Badger

Rockwell International
Energy Systems Group

SUPPORTING DOCUMENT

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ABSTRACT

Data on exposures of Energy Systems Group personnel at NRC-licensed facilities are presented for CY 1981. This summary, in conjunction with previous annual reports, has been used to determine that the NRC License Condition 23 of the Special Nuclear Materials License SNM-21 continues to be met. The exposures to ionizing radiation continue to be below license limits. There have been no significant upward trends in these exposures. There were no serious radiological events during 1981. This experience and the continued level of activity scheduled for 1982 indicate that there is no need for changes in the ESG radiological controls program.

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REV	SUMMARY OF CHANGE	APPROVALS AND DATE
A	Table 5 had shown values for airborne concentrations that were for 1980. These have been revised to show the concentration values for 1981. Page 14, Lines 5 to 9: Change 2.5×10^{-11} to 1.7×10^{-10} Change 2.0×10^{-10} to 5.5×10^{-14} Change 8.5×10^{-12} to 1.5×10^{-10} Change 2×10^{-12} to 4.8×10^{-13} Change 5.5×10^{-13} to 6.9×10^{-13} Change 1×10^{-13} to 1.6×10^{-13} Change 9.0×10^{-12} to 3.8×10^{-14} Add 2.3×10^{-14} Page 14, Line 18: Change 9.0×10^{-12} to 1.5×10^{-12} Page 14, Line 19 Change 2.2×10^{-13} to 2.3×10^{-14} Page 14, Building Description Column: Add radiation symbol to Building No.	[signature] 05/26/84 RJ Tuttle 5/28/84 Release date 5-29-84 [initials]

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INTRODUCTION

Condition 23 of the Atomics International special material 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."

These reports for the years 1975 through 1980[2][3][4][5][6][7] provide a historical basis for the identification of trends. It should be noted that, in some instance, both NRC-licensed and non-NRC-licensed activities take place in the same building. In these cases, certain measurements (e.g., ventilation air exhaust radioactivity) are not possible to be separated for each type of activity. When this occurs, the values are reported unmodified as measured and conservatively may be attributed wholly to licensed activities.

Additionally, it is not possible to separate the integrated personnel radiological doses to that attributable from either nonlicensed activities for the DOE or the licensed activities.

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

1) <u>Fuel Fabrications</u> — 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.

I. PERSONNEL DOSIMETRY

Personnel dosimetry techniques generally consist of two types: those which measure incident radiation on the body from external sources (e.g., film badges) and those which measure internal body organ accumulations of radioactivity via inhalation, ingestion, or possibly through cuts or puncture wounds (e.g., bioassays). An attempt has been made to separate the exposure modes as much as possible along these lines to (1) permit an evaluation of the more significant exposure routes and (2) to allow a differentiation between those exposure sources which are external and controllable in the future and those which may continue to irradiate the body from some time period, regardless of future efforts (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

Personnel external radiation edxposures for the pertinent activities for the year are presented in Table 1, where the number of individuals within a specific annual dose range is shown along with the percentage of employees within each annual dose range or less and the man-rem contributed by each dose range value. It also should be noted (see Summary, Section VI) that all exposures were less than 2 rem and were well below the annual occupational limit of 5 rem. The cumulative log-normal distribution of all those personnel with reported exposures greater than zero is shown in Figure 1.

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 1981, 63 lung scans were made for uranium deposition. Twenty-seven of the scans (on 22 different individuals) showed positive results. However, followup scans showed a steady decrease of these lung burdens (see Table 3).

