Badger

Rockwell International
Energy Systems Group

SUPPORTING DOCUMENT

GO NO.	S/A NO.	PAGE 1 OF	TOTAL PAGES	REV LTR/CHG NO SEE SUMMARY OF CHG	NUMBER
94132	00012	44	44	A	N001TI000144

PROGRAM TITLE
Health, Safety and Radiation Services

DOCUMENT TITLE
Annual Review of Radiological Controls – 1980

DOCUMENT TYPE	KEY NOUNS
Technical Information	Radiation Exposures, Effluents, NRC-Licensed Facilities

ORIGINAL ISSUE DATE	REL. DATE	APPROVALS	DATE
	10-12-83 RK	R. J. Tuttle	7/2/81

PREPARED BY/DATE	DEPT	MAIL ADDR
R. R. Eggleston	779	NB08

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

DISTRIBUTION

*	NAME	MAIL ADDR
*	K. Buttrey	LB15
*	R. Hart	JB05
*	V. Keshishian	LB34
*	W. Kittinger	NB02
*	L. Mountford	LB01
*	M. Remley	NB08
*	R. Tuttle (2)	NB13
*	J. Moore	NB13

779-B.83/sjd

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

ABSTRACT

Data on exposures of Energy Systems Group personnel at NRC-licensed facilities are presented for CY 1980. 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 continue 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 1980. This experience and the continued level of activity scheduled for 1981 indicates that there is no need for changes in the ESG radiological controls program.

RESERVED F

REV	SUMMARY OF CHANGE	APPROVALS AND DATE
A	Table 5 had shown values for airborne concentrations measured in "001 - Fuel Fab (Lapel)" that were actually 40-hr exposures (μCi-hr/cc). These have been revised to show time-weighted averages (μCi/cc) as labeled. Page 13, Line 6: Change 2.5×10^{-9} to 6.2×10^{-11}. Change 2.0×10^{-10} to 5.0×10^{-12}. Rel Date: 10-12-83	R. J. Tuttle M. E. Remley

CONTENTS

TABLES

TABLES

FIGURES

INTRODUCTION

Condition 23 of the Atomics Internatinal 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 1979[2][3][4][5][6] 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) Fuel Fabrications — 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 for some time period, regardless of future efforts (i.e., internal body deposits). Appendix A describes the Personnel Mounting Program and Appendix B describes the Analytical Procedure Summary for Bioassay by Urinalysis.

A. FILM/TLD DATA

Personnel external radiation exposures for the pertinent activities for the year are presented in Table 1 where the number of individuals within a specific annual dose range is shown along with the percentage of employees within each annual dose range or less and the man-rem contributed by each dose range value. These same data are plotted in Figure 1* to show the probability of receiving greater than any selected dose. It also should be noted (see Summary, Section VI) that all exposures were less than the 3 rem and were well below the annual occupational limit of 5 rem.

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

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

TABLE 1

PERSONNEL EXTERNAL EXPOSURES - 1980

Dose Range (rem)	P (People)	C.P. (Cumulative People)	C.P. (%)	Population - Dose* (Man-Rem)
No Meas. Exposure	540	540	60.67	0
<0.10	282	822	92.36	14.10
0.10 - 0.25	38	860	96.63	6.65
0.25 - 0.50	10	870	97.75	3.75
0.50 - 0.75	3	873	98.09	1.88
0.75 - 1.00	5	878	98.65	4.38
1.00 - 2.00	9	887	99.66	13.50
2.00 - 3.00	3	890	100.00	7.50
>3.00	None	890	Total	51.76

