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SURPLUS FACILITIES MANAGEMENT PROGRAM

POST REMEDIAL ACTION SURVEY REPORT FOR BUILDING 003 SANTA SUSANA FIELD LABORATORIES ROCKWELL INTERNATIONAL VENTURA COUNTY, CALIFORNIA

OCTOBER 1981

APRIL 1982

OCCUPATIONAL HEALTH AND SAFETY DIVISION
Health Physics Section
ARGONNE NATIONAL LABORATORY, ARGONNE, ILLINOIS

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Argonne, Illinois 60439

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October 1981
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October 1983

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PREFACE AND EXECUTIVE SUMMARY

Rockwell International's Santa Susana Laboratories in Ventura County, California, have been the site of numerous Federally-funded projects involving the use of radioactive materials. One such project was the System for Nuclear Auxiliary Power (SNAP) Program, which was conducted under the auspices of the U.S. Atomic Energy Commission (AEC). Building 003 on the Santa Susana site was used in conjunction with the SNAP Program and contained a highly shielded area designed for remote manipulation of radioactive materials. Such facilities are commonly referred to as "hot caves."

During the SNAP Program, fuel burnup samples were analyzed and irradiation experiments were evaluated in the Building 003 hot cave. Use of the hot cave facility ended when the SNAP Program was terminated in 1973. Subsequently, the Building 003 facilities were declared excess (i.e., surplus facilities) and were decontaminated and decommissioned during the first half of calendar year 1975. At that time, the building was given a preliminary release.

In 1981, a post-remedial-action (certification) survey of Building 003 was conducted at the request of the Department of Energy by the Radiological Survey Group (RSG) of the Occupational Health and Safety Division, Health Physics Section (OHS/HP) of Argonne National Laboratory (ANL). Significant levels of residual contamination were found in various parts of the building. Consequently, additional decontamination was conducted by Rockwell International. A final post-remedial-action survey was conducted by the ANL-RSG during April 1982, and those areas in Building 003 that had been found contaminated in 1981 were now found to be free of detectable radioactive contamination.

Sludge samples taken from the sewer sump showed elevated levels of enriched uranium contaminant. Hence, all sewer lines within Building 003 were removed. This permitted unconditional release of the building for unrestricted use. However, the sewer lines exterior to the building, which remain in place, must

be considered potentially contaminated and, therefore, subject to restricted use.

This survey was performed under the auspices of the Health Physics Section of the Occupational Health and Safety Division of Argonne National Laboratory, Argonne, Illinois. The following personnel participated: R. A. Wynveen, W. H. Smith, R. L. Mundis*, C. M. Sholeen, A. L. Justus, K. F. Flynn, J. D. Thereon, R. Rodriguez, and D. W. Reilly.

*Now at Los Alamos Scientific Laboratory.

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POST REMEDIAL ACTION SURVEY REPORT FOR BUILDING 003 SANTA SUSANA FIELD LABORATORIES, ROCKWELL INTERNATIONAL

INTRODUCTION

The Santa Susana Laboratories, located in the Santa Susana Mountains in Ventura County northwest of Los Angeles, California (Fig. 1), are owned by Rockwell International and have been used for a number of government-sponsored projects involving radioactive material. One of the buildings on the laboratory site, Building 003, contains a "hot cave" facility that was used in conjunction with the System for Nuclear Auxiliary Power (SNAP) Program conducted for the Atomic Energy Commission. The location of Building 003 in relation to the rest of the site is shown in Figure 2, and a floor plan of the building is shown in Figure 3.

The hot cave in Building 003 has been inactive since the closeout of the SNAP Program in 1973. Prior to that time, the facility had been used for the analysis of SNAP fuel burnup samples and the evaluation of irradiation experiments. After Building 003 facilities were declared excess, Rockwell International carried out decontamination and dismantling of the contaminated facility.[1] The actual decontamination and decommissioning efforts in Building 003 began in January 1975 and ended in June 1975. At that time, the building was given a preliminary release.

A post-remedial-action (certification survey) of the hot cave facility in Building 003 was undertaken in October 1981 on behalf of the Department of Energy by the Radiological Survey Group (RSG) of the Occupational Health and Safety Division, Health Physics Section (OHS/HP), of Argonne National Laboratory (ANL). That survey showed residual contamination in various parts of the building; therefore, additional decontamination work was conducted by Rockwell International. This decontamination work was concluded in March 1982, and the results of a comprehensive radiological survey of the building conducted at the time by Rockwell International personnel were documented in a Rockwell report.[2]

A final (confirmatory) post-remedial-action survey was conducted by the ANL Radiological Survey Group during April 1982. At that time, only those areas that previously had been found contaminated were resurveyed.

The results of the surveys of Building 003 by the ANL-OHS/HP Radiological Survey Group, and the investigators' conclusions and recommendations, are presented in this report. The results of investigations of the SNAP Facility, the KEWB Facility, and the SRE Facility (also located at the Santa Susana site), are included in companion reports.

RADIOLOGICAL SURVEY PROCEDURES

Instrumentation

Four types of portable survey instruments were used to conduct the direct radiological surveys in Building 003. Gas-flow proportional detectors with window areas of 51 cm^2 and 325 cm^2 (using Eberline PAC-4G-3 electronics) were used to monitor for alpha and/or beta-gamma radiation. NaI crystal detectors, 2 in. diameter by 2 mm thick (Eberline PG-2 with Eberine PRM-5-3 electronics), were used to monitor for low energy x-ray and gamma radiation. NaI crystal detectors, measuring 1 in. diameter by 1 in. thick (Eberline PRM-7 μR meter) and calibrated with a ^{226}Ra standard source, were used to measure the ambient external penetrating radiation field in units of μR/h. An end-window Geiger-Mueller (GM) detector (Eberline HP-190 with a 7 mg/cm^2 window and Eberline 530 electronics), calibrated with a ^{226}Ra standard source, was used to measure the contact exposure rate (mR/h) of contaminated areas. Integrated measurements of the ambient penetrating radiation exposure field were taken with a pressurized ionization chamber (Reuter Stokes RSS-111) calibrated with an NBS traceable ^{60}Co gamma-ray source. These instruments and associated calibration procedures are detailed in Appendicies 1 and 2. Since instrument calibrations were to infinitely-thin flat-plate standards, all reported readings should be regarded as minimal values; no corrections were made for absorption by surface media.

Whenever possible, a contaminant was identified by performing gamma spectral analysis on either the contaminated item or on a sample of material taken from a contaminated area. These analyses were performed with a sodium iodide or hyperpure germanium detector coupled to a multichannel analyzer. This instrumentation is also described in Appendix 1.

Smear Surveys

Dry smears were taken at representative locations throughout the building using a 4.75-cm-diameter filter paper (Whatman #1). A smear sample was obtained by applying moderate pressure with the tips of the first two fingers to the back of the filter paper and wiping the surface over an area of ~ 900 cm^2. Smears were taken on original structures and components such as walls, floors, pipes, and vents. A smear of 100 cm^2 was taken from an area or object indicated by a portable survey instrument to have a higher than normal radiation level. A smear area of 100 cm^2 was also taken if the surface was extremely dusty.

To expedite counting of the numerous smear samples, two counting techniques were employed using two types of counters. A large-area, thin-window, gas-flow proportional counter, sensitive to alpha and/or beta-gamma radiation, was used to make an initial count on groups of smears. For more sensitive counts on individual smears, a Nuclear Measurement Corporation Model PC-5 (or 3A), Internal Gas-Flow Proportional Counter (PC counter) with a thin aluminized Mylar window (referred to as a Mylar spun top) was used.

Initial counts were made with the large-area counter on groups of ten smears at one time. Any set of smears indicating a reading above the instrument background was then counted individually in the PC counter. In addition, at least one smear from each group of ten was selected at random and counted in the PC counter. All smears of the areas or objects with elevated direct readings were counted individually in the PC counter. A more detailed description of the instruments and of the counting and calibration techniques used is presented in Appendix 1.

Air Samples

Air-particulate samples were collected using a commercial vacuum cleaner to pull air through filter media (HV-70). A total volume of 26.7 m^3 of air was sampled at a flow rate of 40 m^3/h. A 10% portion (5 cm in diameter) was removed from the filter media after collection and counted for both alpha and beta-gamma activity in the PC counter. Concentrations of radon (^{222}Rn), and thoron (^{220}Rn), and the presence of any long-lived airborne radionuclides, were determined based on the result of several counts at specified intervals on each sample.

Air-particulate samples were also collected on Millipore filter media for 40 minutes at a flow rate of approximately 1.5 m^3/h. A portion of each filter sample was used for alpha spectral analysis to determine the actinon (^{219}Rn) concentration.

Details of the air-sampling techniques and associated calculations are given in Appendix 3.

Sludge, Water, Soil, and Tile Samples

Sludge, water, soil, and tile samples for radiochemical analysis were taken from several areas at the site suspected of being contaminated. All samples were prepared at ANL as detailed in Appendix 4 and submitted to the Analytical Chemistry Laboratory at ANL for radiochemical and gamma spectral analyses.

RADIOLOGICAL SURVEY RESULTS

Instrument and Smear Surveys

The October 1981 post-remedial-action survey of the interior of Building 003 indicated several areas with significant levels of radioactive contamination. Radiation levels measured were as high as 10 k dis/min-100 cm^2 surface alpha (α) activity, 225 k dis/min-100 cm^2 surface alpha-beta-gamma ($\alpha\beta\gamma$) activity, 125 k cts/min low-energy x and gamma (γ) radiation, and 11 $\mu R/h$ ambient radiation level at 3 ft as determined with a μR meter.* The areas of contamination were limited to Rooms 101, 102, 180, 200, 205, and the High Bay Area (see Fig. 3). All other areas of the building interior were free from measurable contamination. All measurements of ambient radiation made with the pressurized ion chamber (RSS-111) were at background levels.

Some contamination (up to 500 cts/min-51 cm^2 surface $\alpha\beta\gamma$) was measured on the south roof (exterior) in the vicinity of the blowers. There was no contamination detected on the north roof. A detailed survey of the outside perimeter of the building revealed no detectable contamination. These survey results are delineated in Table 1.

*The direct measurement results presented in this report are gross readings. Background radiation levels have not been subtracted, nor have conversion factors been applied for specific radioisotopes (see Table 1).

After the October 1981 radiological survey, all areas of contamination were cleaned of residual radioactivity by Rockwell International personnel. The ANL Radiological Survey Group resurveyed those areas during April 1982, and found them to be free of contamination.

Air Samples

Short-term air samples were collected at seven locations within the building to determine if any airborne contamination was present. These samples revealed Working Levels (WL) for radon (^{222}Rn) decay-product concentrations ranging from 0.00064 to 0.0025; radon (^{222}Rn) concentrations ranging from 0.064 pCi/ℓ to 0.25 pCi/ℓ; thoron (^{220}Rn) concentrations ranging from 0.0012 pCi/ℓ to 0.015 pCi/ℓ; and actinon (^{219}Rn) concentrations that were below detectable limits. These values are all well below the limit of 0.02 WL for average annual concentration as specified in the EPA Standard, 40 CFR 192 (see Appendix 6). No long-lived particulate contamination was found in any of these air samples. The results are presented in Table 2. The detailed calculations used in evaluating these samples are shown in Appendix 3.

Sludge, Water, Soil, and Tile Samples

Samples of the contaminated floor tiles in Room 205 (sample 10-T89) and Room 102 (samples 10-T90 and 10-T91) were collected for analysis. The results of the gamma spectral analyses indicated that the contamination was due primarily to ^{155}Eu in Room 205 and ^{137}Cs in Room 102. The results of the uranium fluorometric analyses for these samples indicated uranium concentrations of 6.6 μg/g (10-T89), 78.6 μg/g (10-T90), and 310 μg/g (10-T91). These results are given in Table 3.

These areas were subsequently decontaminated to background levels as confirmed by the ANL Radiological Survey Group in April 1982.

A water sample was taken from the cutting mockup pit in Room 180 (sample 10-W88) and water and sludge samples (10-W87 and 10-SS87) were taken from the sewer sump on the western side of the building. The water samples from the mockup pit and the sewer pump showed no significant contamination; however, the sludge sample from the sewer sump showed elevated levels of uranium. These results are included in Table 3. One sample (10-SS87) contained 314 μg of

uranium (30.7 μg U/g solids). Mass spectrometric analysis of this uranium indicated that the material was slightly enriched in the 235 isotope (1.03% ^{235}U versus 0.72% ^{235}U for normal uranium).

The presence of enriched uranium in the sewer sump rendered all sewer lines suspect. It was decided by Rockwell International to remove the sewer lines within Building 003. Prior to backfilling, the trench where the sewer lines were located was surveyed and nine soil samples (10-S116 through 10-S124) were collected for analysis. The location of the trench and the spots where the soil samples were taken are shown in Figure 4. These soil samples were analyzed for ^{137}Cs, ^{232}Th decay chain, ^{226}Ra decay chain, and ^{60}Co by gamma spectral analysis; for uranium by laser fluorometry; and for ^{90}Sr by radiochemical analysis. No significant levels of radioactive contaminants were found in the soil samples, and the direct survey measurements indicated background levels of radiation (see Table 3).

CONCLUSIONS

The final radiological survey conducted by the ANL Radiological Survey Group in April 1982, following further decontamination by Rockwell International, indicated that both the interior and the exterior of Building 003 are free from radioactive contamination. The criteria specified for the release of facilities for unrestricted use as promulgated by the U.S. Nuclear Regulatory Commission (see Appendix 6) have been met. Hence, this building is suitable for unrestricted use. The adjacent land surfaces immediately around the perimeter of the building are also certified to be free from contamination.

Air samples taken within the building certify that the radon (^{222}Rn) concentration is well below the limit (0.02 WL) prescribed by the Environmental Protection Agency (see Appendix 6).

The sewer lines exterior to the building should be considered potentially contaminated with radionuclides and, therefore, subject to restricted use. This restriction should include proper health physics control and suitable disposal of any material associated with these sewer lines. Effluent from the outfall of this sewer system should be periodically monitored for radionuclides.

REFERENCES

1. "Decontamination and Decommissioning of Facilities Program Plan No. PP-704-990-002." Rockwell International, Canoga Park, CA.

2. F. E. Begley, "Radiation Survey of Building T003-Santa Susana." Internal Letter, Rockwell International. March 15, 1982.

