F. H. Badger - T020

Rockwell International
Energy Systems Group

SUPPORTING DOCUMENT

GO NO.	S/A NO.	PAGE 1 OF	TOTAL PAGES	REV LTR/CHG NO SEE SUMMARY OF CHG	NUMBER
94132	-	52	52	N/C	N001T1000104

PROGRAM TITLE

Health, Safety, and Radiation Services

DOCUMENT TITLE

Annual Review of Radiological Controls - 1978

DOCUMENT TYPE

Technical Information

KEY NOUNS

Radiation Exposures, Effluents, NRC Licensed Facilities

ORIGINAL ISSUE DATE	REL. DATE
	8-6-80 CW

APPROVALS

	DATE
E. R. Specht ER Specht	7/30/80
R. J. Tuttle RJ Tuttle	7/25/80

PREPARED BY/DATE	DEPT	MAIL ADDR
R. S. Hart	731	JB05

IR&D PROGRAM? YES ☐ NO ☒
IF YES, ENTER TPA NO.

DISTRIBUTION

*	NAME	MAIL ADDR
*	D. J. Aubuchon	KA47
*	E. Baumeister	JB10
*	S. Berger	JB05
*	K. Buttrey	LB15
*	R. Hart(2)	JB05
*	R. Hartzler	JB02
*	V. Keshishian	LB34
*	W. Kittinger	NB02
*	W. McCurnin	T020
*	L. Mountford	LB01
*	C. Nealy	NB06
*	M. Remley	NB08
*	O. Roberson	KA02
*	E. Specht	JB05
*	R. Tuttle(2)	NB13
*	J. Walter	T006
*	Radiation and Nuclear Safety Group (11)	NB13

731-M.262/vlg

* COMPLETE DOCUMENT
NO ASTERISK, TITLE PAGE/SUMMARY OF CHANGE PAGE ONLY

ABSTRACT

Data on exposures of Energy System Group personnel at NRC-licensed facilities to radiation and/or radioactivity are presented for CY 1978. 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 effluents can be further reduced under the ALARA concept, and (3) the equipment for effluent and personnel exposure control is performing properly. As for 1978, it is concluded that although the overall personnel exposure continued to increase (110 man-Rem versus 67 man-Rem in 1977, 48 man-Rem in 1976, and 27 man-Rem in 1975) the average exposure was 200 mRem. As in recent years, the majority of the higher exposures were associated with D&D activities.

This report satisfies License Condition 23 of Special Nuclear Materials License SNM-21 for 1978.

RESERVED FOR PROPRIETARY/LEGAL NOTICES

FORM 734-C REV. 12-79

CONTENTS

TABLES

CONTENTS

INTRODUCTION

As set forth in the Energy Systems Group's special material license[1] as Condition 23: "A formal annual report shall be made to the radioisotope review committee of the Nuclear Safety 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 operations or effluent releases, (2) if exposures and effluents might be lowered under the concept of 'as low as reasonably achievable,' and (3) if equipment for effluent and exposure control is being properly used, maintained, and inspected. This report shall include review of other required audits and inspections performed during the past twelve months and review of the data from the following areas: Employee exposures; bioassay results; effluent releases; in-plant airborne radioactivity and environmental monitoring."

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

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

1) Fuel Fabrications - Building 001 and supporting operations in Buildings 001 and 004, De Soto Facility, Canoga Park, California
2) Rockwell International Hot Laboratory - Building 002, Santa Susana Field Laboratories, Santa Susana, California
3) Nuclear Material Development Facility - Building 055, Santa Susana Field Laboratories, Santa Susana, California

*The initial report in this series was for the year 1975[4]. The most recent previous report was for 1977[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 irradiated 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 are presented in Table 1 where the number of individuals within specific annual dose ranges are given. The data for those persons with measured doses greater than zero are plotted in Figure 1 using a log-probability scale. Data which have an essentially log-normal distribution will appear as a straight line on such a scale. This shows that the median dose for personnel exposed to radiation is well below 100 mrem, about 20 mrem.

TABLE 1

PERSONNEL EXTERNAL EXPOSURES - 1978

Dose Range (Rem)	P (People)	CP (Cumulative People)	CP (%)	Pop-Dose* (Man-Rem)
No Measurable	1377	1377	71.1	0
<0.100	432	1089	93.4	21.6
0.1-0.25	60	1869	96.5	10.5
0.25-0.50	18	1887	97.5	6.75
0.50-0.75	16	1903	98.3	10.0
0.75-1.0	4	1907	98.5	3.5
1.0-2.0	18	1925	99.4	27
2.0-3.0	9	1934	99.90	22.5
3.0-4.0	1	1935	99.95	3.5
4.0-5.0	1	1936	100.00	4.5
>5.0	0	1936	-	-
				109.85

*The midpoint of each dose range was assumed for the average dose in calculating the population dose (Man-Rem) for that range.