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

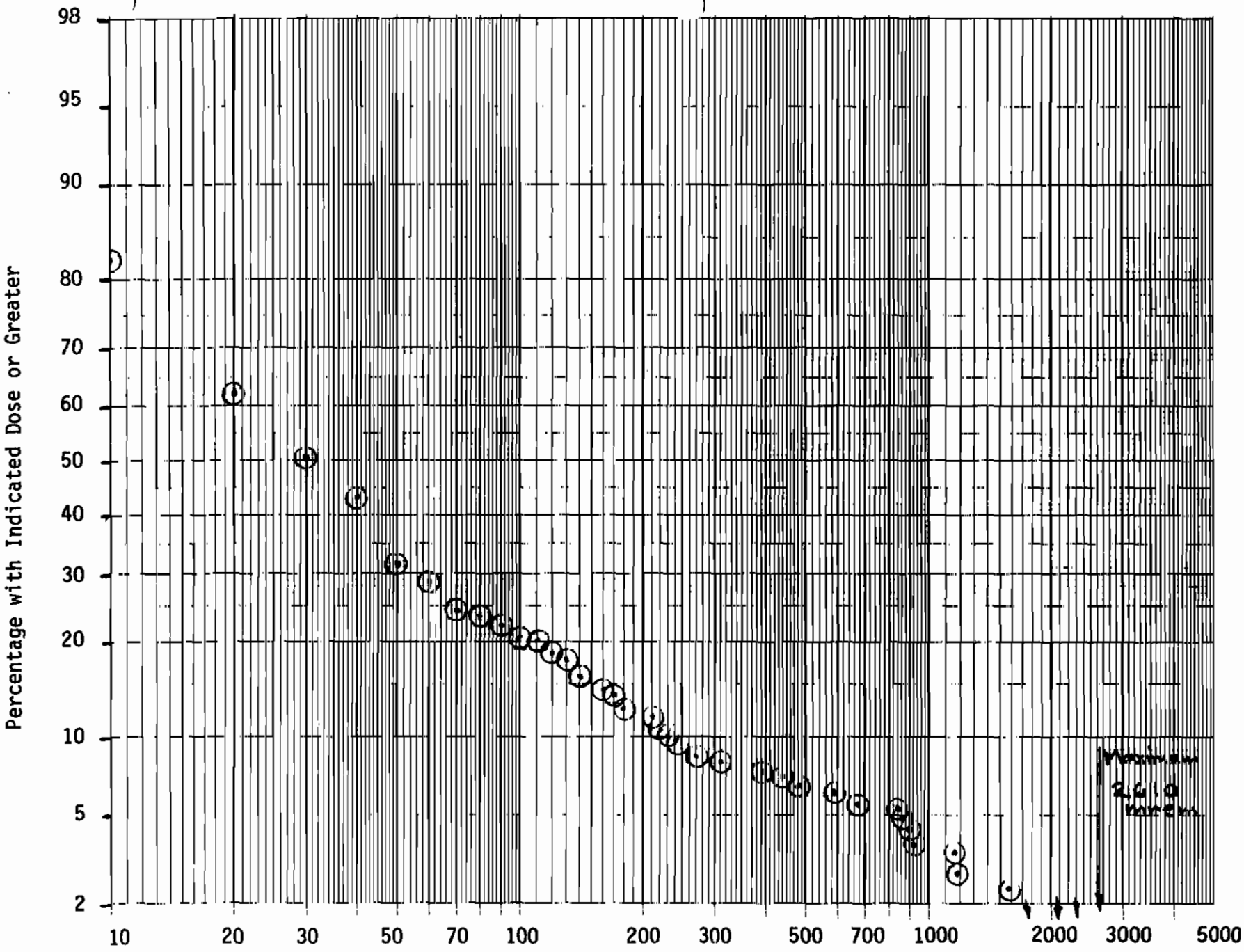

Figure 1. Distribution of Personnel External Radiation Exposures - 1980

the use of a whole body counter.* During 1980, 63 lung scans were made for uranium deposition. Seven of the scans (on six different individuals) showed positive results. However, followup scans showed a steady decrease of these lung burdens (see Table 3).

C. BIOASSAYS

Bioassays normally consist of analysis of urine and occasionally, fecal samples. Personnel whose work assignments potentially expose them to respirable-sized radioactive aerosols are routinely checked in this manner. Normally, urinalyses are performed quarterly and fecal analysis only when gross internal contamination is suspected. The techniques employed are described in the Appendix. A statistical summary of the results for 1980 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 1980 also appear in these tables.

*Helgeson Nuclear Services, Inc., Pleasanton, California

TABLE 2

SUMMARY OF BIOASSAYS - 1980

Measurement*	Type*	Total Tests	Total Positive Results	Total Individuals With Positive Results
U	UF	297	17	9
U	UR	196	9	3
GA	1A	0	0	0
GA	1B	0	0	0
GA	2B	62	0	0
GB	H	0	0	0
PU	A	9	0	0
FP	1	0	0	0
FP	3A	116	7	5
FP	3B	116	4	2
U	IVLC	63	7	6
Sr-90	Sr-90	0	0	0
Cs-137	TBC	0	0	0
H-3	H-3	5	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 - 1980
(Sheet 1 of 2)

