Rockwell International
Energy Systems Group

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ABSTRACT

Data on employee exposures, bioassay results, effluent releases, in-plant airborne radioactivities, and environmental monitoring for ESG operations during 1982 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.

There was a significant increase in exposures to ionizing radiation due to decladding operations with irradiated SEFOR fuel. This, and a reportable overexposure to one worker's hand, resulted in a major modification to the procedures for Rad-Pac encapsulation and for decladding the SEFOR fuel.

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

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

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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 1981[2][3][4][5][6][7][8] 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. When this occurs, the values are reported unmodified as measured and conservatively may be attributed wholly to licensed activities.

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) Fuel Fabrication - Building 001 and supporting operations in Buildings 001 and 004, De Soto Facility, Canoga Park, California
2) Rockwell International Hot Laboratory (RIHL) - Building 020, Santa Susana Field Laboratories
3) Nuclear Material Development Facility (NMDF) - Building 055, Santa Susana Field Laboratories
4) Radiation 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). Appendix A describes the Personnel Monitoring Program, and Appendix B describes the Analytical Procedure Summary for Bioassay by Urinalysis.

A. FILM/TLD DATA

1. Whole Body Monitoring

Personnel external radiation exposures for the pertinent activities for the year are presented in Table 1, where the number of individuals within a specific annual dose range is shown along with the cumulative percentage of employees within each annual dose range or less and the man-rem contributed by each dose range value. It should also 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). The cumulative log-normal distribution of all those personnel with reported exposures greater than zero is shown in Figure 1.

2. Extremity Monitoring

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 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

TABLE 1

PERSONNEL EXTERNAL (WHOLE BODY) EXPOSURES - 1982

Dose Range (rem)	P (People)	C.P. (Cumulative People)	C.P. (%)	Population - Dose* (Man-rem)
No Measured Exposure	555	555	47.89	0
0.10	521	1076	92.84	26.05
0.10 - 0.25	32	1108	95.60	5.60
0.25 - 0.50	8	1116	96.29	3.00
0.50 - 0.75	4	1120	96.64	2.50
0.75 - 1.0	4	1124	96.98	3.50
1.0 - 2.0	16	1140	98.35	24.00
2.0 - 3.0	4	1144	98.71	10.00
3.0 - 4.0	7	1151	99.31	24.50
4.0 - 5.0	8	1159	100.00	36.00
5.0	0			
			Total	135.15

*The midpoint of each dose range was assumed for the average dose in the calculation of the man-rem.

Figure 2. All but one exposure were below the allowable quarterly limit of 18.75 rem. The highest exposure exceeded the allowable limit, was reported to NRC and the State of California, and is described in Section V.

B. 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 1982, 49 lung scans were made for uranium deposition. Nine of the scans (on 8 different individuals) showed positive results. However, follow-up scans showed elimination of these lung burdens in most cases.

*Helgeson Nuclear Services, Inc., Pleasanton, California

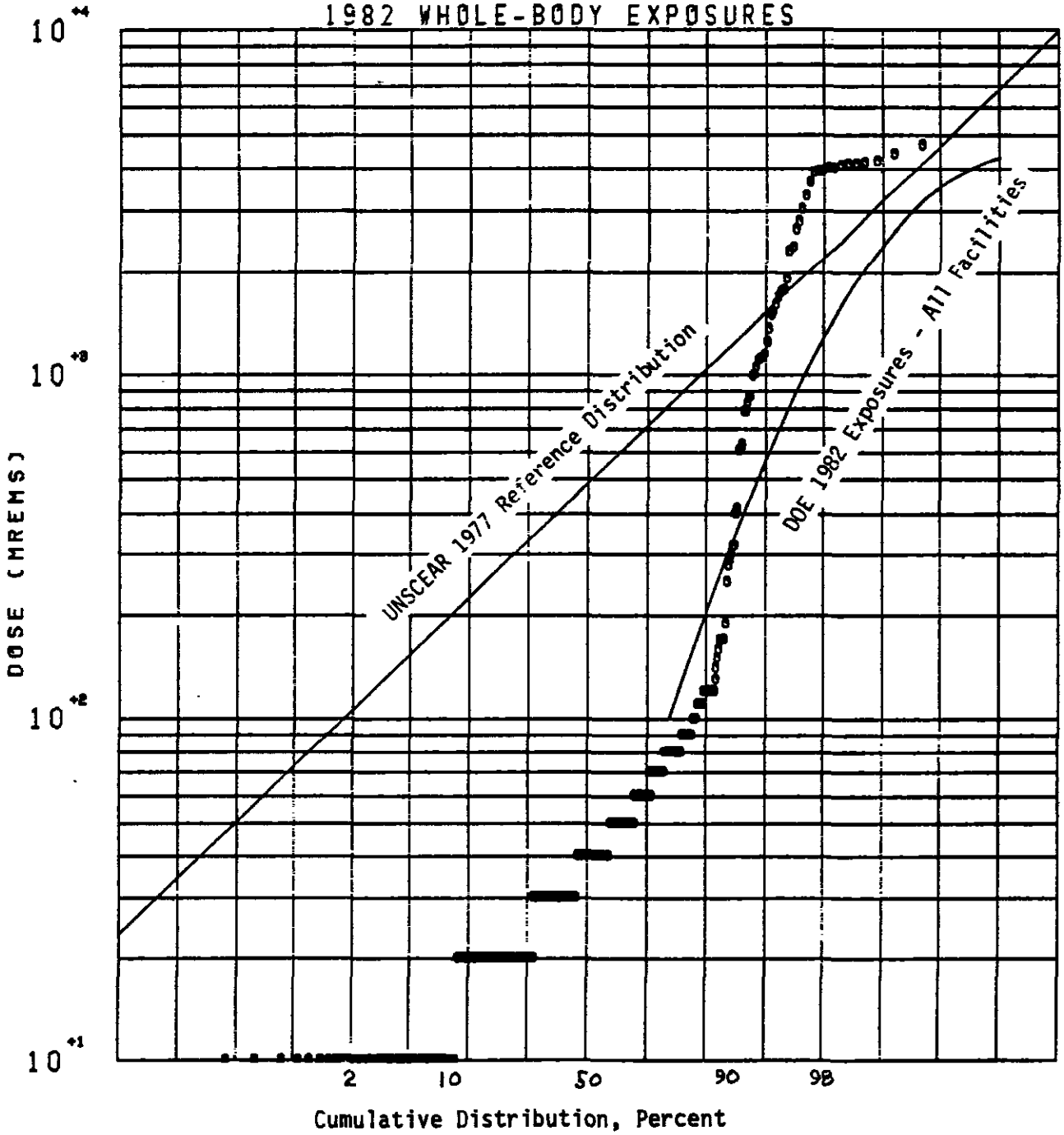

Figure 1. Cumulative Log-Normal Distribution for Whole-Body Radiation Exposures of Occupationally Exposed Individuals in 1982