731-M.262/vlg

Percentage of Personnel With Annual
Dose ≤ to Ordinate Value

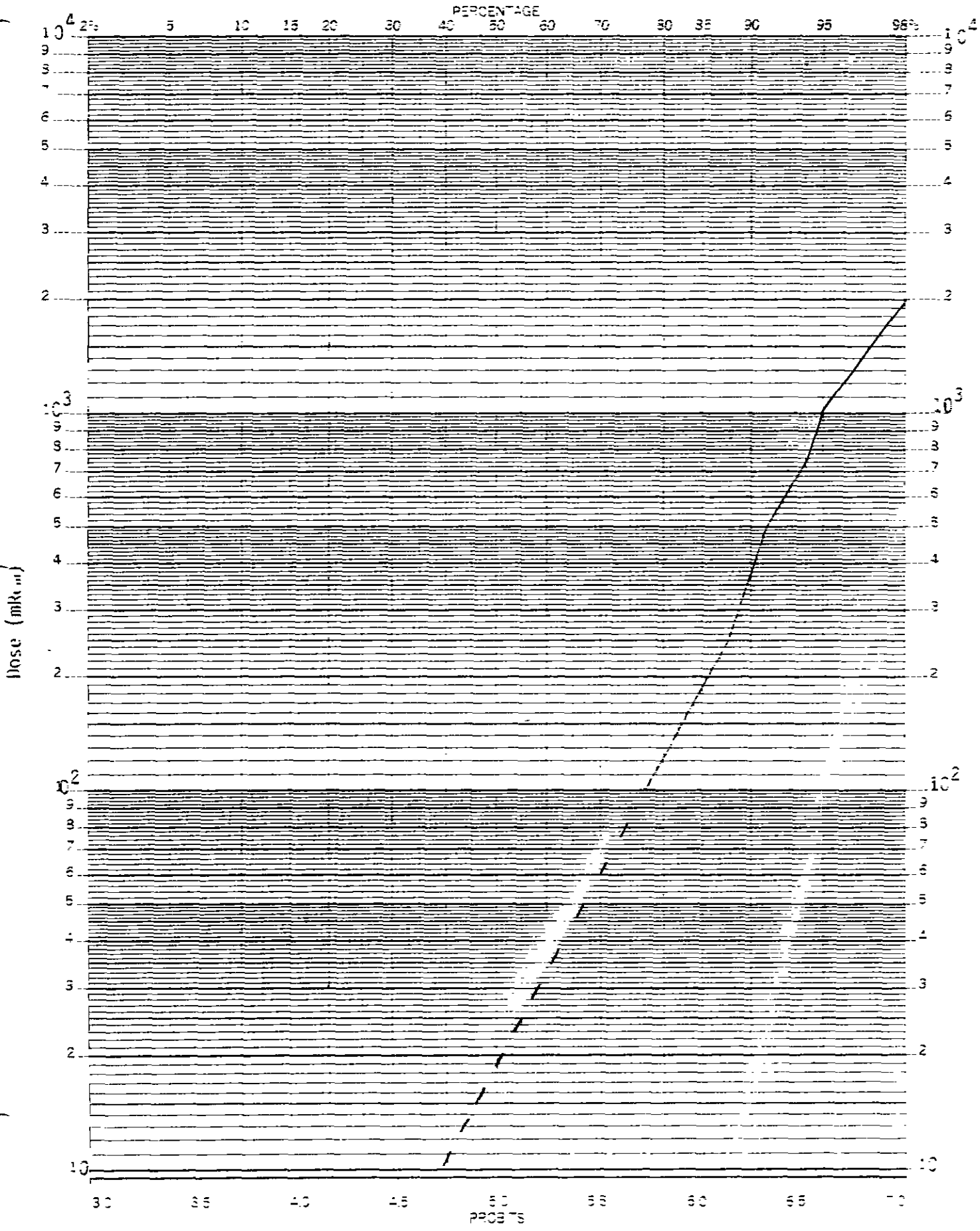

Figure 1

It can also be noted that all exposures are below the annual occupational limit of 5 Rem, with the average exposure (of personnel with non-zero doses) being about 197 mrem or about 4% of this limit. Compared to the previous year (1977) data, the total population dose increased from 67 to 110 man-rem.

B. IN-VIVO LUNG SCANS

Measurements are periodically made of the total radioactive lung burden for specific radioisotopes of those employees who have been potentially exposed to radioactive aerosols in the respirably-sized particle range. These measurements are accomplished through the use of a whole body counter.* During 1978, forty-eight lung scans were made for uranium deposition. Eleven of the scans (on ten different individuals) showed some degree of positive results. However, followup scans showed a steady decrease of these lung burdens in all cases (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 1978 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 1978 also appear in these tables.

*Helgeson Nuclear Services, Inc., Pleasanton, California

TABLE 2A

SUMMARY OF BIOASSAYS - 1978

Measurement*	Type*	Total Tests	Total Positive Results	Total Individuals With Positive Results**
U	UF	205	9	8
U	UR	201	0	0
GA	1A	1	0	0
GB	1B	2	0	0
GB	2B	0	-	-
GB	H	2	0	0
Pu	A	19	0	0
FP	1	0	-	-
FP	3A	271	45	25
FP	3B	271	22	14
U	IVLC	48	11	10
Sr-90	-	2	2	2
Cs-137	TBC	0	-	-

*UF - Uranium-Fluorometric
UR - Uranium-Radiometric
GA - Gross Alpha
GB - Gross Beta
Pu - Gross Plutonium
FP - Fission Products
U-IVLC - Uranium - In-Vivo Lung Count
TBC - Total Body Count

**Tests are sometimes repeated when positive results are obtained

Note: See Appendix for description of various bioassay techniques

731-M.262/vlg

TABLE 23

POSITIVE BIOASSAY RESULTS SUMMARY - 1978

H&S No.	Sample Date	Analysis Type*	Results Per Volume Analysis (ugm)	Results Per 1500 ml/day (ugm)	Specific Radionuclide	Equivalent MPBB** (%)
4215	08/14/78	UF	0.0005	0.75	U	0.75
4215	09/10/78	UF	0.0001	<MDL	U	-
4014	08/28/78	UF	0.0004	0.60	U	0.60
2040	01/11/78	IVLC	84		U-235	20.5 MPLB$^{\Psi}$
2040	05/03/78	IVLC	0		U-235	-
2040	09/07/78	IVLC	70		U-235	-
2040	01/18/79	IVLC	0		U-235	-
4415	09/20/78	UF	0.0014	2.1	U	2.1
1643	01/11/78	IVLC	73		U-235	17.8 MPLB
1643	05/03/78	IVLC	0		U-235	-
1863	09/07/78	IVLC	34		U-235	13.7 MPLB
1863	01/18/79	IVLC	0		U-235	-
4154	01/11/78	IVLC	84		U-235	20.5
4154	05/03/78	IVLC	0		U-235	-
1816	03/06/78	UF	0.0005	0.75	U	0.75
1816	04/04/78	UF	0		U	-
3986	05/03/78	IVLC	78		U-235	17.8 MPLB
3986	09/07/78	IVLC	0		U-235	-
4206	01/11/78	IVLC	65		U-235	15.8 MPLB
4206	05/03/78	IVLC	0		U-235	-
4345	05/03/78	IVLC	31		U-235	7.5 MPLB
4345	09/07/78	IVLC	0		U-235	-
4125	09/07/78	IVLC	36		U-235	14.5 MPLB
4414	12/05/78	UF	0.0007	1.05	U	1.1
4414	12/19/78	UF	0.0001	<MDL	U	-

TABLE 2B

POSITIVE BIOASSAY RESULTS SUMMARY - 1978
(Continued)

H&S No.	Sample Date	Analysis Type*	Results Per Volume Analysis (ugm)	Results Per 1500 ml/day (ugm)	Specific Radionuclide	Equivalent MPBB** (%)
4508	11/19/78	UF	0.0003	0.45	U	0.45
4508	12/15/78	UF	0	-	U	-
4356	08/20/78	UF	0.0004	0.60	U	0.60
4356	09/07/80	UF	0.0001	<MDL	U	-
4136	05/04/78	UF	0.0005	0.75	U	0.75
4136	06/27/78	UF	0	-	U	-
4208	05/03/78	IVLC	41		U-235	10.0
4208	09/07/78	IVLC	0		U-235	-
0812	03/06/78	UF	0.0003	0.45	U	0.45
0812	04/03/78	UF	0	-	U	-
1547	11/28/78	UF	0.0003	0.45	U	0.45
1547	12/17/78	UF	0	-	U	-
4210	01/11/78	IVLC	80		U-235	19.5
4210	05/03/78	IVLC	0		U-235	-

*IVLC - In-Vivo Lung Count
UF - Uranium-Fluorometric

**MPBB - Maximum Permissable Body Burden
MDL - Minimum Detectable Level

ΨMPLB - Maximum Permissable Lung Burden

731-M.262/vlg

II. RADIATION/RADIOACTIVITY MEASUREMENTS

A. AREA RADIATION LEVELS

To roughly characterize the general external levels of penetration radiation which existed at each facility during 1978, 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 believed to be approximately correct, somewhat higher levels possibly could have existed for limited periods in certain locations.