*Helgeson Nuclear Services, Inc., Pleasanton, California

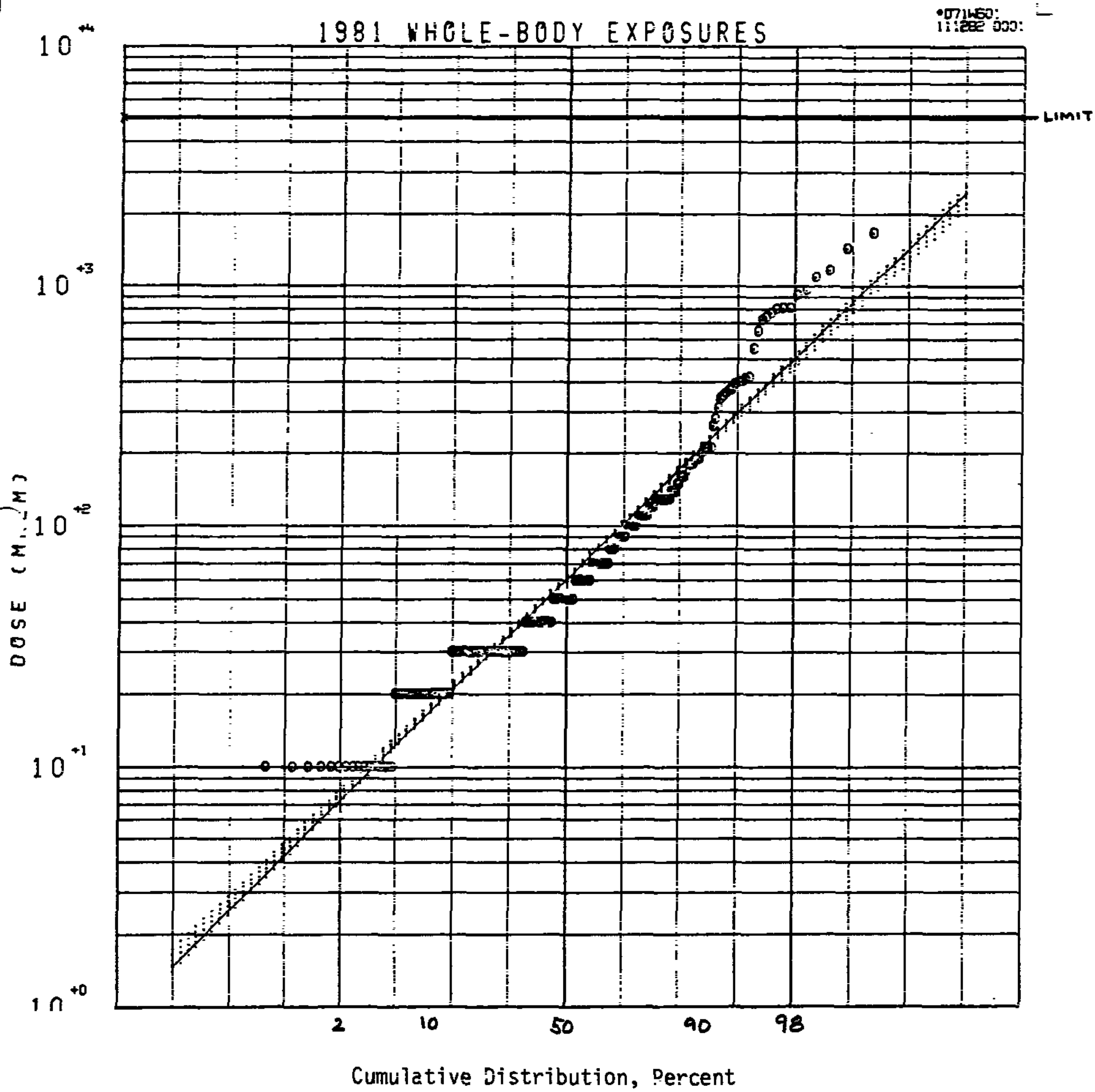

Figure 1. Cumulative Log-Normal Distribution for Whole-Body Radiation Exposures of Occupationally Exposed Individuals in 1981

TABLE 1

PERSONNEL EXTERNAL (WHOLE BODY) EXPOSURES — 1981

Dose Range (rem)	P (People)	C.P. (Cumulative People)	C.P. (%)	Population - Dose* (Man-rem)
No Meas. Exposure	418	418	51.23	0
<0.10	311	729	89.34	15.55
0.10 - 0.25	61	790	96.81	10.68
0.25 - 0.50	12	802	98.28	4.50
0.50 - 0.75	4	806	98.78	2.50
0.75 - 1.00	5	811	99.39	4.38
1.00 - 2.00	5	816	100.00	7.50
2.00 - 3.00	0	816	100.00	0
>3.00	0	816	Total	45.11

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

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. The techniques employed are described in the Appendix. A statistical summary of the results for 1981 appear in Table 2, while a detailed listing of the positive results and followup measurements are shown in Table 3. Data on the In-Vivo lung scans performed in 1981 also appear in these tables.

TABLE 2
SUMMARY OF BIOASSAYS — 1981

Measurement*	Type*	Total Tests	Total Positive Results	Total Individuals With Positive Results
U	UF	161	2	2
U	UR	152	1	1
GA	1A	0	0	0
GA	1B	0	0	0
GA	2B	62	0	0
GB	H	0	0	0
PU	A	11	0	0
FP	1	0	0	0
FP	3A	141	31	23
FP	3B	141	3	2
U	IVLC	63	27	22
Sr-90	Sr-90	8	2	2
Cs-137	TBC	0	0	0
H-3	H-3	29	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
(For a discussion of specific analytical techniques employed, see Appendix B)
U-IVLC = Uranium In-Vivo Lung Count
TBC = Total Body Count
H-3 = Tritium
Th = Thorium
H = High Level

TABLE 3
POSITIVE BIOASSAY RESULT SUMMARY — 1981
(Sheet 1 of 3)

H&S* Number	Sample Date	Analysis Type*	Results: Per Vol. Anal. (dpm)	Results: Per 1500 ml-day (dpm)	Assumed Specific Radionuclide	Assumed Critical Nuclide Equivalent MPBB (%)
3725	06/03/81	IVLC	40 µg		U-235	16.3**
3725	11/01/81	FP3A	5.8	43.5	Sr-90	9.1
3897	12/14/81	FP3A	5.8	43.5	Sr-90	9.1
3762	02/27/81	IVLC	39 µg		U-235	15.9**
4662	10/21/81	IVLC	62 µg		U-235	25.3**
3742	11/23/81	FP3A	8.2	61.5	Sr-90	12.8
4253	10/16/81	FP3A	6.0	45.0	Sr-90	9.4
4253	11/15/81	FP3A	10.4	78.0	Sr-90	16.2
4390	08/24/81	FP3A	33.1	248.0	Sr-90	51.7(1)
4390	08/24/81	Sr-90	21.8	72.7	Sr-90	15.1
4390	08/24/81	FP3A	9.8	73.5	Sr-90	15.3
4136	06/03/81	IVLC	49 µg		U-235	20.0**
2040	02/27/81	IVLC	51 µg		U-235	20.8**
2040	06/03/81	IVLC	49 µg		U-235	20.0**
2040	10/21/81	IVLC	53 µg		U-235	21.6**
2040	12/02/81	UR	0.86	6.45	U-235	6.4
1281	10/21/81	IVLC	37 µg		U-235	15.1**
4681	02/27/81	IVLC	60 µg		U-235	24.5**
4163	08/23/81	FP3A	39.6	297.0	Sr-90	61.9(1)
4163	08/23/81	Sr-90	26.7	89.0	Sr-90	18.5
4163	09/08/81	FP3A	6.3	47.3	Sr-90	9.8
4163	09/29/81	FP3A	4.6	34.5	Sr-90	7.2
4163	10/29/81	FP3A	5.7	42.8	Sr-90	8.9
4486	05/19/81	FP3A	4.2	31.5	Sr-90	6.6
4486	11/23/81	FP3A	10.6	79.5	Sr-90	16.6
4160	12/14/81	FP3A	12.7	95.25	Sr-90	19.8

TABLE 3

POSITIVE BIOASSAY RESULT SUMMARY — 1981
(Sheet 2 of 3)

