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ABSTRACT

Data on employee exposures, bioassay results, effluent releases, in-plant airborne radioactivities, and environmental monitoring for ESG operations during 1983 are reviewed. This review is prepared, as required by License Condition No. 23 of Special Nuclear Materials License No. SNM-21, 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.

Continued high personnel exposures in the SEFOR decladding project, while well below the regulatory limits, led to the development of a formal radiation exposure control program. Internal limits on exposure imposed by this program resulted in significant decreases in personnel exposures during the latter part of the year.

Effluent releases are at insignificant levels compared to regulatory standards, do not show any upward trends and do not appear to be reducible by reasonable means.

To the extent covered by this review, equipment for effluent and exposure control was properly used, maintained, and inspected.

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FORM 734-C REV. 12-84

CONTENTS

TABLES

FIGURES

INTRODUCTION

Condition 23 of the Energy Systems Group special nuclear materials license[1] requires that: "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 12 months and review of the data from the following areas: employee exposures, bioassay results, effluent releases, in-plant airborne radioactivity, and environmental monitoring." While this report is prepared primarily to satisfy a requirement of the NRC license, all operations have been included.

These reports for the years 1975 through 1982[2-9] 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) have not been made separately for each type of activity.

Additionally, it is not possible to separate the integrated personnel radiological doses to that attributable to either nonlicensed activities for the DOE or the activities licensed by NRC or the State of California.

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

1) <u>Fuel Fabrication Decommissioning</u> - Building 001 and supporting operations in Buildings 001 and 004, De Soto Facility, Canoga Park, California

2) <u>Rockwell International Hot Laboratory (RIHL)</u> - Building 020, Santa Susana Field Laboratories

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

4) Radioactive Material Disposal Facility (RMDF) - Buildings 021, 022, and related facilities at Santa Susana Field Laboratories (DOE jurisdiction)

Personnel exposures and dosimetry in all activities with radioactive material are included in this report.

I. PERSONNEL DOSIMETRY

Personnel dosimetry techniques generally consist of two types: those which measure radiation incident on the body from external sources (film badges) and those which measure internal body organ accumulations of radioactivity via inhalation, ingestion, or through cuts or puncture wounds (bioassays). These measurement methods provide a natural separator of the exposure modes 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 (i.e., internal body deposits).

A. FILM/TLD DATA

1. Whole Body Monitoring

Personnel external radiation exposures for the pertinent activities for the year are shown in Figure 1 as a cumulative log-normal distribution. It should be noted (see Summary, Section VI) that all whole-body exposures were less than 5 rem and were well below the allowable annual occupational total of 12 rem (for NRC and State-licensed operations).

For comparison, the distributions of exposures reported for NRC licensees (10) and for DOE contractor[11] for 1983 are shown as solid curves. (In previous reports in this series, the "UNSCEAR Reference Distribution"[12] had been shown. However, in the UNSCEAR 1982 report[13], the significance of this distribution has been revised. In the earlier report, it was implied that the Reference Distribution represented an ALARA situation, and this view was rather widely adopted. UNSCEAR states in the 1982 report, "It was not the intent of the Committee that this reference distribution be considered an ideal or optimal distribution of doses and it should not be so interpreted." Further, ". . . because the attention which has been paid to the reference distribution was more than anticipated, . . ." a "basic characteristic" has been chosen for the purpose of comparing distributions. This is the fraction

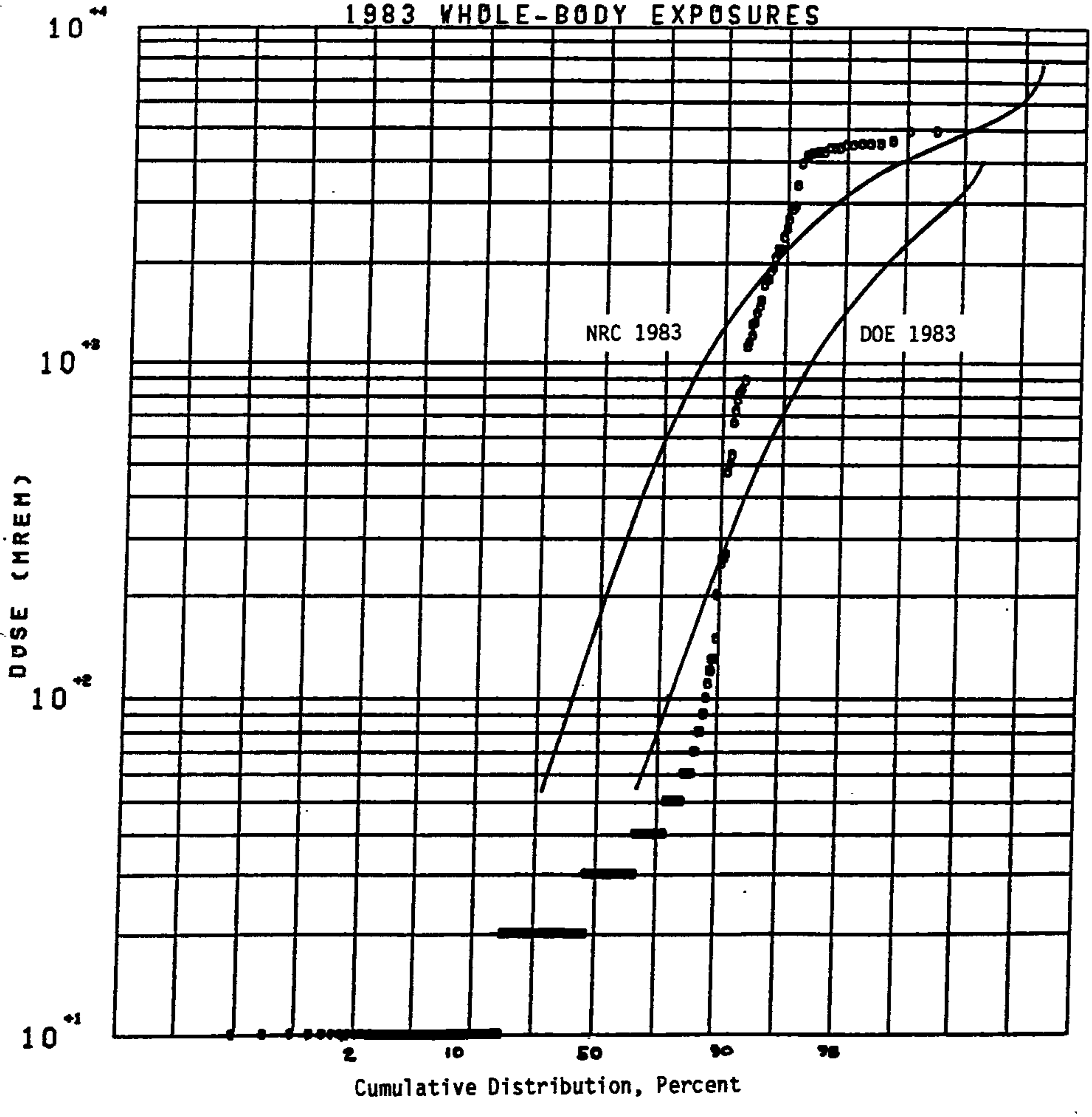

Figure 1. Cumulative Log-Normal Distribution for Whole-Body Radiation Exposures of Occupationally Exposed Individuals in 1983

of the group-dose that is received above 1.5 rem. Considerable arbitrariness in selecting this particular definition is recognized and, while a "range" of 0.03 to 0.6 has been calculated for this parameter, UNSCEAR 1982 recommends a "normal range" of 0.05 to 0.5. The significance of this, other than indicating that one distribution is different from another, is unclear.)

While some differences can be seen between the NRC and ESG distributions (more moderate exposures for NRC licensees, more higher exposures for ESG), a significant comparison can be most readily made in terms of the group dose. The group dose received by ESG employees in 1983 amounted to 138.35 person-rem. *If the distribution of doses had been that shown for NRC licensees, the* group dose would have been 207 person-rem. If the doses had been those shown for DOE, the group dose would have been 65 person-rem. However, comparisons such as these are not very informative because of differences in the work performed between the ESG workforce and both the NRC licensees and the DOE contractors.

2. <u>Extremity Monitoring</u>

Hand exposures were the limiting factor during the first 4 months of the SEFOR decladding project, while the fuel was being removed from the cladding in a "hands-on" manner in the glove box. While the change to manipulators in the glove box in October of 1982 greatly reduced hand exposures, these exposures were carefully monitored for the rest of the project. The results of monitoring for the more highly exposed hand of each individual for each quarter have been combined in Figure 2. All extremity exposures were below the allowable quarterly limit of 18.75 rem.