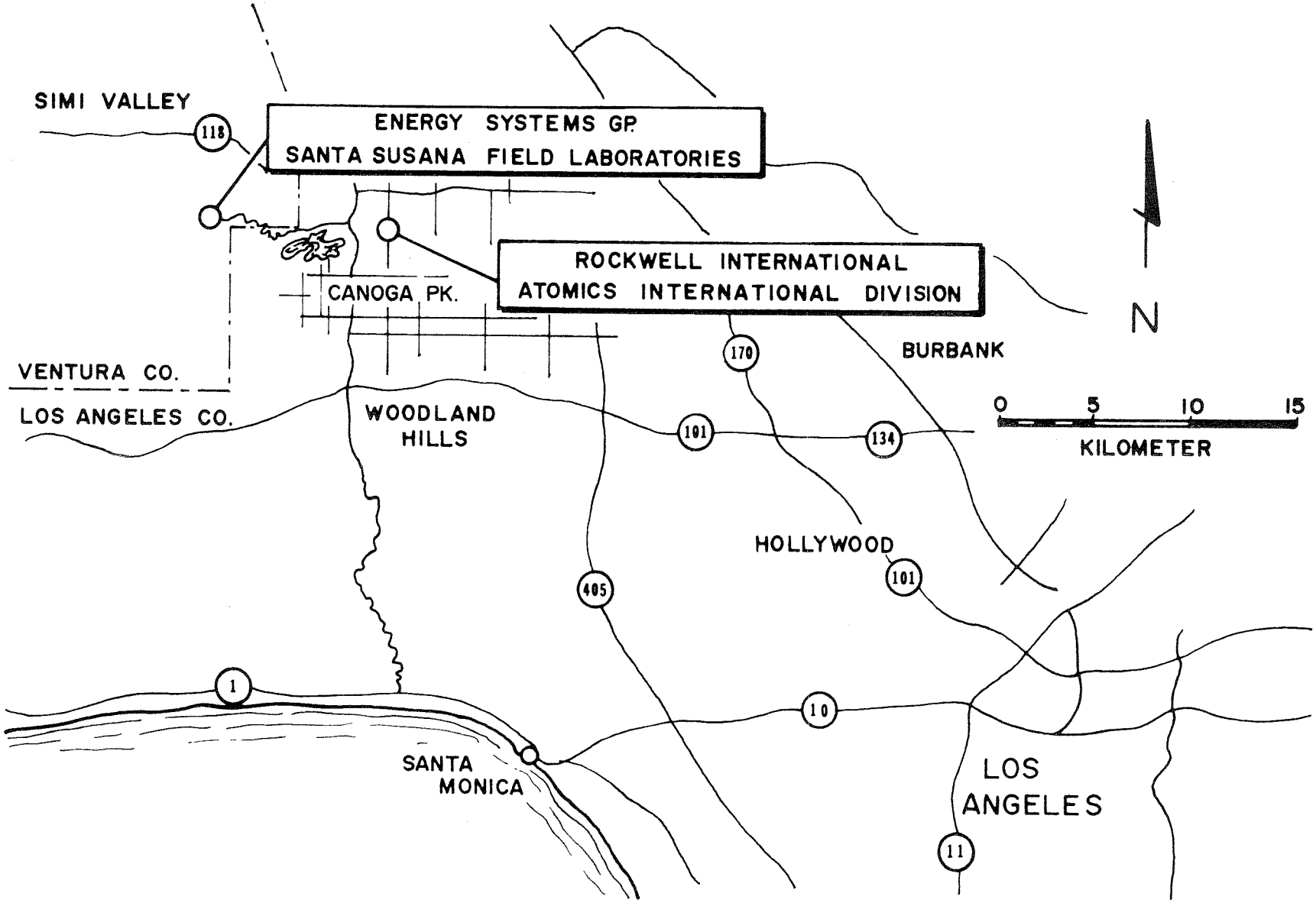

Fig. 1. Site Location of Santa Susana Field Laboratories

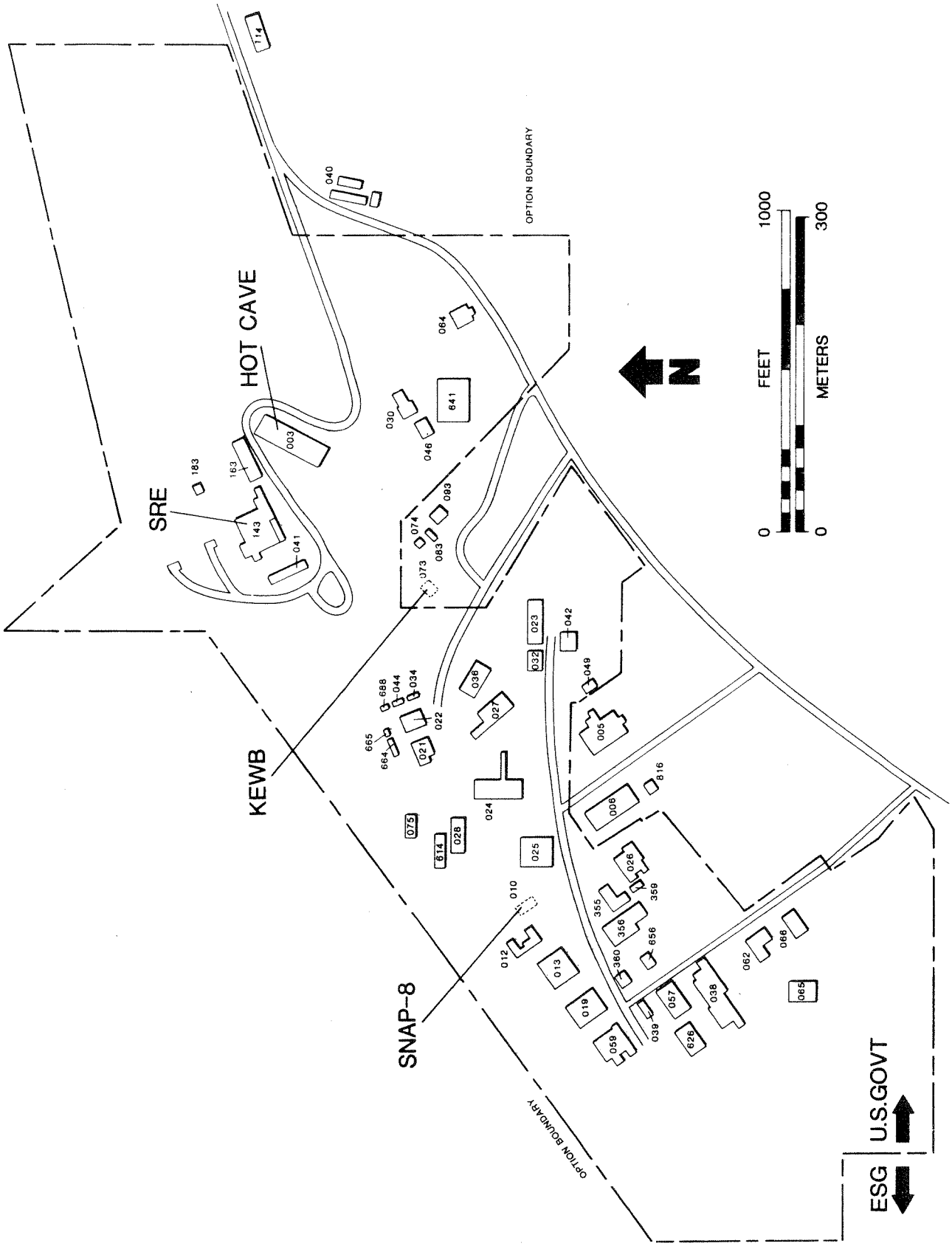

Fig. 2. Energy Systems Group, Santa Susana Field Laboratories

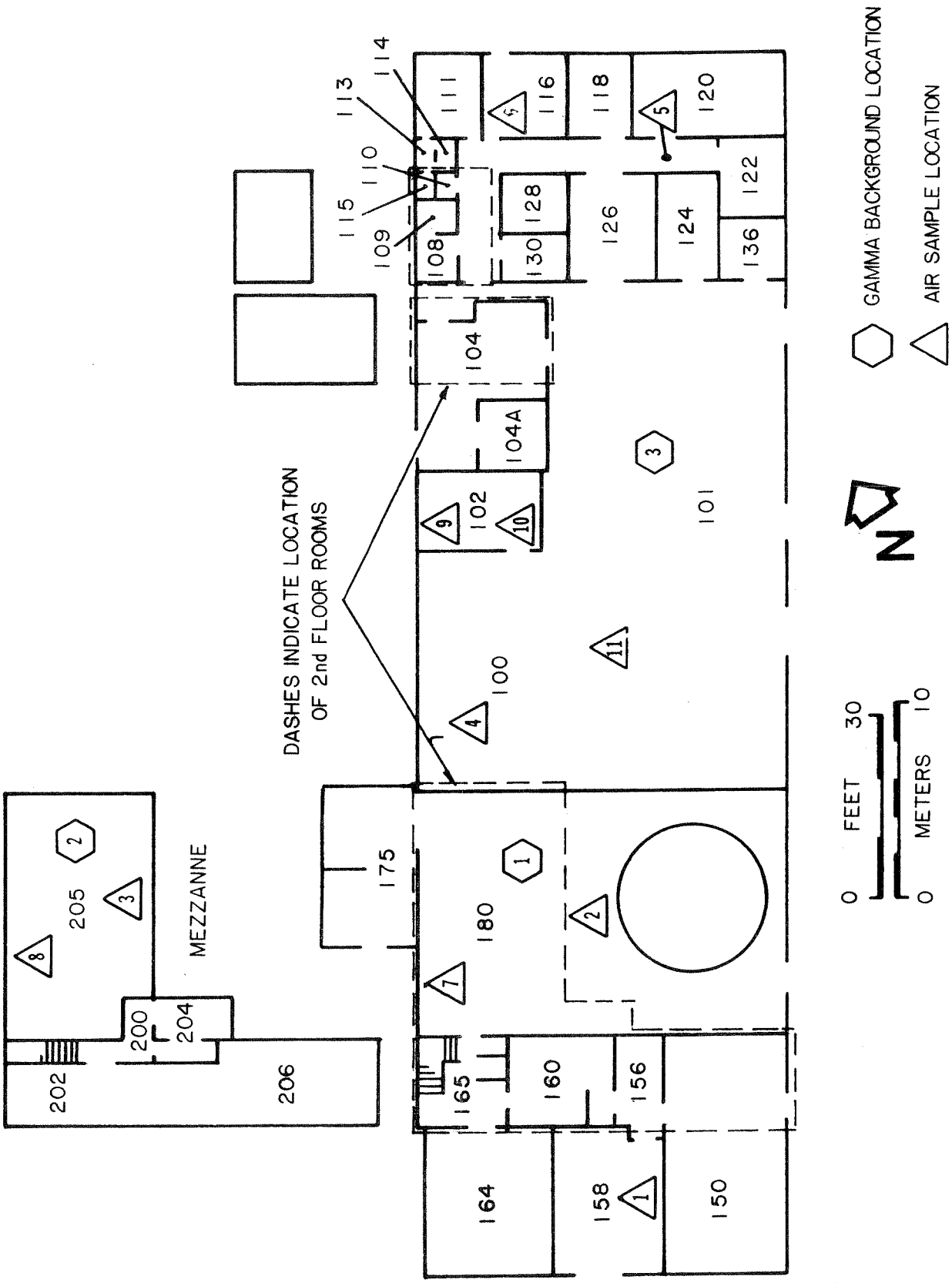

Fig. 3. Building 003

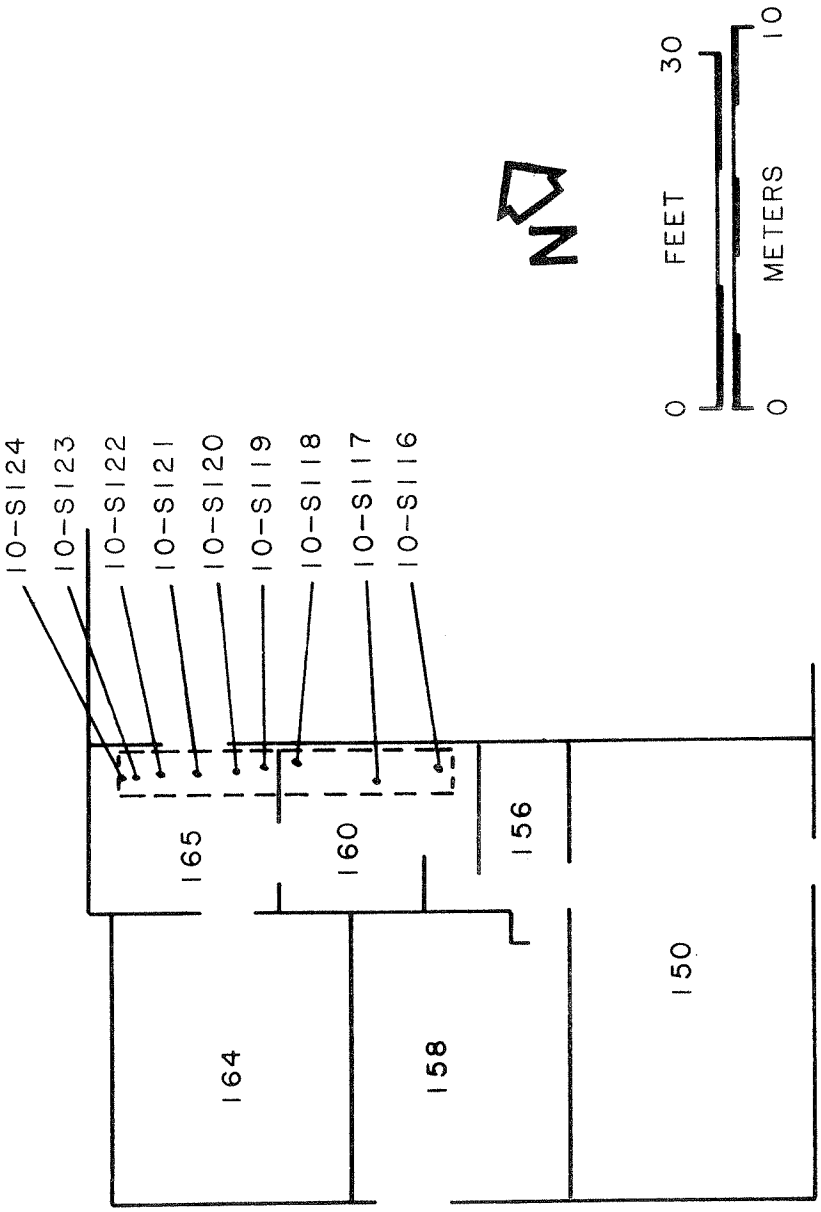

Fig. 4. Building 003 Trench Excavation

TABLE 1
DATA SHEET OF AREA SURVEYS

Room or Area No.	Percent of Area Accessible for Survey Floor	Wall	Air Sample (WL)	PAC 4G-3 Direct Readings[a] (dis/min-100 cm²) Beta	Alpha	End Window GM (mR/h) Contact	PRM-7 (μR/h) 1 meter	PRM-5-3 w/PG-2 (cts/min)	Smear Results (dis/min-100 cm²)	Comments
180	100	90		BKGD	BKGD		BKGD	5 k	NCD	
205 #1	100	90	0.0014	225 k	400	14	BKGD	45 k	NCD	RSS-111 - No. 2 11.2 μR/h
#2				100 k	300	-	BKGD	18 k	NCD	
#3				12 k	100	0.04	BKGD	3.5 k	NCD	
#4				8 k	450	0.1	BKGD	3 k	NCD	
#5				15 k	450	0.04	BKGD	BKGD	NCD	
Hot Speaker				10 k	BKGD	-	BKGD	125 k	NCD	
High Bay	100	90								RSS-111 - No. 3 12.3 μR/h
#1				BKGD	BKGD	BKGD	BKGD	6 k	NCD	
#2				BKGD	BKGD	BKGD	BKGD	17 k	NCD	
165, Bricks	NA	NA		2 k	BKGD	BKGD	BKGD	8 k	NCD	
101	100	90	0.0019	40 k	BKGD	BKGD	BKGD	4 k	NCD	RSS-111 - No. 1 11.6 μR/h
102	100	90	0.0025	15 k	10 k	0.08	11	4 k	NCD	

TABLE 1
DATA SHEET OF AREA SURVEYS

Room or Area No.	Percent of Area Accessible for Survey Floor	Wall	Air Sample (WL)	PAC 4G-3 Direct Readings[a] (dis/min-100 cm²) Beta	Alpha	End Window GM (mR/h) Contact	PRM-7 (μR/h) 1 meter	PRM-5-3 w/PG-2 (cts/min)	Smear Results (dis/min-100 cm²)	Comments
200	199	90		5 k	BKGD	BKGD	BKGD	20 k	NCD	
180, Overhead			0.0006	3 k	BKGD	BKGD	BKGD	BKGD	NCD	
Roof, South	100	NA		0.5 k	BKGD	BKGD	BKGD	BKGD	NCD	

[a] Instrument Backgrounds (included in tabulated data):

PRM-7	7 to 9 μR/h
PAC-4G-3	βγ 200 cts/min α < 50 cts/min
PRM-5-3	1.5 k cts/min (HV-1 gross)
End Window GM	0.03 mR/h
RSS-111	11.7 μR/h

TABLE 2

RADON-CONCENTRATION DETERMINATIONS

Sample Number	Location	WL[a]	Radon (^{222}Rn), pCi/ℓ	Thoron (^{220}Rn), pCi/ℓ
1	Room 158	0.0012	0.112	0.0062
2	Room 180	0.00064	0.064	0.0081
3	Room 205	0.0014	0.138	0.015
4	Room 101	0.0019	0.190	0.0079
5	Room 120, Corridor	0.0018	0.179	0.0041
6	Room 116	0.0015	0.148	0.0012
*				
10	102, Vault	0.0025	0.248	-
*				

[a]A Working Level (WL) is defined in 10 CFR 712 as any combination of short-lived radon-daughter products in 1 liter of air that will result in the ultimate emission of 1.3×10^5 MeV of potential alpha energy. The numerical value of the WL is derived from the alpha energy released by the total decay through RaC' of the short-lived radon-daughter products, RaA, RaB, and RaC at radioactive equilibrium with 100 pCi of ^{222}Rn per liter of air.