H&S* Number	Sample Date	Analysis Type*	Results: Per Vol. Anal. (dpm)	Results: Per 1500 ml-day (dpm)	Assumed Specific Radionuclide	Assumed Critical Nuclide Equivalent MPBB (%)
3746	06/03/81	IVLC	34 μg		U-235	13.9**
1073	10/21/81	IVLC	30 μg		U-235	12.2**
4440	10/21/81	IVLC	40 μg		U-235	16.3**
1863	10/21/81	IVLC	51 μg		U-235	20.8**
4572	02/27/81	IVLC	29 μg		U-235	11.8**
4650	02/27/81	IVLC	45 μg		U-235	18.4**
4650	06/03/81	IVLC	37 μg		U-235	15.1**
4231	11/23/81	FP3A	17.5	131.25	Sr-90	27.3
4430	11/22/81	FP3A	39.4	295.50	Sr-90	61.6
4393	06/03/81	IVLC	40 μg		U-235	16.3**
2307	11/15/81	FP3A	6.3	47.3	Sr-90	9.8
4524	06/01/81	FP3A	9.3	70.0	Sr-90	14.6
4524	09/04/81	FP3A	5.7	42.8	Sr-90	8.9
4710	11/16/81	FP3A	8.5	63.75	Sr-90	13.3
4565	11/23/81	FP3A	9.9	74.25	Sr-90	15.5
2041	05/29/81	FP3A	11.7	87.8	Sr-90	18.3
2041	06/22/81	FP3B	13.0	97.5	Cs-137	0.015
3403	06/17/81	FP3B	12.0	90.0	Cs-137	0.014
3393	11/15/81	FP3A	10.8	81.0	Sr-90	16.9
4211	06/03/81	IVLC	47 μg		U-235	19.2**
3726	11/01/81	UF	0.0004 μg	0.6	U	0.6
3726	11/01/81	FP3A	5.4	40.1	Sr-90	8.4
4840	07/28/81	UF	0.0003 μg	0.45	U	0.45
4125	10/21/81	IVLC	40 μg		U-235	16.3**
4303	09/13/81	FP3A	6.1	45.8	Sr-90	9.5

TABLE 3

POSITIVE BIOASSAY RESULT SUMMARY — 1981
(Sheet 3 of 3)

H&S* Number	Sample Date	Analysis Type*	Results Per Vol. Anal. (dpm)	Results Per 1500 ml-day (dpm)	Assumed Specific Radionuclide	Assumed Critical Nuclide Equivalent MPBB (%)
4303	12/02/81	FP3A	25.3	189.75	Sr-90	39.5
1374	06/22/81	FP3B	9.8	73.5	Cs-137	0.011
1374	06/22/81	FP3A	4.2	31.5	Sr-90	6.6
0606	06/03/81	IVLC	53 μgm		U-235	21.6**
4137	11/23/81	FP3A	6.0	45.0	Sr-90	9.4
4104	06/03/81	IVLC	39 μg		U-235	15.9**
2729	06/03/81	IVLC	36 μg		U-235	14.7**
3078	07/06/81	FP3A	4.7	35.3	Sr-90	7.4
3939	11/02/81	FP3A	5.4	40.5	Sr-90	8.4
4651	02/27/81	IVLC	46 μg		U-235	18.8**
4651	10/21/81	IVLC	41 μg		U-235	16.7**
1547	06/03/81	IVLC	42 μg		U-235	17.1**
1547	06/17/81	FP3A	4.1	30.8	Sr-90	6.4
1547	10/21/81	IVLC	34 μg		U-235	13.9**
4587	10/21/81	IVLC	48 μg		U-235	15.5**

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

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

To roughly characterize the general external levels of penetrating radiation which existed at each facility during the year, the data presented in Table 4 were compiled based on survey measurements made by the assigned HS&RS representative(s) during the year. It should be noted that while these data are typical, higher levels existed for very limited periods in Building 020 during the handling of a fuel cannister.

TABLE 4
RADIATION LEVELS — WORKING AREAS — 1981

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

*Estimated

B. INTERIOR AIR SAMPLES — WORKING AREAS

In those working areas where the nature of the tasks being performed and of the materials in use might lead to the potential for generation of respirably-sized radioactive aerosols, periodic local air sampling is performed. A summary of these results for 1981 is given in Table 5.

TABLE 5

INTERIOR AIR SAMPLE SUMMARY – 1981

Building		Maximum µCi/cc	Average µCi/cc
001-Fuel Fab (Lapel)	α	1.7×10^{-10}	5.5×10^{-14}
004		(Not Required)	--
020 Controlled Areas	β	1.5×10^{-10}	4.8×10^{-13}
Uncontrolled Areas	β	6.9×10^{-13}	1.6×10^{-13}
055	α	3.8×10^{-14}	2.3×10^{-14}

C. SPECIAL AIR SAMPLES – BUILDING 055

The NMDF (Building 055 was not utilized for Pu fuel fabrication or R&D work during 1981. The routine sampling of many different locations in the building continued on a daily basis as in past years until the last week in August. At that time, a review of the results indicated that the measurements of the cumulative weekly exposures were low and unvarying. Air sampling remained on a continuous basis but the filter change frequency was reduced to weekly. The same sampling locations were used. The maximum cumulative weekly exposure at any location was 1.5×10^{-12} µCi-hr/ml. The average weekly concentration (40 hr/wk) is 2.3×10^{-14} µCi/ml, which can be compared to the MPC for the most restrictive radioisotope that could be present (Pu-239), i.e., 2×10^{-12} µCi/ml.

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 these measurements is maintained by CRIS (see Appendix C) to assure proper operation and calibration.

An annual review 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 1981 edition of that review[8] for atmospherically discharged and liquid effluents for the facilities identified above is presented in Tables 6 and 7, respectively.

TABLE 6

ATMOSPHERIC EMISSIONS TO UNRESTRICTED AREAS — 1981

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)	% of Guide[a]	% of Samples with Activity <MDL
001 De Soto	1.4×10^{10}	α	1.6×10^{-16}	$<6.8 \times 10^{-15}$	5.0×10^{-14}	$<2.8 \times 10^{-6}$	<0.23	35
		β	5.4×10^{-16}	$<6.7 \times 10^{-15}$	4.7×10^{-14}	$<2.7 \times 10^{-6}$	<0.002	35
004 De Soto	3.1×10^{10}	α	2.1×10^{-16}	$<4.4 \times 10^{-16}$	2.1×10^{-15}	$<3.9 \times 10^{-7}$	<0.01	56
		β	7.2×10^{-16}	$<4.7 \times 10^{-15}$	2.3×10^{-14}	$<4.1 \times 10^{-6}$	<0.002	12
020 SSFL	1.0×10^{10}	α	0.9×10^{-16}	$<2.3 \times 10^{-16}$	8.7×10^{-16}	$<6.9 \times 10^{-8}$	<0.38	32
		β	3.0×10^{-16}	4.6×10^{-14}	1.7×10^{-13}	1.4×10^{-5}	0.15	0
021-022 SSFL	1.2×10^{10}	α	0.9×10^{-16}	$<2.5 \times 10^{-16}$	1.4×10^{-15}	$<8.7 \times 10^{-8}$	<0.42	38
		β	3.0×10^{-16}	1.2×10^{-14}	3.4×10^{-14}	4.0×10^{-6}	0.04	0
055 SSFL	7.0×10^{9}	α	2.9×10^{-16}	$<2.9 \times 10^{-16}$	6.6×10^{-16}	$<5.9 \times 10^{-8}$	<0.48	88
		β	9.6×10^{-16}	9.8×10^{-15}	4.2×10^{-14}	2.0×10^{-6}	<0.003	0
Annual average ambient air radioactivity concentration (µCi/ml) — 1981		α		$<6.8 \times 10^{-15}$	Total (α + β)	$<3.0 \times 10^{-5}$		
		β		$<1.2 \times 10^{-13}$				