B. 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 evaluated in this manner.

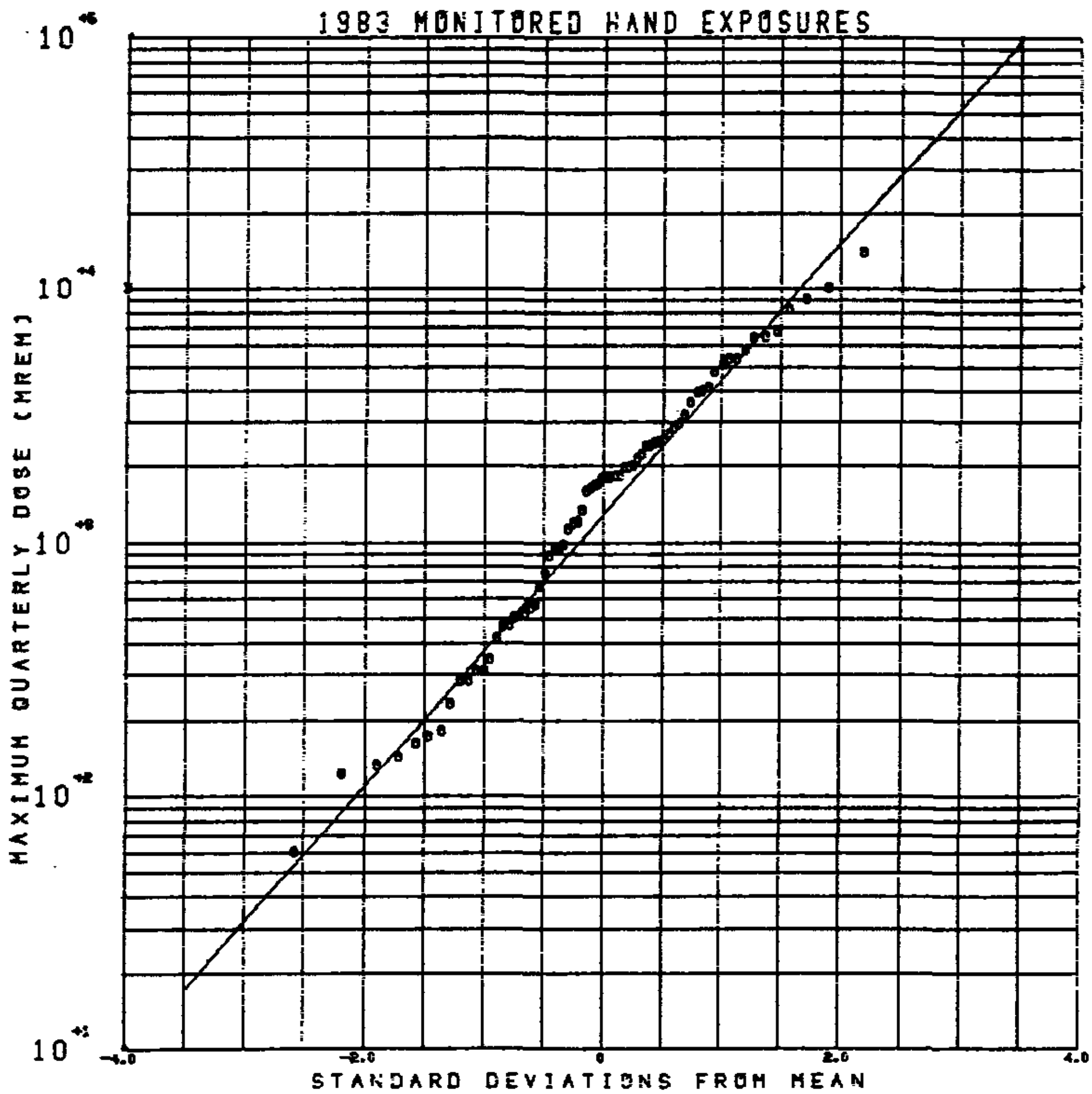

Figure 2. Hand Exposure Values (higher exposed hand from each individual) for SEFOR Project During 1983

Normally, urinalyses are performed quarterly and fecal analysis only when gross internal contamination is suspected. Individuals selected on the basis of either a positive uranium urinalysis result or a high potential for significant exposure to respirable uranium aerosols are monitored with an in-vivo lung count (IVLC) performed on-site by Helgeson Nuclear Services, Inc. A statistical summary of the results for 1983 appears in Table 1, while a detailed listing of the positive results are shown in Table 2. Data on the in-vivo lung scans performed in 1983 also appear in these tables.

Followup results are shown, where available, to indicate the decrease of detected activity to negligible levels.

The excretion rates assumed to be indicative of 1 MPBB for various radionuclides are:

Radionuclides	Standard Excretion Rate
Sr-90	480 dpm/day
Cs-137	660,000 dpm/day
Ra-226	10 dpm/day
Normal U (NU)	100 ug/day
Highly Enriched U (HEU)	220 dpm/day
Pu-239	121.4 dpm/day

These excretion rates are based on an assumption of equilibrium between intake and elimination. Transient elimination following an acute exposure will generally indicate a much higher body burden than actually exists.

The U-235 lung content equivalent to 1 MPLB is approximately 245 ug, or 17 nCi of highly enriched uranium (HEU) alpha activity. The majority of this activity is from U-234.

TABLE 1

SUMMARY OF BIOASSAYS - 1983

Measurement Type*	Total Tests	Total Positive Results	Total Individuals With Positive Results
UF	123	16	14
UR	123	3	3
GA2B	2	0	0
PUA	100	0	0
FP3A	74	0	0
FP3B	76	6	4
UIVLC	27	5	5
U-238 IVLC	2	0	0
Total	527	30	26

*UF = Uranium - Fluorometric
UR = Uranium - Radiometric
GA = Gross Alpha
PU-A = Gross Plutonium-alpha
FP = Fission Products
U-IVLC = Uranium In-Vivo Lung Count
(For a discussion of specific analytical techniques employed, as identified by "TYPE," see Appendix B in Reference 9)

TABLE 2

POSITIVE BIOASSAY RESULT SUMMARY - 1983

H&S Number	Sample Date	Analysis Type*	Results Per Vol. Anal.	Results Per 1500 ml-day	Assumed Specific Radionuclide	Assumed Critical Nuclide Equivalent MPBB (%)
1292	022583	IVLC	52.ug	21.1% MPLB	HEU	21.1
4857	022583	IVLC	42.ug	17.1% MPLB	HEU	17.1
3762	071983	UF	0.0003 ug	0.42 ug	NU	0.42
2762	102883	UF	0.0000 ug	0.0 ug	NU	0.00
NHCI	072683	UF	0.0002 ug	0.34 ug	NU	0.34
4897	042283	UF	0.0027 ug	4.05 ug	NU	4.05
	022585	IVLC	42.ug	17.1% MPLB	HEU	17.1
3486	091583	UF	0.0005 ug	0.75 ug	NU	0.75
	022583	IVLC	37.ug	15.0% MPLB	HEU	15.0
4393	072683	UF	0.0008 ug	1.15 ug	NU	1.15
4884	113183	UR	0.5467 dpm	4.10 dpm	HEU	1.86
4884	113183	UF	0.0010 ug	1.50 ug	NU	1.50
NHCI	102683	UR	0.5180 dpm	3.88 dpm	HEU	1.76
NHCI	072483	UF	0.0002 ug	0.36 ug	NU	0.36
NHCI	102383	UF	0.0001 ug	0.15 ug	NU	0.00
4574	071383	UF	0.0012 ug	1.83 ug	NU	1.83
4574	091283	UF	0.0003 ug	0.45 ug	NU	0.45
4574	102483	UF	0.0002 ug	0.30 ug	NU	0.00
3467	041283	UF	0.0004 ug	0.60 ug	NU	0.60
2729	092283	UF	0.0003 ug	0.45 ug	NU	0.45
2729	111883	UF	0.0000 ug	0.0 ug	NU	0.00
1754	062783	FP3B	15.0000 dpm	112.50 dpm	Cs-137	0.02
1754	091783	FP3B	8.6000 dpm	64.50 dpm	Cs-137	0.01
1754	042384	FP3B	0.0000 dpm	0.0 dpm	Cs-137	0.00
4074	041183	FP3B	8.9000 dpm	66.75 dpm	Cs-137	0.01
4074	062683	FP3B	11.6000 dpm	87.00 dpm	Cs-137	0.01
4074	091183	FP3B	10.9000 dpm	81.75 dpm	Cs-137	0.01
4074	110283	FP3B	0.0000 dpm	0.0 dpm	Cs-137	0.00
4404	070783	FP3B	11.1000 dpm	83.25 dpm	Cs-137	0.01
4404	081983	FP3B	5.9 dpm	dpm	Cs-137	0.00
0000	082283	UF	0.0009 ug	1.35 ug	NU	1.35
0000	090983	UF	0.0003 ug	0.45 ug	NU	0.45
NHCI	071383	UF	0.0003 ug	0.42 ug	NU	0.42
NHCI	092283	UF	0.0003 ug	0.45 ug	NU	0.45
NHCI	102583	UF	0.0002 dpm	0.30 ug	NU	0.00
4187	022583	IVLC	41.ug	16.7% MPLB	HEU	16.7
4187	112183	UR	0.6330 dpm	4.75 dpm	HEU	2.16
4187	112183	UF	0.0004 ug	0.60 ug	NU	0.60

*see following page

*IVLC:	In-Vivo Lung Count	ug:	Microgram
UF:	Uranium - Fluorometric	dpm:	Disintegration per Minute
UR:	Uranium - Radiometric	MPBB:	Maximum Permissible Body Burden
GA:	Gross Alpha	MPLB:	Maximum Permissible Lung Burden
GB:	Gross Beta	TBC:	Total Body Count
Pu:	Gross Plutonium		(For a brief description of the specific analytical techniques, see Appendix B of Reference 9)
FP:	Fission Products		

(FP3A is predominantly Sr-90; FP3B is predominantly Cs-137)

C. IN-VIVO LUNG SCANS

Measurements are periodically made of the total body or lung burden for 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 1983, 27 lung scans were made for uranium deposition. Five of the scans (on 5 different individuals) showed positive results (values exceeding the assigned 2 sigma uncertainty). Two of these individuals had never been exposed to uranium in any form. This is discussed in detail below. Follow-up scans showed elimination of these indicated lung burdens in all cases.