TABLE 3
RADIATION LEVELS - WORKING AREAS - 1978

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	0.1	0.2	Uncontrolled area
	0.8	5.0	Uncontrolled area
055	<1	~20	

*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 respirable-sized radioactive aerosols, periodic air sampling is performed. A summary of these results for 1978 is given in Table 4.

TABLE 4

INTERIOR AIR SAMPLE SUMMARY - 1978

Building/Area	Maximum μCi/cc	Average* μCi/cc	Remarks**
001-Fuel Fab	1×10^{-9}	5×10^{-10}	α
004	Not Sampled	Not Sampled	
020 Controlled Areas	1×10^{-9}	6×10^{-11}	$\beta\gamma$
Uncontrolled Areas	1.5×10^{-12}	1×10^{-13}	$\beta\gamma$
055	9×10^{-12}	6×10^{-13}	α

*Estimated

**Because of the nature of the material in use, $\beta\ \gamma$ activities are not normally measured in Buildings 001 and 055, and α activities are not normally measured in Building 020.

C. SPECIAL AIR SAMPLES - BUILDING 055

In Building 055, air samples were taken routinely at about 30 separate locations adjacent to the glove box trains, as well as at several other locations. The results of these samples for 1978 are tabulated in Table 5 in a descending order of magnitude with the date (month-week) of each measurement noted. In this manner, any unusual airborne activity release is more readily apparent. For example, in 1978, the week ending December 31, and possibly that ending October 27, seem to occur disproportionately often on the tabulation. It may be noted from Table 5 that the highest weekly concentration occurred near glove Box 24NE for the week ending December 22 when the cumulative air concentration was $9 \times 10^{-12}\ \frac{\mu Ci\text{-}h}{cc}$. This value is less than 20% of the weekly integrated MPC for the most restrictive material present (Pu-239).

731-M.262/vlg

TABLE 5
AIR SAMPLES - BUILDING 055 - 1978

Sampling Location	Maximum Cumulative Weekly Exposure (μCi-h/cc)	Week(s) Ending
GB24NE	9-12*	12/22
Support Lab	6.3-12	12/1
GB3S	5.4-12	10/27
Chemistry Lab	4.8-12	12/15
GB9S	4.5-12	10/27
Support Area	4.5-12	10/27
GB1A	4.2-12	12/31
GB8N	4.2-12	12/31
GB11N	4.2-12	12/31
GB24SE	4.2-12	12/31
GB5S	3.9-12	10/27
GB27S	3.6-12	12/31
GB19S	3.6-12	12/31
GB18SW	3.6-12	10/27
GB6N	3.6-12	12/08
GB18S	3.6-12	10/31
GB20S	3.3-12	11/17, 11/30
GB4N	3.3-12	11/17
GB3AN	3.3-12	12/08, 10/27
GB1A	3.3-12	10/27
GB15A	3-12	10/27
Fan Room	2.7-12	11/24, 12/31
STA B	2.4-12	11/24
Vault	2.4-12	12/31
STA A	2.1-12	11/24
GB24SW	1.5-12	08/25
STA E	1.2-12	11/24
STA C	1.2-12	11/03
STA D	9-13	11/02

*9-12 = 9×10^{-12}. To obtain average concentration in μCi/cc, divide value shown by 40.

731-M.262

III. EFFLUENT MONITORING - 1978*

Effluents which may contain radioactive material are generated at 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 as the De Soto 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 and/or particulate aerosol effluents. Contaminated liquids are not 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 any remaining 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 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. Particulate filtration efficiency increases above and below this size.

---

*A separate and comprehensive report on facility effluents and environmental monitoring is prepared annually. The data presented in Section III and IV of this report were almost wholly abstracted from this report for 1978.[3]

FORM 719-P REV. 2-80

TABLE 6

ATMOSPHERICALLY DISCHARGED EFFLUENT RELEASED TO UNRESTRICTED AREAS - 1978

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	2.3×10^{10}	α	1.7×10^{-16}	$<1.8 \times 10^{-14}$	1.4×10^{-13}	$<1.6 \times 10^{-5}$
			β	5.4×10^{-16}	$<5.8 \times 10^{-15}$	3.8×10^{-14}	$<5.0 \times 10^{-6}$
004	Stack Exit	3.4×10^{10}	α	2.7×10^{-16}	$<6.8 \times 10^{-16}$	4.7×10^{-15}	$<6.5 \times 10^{-7}$
			β	9.2×10^{-16}	$<4.5 \times 10^{-15}$	4.6×10^{-14}	$<4.3 \times 10^{-6}$
020	Stack Exit	1.1×10^{10}	α	0.9×10^{-16}	$<4.0 \times 10^{-16}$	1.8×10^{-15}	$<1.3 \times 10^{-7}$
			β	3.0×10^{-16}	1.4×10^{-13}	1.0×10^{-12}	5.9×10^{-5}
055	Stack Exit	8.0×10^{9}	α	2.4×10^{-16}	$<3.5 \times 10^{-16}$	9.7×10^{-16}	$<8.1 \times 10^{-8}$
TOTAL		7.6×10^{10}					$<1.7 \times 10^{-5}\alpha$ $<6.8 \times 10^{-5}\beta$

Annual average ambient air radioactivity concentration - 1978

α 7.5×10^{-15} $\mu C/m\ell$

β 8.8×10^{-14} $\mu C/m\ell$

*Effluent radioactivity is generally less than ambient air radioactivity. If released concentrations were at the ambient air radioactivity concentrations, a total of 2.4×10^{-4} Ci would be released, which value is ~2.5 times greater than the actual release.

731-M.262

The average concentration and total radioactivity in gaseous effluent released 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 is less radioactive than the ambient air, which is indicative that there are not any radioactivity releases during normal facility operations.

Liquid wastes released to sanitary sewers, a controlled area as provided for by CAC 17 and 10 CFR 20, are generated at the De Soto site only. Liquid wastes are discharged from Building 001 following analysis on a volume batch basis only. 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 sewage system. No radioactive liquid effluents are released from the Santa Susana Buildings 020 or 055 except as controlled liquid radioactive waste solidified and shipped offsite for land burial at approved disposal sites. The average concentration and total radioactivity in liquid effluents discharged in 1978 are shown in Table 7.

TABLE 7

LIQUID EFFLUENT DISCHARGED TO SANITARY SEWER - 1978

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	Retention Tank	40,500	α	1.2×10^{-9}	1.5×10^{-7}	3.7×10^{-7}	2.3×10^{-5}
			β	4.1×10^{-9}	1.1×10^{-7}	3.3×10^{-7}	1.7×10^{-5}
004	Proportional Sampler	1,039,600	α	1.2×10^{-9}	$<2.1 \times 10^{-8}$	1.5×10^{-7}	$<8.0 \times 10^{-5}$
			β	4.1×10^{-9}	7.8×10^{-8}	1.4×10^{-6}	3.1×10^{-4}
020*	-	0		-	-	-	-
055*	-	0		-	-	-	-

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

IV. ENVIRONMENTAL MONITORING

A. INTRODUCTION

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 were observed.