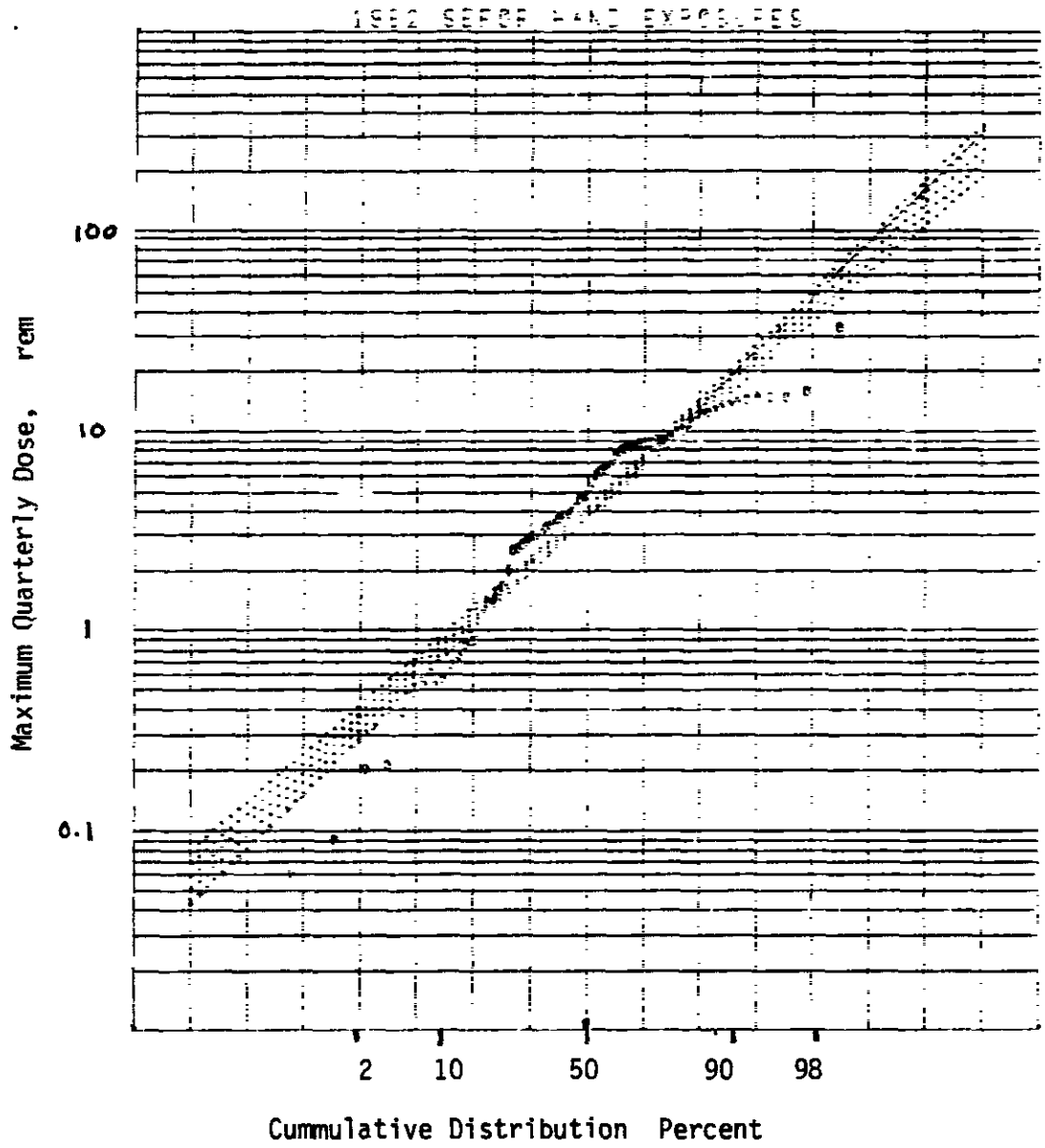

Figure 2. Hand Exposure Values (higher exposed hand from each individual) for SEFOR Project During 1982

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 evaluated in this manner. 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 1982 appears in Table 2, while a detailed listing of the positive results are shown in Table 3. Data on the in-vivo lung scans performed in 1982 also appear in these tables.

The analytical laboratory experienced some difficulty in routinely correcting for K-40 beta activity carried over in high-residue urine samples analyzed for "insoluble" fission products (analysis FP 3A). Results from this analysis are treated as representing Sr-90. Because of the bias toward higher activity caused by the K-40 activity, results from these analyses are therefore interpreted incorrectly as indicating a significant level of Sr-90 activity. The laboratory informed us of this problem on August 5, 1982, and by September had adopted some additional procedures to reduce this occurrence in the future.

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

TABLE 2

SUMMARY OF BIOASSAYS - 1982

Measurement*	Type*	Total Tests	Total Positive Results	Total Individuals With Positive Results
U	UF	155	6	5
U	UR	157	12	2
GA	1A	0	0	0
GA	1B	0	0	0
GA	2B	12	1	1
GB	H	0	0	0
PU	A	22	2	1
FP	1	0	0	0
FP	3A	174	32	22
FP	3B	171	4	4
U	IVLC	49	9	8
Sr-90	Sr-90	1	0	0
Cs-137	TBC	0	0	0
H-3	H-3	1	0	0
Th	Th	0	0	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
H-3 = Tritium
Th = Thorium
H = High Level
(For a discussion of specific analytical techniques employed, as identified by "TYPE," see Appendix B)

TABLE 3

POSITIVE BIOASSAY RESULT SUMMARY - 1982

H&S Number	Sample Date	Analysis Type*	Results		Assumed Specific Radionuclide See Note 1	Assumed Critical Nuclide Equivalent MPBB (%)
			Per Vol. Anal. (dpm)	Per 1500 ml-day (dpm)		
4892	052782	FP3A	5.10	38.2	Sr-90	8.0
4892	070281	FP3A	2.6	-		
4382	062182	FP3A	6.60	49.5	Sr-90	10.3
4382	100482	FP3A	2.5	-		
3758	031482	FP3A	6.0	45.00	Sr-90	9.4
3758	041782	FP3B	12.40	93.00	Cs-137	0.01
3758	052782	FP3A	0.6	-		
3758	052782	FP3B	1.6	-		
4299	041582	FP3A	5.60	42.00	Sr-90	9.4
	Terminated					
3892	062382	FP3A	18.10	135.7	Sr-90	28.3
4253	030982	FP3A	32.60	244.50	Sr-90	50.9
4253	030982	FP3B	9.70	72.75	Cs-137	0.01
4253	041982	FP3A	2.5	-		
4253	041982	FP3B	5.1	-		
4253	042582	FP3A	13.90	104.2	Sr-90	21.7
4253	081982	FP3A	1.5	-		
4390	062782	FP3A	6.40	48.0	Sr-90	10.0
4390	071482	FP3A	3.1	-		
2040	011282	UR	1.24	9.3	EU	4.2
2040	011282	UF	0.00	0.4	U	0.4
2040	011282	UR	0.77	5.7	EU	2.6
2040	020782	UR	0.79	5.92	EU	2.7
2040	021582	UR	1.17	8.77	EU	4.0
2040	030882	UR	0.62	4.65	EU	2.1
2040	031782	UR	0.92	6.90	EU	3.1
2040	032882	UR	1.03	7.73	EU	3.5
2040	040582	UR	0.54	4.05	EU	1.8
2040	041182	UR	0.66	4.95	EU	2.2
2040	042582	UR	0.32	-		
2040	042582	UR	0.00	-		
2040	051982	IVLC	0.ug	-		
1281	101582	IVLC	38.ug	-	EU	15.4
	Terminated					

TABLE 3

POSITIVE BIOASSAY RESULT SUMMARY - 1982
(Continued)

H&S Number	Sample Date	Analysis Type*	Results Per Vol. Anal. (dpm)	Results Per 1500 ml-day (dpm)	Assumed Specific Radionuclide See Note 1	Assumed Critical Nuclide Equivalent MPBB (%)
4163	042182	FP3A	38.30	287.2	Sr-90	59.8
4163	042182	Sr-90	1.3	-		
4163	071482	FP3A	5.10	38.2	Sr-90	8.0
4163	071482	FP3A	5.20	39.0	Sr-90	8.1
4163	081282	FP3A	5.50	41.2	Sr-90	8.6
4163	101982	FP3A	2.9	-		
4486	032682	FP3A	13.60	102.00	Sr-90	21.2
4486	032682	FP3A	22.70	170.25	Sr-90	35.5
	Terminated					
4160	042382	FP3A	6.50	48.7	Sr-90	10.2
4160	042382	FP3A	13.40	100.5	Sr-90	20.9
4160	060182	FP3A	1.9	-		
4440	091782	UF	0.10	10.7	U	10.7
4440	091782	UR	44.80	239.00	EU	108.6
4440	092082	UF	0.16	7.9	U	7.9
4440	092082	UR	21.90	53.7	EU	24.4
4440	101082	UR	0.76	5.7	EU	2.6
4440	101582	IVLC	0.ug	-		
4440	102982	UF	0.0	-		
4440	102982	UR	0.28			
4572	020982	IVLC	63.ug	-	EU	25.7
4572	051982	IVLC	0.ug	-		
4650	051982	IVLC	38.ug	-	EU	15.4
4650	101582	IVLC	0.ug	-		
3912	062182	FP3A	8.70	65.2	Sr-90	13.6
3912	110282	FP3A	1.2	-		
3912	121382	UF	0.00	0.9	U	0.9
3703	031182	GA2	2.40	18.00	Ra-226	180.0
3703	042082	GA2	0.0	-		

TABLE 3

POSITIVE BIOASSAY RESULT SUMMARY - 1982
(continued)