So-called "lung counts" have been done on our workers involved with potential inhalation of enriched uranium since 1966. These involve measurement of the 186-keV gamma ray from the decay of U-235 by means of gamma-ray scintillation spectrometry.

Initially, in response to apparent overexposure conditions that developed in uranium aluminide fuel fabrication in 1966, measurements were made with a medical system at UCLA. This system was calibrated for U-235 by use of comparative measurements done on one individual, at ORNL. Since 1968, these analyses have been done by Helgeson Nuclear Services with equipment and analytical techniques specifically developed to measure lung deposits of highly enriched uranium. Calibration of this equipment is based on measurements with a

*Helgeson Nuclear Services, Inc., Pleasanton, California phantom containing a specified amount of U-235 and having a certain chest-wall thickness. For each subject, the gross count pulse-height spectrum obtained during a counting time of 20 min is adjusted for background by subtracting an assumed background spectrum that consists of a straight line passing through the gross count values just below and just above the pulse-height region corresponding to 186 keV. The net count is then adjusted for chest-wall thickness and then converted to mass of U-235, based on the instrument calibration.

Until 1981, the results of these lung count measurements were reported as zero if the result were below the assigned 2-sigma uncertainty, based upon counting statistics, and as the actual value if equal to or above the uncertainty.

By 1981, we had concluded that, since values other than zero occurred rarely, much of the information from these measurements was being lost. Accordingly, Helgeson was requested to report all values, regardless of the assigned uncertainty.

Early review of those results showed that the values were somewhat uniformly distributed from a low value (greater than zero) to the maximum observed. That is, the results did not seem to be divided into the expected two groups of people: "no U-235 inhalation" and "some U-235 inhalation." All the results indicated some U-235 inhalation and deposition, regardless of the worker's history and urinalysis results.

The results of 8 sets of lung counts, involving 139 counts, are shown in Figure 3 as a cumulative frequency distribution. (Of these 139 values, 38 were greater than the assigned 2-sigma uncertainties, denoting a positive result. The uncertainties shown in the graph are 1-sigma values.) This plot shows that, except for perhaps 8 values at the low end and 4 values at the high end, the observed (reported) values match a Gaussian with a mean of about 32 micrograms U-235 and a standard deviation of 11. While many individuals in this group had some potential for inhalation of highly enriched uranium,

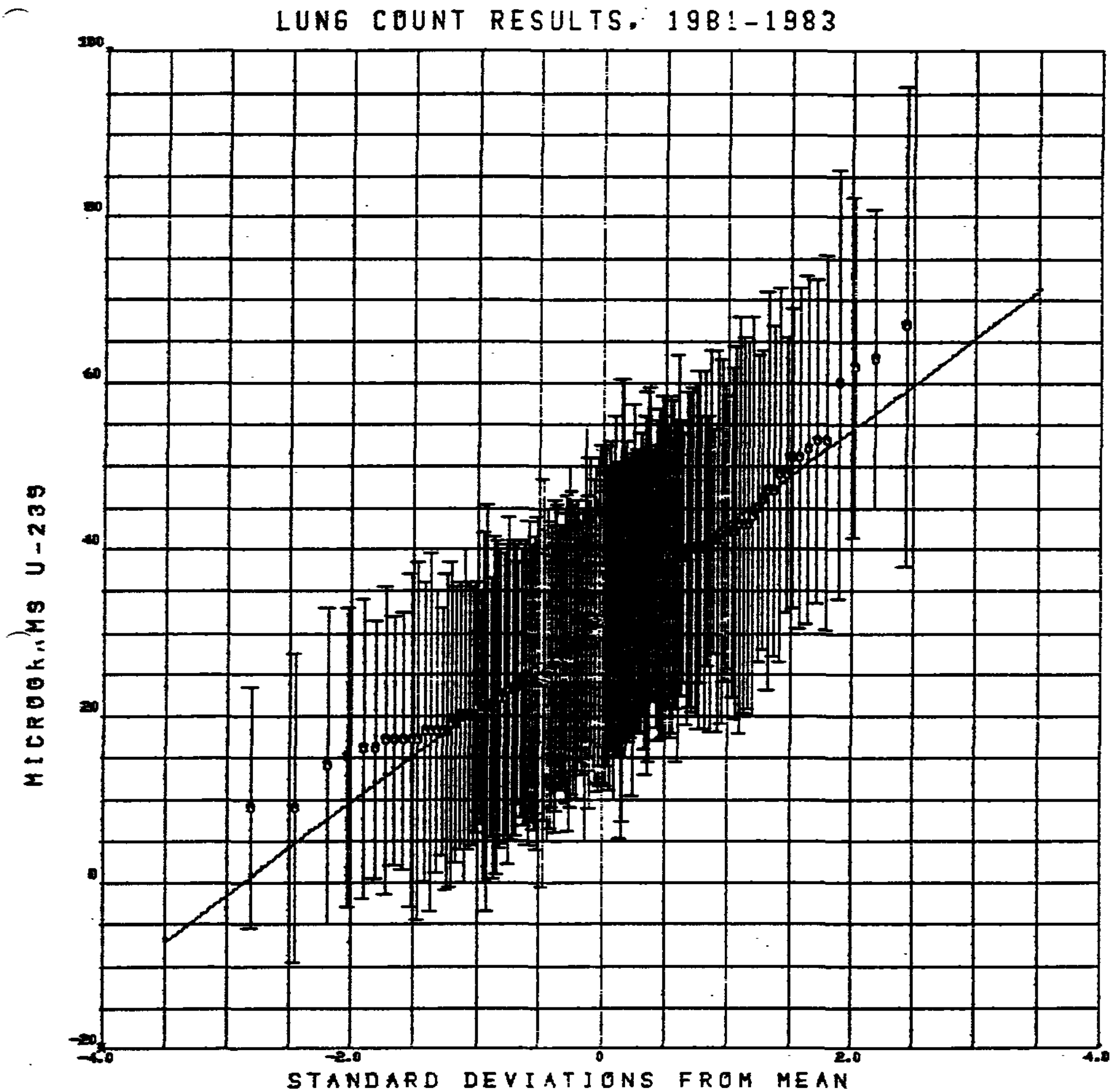

Figure 3. Reported Results of Lung-Counts for U-235 During 1981-1983

several had relatively little potential, yet had frequently shown more U-235 than those with higher risk. In February 1983, three individuals with no history of exposure to uranium were counted. These individuals were selected from the Publications Department, Data Processing, and Contract Administration, and had never worked in an area with potential for airborne uranium. Even though these three individuals presumably have no U-235 (other than the trivial amount due to the natural uranium content of the body, approximately 90 micrograms, including less than 1 microgram U-235, distributed through the entire body) the reported results showed:

42 ± 16.5 (all ± 1 standard deviation)
37 ± 15
33 ± 17

These values greatly exceed what would be expected from naturally present uranium.

On the basis of the similarity of results regardless of work history, and the fact that the individuals with no exposure showed measurable amounts (that would have been reported as positive results in two cases), it must be concluded that the majority of the measurements do not represent true depositions of U-235, but reflect a bias of the analysis. This bias could result, for example, from the use of a linear approximation to the background spectrum used in determining the net count. This would result in apparent small amounts of U-235 in persons with none but would be a minor effect for amounts approaching or exceeding a maximum permissible lung burden (1 MPLB = 245 micrograms).

Only those 4 values above 60 micrograms appear to represent true depositions, and those probably overestimate the actual value by a significant fraction of the bias. The effect of this bias may be roughly estimated by assuming that it results in an overestimate of 32.5 micrograms when none is present, and no overestimate when 1 MPLB (245 micrograms) is present, and has a linear behavior in this range.

With these assumptions, the 4 highest values are adjusted as below.

Observed Value	Adjusted Value
60	31.7
62	34.0
63	35.2
67	39.8

These results should be recognized as only approximate, but the review strongly indicates that few of the lung counts reported as positive really were, and the remaining true positives are significantly over-estimated.

II. RADIATION/RADIOACTIVITY MEASUREMENTS

The measurements and surveillance performed to determine local radiation levels in the working areas where licensed activities are performed are described below.

A. AREA RADIATION LEVELS

Film badges ("location badges") are placed throughout the facilities, and are kept in place during the entire calendar quarter. Some of these are in nominally low-exposure areas while some are in relatively high-exposure (but low-occupancy) areas. The average and maximum exposure rates determined for each quarter are shown in Table 3.