In addition to environmental monitoring, work area air and atmospherically discharged effluents are continuously monitored or sampled, as appropriate. This provices a direct measure of the effectiveness of engineering controls and allows remedial action to be taken before a significant release of hazardous material can occur.

Soil and vegetation sample collection and analysis for radioactivity were initiated in 1952, in the Downey, California area, where the ESG was originally located. Environmental sampling was subsequently extended to the proposed Sodium Reactor Experiment (SRE) site in the Simi hills in May 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 Group relocated to Canoga Park in 1955. The primary purpose of the environmental monitoring program is to survey environmental radioactivity adequately to ensure that ESG operations do not contribute significantly to environmental radioactivity.

B. GENERAL DESCRIPTION

Environmental sampling stations that are located within the boundaries of ESG sites are referred to as "onsite" stations; those located within a 10-mile radius of the sites are referred to as "offsite" stations. The onsite environs of the De Soto and SSFL sites are sampled monthly to determine the concentration of radioactivity in typical surface soil, vegetation, and water. Soil is also sampled onsite semiannually for plutonium analysis. Similar offsite environmental samples, except for plutonium analysis, are obtained quarterly. Continuous onsite and offsite ambient air sampling provides information concerning long-lived airborne particulate radioactivity. A site ambient radiation monitoring program, utilizing thermoluminescent dosimetry (TLD), begun in 1971 was expanded during 1978. The locations of sampling stations are shown in Figures 5 through 7 and listed in Table 8.

No intentional releases of any liquids containing radioactivity are made to unrestricted areas. Liquid wastes generated at the De Soto site are discharged into the city sewage system. This effluent is sampled for determination of contained radioactivity. Sanitary sewage from all DOE and ESG facilities at the SSFL site is treated at an onsite sewage plant. The plant effluent drains into a retention pond, located on the adjoining Rocketdyne Division site. The surface water drainage system of the SSFL is composed of catch ponds and open drainage ditches leading to the Rocketdyne retention pond. This pond also receives the ESG site sewage plant effluent. Water from the pond may be reclaimed as industrial process water, or it may be released offsite into Bell Creek, a tributary of the Los Angeles River. The pond also was monitored at discharge for tritium and for nonradioactive pollutants by Rocketdyne Division as required by discharge permits issued to Rocketdyne by the California Regional Water Quality Control Board.

Air sampling is performed continuously at the De Soto and SSFL sites with automatic air samplers, operating on 24-h sampling cycles. Airborne particulate

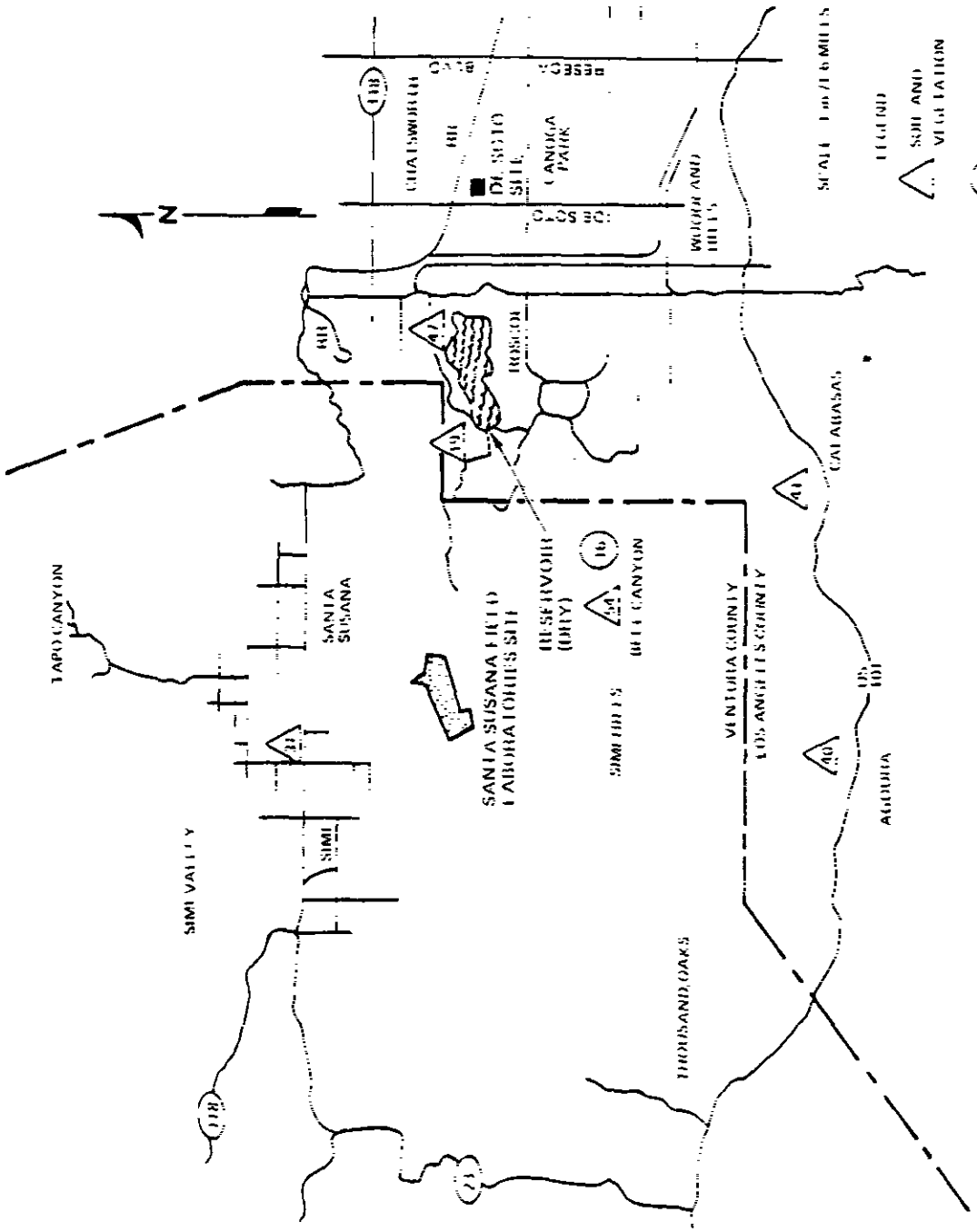

Figure 5. Map of Canoga Park, Simi Valley, Agoura and Calabasas Sampling Stations