H&S Number	Sample Date	Analysis Type*	Results: Per Vol. Anal. (dpm)	Results: Per 1500 ml-day (dpm)	Assumed Specific Radionuclide See Note 1	Assumed Critical Nuclide Equivalent MPBB (%)
4175	020982	IVLC	67.ug	-	EU	27.3
4175	051982	IVLC	0.ug	-		
4565	062182	FP3A	8.70	65.2	Sr-90	13.6
1639	062282	FP3A	5.60	42.0	Sr-90	8.8
3459	072782	FP3B	24.80	186.0	Cs-137	0.03
3459	091482	FP3B	1.7	-		
4535	062982	UF	0.00	0.4	U	0.4
4535	100582	UF	0.0	-		
4303	031182	FP3A	5.00	37.50	Sr-90	7.8
4303	04282	FP3A	0.4	-		
4137	062082	FP3A	5.0	37.5	Sr-90	7.8
4137	062783	FP3A	0.8	-		
2729	101582	IVLC	38.ug	-	EU	15.4
2729	022583	IVLC	0.ug	-		
0000	071582	FP3A	6.30	47.2	Sr-90	9.8
0000	102082	FP3A	3.7	-		
4336	020982	IVLC	33.ug	-	EU	13.5
4336	060782	UR	0.0	-		
3453	060982	FP3A	10.2	76.5	Sr-90	15.9
3453	070282	FP3A	4.20	31.5	Sr-90	6.6
	Terminated					
4268	120382	UF	0.00	0.6	U	0.6
	Terminated					
4651	051982	IVLC	35.ug	-	EU	14.2
4651	101582	IVLC	0.ug			
1547	062182	FP3A	18.10	135.7	Sr-90	28.3
1547	080282	FP3A	0.0	-		

TABLE 3

POSITIVE BIOASSAY RESULT SUMMARY - 1982
(continued)

H&S Number	Sample Date	Analysis Type*	Results Per Vol. Anal. (dpm)	Results Per 1500 ml-day (dpm)	Assumed Specific Radionuclide See Note 1	Assumed Critical Nuclide Equivalent MPBB (%)
4891	041882	FP3A	7.70	57.7	Sr-90	12.0
4891	041882	FP3A	10.10	75.7	Sr-90	15.8
4891	052782	FP3A	0.8	-		
4891	101382	PUA (fecal)	6.29	33.3	Pu-239	27.4
4891	101482	PUA (fecal)	1.40	9.3	Pu-239	7.7
4891	110482	PUA	0.0	-		
4187	051982	IVLC	24.ug	-	EU	9.8
4187	101582	IVLC	35.ug	-	EU	14.2
4187	060883	IVLC	0.ug	-		

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

ug: Microgram
MDL: Minimum Detectable Level
MPBB: Maximum Permissible Body Burden
**MPLB: Maximum Permissible Lung Burden
TBC: Total Body Count
(For a brief description of the specific analytical techniques, see Appendix B)

(3A is predominantly Sr-90; 3B is predominatly Cs-137)

Note 1: During this period, the radioanalytical laboratory experienced difficulties in separating naturally occurring K-40 activity from possible Sr-90 activity, resulting in frequent overestimates of the assumed Sr-90 activity.

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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. The equipment used for these measurements is maintained by CRIS (see Appendix C) to assure its proper maintenance and calibration.

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 4.

TABLE 4
LOCATION BADGE RADIATION EXPOSURE - 1982

Facility	Calendar Quarter Q1	Q2	Q3	Q4
	Average Exposure Rate (mR/h)			
	Maximum Exposure Rate (mR/h)			
Fuel Fabrication (C001)	0.03 / 0.15	0.03 / 0.17	0.03 / 0.21	0.03 / 0.21
RIHL (T020)	0.06 / 0.46	0.11 / 0.65	0.07 / 0.60	0.16 / 1.21
NMDF (T055)	0.02 / 0.05	0.02 / 0.05	0.02 / 0.06	0.02 / 0.02
RMDF	6.7 / 36.3	7.6 / 42.2	2.0 / 9.2	0.67 / 1.4

The maximum exposure in the NMDF was associated with some packaged waste stored in the rear air lock. This waste was shipped out in the fourth quarter, bringing all areas in the NMDF down to nearly background levels of exposure. The high exposure in the fourth quarter in the RIHL was due to some hot samples stored in the fume hood/glove box in the chem lab.

Location badges placed at the RMDF are generally in the more highly radioactive areas and are not typical of the occupied areas. However, considerable improvement was made in reducing the exposure rate in the Packaging Room and some other areas during this year.

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 1982 is given in Table 5.

TABLE 5

INTERIOR AIR SAMPLE SUMMARY - 1982

Area	Sample	Airborne Activity Concentration (μCi/ml) Calendar Quarter Q1	Q2	Q3	Q4	MPC
Fuel Fab	Lapel Max Week*	4×10^{-11}	1×10^{-10}	4×10^{-11}	5×10^{-11}	1×10^{-10}
(C001)	Average*	2×10^{-13}	2×10^{-12}	2×10^{-12}	8×10^{-13}	1×10^{-10}
	Stationary Max Week	1×10^{-10}	3×10^{-11}	5×10^{-11}	2×10^{-11}	1×10^{-10}
	Average	8×10^{-13}	8×10^{-12}	8×10^{-12}	1×10^{-12}	1×10^{-10}
RIHL	Unposted α	-	-	-	1×10^{-15}	2×10^{-12}
	β	$<3 \times 10^{-13}$	$<3 \times 10^{-13}$	$<3 \times 10^{-13}$	1×10^{-14}	1×10^{-9}
	Posted α	-	-	-	1×10^{-14}	2×10^{-12}
	β	$<8 \times 10^{-13}$	$<8 \times 10^{-13}$	$<8 \times 10^{-13}$	1×10^{-14}	1×10^{-9}
	Maximum α	-	-	-	4×10^{-14}	2×10^{-12}
	β	1×10^{-10}	1×10^{-10}	1×10^{-10}	2×10^{-11}	1×10^{-9}
NMDF	Stationary Max Week	4×10^{-14}	4×10^{-14}	4×10^{-14}	4×10^{-14}	1.3×10^{-12}
	Average	2×10^{-15}	2×10^{-15}	2×10^{-15}	2×10^{-15}	1.3×10^{-12}

*Adjusted for respirator protection factor:

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

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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. Equipment used for measurement of radioactivity in these effluents is maintained by CRIS (see Appendix C) to assure proper operation and calibration.

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 1982 edition of that report[9] for atmospherically discharged and liquid effluents for the facilities identified above is presented in Tables 6 and 7, respectively.

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TABLE 6

ATMOSPHERIC EMISSIONS TO UNRESTRICTED AREAS - 1982

Building	Approximate Emissions Volume (ft^3)	Activity Monitored	Approximate Minimum Detection Level ($\mu Ci/ml$)	Annual Average Concentration ($\mu Ci/ml$)	Sampling Period Maximum Observed Concentration ($\mu Ci/ml$)	Total Radioactivity Released (Ci)	Percent of Guide[a]	Percent of Samples With Activity <MDL
001 De Soto	1.4×10^{10}	α	1.6×10^{-16}	3.0×10^{-15}	1.5×10^{-14}	1.2×10^{-6}	0.10	35
		β	5.4×10^{-16}	2.4×10^{-15}	1.7×10^{-14}	9.4×10^{-7}	0.0008	33
004 De Soto	2.4×10^{10}	α	2.1×10^{-16}	3.4×10^{-16}	2.5×10^{-15}	2.4×10^{-7}	0.01	46
		β	7.2×10^{-16}	1.5×10^{-15}	1.3×10^{-14}	1.1×10^{-6}	0.0005	43
020 SSFL	1.0×10^{10}	α	0.9×10^{-16}	1.0×10^{-16}	4.4×10^{-16}	3.1×10^{-8}	0.16	68
		β	3.0×10^{-16}	4.7×10^{-14}	9.0×10^{-13}	1.4×10^{-5}	0.16	0
021-022 SSFL	1.2×10^{10}	α	0.9×10^{-16}	6.8×10^{-16}	1.9×10^{-16}	2.4×10^{-8}	1.13	82
		β	3.0×10^{-16}	1.8×10^{-15}	4.4×10^{-15}	6.1×10^{-7}	0.006	0
055 SSFL	7.0×10^{9}	α	2.9×10^{-16}	1.1×10^{-16}	7.6×10^{-16}	2.3×10^{-8}	0.18	84
		β	9.6×10^{-16}	5.2×10^{-15}	1.7×10^{-14}	1.0×10^{-6}	0.17	0
Annual average ambient air radioactivity concentration[b] ($\mu Ci/ml$) – 1982		α		2.0×10^{-15}	Total	1.9×10^{-5}		
		β		4.0×10^{-14}				

[a]Assuming all radioactivity detected is from ESG operations.
Guide: De Soto site: 3×10^{-12} $\mu Ci/ml$ alpha, 3×10^{-10} $\mu Ci/ml$ beta; 10 CFR 20 Appendix B.
SSFL site: 6×10^{-14} $\mu Ci/ml$ alpha, 3×10^{-11} $\mu Ci/ml$ beta, 3×10^{-12} $\mu Ci/ml$ 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.