TABLE 3
LOCATION BADGE RADIATION EXPOSURE - 1983

Facility	Calendar Quarter Q1	Q2	Q3	Q4
	Average Exposure Rate (mR/h) / Maximum Exposure Rate (mR/h)			
Fuel Fabrication	0.02 / 0.02	0.02 / 0.02	0.03 / 0.03	*
RIHL	0.58 / 5.41	0.55 / 3.86	0.67 / 6.42	0.08 / 0.59
NMDF	0.02 / 0.05	0.01 / 0.05	0.01 / 0.05	0.01 / 0.05
RMDF	0.85 / 2.50	0.76 / 1.96	0.71 / 2.10	0.98 / 4.15

*Decommissioned

B. *INTERIOR AIR SAMPLES - WORKING AREAS*

In those working areas where the nature of the tasks being performed and the materials in use might lead to the potential for generation of respirable airborne radioactivity, periodic local air sampling is performed. A summary of these results for 1983 is given in Table 4.

TABLE 4

INTERIOR AIR SAMPLE SUMMARY - 1983

		Airborne Activity Concentration (Ci/ml)				
		Calendar Quarter				
Area	Sample	Q1	Q2	Q3	Q4	MPC
Fuel Fab	Lapel Max Week*	3.7×10^{-11}	3.5×10^{-11}	2.2×10^{-11}	†	1×10^{-10}
(C001)	Average*	1.4×10^{-12}	1.2×10^{-12}	8×10^{-13}	-	1×10^{-10}
	Stationary Max Week	5.2×10^{-11}	-	-	-	1×10^{-10}
	Average	1.9×10^{-12}	-	-	-	1×10^{-10}
RIHL	Unposted	1×10^{-15}	1×10^{-15}	1×10^{-15}	1×10^{-15}	2×10^{-12}
	β	1×10^{-14}	1×10^{-14}	1×10^{-14}	1×10^{-14}	1×10^{-9}
	Posted	1×10^{-14}	1×10^{-14}	1×10^{-14}	1×10^{-14}	2×10^{-12}
	β	1×10^{-13}	1×10^{-13}	1×10^{-13}	1×10^{-13}	1×10^{-9}
	Maximum	1×10^{-12}	2×10^{-12}	2×10^{-11}	2×10^{-13}	2×10^{-12}
	β	2×10^{-11}	4×10^{-12}	1×10^{-11}	1×10^{-14}	1×10^{-9}
NMDF	Stationary Max Week	9×10^{-15}	1.2×10^{-14}	1.6×10^{-14}	1×10^{-14}	1.3×10^{-12}
	Average	5×10^{-15}	5×10^{-15}	7×10^{-15}	7×10^{-14}	1.3×10^{-12}

**Adjusted for respirator protection factor:*

No mask = 1
Full face air purifying = 50
Airline supplied full face = 2000

†Decommissioned

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III. EFFLUENT MONITORING

Effluents which may contain radioactive material are generated at certain ESG facilities as a result of operations performed either under contract to DOE, or under the NRC Special Nuclear Materials License SNM-21, or under the State of California Radioactive Material License 0015-70. The specific facilities identified with the NRC license are Buildings 001 and 004 at the headquarters site and Buildings 020 and 055 at the SSFL at Santa Susana.

An annual report of effluent releases, prepared by Radiation & Nuclear Safety in the HS&RS Department, describes in detail the monitoring program at ESG for gaseous and liquid effluents from the ESG facilities. The data reported in the 1983 edition of that report[14] for atmospherically discharged and liquid effluents for the facilities identified above is presented in Tables 5 and 6, respectively.

TABLE 5

ATMOSPHERIC EMISSIONS TO UNRESTRICTED AREAS – 1983

Building	Approximate Emissions Volume (ft^3)	Activity Monitored	Approximate Minimum Detection Level (μCi/ml)	Annual Average Concentration (μCi/ml)	Sampling Period Maximum Observed Concentration (μCi/ml)	Total Radio-activity Released (Ci)	Percent of Guide[a]	Percent of Samples With Activity <MDL
001[c] De Soto	9.0×10^{9}	α β	1.6×10^{-16} 5.4×10^{-16}	2.5×10^{-13} 8.8×10^{-14}	7.9×10^{-13} 2.8×10^{-13}	5.2×10^{-5} 1.9×10^{-5}	8.33 0.029	0 0
004 De Soto	1.2×10^{10}	α β	2.1×10^{-16} 7.2×10^{-16}	2.6×10^{-15} 2.8×10^{-15}	3.1×10^{-14} 1.7×10^{-14}	1.1×10^{-6} 1.1×10^{-6}	0.087 0.0009	33 25
020 SSFL	1.2×10^{10}	α β	0.9×10^{-16} 3.0×10^{-16}	7.8×10^{-17} 4.1×10^{-15}	4.0×10^{-16} 1.6×10^{-14}	2.4×10^{-8} 1.3×10^{-6}	0.13 0.014	63 0
021-022 SSFL	1.2×10^{10}	α β	0.9×10^{-16} 3.0×10^{-16}	1.4×10^{-16} 3.4×10^{-15}	1.1×10^{-15} 1.6×10^{-14}	4.7×10^{-8} 1.1×10^{-6}	0.23 0.011	50 0
055 SSFL	9.1×10^{9}	α β	2.9×10^{-16} 9.6×10^{-16}	3.8×10^{-16} 4.7×10^{-15}	3.1×10^{-15} 1.3×10^{-14}	8.0×10^{-8} 1.1×10^{-6}	0.63 0.16	63 0
Total	5.4×10^{10}				Total	7.7×10^{-5}		
Annual average ambient air radioactivity concentration[b] (μCi/ml) – 1983		α β		1.8×10^{-15}[b] 3.2×10^{-14}[b]	Ambient equivalent[d]	1.83×10^{-3}		

[a]Assuming all radioactivity detected is from ESG operations.
Guide: De Soto site: 3×10^{-12} μCi/ml alpha, 3×10^{-10} μCi/ml beta; 10 CFR 20 Appendix B.
SSFL site: 6×10^{-14} μCi/ml alpha, 3×10^{-11} μCi/ml beta, 3×10^{-12} Ci/ml beta (055 only); 10 CFR 20 Appendix B, CAC-17, and DOE Order 5480.1 Chapter XI.
[b]Averaged result for 7-day (202 m^3) De Soto continuous air sampler.
[c]System permanently removed from service on July 29, 1983, at 0930.
[d]Natural radioactivity contained in volume of air discharged through ESG exhaust systems after filtration.

Note: All release points are at the stack exit.

TABLE 6

LIQUID EFFLUENT DISCHARGED TO SANITARY SEWER — 1983

Building	Point of Release	Approximate Volume (gal)	Activity Monitored	Approximate MDL (μCi/ml)	Annual Average Concentration (μCi/ml)	Diluted Concentration (μCi/ml)	Sample Maximum Observed Concentration (μCi/ml)	Total Radio-activity Released (Ci)	Percent of Guide[b]
001	Retention tank	29,750		1.0×10^{-9} 3.7×10^{-9}	3.8×10^{-7} 8.5×10^{-7}	0.9×10^{-9} 2.1×10^{-9}	1.3×10^{-6} 7.6×10^{-6}	4.5×10^{-5} 1.0×10^{-4}	0.042 0.085
004	Flow sampler	366,700		1.0×10^{-9} 3.7×10^{-9}	4.3×10^{-9}[c] 1.3×10^{-7}[c]	0.1×10^{-9} 3.9×10^{-9}	1.6×10^{-8} 7.8×10^{-7}	6.2×10^{-6} 1.8×10^{-4}	<0.001 0.013
Supply water		396,450		2.3×10^{-10} 6.4×10^{-10}	0.34×10^{-9} 3.53×10^{-9}		0.88×10^{-9} 5.1×10^{-9}	5.1×10^{-7} 5.3×10^{-6}	
Effluent water		12,100,000			d			6.0×10^{-5} 2.6×10^{-4}	
020[a]	—	0	—	—		—	—	—	—
021-022[a]	—	0	—		—	—	—		
055[a]	—	0	—	—		—	—		

[a]All liquid radioactive wastes are solidified and land buried as dry waste

[b]Guide: 9×10^{-4} μCi/ml alpha, 1×10^{-3} μCi/ml beta; 10 CFR 20 Appendix B, CAC-17

[c]Percent of samples < MDL: 15.7% alpha activity, 0% beta activity

[d]Assumed equal to supply water for this comparison

IV. ENVIRONMENTAL MONITORING PROGRAM

The basic policy for control of radiological and toxicological hazards at ESG requires that adequate containment of such materials be provided through engineering controls 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 ESG safety procedures and of the engineering safeguards incorporated into facility designs. Specific radionuclides in facility effluent or environmental samples are not routinely identified due to the extremely low radioactivity levels normally detected, but may be identified by analytical or radiochemistry techniques if significantly increased radioactivity levels are observed.

The annual report of environmental monitoring, prepared by Radiation & Nuclear Safety in the HS&RS Department, describes in detail the ESG environmental monitoring program.