[a]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} µCi/ml beta (055 only); 10 CFR 20 Appendix B, CAC-17, and DOE Order 5480.1 Chapter XI.

Note: All release points are at the Stack Exit.

TABLE 7

LIQUID EFFLUENT DISCHARGED TO SANITARY SEWER 1981

Building	Point of Release	Approximate Effluent Volume (gal)	Activity Monitored	Approximate MDL (μCi/ml)	Annual Average Concentration (μCi/ml)	Sample Maximum Observed Concentration (μCi/ml)	Total Radioactivity Released (Ci)	% of Guide[b]
001	Retention tank	25,500	α	1.0×10^{-9}	1.8×10^{-7}	5.4×10^{-7}	9.3×10^{-6}	0.020
			β	3.7×10^{-9}	9.6×10^{-8}	1.8×10^{-7}	1.7×10^{-5}	0.010
004	Flow sampler	1,516,000	α	1.1×10^{-9}	$<1.3 \times 10^{-8}$[c]	7.0×10^{-8}	7.5×10^{-5}	0.001
			β	3.7×10^{-9}	$<3.7 \times 10^{-8}$[c]	1.6×10^{-7}	2.1×10^{-4}	0.004
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]% of samples <MDL: 59.7% alpha activity, 47.8% beta activity
Note: The average radioactivity concentration in De Soto site domestic water supplied by the Los Angeles City Department of Water and Power during 1981 was 3.7×10^{-10} μCi/ml alpha and 3.8×10^{-9} μCi/ml beta.

IV. ENVIRONMENTAL MONITORING PROGRAM

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

The annual review of radiological controls prepared by Radiation & Nuclear Safety in the HS&RS Department also describes in detail the ESG environmental monitoring program.

Some of the data reported in the 1981 edition of that review(8) are presented here. It is important to remember that the radiological activity levels reported can be attributed not only to all activities at ESG-NRC licensed, DOE-sponsored, and State of California-licensed, but also to external influences such as fallout from nuclear weapon testing.

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 GROSS RADIOACTIVITY DATA — 1981

Area	Activity	No. Samples	Gross Radioactivity (μCi/g) Annual Average Value (Standard Deviation)	Gross Radioactivity (μCi/g) Maximum Observed Value[a] and Month Observed
Onsite (monthly)	α	144	$(0.69 \pm 0.20)\ 10^{-6}$	1.32×10^{-6} (December)
	β	144	$(25.4 \pm 3.5)\ 10^{-6}$	38.2×10^{-6} (May)
Offsite (quarterly)	α	48	$(0.64 \pm 0.23)\ 10^{-6}$	1.26×10^{-6} (July)
	β	48	$(22.8 \pm 4.5)\ 10^{-6}$	33.2×10^{-6} (April)

[a]Maximum value observed for single sample.

TABLE 9

SOIL PLUTONIUM RADIOACTIVITY DATA — 1981

Sample Location	8 July 1981 Survey Results		17 December 1981 Survey Results	
	Pu^{238} (μCi/g)	$Pu^{239} + Pu^{240}$ (μCi/g)	Pu^{238} (μCi/g)	$Pu^{239} + Pu^{240}$ (μCi/g)
S-56	$(-2.4 \pm 2.0)\ 10^{-9}$	$(2.0 \pm 2.4)\ 10^{-9}$	$(-2.9 \pm 2.1)\ 10^{-9}$	$(3.5 \pm 3.8)\ 10^{-9}$
S-57	$(-3.7 \pm 1.4)\ 10^{-9}$	$(0.04 \pm 1.7)\ 10^{-9}$	$(-3.4 \pm 1.5)\ 10^{-9}$	$(4.8 \pm 3.1)\ 10^{-9}$
S-58	$(-2.2 \pm 2.0)\ 10^{-9}$	$(4.2 \pm 2.7)\ 10^{-9}$	$(-3.6 \pm 0.8)\ 10^{-9}$	$(0.3 \pm 1.4)\ 10^{-9}$
S-59	$(-3.2 \pm 1.3)\ 10^{-9}$	$(4.5 \pm 2.7)\ 10^{-9}$	$(-3.4 \pm 0.7)\ 10^{-9}$	$(1.1 \pm 1.6)\ 10^{-9}$
S-60	$(-8.8 \pm 4.8)\ 10^{-9}$	$(15.9 \pm 5.5)\ 10^{-9}$	$(-2.5 \pm 1.5)\ 10^{-9}$	$(5.2 \pm 3.1)\ 10^{-9}$

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

TABLE 10

VEGETATION RADIOACTIVITY DATA — 1981

Area	Activity	No. Samples	Gross Radioactivity (μCi/g): Dry Weight Annual Average Value	Gross Radioactivity (μCi/g): Ash: Annual Average Value (Standard Deviation)	Gross Radioactivity (μCi/g): Ash: Maximum Value[a] and Month Observed	% of Samples with Activity <MDL[b]
Onsite (monthly)	α	144	$(0.03 \pm 0.04)\ 10^{-6}$	$(0.16 \pm 0.20)\ 10^{-6}$	1.23×10^{-6} (January)	51
	β	144	$(21.2 \pm 9.6)\ 10^{-6}$	$(137.0 \pm 52.0)\ 10^{-6}$	257.0×10^{-6} (July)	0
Offsite (quarterly)	α	48	$(0.04 \pm 0.07)\ 10^{-6}$	$(0.18 \pm 0.17)\ 10^{-6}$	0.62×10^{-6} (January)	44
	β	48	$(26.1 \pm 13.9)\ 10^{-6}$	$(129.0 \pm 58.0)\ 10^{-6}$	221.0×10^{-6} (April)	0