Note: All release points are at the stack exit.

TABLE 7

LIQUID EFFLUENT DISCHARGED TO SANITARY SEWER - 1982

Building	Point of Release	Approximate Effluent Volume (gal)	Activity Monitored	Approximate MDL (μCi/ml)	Annual Average Concentration (μCi/ml)	Sample Maximum Observed Concentration (μCi/ml)	Total Radio-activity Released (Ci)	Percent of Guide[a]
001	Retention tank	25,500	α	1.0×10^{-9}	3.2×10^{-7}	9.8×10^{-7}	3.1×10^{-5}	0.027
			β	3.7×10^{-9}	2.4×10^{-7}	8.0×10^{-7}	2.3×10^{-5}	0.024
004	Flow sampler	690,000	α	1.1×10^{-9}	5.9×10^{-8}[c]	7.0×10^{-8}	7.4×10^{-5}	0.006
			β	3.7×10^{-9}	3.2×10^{-7}[c]	1.6×10^{-7}	2.8×10^{-4}	0.032

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

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. Equipment used for these measurements is maintained by CRIS (see Appendix C) to assure proper maintenance and calibration.

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 1982 edition of that report[9] 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 8
- Soil plutonium radioactivity data presented in Table 9
- Vegetation radioactivity data presented in Table 10
- SSFL Site - Domestic water radioactivity data presented in Table 11
- Bell Creek and Rocketdyne site retention pond radioactivity data presented in Table 12
- Ambient air radioactivity data presented in Table 13.

TABLE 8

SOIL RADIOACTIVITY DATA - 1982

Area	Activity	Number of Samples	Gross Radioactivity (μCi/g)	
			Annual Average Value	Maximum Observed Value[a] and Month Observed
Onsite (monthly)	α	144	$(0.69 \pm 0.20)\ 10^{-6}$	1.18×10^{-6} (November)
	β	144	$(24.6 \pm 2.3)\ 10^{-6}$	30.1×10^{-6} (September)
Offsite (quarterly)	α	48	$(0.68 \pm 0.22)\ 10^{-6}$	1.21×10^{-6} (October)
	β	48	$(23.3 \pm 3.7)\ 10^{-6}$	32.9×10^{-6} (April)

[a]Maximum value observed for single sample.

TABLE 9

SOIL PLUTONIUM RADIOACTIVITY DATA - 1982

Sample Location	9 July 1982 Survey Results		16 December 1982 Survey Results	
	Pu^{238} (μCi/g)	$Pu^{239} + Pu^{240}$ (μCi/g)	Pu^{238} (μCi/g)	$Pu^{239} + Pu^{240}$ (μCi/g)
S-56	$(-3.2 \pm 1.0)\ 10^{-9}$	$(0.7 \pm 1.2)\ 10^{-9}$	$(-7.6 \pm 1.1)\ 10^{-9}$	$(0.6 \pm 1.4)\ 10^{-9}$
S-57	$(1.0 \pm 0.4)\ 10^{-9}$	$(3.9 \pm 0.6)\ 10^{-9}$	$(-8.0 \pm 1.5)\ 10^{-9}$	$(5.1 \pm 2.2)\ 10^{-9}$
S-58	$(-3.2 \pm 1.2)\ 10^{-9}$	$(7.1 \pm 2.3)\ 10^{-9}$	$(0.8 \pm 2.9)\ 10^{-9}$	$(7.3 \pm 2.8)\ 10^{-9}$
S-59	$(0.6 \pm 0.3)\ 10^{-9}$	$(2.3 \pm 0.6)\ 10^{-9}$	$(-7.0 \pm 1.2)\ 10^{-9}$	$(2.4 \pm 1.8)\ 10^{-9}$
S-60	$(0.7 \pm 0.3)\ 10^{-9}$	$(4.6 \pm 0.6)\ 10^{-9}$	$(-7.6 \pm 1.5)\ 10^{-9}$	$(5.6 \pm 2.2)\ 10^{-9}$
S-61	$(1.8 \pm 1.9)\ 10^{-9}$	$(5.0 \pm 2.7)\ 10^{-9}$	$(-8.0 \pm 2.2)\ 10^{-9}$	$(0.4 \pm 2.5)\ 10^{-9}$

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

TABLE 10

VEGETATION RADIOACTIVITY DATA - 1982

Area	Activity	Number of Samples	Gross Radioactivity (μCi/g) Dry Weight: Annual Average Value	Ash: Annual Average Value	Ash: Maximum Value[a] and Month Observed	Percent of Samples With Activity <MDL[b]
Onsite (monthly)	α	144	$(0.03 \pm 0.07)\ 10^{-6}$	$(0.16 \pm 0.22)\ 10^{-6}$	1.25×10^{-6} (February)	60
	β	144	$(23.6 \pm 9.6)\ 10^{-6}$	$(140.2 \pm 48.2)\ 10^{-6}$	260.1×10^{-6} (October)	0
Offsite (quarterly)	α	48	$(0.03 \pm 0.02)\ 10^{-6}$	$(0.17 \pm 0.14)\ 10^{-6}$	0.59×10^{-6} (October)	50
	β	48	$(25.2 \pm 12.2)\ 10^{-6}$	$(129.7 \pm 51.5)\ 10^{-6}$	258.4×10^{-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 11

DOMESTIC WATER RADIOACTIVITY DATA - 1982

Area	Activity	Number of Samples	Gross Radioactivity (μCi/ml) Average Value	Gross Radioactivity (μCi/ml) Maximum Value[a] and Month Observed
ESG-De Soto (monthly)	α	12	$(0.36 \pm 0.23)\ 10^{-9}$	0.79×10^{-9} (February)
	β	12	$(3.97 \pm 1.19)\ 10^{-9}$	6.6×10^{-9} (September)
ESG-SSFL (monthly)	α	24	$(0.14 \pm 0.12)\ 10^{-9}$	0.38×10^{-9} (August)
	β	24	$(3.01 \pm 0.67)\ 10^{-9}$	4.91×10^{-9} (September)

[a] Maximum value observed for single sample

TABLE 12

BELL CREEK AND ROCKETDYNE SITE RETENTION POND RADIOACTIVITY DATA - 1982

Area (Monthly)	Activity	Number of Samples	Gross Radioactivity Concentration			
			Average Value	Maximum Value[a] and Month Observed	Percent of Guide[b]	Percent of Samples With Activity <MDL[c]
Bell Creek mud no. 54 (uCi/g)	α	12	(0.64 ± 0.14) 10^{-6}	0.92 x 10^{-6} (February)	NA	0
	β	12	(23.5 ± 2.1) 10^{-6}	28.0 x 10^{-6} (July)	NA	0
Pond R-2A mud no. 55 (uCi/g)	α	12	(0.74 ± 0.14) 10^{-6}	0.99 x 10^{-6} (January)	NA	0
	β	12	(25.5 ± 3.2) 10^{-6}	26.0 x 10^{-6} (December)	NA	0
Bell Creek vegetation no. 54 (uCi/g-ash)	α	12	(0.06 ± 0.08) 10^{-6}	0.32 x 10^{-6} (December)	NA	75
	β	12	(160.5 ± 67.7) 10^{-6}	280.7 x 10^{-6} (November)	NA	0
Bell Creek vegetation no. 54 (uCi/g dry weight)	α	12	(0.02 ± 0.02) 10^{-6}	0.07 x 10^{-6} (December)	NA	75
	β	12	(31.0 ± 12.2) 10^{-6}	51.5 x 10^{-6} (February)	NA	0
Bell Creek water no. 16 (uCi/ml)	α	12	(0.03 ± 0.06) 10^{-9}	0.14 x 10^{-9} (June)	d	100
	β	12	(3.29 ± 0.7) 10^{-9}	4.4 x 10^{-9} (July)	d	0
Pond water no. 6 (uCi/ml)	α	12	(0.17 ± 0.08) 10^{-9}	0.35 x 10^{-9} (June)	d	83
	β	12	(3.91 ± 1.08) 10^{-9}	5.34 x 10^{-9} (October)	d	0
SSFL pond R-2A water no. 12 (uCi/ml)	α	12	(0.11 ± 0.13) 10^{-9}	0.28 x 10^{-9} (July)	d	75
	β	12	(3.93 ± 0.83) 10^{-9}	5.81 x 10^{-9} (September)	d	0