Some of the data reported in the 1983 edition of that report(14) are presented here. It is important to remember that the radiological activity levels reported can be attributed not only to operations at NRC licensed, DOE-sponsored, and State of California-licensed facilities, but also to external influences such as fallout from nuclear weapon testing and naturally occurring radioactive materials.

These data are:

- Soil gross radioactivity data presented in Table 7
- Soil plutonium radioactivity data presented in Table 8
- Vegetation radioactivity data presented in Table 9
- SSFL Site - Domestic water radioactivity data presented in Table 10
- Bell Creek and Rocketdyne site retention pond radioactivity data presented in Table 11
- Ambient air radioactivity data presented in Table 12
- Ambient air radioactivity data presented in Table 13.

TABLE 7

SOIL RADIOACTIVITY DATA - 1983

Area	Activity	Number of Samples	Gross Radioactivity (μCi/g) Annual Average Value and Dispersion	Gross Radioactivity (μCi/g) Maximum Observed Value[a] and Month Observed
Onsite (monthly)	α	144	$(0.61 \pm 0.19)\ 10^{-6}$	1.13×10^{-6} (March)
	β	144	$(24.2 \pm 2.0)\ 10^{-6}$	29.7×10^{-6} (October)
Offsite (quarterly)	α	48	$(0.59 \pm 0.18)\ 10^{-6}$	1.11×10^{-6} (January)
	β	48	$(23.0 \pm 2.8)\ 10^{-6}$	27.8×10^{-6} (April)

[a]Maximum value observed for single sample.

TABLE 8

SOIL PLUTONIUM RADIOACTIVITY DATA - 1983

Sample Location	22 June 1983 Survey Results		7 December 1983 Survey Results	
	Pu^{238} (μCi/g)	$Pu^{239} + Pu^{240}$ (μCi/g)	Pu^{238} (μCi/g)	$Pu^{239} + Pu^{240}$ (μCi/g)
S-56	$(0.0 \pm 1.7)\ 10^{-9}$	$(2.6 \pm 3.5)\ 10^{-9}$	$(1.3 \pm 1.1)\ 10^{-9}$	$(14.4 \pm 3.9)\ 10^{-9}$
S-57	$(1.4 \pm 1.9)\ 10^{-9}$	$(2.4 \pm 2.4)\ 10^{-9}$	$(0.3 \pm 0.6)\ 10^{-9}$	$(3.8 \pm 2.2)\ 10^{-9}$
S-58	$(1.5 \pm 4.3)\ 10^{-9}$	$(10.6 \pm 8.6)\ 10^{-9}$	$(0.2 \pm 0.5)\ 10^{-9}$	$(4.3 \pm 2.1)\ 10^{-9}$
S-59	$(1.7 \pm 2.2)\ 10^{-9}$	$(1.1 \pm 2.0)\ 10^{-9}$	$(0.5 \pm 0.7)\ 10^{-9}$	$(8.1 \pm 3.0)\ 10^{-9}$
S-60	$(0.5 \pm 1.5)\ 10^{-9}$	$(3.2 \pm 2.8)\ 10^{-9}$	$(0.0 \pm 0.2)\ 10^{-9}$	$(1.7 \pm 1.0)\ 10^{-9}$
S-61*	$(0.0 \pm 1.8)\ 10^{-9}$	$(7.1 \pm 5.3)\ 10^{-9}$	$(0.3 \pm 0.5)\ 10^{-9}$	$(6.8 \pm 2.8)\ 10^{-9}$

*Offsite location

TABLE 9

VEGETATION RADIOACTIVITY DATA - 1983

Area	Activity	Number of Samples	Gross Radioactivity (μCi/g) Dry Weight: Annual Average Value and Dispersion	Gross Radioactivity (μCi/g) Ash: Annual Average Value and Dispersion	Ash: Maximum Value[a] and Month Observed	Percent of Samples With Activity < MDL[b]
Onsite (monthly)	α	144	$(0.04 \pm 0.04)\ 10^{-6}$	$(0.18 \pm 0.14)\ 10^{-6}$	0.91×10^{-6} (January)	40
	β	144	$(29.9 \pm 9.6)\ 10^{-6}$	$(149.3 \pm 42.4)\ 10^{-6}$	240.8×10^{-6} (May)	0
Offsite (quarterly)	α	48	$(0.05 \pm 0.07)\ 10^{-6}$	$(0.24 \pm 0.28)\ 10^{-6}$	1.51×10^{-6} (January)	38
	β	48	$(28.3 \pm 9.8)\ 10^{-6}$	$(142.7 \pm 47.0)\ 10^{-6}$	226.6×10^{-6} (July)	0

[a]Maximum value observed for single sample
[b]Minimum detection level: 0.12×10^{-6} μCi/g alpha; 0.36×10^{-6} μCi/g beta (ash).

TABLE 10

DOMESTIC WATER RADIOACTIVITY DATA - 1983

Area	Activity	Number of Samples	Gross Radioactivity (μCi/ml): Average Value and Dispersion	Gross Radioactivity (μCi/ml): Maximum Value[a] and Month Observed
ESG-De Soto (monthly)	α	12	$(0.34 \pm 0.23)\ 10^{-9}$	0.88×10^{-9} (March)
	β	12	$(3.53 \pm 0.97)\ 10^{-9}$	5.1×10^{-9} (March)
ESG-SSFL (monthly)	α	24	$(0.12 \pm 0.13)\ 10^{-9}$	0.41×10^{-9} (July)
	β	24	$(3.00 \pm 0.60)\ 10^{-9}$	4.45×10^{-9} (October)

[a]Maximum value observed for single sample

TABLE 11

BELL CREEK AND ROCKETDYNE SITE RETENTION POND RADIOACTIVITY DATA - 1983

Area (Monthly)	Activity	Number of Samples	Gross Radioactivity Concentration			
			Average Value and Dispersion	Maximum Value[a] and Month Observed	Percent of Guide[b]	Percent of Samples With Activity < MDL[c]
Bell Creek mud no. 54 (µCi/g)	α	12	$(0.54 \pm 0.17)\ 10^{-6}$	0.90×10^{-6} (August)	NA	0
	β	12	$(23.4 \pm 2.5)\ 10^{-6}$	30.0×10^{-6} (August)	NA	0
Pond R-2A mud no. 55 (µCi/g)	α	12	$(0.76 \pm 0.15)\ 10^{-6}$	1.00×10^{-6} (March)	NA	0
	β	12	$(24.4 \pm 1.6)\ 10^{-6}$	27.2×10^{-6} (March)	NA	0
Bell Creek vegetation no. 54 (µCi/g-ash)	α	12	$(0.12 \pm 0.10)\ 10^{-6}$	0.39×10^{-6} (April)	NA	58
	β	12	$(136.2 \pm 31.9)\ 10^{-6}$	175.6×10^{-6} (March)	NA	0
Bell Creek vegetation no. 54 (µCi/g dry weight)	α	12	$(0.02 \pm 0.01)\ 10^{-6}$	0.04×10^{-6} (July)	NA	58
	β	12	$(24.4 \pm 8.9)\ 10^{-6}$	41.8×10^{-6} (June)	NA	0
Bell Creek water no. 16 (µCi/ml)	α	12	$(0.08 \pm 0.12)\ 10^{-9}$	0.39×10^{-9} (December)	d	92
	β	12	$(3.30 \pm 0.6)\ 10^{-9}$	4.2×10^{-9} (August)	d	0
Pond water no. 6 (µCi/ml)	α	12	$(0.12 \pm 0.11)\ 10^{-9}$	0.27×10^{-9} (August)	d	83
	β	12	$(3.57 \pm 0.92)\ 10^{-9}$	4.80×10^{-9} (September)	d	0
SSFL pond R-2A water no. 12 (µCi/ml)	α	12	$(0.13 \pm 0.12)\ 10^{-9}$	0.35×10^{-9} (December)	d	67
	β	12	$(4.44 \pm 1.84)\ 10^{-9}$	9.15×10^{-9} (August)	d	0

[a]Maximum value observed for single sample
[b]Guide: 5×10^{-6} µCi/ml alpha, 3×10^{-7} µCi/ml beta; 10 CFR 20 Appendix B, CAC 17, DOE Order 5480.1
[c]Minimum detection level: 0.23×10^{-9} µCi/ml alpha; 0.64×10^{-9} µCi/ml beta
NA — not applicable, no Guide value having been established.
[d]Activity essentially the same as local domestic supply water.