H&S* Number	Sample Date	Analysis Type*	Results: Per Vol. Anal. (dpm)	Results: Per 1500 ml-day (dpm)	Specific Radionuclide	Equivalent MPBB (%)
4158	07/20/80	FP3A	11.7	87.8	Sr-90	18.3
4158	09/25/80	FP3A	37.6	282.0	Sr-90	58.8
2040	05/08/80	IVLC	43 μg		U-235	17.6**
4614	09/07/80	UR	0.67	5.03	U-235	2.5
4614	09/07/80	UF	0.0022 μg	3.30 μg	U	3.3
2817	05/04/80	FP3A	27.1	203.0	Sr-90	42.3
4576	03/02/80	FP3B	38.0	285.0	Cs-137	0.04
4576	04/16/80	FP3B	28.9	217.0	Cs-137	0.03
4572	02/06/80	IVLC	34.0 μg		U-235	15.7**
4650	05/08/80	IVLC	48.0 μg		U-235	19.6**
4650	11/10/80	IVLC	39.0 μg		U-235	14.7**
FAP	07/13/80	UF	0.0042 μg	6.3 μg	U	6.3
FAP	08/11/80	UF	0.0098 μg	14.7 μg	U	14.7
FAP	09/08/80	UR	1.13	8.48	U-235	4.2
FAP	09/08/80	UF	0.0083 μg	12.45 μg	U	12.45
FAP	09/15/80	UF	0.0067 μg	10.1 μg	U	10.1
FAP	09/15/80	UR	1.04	7.80	U-235	3.9
FAP	09/23/80	UF	0.0070 μg	10.5 μg	U	10.5
FAP	09/29/80	UF	0.0021 μg	3.15 μg	U	3.15
FAP	10/13/80	UF	0.0039 μg	5.91 μg	U	5.91
FAP	10/20/80	UF	0.0021 μg	3.16 μg	U	3.16
FAP	10/27/80	UF	0.0023 μg	3.45 μg	U	3.45
4393	05/08/80	IVLC	49.0 μg		U-235	20.0**
0307	11/10/80	IVLC	52.0 μg		U-235	21.2**
4212	04/28/80	UF	0.0012 μg	1.80 μg	U	1.8
3774	09/07/80	UF	0.0003 μg	0.45 μg	U	0.45
FAP	09/26/80	UF	0.0004 μg	0.60 μg	U	0.60
4660	06/28/80	UF	0.0003 μg	0.45 μg	U	0.45
4303	06/13/80	FP3A	6.3	47.3	Sr-90	9.8
0606	04/28/80	UR	1.11	16.7	U-235	8.4
0606	04/28/80	UF	0.0037 μg	5.55 μg	U	5.55
4617	10/27/80	FP3A	5.0	37.5	Sr-90	7.8

*FAP refers to an individual employed on the Frankford Arsenal Project, who came with a detectable deposition of depleted uranium.

TABLE 3

POSITIVE BIOASSAY RESULT SUMMARY - 1980
(Sheet 2 of 2)

H&S Number	Sample Date	Analysis Type*	Results: Per Vol. Anal. (dpm)	Results: Per 1500 ml-day (dpm)	Specific Radionuclide	Equivalent MPBB (%)
3072	05/08/80	IVLC	40.0 μg		U-235	16.3**
1298	09/09/80	UF	0.0003	0.45	U	0.45
4404	07/20/80	FP3B	36.1	271.0	Cs-137	0.04
4404	07/20/80	FP3A	17.4	131.0	Sr-90	27.3
4404	09/18/80	FP3B	21.8	163.5	Cs-137	0.02
4404	10/31/80	FP3A	29.1	218.0	Sr-90	45.4
4337	05/05/80	UF	0.0005 μg	0.75 μg	U	0.75

*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 – 1980

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

*Estimated

B. INTERIOR AIR SAMPLES – WORKING AREAS

In those working areas where the nature of the tasks being performed and of the materials in use might lead to the potential for generation of respirably-sized radioactive aerosols, periodic local air sampling is performed. A summary of these results for 1980 is given in Table 5.

TABLE 5

INTERIOR AIR SAMPLE SUMMARY — 1980

Building	Maximum µCi/cc	Average* µCi/cc
001-Fuel Fab (Lapel)	(α) 6.2×10^{-11}	(α) 5.0×10^{-12}
004	(Not Required)	--
020 Controlled Areas	(β,γ)8.5×10^{-12}	(β,γ)2×10^{-12}
Uncontrolled AReas	(β,γ)5.5×10^{-13}	(β,γ)1×10^{-13}
055	(α) 9.0×10^{-12}	

*Estimated

C. SPECIAL AIR SAMPLES — BUILDING 055

The NMDF (Building 055) was not utilized for Pu fuel fabrication or R&D work during 1980. 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. Consequently, the sampling was reduced to once a week. The same sampling locations were used. The maximum cumulative weekly exposure at any location was 9.0×10^{-12} µCi-hr/ml. The average weekly concentration (40 hr/wk) is 2.2×10^{-13} 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 0015-71. 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 1980 edition of that review[7] for atmospherically discharged and liquid effluents for the facilities identified above is presented in Tables 6 and 7, respectively.