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

TABLE 13

AMBIENT AIR RADIOACTIVITY DATA - 1982

Site Location (Continuous)	Activity	Number of Samples	Average Value	Maximum Value[a] and Date Observed	Percent of Guide[b]	Percent of Samples With Activity <MDL
De Soto Onsite (μCi/ml)	α	727	$(1.7 \pm 3.1)\ 10^{-15}$	3.9×10^{-14} (07/17)	0.06	93[c]
	β		$(2.6 \pm 1.4)\ 10^{-14}$	2.6×10^{-13} (01/17)	0.009	20[d]
SSFL Onsite (μCi/ml)	α	1690	$(1.1 \pm 2.6)\ 10^{-15}$	3.0×10^{-14} (07/17)	1.8	96[c]
	β		$(2.1 \pm 1.6)\ 10^{-14}$	1.8×10^{-13} (06/13)	0.07	27[d]
SSFL sewage treatment plant Offsite (μCi/ml)	α	312	$(1.6 \pm 3.1)\ 10^{-15}$	1.6×10^{-14} (06/07)	2.7	95[c]
	β		$(2.2 \pm 1.1)\ 10^{-14}$	7.3×10^{-14} (10/07)	0.07	26[d]
SSFL Control Center Offsite (μCi/ml)	α	345	$(1.8 \pm 2.5)\ 10^{-15}$	1.3×10^{-14} (07/12)	3.0	95[c]
	β		$(2.3 \pm 1.3)\ 10^{-14}$	8.8×10^{-14} (04/25)	0.08	21[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 5480.1

[c] MDL = 6.4×10^{-15} μCi/ml alpha

[d] MDL = 1.3×10^{-14} μCi/ml beta.

TABLE 14

DE SOTO AND SSFL SITES - AMBIENT RADIATION DOSIMETRY DATA - 1982

TLD Location	Quarterly Exposure (mR)				Annual Exposure (mR)	Equivalent Exposure Rate (μR/h)
	Q-1	Q-2	Q-3	Q-4		
1. De Soto	28	29	29	31	117	13
2. De Soto	28	28	28	29	113	13
3. De Soto	28	28	29	28	113	13
4. De Soto	25	32	32	32	121	14
5. De Soto	28	30	29	29	116	13
6. De Soto	34	36	33	32	135	15
7. De Soto	28	26	28	28	110	13
Mean value					118	13.4
1. SSFL	31	32	33	a	128	15
2. SSFL	31	34	34	36	135	15
3. SSFL	34	32	39	38	143	16
4. SSFL	a	34	37	37	144	17
5. SSFL	a	31	36	34	135	15
6. SSFL	25	27	28	28	108	12
Mean value					132	15.0
1. Offsite control	29	30	32	32	123	14
2. Offsite control	29	30	33	31	123	14
3. Offsite control	29	31	28	31	119	14
4. Offsite control	32	32	31	32	127	14
5. Offsite control	27	33	34	34	128	15
Mean value					124	14.2

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

NOTE: The elevation for the De Soto and offsite dosimeters is about 1000 ft less than those for the SSFL site. From sea level to a few thousand feet in elevation, the increase in annual exposure is approximately 15 mR/1000 ft. This amount subtracted from the SSFL site results provides good agreement between the three data sets.

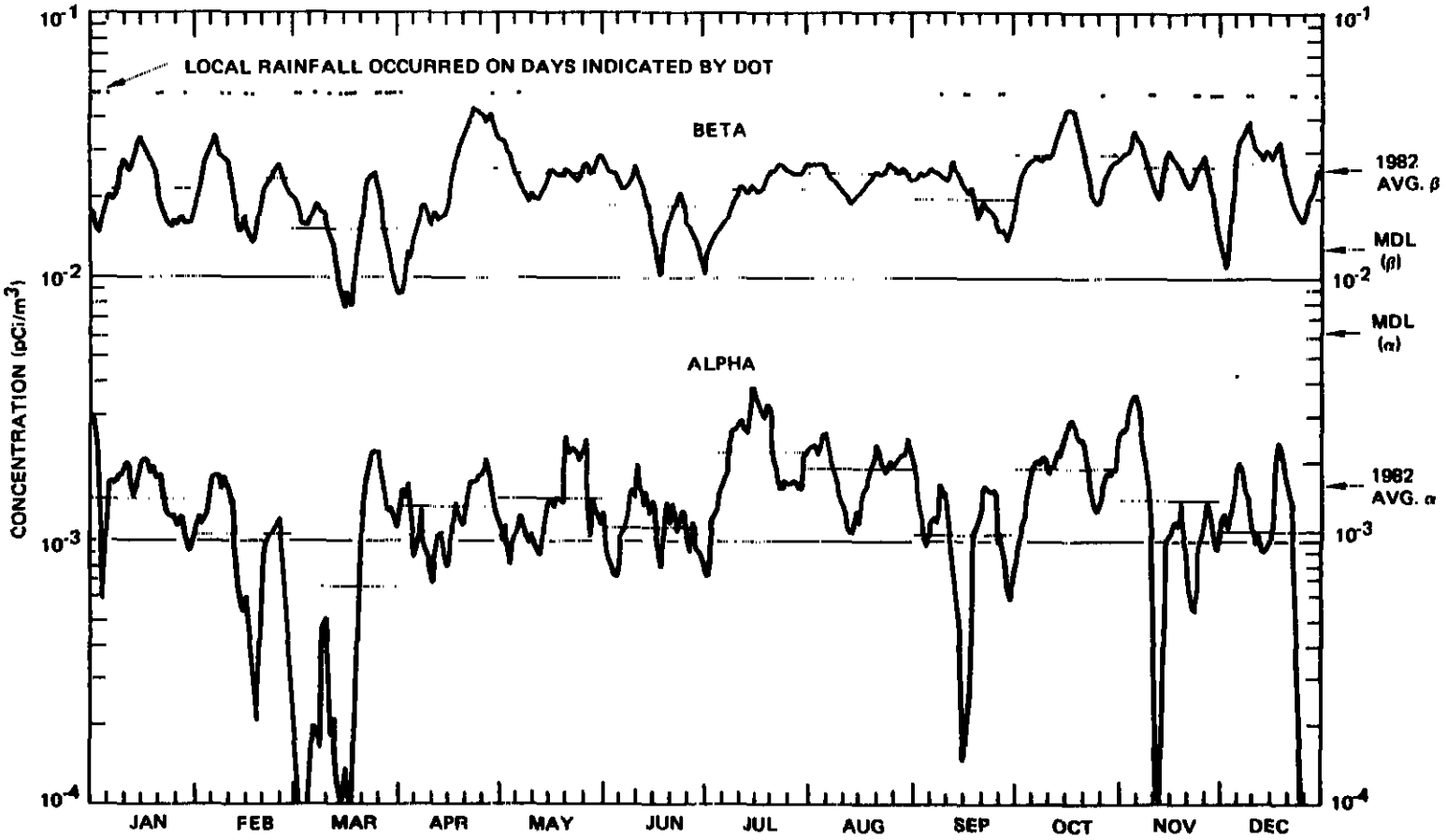

Figure 3. Average Long-Lived Airborne Radioactivity at the De Soto and Santa Susana Field Laboratories Sites - 1982

V. UNUSUAL EVENTS

There were several unusual events at faciities involving radiation or radioactive materials. One of these events was reportable under NRC and State of California regulations. These events are summarized below.

A. REPORTABLE INCIDENT

During the second calendar quarter, in work with NRC- and state-licensed radioactive materials, a series of exposures occurred to an employee's hand that exceeded the regulatory limit of 18.75 rems per calendar quarter. This occurrence was reported to NRC and the State of California as follows.

REPORT OF RADIATION EXPOSURE AT
ENERGY SYSTEMS GROUP, ROCKWELL INTERNATIONAL

During the second calendar quarter of 1982, one individual ("Employee A"), an employee of the Energy Systems Group, Rockwell International, was exposed to radiation to his right hand in excess of applicable limits. This occurred while working with neutron-activated tantalum authorized under the State of California Radioactive Materials License No. 0015-70 and with irradiated nuclear reactor fuel elements under the NRC Special Nuclear Materials License No. SNM-21.