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

TABLE 11

SSFL SITE — DOMESTIC WATER RADIOACTIVITY DATA — 1981

Area	Activity	No. Samples	Gross Radioactivity (μCi/ml): Average Value (Standard Deviation)	Gross Radioactivity (μCi/ml): Maximum[a] Value and Month Observed
ESG-SSFL (monthly)	α	24	$(0.11 \pm 0.12)\ 10^{-9}$	0.44×10^{-9} (December)
	β	24	$(2.79 \pm 0.55)\ 10^{-9}$	3.65×10^{-9} (October)

[a]Maximum value observed for single sample.

TABLE 12

BELL CREEK AND ROCKETDYNE SITE RETENTION POND RADIOACTIVITY DATA — 1981

Area (Monthly)	Activity	No. Samples	Gross Radioactivity Concentration: Average Value (Standard Deviation)	Maximum[a] Value and Month Observed	% of Guide[b]	% of Samples with Activity MDL[c]
Bell Creek mud no. 54 (μCi/g)	α	12	$(0.59 \pm 0.16)\ 10^{-6}$	0.77×10^{-6} (December)	NA	0
	β	12	$(23.5 \pm 2.1)\ 10^{-6}$	27.3×10^{-6} (November)	NA	0
Pond R-2A mud no. 55 (μCi/g)	α	12	$(0.79 \pm 0.23)\ 10^{-6}$	1.12×10^{-6} (December)	NA	0
	β	12	$(25.5 \pm 3.2)\ 10^{-6}$	32.9×10^{-6} (February)	NA	0
Bell Creek vegetation no. 54 (μCi/g-ash)	α	12	$(0.07 \pm 0.07)\ 10^{-6}$	0.25×10^{-6} (September)	NA	83
	β	12	$(103.0 \pm 45.1)\ 10^{-6}$	188.0×10^{-6} (May)	NA	0
Bell Creek vegetation no. 54 (μCi/g dry weight)	α	12	$(0.01 \pm 0.01)\ 10^{-6}$	0.05×10^{-6} (September)	NA	83
	β	12	$(16.0 \pm 6.44)\ 10^{-6}$	31.6×10^{-6} (October)	NA	0
Bell Creek water no. 16 (μCi/ml)	α	12	$(0.05 \pm 0.09)\ 10^{-9}$	0.20×10^{-9} (January)	<0.001	100
	β	12	$(3.78 \pm 0.65)\ 10^{-9}$	5.0×10^{-9} (October)	1.3	0
Pond water no. 6 (μCi/ml)	α	12	$(0.05 \pm 0.08)\ 10^{-9}$	0.17×10^{-9} (February)	<0.001	100
	β	12	$(4.25 \pm 0.63)\ 10^{-9}$	5.26×10^{-9} (December)	1.4	0
SSFL pond R-2A water no. 12 (μCi/ml)	α	12	$(0.07 \pm 0.15)\ 10^{-9}$	0.37×10^{-9} (January)	<0.001	83
	β	12	$(5.16 \pm 1.22)\ 10^{-9}$	8.30×10^{-9} (December)	1.7	0

[a]Maximum value observed for single sample
[b]Guide: 5×10^{-6} μCi/mlα, 3×10^{-7} μCi/mlβ; 10 CFR 20 Appendix B, CAC 17, DOE Order 5480.1
[c]Minimum detection level: 0.23×10^{-9} μCi/mlα; 0.64×10^{-9} μCi/mlβ.
NA — not applicable, no Guide value having been established

TABLE 13

AMBIENT AIR RADIOACTIVITY DATA — 1981

Site Location (Continuous)	Activity	No. Samples	Average Value (Standard Deviation)	Maximum[a] Value and Date Observed	% of Guide[b]	% of Samples with Activity < MDL
De Soto Onsite (µCi/ml)	α	704	$(<6.9 \pm 7.7)\ 10^{-15}$	2.6×10^{-14} (01/02)	<0.23	95[c]
	β		$(<1.2 \pm 0.2)\ 10^{-13}$	1.1×10^{-12} (03/01)	<0.04	10[c]
SSFL Onsite (µCi/ml)	α	1799	$(<6.8 \pm 7.9)\ 10^{-15}$	3.5×10^{-14} (11/18)	<11.3	96[c]
	β		$(<1.2 \pm 0.2)\ 10^{-13}$	1.1×10^{-12} (03/31)	<0.40	10[d]
SSFL sewage treatment plant Offsite (µCi/ml)	α	364	$(<6.9 \pm 7.4)\ 10^{-15}$	2.2×10^{-14} (01/24)	<11.5	94[c]
	β		$(<1.1 \pm 0.2)\ 10^{-13}$	5.5×10^{-13} (03/31)	<0.36	9[c]
SSFL Control Center Offsite (µCi/ml)	α	355	$(<6.8 \pm 7.0)\ 10^{-15}$	1.6×10^{-14} (02/23)	<11.3	96[c]
	β		$(<1.3 \pm 0.2)\ 10^{-13}$	1.6×10^{-12} (01/02)	<0.12	10[c]

[a]Maximum value observed for single sample

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

[c]MDL = 6.4×10^{-15} µCi/ml α — Individual daily samples with activity levels of 0 to 6.4×10^{-15} µCi/ml are recorded and averaged as 6.4×10^{-15} µCi/ml

[d]MDL = 1.3×10^{-14} µCi/ml β — Individual daily samples with activity levels of 0 to 1.3×10^{-14} µCi/ml are recorded and averaged as 1.3×10^{-14} µCi/ml. Indicated average values are upper limits, since some data were below the minimum detection levels.