TABLE 6

RADIOACTIVITY CONTENT OF ATMOSPHERICALLY DISCHARGED EFFLUENT RELEASED TO UNRESTRICTED AREAS – 1980

Building	Point of Release	Approximate Effluent Volume (ft^3)	Activity Monitored	Approximate Minimum Detection Limit ($\mu Ci/m\ell$)	Annual* Average Concentration ($\mu Ci/m\ell$)	Sampling Period Maximum Observed Concentration ($\mu Ci/m\ell$)	Radioactivity Released (Ci)
001	Stack Exit	1.9×10^{10}	α	1.9×10^{-16}	$<1.0 \times 10^{-14}$	1.8×10^{-13}	$<5.3 \times 10^{-6}$
			β,γ	6.1×10^{-16}	$<8.2 \times 10^{-15}$	9.4×10^{-14}	$<4.3 \times 10^{-6}$
004	Stack Exit	2.4×10^{10}	α	2.5×10^{-16}	$<1.2 \times 10^{-15}$	1.4×10^{-14}	$<1.0 \times 10^{-6}$
			β,γ	8.7×10^{-16}	$<7.3 \times 10^{-15}$	1.1×10^{-13}	$<4.9 \times 10^{-6}$
020	Stack Exit	1.3×10^{10}	α	0.9×10^{-16}	$<4.6 \times 10^{-16}$	1.4×10^{-15}	$<1.7 \times 10^{-7}$
			β,γ	3.0×10^{-16}	4.6×10^{-14}	2.4×10^{-13}	1.7×10^{-5}
055	Stack Exit	7.9×10^{9}	α	2.7×10^{-16}	$<3.7 \times 10^{-16}$	4.8×10^{-15}	$<8.2 \times 10^{-8}$
			β,γ	8.9×10^{-16}	$<4.9 \times 10^{-15}$	3.5×10^{-14}	$<1.1 \times 10^{-6}$

*Annual average ambient air radioactivity concentrations 1980

α $<6.4 \times 10^{-15}$ $\mu Ci/m\ell$

β,γ $<3.6 \times 10^{-14}$ $\mu Ci/m\ell$

Totals (all facilities) α $<6.7 \times 10^{-6}$ Ci

β,γ $<2.5 \times 10^{-5}$ Ci

TABLE 7

LIQUID EFFLUENT DISCHARGED TO SANITARY SEWER — 1980

Building	Point of Release	Approximate Effluent Volume (gal)	Activity Monitored	Approximate MDL (μCi/mℓ)	Annual Average Concentration (μCi/mℓ)	Sample Maximum Observed Concentration (μCi/mℓ)	Total Radioactivity Released (Ci)
001	Retention Tank	48,000	α	7.0×10^{-9}	$<7.3 \times 10^{-8}$	4.1×10^{-7}	1.3×10^{-5}
			β	3.7×10^{-9}	$<6.2 \times 10^{-8}$	2.8×10^{-7}	1.1×10^{-5}
004	Proportional Sampler	1,717,000	α	1.2×10^{-9}	$<2.4 \times 10^{-8}$	1.5×10^{-7}	$<1.6 \times 10^{-4}$
			β	3.7×10^{-9}	$<6.3 \times 10^{-8}$	6.5×10^{-7}	$<4.1 \times 10^{-4}$
020*	—	0	—	—	—	—	—
055*	—	0	—	—	—	—	—

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

IV. ENVIRONMENTAL MONITORING PROGRAM

The basic policy for control of radiological and toxicological hazards at ESG requires that through engineering controls adequate containment of such materials be provided and, through rigid operational controls, that facility effluent releases and external radiation levels are reduced to a minimum. The environmental monitoring program provides a measure of the effectiveness of the Group safety procedures and of the engineering safeguards incorporated into facility designs. Specific radionuclides in facility effluent or environmental samples, although not routinely identifed 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 Appendic 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 1980 edition of that review[7] is 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 fall-out 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 — 1980

