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**SURPLUS FACILITIES
MANAGEMENT PROGRAM**

**POST-REMEDIAL-ACTION SURVEY REPORT
FOR
SNAP-8 EXPERIMENTAL REACTOR FACILITY
BUILDING 010 SITE
SANTA SUSANA FIELD LABORATORIES
ROCKWELL INTERNATIONAL
VENTURA COUNTY, CALIFORNIA**

**November 1979
October and November 1981**

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

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PREFACE AND EXECUTIVE SUMMARY

Between 1959 and 1965, two "System for Nuclear Auxilliary Power" (SNAP) space-type compact experimental reactors were tested in Building 010 at the Santa Susana Field Laboratories (SSFL) of Rockwell International's Energy Systems Group (ESG) in Ventura County, California. The first was a 50-kW SNAP-2 reactor and the second was a 600-kW SNAP-8 reactor. In 1974, the SNAP-8 facility was declared excess to the government's program needs and a de-commissioning program was begun in 1976 that provided for the complete removal of all radioactive materials remaining from the operation of the SNAP reactors.

The Department of Energy (DOE), Office of Operational Safety (OOS), requested the Radiological Survey Group (RSG) from Argonne National Laboratory (ANL) to conduct a post-remedial-action assessment (certification) survey of the facility. Since the area had been converted to an asphalt-paved parking area prior to the first visit by the survey team in 1979, the primary mode of assessment was subsurface investigation. However, the surface was also surveyed for alpha and beta-gamma radiation with gas-flow proportional detectors; for low energy x and gamma radiation with thin (2-mm thick) NaI(Tl) crystal detectors; and for ambient external penetrating radiation with thick (2.5-cm thick) NaI(Tl) crystal detectors. Twenty-two boreholes were drilled through the asphalt pad. A NaI(Tl) scintillation detector was lowered into the boreholes to log the gamma-ray spectra at 2-ft intervals and split-spoon soil samples were removed from all of the holes in 1-ft increments. The gamma-ray spectral logging revealed ^{60}Co in the material adjacent to six of the holes. Gamma spectral and uranium fluorometric analysis of the soil indicated ^{60}Co in samples from eight of the holes, ^{152}Eu in samples from one hole, and uranium in samples from two holes. However, the measured concentrations of the radionuclides (up to 48 pCi/g ^{60}Co , up to 0.8 pCi/g ^{152}Eu , and up to 12.8 pCi/g uranium) were below guidelines for decontamination of the site (100 pCi/g gross detectable beta activity). The activity detected during the borehole logging appears to be from the activity in the soil. The origin of this activity was never determined but probably is the result of the activation of either the soil itself or the Building 010 construction material by the SNAP reactors. Originally it was suspected that a stainless steel rod or bolt (an object containing ^{60}Ni) or stellite material (an object containing ^{59}Co) had been left behind during the

D&D operations. Therefore, the second trip in 1981 was organized to drill additional boreholes. The results from these boreholes indicated the radiation was in the soil itself.

This survey was performed by the following Health Physics personnel of the Occupational Health and Safety Division, Argonne National Laboratory, Argonne, Illinois: W. H. Smith, R. L. Mundis (now at Los Alamos National Laboratory), C. Boggs Mayes (now at Exxon Nuclear Idaho, Inc.), A. L. Justus, J. D. Thereon, R. Rodriguez and D. W. Reilly.

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INTRODUCTION

The Argonne National Laboratory (ANL) Radiological Survey Group (RSG) conducted a series of post-remedial-action assessment (certification) radiological measurements and analyses at the former site of the System for Nuclear Auxilliary Power-8 (SNAP-8) Experimental Reactor Facility in the Santa Susana Mountains at Rockwell International's Santa Susana Field Laboratories (SSFL) in Ventura County, California, (see Fig. 1). The assessment was conducted at the request of the U.S. Department of Energy (DOE) in accordance with the programmatic post-remedial-action responsibilities of the Office of Operational Safety (OOS). Radiological measurements were conducted initially during November 1979 and again during October and November 1981.