TABLE 14

DE SOTO AND SSFL SITES — AMBIENT RADIATION DOSIMETRY DATA — 1981

TLD Location	Quarterly Exposure (mR)				Annual Exposure (mR)	Equivalent Exposure Rate (μR/h)
	Q-1	Q-2	Q-3	Q-4		
1. De Soto	35	32	32	40	140	16
2. De Soto	36	35	32	41	144	16
3. De Soto	33	29	31	47	140	16
4. De Soto	36	36	44	43	159	18
5. De Soto	36	29	32	39	136	15
6. De Soto	40	33	33	48	154	17
7. De Soto	35	31	31	38	135	15
Mean value					144 ± 9	16.1
1. SSFL	39	37	35	48	159	18
2. SSFL	43	33	36	54	166	19
3. SSFL	44	40	46	58	188	21
4. SSFL	(44)[a]	40	39	50	173	20
5. SSFL	39	31	36	44	150	17
6. SSFL	37	31	30	39	137	16
Mean value					162 ± 18	18.5
1. Offsite control	38	32	36	32	138	16
2. Offsite control	39	33	35	55	162	18
3. Offsite control	35	31	33	33	132	15
4. Offsite control	44	34	37	37	152	17
5. Offsite control	40	36	35	45	156	18
Mean value					148 ± 13	16.8

[a]Missing dosimeter, assumed value

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 would provide good agreement between the three data sets.

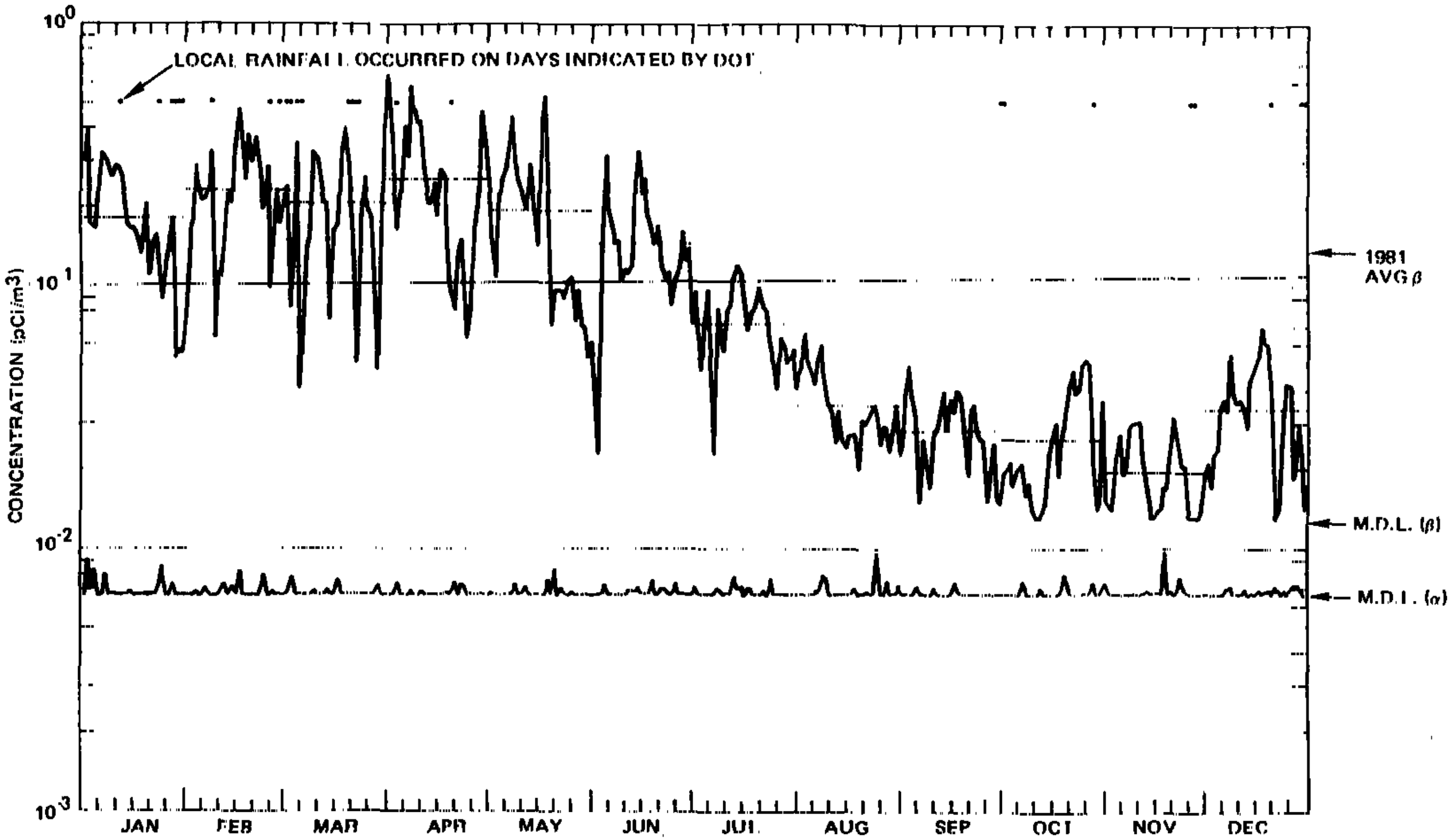

Figure 1. Daily Averaged Long-Lived Airborne Radioactivity at the De Soto and Santa Susana Field Laboratories Sites 1981

V. UNUSUAL EVENTS

There were a number of unusual events at facilities performing licensed activities. None of these events was reportable under the licence requirements. These events at each of the facilities are summarized below.

Fuels Area (Building 001)

On January 9, 1981, a powder room worker's breathing zone air sampler was 706 dpm, which increased the individual's total exposure for the week to 2.8×10^{-9} μCi-h/cc or 70% of the 40-h time-weighted occupational exposure limit. The cause of the elevated exposure could not be attributed to any specific event. The worker indicated that he had not performed any tasks involving nuclear material since the inventory performed on January 5, 1981.

A radioisotope shipment received from CEN/SCK Laboratories, Mol, Belgium, on February 24, 1981, was stored in the SNM vault pending transfer to the user. On March 3, 1981, the shipping container was surveyed for external radiation levels, which were approximately 700 mR/h surface and 15 mR/h at 3 ft. The appropriate limits are 200 mR/h and 10 mR/h, respectively. The NRC was informed of the shipment; a review of the shipment including routing, packaging, and external radiation levels was conducted by NRC as part of an unannounced compliance inspection. Although the circumstances of the shipment were nonreportable, since it was an international shipment, the NRC holds jurisdiction for investigating and, if deemed necessary, reporting the circumstances to the IAEA.