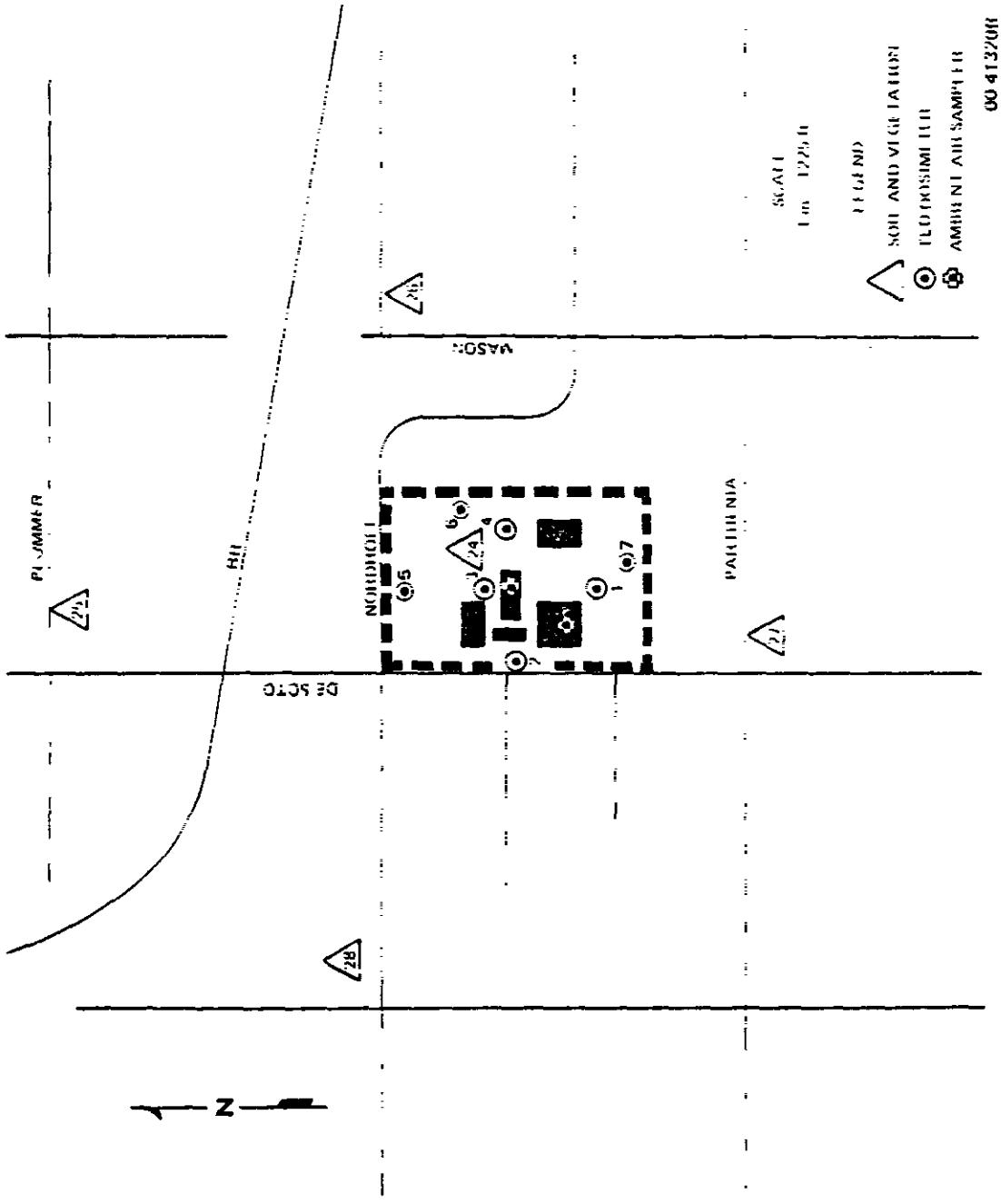

Figure 6. Map of De Soto Site and Vicinity Sampling Stations

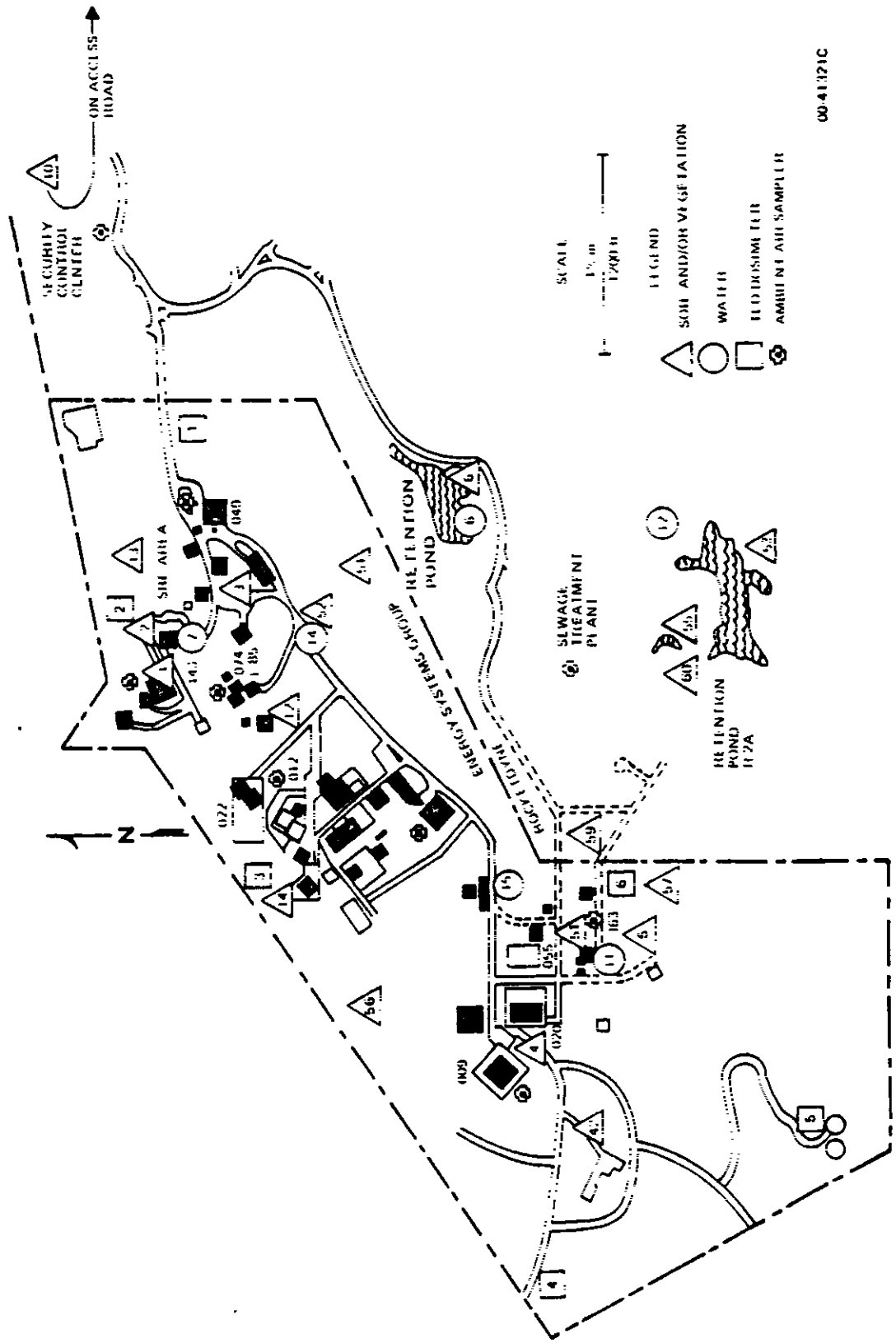

Figure 7. Map of Santa Susana Field Laboratories Site Sampling Stations

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 Norchoff 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 Muc)
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. 354
TLD-1	Off Site, Northridge
TLD-2	Off Site, Simi Valley
TLD-3	Off Site, Northridge

TLD — Thermoluminescent Dosimeter Location

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 20-h decay period to eliminate radon particulate daughters. The volume of a typical daily ambient air sample is approximately 25 m^3.

