimeter locations. Since the data for the on-site and off-site locations are nearly identical, no measurable radiation dose to the general population or to individuals in uncontrolled areas resulted from ESG operations.

1. Ambient Air

Air sampling is performed continuously at the De Soto and SSFL sites with automatic air samplers, operating on 24-h sampling cycles. Airborne particulate radioactivity is collected on Type HV-70 filter media, which are automatically changed daily at the end of each sampling period. The samples are counted for alpha and beta radiation following a minimum 120-h decay period. The volume of a typical daily ambient air sample is approximately 25 m^3.

Figure 5 is a graph of the daily averaged long-lived alpha and beta ambient air particulate radioactivity concentrations for the De Soto and SSFL sites during 1979. The average beta concentration for each month also is indicated by horizontal bars. The graph shows that no prominent peaks occurred during the year, and that radioactivity concentrations were essentially constant.

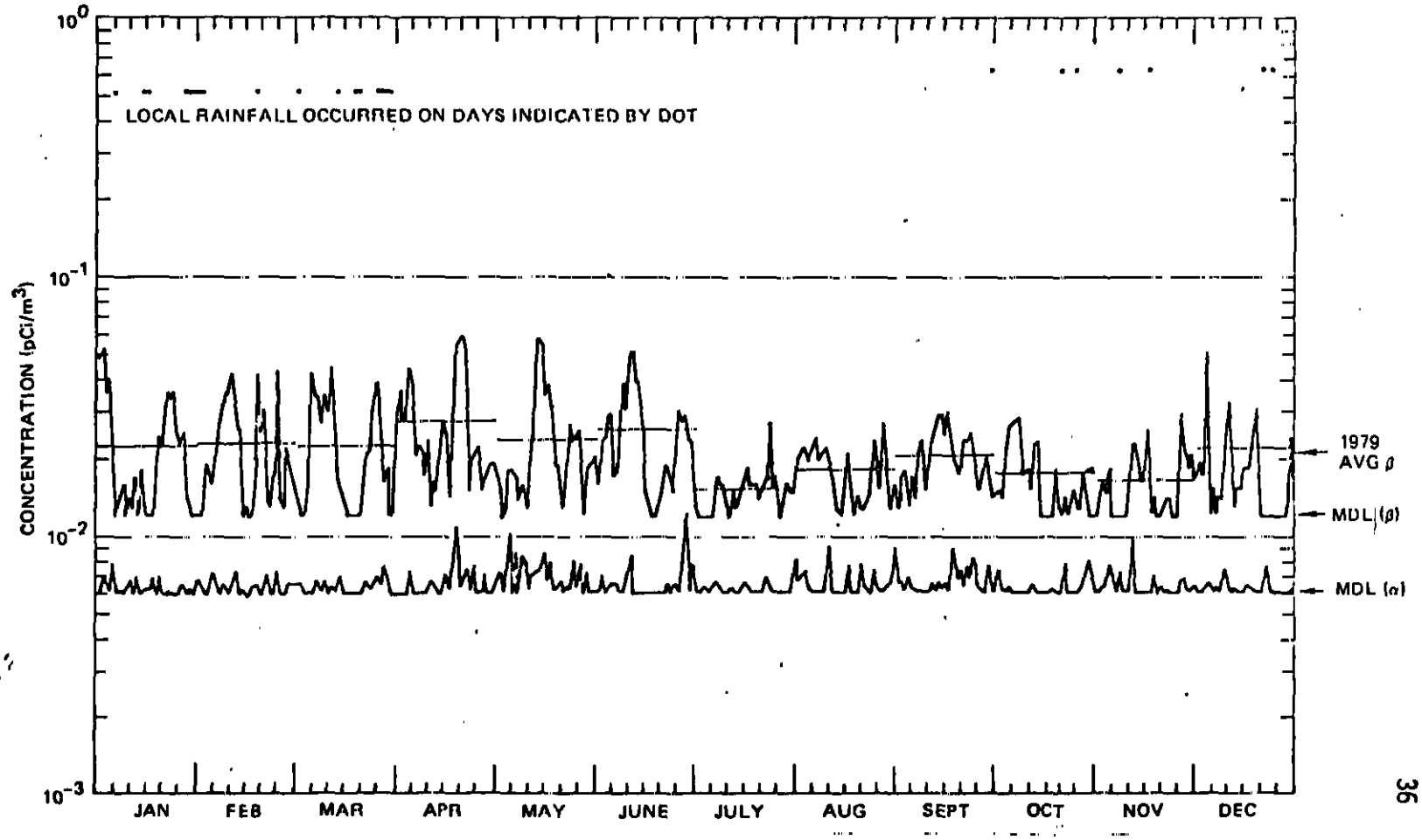

Figure 5. Daily Averaged Long-Lived Airborne Radioactivity at Headquarters and SSFL -- 1978

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V. UNUSUAL EVENTS

Building 055

At 1300 on June 26, 1979, some equipment was being removed from a glove box by three men, including a representative of Health and Safety. Full precautions (protective clothing, respirators, etc) were in force. Unnoticed for a moment, a small piece of highly contaminated insulation material was caught in the flange seat and was protruding outside the containment. The air monitor alarmed and the three workers immediately evacuated the room. After a short rest and discussion of the recovery procedures, four men reentered the area with full protective clothing including full face masks. Surface contamination of the order of 75 dpm/100 cm^2 was found in the vicinity of the mishap. The affected areas were decontaminated. All personnel in the room during the release submitted nasal smears which showed no detectable activity. Based on subsequent surveys, an area of $\sim 10^5$ cm^2 was contaminated to an average value of 75 dpm/100 cm^2. Momentarily, at least, the air concentration could have reached $\sim 2 \times 10^4 \times$ MPC if all the activity was assumed contained in 1 m^3 of air.