The exposures were determined by use of LiF TLD chip finger rings. These rings are furnished and processed as part of the services for external dosimetry supplied by R. S. Landauer Jr., and Company. Whole-body exposures are determined by use of film badges that are processed quarterly or more often as necessary, also from Landauer. In addition, whole-body exposures are monitored on a daily basis by use of personally assigned direct-reading pocket dosimeters.

The reported hand exposures (in rems) determined by these TLD rings for the periods of activity involving hand-exposure work in the second quarter are:

	Right Hand	Left Hand
April 23-27	0.16	0.17
May 22	1.40	0.54
June 4-9	18.07	3.65
June 22-28	13.33	8.78
Total	32.96	13.14

The April exposure results were reported early in May, while the May exposure results were reported early in June. Both June exposure results were reported by telephone on July 6, 1982.

The April exposure was associated with crimping closed aluminum transfer tubes containing Fermi reactor irradiated fuel scrap. The May and first June exposures were associated with work on irradiated tantalum "Rad-Pac" sources. This activity normally results in a relatively low hand exposure as discussed in some detail below. The second June exposure was associated with various operations in the decladding of irradiated SEFOR fuel rods. These initial decladding operations were anticipated to cause relatively high hand exposures.

The primary causes of this over-exposure are the unexpectedly high exposure during the Rad-Pac work and the high exposure during the SEFOR work.

The Rad-Pac sources are wires of tantalum metal, activated by irradiation in the Northrop Space Laboratories TRIGA reactor to produce 115-day tantalum-182. After return to the Rockwell Interntional Hot Laboratory (RIHL), these wires are sorted according to activity and sealed in capsules or stored for future use. This work is done in a partially shielded work compartment using 3-inch- and 12-inch-long forceps, and normally results in hand exposures of 0.5-1 rems. Rad-Pac sources are usually prepared twice a year, and so the relatively low hand exposures do not normally pose an exposure control problem. However, work on this particular set of wires departed from the normal operating sequence and resulted in a higher-than-normal exposure.

When the irradiation capsule is loaded into the TRIGA reactor, a spacer rod is normally loaded ahead of it, to position the capsule near the mid-plane of the core. This was inadvertently omitted by the Northrop reactor staff for this irradiation, so that the wires received less radiation than required, and the irradiation was nonuniform. The induced activity in the wires was too low and too variable for the wires to be used as Rad-Pac sources.

This low and variable activity was found after the wires were unloaded by Employee A, and the individual wire activities were measured by a health physicist. This work was accomplished normally and, due to the lower activities, probably resulted in a lower than usual hand exposure.

Upon determination that the activated wires were unusable, together with identification of the cause and suitable corrective action, the wires were reloaded for additional irradiation in the reactor. We believe that it was during this task that a large fraction of the 18-rem exposure occurred.

The wires are of two sizes: 0.100 in. diameter by 0.500 in. long and 0.150 in. diameter by 0.865 in. long. For irradiations, the smaller ones are loaded 24 to a holder; the larger ones are loaded 15 to a holder. The holder consists of a short section of thick-walled aluminum (Type 1100) tubing, with circular cross-section grooves drilled along the outer surface. The wires are placed in the grooves, and each holder is loaded into the irradiation capsule. When activated wires are unloaded, the holder contents are quickly dumped into a shielded container. When wires are loaded for irradiation, the holder remains near the worker's hand as each wire is loaded. The extent of this exposure is not normally a problem, since the wires are either new material or previously irradiated material in which much of the activity has decayed and the exposure rate is low. Thus, the step of loading the irradiation capsule is normally a routine operation with low exposures and is not monitored by a health physicist.

In the present case, the irradiated wires contained a total of about 1-2 Ci, and therefore produced a significant exposure rate at the worker's

hands. The worker was aware of the high-exposure rate, because of the measurements made during unloading, and worked as quickly and carefully as possible. He did not feel that he had received a significant exposure.

In order to maintain schedule in spite of the delay resulting from the need to reirradiate the wires, the capsule was immediately unloaded on June 9, after return from the second irradiation. This also differs from normal practice which allows ativity induced in the aluminum (probably 15-hour sodium-24) to decay during a cooling time of several days. This work was monitored by a health physicist, who noted the higher-than-normal exposure rate associated with the aluminum parts, which were promptly moved away.

On June 11, fifteen Rad-Pac sources were shipped to the customer. On June 23, 24, 25, and 28, Employee A was assigned to work on the decladding of irradiated SEFOR reactor fuel. This process was still being developed, and the job was expected to produce relatively high hand exposures, although still below the allowable limit.

Because of the concentration on exposures related to the SEFOR work, neither the facility health physicist nor the facility managers gave appropriate consideration to the earlier Rad-Pac work and the associated hand exposure, and approved the assignment of Employee A to the decladding operation. This assignment was consistent with the whole-body exposure data for this employee at the time.

Employee A's whole body and hand exposure record is summarized below.

	Exposures (rem)		
Period	Whole Body	Right Hand	Left Hand
Prior to 1981	3.4	0.49	0.09
1981	1.14	4.7	1.45
First quarter 1982	0.02	0	0
Second quarter 1982	1.22	32.96	13.14

Several steps have been taken to prevent the recurrence of excessive hand exposures. These corrective actions relate to both the Rad-Pac and SEFOR work specifically, as well as to the general exposure control system.

1) Future Rad-Pac irradiation capsules will be allowed to cool for at least 3 days before unloading to eliminate the radiation from the aluminum parts.

2) A shielding block will be designed and fabricated for use in loading previously irradiated wires into the holders.

3) SEFOR operations involving high hand exposure are being modified to reduce significantly these exposures.

4) Finger rings for recording hand exposures are being sent more frequently and promptly to Landauer for processing and reporting.

5) An exposure reporting form has been established for all personnel working on SEFOR to make the exposure data (head, trunk, and hands) more readily interpreted.

6) A TLD reader for processing LiF TLD chips has been set up at the RIHL so that these chips may be used to measure hand exposure on a job-by-job or day-by-day basis. This will provide continuous current information on hand exposure, for use in conjunction with the direct-reading dosimeter data for head and trunk exposure, to control work assignments and exposures.

It is expected that these changes will effectively eliminate the potential for additional over-exposures to the hands at this facility. The results of these changes will be closely monitored by management.

(END OF EXPOSURE REPORT)

This exposure was investigated by NRC Region V Office of Inspecion and Enforcement and the State of California Division of Occupational Safety and Health (DOSH). At the conclusion, NRC (accompanied by the DOSH inspector) conducted a management conference with upper level ESG management.

Considerable changes were made in the tools and procedures associated with the Rad-Pac source fabrication. The SEFOR decladding job was modified to essentially eliminate the hands-on work at the glove box and therefore considerably reduce the hand exposure on this job.

B. NON-REPORTABLE INCIDENTS

On February 16, an industrial radiographer reported an off-scale dosimeter (greater than 200 mrem). He was restricted from further radiation work and his film badge was sent to Landauer for processing. The reported result was "M" (less than 10 mrem). No cause for the off-scale reading was determined.

On March 2, a can containing enriched UAl_x briquettes collapsed and released radioactive material as it was being removed from a vacuum desiccator. While the shelf and floor were contaminated, no contamination was found on the two individuals involved, and subsequent bioassays (radiometric and fluorometric urinalyses) did not show significant amounts of uranium.

On March 9, during removal of the UAl_x crusher from the glove box in the ATR powder room, airborne contamination was released and the local air monitor alarmed. Both workers involved were using full-face respirators with a protection factor of 50. A breathing-zone (lapel) air sampler showed an adjusted exposure of 2.2 MPC-hours. A bioassay sample taken 3 months later showed no detectable uranium. It was recommended that the crusher box be provided with a tent to reduce the spread of airborne radioactivity when the window must be removed for access to the cursher.

On March 22, while zirconium fines from decladding Fermi fuel were being oxidized in Cell 1 at the RIHL, a small but violent reaction occurred which blew out the seal of the vacuum furnace. Power to the furnace was turned off and the nitrogen purge was started. Some airborne radioactivity was released to the operating gallery and to the exhaust system. Oxidation of zirconium fines was discontinued.

On March 24, while compacting waste in a glove box at NMDF, the operator suffered a cut hand when a glass Petri dish broke. No contamination was detected on his hand or in the wound.