These WL values reported above are all below the limit of 0.02 WL for average annual concentration as specified in the EPA Standard (40 CFR 192).

*Air samples 7, 8, 9, and 11, were "actinon" samples and showed no measurable activity (see Appendix 3).

TABLE 3

GAMMA SPECTRAL, URANIUM FLUOROMETRIC AND RADIOCHEMICAL ANALYSES OF SAMPLES

	Gamma Spectra, pCi/g±σ			Uranium Fluorometric		Radiochemical
Sample No.	^{137}Cs	^{232}Th Decay Chain	^{226}Ra Decay Chain	µg/g ± 10%	pCi/g ± 10%	^{90}Sr pCi/g ± σ
10-T-89*	2.32± 0.23	<0.06	<0.02	6.6± 0.6	4.5± 0.5	
10-T-90-a	0.04± 0.02	<0.06	<0.02	16.8± 1.7	11.5± 1.2	
10-T-90-b	0.08± 0.03	<0.06	0.29±0.03	13.4± 1.3	9.2± 0.9	
10-T-90-c	<0.03	<0.06	0.15±0.02	78.6± 7.9	54.0± 5.4	
10-T-91	220.0 ±22.0	<0.06	<0.02	310.0±31.0	213.0±21.0	
10-W-87	<0.03	<0.06	<0.02	8.6± 0.9	5.9± 0.6	
10-W-88	0.07±0.04	<0.06	<0.02	20.0± 2.0	13.7± 1.4	
10-SS-87***	0.05±0.03	0.48±0.07	0.12±0.02	30.7± 3.0	21.1± 2.1	
10-S-116	<0.03	1.20±0.12	1.12±0.11	2.5± 0.2	1.7± 0.2	0.33±0.03
10-S-117	0.13±0.04	1.39±0.14	1.22±0.12	2.4± 0.2	1.6± 0.2	0.09±0.03
10-S-118	0.07±0.03	1.48±0.15	1.17±0.12	2.4± 0.2	1.6± 0.2	0.12±0.04
10-S-119	<0.03	1.53±0.15	1.21±0.12	2.1± 0.2	1.4± 0.2	0.06±0.03
10-S-120	BDL**	1.39±0.14	0.93±0.09	2.3± 0.2	1.6± 0.2	0.05±0.03
10-S-121	<0.03	1.88±0.19	1.15±0.12	2.1± 0.2	1.4± 0.2	0.14±0.04
10-S-122	<0.03	1.57±0.16	1.13±0.11	2.2± 0.2	1.5± 0.2	0.10±0.03
10-S-123	<0.03	1.87±0.19	0.84±0.08	2.2± 0.2	1.5± 0.2	0.08±0.02
10-S-124	<0.03	1.50±0.15	0.78±0.08	1.8±0.2	1.2±0.2	0.07±0.02

*Major activity has been identified as ^{155}Eu at a level of 7±4 pCi/g.
**Below detectable limits.
***Mass spectrometric analysis indicated enriched U (i.e., 1.03% ^{235}U, 98.96% ^{238}U).

APPENDIX 1

INSTRUMENTATION

I. PORTABLE RADIATION SURVEY METERS

A. Gas-Flow Proportional Survey Meters

The Eberline PAC-4G-3 was the primary instrument used for surveying. This instrument is a gas-flow proportional counter which utilizes a propane gas-proportional detector, 51 cm^2 (PAC-4G-3) or 325 cm^2 (FM-4G) in area, with a thin double-aluminized Mylar window (~ 0.85 mg/cm^2).

Since this instrument has multiple high-voltage positions, it can be used to distinguish between alpha and beta-gamma contamination. This instrument was initially used in the beta mode. In the beta mode, the detector responds to alpha and beta paticles and x- and gamma-rays. When areas indicated a higher count rate than the average instrument background, the beta-mode reading was recorded, and the instrument was then switched to the alpha mode to determine any alpha contribution. In the alpha mode, the instrument responds only to particles with high-specific ionization. This instrument is calibrated in the alpha mode with a flat-plate infinitely-thin NBS traceable ^{239}Pu standard, and in the beta mode with a flat-plate infinitely-thin NBS traceable ^{90}Sr-^{90}Y standard. The PAC-4G-3 instruments are calibrated to an apparent 50% detection efficiency.

B. Beta-Gamma End Window Survey Meter

When an area of contamination was found with a PAC instrument, a reading was taken with an Eberline Beta-gamma Geiger-Mueller Counter Model E-530 with a HP-190 probe. This probe has a thin mica end window and is, therefore, sensitive to alpha and beta particles and x- and gamma-rays. A thin piece of aluminum is added to the mica, thus making the window density ~7 mg/cm^2. At this density the instrument is not sensitive to alpha particles. A maximum reading is obtained with the probe placed in contact with the area of contamination. In this position, the response (in mR/h) to gamma radiation is generally conservative relative to a determination of mrad/h at 1 cm; however, the response (in mR/h) to beta radiation is nonconservative by a factor of up to abour four relative to a determination of mard/h through 7 mg/cm^2. Another reading is obtained with the probe held 1 m from the contaminated area. This instrument is calibrated in mR/h with a ^{226}Ra standard source.

C. Low Energy Gamma Scintillation Survey Meter

An Eberline Model PRM-5-3 with a PG-2 gamma scintillation detector was used to determine low energy x and gamma radiation. The PG-2 detector consists of a thin NaI(Tℓ) scintillation crystal 5 cm in diameter by 2 mm thick. This instrument is calibrated on three separate discriminators for three energy regions using ^{239}Pu (17 keV), ^{241}Am (59.5 keV) and ^{235}U (185.7 keV) sources. This instrument can be operated in either a differential (to discriminate between different energy regions) or integral mode.

APPENDIX 1
(cont'd.)

D. High Energy Micro "R" Scintillation Survey Meter

An Eberline Micro "R" meter model PRM-7 was used to detect high-energy gamma radiation. This instrument contains an internally mounted NaI(Tℓ) scintillation crystal 2.5 cm in diameter by 2.5 cm thick and can be used for measuring fields of low-level radiation between 10 μR/h and 5000 μR/h. This instrument is also calibrated with a ^{226}Ra standard source.

E. Integrating Radiation Meter

In addition to the PRM-7, a pressurized ion chamber (Reuter Stokes Model RSS-111) was used at selected locations to determine the ambient radiation field. The RSS-111 has three output modes; (1) instantaneous exposure rate, (2) strip chart differential readout, and (3) integrated exposure. This instrument is mounted on a tripod, 3 ft (~ 1 m) above the surface and has a uniform energy response from about 0.2 MeV to about 4 MeV. A 3-h period of operation is usually sufficient to obtain significant data.

II. SMEAR COUNTING INSTRUMENTATION

An ANL-designed gas-flow proportional detector connected to an Eberline Mini Scaler Model MS-2 was used to count multiple smears simultaneously. This detector has a double-aluminized Mylar window (400 cm^2) and uses P-10 (90% argon and 10% methane) as the counting gas. The metal sample holder for this detector has been machined to hold ten smear papers. This particular system consists of two Mini Scalers and two detectors. One is used for counting in the alpha mode; the other is used in the beta mode. Up to ten samples can be counted simultaneously.

Any smear taken from a contaminated area was counted individually in a Nuclear Measurements Corporation PC-5 gas-flow proportional counter. This instrument has been modified to contain a double-aluminized Mylar spun top. This top is placed over non-conducting media (e.g. paper smears) to negate the dielectric effect on the counter. This counter also uses P-10 counting gas. Smears are counted in both the alpha and beta modes. This instrument is calibrated by determining the input sensitivity using an alpha source.

III. AIR SAMPLING DEVICE

Air samples were collected using a commercially available (ANL-modified filter queen) vacuum cleaner identified as a "Princess Model." The air was drawn through a filter media at a flow rate of 40 m^3/h. The filter media consist of 200 cm^2 sheets of Hollingsworth-Vose (HV-70 or LB5211-9 mil) filter paper. The collection efficiency at these flow rates for 0.3-micron particles is about 99.9%.

A separate air sample can be taken with a positive displacement pump drawing about 20 liters/min through a millipore (0.5 to 0.8 micron) filter paper

APPENDIX 1
(cont'd.)

for about one hour. An alpha spectrum can be measured from a section of this filter paper. The ratio of actinon (^{219}Rn - 6.62 MeV α AcC) to radon (^{222}Rn - 7.69 MeV α RaC') can be determined from this spectrum.

IV. GAMMA SPECTRAL INSTRUMENTATION

A Nuclear Data Multichannel Analyzer Model ND-100, utilizing a 7.6-cm-diameter by 7.6-cm-thick NaI(Tℓ) scintillation crystal is commonly used for determining gamma spectrum. This instrument is calibrated with NBS traceable gamma sources. Samples from contaminated areas were analyzed using this sytem and the contamination radionuclides were identified.

Hyperpure Germanium detectors (ORTEC - 17% efficiency right-circular cylinders) were used when more sophisticated gamma-ray analyses were required. These detectors are coupled to Nuclear Data Multichannel Analyzers (Models ND-60, ND-66 or ND-100).

APPENDIX 2

CONVERSION FACTORS

I. INSTRUMENTATION

The factors used to convert the instrument readings to units of disintegrations per minute per 100 cm^2 (dis/min-100 cm^2) and the derivation of those factors are listed below.

A. Conversion Factors

	PAC-4G-3		Floor Monitor (FM-4G)	
	Alpha	Beta	Alpha	Beta
To 100 cm^2	1.96	1.96	0.31	0.31
cts/min to dis/min for ^{90}Sr-^{90}Y	-	2	-	2
cts/min to dis/min for ^{239}Pu	2	-	2	-
cts/min to dis/min for normal U	5.9	3.5	5.9	3.5
cts/min to dis/min ^{226}Ra plus daughters	1.6	4.7	1.6	4.7

B. Derivation of Conversion Factors

. Floor Monitor

Window Area: ~ 325 cm^2
Conversion to 100 cm^2 = 0.31 times Floor Monitor readings

. PAC-4G-3

Window Area: ~ 51 cm^2
Conversion to 100 cm^2 = 1.96 times PAC reading

. 2π Internal Gas-Flow Counter, PC counter

Geometry: Solid Steel Spun Top - 0.50

Geometry: Mylar Spun Top - 0.43
Mylar spun top counting {double-aluminized Mylar window (~ 0.85 mg/cm^2)} utilizes the well of the PC counter and is a method developed and used by the Argonne National Laboratory Health Physics Section for negating the dielectric effect in counting samples on nonconducting media.

APPENDIX 2
(cont'd.)

Using a flat-plate, infinitely thin ^{226}Ra plus short-lived daughters standard as a source of alpha emissions, the plate was counted in the well of a 2π Internal Gas-Flow Counter (PC counter) with the source leveled to an apparent 2π geometry. This instrument was calibrated using ^{239}Pu NBS traceable alpha sources. The alpha counts per minute (cts/min) reading was found to be 1.8×10^4 cts/min, or $1.8 \times 10^4 \div 0.51^* = 3.5 \times 10^4$ disintegrations per minute (dis/min) alpha. Since the source was infinitely-thin, the alpha component was used as the total alpha dis/min of the source.

The same ^{226}Ra plus daughters source, when counted with the PAC instrument in the alpha mode, was found to be 2.2×10^4 cts/min at contact. The conversion factor for cts/min to dis/min for the PAC instrument is $3.5 \times 10^4 \div 2.2 \times 10^4 = 1.6$ dis/min alpha to cts/min alpha.

The same source was covered with two layers of conducting paper, each 6.65 mg/cm^2, to absorb the alpha emissions. With the PAC-4G-3 in the beta mode and in contact with the covered source in the center of the probe, the count was found to be 7.5×10^3 cts/min. This indicates a conversion factor of $3.5 \times 10^4 \div 7.5 \times 10^3 = 4.7$ dis/min alpha to cts/min beta-gamma.

A similar method was used to determine the conversion factors for normal uranium.

II. SMEAR COUNT

The conversion factors for cts/min-100 cm^2 to dis/min-100 cm^2 for smear counts are given below:

A. Conversion Equation (Alpha)

$$\frac{\text{cts/min - (Bkgd)}}{\text{g x bf x sa x waf}} = \text{dis/min } \alpha$$

A geometry (g) of 0.43 is standard for all flat-plate counting using the Mylar spun top.

A backscatter factor (bf) of 1.0 was used when determining alpha activity on a filter media.

The self-absorption factor (sa) was assumed to be 1, unless otherwise determined.

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*The value of 0.51 includes the following factors: geometry (g) = 0.50; backscatter factor (bf) = 1.02; sample absorption factor (sa) = 1.0; window air factor (waf) = 1.0. The product of g x bf x sa x waf is 0.51.

APPENDIX 2
(Cont'd.)

If the energies of the isotope were known, the appropriate window air factor (waf) was used; if the energies of the isotopes were not known, the (waf) of ^{239}Pu (0.713) was used.

The (waf) for alpha from ^{226}Ra plus daughters is 0.55.

B. Conversion Equation (Beta)

$$\frac{\text{cts/min} - \{\beta \text{ Bkgd (cts/min)} + \alpha \text{ cts/min}\}}{g \times bf \times sa \times waf} = \text{dis/min } \beta$$

A geometry (g) of 0.43 is standard for all flat-plate counting using the Mylar spun top.

A backscatter factor (bf) of 1.1 was used when determining beta activity on a filter media.

A self-absorption factor (sa) was assumed to be 1, unless otherwise determined.

If the energies of the isotopes were known, the appropriate window air factor (waf) was used; if the energies of the isotopes were unknown, the (waf) of ^{90}Sr-^{90}Y (0.85) was used.

The (waf) for betas from ^{226}Ra plus daughters is 0.85.