TABLE 12

AMBIENT AIR RADIOACTIVITY DATA - 1983

Site Location (Continuous)	Activity	Number of Samples	Average Value and Dispersion	Maximum Value[a] and Date Observed	Percent of Guide[b]	Percent of Samples With Activity <MDL
De Soto Onsite (μCi/ml)	α	644	$(2.4 \pm 3.8)\ 10^{-15}$	6.0×10^{-14} (08/11)	0.08	88[c]
	β		$(2.6 \pm 2.1)\ 10^{-14}$	1.3×10^{-13} (10/19)	0.009	18[d]
SSFL Onsite (μCi/ml)	α	1639	$(0.9 \pm 5.4)\ 10^{-15}$	2.4×10^{-14} (07/03)	1.5	95[c]
	β		$(2.3 \pm 1.7)\ 10^{-14}$	1.8×10^{-13} (08/29)	0.08	26[d]
SSFL sewage treatment plant Offsite (μCi/ml)	α	359	$(1.4 \pm 3.2)\ 10^{-15}$	2.0×10^{-14} (09/02)	2.3	95[c]
	β		$(2.5 \pm 1.3)\ 10^{-14}$	2.8×10^{-13} (10/07)	0.08	21[d]
SSFL Control Center Offsite (μCi/ml)	α	330	$(1.0 \pm 2.5)\ 10^{-15}$	1.1×10^{-14} (07/25)	1.7	97[c]
	β		$(2.5 \pm 2.1)\ 10^{-14}$	1.7×10^{-13} (04/25)	0.08	22[d]

[a]Maximum value observed for single sample

[b]Guide: De Soto site: 3×10^{-12} μCi/ml alpha, 3×10^{-10} μCi/ml beta, 10 CFR 20 Appendix B. SSFL site: 6×10^{-14} μCi/ml alpha, 3×10^{-11} μCi/ml beta; 10 CFR 20 Appendix B, CAC 17, and DOE Order 5480.1A

[c]MDL = 6.4×10^{-15} μCi/ml alpha

[d]MDL = 1.3×10^{-14} μCi/ml beta.

TABLE 13

DE SOTO AND SSFL SITES - AMBIENT RADIATION DOSIMETRY DATA - 1983

TLD Location	Quarterly Exposure (mR)				Annual Exposure (mR)	Equivalent Exposure at 1000 ft ASL	
	Q-1	Q-2	Q-3	Q-4		(mR)	(μR/h)
1. De Soto	29	29	28	27	113	115	13
2. De Soto	28	27	26	25	106	108	12
3. De Soto	28	24	26	25	103	105	12
4. De Soto	32	30	31	30	123	125	14
5. De Soto	28	26	28	24	106	108	12
6. De Soto	31	28	32	28	119	121	14
7. De Soto	27	25	27	24	103	105	12
8. De Soto	31	27	27	24	109	111	13
Mean value	29	27	28	26	110	112	13
1. SSFL	31	a	31	29	121	109	12
2. SSFL	31	32	32	31	126	114	13
3. SSFL	33	32	33	33	131	119	14
4. SSFL	30	32	37	31	130	117	13
5. SSFL	30	30	31	29	120	107	12
6. SSFL	26	27	29	26	108	97	11
7. SSFL	38	34	35	29	136	124	14
Mean value	31	31	33	31	126	112	13
1. Offsite	29	30	31	27	117	120	14
2. Offsite	28	27	28	25	108	106	12
3. Offsite	30	27	27	25	109	111	13
4. Offsite	30	31	33	27	121	119	14
5. Offsite	32	31	32	28	123	124	14
Mean value	30	29	30	26	115	116	13

[a]Missing dosimeter; annual exposure based on data for three quarters.

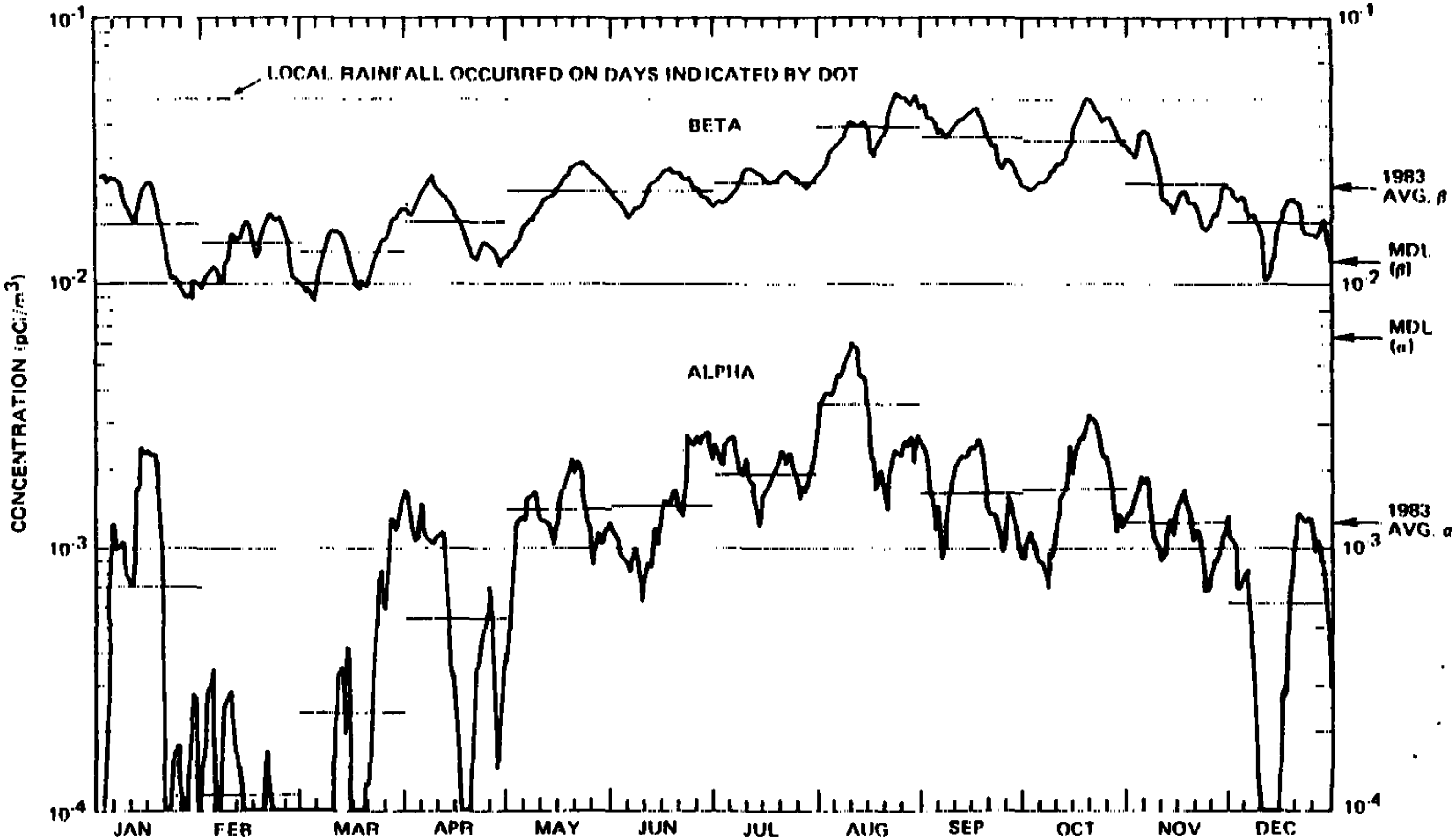

Figure 4. Average Long-Lived Airborne Radioactivity at the De Soto and Santa Susana Field Laboratories Sites - 1983

V. UNUSUAL EVENTS

There were several unusual events at faciities involving radiation or radioactive materials. These events are summarized below.

A. REPORTABLE INCIDENTS

None.

B. NONREPORTABLE INCIDENTS

<u>February 7, 1983</u>

While disassembling a balance in glove box 3A in the NMDF, an employee accidentally jammed a small screwdriver through the box glove and surgeon's glove, resulting in a puncture wound at the base of his left index finger. He immediately notified Radiation and Nuclear Safety. A survey of the wound and finger indicated approximately 3500 d/m α. His surgeon's glove indicated approximately 30,000 d/m α. Blood samples of his finger indicated 414 d/m α. The area was flushed with clean water and encouraged to bleed. The wound area was allowed to dry and a second blood sample taken which indicated 39 d/m α. Bleeding was again encouraged and the area washed with Bactine. A third blood sample indicated 15 d/m α. A smear from around the wound area indicated 4 d/m α. A wound monitor probe did not indicate any activity. A survey of the wound area with a FM-5 count rate meter with alpha scintillation probe indicated significant alpha activity still present. The employee was then taken to T020 for an In-Vivo scan on the Canberra multichannel (MCA) analyzer with Ge detector and for further decon. The area around the wound was decontaminated to background level and the wound irrigated and swabbed with Bactine. A 1000-second count was performed with the Ge detector and analyzed. A significant photopeak was observed at 59.5 keV with 1216 counts per 1000 seconds. No other photopeaks were observed. A background count indicated 450 counts per 1000 seconds for that energy region. Calibration of a similar detector and system for Am-241 at De Soto indicated an efficiency of 1 gamma detected in 50 decays (2%). The count data thus converts to 2300 d/m of Am-241. The plutonium isotopic mixture for the active material involved calculates to a 7:1 ratio of plutonium (alpha) to Am-241 activity. Thus, the total activity in the wound was assumed to be about 7 nCi or about 20% of a systemic body-burden. The Medical Department was informed of the incident and requested to provide medical support at SSFL so that the MCA could be used to monitor the wound cleaning progress. An area was set up at the SSFL First Aid Station and debridement of the wound site was performed. All equipment and waste were collected, and the surgery area surveyed clean upon completion. A second MCA scan was performed which indicated 750 counts per 1000 seconds in the energy region-of-interest with a 450 counts per 1000 seconds background, equivalent to approximately 2.8 nCi of plutonium plus americium. No further effort was made to remove active material from the wound at this time. Additional radioactive material was surgically removed from the wound site on March 16, 1983.