Area	Activity	No. Samples	Gross Radioactivity (μCi/g) Annual Average Value (95% Confidence Level)	Gross Radioactivity (μCi/g) Maximum Observed Value*
On Site	α	144	$(6.0 \pm 1.5)\ 10^{-7}$	11.4×10^{-7}
	β	144	$(2.4 \pm 0.1)\ 10^{-5}$	11.0×10^{-5}
Off Site	α	48	$(5.8 \pm 1.5)\ 10^{-7}$	10.3×10^{-7}
	β	48	$(2.3 \pm 0.1)\ 10^{-5}$	3.0×10^{-5}

*Maximum value observed for single sample

TABLE 9

SOIL PLUTONIUM RADIOACTIVITY DATA — 1980

Sample Location	July 9, 1980, Survey Results Pu^{238} (μCi/g)	July 9, 1980, Survey Results $Pu^{239} + Pu^{240}$ (μCi/g)	December 22, 1980, Survey Results Pu^{238} (μCi/g)	December 22, 1980, Survey Results $Pu^{239} + Pu^{240}$ (μCi/g)
S-56	$(0.7 \pm 2.4)\ 10^{-9}$	$(0.5 \pm 1.3)\ 10^{-9}$	$(-0.8 \pm 1.4)\ 10^{-9}$	$(13.0 \pm 3.0)\ 10^{-9}$
S-57	$(1.4 \pm 3.4)\ 10^{-9}$	$(9.5 \pm 4.8)\ 10^{-9}$	$(-0.3 \pm 2.3)\ 10^{-9}$	$(5.6 \pm 3.2)\ 10^{-9}$
S-58	$(-0.2 \pm 2.4)\ 10^{-9}$	$(1.6 \pm 2.0)\ 10^{-9}$	$(0.4 \pm 3.1)\ 10^{-9}$	$(9.9 \pm 4.6)\ 10^{-9}$
S-59	$(-1.2 \pm 2.3)\ 10^{-9}$	$(8.2 \pm 3.8)\ 10^{-9}$	$(-0.8 \pm 2.1)\ 10^{-9}$	$(4.2 \pm 2.7)\ 10^{-9}$
S-60	$(-1.1 \pm 1.9)\ 10^{-9}$	$(1.7 \pm 1.8)\ 10^{-9}$	$(-1.6 \pm 1.7)\ 10^{-9}$	$(29.5 \pm 6.5)\ 10^{-9}$

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

TABLE 10

VEGETATION RADIOACTIVITY DATA – 1980

Area	Activity	No. Samples	Gross Radioactivity (μCi/g) Dry Weight Annual Average Value	Ash Annual Average Value (95% Confidence Level)	Ash Maximum Observed Value*
Onsite	α	144	$(<3.1 \pm 2.1)\ 10^{-8}$	$(<2.5 \pm 1.7)\ 10^{-7}$	1.3×10^{-6}
	β	144	$(2.1 \pm 0.04)\ 10^{-5}$	$(1.60 \pm 0.03)\ 10^{-4}$	2.71×10^{-4}
Offsite	α	48	$(<4.2 \pm 3.3)\ 10^{-8}$	$(<1.9 \pm 1.5)\ 10^{-7}$	5.2×10^{-7}
	β	48	$(3.1 \pm 0.06)\ 10^{-5}$	$(1.42 \pm 0.03)\ 10^{-4}$	2.21×10^{-4}

*Maximum value observed for single sample

TABLE 11

SSFL SITE – DOMESTIC WATER RADIOACTIVITY DATA – 1980

Area	Activity	No. Samples	Gross Radioactivity (μCi/mℓ) Average Value (95% Confidence Level)	Maximum* Observed Value
ESG-SSFL	α	24	$(<2.2 \pm 2.7)\ 10^{-10}$	$<2.2 \times 10^{-10}$
	β	24	$(2.4 \pm 0.7)\ 10^{-9}$	3.4×10^{-9}