APPENDIX 3

RADON-DETERMINATION CALCULATIONS

Calculations for air samples collected with an Argonne National Laboratory-designed air sampler using HV-70 or LB5211 filter media are summarized in this appendix. The appendix also includes the basic assumptions and calculations used to derive the air concentrations.

I. RADON CONCENTRATIONS

The following postulates are assumed in deriving the radon (^{222}Rn) concentrations based on the RaC' alpha count results.

A. RaA, RaB, RaC, and RaC' are in equilibrium.

B. RaA is present only in the first count and not the 100-minute decay count.

C. One-half of the radon progeny is not adhered to airborne particulates (i.e., unattached fraction) and, therefore, is not collected on the filter media.

D. The geometry factor (g) is 0.43 for both the alpha and beta activity.

E. The backscatter factor (bf) of 1.0 is used for the alpha activity.

F. The sample absorption factor (sa) for RaC' is 0.77.

G. The window air factor (waf) for RaC' is 0.8.

H. RaB and RaC, being beta emitters, are not counted in the alpha mode.

I. The half-life of the radon progeny is approximately 36 minutes, based on the combined RaB and RaC half-lives.

J. Thoron and long-lived alpha emitters are accounted for using the 360 count and the seven-day count, respectively.

K. For all practical purposes, RaC' decays at the rate of the composite of RaB and RaC, which is about 36 minutes.

The following postulates are assumed in deriving the thoron (^{220}Rn) concentrations.

L. ThA, ThB, ThC and ThC' are in equilibrium.

M. ThA and RaC' have decayed by the 360-minute decay count.

APPENDIX 3
(cont'd.)

N. The geometry factor (g), backscatter factor (bf), sample absorption factor (sa) and window air factor (waf) are all the same for thoron as for radon.

O. ThB and 64% of ThC, being beta emitters, are not counted in the alpha mode.

P. The half-life of the thoron progeny is 10.64 hours (638.4 minute) based on the ThB half-life.

Q. For all practical purposes 36% of the ThC (alpha branch) and the ThC' decay at the decay rate of ThB which is 638.4 minute.

R. The counter does not differentiate between the ThC alphas and the ThC' alphas.

The following postulates are assumed in deriving the actinon (^{219}Rn) concentrations.

S. AcA, AcB and AcC are in equilibrium.

T. AcA has decayed by the 100 minute decay count.

U. The geometry (g), backscatter (bf), sample absorption (sa) and window air factor (waf) factors are all the same for actinon as for radon.

V. AcB, being a beta emitter, is not counted in the alpha mode.

W. The half-life of the actinon progeny is 36.1 minutes based on the AcB half-life.

X. For all practical purposes, the AcC decays at the decay rate of AcB which is 36.1 minutes.

Y. 84% of the AcC decays by 6.62 MeV α emissions and 16% decays by 6.28 MeV α emissions.

The following postulate is assumed in deriving the long-lived concentration. The long-lived activity, as determined from the seven-day count, is assumed to be constant during the entire counting periods. This assumption is valid for isotopes with half-lives longer than a few years.

II. EQUATIONS USED TO DERIVE AIR CONCENTRATIONS

$$A_o = \frac{A}{e^{-\lambda t}}$$

Where: A_o = Activity (dis/min) present at the end of the sampling period (usually 40 minutes)

APPENDIX 3
(cont'd.)

A = Activity (dis/min) at some time, t, after end of the sampling period

t = Time interval (minute) from end of sampling period to counting interval (usually $\sim$ 100 minutes)

$$\lambda = \frac{0.693}{t_{\frac{1}{2}}}$$

$t_{\frac{1}{2}}$ = Half-life of isotope (min)

Concentration is determined by the equation:

$$C = \frac{A_o \lambda}{f} \times \frac{1}{1-e^{-\lambda t_s}}$$

Where: C = Concentration (dis/min-m^3)

A_o = Activity on filter media at end of sampling period (dis/min)

f = Sampling rate (m^3/min = m^3/h x 1 h/60 minutes)

t_s = Length of sampling time (minute)

$$\lambda = \frac{0.693}{t_{\frac{1}{2}}}$$

$t_{\frac{1}{2}}$ = Half-life of isotope or controlling parent (minute)

III. ACTINON CORRECTION

Since the actinon (^{219}Rn) progeny (AcA, AcB & AcC) decays at the AcB half-life of 36 minutes it cannot be distinguished from the radon (^{222}Rn) progeny using standard air sampling with HV-70 or LB5211 filter media and standard alpha counting techniques. A positive displacement pump is used to collect a sample on millipore (0.5 to 0.8 micron) filter media. The sample rate is approximately 20 liters/minute for a sampling time of at least 90 minutes. The center portion of the sample is removed and counted in an alpha spectrometer which exhibits the 6.62 MeV AcC alpha emissions and the 7.69 MeV RaC' alpha emissions. If these two peaks are observed in the spectrum, then the following calculations are performed:

APPENDIX 3
(cont'd.)

$$B_j = \sum_{i=1}^{n} b_{ij}$$

Where: B_j = summation of n channels under peak j.

b_{ij} = the number of counts in channel i of peak j

j = 1 for the 6.62 MeV peak of actinon; 2 for the 7.69 MeV peak of radon.

n = total number of channels in the summation.

The fraction of the activity with a 36-minute half-life due to actinon and radon are then:

$$\text{Actinon} = \frac{B_1/0.84}{B_1/0.84+B_2}$$

$$\text{Radon} = \frac{B_2}{B_1/0.84+B_2}$$

where 1 refers to actinon progeny and 2 refers to radon progeny.

IV. EXAMPLE CALCULATION

Data have been created to correspond to values likely to occur if all possible types of contamination are present in the air of a room where a sample is collected. The application of the equations for determining all types of activity and their concentrations are given below.

Data	$f = 40\ m^3/60\ min$	$t_s = 40\ min$
at	$t = 100\ min$	$A = 2000\ dis/min$
at	$t = 360\ min$	$A = 140\ dis/min$
at	$t = 7\ days$	$A = 5\ dis/min$

For long-lived activity:

$$A_o = A = 5\ dis/min$$

$$C(L) = A_o/fxt_s = \frac{5}{40/60x40} = 0.19\ dis/min\text{-}m^3$$

APPENDIX 3
(cont'd.)

For thoron:

$$A_o = \frac{140-5}{\exp - \frac{0.693 \times 360}{638.4}} = 199.6 \text{ dis/min}$$

$$C(Tn) = \frac{199.6 \times \frac{0.693}{638.4}}{40/60} \times \frac{1}{1 - \exp - \frac{0.693 \times 40}{638.4}} = 7.6 \text{ dis/min-m}^3$$

For radon (^{222}Rn) and actinon (^{219}Rn), activity due to thoron at t = 100 min:

$$A = \frac{135}{\exp - \frac{0.693 \times 260}{638.4}} = 179 \text{ dis/min}$$

Activity due to the isotopes with a 36 minute half-life:

$$A = 2000 - 179 - 5 = 1816 \text{ dis/min}$$

$$A_o = \frac{1816}{\exp - \frac{0.693 \times 100}{36}} = 12,454 \text{ dis/min}$$

$$C(36) = \frac{12,454 \times \frac{0.693}{36}}{40/60} \times \frac{1}{1 - \exp \frac{-0.693 \times 40}{36}} = 669.7 \text{ dis/min-m}^3$$

When an actinon peak is seen at 6.62 MeV, then the counts under the two peaks are summed. For example, if 10 channels are summed, the following counts are found:

For 6.62 MeV peak: 44 in 10 channels, where the 6.62 alpha emissions are 84% of the total.

For 7.69 MeV peak: 601 counts in 10 channels, where the 7.69 MeV alpha emissions are 100% of the total.

APPENDIX 3
(cont'd.)

$B_1 = 44$

$B_1/0.84 = 52$ counts

$B_2 = 601$ counts

Actinon = 52/653 = 0.08

Radon = 601/653 = 0.92

C(Rn) = C(36) x Radon% = 669.7 x 0.92 = 616.1 dis/min-m^3

C = C(36) x Actinon% = 669.7 x 0.08 = 53.6 dis/min-m^3

Since we assume that on the average half of the progeny is not adhered to the airborne particulates, the above concentrations are then multiplied by 2 to determine actual concentrations. We assume that there is no unattached fraction for the long-lived activity.

C actual = C measured x progeny correction factor

C(L) = 0.19 dis/min-m^3

C(Tn) = 7.6 dis/min-m^3 x 2 = 15.2 dis/min-m^3

C(An) = 53.6 dis/min-m^3 x 2 = 107.2 dis/min-m^3

C(Rn) = 616 dis/min-m^3 x 2 = 1232 dis/min-m^3

These would then be the resulting concentrations in dis/min-m^3. To convert to pCi/ℓ, divide the concentrations by 2.2 x 10^3.

$$C(L) = \frac{0.19 \text{ dis/min-m}^3}{2{,}220 \text{ dis/min-m}^3/\text{pCi}/\ell} = 8.6 \times 10^{-5} \text{ pCi}/\ell$$

$$C(Tn) = \frac{15.2 \text{ dis/min-m}^3}{2{,}220 \text{ dis/min-m}^3/\text{pCi}/\ell} = 0.0068 \text{ pCi}/\ell$$

$$C(An) = \frac{107.2 \text{ dis/min-m}^3}{2{,}220 \text{ dis/min-m}^3/\text{pCi}/\ell} = 0.048 \text{ pCi}/\ell$$

$$C(Rn) = \frac{1232 \text{ dis/min-m}^3}{2{,}220 \text{ dis/min-m}^3/\text{pCi}/\ell} = 0.55 \text{ pCi}/\ell$$

APPENDIX 4

SAMPLE PREPARATION AND ANALYSIS GENERIC PROTOCOL

I. SOIL-SAMPLE PREPARATION

Soil samples are acquired as previously described. These samples are bagged and identified at the collection site and returned to ANL. If there is an indication of radioactive contamination, the sample is sealed in a Nalgene jar. At ANL, the soil samples are logged into the soil sample book and weighed. Each soil sample is weighed (on a tared balance scale) and the weight is marked on the container. This weight is recorded in the soil book as a "net weight."

After all samples are marked, weighed, and recorded, they are dried. Each sample is placed in a pyrex beaker marked with the sample identification number. If more than one beaker is necessary, additional numbers (e.g., 1-3, 2-3, 3-3) are used. The original containers are saved for repackaging the dried samples. The beaker is set in an 80°C oven until the soil is dry (approximatley 48 hours). Visual inspection of the soil is sufficient to determine when the soil is dry. The sample is returned to the original container and reweighed using a tared balance scale. This weight is also marked on the container and in the soil sample book where it is referred to as a "dry weight."

After all the samples are returned to their original containers, the milling process is started. Each dried sample is transferred to a 2.3-gallon ceramic mill jar containing mill balls (1¼" x 1¼" Burundum cylinders). The mill jar number is marked on the original container. The jars are sealed and the samples are milled for two hours or until sufficient material is produced to obtain 100 g and 5 g samples for analyses. The samples are milled six at a time. A second set of six jars is prepared while the milling of the first set is proceeding. After each sample is milled, the mill balls are removed with tongs and placed in a tray. A large plastic bag is inverted over the mill jar. Both are inverted and shaken until all the soil is transferred to the bag. If the soil plates the inside of the mill jar, a small paint brush is used to loosen the soil before the jar is inverted. A separate brush is used for each jar to prevent cross-contamination of the soil samples.

After milling, each sample is sieved through a number 30 standard testing sieve (600 micron) and transferred to a 12" x 12" ziplock bag. Rocks and dross are bagged separately. The bags are marked with the sample number, the sieve number and R(rocks) or S(soil). The balance is tared and the weights of the soil (or rocks) are measured and recorded in the soil sample book. A 100-g sample of the sieved material is transferred to a 4-oz. Nalgene bottle. These samples are analyzed by suitable analytical techniques, including, as a minimum, gamma spectroscopy (GeLi) and radiochemical analyses for uranium. A 5-g sample of the sieved material is transferred to a 1-oz Nalgene bottle. This sample is used for the determination of uranium by laser fluorometry. The bottles containing these weighed samples are marked with sample number and date and this information is recorded in the soil sample book. The rocks (and dross) and remaining soil are placed in storage.

APPENDIX 4
(cont'd.)

The sieves, mill jars, and Burundum milling balls used in this work are classified in two sets. One set is used for background samples exclusively. The other set is used for all samples from suspect areas. Soil samples with elevated levels of radioactivity based on instrument measurements are milled in one-gallon Nalgene bottles using Burundum balls from the set used for suspect samples. After use, these balls are either decontaminated (see below) or disposed of as radioactive waste. The Nalgene bottles are always disposed of as radioactive waste. The sieves used for these samples are also from the set used for suspect samples and are decontaminated after use.

II. EQUIPMENT DECONTAMINATION

The care of the milling apparatus is as important as the actual sample preparation. Proper care prevents cross-contamination of successive samples. The beakers used to dry the samples are washed thoroughly by placing a small amount of Haemo-Sol in each beaker and filling with warm water. The beaker is then scrubbed thoroughly on the inside and scoured on the outside with scouring powder. The beakers are then rinsed with tap water (three times) followed by demineralized water (three times) and finally dried thoroughly before reuse.

The milling apparatus (tongs, brushes, milling jars, lids and milling balls) are rinsed. The tongs and brushes are washed thoroughly with Haemo-Sol. Eight Burundum balls are returned to each milling jar along with about one pint of clean road gravel, one spoon of Haemo-Sol, one spoon of scouring powder with bleach, and one quart of water. The lid is tightened on the jar and the jar is placed on the rolling mill and rolled for approximately two hours or until the balls and the inside of the jar appear to be physically clean. After this time, the mill jar is removed from the rolling mill and its contents are dumped into a screen or basket. The lid and balls are then rinsed thoroughly three times with tap water followed by three times with demineralized water. The inside of the jar is rinsed until it is absolutely clean. The milling apparatus is air dried using warm air until absolutely dry. Air is blown through a hose from the oven to the inside of the ceramic jar to dry the jar.

The sieves are rinsed, washed in Haemo-Sol, thoroughly rinsed (three times with tap water, followed by three rinses with demineralized water) and then air dried as above before reuse.

III. WATER AND SLUDGE

Water samples are collected in 0.1-liter, 0.5-liter and/or 1-liter quantities as deemed appropriate. These samples are forwarded directly to a certified radiochemistry laboratory for preparation and analysis. The customary analysis procedure consists of filtration to obtain the suspended solids followed by evaporation to obtain the dissolved solids. Both suspended and dissolved solids are analyzed by appropriate radiochemical analytical techniques.

APPENDIX 4
(cont'd.)

Sludge samples are collected in 0.1-liter bottles and are processed as outlined above for water samples.

IV. VEGETATION, TRASH AND RUBBLE

Samples of potentially contaminated vegetation, trash (e.g. piping, ducts, conduit, etc.) and rubble are collected, bagged, and labeled at the site and returned to ANL for analysis.

Vegetation samples are initially weighed and transferred to Marinelli beakers for gamma spectrometric analysis. Then they are ashed, reweighed, and analyzed by appropriate analytical techniques.

Trash and rubble samples are forwarded to a certified radiochemistry laboratory for analysis.