The employee was placed on a total bioassay (urine) collection program. The bioassay urinalysis results showed that a very rapid clearance of a small soluble component of the deposited plutonium occurred within 48 h of the incident. All subsequent urinalysis results were below the minimum detection level. A fecal specimen voided 11 days after the incident was negative for plutonium which indicates that the deposited plutonium was being retained at the wound site with no translocation and elimination through the liver and associated elimination pathways. The UR urinalysis results for March 13 and 29, and April 3 specimens are not significant since these values are below the minimum level of detection for the analysis and were reported as a "zero" value by the bioassay laboratory. The In-Vivo data showed that the deposited activity remained at the wound site until surgically removed on March 16, 1983. A post-surgery scan result indicated that between 90 and 100% of the active material was removed. No further bioassay services are planned for the employee as a result of this incident.

March 27, 1983

The Protective Services Control Center notified the RIHL health physicist of a stack monitor alarm. On investigation, the stack monitor recorder chart showed an increasing count trend for about 1 h to 26 dpm. The filter was removed and counted for alpha radioactivity. The results was 7 dpm α immediate and 3 dpm α after 19 h decay. The final count (>72 h decay) on this stack sample for the facility release record was 0 dpm for a 100-min counting time. The event is attributed to the effect of atmospheric inversion conditions on naturally occurring airborne radioactivity. The facility exhaust monitoring system is sensitive to inversion-induced airborne radioactivity increases due to bypass air intakes at the stack which allows unfiltered ambient air to be mixed into the discharge airstream which is sampled at the stack exit point. The alarm was reset and remained at normal levels thereafter.

July 15, 1983

An incident involving irradiated plutonium fines and fission product radioactivity occurred at the Decon Room 4 alpha glove box in the RIHL.

While preparing to replace a leaded glove with a plastic bag in the SEFOR glove box in Decon Room 4, the glove box operator checked his gloved hands and detected alpha contamination. He proceeded to the decon room doorway to request support and plastic bags to collect waste. At this point, the air monitor at the glove box alarmed. The Radiation and Nuclear Safety representative and the operator decided to complete the change-out. They donned respiratory protection with the operator additionally putting on a hood. They reentered the area. An alpha survey indicated contamination only on the glove port, so they proceeded with the change-out. During this operation, the inside glove keeper rings, installed as an additional precaution because of the stiffness of the leaded glove, snapped off. A quick survey indicated gross alpha contamination on the hood, mask, lab coat of the operator (>50,000 c/m), as well as the glove port and the immediate surrounding area.

Evacuation was ordered and additional support requested. The area in the service gallery in front of Decon Room 4 was covered with plastic, and the remainder of the service gallery was made a shoe-cover area and secured.

Wearing respiratory protection, gloves, etc., two Radiation and Nuclear Safety representatives carefully stripped contaminated articles from the operator, leaving his full-face respirator until last. Activity levels up to 50,000 c/m alpha were detected, with 5,000 c/m alpha on the right lower part of the pant leg. The gross activity was removed (to less than 200 c/m alpha) and his mask removed. Decon efforts were satisfactorily completed, and the operator was released to the showers.

Another employee who was fully suited in coveralls and full facepiece respirator entered the decon room and completed the glove change-out. The area was surveyed and completely deconned to background levels.

Nasal smears of involved personnel indicated no significant radioactivity.

The breathing zone air sample data revealed 161 MPC-hours with the major portion of the activity coming while personnel were wearing respiratory protection and particularly during the recovery operation. The respiratory protection factor for a full facepiece respirator with ultra filter is 50, thus reducing the exposure to 3.2 MPC-hours.

Urine and fecal samples were requested and submitted for analyses. The urinalysis results were negative; the fecal specimen was lost at the laboratory during processing.

October 11, 1983

A spill of liquid containing about 30 μCi of fission and activation product radioactivity plus a small amount of "TRU" (alpha activity) occurred at the face of cell 3 at the RIHL.

The incident occurred while distilled water was being pumped into cell 3 via a cell face through-tube to aid in neutralizing and solidifiction of SEFOR fuel cladding electropolish acid. The in-line manual antisiphon valve was not closed when the pump was shut off, and acid solution siphoned back into a 5-gallon-capacity, blue-colored (opaque) container of clean distilled water which became contaminated to an activity concentration of about 2 to 6 μCi/ml.

A second pump from cell 4 was concurrently being tested for use in cell 3. The pump supply and return lines were placed in the assumed "clean" container of distilled water. The cell 4 pump functioned normally, the supply and return lines were removed, and the pump was taken into the shop area to shorten the electric power cable. Water was spilled at the disconnect point, onto the work table floor area, and at the shop work bench.

A cell operator working at cell 3 face had completed another operation, performed a routine survey of his hands, and discovered widespread radioactive contamination. Radiation and Nuclear Safety personnel were called, and a preliminary radiation survey indicated the presence of contamination on the floor and on the shoes of several persons. A P.A. system announcement was made for everyone to remain in place until monitored. No persons were permitted to enter or leave the building during this time.

Subsequent surveys of the RIHL indicated that radioactive contamination had spread throughout the building except for the Equipment Room, Generator Room, Battery Room, and their access aisles. Other areas had contamination levels from 6,000 dpm beta to greater than 450,000 dpm beta when monitored with pancake-type G-M detectors. No contamination was found outside of the RIHL. The radioactivity detected by the facility surveys was primarily due to the presence of Cs-137 (~95%) and Co-60 (~5%) with no alpha radiation detected. A sample from the distilled water reservoir indicated a small amount of Am-241, indicative of the presence of "TRU" in the water.

All persons in the building were monitored, and 11 had contaminated shoes. Three persons had contaminated clothing and hands. Contamination levels ranged up to 100,000 cpm beta. Seven pairs of shoes were decontaminated to background level and released. Four pairs of shoes, one pair of pants, two shirts, and one jacket were disposed of as radioactive waste. Two persons' hands were cleaned to background level, and one person was released with 250 cpm beta fixed-skin contamination which was successfully removed the following day.

Decontamination and surveys of the suspect and low-level contamination areas were done with Maslin-cloth mops. The lightly oiled mop was gamma scanned after use to identify radioisotopes present. Areas of higher contamination were hand wiped, stripped, and/or disposed of as radioactive waste. A section of floor tile in front of cell 3 was replaced as was some front office area carpeting.

Ninety percent of the facility was released for normal use by 1800 on October 11, with the remainder released by 1300 on the following day. No significant airborne radioactivity was detected, and all nasal wipes were negative for radioactive contamination. Bioassay samples collected on October 17, from persons who had skin contamination, indicated that no detectable internal exposure had occurred.

5026Y/jbv

VI. SUMMARY/TRENDS - EXPOSURE, EFFLUENTS

A. PERSONNEL EXPOSURES

Personnel exposures are summarized by year in the following table:

	Number of Persons in Exposure Range (rem)											
Year	0 0.1	0.1 0.25	0.25 0.5	0.5 0.75	0.75 1.0	1.0 2.0	2.0 3.0	3.0 4.0	4.0 5.0	Total Exposed Persons	Group Dose (Person rems)	Average Dose (rems)
1983	281	9	5	4	5	13	8	2	17	344	138	0.402
1982	349	29	8	3	6	15	4	7	8	429	116	0.271
1981	192	55	13	4	6	4				274	33	0.121
1980	357	39	10	3	5	9	3			426	56*	0.131*
1979	347	39	19	10	4	15	8	2		444	91*	0.204*
1978	432	60	18	16	4	18	9	1	1	559	110*	0.197*
1977	340	31	29	7	5	11	13			436	91*	0.209*
1976	295	38	17	14	5	9	2			380	59*	0.156*
1975	170	24	12	4	5	6	1	1		223	39*	0.175*

*Determined by use of mid-point of range

Data shown for 1980 and prior years include visitors. Visitor exposures rarely exceed 0.25 rem. Data for 1981, 1982, and 1983 represent occupationally exposed ESG employees. The group dose was calculated exactly for these three years. This results in values that are approximately 10% lower than those calculated by use of the mid point of the exposure ranges.

Exposures during 1983 continued to be considerably greater than in prior years. This resulted from exposures received during decladding the SEFOR fuel. At the end of the decladding itself, additional controls were exerted in order to minimize the total annual exposure.

A formal radiation exposure control program was developed and issued as Health and Safety Procedure No. 37. This procedure established planning guides, action guides, and internal limits significantly below the applicable regulatory limits. These are summarized below.