C. RESULTS

The average radioactivity concentrations in local soil, vegetation, surface water, and in ambient air for 1978 are presented in Tables 9 through 12. 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. Except for ambient air radioactivity, none of the maximum observed values, which occurred randomly during the year, show a great increase over the average values beyond natural variability. The air sample data reflect March and December atmospheric nuclear device detonations which resulted in marked but transient increases in local airborne radioactivity levels.

The results reported in Tables 9 and 11 show no significant difference in radioactivity content between onsite and offsite soil and vegetation samples. Table 10 shows no significant variations in soil plutonium concentrations for the

TABLE 9

SOIL RADIOACTIVITY DATA — 1978

Area	Activity	No. Samples	Gross Radioactivity (μCi/g)	
			Annual Average Value (95% Confidence Level)	Maximum Observed Value*
On Site	α	144	$(6.3 \pm 1.5)\ 10^{-7}$	9.9×10^{-7}
	β	144	$(2.4 \pm 0.09)\ 10^{-5}$	4.8×10^{-5}
Off Site	α	48	$(5.1 \pm 1.4)\ 10^{-7}$	10.0×10^{-7}
	β	48	$(2.4 \pm 0.09)\ 10^{-5}$	3.4×10^{-5}

*Maximum value observed for single sample

TABLE 10

SOIL PLUTONIUM RADIOACTIVITY DATA — 1978

Sample Location	June 8, 1978 Survey Results		December 13, 1978 Survey Results	
	Pu^{238} (μCi/g)	$Pu^{239} - Pu^{240}$ (μCi/g)	Pu^{238} (μCi/g)	$Pu^{239} + Pu^{240}$ (μCi/g)
S-56	$(-2.90 \pm 2.3)\ 10^{-9}$	$(1.04 \pm 2.3)\ 10^{-9}$	*	$(9.05 \pm 3.6)\ 10^{-9}$
S-57	$(-2.13 \pm 2.2)\ 10^{-9}$	$(1.14 \pm 2.2)\ 10^{-9}$	*	$(6.21 \pm 3.5)\ 10^{-9}$
S-58	$(-3.59 \pm 2.3)\ 10^{-9}$	$(1.46 \pm 3.0)\ 10^{-9}$	*	$(5.54 \pm 2.8)\ 10^{-9}$
S-59	$(-3.61 \pm 2.3)\ 10^{-9}$	$(3.10 \pm 3.4)\ 10^{-9}$	$(1.22 \pm 2.4)\ 10^{-9}$	$(5.98 \pm 2.8)\ 10^{-9}$
S-60	$(-2.17 \pm 3.2)\ 10^{-9}$	$(3.35 \pm 3.9)\ 10^{-9}$	$(-3.66 \pm 1.7)\ 10^{-9}$	$(8.28 \pm 4.4)\ 10^{-9}$

Note: Minus (-) indicates sample value less than reagent blank.
*Result significantly less than the minimum detection level.

TABLE 11

VEGETATION RADIOACTIVITY DATA – 1978

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	$(<5.8 \pm 4.1)\ 10^{-8}$	$(<2.4 \pm 1.7)\ 10^{-7}$	9.6×10^{-7}
	β	144	$(2.6 \pm 0.05)\ 10^{-5}$	$(1.66 \pm 0.03)\ 10^{-4}$	3.19×10^{-4}
Off Site	α	48	$(<6.6 \pm 4.4)\ 10^{-8}$	$(<2.4 \pm 1.6)\ 10^{-7}$	6.6×10^{-7}
	β	48	$(3.3 \pm 0.07)\ 10^{-5}$	$(1.43 \pm 0.03)\ 10^{-4}$	2.41×10^{-4}

*Maximum value observed for single sample

TABLE 12

SSFL SITE – DOMESTIC WATER RADIOACTIVITY DATA – 1978

Area	Activity	No. Samples	Gross Radioactivity (μCi/mℓ)	
			Average Value (95% Confidence Level)	Maximum* Observed Value
ESG-SSFL	α	24	$(<2.6 \pm 2.8)\ 10^{-10}$	4.4×10^{-10}
	β	24	$(3.0 \pm 0.8)\ 10^{-9}$	3.6×10^{-9}

*Maximum value observed for single sample

TABLE 13

BELL CREEK AND ROCKETDYNE SITE RETENTION POND
RADIOACTIVITY DATA — 1978

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	$(4.2 \pm 1.3)\ 10^{-7}$	7.4×10^{-7}	NA
	β	12	$(2.3 \pm 0.08)\ 10^{-5}$	2.7×10^{-5}	NA
Pond R-2A Mud No. 55 (μCi/g)	α	12	$(6.5 \pm 1.6)\ 10^{-7}$	9.0×10^{-7}	NA
	β	12	$(2.5 \pm 0.09)\ 10^{-5}$	2.8×10^{-5}	NA
Bell Creek Vegetation No. 54 (μCi/g ash)	α	12	$(<2.6 \pm 1.7)\ 10^{-7}$	5.1×10^{-7}	NA
	β	12	$(1.56 \pm 0.03)\ 10^{-4}$	2.14×10^{-4}	NA
Bell Creek Vegetation No. 54 (μCi/g dry weight)	α	12	$(<4.6 \pm 3.8)\ 10^{-8}$	1.0×10^{-7}	NA
	β	12	$(2.8 \pm 0.05)\ 10^{-5}$	4.4×10^{-5}	NA
Bell Creek Water No. 16 (μCi/mℓ)	α	12	$(<2.4 \pm 2.8)\ 10^{-10}$	$<2.4 \times 10^{-10}$	<0.005
	β	12	$(2.5 \pm 0.8)\ 10^{-9}$	3.5×10^{-9}	0.8
Pond Water No. 6 (μCi/mℓ)	α	12	$(<2.5 \pm 2.8)\ 10^{-10}$	3.5×10^{-10}	<0.005
	β	12	$(4.3 \pm 0.8)\ 10^{-9}$	7.0×10^{-9}	1.4
SSFL Pond R-2A Water No. 12 (μCi/mℓ)	α	12	$(<2.5 \pm 2.8)\ 10^{-10}$	2.7×10^{-10}	<0.005
	β	12	$(4.6 \pm 0.8)\ 10^{-9}$	6.3×10^{-9}	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.