V. TRITIUM FROM SOLID MATERIALS

Samples of solid materials (e.g., concrete) suspected of containing tritium are collected, broken into small pieces and submitted to a certified radiochemistry laboratory for analysis. The standard analytical procedure consists of transferring a 20-40 g sample to a ceramic boat followed by heating in a tube furnace at 425°C for a period of two hours (~ 40 min to reach temperature and ~ 80 min heating at temperature). Helium is used as a flow gas through the tube during heating, and the tritium is collected in two traps on the downstream side of the furance. The first trap is immersed in an ordinary ice bath (0°C); the second trap is immersed in a CO_2-Freon bath (-57°C). The collected tritiated water from both traps is combined, made up to a known volume, and an aliquot taken for liquid scintillation counting of the tritium.

VI. ANALYSIS PROCEDURES

A 100-g fraction from each soil sample is analyzed by high resolution gamma-ray spectroscopy using a germanium crystal detector coupled to a multichannel analyzer. This analysis allows for a quantitative determination of the ^{226}Ra decay chain (via the 609 keV γ-ray of ^{214}Bi) and the ^{232}Th decay chain (via the 908 keV γ-ray of ^{228}Ac as well as any other gamma emitting radionuclide (e.g. ^{137}Cs) present in the soil.

The total uranium (elemental) present in the soil is determined by an acid leach of the soil sample followed by laser fluorometry of the leached sample.

Thorium analysis consists of an acid leach of the soil (using a ^{234}Th spike for yield determination) followed by plating a thin source of the radiochemically separated thorium and determining the thorium isotopes (^{228}Th and ^{232}Th) by alpha spectroscopy.

APPENDIX 4
(cont'd.)

The results of the above measurements allow for quantitative determination of the relative amounts of normal uranium, natural uranium, tailings (i.e., ^{226}Ra decay chain), thorium (^{232}Th), mesothorium (^{228}Ra decay chain) and thorium (^{228}Th) decay chain present in the contaminated material.

A mass spectrometric analysis of the uranium fraction is conducted when it is known or is is surmised that depleted or enriched uranium might be present.

APPENDIX 5

CALCULATION OF NORMAL-URANIUM SPECIFIC ACTIVITY

The specific activity for normal uranium was obtained by summing the measured specific activities for the individual isotopes weighted according to their normal abundances. Best values for these specific activities were taken from A. H. Jaffey, et al. Phys. Rev. 4 1889 (1971). The percent abundance and half-life for each isotope were taken from the "Table of Isotopes," 7th Edition by C. M. Lederer and V. S. Shirley (1978). Atomic weights were taken from the Handbook of Chemistry and Physics, 52nd Edition (1971).

Isotope	Specific Activity	Half-life (years)	Abundance (atom %)	Atomic Weight (grams)	Abundance (wt %)
^{234}U	-	2.446×10^5	0.0054	234.0409	0.0053
^{235}U	4.798 dis/min-μg	7.038×10^8	0.7196	235.0439	0.7106
^{238}U	0.746 dis/min-μg	4.4683×10^9	99.2747	238.0508	99.2841
			99.9997		100.0000

where $(wt\ \%)_i =$

$$\frac{(atom\ \%)_i \quad (atomic\ weight)_i}{\sum_j (atom\ \%)_j \quad (atomic\ weight)_j} = \frac{(atom\ \%)_i \quad (atomic\ weight)_i}{238.02985}$$

Specific activity for normal uranium:

$$0.746 \times 0.99284 \times 2 = 1.481 \quad \text{dis/min-μg from } ^{234} \text{ \& } ^{238}U$$
$$4.798 \times 0.00711 = \underline{0.0341} \quad \text{dis/min-μg from } ^{235}U$$
$$1.515 \quad \text{dis/min-μg for normal U}$$

or $(1.515\ \text{dis/min-μg})/(2.22\ \text{dis/min-pCi}) = 0.683\ \text{pCi/μg}$

where ^{234}U is assumed to be in secular equilibrium with the ^{238}U parent.

Note that 2.25% of the total activity is due to ^{235}U and 48.87% each is due to ^{234}U and ^{238}U.

APPENDIX 6

PERTINENT RADIOLOGICAL REGULATIONS, STANDARDS, AND GUIDELINES

Excerpts From

I. DRAFT AMERICAN NATIONAL STANDARD

N13.12

Control of Radioactive Surface Contamination on Materials, Equipment, and Facilities to be Released for Uncontrolled Use

Where potentially contaminated surfaces are not accessible for measurement (as in some pipes, drains, and ductwork), such property shall not be released pursuant to this standard, but shall be made the subject of case-by-case evaluation.

Property shall not be released for uncontrolled use unless measurements show the total and removable contamination levels to be no greater than the values in Table 1 or Table 2. (The values in Table 2 are easier to apply when the contaminants cannot be individually identified.)

Coatings used to cover the contamination shall not be considered a solution to the contamination problem. That is, the monitoring techniques shall be sufficient to determine, and such determination shall be made, that the total amount of contamination present on and under any coating does not exceed the Table 1 or Table 2 values before release.

APPENDIX 6
(Cont'd.)

TABLE 1

SURFACE CONTAMINATION LIMITS*

Contaminants			Limit (Activity) (dis/min-100 cm^2)+	
Group	Description	Nuclides (Note 1)	Removable	Total (Fixed plus Removable)
1	Nuclides for which the nonoccupational MPC$_a$ (Note 2) is 2×10^{-13} Ci/m^3 or less or for which the nonoccupational MPC$_w$ (Note 4) is 2×10^{-7} Ci/m^3 or less	^{227}Ac $^{241,242m,243}Am$ $^{249,250,251,252}Cf$ $^{243,244,245,246,247,248}Cm$ $^{125,129}I$ ^{237}Np ^{231}Pa ^{210}Pb $^{238,239,240,242,244}Pu$ $^{226,228}Ra$ $^{228,230}Th$	20	Nondetectable (Note 3)
2	Those nuclides not in Group 1 for which the nonoccupational MPC$_a$ (Note 2) is 1×10^{-12} Ci/m^3 or less for which the nonoccupational MPC$_w$ (Note 4) is 1×10^{-6} Ci/m^3 or less	^{254}Es ^{256}Fm $^{126,131,133}I$ ^{210}Po ^{223}Ra ^{90}Sr ^{232}Th ^{232}U	200	2000 α Nondetectable β,γ (Note 5)
3	Those nuclides not in Group 1 or Group 2		1000	5000

APPENDIX 6
(Cont'd.)

SURFACE CONTAMINATION LIMITS

*The levels may be averaged over one square meter provided the maximum activity in any area of 100 cm^2 is less than three times the limit value. For purposes of averaging with regard to isolated spots of activity, any square meter of surface shall be considered to be contaminated above the limit L, applicable to 100 cm^2, if (1) from measurements of a representative number n of sections it is determined that $1/n \sum_{i}^{n} S_i \geq L$, where S_i is the dis/min-100 cm^2 determined from measurement of section i; or (2) it is determined that the activity of all isolated spots or particles in any area less than 100 cm^2 exceeds 3 L.

+Disintegrations per minute per square decimeter.

NOTES:

(1) Values presented here are obtained from the Code of Federal Regulations, Title 10, Part 20, April 30, 1975. The most limiting of all given MPC values (for example, soluble versus insoluble) are to be used. In the event of the occurrence of mixtures of radionuclides, the fraction contributed by each constituent of its own limit shall be determined and the sum of the fraction shall be less than 1.

(2) Maximum permissible concentration in air applicable to continuous exposure of members of the public as published by or derived from an authoritative source such as the National Committee on Radiation Protection and Measurements (NCRP), the International Commission on Radiological Protection (ICRP), or the Nuclear Regulatory Commission (NRC). From the Code of Federal Regulations, Title 10, Part 20, Appendix B, Table 2, Column 1.

(3) The instrument utilized for this measurement shall be calibrated to measure at least 100 pCi of any Group 1 contaminants uniformly spread over 100 cm^2.

(4) Maximum permissible concentration in water applicable to members of the public.

(5) The instrument utilized for this measurement shall be calibrated to measure at least 1 nCi of any Group 2 beta or gamma contaminants uniformly spread over an area equivalent to the sensitive area of the detector. Direct survey for unconditional release should be performed in areas where the background is $\leq$ 100 counts per minute. When the survey must be performed in a background exceeding 100 counts per minute, it may be necessary to use the indirect survey method to provide the additional sensitivity required.

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(Cont'd.)

TABLE 2
ALTERNATE SURFACE CONTAMINATION LIMITS

(All Alpha Emitters, except U_{nat} and Th_{nat}, Considered as a Group)*

Contamination Contingencies	Limit (Activity) (dis/min-100 cm^2)+ Removable	Total (Fixed Plus Removable
If the contaminant cannot be identified; or if alpha emitters other than U_{nat} (Note 1) and Th_{nat} are present; or if the beta emitters comprise ^{227}Ac or ^{228}Ra.	20	Nondetectable (Note 2)
If it is known that all alpha emitters are generated from U_{nat} (Note 1) and Th_{nat}; and if beta emitters are present that, while not identified, do not include ^{227}Ac, ^{125}I, ^{226}Ra, and ^{228}Ra.	200	2000 α Nondetectable β,γ (Note 3)
If it is known that alpha emitters are generated only from U_{nat} (Note 1) and Th_{nat} in equilibrium with its decay products; and if the beta emitters, while not identified, do not include ^{227}Ac, ^{125}I, ^{129}I, ^{90}Sr, ^{223}Ra, ^{228}Ra, ^{126}I, ^{131}I and ^{133}I.	1000	5000

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ALTERNATE SURFACE CONTAMINATION LIMITS

* The levels may be averaged over one square meter provided the maximum activity in any area of 100 cm^2 is less than three times the limit value. For purposes of averaging with regard to isolated spots of activity, any square meter of surface shall be considered to be contaminated above the limit L, applicable to 100 cm^2, if (1) from measurements of a representative number n of sections it is determined that $1/n \sum_{i}^{n} S_i \geq L$, where S_i is the dis/min-100 cm^2 determined from measurement of section i; or (2) it is determined that the activity of all isolated spots or particles in any area less than 100 cm^2 exceeds 3 L.

+ Disintegrations per minute per square decimeter.

NOTES:

(1) U_{nat} and decay products.

(2) The instrument utilized for this measurement shall be calibrated to measure at least 100 pCi of any Group 1 contaminants uniformly spread over 100 cm^2.

(3) The instrument utilized for this measurement shall be calibrated to measure at least 1 nCi of any Group 2 beta or gamma contaminants uniformly spread over an area equivalent to the sensitive area of the detector. Direct survey of unconditional release should be performed in areas where the background is $\leq$ 100 counts per minute. When the survey must be performed in a background exceeding 100 counts per minute, it may be necessary to use the indirect survey method to provide the additional sensitivity required.

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II. U.S. NUCLEAR REGULATORY COMMISSION, DIVISION OF FUEL CYCLE AND MATERIAL SAFETY, WASHINGTON, D.C., July 1982

GUIDELINES FOR DECONTAMINATION OF FACILITIES AND EQUIPMENT PRIOR TO RELEASE FOR UNRESTRICTED USE OR TERMINATION OF LICENSES FOR BY-PRODUCT SOURCE, OR SPECIAL NUCLEAR MATERIAL

(These have been retyped for purposes of this report)

The instructions in this guide, in conjunction with Table 1, specify the radioactivity and radiation exposure rate limits which should be used in accomplishing the decontamination and survey of surfaces or premises and equipment prior to abandonment or release for unrestricted use. The limits in Table 1 do not apply to premises, equipment, or scrap containing induced radioactivity for which the radiological considerations pertinent to their use may be different. The release of such facilities or items from regulatory control will be considered on a case-by-case basis.

1. The licensee shall make a reasonable effort to eliminate residual contamination.

2. Radioactivity on equipment or surfaces shall not be covered by paint, plating, or other covering material unless contamination levels, as determined by a survey and documented, are below the limits specified in Table 1 prior to applying the covering. A reasonable effort must be made to minimize the contamination prior to use of any covering.

3. The radioactivity on the interior surfaces of pipes, drain lines, or duct work shall be determined by making measurements at all traps, and other appropriate access points, provided that contamination at these locations is likely to be representative of contamination on the interior of the pipes, drain lines, or duct work. Surfaces of premises, equipment, or scrap which are likely to be contaminated but are of such size, construction, or location as to make the surface inaccessible for purposes of measurement shall be presumed to be contaminated in excess of the limits.

4. Upon request, the Commission may authorize a licensee to relinquish possession or control of premises, equipment, or scrap having surfaces contaminated with materials in excess of the limits specified. This may include, but would not be limited to, special circumstances such as razing of buildings, transfer of premises to another organization continuing work with radioactive materials, or conversion of facilities to a long-term storage or standby status. Such request must:

   a. Provide detailed, specific information describing the premises, equipment or scrap, radioactive contaminants, and the nature, extent, and degree of residual surface contamination.

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b. Provide a detailed health and safety analysis which reflects that the residual amounts of materials on surface areas, together with other considerations such as prospective use of the premises, equipment or scrap, are unlikely to result in an unreasonable risk to the health and safety of the public.

5. Prior to release of premises for unrestricted use, the licensee shall make a comprehensive radiation survey which establishes that contamination is within the limits specified in Table 1. A copy of the survey report shall be filed with the Division of Fuel Cycle and Material Safety, USNRC, Washington, D.C. 20555, and also the Director of the Regional Office of the Office of Inspection and Enforcement, USNRC, having jurisdiction. The report should be filed at least 30 days prior to the planned date of abandonment. The survey report shall:

a. Identify the premises.

b. Show that reasonable effort has been made to eliminate residual contamination.

c. Describe the scope of the survey and general procedures followed.

d. State the findings of the survey in units specified in the instruction.

Following review of the report, the NRC will consider visiting the facilities to confirm the survey.

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TABLE 3

ACCEPTABLE SURFACE CONTAMINATION LIMITS

NUCLIDES[a]	AVERAGE[bcf]	MAXIMUM[bdf]	REMOVABLE[bef]
U-nat, ^{235}U, ^{238}U and associated decay products	5000 dis/min-100 cm^2 α	15,000 dis/min-100 cm^2 α	1000 dis/min-100 cm^2 α
Transuranics, ^{226}Ra, ^{228}Ra, ^{230}Th, ^{228}Th, ^{231}Pa, ^{227}Ac, ^{125}I, ^{129}I	100 dis/min-100 cm^2	300 dis/min-100 cm^2	20 dis/min-100 cm^2
Th-nat, ^{232}Th ^{90}Sr, ^{223}Ra, ^{224}Ra, ^{232}U, ^{126}I, ^{131}I, ^{133}I	1000 dis/min-100 cm^2	3,000 dis/min-100 cm^2	200 dis/min-100 cm^2
Beta-gamma emitters (nuclides with decay modes other than alpha emission or spontaneous fission) except ^{90}Sr and others noted above.	5000 dis/min-100 cm^2 $\beta\gamma$	15,000 dis/min-100 cm^2 $\beta\gamma$	1000 dis/min-100 cm^2 $\beta\gamma$

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(Cont'd.)

TABLE 3

ACCEPTABLE SURFACE CONTAMINATION LEVELS

[a]Where surface contamination by both alpha and beta-gamma emitting nuclides exists, the limits established for alpha and beta-gamma emitting nuclides should apply independently.

[b]As used in this table, dis/min (disintegrations per minute) means the rate of emission by radioactive material as determined by correcting the counts per minute observed by an appropriate detector for background, efficiency, and geometric factors associated with the instrumentation.

[c]Measurements of average contaminant should not be averaged over more than 1 square meter. For objects of less surface area, the average should be derived for each such object.

[d]The maximum contamination level applies to an area of not more than 100 cm^2.