ESG EXPOSURE GUIDES AND LIMITS
(rems)

Type of Exposure	Planning Guide	Action Guide	Internal Limit	Regulatory Limit
Whole body, head and trunk. Lens of the eye; active blood-forming organs	1/quarter 2/year	2.5/quarter* 2.5/year	2.9/quarter* 3.0/year	3/quarter* 5/year†
Skin	2/quarter 5/year	4.5/quarter 7/year	4.9/quarter 10/year	5/quarter† 15/year†
Hands	2/quarter 5/year	4.5/quarter 7/year	4.9/quarter 10/year	18.75/quarter§
Forearms	2/quarter 5/year	4.5/quarter 7/year	4.9/quarter 10/year	10/quarter† 30/year†
Feet	5/quarter 10/year	10/quarter 30/year	12/quarter 45/year	18.75/quarter§
Bone	2/quarter 5/year	4.5/quarter 10/year	7/quarter 20/year	10/quarter† 30/year†
Other organs	2/quarter 5/year	4.5/quarter 7/year	4.9/quarter 10/year	5/quarter† 15/year†
Internal exposure	5% MPBB	25% MPBB	50% MPBB	100% MPBB

*Exposures above 1.25 rem/quarter are permissible only if the individual's prior radiation exposure history is on file and has been reviewed to determine that the exposure will not exceed an accumulation of 5 (N-18) rems, where N is age in years.
†DOE limit
§NRC and State of California limits

B. AMBIENT (ENVIRONMENTAL) RADIATION EXPOSURE

Ambient (environmental) radiation exposure rates as measured by CaF_2:Mn TLDs and averaged for all locations are shown below.

	Quarterly Dose (mrem)				Annual Dose (mrem)
	Jan-Mar	Apr-Jun	Jul-Sep	Oct-Dec	
1983	30.1	28.9	30.2	27.4	116.6
1982	29.1	30.8	31.8	31.9	123.8
1981	38.2	33.5	35.2	43.9	150.8
1980	35.0	34.4	37.7	49.1	157.3
1979	32.1	38.1	38.0	39.4	147.8
1978	27.3	35.5	33.4	36.6	133.1
1977	24.2	29.2	32.9	30.9	117.5
1976	21.6	24.8	22.5	25.0	93.9
1875	21.3	24.6	26.2	25.4	97.6

The quarterly doses are plotted as a histogram in Figure 5. This graph, and the tabulated annual doses show a clear increase from 1976 to 1980, followed by a decrease for 1981, 1982, and 1983. All data prior to 1982 were obtained using an EG&G TL-3 reader. The 1982 and 1983 data were obtained using a Victoreen Model 2810. This is a new reader, built on the basic design of the TL-3 reader, but with modern electronics and digital adjustments and readout.

The increasing trend (from 1976 to 1980) was also observed in data for the Rocky Flats Plant, the only other DOE facility where the same type dosimeters are used, but not at any other facility. The cause has not been identified, but since the trend exists equally for the De Soto, Santa Susana, and off-site TLDs, at this time it is assumed to be either a true environmental effect, or an artifact of the TLD reading or calibration.

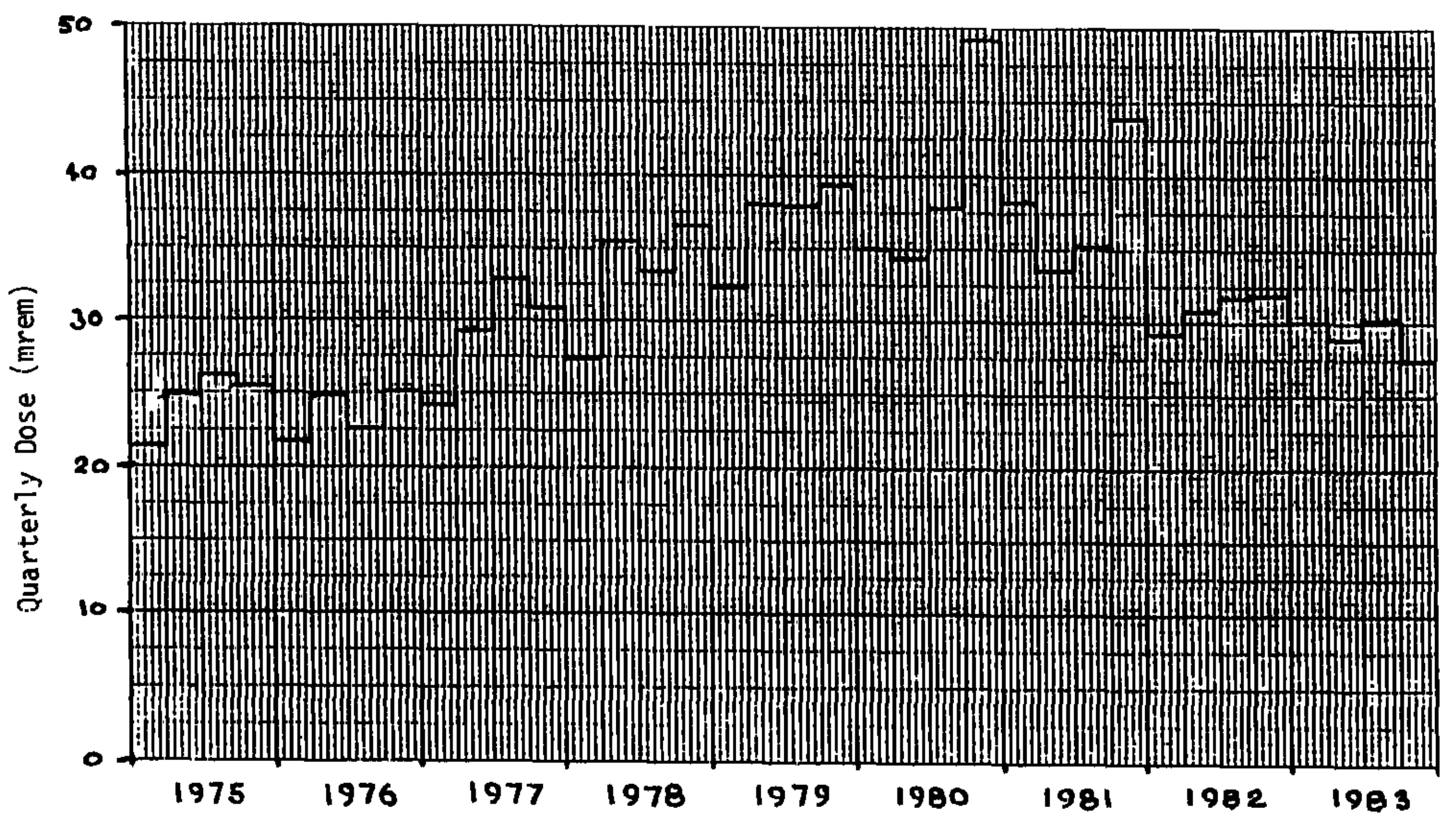

Figure 5. Averaged Quarterly Dose Recorded by Environmental TLDs

C. ATMOSPHERIC EFFLUENT RELEASES

Atmospheric effluent releases are monitored by use of stack samplers at the major facilities. The results are shown below in terms of the total activity released. In some cases, the releases were at concentrations less than the ambient (natural) airborne radioactivity; in others, much of the activity is natural resulting from the use of unfiltered bypass or dilution air in the exhaust system.

A significant change has been made in the manner in which those releases are calculated from the effluent sampling measurements. Prior to 1982, all concentration values less than the minimum detection level (MDL) were set equal to the MDL in calculating the average concentration release. This was done on the basis of DOE requirements. It was recognized that this practice biased the reported results upwards by a considerable amount, and DOE changed its guidance. Now, all measured values, even zeroes and negative ("less than background") values, are used in the calculation.

The major fluctuations observed in the beta activity released from the RIHL is due primarily to changes in the work in the hot cells.

RADIOACTIVITY DISCHARGED TO ATMOSPHERE
(microcuries)

	De Soto		Santa Susana		
	001	004	RIHL	RMDF	NMDF
1983α	52.0	1.1	0.047	0.047	0.08
β	19.0	1.1	1.3	1.1	1.1
1982α	1.2	0.24	0.03	0.024	0.023
β	0.94	1.1	14.0	0.61	1.0
1981α	2.8	0.39	0.069	0.087	0.059
β	2.7	4.1	14.0	4.0	2.0
1980α	5.3	1.0	0.17	0.061	0.082
β	4.3	4.9	17.0	1.7	1.1
1979α	2.1	1.1	0.18	0.085	0.053
β	5.8	5.7	44.0	2.7	0.21
1978α	16.0	0.65	0.13	0.1	0.081
β	5.0	4.3	59.0	11.0	-
1977α	10.0	0.88	0.1	0.11	0.15
β	4.1	7.5	13.0	3.0	-
1976α	64.0	8.1	0.15	0.23	0.15
β	17.0	8.9	5.8	1.1	-
1975α	3.7	5.4	0.15	0.45	0.19
β	2.6	12.0	6700.0*	10.0	-

*Released from burned fuel slug.

VII. ANTICIPATED ACTIVITIES DURING NEXT REPORTING PERIOD (1984)

Buildings 001/004

Completion of the decontamination and decommissioning of all areas involved in special nuclear material operations.

Building 020

Complete cleanup and waste disposal following the SEFOR fuel decladding program. Start decladding EBR-II blanket fuel.

Buildings 021/022

Shipment of declad SEFOR fuel and scrap. Receive, store, and transfer EBR-II blanket fuel for decladding. Storage and transfer of declad SEFOR fuel and scrap.

Building 055

Complete the decontamination of glove boxes and ship for disposal.

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