The SNAP-8 facility, also known as Building 010 (see Fig. 2), was built for the testing of small (less than 2-ft diameter), space-type reactors. It was originally constructed in 1959 for testing the 50-kW SNAP-2 Experimental Reactor. That testing was completed in 1960, and the facility was modified and improved for a similar testing program that involved the higher power level 600-kW SNAP-8 Experimental Reactor. However, the fixed-concrete shielding limited the reactor power rating to less than 550-kW. Since most of the reactor operation was at or near full power, it can be concluded that the reactor components were exposed to a neutron flux from 5.4×10^6 kW of power generation over 688 days. The program involving the SNAP-8 Experimental Reactor was completed in 1965. The SNAP-8 facility and its components were owned by the U.S. Government under the responsibility of the Atomic Energy Commission and its predecessor organizations. The land is owned by Rockwell International and is under a long-term lease/option to the U.S. Government.

Building 010 was a rigid steel frame structure with corrugated metal siding and roofing (see Fig. 3). The foundation and floors were constructed of steel-reinforced concrete. Subsurface structures included three steel-reinforced

concrete vaults, two of which were steel-lined, that extended to a maximum depth of 14 ft below grade. The reactors were operated in a below grade, concrete shielded, containment vessel. The shielded containment vessel consisted of a 4-ft diameter by 14-ft deep, carbon steel pressure vessel embedded in concrete (see Fig. 4).

After completing the testing program, the SNAP facility was dismantled. At the time of the 1979 and 1981 surveys, the SNAP-8 site consisted of an asphalt-paved parking area (see Fig. 5). All reactor components, equipment, structures, and appurtenances thereto, including the radioactive gas hold-up tanks, had been removed and disposed of (see Fig. 6), with the exception of an unknown quantity of nonradioactive reinforced concrete, used as backfill and a previously abandoned sanitary leach field. The septic tank, a part of the sanitary leach field, had been removed when the facility was connected to the central sewer system.

The results of the post-remedial-action assessment of the site and the conclusions of the Argonne National Laboratory Radiological Survey Group (ANL-RSG) health physicists are presented in this report. Three other sites located at SSFL were also investigated by ANL-RSG (see Fig. 2). The post-remedial-action survey reports for those other sites are presented in companion reports: the Building 003 Hot Cave Facility in ANL-OHS/HP-83-109; the Kinetic Experiment Water Boiler Facility (KEWB) in ANL-OHS/HP-83-110; and the Sodium Reactor Experiment Facility (SRE) in ANL-OHS/HP-84-101.

SURVEY AND ANALYTICAL TECHNIQUES

General

Decontamination and decommissioning (D&D) of the SNAP-8 facility was completed by Rockwell International in 1978, converting the site to its present status, an asphalt-paved parking area. The D&D activity was performed while the facility was under U.S. Government ownership and subject to DOE requirements. It was reported by Rockwell International that their guidelines for reducing radioactive contamination included the "As Low As Practicable" (ALAP) concept and in all cases, decontamination to levels less than the following limits:

Surface Contamination

<u>Source</u>	<u>Total</u>	<u>Removable</u>
Beta-gamma emitters	0.1 mrad/h at 1 cm with 7 mg/cm ² absorber	100 dis/min-100 cm ²
Alpha emitters	100 dis/min-100 cm ²	20 dis/min-100 cm ²

Soil Concentrations

100 pCi/g gross detectable beta activity.

Gross detectable beta activity background readings at the site ranged from 20 to 30 pCi/g of soil.

The initial post-remedial-action radiological assessment conducted by the ANL-RSG during November 1979 included direct surface surveys, background measurements, subsurface investigation that included 1-ft continuous split-spoon soil sampling and borehole logging, and 1-ft peripheral environmental soil corings. Two background soil corings were collected at the Rockwell International site in Canoga Park, California.

Since the ANL-RSG was not involved in an initial radiological assessment or a concurrent radiological assessment (overview) during the D&D operations, it is not possible to comment in this report on the radiological status and disposition of the SNAP-8 facility structures and equipment.