[e]The amount of removable radioactive material per 100 cm^2 of surface area should be determined by wiping that area with dry filter or soft absorbent paper, applying moderate pressure, and assessing the amount of radioactive material on the wipe with an appropriate instrument of known efficiency. When removable contamination on objects of less surface area is determined, the pertinent levels should be reduced proportionally and the entire surface should be wiped.

[f]The average and maximum radiation levels associated with surface contamination resulting from beta-gamma emitters should not exceed 0.2 mrad/h at 1 cm and 1.0 mrad/h at 1 cm, respectively, measured through not more than 7 milligrams per square centimeter of total absorber.

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III. SURGEON GENERAL'S GUIDELINES
as included in 10 CFR Part 712
Grand Junction Remedial Action Criteria

712.1 Purpose

(a) determination by DOE of the need for, priority of and selection of appropriate remedial action to limit the exposure of individuals in the area of Grand Junction, Colorado, to radiation emanating from uranium mill tailings which have been used as construction-related material.

(b) The regulations in this part are issued pursuant to Pub. L. 92-314 (86 Stat. 222) of June 16, 1972.

712.2 Scope

The regulations in this part apply to all structures in the area of Grand Junction, Colorado, under or adjacent to which uranium mill tailings have been used as a construction-related material between January 1, 1951, and June 16, 1972, inclusive.

712.3 Definitions

As used in this part:

(a) "Administrator" means the Administrator of Energy Research and Development or his duly authorized representative.

(b) "Area of Grand Junction, Colorado," means Mesa County, Colorado.

(c) "Background" means radiation arising from cosmic rays and radioactive material other than uranium mill tailings.

(d) "DOE" means the U.S. Department of Energy or any duly authorized representative thereof.

(e) "Construction-related material" means any material used in the construction of a structure.

(f) "External gamma radiation level" means the average gamma radiation exposure rate for the habitable area of a structure as measured near floor level.

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(g) "Indoor radon daughter concentration level" means that concentration of radon daughters determined by: (1) averaging the results of six air samples each of at least 100 hours duration, and taken at a minimum of 4-week intervals throughout the year in a habitable area of a structure, or (2) utilizing some other procedure approved by the Commission.

(h) "Milliroentgen" (mR) means a unit equal to one-thousandth (1/1000) of a roentgen which roentgen is defined as an exposure dose of X or gamma radiation such that the associated corpuscular emission per 0.001293 gram of air produces, in air, ions carrying one electrostatic unit of quantity of electricity of either sign.

(i) "Radiation" means the electromagnetic energy (gamma) and the particulate radiation (alpha and beta) which emanate from the radioactive decay of radium and its daughter products.

(j) "Radon daughters" means the consecutive decay products of radon-222. Generally, these include Radium A (polonium-218), Radium B (lead-214), Radium C (bismuth-214), and Radium C' (polonium-214).

(k) "Remedial action" means any action taken with a reasonable expectation of reducing the radiation exposure resulting from uranium mill tailings which have been used as construction-related material in and around structures in the area of Grand Junction, Colorado.

(l) "Surgeon General's Guidelines" means radiation guidelines related to uranium mill tailings prepared and released by the Office of the U.S. Surgeon General, Department of Health, Education and Welfare on July 27, 1970.

(m) "Uranium mill tailings" means tailings from a uranium milling operation involved in the Federal uranium procurement program.

(n) "Working Level" (WL) means any combination of short-lived radon daughter products in 1 liter of air that will result in the ultimate emission of 1.3×10^5 MeV of potential alpha energy.

712.4 Interpretations

Except as specifically authorized by the Administrator in writing, no interpretation of the meaning of the regulations in this part by an officer or employee of DOE other than a written interpretation by the General Counsel will be recognized to be binding upon DOE.

712.5 Communications

Except where otherwise specified in this part, all communications concerning the regulations in this part should be addressed to the Director, Division of Safety, Standards, and Compliance, U.S. Department of Energy, Washington, D.C. 20545.

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712.6 General radiation exposure level criteria for remedial action.

The basis for undertaking remedial action shall be the applicable guidelines published by the Surgeon General of the United States. These guidelines recommended the following graded action levels for remedial action in terms of external gamma radiation level (EGR) and indoor radon daughter concentration level (RDC) above background found within dwellings constructed on or with uranium mill tailings.

EGR	RDC	Recommendation
Greater than 0.1 mR/h	Greater than 0.05 WL	Remedial action indicated.
From 0.05 to 0.1 mR/h	From 0.01 to 0.05 WL	Remedial action may be suggested.
Less than 0.05 mR/h	Less than 0.01 WL	No remedial action indicated

712.7 Criteria for determination of possible need for remedial action

Once it is determined that a possible need for remedial action exists, the record owner of a structure shall be notified of that structure's eligibility for an engineering assessment to confirm the need for remedial action and to ascertain the most appropriate remedial measure, if any. A determination of possible need will be made if as a result of the presence of uranium mill tailings under or adjacent to the structure, one of the following criteria is met:

(a) Where DOE approved data on indoor radon daughter concentration levels are available

(1) For dwellings and schoolrooms: An indoor radon daughter concentration level of 0.01 WL or greater above background.

(2) For other structures: An indoor radon daughter concentration level of 0.03 WL or greater above background.

(b) Where DOE approved data on indoor radon daughter concentration levels are not available:

(1) For dwellings and schoolrooms:

(i) An external gamma radiation level of 0.05 mR/h or greater above background.

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(ii) An indoor radon daughter concentration level of 0.01 WL or greater above background (presumed).

(A) It may be presumed that if the external gamma radiation level is equal to or exceed 0.02 mR/h above background, the indoor radon daughter concentration level equals or exceeds 0.01 WL above background.

(B) It should be presumed that if the external gamma radiation level is less than 0.001 mR/h above background, the indoor radon daughter concentration level is less than 0.01 WL above background, and no possible need for remedial actions exists.

(C) If the external gamma radiation level is equal to or greater than 0.001 mR/h above background but is less than 0.02 mR/h above background, measurements will be required to ascertain the indoor radon daughter concentration level.

(2) For other structures:

(i) An external gamma radiation level of 0.15 mR/h above background averaged on a room-by-room basis.

(ii) No presumptions shall be made on the external gamma radiation level/indoor radon daughter concentration level relationship. Decisions will be made in individual cases based upon the results of actual measurements.

712.8 Determination of possible need for remedial action where criteria have not been met.

The possible need for remedial action may be determined where the criteria in 712.7 have not been met if various other factors are present. Such factors include, but are not necessarily limited to, size of the affected area, distribution of radiation levels in the affected area, amount of tailings, age of individuals occupying affected area, occupancy time, and use of the affected area.

712.9 Factors to be considered in determination of order of priority for remedial action.

In determining the order or priority for execution of remedial action, consideration shall be given, but not necessarily limited to, the following factors:

(a) Classification of structure. Dwellings and schools shall be considered first.

(b) Availability of data. Those structures for which data on indoor radon daughter concentration levels and/or external gamma radiation levels are available when the program starts and which meet the criteria in 712.7 will be considered first.

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(c) Order of application. Insofar as feasible remedial action will be taken in the order in which the application is received.

(d) Magnitude of radiation level. In general, those structures with the highest radiation levels will be given primary consideration.

(e) Geographical location of structures. A group of structures located in the same immediate geographical vicinity may be given priority consideration particularly where they involve similar remedial efforts.

(f) Availability of structures. An attempt will be made to schedule remedial action during those periods when remedial action can be taken with minimum interference.

(g) Climatic conditions. Climatic conditions or other seasonable considerations may affect the scheduling of certain remedial measures.

712.10 Selection of appropriate remedial action.

(a) Tailings will be removed from those structures where the appropriately averaged external gamma radiation level is equal to or greater than 0.05 mR/h above background in the case of dwellings and schools and 0.15 mR/h above background in the case of other structures.

(b) Where the criterion in paragraph (a) of this section is not met, other remedial action techniques, including but not limited to sealants, ventilation, and shielding may be considered in addition to that of tailings removal. DOE shall select the remedial action technique or combination of techniques, which it determines to be the most appropriate under the circumstances.

IV. 40 CFR Part 192
HEALTH AND ENVIRONMENTAL PROTECTION STANDARDS FOR URANIUM MILL TAILINGS

SUBPART A--Standards for the Control of Residual Radioactive Materials from Inactive Uranium Processing Sites

192.00 Applicability

This subpart applies to the control of residual radioactive material at designated processing or depository sites under Section 108 of the Uranium Mill Tailings Radiation Control Act of 1978 (henceforth designated "the Act"), and to restoration of such sites following any use of subsurface minerals under Section 104(h) of the Act.

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192.01 Definitions

(a) Unless otherwise indicated in this subpart, all terms shall have the same meaning as in Title I of the Act.

(b) Remedial action means any action performed under Section 108 of the Act.

(c) Control means any remedial action intended to stabilize, inhibit future use of, or reduce emissions or effluents from residual radioactive materials.

(d) Disposal site means the region within the smallest perimeter of residual radioactive material (excluding cover materials) following completion of control activities.

(e) Depository site means a disposal site (other than a processing site) selected under Section 104(b) or 105(b) of the Act.

(f) Curie (Ci) means the amount of radioactive material that produces 37 billion nuclear transformation per second. One picocurie (pCi) = 10^{-12} Ci.

192.02 Standards

Control shall be designed* to:

(a) be effective for up to one thousand years, to the extent reasonably achievable, and, in any case, for at least 200 years, and,

(b) provide reasonable assurance that releases of radon-222 from residual radioactive material to the atmosphere will not:

(1) exceed an average** release rate of 20 picocuries per square meter per second, or

---

*Because the standard applies to design, monitoring after disposal is not required to demonstrate compliance.

**This average shall apply over the entire surface of the disposal site and over at least a one-year period. Radon will come from both residual radioactive materials and from materials covering them. Radon emissions from the covering materials should be estimated as part of developing a remedial action plan for each site. The standard, however, applies only to emissions from residual radioactive materials to the atmosphere.

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(2) increase the annual average concentration of radon-222 in air at or above any location outside the disposal site by more than one-half picocurie per liter.

SUBPART B--Standards for Cleanup of Open Lands and Buildings Contaminated with Residual Radioactive Materials from Inactive Uranium Processing Sites

192.10 Applicability

This subpart applies to land and buildings which are part of any processing site designated by the Secretary of Energy under Pub. L. 95-604, Section 102. Section 101 of Pub. L. 95-604, states that "processing site" means--

(a) any site, including the mill, containing residual radioactive materials at which all or substantially all of the uranium was produced for sale to any Federal agency prior to January 1, 1971, under a contract with any Federal agency, except in the case of a site at or near Slick Rock, Colorado, unless--

(1) such site was owned or controlled as of January 1, 1978, or is thereafter owned or controlled, by an Federal agency, or

(2) a license [issues by the (Nuclear Regulatory) Commission or its predecessor agency under the Atomic Energy Act of 1954 or by a State as permitted under Section 274 of such Act] for the production at such site of any uranium or thorium product dervied from ores is in effect on January 1, 1978, or is issued or renewed after such date; and

(b) Any other real property or improvement thereon which--

(1) is in the vicinity of such site, and

(2) is determined by the Secretary, in consultation with the Commission, to be contaminated with residual radioactive materials derived from such site.

192.11 Definitions

(a) Unless otherwise indicated in this subpart, all terms shall have the same meaning as defined in Title I of the Act or in Subpart A.

(b) Land means any surface or subsurface land that is not part of a disposal site and is not covered by an occupiable building.

(c) Working Level (WL) means combination of short-lived radon decay products in one liter of air that will result in the ultimate emission of alpha particles with a total energy of 130 billion electron volts.

(d) Soil means all unconsolidated materials normally found or near the surface of the earth including, but not limited to silts, clays, sands, gravel, and small rocks.

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192.12 Standards

Remedial actions shall be conducted so as to provide reasonable assurance that, as a result of residual radioactive materials from any designated processing site:

(a) the concentration of radium-226 in land averaged over any area of 100 square meters shall not exceed the background level by more than---

(1) 5 pCi/g, averaged over the first 15 cm of soil below the surface, and

(2) 15 pCi/g, averaged 15 cm thick layers of soil more than 15 cm below the surface.

(b) in any occupied or habitable building---

(1) the objective of remedial action shall be, and reasonable effort shall be made to achieve, an annual average (or equivalent) radon decay product concentration (including background) not to exceed 0.02 WL. In any case, the radon decay product concentration (including background) shall not exceed 0.03 WL, and

(2) the level of gamma radiation shall not exceed the background level by more than 20 microroentgens per hour.

SUBPART C--Implementation

192.20 Guidance for Implementation

Section 108 of the Act requires the Secretary of Energy to select and perform remedial actions with the concurrence of the Nuclear Regulatory Commission and the full participation of any State that pays part of the cost, and in consultation, as appropriate, with affected Indian Tribes and the Secretary of the Interior. These parties, in their respective roles under Section 108, are referred to hereafter as "the implementing agencies."

The implementing agencies shall establish methods and procedures to provide "reasonable assurance" that the provisions of Subparts A and B are satisfied. This should be done as appropriate through use of analytic models and site-specific analyses, in the case of Subpart A, and for Subpart B through measurements performed within the accuracy of currently available types of field and laboratory instruments in conjunction with reasonable survey and sampling procedures. These methods and procedures may be varied to suit conditions at specific sites. In particular:

(a) The purpose of Subpart A is to provide for long-term stabilization and isolation in order to inhibit misuse and spreading of residual radioactive materials, control releases of radon to air, and protect water. Subpart A may be implemented through analysis of the physical properties of the site and the

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control system and projection of the effects of natural processes over time. Events and processes that could significantly affect the average radon release rate from the entire disposal site should be considered. Phenomena that are localized or temporary, such as local cracking or burrowing of rodents, need to be taken into account only if their cumulative effect would be significant in determining compliance with the standard. Computational models, theories, and prevalent expert judgment may be used to decide that a control system design will satisfy the standard. The numerical range provided in the standard for the longevity of the affectiveness of the control of residual radioactive materials allows for consideration of the various factors affecting the longevity of control and stabilization methods and their costs. These factors have different levels of predictability and may vary for the different sites.

Protection of water should be considered in the analysis of reasonable assurance of compliance with the provisions of Section 192.02. Protection of water should be considered on a case-specific basis, drawing on hydrological and geochemical surveys and all other relevant data. The hydrologic and geologic assessment to be conducted at each site should include a monitoring program sufficient to establish background groundwater quality through one or more upgradient wells, and identify the presence and movement of plumes associated with the tailings piles.

If contaminants have been released from a tailings pile, an assessment of the location of the contaminants and the rate and direction of movement of contaminated groundwater, as well as its relative contamination, should be made. In addition, the assessment should identify the attenuative capacity of the unsaturated and saturated zone to determine the extent of plume movement. Judgments on the possible need for remedial or protective actions for groundwater aquifers should be guided by relevant considerations described in EPA's hazardous waste management system (47 FR 32274, July 26, 1982) and by relevant State and Federal Water Quality Criteria for anticipated or existing uses of water over the term of the stabilization. The decision on whether to institute remedial action, what specific action to take, and to what levels an aquifer should be protected or restored should be made on a case-by-case basis taking into account such factors as technical feasibility of improving the aquifer in its hydrogeologic setting, the cost of applicable restorative or protective programs, the present and future value of the aquifer as a water resource, the availability of alternative water supplies, and the degree to which human exposure is likely to occur.