On August 14, 1981, at 0850, fuel production area personnel were evacuated to EAA-1 due to a high-radiation alarm from the fuel vault RAS. The alarm was inadvertently actuated by a Plant Services electrician who was performing routine calibration of the RAS unit with a radiation source. The electrician notified the Protective Services Control Center that the alarm would be tested and then proceeded before the Control Center operator could disable the alarm annunciator system within the building.

A powder room glove box operator's breathing zone (lapel) air sampler filter had a total count of 2162-dpm alpha for October 12, 1981. The individual was restricted from further work in the powder room pending completion of bioassay analyses to evaluate any incurred internal exposure. The blend glove box stationary air sampler was then counted and had a decayed total count of 8830-dpm alpha, which is greater than the 40-h limit for that work station. A smear survey of the blend box front panel indicated 1100 dpm/100 cm^2 removable alpha contamination. The glove box was found to be positively pressurized and was, therefore, removed from service pending repairs.

On investigation, an electrical problem was found in the photohelic gauge which controls the glove box pressure level. The vacuum and pressure solenoid valves were not being properly actuated; therefore, the glove box was continually pressurized, resulting in external contamination. The results of urinalyses for uranium and of an In-Vivo count given on October 21, 1981, to the individual involved showed that no internal exposure occurred, and he was released from the powder room work restriction.

NMDF (Building 055)

On May 30, at 1920 h, the SSFL Protective Service Control Center received a stack monitor failure alarm. The vacuum pump serving the stack monitor had failed. Repair was arranged by facility management and was completed by noon on May 31. The vacuum pump was not reconnected to the stack monitor at completion of repair. This condition was corrected 3 or 4 days later when found. No release of material from the facility occurred during the nonmonitoring interval due to the low level of operations within the facility, and alternate facility engineering safeguards were normal. The recommendations presented in the incident summary were (1) HS&RS should have been notified of the vacuum pump failure and (2) Operations should be familiar with corrective measures.

A high-airborne radioactivity alarm for the glove box room was received at the Protective Services Control Center on the morning of November 7, 1981. On entry, the Eberline Alpha-3 air monitor for Station C was in "alarm" mode. Smear and direct survey of the area around the air monitor showed no radioactive contamination.

RIHL (Building 020)

On August 21, 1981, at approximately 1430, three facility workers had loaded bagged contaminated equipment from the Hot Storage room into a 19A disposal box located in the airlock. The routine personal survey subsequently performed detected nasal, face, and hair radioactivity contamination up to 440 dpm on a nares wipe. All personal contamination was reduced to below 50 dpm, and post-exposure bioassay analyses indicated that no significant internal exposure was incurred by any of the individuals involved.

On September 9, 1981, at 1919, a high-radiation alarm for the T/020 Liquid Waste Facility was received by the SSFL Protection Services Control Center. No abnormal radiation levels were observed within the facility during the emergency response entry, and the facility was pronounced all clear at 2038. The exact cause was unknown, however, an electrical power dip was reported at the time of the alarm, and also Pacific Telephone employees were working onsite at that time.

On October 20, 1981, a ladder being used in an unposted area was monitored prior to release to another facility. A maximum of 30 mrad/h fixed and 5000 dpm/100 cm^2 beta contamination was found. The ladder had previously been used in painting the service gallery, Decon Cell 4 area. The ladder was covered with plastic and returned to a posted area. All individuals involved, and the area involved, were monitored. No additional significant contamination was found.

VI. SUMMARY/TRENDS — EXPOSURE, EFFLUENTS

A review of the data of this and previous years shows:

1) The upward trend from 1975 through 1978 in the occupational population dose has been reversed in 1979, 1980, and 1981 as shown below:

Year	Population Dose
1975	27 man-rem
1976	48 man-rem
1977	67 man-rem
1978	110 man-rem
1979	91 man-rem
1980	52 man-rem
1981	45 man-rem

This decrease in part probably reflect the somewhat lessened overall activity; however, it does substantiate that the controls provided continue to be effective.

2) The annual average personnel exposures over the last 7 years were:

Year	Average Exposure
1975	115 mrem
1976	200 mrem
1977	312 mrem
1978	198 mrem
1979	205 mrem*
1980	149 mrem
1981	113 mrem

*This number was incorrectly reported in Reference 6.

The average personnel exposure is computed on the basis that only those personnel who actually received an exposure were radiation workers subject to exposure. This eliminates many personnel that were provided film badges during each year because of duties for which exposure was possible, but did not receive any exposure. This average exposure is small (currently less than 3% of the occupational limit of 5 Rem/yr) and has not shown any significant trends.

3) Engineered and operational controls continue to effectively limit releases of radioactive materials to the environment. The summary of annual atmospherically discharged effluents and liquid effluent radioactivity for 1975-1981 is presented below:

Atmospherically Discharged Effluent Radioactivity (curies)

1975	9.4×10^{-5}	α
	6.7×10^{-3}	β,γ
1976	7.3×10^{-5}	α
	3.3×10^{-5}	β,γ
1977	1.1×10^{-5}	α
	2.5×10^{-5}	β,γ
1978	$<1.7 \times 10^{-5}$	α
	$<6.8 \times 10^{-5}$	β,γ
1979	$<2.2 \times 10^{-5}$	α
	$<5.8 \times 10^{-5}$	β,γ
1980	$<6.7 \times 10^{-6}$	α
	$<2.5 \times 10^{-5}$	β,γ
1981	$<3.0 \times 10^{-5}$	α,β

Liquid Effluent Discharged to Sanitary Sewer (Curies)

1975	6.9×10^{-5} α
	3.3×10^{-4} β,γ
1976	$<1.6 \times 10^{-4}$ α
	$<3.4 \times 10^{-4}$ β,γ
1977	1.1×10^{-5} α
	2.5×10^{-5} β,γ
1978	$<1.0 \times 10^{-4}$ α
	3.3×10^{-4} β,γ
1979	$<9.9 \times 10^{-5}$ α
	$<2.1 \times 10^{-4}$ β,γ
1980	$<1.8 \times 10^{-4}$ α
	$<4.2 \times 10^{-4}$ β,γ
1981	$<8.4 \times 10^{-5}$ α
	$<2.3 \times 10^{-4}$ β

From the above listings, it can be seen that the radioactivity content of both liquid and atmospheric discharges has remained relatively constant over the period 1975-1981, accounting to a few, or at most a few hundreds of microcuries per year. *No trends can be discerned from these discharge data that are meaningful;* however, the controls provided are effective and adequate.