On July 7, at the completion of a fluoroscopic inspection of an ATR fuel plate, the X-ray tubes remained on after the shut-off button was pushed and exposed the operator slightly as he opened the shielded window. The window interlock switches also failed to deactivate the X-ray tubes. The operator's film badge, worn on his chest, was processed and showed "M" (less than 10 mrem). Film badges exposed in a simulation of the incident showed "M" except for one located at waist level, which showed 10 mrem. The electrical failures were investigated and repaired, but a second incident occurred on August 10 (see below).

On July 13, radioactive material was released in Decon 4 at the RIHL while a SEFOR fuel transfer tube was being crimped. Three individuals became contaminated, but there was no indiction of airborne activity. Nasal smears showed no contamination, and urinalysis for three individuals showed no significant activity. Two individuals showed fission product activity slightly above the MDA but this cleared out immediately. Greater protective clothing and respiratory protective equipment was specified for future crimping operations. The alignment of the crimping tool was adjusted to improve its performance.

At the end of the second quarter, a film badge report showed an unusually high exposure (330 mrem) for an inspection manager. It was determined that the individual frequently left his badge near a rack of ATR fuel elements which accounted for essentially all the indicated exposure. He was instructed to hang his badge on the film badge rack when not wearing it.

On July 30, approximately 5 ml of irradiated fuel solution leaked through an inline HEPA filter as nitrogen gas was flowed through the drained L-85 core tank to dry it. The gas flow was stopped and the radioactive liquid was absorbed. Subsequent drying opertions were done with a Drierite-filled liquid trap inline.

On August 10, a second fluoroscope exposure occurred in the ATR facility. The failure was essentially the same as the previous case.

Investigation this time located a burned closed relay contact and a loss of spring force in the interlock switches. These faults were corrected.

On August 21, three employees became contaminated as a result of cutting open a bag of contaminated waste and the breaking of another bag of contaminated waste in the Hot Storage Room at RIHL. Urinalysis for one worker showed fission product activity slightly above the MDA, but this had cleared out 2 months later.

On September 16, an ATR powder room operator opened a can outside of a glove box. The can contained loose UAl_x power, which splattered out when the lid fell into the can. The operator was restricted from further work in the power room. This was the last can of UAl_x to be used in this project. Fecal samples submitted the next day showed uranium activity greater than the MDA. Urine samples showed no significant activity except for one sample, slightly above the MDA, taken 3 weeks after the incident. An in-vivo lung count one month after the incident showed no uranium above the MDA.

On September 19, water was found coming out of the ATR facility on the adjacent floor. The water had originated in a corroded fire sprinkler line in the exhaust duct for the nitric acid cleaning tank in the cleaning room. The activity in the water was less than 5% of MPC for a restricted area. The water was drained to the holdup tank. Floors were wet vacuumed and surveyed. No contamination was found to have resulted from this incident.

On October 11, a small leak in a glove on the SEFOR decladding glove box resulted in contamination on the operator's labcoat, forehead and hand, and a coworker's shirt and the floor. Fecal bioassay showed a small amount of contamination. Later urinalysis showed none.

On October 26, as UAl_x scrap powder was being sampled in a glove box, it ignited. The fire was put out immediately by use of a Met-L-X extinguisher with a bayonet inserted through a glove. No contamination was released from the glove box.

0106Y/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	Average Dose (rems)	Group Dose (Person rems)
1982	349	29	8	3	6	15	4	7	8	429	0.271	116
1981	192	55	13	4	6	4				274	0.121	33
1980	357	39	10	3	5	9	3			426	0.131	56*
1979	347	39	19	10	4	15	8	2		444	0.204	91*
1978	432	60	18	16	4	18	9	1	1	559	0.197	110*
1977	340	31	29	7	5	11	13			436	0.209	91*
1976	295	38	17	14	5	9	2			380	0.156	59*
1975	170	24	12	4	5	6	1	1		223	0.175	39*

*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 and 1982 represent occupationally exposed Rockwell employees. The group dose was calculated exactly for these two 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 1982 were considerably greater than in prior years. This resulted from exceptionally difficult working conditions in decladding the SEFOR fuel. This fuel was composed of oxide pellets in heavy-walled cladding with a helium bond. The job had been planned on the assumption that the majority of the pellets would be intact and would slide out of the cladding when the ends of the rods were cut off. On this basis, the job was planned to be done in a glove box with operators working "hands-on" to the material.

However, all pellets were badly cracked and efforts to push them out of the cladding failed due to jamming of the broken pieces. A crusher was devised, which broke the pellets into pieces small enough to pour out of the cladding. This operation resulted in unacceptably high hand exposures. Modifications to reduce these exposures were proposed in a meeting between Nuclear Operations and HS&RS early in August, about 2 months after the start of decladding. A second meeting a week later resulted in the conclusion that the modifications could not be sufficiently effective, and the "hands-on" approach was discontinued, in favor of master-slave manipulators installed in the glove box. With the operators at a greater distance from the fuel and cladding, hand exposures were no longer limiting. Whole-body exposures were anticipated to be quite high, but the potential reduction in group dose was judged to be not worth the impact on cost and schedule that would result from moving the operation into one of the hot cells. Continuing efforts were made to keep exposures as low as reasonably achievable throughout the rest of the project, subject to the constraint of working at the glove box, out-of-cell.

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	
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 4. This graph, and the tabulated annual doses, show a clear increase from 1976 to 1980, followed by a decrease for 1981 and 1982. All data prior to 1982 were obtained using an EG&G TL-3 reader. The 1982 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.

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.

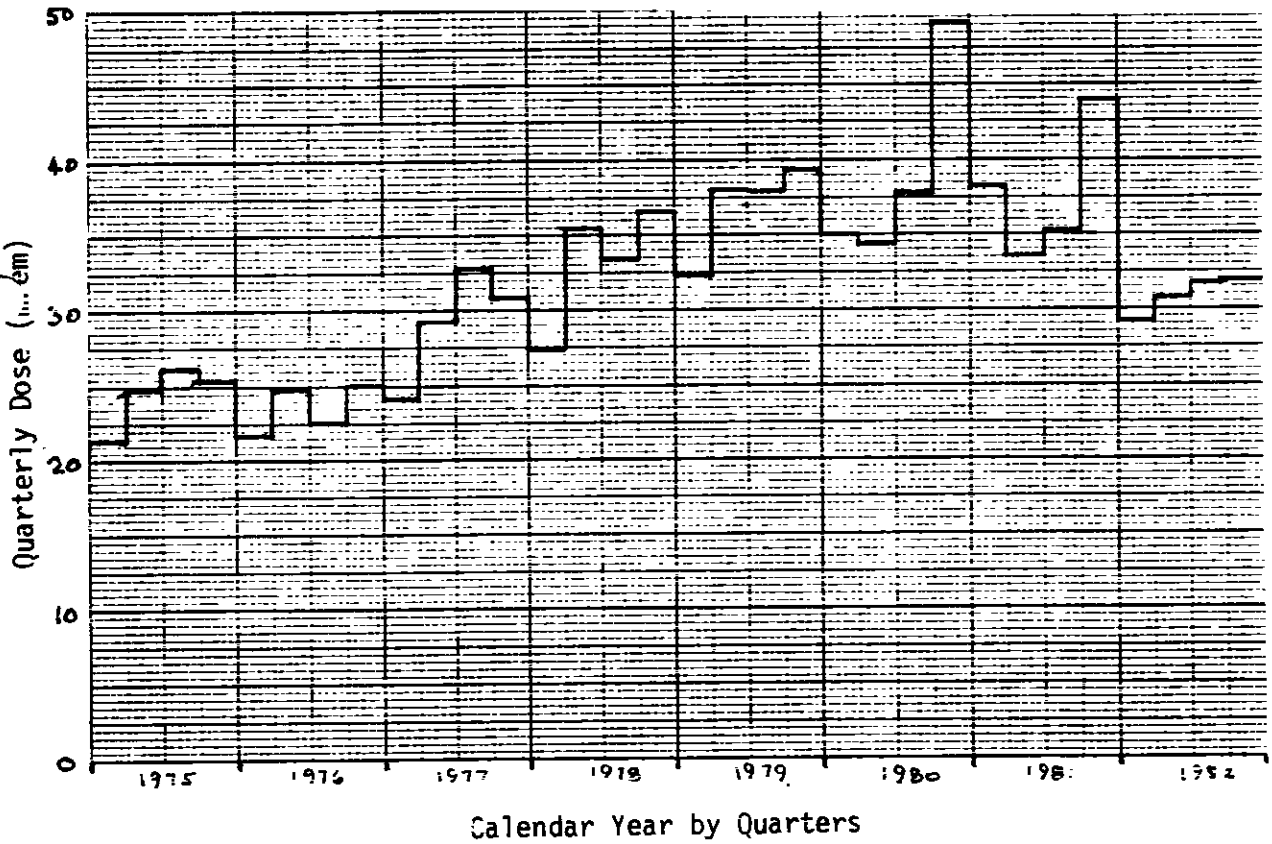

Figure 4. Averaged Quarterly Dose Recorded by Environmental TLDs

RADIOACTIVITY DISCHARGED TO ATMOSPHERE
(microcuries)

	De Soto		Santa Susana		
	001	004	RIHL	RMDF	NMDF
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.