(b) Complinace with Subpart B, to the extent practical, should be demonstrated through radiation surveys. Such surveys may, if appropriate, be restricted to locations likely to contain residual radioactive materials. These surveys should be designed to provide for compliance averaged over limited areas rather than point-by-point compliance with the standards. In most cases, measurement of gamma radiation exposure rates above and below the land surface can be used to show compliance with Section 192.12(a). Protocols for making such measurements should be based on realistic radium distributions near the surface rather than extremes rarely encountered.

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In Section 192.12(a), the "background level" refers to native radium concentration in soil. Since this may not be determinable in the presence of contamination by residual radioactive materials, a surrogate "background level" may be established by simple or indirect (e.g., gamma radiation) measurements performed nearby but outside of the contaminated location.

Compliance with Section 192.12(b) may be demonstrated by methods that the Department of Energy has approved for use under PL 92-314 (10 CFR 712), or by other methods that the implementing agencies determine are adequate. Residual radioactive materials should be removed from buildings exceeding 0.03 WL so that future replacement buildings will not pose a hazard [unless removal is not practical--see Section 192.21(c)]. However, sealants, filtration, and ventilation devices may provide reasonable assurance of reductions from 0.03 WL to below 0 02 WL. In unusual cases, indoor radiation may exceed the levels specified in Section 192.12(b) due to sources other than residual radioactive materials. Remedial actions are not required in order to comply with the standard when there is reasonable assurance that residual radioactive materials are not the cause of such an excess.

192.21 Critiera for Applying Supplemental Standards

The implementing agencies may (and in the case of Subsection (f) shall) apply standards under Section 192.22 in lieu of the standards of Subparts A or B if they determine that any of the following circumstances exists:

(a) Remedial actions required to satsify Subparts A or B would pose a clear and present risk of injury to workers or to members of the public, notwithstanding reasonable measures to avoid or reduce risk.

(b) Remedial actions to satisfy the cleanup standards for land, Section 192.12(a), or the acquisition of minimum materials required for control to satisfy Section 192.02(b), would, notwithstanding reasonable measures to limit damage, directly produce environmental harm that is clearly excessive compared to the health benefits to persons living on or near the site, now or in the future. A clear excess of environmental harm is harm that is long-term, manifest, and grossly disproportionate to health benefits that may reasoably be anticipated.

(c) The estimate cost of remedial action to satisfy Section 192.12(a) at a "vicinity" site [described under Section 101(6)(B) of the Act] is unreasonably high relative to the long-term benefits, and the residual radioactive materials do not pose a clear present or future hazard. The likelihood that buildings will be erected or that people will spend long periods of time at such a vicinity site should be considered in evaluating this hazard. Remedial action will generally not be necessary where residual radioactive materials have been placed semi-permanently in a location whre site-specific factors limit their hazard and from which they are costly or difficult to remove, or where only minor quantities of residual radioactive materials are involved. Examples are

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residual radioactive materials under hard surface public roads and sidewalks, around public sewer lines, or in fence post foundations. Supplemental standards should not be applied at such sites, however, if individuals are likely to be exposed for long periods of time to radiation from such materials at levels above those that would prevail under Section 192.12(a).

(d) The cost of a remedial action for cleanup of a building under Section 192.12(b) is clearly unreasonably high relative to the benefits. Factors that should be included in this judgment are the anticipated period of occupancy, the incremental radiation level that would be affected by the remedial action, the residual useful lifetime of the building, the potential for future construction at the site, and the applicability of less costly remedial methods than removal of residual radioactive materials.

(e) There is no known remedial action.

(f) Radionuclides other than radium-226 and its decay products are present in sufficient quantity and concentration to constitute a significant radiation hazard from residual radioactive materials.

192.22 Supplemental Standards

Federal agencies implementing Subparts A and B may in lieu thereof proceed pursuant to this section with respect to generic or individual situations meeting the eligibility requirements of Section 192.21.

(a) When one or more of the criteria of Section 192.21(a) through (e) applies, the implementing agencies shall select and perform remedial actions that come as close to meeting the otherwise applicable standard as is reasonable under the circumstances.

(b) When Section 192.21(f) applies, remedial actions shall, in addition to satisfying the standards of Subparts A and B, reduce other residual radioactivity to levels that are as low as is reasonably achievable.

(c) The implementing agencies may make general determinations concerning remedial actions under this Section that will apply to all locations with specified characteristics, or they may make a determination for a specific location. When remedial actions are proposed under this Section for a specific location, the Department of Energy shall inform any private owners and occupants of the affected location and solicit their comments. The Department of Energy shall provide any such comments to the other implementing agencies. The Department of Energy shall also periodically inform the Environmental Protection Agency of both general and individual determinations under the provisions of this section.

APPENDIX 6
(cont'd.)

V. EXCERPTS FROM LA-UR-79-1865-Rev.,

"Interim Soil Limits for D&D Projects"

Table XXIII. Recommended Soil Limits[a,b] (in pCi/g)

		Ingestion			
	Inhalation	Home Gardener	Full Diet	External Radiation	All Pathways[c]
^{231}Pa	50	740	150	250	40
^{227}Ac	200[d]	4,900	1,000	300	120[d]
^{232}Th	45	670	140	40	20
^{228}Th	1,000	37,000	7,800	55	50
^{230}Th (No Daught.)	300	4,400	940	36,000	280
^{238}U-^{234}U	750	44	8	6,000	40
^{90}Sr	$2x10^6$	100	19	-	100
^{137}Cs	$7x10^6$	800	1	90	80

[a] Soil limits for ^{241}Am and $^{239,240}Pu$ are available from EPA recommendations, and a soil limit for ^{226}Ra has been reported by Healy and Rodgers.

[b] Limits are to apply to only one nuclide present in the soil. If more than one is present, a weighted average should apply.

[c] Based on a diet of a home gardener.

[d] Modified from LA-UR-79-1865-Rev. values to correct error.

APPENDIX 6
(cont'd.)

VI. EXCERPTS FROM DOE 5480.1, Chapter XI

"Requirements for Radiation Protection"

Exposure of Individuals and Population Groups in Uncontrolled Areas. Exposures to members of the public shall be as low as reasonably achievable levels within the standards prescribed below.

Radiation Protection Standards
for Internal and External Exposure
of Members of the Public

	Annual Dose Equivalent or Dose Commitment	
Type of Exposure	Based on Dose to Individuals at Points of Maximum Probable Exposure	Based on Average Dose to a Suitable Sample of the Exposed Population
Whole body, gonads, or bone marrow	0.5 rem (or 500 mrem)	0.17 rem (or 170 mrem)
Other organs	1.5 rem (or 1500 mrem)	0.5 rem (or 500 mrem)

CONCENTRATIONS IN AIR AND WATER ABOVE NATURAL BACKGROUND

			Table I Controlled Area		Table II Uncontrolled Area	
Element (atomic number)	Isotope	(I) soluble (S)	Column 1 Air (pCi/ℓ)	Column 2 Water (pCi/ℓ)	Column 1 Air (pCi/ℓ)	Column 2 Water (pCi/ℓ)
Radon (86)	Rn 220	S	300		10	
	Rn 222	S	100		3	

APPENDIX 7

ESTIMATED EXTENT OF CONTAMINATION

Estimates of the extent of the contamination at the assessed site are based on the total volume, mass, and quantity of radioactive material in the contaminated area. The volume is the product of the surface area and the depth of the contamination. The mass is the product of the volume and the density of the material. A density of 1.5 g/cm^3 is used for soil. The concentration (pCi/g) of the specific radioisotope is determined by radiochemical analysis of the soil The total quantity of radioactive material is the product of the concentration of the specific radioisotope and the total mass of material.

Often there is more than one contaminant in the soil (or contaminated material) and the contaminants are not uniformly distributed throughout the material. In these cases, it is necessary to estimate the fraction of the material containing each contaminant in order to assess the total quantity of the radioactive material. This estimate of the fraction of the material containing each contaminant is based on the radiochemical analysis of randomly selected samples.

Estimates of the extent of contamination are usually determined for averaged (Option 1) and maximum or worst-case (Option 2) conditions. A sample calculation for the extent of contamination in a typical site is as follows:

Volume (Average) = 34,800 ft^2 (area) x 3.6 ft (avg. depth) = 125,000 ft^3
= 3,550 m^3

Volume (Maximum) = 34,800 ft^2 (area) x 9 ft (max. depth) = 314,000 ft^3
= 8,880 m^3

Mass (Average) = 3,550 m^3 x 1,500 kg/m^3 = 5.33 x 10^6 kg
Mass (Maximum) = 8,880 m^3 x 1,500 kg/m^3 = 1.33 x 10^7 kg

Estimated Total Activity for ^{226}Ra (chain)

Average: 5.33 x 10^6 kg x 14 x 10^{-12} Ci/g x 10^3 g/kg x .05 (fraction)* = 0.004 Ci

Maximum: 1.33 x 10^7 kg x 16 x 10^{-12} Ci/g x 10^3 g/kg x .05 (fraction)* = 0.011 Ci

*This represents the estimate of the fraction of the total mass contaminated with the ^{226}Ra chain.

APPENDIX 8

EVALUATION OF RADIATION EXPOSURES

INTRODUCTION

A. Types of Radiation

Radiation is the emission or transmission of energy in the form of waves or particles. Examples are acoustic waves (i.e., sound), electromagnetic waves (such as radio, light, x- and gamma-rays), and particulate radiations (such as alpha particles, beta particles, neutrons, protons, and other elementary particles).

The class of radiation of importance to this report is known as ionizing radiation. Ionizing radiations are those, either electromagnetic or particulate, with sufficient energy to ionize matter, i.e., to remove or displace electrons from atoms and molecules. The most common types of ionizing radiation are x- and gamma-rays, alpha particles, beta particles, and neutrons.

X- and gamma-rays are electromagnetic waves of pure energy, having no charge and no mass or existence at rest. Gamma-rays and x-rays are identical except that x-rays originate in the atom and gamma-rays originate in the nucleus of an atom. X- and gamma-rays are highly penetrating and can pass through relatively thick materials before interacting. Upon interaction, some or all of the energy is transferred to electrons which, in turn, produce additional ionizations while coming to rest.

Alpha particles are positively charged particulates composed of two neutrons and two protons, identical to the nucleus of a helium atom. Due to its comparatively large mass and double charge, an alpha particle interacts readily with matter and penetrates only a very short distance before coming to rest, causing intense ionization along its path.

Beta particles are negatively charged free electrons moving at high speeds. Due to its comparatively small mass and single charge, a beta particle's penetration through matter is intermediate between that of the alpha particle and the gamma-ray, causing fewer ionizations per unit path length than an alpha particle.

B. Sources of Radiation

Ionizing radiations arise from terrestrial radioactive materials (both naturally occurring and man-made), extra-terrestrial (cosmic) sources, and radiation-producing machines. The sources of ionizing radiation important to this report are radioactive materials and cosmic sources.

Most atoms of the elements in our environment remain structurally stable. With time, an atom of potassium, for instance, may change its association with other atoms in chemical reactions and become part of other compounds, but it will always remain a potassium atom. Radioactive atoms, on the other hand, are

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(cont'd.)

not stable and will spontaneously emit radiation in order to achieve a more stable state. Because of this spontaneous transformation, the ratio of protons and neutrons in the nucleus of an atom is altered toward a more stable condition. Radiation may be emitted from the nucleus as alpha particles, beta particles, neutrons, or gamma-rays, depending uniquely upon each particular radionuclide. Radionuclides decay at characteristic rates dependent upon the degree of stability and characterized by a period of time called the half-life. In one half-life, the number of radioactive atoms and, therefore, the amount of radiation emitted, decrease by one half.

The exposure of man to terrestrial radiation is due to naturally occurring radionuclides and also to "man-made" or technologically enhanced radioactive materials. Several dozen radionuclides occur naturally, some having half-lives of at least the same order of magnitude as the estimated age of the earth. The majority of these naturally occurring radionuclides are isotopes of the heavy elements and belong to three distinct radioactive series headed by uranium-238, uranium-235, and thorium-232. Each of these decays to stable isotopes of lead (Pb) through a sequence of radionuclides of widely varying half-lives. Other naturally occurring radionuclides, which decay directly to a stable nuclide, are potassium-40 and rubidium-87. It should be noted that even though the isotopic abundance of potassium-40 is less than 0.012%, potassium is so widespread that potassium-40 contributes about one-third of the radiation dose received by man from natural background radiation. A major portion of the exposure (dose) of man from external terrestrial radiation is due to the radionuclides in the soil, primarily potassium-40 and the radioactive decay-chain products of thorium-232 and uranium-238. The naturally occurring radionuclides deposited internally in man through uptake by inhalation/ingestion of air, food, and drinking water containing the natural radioactive material also contribute significantly to his total dose. Many other radionuclides are referred to as "man made" in the sense that they can be produced in large quantities by such means as nuclear reactors, accelerators, or nuclear weapons tests.

The term "cosmic radiation" refers both to the primary energetic particles of extra-terrestrial origin that are incident on the earth's atmosphere and to the secondary particles that are generated by the interaction of these primary particles with the atmosphere, and reach ground level. Primary cosmic radiation consists of "galactic" particles externally incident on the solar system, and "solar" particles emitted by the sun. This radiation is composed primarily of energetic protons and alpha particles. The first generation of secondary particles (secondary cosmic radiation), produced by nuclear interactions of the primary particles with the atmosphere, consists predominantly of neutrons, protons, and pions. Pion decay, in turn, results in the production of electrons, photons, and muons. At the lower elevations, the highly penetrating muons and their associated decay and collision electrons are the dominant components of the cosmic-ray particle flux density. These particles, together with photons from the gamma-emitting, naturally occurring radionuclides in the local environment, form the external penetrating component of the background environmental radiation field which provides a significant portion of the whole-body radiation dose to man.

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(cont'd.)

In addition to the direct cosmic radiation, cosmic sources include cosmic-ray-produced radioactivity, i.e., cosmogenic radionuclides. The major production of cosmogenic radionuclides is through interaction of the cosmic rays with the atmospheric gases through a variety of spallation or neutron-capture reactions. The four cosmogenic radionuclides that contribute a measurable radiation dose to man are carbon-14, sodium-22, beryllium-7, and tritium (hydrogen-3), all produced in the atmosphere.

BACKGROUND RADIATION DOSES

Background radiation doses are comprised of an external component of radiation impinging on man from outside the body and an internal component due to radioactive materials taken into the body by inhalation or ingestion.

Radiation dose may be expressed in units of rads or rems, depending upon whether the reference is to the energy deposited or to the biological effect. A rad is the amount of radiation that deposits a certain amount of energy in each gram of material. It applies to all radiations and to all materials which absorb that radiation.

Since different types of radiation produce ionizations at different rates as they pass through tissue, differences in damage to tissues (and hence the biological effectiveness of different radiations) has been noticed. A rem is defined as the amount of energy absorbed (in rads) from a given type of radiation multiplied by the factor appropriate for the particular type of radiation in order to approximate the biological damage that it causes relative to a rad of x or gamma radiation. The concept behind the unit "rem" permits evaluation of potential effects from radiation exposure without regard to the type of radiation or its source. One rem received from cosmic radiation results in the same biological effects as one rem from medical x-rays or one rem from the radiations emitted by naturally occurring or man-made radioactive materials.