Based on nasal smears taken immediately after the release, it was deemed unlikely that any significant internal exposures occurred. However, fecal and urine samples were taken from involved personnel.[4] No significant activity was detected by these analyses.

Building 001

On August 1, an employee received an estimated 50% of the allowable weekly exposure (airborne) when a glove became unsealed from its port on a glove box containing enriched uranium. Both bioassay and a lung scan gave negative results. His lapel air monitor showed $3.6 \times 10^{-9} \frac{\mu Ci\text{-}hr}{cc}$ for the week.[5]

On September 27, the crusher glove box operator received a cut on his hand which required stitching. However, wound monitoring for radioactivity proved negative.[5]

Building 020

None

Building 004

None

VI. SUMMARY/TRENDS – EXPOSURE, EFFLUENTS

A comparison of the 1979 data with that of previous years shows:

1) The previous (1975-78) upward trend in the occupational population dose was reversed as shown below:

Year	Population Dose
1975	27 man-rem
1976	48 man-rem
1977	67 man-rem
1978	110 man-rem
1979	91 man-rem

Although this decrease (~17%) is not large and probably reflects somewhat lessened overall activity, it does substantiate that the controls provided continue to be effective.

2) The annual average personnel exposures over the last 5 years were:

Year	Population Dose
1975	110 mrem
1976	220 mrem
1977	310 mrem
1978	200 mrem
1979	54 mrem

Again, the 1979 value represents a significant drop from the previous four years and is just over 1% of the occupational limit of 5 rem/year.

3) Engineered and operational controls continue to effectively limit releases of radioactive materials to the environment. In

substantiation of this conclusion, a summary of annual atmospherically discharged effluents and liquid effluent radioactivity for 1975-1979 is presented below:

Atmospherically Discharged Effluent Radioactivity (curies)

1979	$<2.2 \times 10^{-5}$	α
	$<5.8 \times 10^{-5}$	β,γ
1978	$<1.7 \times 10^{-5}$	α
	$<6.8 \times 10^{-5}$	β,γ
1977	1.1×10^{-5}	α
	2.5×10^{-5}	β,γ
1976	7.3×10^{-5}	α
	3.3×10^{-5}	β,γ
1975	9.4×10^{-5}	α
	6.7×10^{-3}	β,γ

Similarily, the radioactive contents of the liquid effluents were:

1979	$<9.9 \times 10^{-5}$	α	curies
	$<2.1 \times 10^{-4}$	β,γ	curies
1978	$<1.0 \times 10^{-4}$	α	curies
	3.3×10^{-4}	β,γ	curies
1977	1.1×10^{-5}	α	curies
	2.5×10^{-5}	β,γ	curies

1976	$<1.6 \times 10^{-4}$	α curies
	$<3.4 \times 10^{-4}$	β,γ curies
1975	6.9×10^{-5}	α curies
	3.3×10^{-4}	β,γ curies

From the above listings, it can be seen that the radioactivity content of both liquid and atmospherically discharges has remained relatively constant over the period 1975-1979, amounting to a few, or at most a few hundreds of microcuries per year. Although it is difficult to establish trends at such low levels, it would appear that there has been an overall decrease in both personnel exposures, as well as in the radioactivity discharged in effluent streams. Some of this decrease can no doubt be attributed to a reduction in potentially contaminating activities and materials, but in any case, the controls provided appear to be effective and adequate.

VII. ANTICIPATED ACTIVITIES DURING NEXT REPORTING PERIOD (1980)

Building 001/004

Removal of L-77 reactor core and shell (fuel was removed previously).

Continued fabrication of test reactor fuel elements.

Building 020

Continued cleanup of the cell from HNPF fuel decladding operations. Initiate EBR-1 fuel (Pu) declad operations.

Building 055

The major operation to be conducted at Bldg. 055 during the next year will be the fabrication of depleted uranium carbide blanket elements for FFTF.

REFERENCES

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

2. "Annual Review of Radiological Controls — 1978," R. S. Hart, Energy Systems Group, Rockwell International, N001TI000104, August 6, 1980

3. "Energy System Group Environmental Monitoring and Facility Effluent Annual Report — 1979," J. D. Moore, Rockwell International, Energy Systems Group, ESG 80-7,

4. IL, "Radiological Safety Incident Report," Radiation and Nuclear Safety Group from F. W. Gardner, Rockwell International, August 23, 1979

5. Personal communication, Joe Wallace, Rockwell International, August 21, 1980

APPENDIX

A. PERSONNEL MONITORING PROGRAM

Film badges are furnished by a vendor service, the R. S. Landauer, Jr. & Co.* A description of the badge components and their functions follow:

1) Type of film — Eastman Kodak film stock of type 2 and type NTA specially packaged by R. S. Landauer, Jr. & Co.