*Maximum value observed for single sample

TABLE 12

BELL CREEK AND ROCKETDYNE SITE RETENTION POND
RADIOACTIVITY DATA — 1980

Area	Activity	No. Samples	Gross Radioactivity Concentration: Average Value (95% Confidence Level)	Gross Radioactivity Concentration: Maximum* Observed Value	% of Guide†
Bell Creek Mud No. 54 (μCi/g)	α	12	$(5.1 \pm 1.4)\ 10^{-7}$	9.3×10^{-7}	NA
	β	12	$(2.3 \pm 0.1)\ 10^{-5}$	3.4×10^{-5}	NA
Pond R-2A Mud No. 55 (μCi/g)	α	12	$(5.1 \pm 1.4)\ 10^{-7}$	7.4×10^{-7}	NA
	β	12	$(2.2 \pm 0.1)\ 10^{-5}$	2.4×10^{-5}	NA
Bell Creek Vegetation No. 54 (μCi/g ash)	α	12	$(<1.8 \pm 1.5)\ 10^{-7}$	3.6×10^{-7}	NA
	β	12	$(1.5 \pm 0.03)\ 10^{-4}$	2.22×10^{-4}	NA
Bell Creek Vegetation No. 54 (μCi/g) dry weight	α	12	$(<3.1 \pm 2.6)\ 10^{-8}$	1.1×10^{-7}	NA
	β	12	$(2.4 \pm 0.1)\ 10^{-5}$	4.4×10^{-5}	NA
Bell Creek Water No. 16 (μCi/mℓ)	α	12	$(<2.3 \pm 2.7)\ 10^{-10}$	$<2.3 \times 10^{-10}$	<0.005
	β	12	$(2.9 \pm 0.8)\ 10^{-9}$	5.2×10^{-9}	1.0
Pond Water No. 6 (μCi/mℓ)	α	12	$(<2.3 \pm 2.7)\ 10^{-10}$	$<2.3 \times 10^{-10}$	<0.005
	β	12	$(2.9 \pm 0.7)\ (10^{-9}$	4.7×10^{-9}	1.0
SSFL Pond R-2A Water No. 12 (μCi/mℓ)	α	12	$(<2.3 \pm 2.7)\ 10^{-10}$	$<2.3 \times 10^{-10}$	<0.005
	β	12	$(3.9 \pm 0.8)\ 10^{-9}$	5.7×10^{-9}	1.3

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

TABLE 13

AMBIENT AIR RADIOACTIVITY DATA – 1980

Site Location	Activity	No. Samples	Average Value (95% Confidence Level)	Maximum* Observed Value (daily)	% of Guide÷
De Soto Onsite (μCi/mℓ)	α§	685	$(<6.5 \pm 7.7)\ 10^{-15}$	4.5×10^{-14}	<0.22
	β**	685	$(<3.9 \pm 1.4)\ 10^{-14}$	3.8×10^{-13}	<0.013
SSFL Onsite (μCi/mℓ)	α§	1611	$(<6.4 \pm 7.8)\ 10^{-15}$	2.5×10^{-14}	<10.7
	β**	1611	$(<3.6 \pm 1.4)\ 10^{-14}$	4.5×10^{-13}	<0.12
SSFL Sewage Treatment Plant Offsite (μCi/mℓ)	α§	366	$(<6.3 \pm 7.6)\ 10^{-15}$	1.8×10^{-14}	<10.5
	β**		$(<3.2 \pm 1.6)\ 10^{-14}$	3.1×10^{-13}	<0.11
SSFL Control Center Offsite (μCi/mℓ)	α§	365	$(<6.3 \pm 7.9)\ 10^{-15}$	2.0×10^{-14}	<10.5
	β**		$(<3.7 \pm 1.4)\ 10^{-14}$	3.6×10^{-13}	<0.12

*Maximum value observed for single sample.

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

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

**MDL = 1.2×10^{-14} μCi/mℓ – Individual daily samples with activity levels of 0 to 1.2×10^{-14} μCi/mℓ are recorded and averaged as 1.2×10^{-14} μCi/mℓ. Indicated average values are upper limits, since some data were below the minimum detection levels.

- De Soto and SSFL Sites - Ambient Radiation Dosimetry Data presented in Table 14
- Daily Averaged Long-Lived Airborne Radioactivity at Headquarters and SSFL presented in Figure 2

Reference 7 should be utilized for the interpretation of some of this data in light of the dominance of fallout in influencing the results.

TABLE 14

DE SOTO AND SSFL SITES — AMBIENT RADIATION DOSIMETRY DATA — 1980

TLD Location*	Average Dose Rate (mRem/h)	Equivalent Annual Dose (mRem)
1 De Soto	0.019	165
2 De Soto	0.017	147
3 De Soto	0.016	144
4 De Soto	0.018	154
5 De Soto	0.019	166
6 De Soto	0.022	189
7 De Soto	0.020	179
1 SSFL	0.018	163
2 SSFL	0.019	166
3 SSFL	0.021	185
4 SSFL	0.020*	177
5 SSFL	0.018†	155
6 SSFL	0.018	162
1 Offsite Control	0.018	163
2 Offsite Control	0.019	165
3 Offsite Control	0.019	166

*Excludes second-quarter data due to missing dosimeter.
†Excludes third-quarter data due to dosimeter damage by brush fire.