TABLE 14

AMBIENT AIR RADIOACTIVITY DATA — 1978

Site Location	Activity	No. Samples	Average Value (95% Confidence Level)	Maximum* Observed Value (daily)	% of Guide†
De Soto On Site (μCi/mℓ)	α§	713	$(<8.4 \pm 8.1)\ 10^{-15}$	9.5×10^{-14}	<0.28
	β**	713	$(<9.1 \pm 1.7)\ 10^{-14}$	1.4×10^{-12}	<0.030
SSFL On Site (μCi/mℓ)	α§	1724	$(<7.2 \pm 7.9)\ 10^{-15}$	2.1×10^{-14}	<12.0
	β**	1724	$(<8.8 \pm 1.7)\ 10^{-14}$	1.5×10^{-12}	<0.29
SSFL Sewage Treatment Plant Off Site (μCi/mℓ)	α§	327	$(<7.3 \pm 7.3)\ 10^{-15}$	4.4×10^{-14}	<12.2
	β**		$(<8.4 \pm 1.6)\ 10^{-14}$	1.2×10^{-12}	<0.28
SSFL Control Center Off Site (μCi/mℓ)	α§	351	$(<7.1 \pm 7.3)\ 10^{-15}$	3.4×10^{-14}	<11.8
	β**		$(<8.9 \pm 1.6)\ 10^{-14}$	1.3×10^{-12}	<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 = 6.0×10^{-15} μCi/mℓ — Individual daily samples with activity levels of 0 to 6.0×10^{-15} μCi/mℓ are recorded and averaged as 6.0×10^{-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.

1978 sample sets. The detected radioactivity 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 onsite by the same piping system previously used when all facility process water was obtained from onsite wells. Conversion to the domestic water supply was completed during 1969. Two onsite water wells were operated during 1978 to reduce consumption of Ventura County domestic water as a conservation measure due to local drought conditions. The well water proportion in the blend averaged about 38% for the year for a total well water consumption of approximately 6.3×10^{6} gal. Pressure for the water system is provided by elevated storage tanks.

Water from the system is sampled monthly at two widely separated SSFL site locations. The average domestic water radioactivity concentration for 1978 is presented in Table 12.

As discussed earlier, 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 values of 5×10^{-6} $\mu Ci/m\ell\alpha$ and 3×10^{-7} $\mu Ci/m\ell\beta$, for Pu^{236} and for Sr^{90}, respectively, are appropriate. The correspondingly most restrictive guide value for De Soto site wastewater radioactivity discharged to the sanitary sewers, 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 mixed 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 depleted uranium). The guide value of 3×10^{-10} $\mu Ci/m\ell\beta$ for Co^{60} for ambient air radioactivity is appropriate since it is the most restrictive limit for beta-emitting radionuclides present at the De Soto site. Guide value percentages are not presented for soil or vegetation data since concentration guide values have not been established.

Ambient air sampling for long-lived particulate alpha and beta radioactivity is performed continuously with automatic sequential samplers at both the De Soto 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 radon particulate daughters. The average concentrations of ambient air alpha and beta radioactivity are presented separately in Table 14.

Radioactivity levels observed in environmental samples for 1978, reported in Tables 11 through 13, 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 continuing, although infrequent, foreign atmospheric nuclear tests continue to be occasionally observable in daily ambient air radioactivity levels; however, the long-term effects on surface sample radioactivity levels are not discernible. The continued relative constancy in current levels of environmental radioactivity 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 6 and 7) on or near the perimeters of the De Soto and SSFL sites. Each dosimeter, sealed in a light-proof energy compensation shield, is 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 onsite TLD monitoring locations used during the year. Three additional dosimeter sets, located at locations up to 10 miles from the ESG sites, are similarly evaluated to determine the local area offsite ambient radiation level, which averaged 0.014 mrem/h for 1978. The average radiation dose rate and equivalent annual dose monitored at each dosimeter location are presented in Table 15.

The table shows that radiation dose rates and equivalent annual doses monitored onsite are nearly identical to levels monitored at three widely separated

TABLE 15

DE SOTO AND SSFL SITES — AMBIENT RADIATION DOSIMETRY DATA — 1978

	TCD	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	0.015	131
5	De Soto	0.015	131
6	De Soto	0.013*	114
7	De Soto	0.014	123
1	SSFL	0.017	149
2	SSFL	0.017	149
3	SSFL	0.017	149
4	SSFL	0.017	149
5	SSFL	0.011+	96
6	SSFL	0.016	140
1	Off-Site Control	0.014	123
2	Off-Site Control	0.014	123
3	Off-Site Control	0.015	131

*Excludes second quarter data due to missing dosimeter.
+Dosimeter partially shielded by water tank. Relocated to 360° exposure field beginning third quarter. Unshielded annual average dose rate is 0.014 mrem/h, equivalent to an annual dose of 123 mrem.

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 in the atmosphere in addition to radioactive fallout from nuclear weapons tests. Locally, this is approximately 125 mrem/yr. 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 onsite and offsite locations are nearly identical, no measurable radiation dose to the general population or to individuals in uncontrolled areas resulted from ESG operations.

Figure 8 is a graph of the daily averaged long-lived alpha and beta ambient air radioactivity concentrations for the De Soto and SSFL sites during 1978. The average beta concentration for each month is also indicated by horizontal bars. The graph shows two prominent peaks occurring during the year, a moderate spring increase in concentration, and subsequent decreasing levels through the year's end.