The external penetrating radiation dose to man derives from both terrestrial radioactivity and cosmic radiation. The terrestrial component is due primarily to the gamma dose from potassium-40 and the radioactive decay products of thorium-232 and uranium-238 in soil as well as from the beta-gamma dose from radon daughters in the atmosphere. Radon is a gaseous member of the uranium-238 chain. The population-weighted external dose to an individual's whole body from terrestrial sources in the United States has been estimated as 15 mrem per year for the Atlantic and Gulf Coastal Plain, 57 mrem per year for an indeterminate area along the Rocky Mountains, and 29 mrem per year for the majority of the rest of the United States. The overall population-weighted external dose for the U.S. population as a whole has been estimated to be 26 mrem per year.

The cosmic radiation dose, due to the charged particles and neutrons from secondary cosmic rays, is typically about 30% to 50% of the total from all external environmental radiation. The cosmic-ray dose to the population is estimated to be 26 mrem per year for those living at sea level, and increases with increasing altitude. Considering the altitude distribution of the U.S.

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(cont'd.)

population, the population-weighted external cosmic-ray dose is 28 mrem per year. The population-weighted total external dose from terrestrial plus cosmic sources is thus 54 mrem per year for the U.S. population as a whole.

The internal radiation doses derive from terrestrial and cosmogenic radionuclides deposited within the body through uptake by inhalation/ingestion of air, food, and drinking water. Once deposited in the body, many radioactive materials can be incorporated into tissues because the chemical properties of the radioisotopes are identical or similar to stable isotopes in the tissues. Potassium-40, for instance, is incorporated into tissues in the same manner as stable potassium atoms because the chemical properties are identical; radioactive radium and strontium can be incorporated into tissues in the same manner as calcium because their chemical properties are similar. Once deposited in tissue, these radionuclides emit radiation that results in the internal dose to individual organs and/or the whole body as long as it is in the body.

The internal dose to the lung is due primarily to the inhalation of polonium-218 and -214 (radon daughters), lead-212 and bismuth-212 (thoron daughters) and polonium-210 (one of the longer-lived radon decay products). The dose to the lung is about 100 mrem per year from inhaled natural radioactivity. The internal dose from subsequent incorporation of inhaled or ingested radioactivity is due to a beta-gamma dose from incorporation of potassium-40, rubidium-87, and cosmogenic nuclides, and an alpha dose from incorporation of primarily polonium-210, radium-226 and -228, and uranium-238 and -234. The dose to man from internally incorporated radionuclides is about 28 mrem per year to the gonads, about 25 mrem per year to the bone marrow, lung, and other soft tissues, and about 117 mrem per year to the bone (osteocytes). The bone dose arises primarily from the alpha-emitting members of the naturally occurring series, with polonium-210 being the largest contributor. The gonadal and soft tissue doses arise primarily from the beta and gamma emissions from potassium-40. The total internal dose from inhaled plus incorporated radioactivity is about 28 mrem per year to the gonads (or whole-body dose), about 125 mrem per year to the lung, about 25 mrem per year to the bone marrow, and about 117 mrem per year to the bone (osteocytes).

The total natural background radiation dose is the sum of the external and internal components. The population-weighted dose for the U.S. population as a whole is about 82 mrem per year to the gonads or whole body, about 179 mrem per year to the lung, about 79 mrem per year to the bone marrow, and about 171 mrem per year to the bone (osteocytes).

Besides the natural background radiation, background radiation doses include contributions from man-made or technologically enhanced sources of radiation. By far, the most significant are x-ray and radiopharmaceutical medical examinations. These contribute a population-averaged dose estimated to be 70 mrem per year for the U.S. population as a whole. Fallout from nuclear weapons testing through 1970 has contributed 50-year dose commitments estimated as 80 mrem external, and 30, 20, and 45 mrem internal to the gonads, lung, and bone marrow, respectively. Contributions from the use of fossil fuels (natural

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(cont'd.)

gas and coal) and nuclear reactors; mining, milling, and tailings piles; television sets, smoke detectors, and watch dials could be responsible for an additional 5 mrem per year, averaged over the U.S. population as a whole. In addition, the use of radiation or radioactivity for scientific, industrial, or medical purposes may cause workers in the industry and, to a lesser extent, members of the general public, to receive some radiation exposure above natural background.

EVALUATION OF RADIATION DOSE AND POTENTIAL HAZARD

Radiation, regardless of its sources, is considered to be a hazard because of its potential for producing adverse effects on human life. Very large amounts of radiation received over a brief period, i.e., hundreds of rem delivered within a few hours, can produce severe injury or death within days or weeks. Distributed over longer intervals, however, these same doses would not cause early illness or fatality. At doses and rates too low to produce these immediate symptoms, chronic or repeated exposure to radiation can bring about biological damage which does not appear until years or decades later. These low-level effects are stochastic in nature; their probability rather than their severity increases with dose. Primary among these latent or delayed effects are somatic effects, where insults such as cancers occur directly to the individual exposed, and genetic defects, where, through damage to the reproductive cells of the exposed individual, disability and disease ranging from subtle to severe are transmitted to his offspring.

Clinical or observed evidence of a relationship between radiation and human cancers arise from several sources. The most important data come from the victims of Hiroshima and Nagasaki, patients exposed during medical therapy, radium dial painters, and uranium miners. Data exist only for relatively large doses; there have been no direct measurements of increased incidence of cancer for low-level radiation exposures. Evaluation of the available data has lead to estimates of the risk of radiation-induced cancer; estimated risks for the lower doses have been derived by linear extrapolation from the higher doses. All radiation exposures then, no matter how small, are assumed to be capable of increasing an individual's risk of contracting cancer.

Data on genetic defects resulting from radiation exposure of humans is not available to the extent necessary to allow an estimate of the risk of radiation-induced effects. Data from animals, along with general knowledge of genetics, have been used to derive an estimate of the risks of genetic effects.

Estimates of health effects from radiation doses are usually based on risk factors as provided in International Commission on Radiological Protection (ICRP), National Research Council Advisory Committee on the Biological Effects of Ionizing Radiation (BEIR), or United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) reports. Multiplying the estimated dose by the appropriate risk factor provides an estimate of the risk or probability of induction of health effects to an individual or his descendants as a result of that exposure. The evaluation of these risk factors is presently subject to large uncertainties and, therefore, potential continual revision. The risk

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(cont'd.)

factors recommended by the ICRP for cancer mortality and hereditary ill health to the first and second generations are 10^{-4} per rem of whole-body dose and 4×10^{-5} per rem of gonadal dose, respectively. As an example, a whole-body dose of 1 rem would be estimated to add a risk of cancer mortality to the exposed invididual of 10^{-4}, i.e., 1 chance in 10,000. However, a precise numerical value cannot be assigned with any certainty to a particular individual's increase in risk attributable to radiation exposure. The reasons for this are numerous and include the following: (1) uncertainties over the influence of the individual's age, state of health, personal habits, family medical history, and previous or concurrent exposure to other cancer-causing agents, (2) the variability in the latent period (time between exposure and physical evidence of disease), and (3) the uncertainty in the risk factor itself.

To be meaningful, an attempt should be made to view such risk estimates in the appropriate context. One useful comparison is with risks encountered in normal life. Another comparison, potentially more useful, is with an estimation of the risks attributable to natural background radiation. Radiation from natural external and internal radioactivity results in the same types of interactions with body tissues as that from "man-made" radioactivity. Hence, the risks from a specified dose are the same regardless of the source. Rather than going through an intermediate step involving risk factors, doses can also be compared directly to natural background radiation doses.

Besides estimation of risks and comparisons to natural background, doses may be compared to standards and regulations. The appropriate standards, the Department of Energy's "Requirements for Radiation Protection," give limits for external and internal exposures for the whole body and specified organs which are expressed as the permissible dose or dose commitment annually in addition to natural background and medical exposures. There are, in general, two sets of limits, one applicable to occupationally exposed persons and the second applicable to individuals and population groups of the general public. The limits for individuals of the public are one-tenth of those permitted for occupationally exposed individuals. The set of limits important to this report are those applicable to individuals and population groups of the public. The limits for individuals of the public are 500 mrem per year to the whole body, gonads, or bone marrow and 1500 mrem per year to other organs. The limits for population groups of the public are 170 mrem to the whole body, gonads, or bone marrow and 500 mrem per year to other organs, averaged over the group. In either case, exposures are to be limited to the lowest levels reasonably achievable within the given limits.

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(cont'd.)

DOSE DETERMINATION CALCULATION

External Exposure

External penetrating radiation dose rates are measured on contact with an end-window beta-gamma Geiger-Mueller (GM) detector (7 mg/cm^2 window), and at 1 meter with a NaI crystal detector (1 in diameter by 1 in thick) manufactured by Eberline Corporation (PRM-7 μR Meter). For the purpose of these calculations, the following conservative assumptions are made. First, it is assumed that the half-life of the contaminant is long and, therefore, the dose rate is constant with respect to time. Second, it is assumed that a person is stationary at the location of maximum dose for 40 hours per week for 50 weeks per year. For such a situation, the annual dose (A) for a 0.3 mR/h radiation field (about ten times normal background) would be:

$$A = 40 \text{ hr/week} \times 50 \text{ weeks/yr} \times 0.3 \text{ mR/h} = 600 \text{ mR/yr}$$

For the purposes of this example it is assumed that one milliRoentgen of penetrating radiation is equivalent to one millirem of dose. Hence, the maximum dose for this case would be 600 mrem. This value is then compared with the allowable limit of 500 mrem per year for a person non-occupationally exposed.[1]

Internal Exposure

The internal radiological hazard from inhalation/ingestion of contamination is assessed by postulating hypothetical "worst case" scenarios. To this end two cases are considered. The first case is based on the situation whereby a child would eat 100 g per year of the contaminated soil. The second case assumes a home gardener would rototill the contaminated soil (dry) to a 1-ft depth for a working day (eight hours) once a year. For this latter case, a resuspension factor of 10^{-6} m^{-1}, a breathing rate of 9.6 m^3/working day and a soil density of 1.5 g/cm^3, are used. In both cases it is assumed that the average concentration of contaminants in the soil is equal to the maximum measured value (a conservative assumption). All calculations are based on methods outlined in ORNL/NUREG/TM-190, Vol. 3.[2] These calculations approximate the ICRP-30 guidelines for hazard analysis.

The adult inhalation and ingestion dose commitment factors for the bone, lung and total body from ^{238}U, ^{235}U and ^{232}Th (and all their significant daughters), as determined in Reference 2, are given in Table 1. For the purposes of these calculations, normal uranium is assumed to be composed of 2.26% ^{235}U, and 97.74% ^{234}U and ^{238}U (in equilibrium) by activity.

Based on the above scenarios and assuming a soil contamination of 5 pCi/g of radium (EPA limits) the following hazard levels (50-year dose commitment for one year of intake) can be calculated.

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(cont'd.)

Ingestion (consumption of 100 g/yr of soil):

Bone: 6.5×10^{-2} mrem/pCi x 5 pCi/g x 100 g = 32.5 mrem
Total Body: 5.5×10^{-3} mrem/pCi x 5 pCi/g x 100 g = 2.75 mrem

Inhalation (rototilling and breathing ensuing aerosol):

Lung: 1.1×10^{-1} mrem/pCi x 2.285 pCi/m^3 x 9.6 m^3 = 2.4 mrem
Bone: 9.2×10^{-2} mrem/pCi x 2.285 pCi/m^3 x 9.6 m^3 = 2.0 mrem
Total Body: 9.5×10^{-3} mrem/pCi x 2.285 pCi/m^3 x 9.6 m^3 = 0.2 mrem

These values are then compared with the allowable limit of 170 mrem per year for a person non-occupationally exposed.[1]

TABLE 1

FIFTY YEAR DOSE COMMITMENT FACTORS (Inhalation/Ingestion)[a]
(mrem/pCi assimilated)

Nuclide[b]	Inhalation			Ingestion	
	Bone	Lung	Total Body	Bone	Total Body
^{238}U	7.1×10^{-3}	4.8×10^{-1}	1.5×10^{-2}	2.8×10^{-4}	2.1×10^{-5}
^{234}U	7.9×10^{-3}	5.4×10^{-1}	1.6×10^{-2}	3.1×10^{-4}	2.4×10^{-5}
^{230}Th	3.1×10^{-1}	5.3×10^{-1}	3.8×10^{-2}	1.2×10^{-3}	9.2×10^{-5}
^{226}Ra	4.9×10^{-2}	5.6×10^{-2}	4.7×10^{-3}	4.3×10^{-2}	3.4×10^{-3}
^{210}Po	8.1×10^{-4}	4.5×10^{-2}	1.3×10^{-3}	5.2×10^{-4}	4.1×10^{-4}
^{210}Pb	4.2×10^{-2}	6.2×10^{-3}	3.5×10^{-3}	2.1×10^{-2}	1.7×10^{-3}
Total Chain	4.2×10^{-1}	1.7	7.9×10^{-2}	6.6×10^{-2}	5.6×10^{-3}
^{226}Ra Chain	9.2×10^{-2}	1.1×10^{-1}	9.5×10^{-3}	6.5×10^{-2}	5.5×10^{-3}
$^{238}U+^{234}U$	1.5×10^{-2}	1.0	3.1×10^{-2}	5.9×10^{-4}	4.5×10^{-5}
^{235}U	7.2×10^{-3}	4.8×10^{-1}	1.5×10^{-2}	2.8×10^{-4}	2.2×10^{-5}
^{231}Pa	9.6×10^{-1}	5.9×10^{-1}	1.4×10^{-1}	1.8×10^{-2}	2.1×10^{-3}
^{227}Ac	5.4×10^{-1}	1.0	1.0×10^{-1}	1.2×10^{-2}	1.3×10^{-3}
^{223}Ra	9.6×10^{-4}	4.6×10^{-2}	8.6×10^{-4}	1.2×10^{-3}	2.3×10^{-4}
^{227}Th	4.8×10^{-4}	6.9×10^{-2}	1.1×10^{-3}	2.7×10^{-5}	6.0×10^{-6}
Total Chain	1.5	2.2	2.6×10^{-1}	3.2×10^{-2}	3.6×10^{-3}
^{232}Th	3.3×10^{-1}	4.5×10^{-1}	3.8×10^{-2}	1.3×10^{-3}	9.6×10^{-5}
^{228}Th	4.4×10^{-2}	7.2×10^{-1}	1.9×10^{-2}	4.5×10^{-4}	3.8×10^{-5}
^{228}Ra	2.9×10^{-2}	4.8×10^{-3}	2.5×10^{-3}	2.1×10^{-2}	1.7×10^{-3}
^{224}Ra	3.0×10^{-4}	8.8×10^{-3}	1.8×10^{-4}	4.0×10^{-4}	7.5×10^{-5}
^{212}Pb	2.6×10^{-5}	1.8×10^{-3}	2.9×10^{-5}	1.8×10^{-5}	2.6×10^{-6}
Total Chain	4.0×10^{-1}	1.2	6.0×10^{-2}	2.3×10^{-2}	1.9×10^{-3}

[a] Data taken from Reference 2.

[b] Nuclides in the chain that contribute negligibly (e.g. $<10^{-6}$ mrem) have not been included.

APPENDIX 8
(cont'd.)

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1. U.S. Department of Energy 1981. "Requirements for Radiation Protection" DOE 5480.1, Chapter XI.

2. D. E. Dunning, Jr., et al. 1981. "Estimates of Internal Dose Equivalent to 22 Target Organs for Radionuclides Occurring in Routine Releases from Nuclear Fuel-Cycle Facilities, Volume III." ORNL/NUREG/TM-190/V3. Oak Ridge National Laboratory for U.S. Nuclear Regulatory Commission.

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