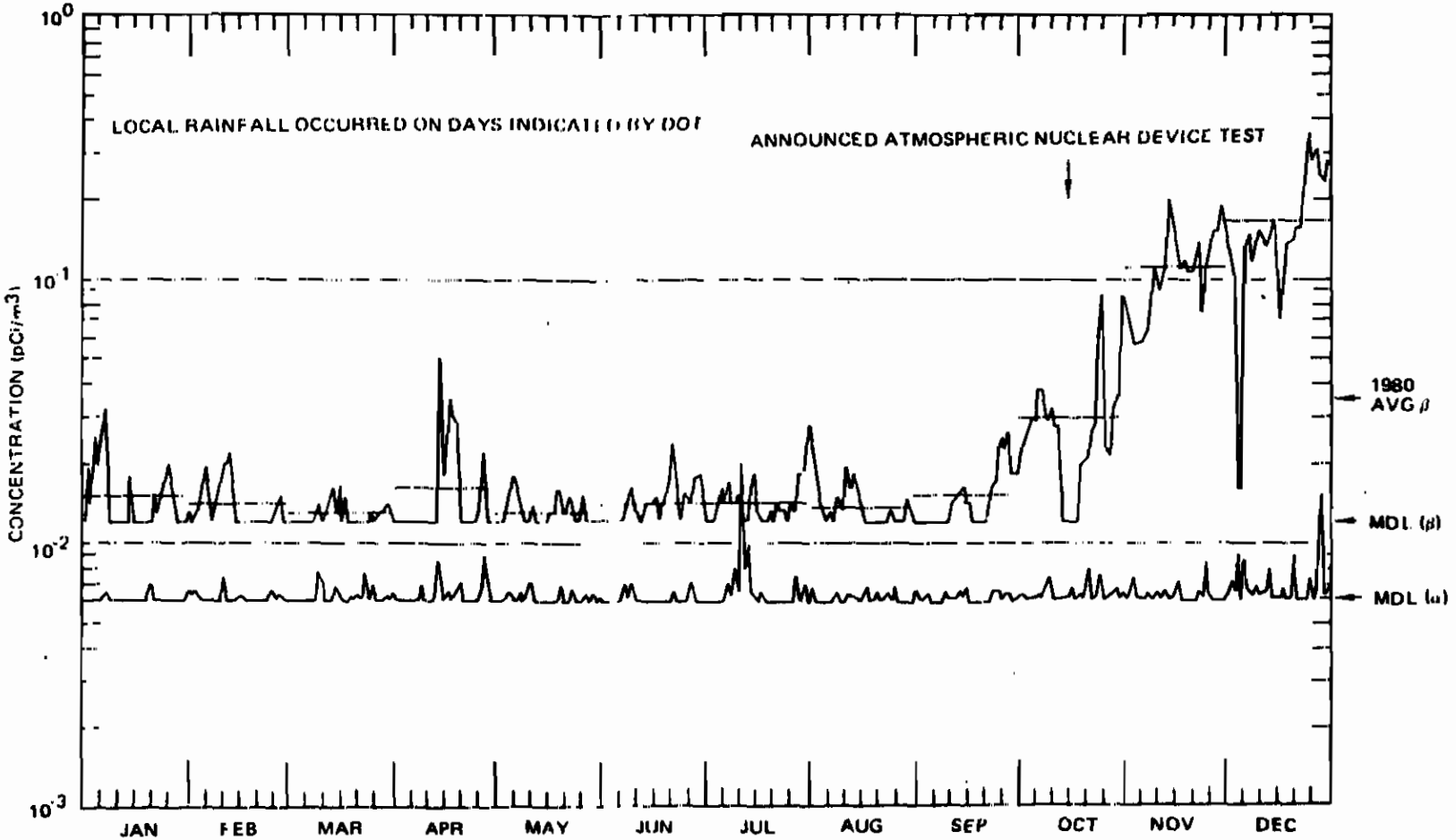

Figure 2. Daily Averaged Long-Lived Airborne Radioactivity at Headquarters and SSFL

V. UNUSUAL EVENTS

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

Fuels Area (Building 001)

On Wednesday, July 23, 1980, a lapel air sampler filter worn by a powder room employee had a count rate of 577 dpm alpha. The employee's weekly integrated exposure was 613 dpm alpha, equivalent to approximately 50% of the 40-hr time-weighted occupational exposure limit. The employee was restricted from powder room work for the remainder of the week. On investigation, it was determined that the employee was repairing a uranium powder blender without benefit of respiratory protection which would have lowered the apparent exposure to 1% of the weekly limit. Bioassay studies by excretion analysis and by in-vivo counting performed subsequent to the incident indicated that the employee incurred no resultant internal exposure to radioactive materials.