Instrumentation

Four types of portable survey instruments were used to conduct the direct radiological surveys. Two sizes of gas-flow proportional detectors were used with Eberline PAC-4G-3 electronics to monitor for alpha and/or beta-gamma radiation. The larger floor monitor probe with a 325-cm² detection area (AC-22) was used to survey the paved parking area. The smaller probe with a 51-cm² detection area (AC-21) was used to survey any areas inaccessible with the floor monitor. NaI(Tl) crystal detectors, 5 cm (2 in) in diameter by 2 mm thick (Eberline PG-2 detector with Eberline PRM-5-3 electronics), were used to monitor for low-energy x and gamma radiation as close to the surfaces as possible (usually 1 cm from the surface). NaI(Tl) crystal detectors, 2.5 cm (1 in) in diameter by 2.5 cm (1 in) thick (Eberline PRM-7 μ R meter), calibrated with a

^{226}Ra standard source, were used to measure the ambient external penetrating radiation field in units of $\mu\text{R/h}$, approximately 1 m (3 ft) from the ground surface. All accessible areas were surveyed with the above instruments. These four types of instruments are described in more detail in Appendix 1, as are other instruments and techniques with site-specific applications for similar surveys. The appendices are written to be generic and the same appendices were used in the reports for all SSFL sites.

Although ^{239}Pu and ^{90}Sr - ^{90}Y standards were used to calibrate the gas-flow instruments, it should be noted that the numerous isotopes that could be encountered exhibit emission energies differing from those of the standards used in the calibration. Thus, when detecting known isotopes that emit beta energies differing from those of the standards (such as uranium), a conversion factor for the particular radionuclide was used to determine the appropriate yield. The methods used to determine the conversion factors are described in Appendix 2. All readings of disintegrations per minute per 100 cm^2 (dis/min-100 cm^2), as reported in Table 2, are equated to ^{90}Sr - ^{90}Y unless stated otherwise. (NOTE: no alpha contamination was found.) 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.

Ambient External Radiation

A Reuter-Stokes, RSS-111, pressurized ionization chamber was used at selected locations to monitor the ambient external penetrating radiation level, in microroentgens per hour ($\mu\text{R/h}$) at 1 m (3 ft) above the pad.

Soil Corings

Environmental soil corings were taken at four locations adjacent to the paved area and at two additional locations at the Canoga Park site in Los Angeles County. The samples were taken with a 10-cm (4-in) diameter, 15-cm (6-in) long right-circular-cylinder cutting tool commonly used as a golf-green hole-cutter. Each sample was taken in three 5-cm (2-in) segments and labeled A, B, and C. A final segment from 15 cm (6 in) to 30 cm (1 ft) was labeled D (see Figs. 5 and 7). The sample number consists of four parts: a site number (10), a sample type (S for soil), a sequential number (between 8 and 16), and a depth indicator as described above.

The segmented coring technique was used to determine if any contaminant migration had occurred; to reduce the dilution of lower-level soil with the upper-level segments with respect to the surface deposition of contaminants, or vice versa; and to reveal if any overburden or backfill had been added.

Soil Borings

Twenty-two boreholes were drilled through the asphalt pad with a commercial drilling rig. The holes were drilled to depths of up to 9.4 m (31 ft) and sampled in sequential 30-cm (1-ft) sections utilizing a split-spoon sampler (~ 4-cm [1.5 in] inside diameter). Each borehole was identified with a site identification number (10); a sample type (S for soil or SB for background soil); a sequential number (between 1 and 7 or between 17 and 19 for the samples collected in November 1979, and between 64 and 76 for the samples collected in 1981); and a depth indicator. For example, sample 10-S1-2.0 was the soil sample taken from the 1.0 to 2.0-ft level in hole number 1. The depths were reported to the nearest tenth of a foot. These samples were taken to determine if any contaminated material remained under the asphalt cap. An additional background borehole was drilled in the northwestern section of the paved area to determine the soil background concentrations in this area.