Exposure Range

a) Gamma and x-ray (>100 keV) — 10 mr — 500R ± 10% or 20 mr, whichever is greater.

b) Gamma and x-ray (<100 keV) — 10 mr — 60R ± 20% or 30 mr, whichever is greater.

c) Beta (>1.5 meV — 40 mr — 100R ± 20% or 30 mr, whichever is greater. (A beta-energy corrected exposure can be reported for energies down to 0.5 meV as a service option).

d) Neutron (1 meV — 14 meV) — 20 mrem — 25 rem ± 30% or 30 mr, whichever is greater. (Where exposure to ionizing radiation is small, calibrations are based on AmBe spectra unless otherwise designated).

e) Thermal neutron — 10 mrem — 20 rem ± 20% or 30 mrem, whichever is greater.

2) Filters and Specifications

*This represents a change in vendors. Previously, the film badges were supplied by the Radiation Detection Company.

a) Holder design and dimension — A plastic holder measuring approximately 2-1/4 in. by 3/4 in. by 3/8 in. with an attached fiberglass filled nylon clip is supplied. This holder contains patented "U" absorbers consistent with the radiation requirements selected by the user.

b) Absorber description — The following absorbers are included in all holders or are optionally used depending on badge type requirements or optionally used depending on service options selected by the user.

Filter Position	Mass (mg/cm^2)
Open Window	25 (wrapper and label)
Plastic-I	100 (includes 1. above)
Plastic-II	175 (includes 1. above)
Plastic-III	325 (includes 1. above)
Aluminum	375 (includes 75/mg/cm^2 plastic and 1. above)
Lead (60%) + tin (40%)	1660 (includes 140 mg/cm^2 plastic and 1. above)
Lead (60%) + cadmium (40%)	1660 (includes 140 mg/cm^2 plastic and 1. above)

3) Sources which film are calibrated to:

a) Cs^{137}
b) Sr^{90} and Uranium
c) X-rays (18 kVE — 140 kVE)

4) Criticality Function

Indium Foil — 1/2 in. by 5/8 in. by 0.015 in. thick; primarily a thermal neutron detector used as an indicator in case of a criticality event. Plastic in front of this foil is 0.040 in. thick. By using a thin window radiation survey meter, an estimate

of the magnitude of exposure can be made. Although the foil may reach saturation, highly exposed individuals can be separated from those less likely to have had exposure. The half life is approximately 54 minutes – 320 mg; 99.97% purity.

Gold Foil – There are two pieces of gold foil used; one is covered with cadmium and the other is bare. Each piece is 3/6 in. by 5/8 in. by 0.005 in. thick. The cadmium cover is 0.015 in. thick. The gold interacts with a wide range of neutron energies – 200 mg; 99.95% purity.

The cadmium cover over the gold eliminates almost all neutrons below the cut-off at approximately 0.4 eV. The difference between activation of the bare gold and the cadmium covered gold is used in the determination of the magnitude and the ratio of the neutron distribution above and below 0.4 eV. Detection is provided for less than 100 rem,to several magnitudes higher than that. The approximate half life is 2.7 days.

Sulphur – Pellet form; 1-2 in. in diameter and 1/8 in. thick, has a threshold of 2.8 meV. Determination of the activity can be made by direct measurement, if of a sufficiently high activity; or by charring the sulphur and measuring the remaining phosphorous-32. The sulphur provides a measure of the fast neutron dosage of 2.8 meV – 450 mg, 99+% purity.

Options

Lithium Fluoride – One or two LiF rods are optionally provided for high range gamma dosimetry. The rods used are Harshaw extruded dosimeters 0.5 mm in diameter and 6 mm long. If two are used, one is placed in a brass sleeve having a wall thickness of 0.20 in., thus allowing for some gamma energy determination.

Glass Rods — Two glass rods are provided at extra cost. One is bare and the other is shielded using a brass sleeve with a wall thickness of 0.020 in. The rods used are Bausch and Lomb Low "Z" microdosimeters. They are of silver activated phosphate glass, 6 millimeters long and 1 millimeter in diameter.

Several different types of detector systems can be used for the determination of the activity of the gold and sulphur, depending on the magnitude of the activity. One can utilize a shielded end window Geiger counter, an internal gas flow counter, or any one of a number of scintillation detector systems. Because of the possibility of activation of small amounts of foreign material, it is advisable to use a single (or multi) channel analyzer. Some sort of provision for cleaning the gold may be advisable also.

All personal film badges are processed routinely by the AI film badge vendor (R. S. Landauer, Jr. & Co.) according to the methods described above.

Certain operations, such as hot cell entries, which may 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.

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, R. S. Landauer also sends an individual ERDA Form-5 on each person on the film badge roster with a summary of the above information.

B. 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 electro-deposited 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 A 1

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.

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TABLE 3.1 (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
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.