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VII. ANTICIPATED ACTIVITIES DURING NEXT REPORTING PERIOD (1983)

Buildings 001/004

Decontamination and decommissioning of all areas involved in special nuclear material operations. Current production run of test reactor fuel elements will be completed. No further production of fuel elements is planned.

Building 020

Complete the SEFOR fuel decladding program.

Buildings 021/022

Storage and transfer of declad SEFOR fuel and scrap.

Building 055

This facility is inactive. Decontamination of glove boxes and preparation for shipment for disposal.

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

3. "Annual Review of Radiological Controls - 1976," R. S. Hart, Energy Systems Group, Rockwell International, N001TI990003, April 2, 1980

4. "Annual Review of Radiological Controls - 1977," R. S. Hart, Energy Systems Group, Rockwell International, N001TI990098, May 27, 1980

5. "Annual Review of Radiological Controls - 1978," R. S. Hart, Energy Systems Group, Rockwell International, N001TI990104, August 6, 1980

6. "Annual Review of Radiological Controls - 1979," R. S. Hart, Energy Systems Group, Rockwell International, N001TI990113, September 15, 1980

7. "Annual Review of Radiological Controls - 1980," R. R. Eggleston, Energy Systems Group, Rockwell International, N001TI990144, July 14, 1981

8. "Annual Review of Radiological Controls - 1981," R. R. Eggleston, Energy Systems Group, Rockwell International, N001TI990176, January 21, 1983

9. "Energy Systems Group Environmental Monitoring and Facility Effluent Annual Report - 1982," J. D. Moore, Rockwell International, Energy Systems Group, ESG-83-17, June 1983

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APPENDICES

APPENDIX A
PERSONNEL MONITORING PROGRAM

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

1. Type of film - Eastman Kodak film stock of Type 2 and Type NTA specially packaged by R. S. Landauer, Jr. & Company

   Exposure Range

   a) Gamma and X-ray (<100 keV) - 10 mR - 500R ± 10% or 10 mR, whichever is greater.

   b) Gamma and X-ray (>100 keV) - 10 mR - 60R ± 20% or 30 mR, whichever is greater.

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

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

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

2. Filters and Specifications

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

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

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

3. Sources which film are calibrated to:

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

4. Criticality Dosimeter (a separate packet that is attached to the film badge)

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.

Indium Foil - 1/2 in. by 5/8 in. by 0.015 in. thick; primarily a thermal neutron detector used as an indicator in case of a criticality event. Plastic in front of this foil is 0.040 in. thick. By using a thin window radiation survey meter, an estimate of the magnitude of exposure can be made. Although the foil may reach saturation, highly exposed individuals can be separated from those less likely to have had exposure. The half life is approximatel;y 54 minutes - 320 mg; 99.97% purity.

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

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

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

Options

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

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

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

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

Certain operations, such as hot cell entries, which may pose a high exposure potential, require the use of special badgess, which are badges worn for a single operation in place of personal badges. When special badges are required, two badges are worn by each individual. Special badges are evaluated according to the method previously described; however, the average reading of the two badges is recorded as the dose.

The Film Badge Dosimetry Report also contains the following information on monitored personnel:

1) Social security number
2) Name
3) Date of birth
4) Badge number
5) Current Dose X + Gamma, Neutron, Beta
6) Calendar quarter dose penetrating, nonpenetrating
7) Calendar year dose penetrating, nonpenetrating
8) Lifetime dose penetrating, nonpenetrating

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

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APPENDIX B
ANALYTICAL PROCEDURE SUMMARY FOR BIOASSAY BY URINALYSIS

The following summary of analytical procedures is limited to the most frequently performed bioassays by 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 the 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 the 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 rsults will be reported separately as FP3a and the oxalate soluble results will be reported separately as FP3b.

Plutonium (PUA,PUB)

After reduction to plutonium (III) and (IV) with hydroxylamine hydrochloride, plutonium is precipitated with lanthanum fluoride. This isolates the plutonium from most elements including uranium, except thorium, the rare earths and actinides.

After oxidation of plutonium with sodium nitrate in acid media, extraction of plutonium is carried out with 0.5 M thenoyltrifluoro acetone in xylene. Following extraction, the aqueous solution containing plutonium is neutralized and concentrated by heating. After oxidation of the plutonium in a basic media, it is electrodeposited on a stainless steel disc. The plutonium activity is determined by autoradiography (PUA) for greater sensitivity, or counted for alpha radiation with a proportional counter (PUB).

Gross Beta, High Level (GBH)

The gross sample is evaporated to dryness, followed by organic digestion by hydrogen peroxide and nitric acid. Natural potassium (K-40) correction is determined by diluting the ashed salts to a known volume, and removing an aliquot for flame spectrophotometry. The remaining solution is evaporated to near dryness, planchetted, and counted for beta radiation with a proportional counter. The radioactivity in the urine sample due to K-40 is subtracted from the gross count.

Gross Alpha (GA1a)

Specific for uranium and/or plutonium which is extracted frm 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 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 conted for alpha radiation with a proportional counter.

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

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	100 ug/day
Enriched U	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.

TABLE B-1

SUMMARY OF BIOASSAY SERVICES AVAILABLE FROM UNITED STATES TESTING COMPANY, INC.

Analysis Type	Listing Code	Analysis Specific For	Sensitivity/ 1500 ml	Accuracy at Minimum Sensitivity	Minimum Volume Required	Remarks
Fluorometric Uranium	UF	Normal or depleted uranium	0.3 µg	±50%	10 ml	
Radiometric Uranium	UR	Enriched uranium	3.75 dpm	±50%	200 ml	
Fission Products (1)	FP1	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)	FP2	Same as FP1 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)	FP3	Same as FP2 with insoluble and soluble oxalate results reported separately as FP3a and FP3b, respectively.	30 dpm FB3a 60 dpm FB3b		300 ml	Volatile fission products lost.
Tritium	H3	Tritium	2.25×10^6 dpm	±50%	10 ml	
Plutonium (A)	PUA	Plutonium	0.0495 dpm	±50%	1000 ml	Greater accuracy than PUB analysis.

TABLE B-1

SUMMARY OF BIOASSAY SERVICES AVAILABLE FROM UNITED STATES TESTING COMPANY, INC.

(continued)

Analysis Type	Listing Code	Analysis Specific for	Sensitivity/ 1500 ml	Accuracy at Minimum Sensitivity	Minimum Volume Required	Remarks
Plutonium (B)	PUB	Plutonium	0.0495 dpm	±75%	1000 ml	Double precipitations, washes, and extractions are eliminated for faster analysis at reduced accuracy.
Plutonium (B) (Optional)	PUB	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 µg	±50%	1000 ml	
Gross Beta High Level	GBH	All beta emitters except halogens	750 dpm	±75%	50 ml	K-40 corrected
Gross Alpha (1a)	GA1A	Uranium and plutonium	1.5 dpm	±50%	100 ml	Sample electrodeposited on SS disc and auto-radiographed.
Gross Alpha (1b)	GA1B	Uranium and plutonium	9 dpm	±50%	100 ml	Sample planchetted and proportional counted for alpha.

APPENDIX C

CALIBRATION, RECALL, AND INVENTORY SYSTEM (CRIS)

Instrumentation utilized in making radiological and environmental measurements receive preventative maintenance on schedules maintained by a computerized system (Calibration, Recall, and Inventory System - CRIS). This system provides notification to the users that specific instruments are due for scheduled inspection, calibration, or adjustment. If the instruments are not returned for this service as required, delinquency reports are printed out for management attention.

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