VII. ANTICIPATED ACTIVITIES DURING NEXT REPORTING PERIOD (1982)

Building 001/004

Current production run of test reactor fuel elements will be completed. No further production of fuel elements is planned.

Building 020

Continue the SEFOR fuel decladding program. May initiate Saxton/EBWR fuel decladding program.

Building 055

This facility is inactive. Decommissioning may be proposed.

REFERENCES

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

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, N01TI990003, April 2, 1980

4. "Annual Review of Radiological Controls — 1977," R. S. Hart, Energy Systems Group, Rockwell International, N01TI000098, May 27, 1980

5. "Annual Review of Radiological Controls — 1978," R. S. Hart, Energy Systems Group, Rockwell International, N01TI000104, August 6, 1980

6. "Annual Review of Radiological Controls — 1979," R. S. Hart, Energy Systems Group, Rockwell International, N01TI000113, September 15, 1980

7. "Annual Review of Radiological Controls — 1980," R. R. Eggleston, Energy Systems Group, Rockwell International N001TI000144, July 14, 1981

8. "Energy Systems Group Environmental Monitoring and Facility Effluent Annual Report — 1981," J. D. Moore, Rockwell International, Energy Systems Group, ESG-82-21, July 15, 1982

APPENDIX

A. PERSONNEL MONITORING PROGRAM

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

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

<u>Exposure Range</u>

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

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

c) Beta (>1.5 MeV — 40 m rad — 100 rad ± 20% or 30 m rad, 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" absorbers consistent with the radiation requirements selected by the user.

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

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

3) Sources which film are calibrated to:

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

4) Criticality Function

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

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

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

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

Options

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

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

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

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

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

In the event of an accidental criticality incident, the film badge holder also contains additional components for the measuring of high-level gamma and neutron exposures generally associated in this type incident. Excessive film blackening prevents the microscopic identification of proton tracks. Therefore, neutron exposures above 10 rad are determined by means of sulfur pellets, gold and indium foils, and a copper washer which are incorporated into the film holder.

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

(1) Social Security Number

(2) Name

(3) Date of Birth

(4) Badge Number

(5) Current Dose X + Gamma, Neutron, Beta

(6) Calendar Quarter Dose Penetrating, Nonpenetrating

(7) Calendar Year Dose Penetrating, Nonpenetrating

(8) Lifetime Dose Penetrating, Nonpenetrating

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

B. ANALYTICAL PROCEDURE SUMMARY FOR BIOASSAY BY URINALYSIS

The following summary of analytical procedures is limited to the most frequently performed 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 to ketone. The water solution is planchetted for alpha counting for the UR analysis. Fluorometric analysis requires that an appropriate aliquot of the water solution be removed prior to planchetting for pelletizing with NaF, LiF. The pellet is then analyzed for uranium with a fluorometer.

Mixed Fission Products (FP1)

Mixed fission products will precipitate from a basic oxalate media. By adjustment of pH and oxalate concentrations, those elements which are amphoteric, or which form oxalate complexes in the form of excess oxalate, will also precipitate. Alkali metals such as Cs^{137} will not precipitate. Also, volatile fission products such as I^{131} will be lost.

The precipitate is washed with NaOH and water and planchetted for counting.

Mixed Fission Products (FP2)

Same extraction procedure as FP1, however, the soluble oxalate precipitates are gamma counted for Cs^{137} and other gamma emitters. The results from its FP1 analysis and the FP2 analysis are summed and reported as a single value.

Mixed Fission Products (FP3)

Same as FP2 except that the oxalate insoluble results will be reported separately as FP3a and the oxalate soluble results will be reported separately as FP3b.

Plutonium (PUA, PUB)

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

After xoidation 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 electrofeposited 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 ali- quot 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 (GAla)

Specific for uranium and/or plutonium which is extracted from ashed urine salts using aluminum nitrate, tetrapropylammonium hydroxide, and methyl isobutyl ketone. Transuranics do not extract to any appreciable extent. Uranium and/or plutonium are electrodeposited on a stainless steel disc and autoradiographed.

Gross Alpha (GAlb)

Same as GAla except the extraction solution is planchetted and counted for alpha radiation with a proportional counter.

Gross Alpha (GA2)

Specific for all alpha emitters. Metabolized actinides are converted to states suitable for coprecipitation with alkaline earth phosphates by digesting the gross urine sample in 10% nitric acid. The actinides are coprecipitated with calcium phosphate by neutralizing the acid solution with ammonia. The precipitate is washed, planchetted, and counted for alpha radiation with a proportional counter.

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

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 autoradiographed.
Gross Alpha (1b)	GA1B	Uranium and plutonium	9 dpm	±50%	100 ml	Sample planchetted and proportional counted for alpha.

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
Gross Alpha (2)	GA2	All other alpha emitters including Th, Pa, U, Np, Pu, Am, Cm, Po, and Ra	15 dpm	±50%	100 ml	Sample planchetted and proportional counted for alpha.
Iodine-131	I131	Iodine-131	300 dpm	±50%	250 ml	Decay corrected to sampling date.

SUMMARY OF IN-VIVO COUNTING SERVICES AVAILABLE FROM HELGESON NUCLEAR SERVICES, INC.

Analysis Type	Listing Code	Analysis Specific For	Sensitivity/ 1500 ml	Accuracy at Minimum Sensitivity	Minimum Volume Required	Remarks
Fission Products	TBC	All gamma emitting radionuclides 0.06 <2.6 MeV in the total body		Nuclide dependent, i.e., Cs-137 = 2.8 nCi, Cr-51 = 28.7 nCi		Technique can identify centers of localized deposition due to detector scan over body.
IVLC	IVLC	U-235 in the lung		40 μgm Ux ≈ 16[illegible] MPLB		Femoral bone is used as background control. Results are reported separately for thorax and femoral counts.

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.