On Thursday, November 6, 1980, a powder room employee lapel air sampler filter had an activity level of 554 dpm alpha which resulted in an integrated exposure for the week of 706 dpm alpha. This value is equivalent to 2.2×10^{-9} $\mu Ci\text{-}hr/cm^3$ or 55% of the 40-hr time-weighted occupational exposure limit. On investigation is was determined the person involved was cleaning powder sieves contaminated with UAl_x and placing them in drums for disposal in a manner that produced a uranium aerosol. An in-vivo lung count of the person on November 10, 1980, indicated that he incurred an internal exposure equivalent to 14.7% of a maximum permissible lung burden for the uranium isotopic composition involved in the exposure. Additionally, bioassay by urinalysis specific for uranium which was performed after the exposure incident was negative, which further indicates that an acute internal exposure did not occur.

On Wednesday, November 12, 1980, the Alpha-3 continuous air monitor located next to the powder room crusher box alarmed at 0825. Two persons present in the powder room immediately left. The air monitor filter was recovered and counted for alpha radioactivity. The immediate count result of 1180 dpm confirmed that a uranium aerosol had been generated in the powder room. The crusher glove box was found to be slightly pressurized, and there were indications of a leak around the front window. Beyond this, all of the window bolts were loosened due to crusher vibration. Three bolts next to the air monitor were completely stripped. The crusher box window had been removed on October 24, 1980, for crusher repairs and was reportedly properly retorqued into position when reinstalled. Nasal smears for the last person to work the crusher were negative for uranium as were excretion samples analyzed specifically for uranium.

On Wednesday, November 19, 1980, a powder room employee was removing a tamper seal from a fuel can with a razor blade and accidentally cut a finger. An instrument and smear survey of the wound indicated no radioactive contamination present.

On Thursday, December 11, 1980, an ATR assembly room machinist caught his rubber-gloved right hand in the rotating "Tee Slot" cutter of the end assembly milling machine. The cutter blade went to and partially through the bone of the medial phalanx of the third finger. A smear and instrument survey of the blood, bone fragments, and tissue indicated no radioactive contamination, as expected, since the room is a radiologically clean area.

NMDF Building 055)

On Friday, October 31, 1980, the facility health physicist surveyed four pairs of green-lined coveralls that were returned from the laundry along with the blue-lined laundry normally worn in the facility. A small tar ball found inside the beltline of one garment was contaminated to 4.5×10^5 dpm of mixed fission products. The commercial laundry was notified of their washing and packaging error.

RIHL (Building 020)

None.

Chem Labs (Building 004)

None.

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 and 1980 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

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 6 years were:

Year	Average Exposure
1975	114 mRem
1976	200 mRem
1977	312 mRem
1978	198 mRem
1979	205 mRem*
1980	149 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 1975-1980 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

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

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 1975-1980, amounting 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 (1980)

Building 001/004

Continued fabrication of test reactor fuel elements with the possibility of increased activity (pending proposal success).

Building 020

Initiate the SEFOR 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. "Energy Systems Group Environmental Monitoring and Facility Effluent Annual Report — 1980, J. D. Moore, Rockwell International, Energy Systems Group, in preparation

APPENDIX

A. PERSONNEL MONITORING PROGRAM

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

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

Exposure Range

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

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

c) Beta (>1.5 MeV — 40 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 on the dose. All special badges are processed at ESG by the Radiation and Nuclear Safety Group.

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

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

(1) Social Security Number

(2) Name

(3) Date of Birth

(4) Badge Number

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

(6) Calendar Quarter Dose Penetrating, Nonpenetrating

(7) Calendar Year Dose Penetrating, Nonpenetrating

(8) Lifetime Dose Penetrating, Nonpenetrating

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

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

Gross Alpha (GA1a)

Specific for uranium and/or plutonium which is extracted from ashed urine salts using aluminum nitrate, tetrapropylammonium hydroxide, and methyl isobutyl ketone. Transuranics doe 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 counted for alpha radiation with a proportional counter.

Some date pertinent to these bioassay services are shown in 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 oxalated including alkaline earths, transition elements, lanthanides, antimony, phosphates. Excludes soluble oxalates, i.e., Cs 137	30 dmp	±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.

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.

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%MPLB		Femoral bone is used as background control. Results are reported separately for thorax and femoral counts.

779-B.83/dmb

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.