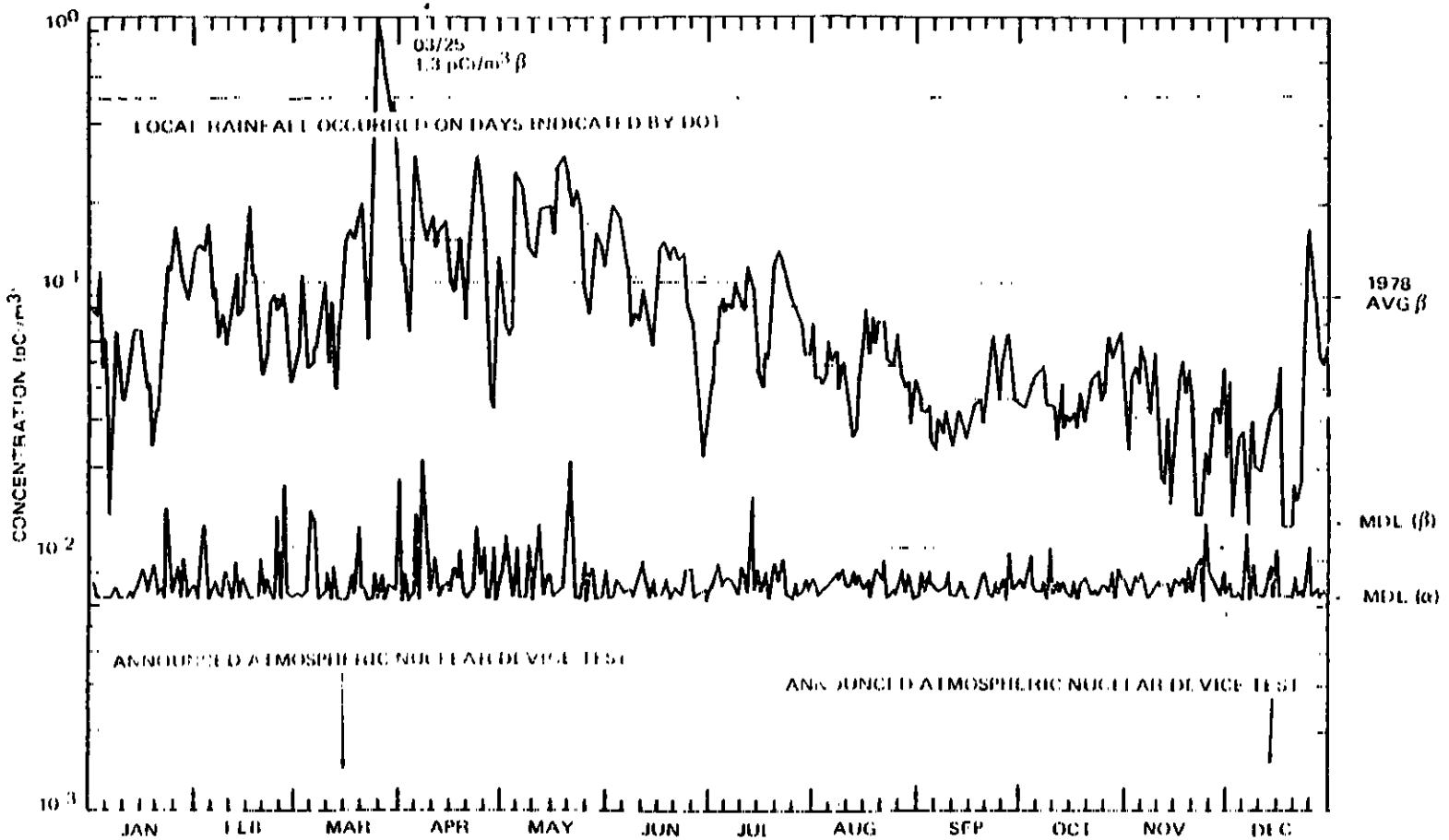

Figure 8. Daily Averaged Long-Lived Airborne Radioactivity at the De Soto and Santa Susana Field Laboratories Sites - 1978

V. UNUSUAL EVENTS

A. BUILDING 001

On January 9, 1978, a small UAL_x alloy fire occurred in the crusher glove box. An argon gas purge was initiated while the remainder of the uninvolved material was removed from the box. The fire was extinguisned as soon as the argon gas was contacted. All smears taken around the box and a nasal smear of the HS&RS representative were evaluated as ~20 dpm, alpha. The crusher box was examined and a leak was detected at the bottom of the front glass. The oxygen analyzer was replaced and the front glass was replaced with new glass that has an extra gasket. The alpha air monitor located next to the involved glove box did not alarm. No detectable internal or external exposures occurred as a result of the fire.(6)

B. BUILDING 004

None

C. BUILDING 020

On April 25, an employee was accidently splattered with spent sodium digester alcohol over ~25% of his body. The protective clothing (coveralls) were soaked. He was wearing a respirator at the time of the accident. A radiation reading of a maximum of 12 mRem/h was measured at the clothing surface. The employee removed the contaminated clothing and showered until no further radiation readings could be measured at his body surface.(5)

On September 20, reactor irradiated material was unloaded in an improperly ventilated area which allowed some mixed fission product contamination to spread to the clean area of the service gallery. A maximum surface area radiation level of ~10 mR/h resulted. No significant personnel exposures occurred and the area was subsequently decontaminated.(5)

D. BUILDING 055

None

VI. SUMMARY/TRENDS - EXPOSURES, EFFLUENTS

A comparison of the 1978 data with that of previous years shows:

1) The population dose has increased as:

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

This increase has been due almost entirely to decontamination and disposition (D&D) of excess DOE facilities.

2) Median and average exposures show no significant trend:

Year	Average	Median
1975	110	22 mrem
1976	200	40 mrem
1977	310	60 mrem
1978	200	20 mrem

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

VII. ANTICIPATED ACTIVITIES DURING NEXT REPORTING PERIOD (1979)

A. BUILDINGS 001/004

Continued ATR fuel element fabrication. Also, commence modifying equipment for a low enrichment (20%) powder line.

B. BUILDING 020

Completion and cleanup of HNPF fuel decladding. Commence study of Fermi fuel declad process.

C. BUILDING 055

The preparation of depleted uranium and carbide pellets will be continued.

REFERENCES

1. U.S. Nuclear Regulatory Commission - Special Nuclear Materials License SNM-21, USNRC, September 15, 1977

2. "Annual Review of Radiological Controls - 1977," R. S. Hart, Rockwell International, Energy Systems Group, N001TI000098 (to be issued)

3. "Energy Systems Group Environmental Monitoring and Facility Effluent Annual Report - 1978," J. D. Moore, Rockwell International, Energy Systems Group, ESG-79-7, April 1979

4. "Annual Review of Radiological Controls - 1975," R. S. Hart, Rockwell International, Energy Systems Group, N001TI990002, July 2, 1979

5. Personal communication, F. H. Badger, Rockwell International, Energy Systems Group, May 15, 1980

6. IL, "ATR Powder Room Crusher Glove Box Fire," J. H. Wallace to R. J. Tuttle, January 19, 1978, Energy Systems Group, Rockwell International

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 wnen 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.

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	1.6	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.

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

The very high thermal neutron sensitivity of indium makes it extremely useful as an exposure indicator. In the event of an accidental criticality, the high-energy neutrons will be moderated and reflected by the body, thereby producing thermal and intermediate energy neutrons that will activate the indium. By using a G.M. survey instrument, those exposed can be detected for 5 hr following an incident.

Maximum 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 nonpenetrating radiation.

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.

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

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

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 isobutyl 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 with 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 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 g	±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 FP 3a 60 dpm FP 3b	±50%	300 ml	Volatile fission products lost.
Tritium	H3	Tritium	2.25×10^6 dpm	±50%	10 ml	
Plutonium	PU A	Plutonium	0.0495 dpm	±50%	1000 ml	Greater accuracy than PU B analysis.
Plutonium (B)	PU B	Plutonium	0.0495 dpm	±75%	1000 ml	Double precipitations, washes and extractions are eliminated for faster analysis at reduced accuracy.

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
Plutonium (B) (Optional)	PU B	Plutonium	0.75 dpm	±100% alpha counting	1000 ml	Sample proportional counted for Alpha-radiation for immediate result. Sample may be later autoradiographed.
Strontium-90	SR90	Strontium-90	30 dpm	±50%	200 ml	
Thorium	TH	Thorium	0.99 g	±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 of SS disc and autoradiographed.
Gross Alpha (1b)	GA1B	Uranium and Plutonium	9 dpm	±50%	100 ml	Sample planchetted and proportional counted for alpha.
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	I131	Iodine-131	300 dpm	±50%	250 ml	Decay corrected to sampling date.