Gamma-ray spectra of all boreholes were logged in 60-cm (2-ft) increments. A 5 cm x 5 cm (2 in x 2 in) NaI(Tl) crystal detector was lowered into the borehole and the gamma-ray spectrum was collected for either 100 seconds or 300 seconds on a multichannel analyzer. Concentrations of the gamma-emitting radionuclide ^{60}Co were detected. This provides a qualitative assessment of the soil condition. However, it is not possible to make a quantitative assessment of soil concentrations from this data since there is no way to determine the distance of the radionuclides from the NaI(Tl) detector.

Three holes (10-S17 through 10-S19) were shallow because a pipe or concrete was hit by the drill rig. Two of these holes, 10-S17 and 10-S19 were drilled in the area that had contained the radioactive gas hold-up tank (see Fig. 5).

Soil Sample Analysis

All soil corings and borehole samples were initially processed at ANL (Fig. 8) and sent to either a commercial laboratory (LFE Environmental Analysis

Laboratory) or the Analytical Chemistry Section of the Chemical Engineering Division at ANL for gamma-spectral analysis. Program direction in 1979 restricted soil analysis to gamma-spectral. However, in 1981, radiochemical (uranium fluorometric) analysis was also permitted. Therefore, while all samples with a sequential number greater than 64 were analyzed for uranium concentration, uranium analyses were performed only on a few samples with sequential numbers less than 19.

Sample preparation consisted of weighing the samples in their entirety and then drying them for about 48 hours at 80°C. All samples were then reweighed, placed into mill jars (8.7 l), and milled until a sufficient amount of the soil sample could pass through a No. 30 (600-micron) standard stainless steel sieve. At no point were the rocks and heavy material ground or pulverized, since this material would act as a diluent and, hence, lower the concentration of deposited material. The rocks and dross vs. sieved material were segregated, bagged, and weighed separately. Soil weights are given in Table 1. Aliquots of the sieved material were then loaded into screwtop plastic containers. The amount varied according to the type of analysis to be performed--100 grams (if available) for gamma-spectral and 10 grams for radiochemical (fluorometric).

Every effort was made throughout the sample-preparation operations to eliminate cross-contamination. Soil samples suspected of containing elevated amounts of radioactivity were processed in equipment separate from the soil samples considered to contain background levels. All items of equipment were scrubbed and air dried before introduction of the next sample. A detailed generic description of sample preparation and analysis techniques is presented in Appendix 4.

SURVEY RESULTS

General

All surface surveys and background measurements were somewhat compromised by a variable low-level gamma background (shine) emanating from radioactive material stored in the Radioactive Materials Disposal Facility (RMDF) composed of Buildings 021 and 022 (see Fig. 2). The RMDF is approximately 700 to 800 ft northeast of the former Building 010 site. This, in effect, made the subsurface investigation the primary mode of assessment for the site material, i.e., back-fill and concrete.

Instrument Survey and Ambient External Radiation

Elevated readings were obtained on the asphalt pad during the instrument survey. The floor monitor readings ranged from 300 to 1600 dis/min-100 cm² beta-gamma above the instrument background. The μ R readings ranged from 15-30 μ R/h (natural background is 9-15 μ R/h). The PRM-5-3 (instrument background of 500 to 1500 cts/min) readings averaged 7×10^3 cts/min. Further investigation revealed that these elevated readings were probably due not to contaminated items buried underneath the pad but more likely to shine from radioactive materials stored on a hill to the east.

Integrated exposure rates were obtained at two locations (Locations 1 and 2 on Fig. 9) around the Building 10 site with a Reuter-Stokes RSS-111. The exposure rates as presented in Table 2 indicate a high ambient background.

Soil Corings

Four environmental soil corings were taken about the periphery of the Building 010 site (see Fig. 5). In addition, two environmental corings were taken in the Canoga Park site of Rockwell International's, Atomics International Division (see Fig. 7). These latter two corings were collected to determine the background soil concentrations in this area.

Gamma spectral and uranium fluorometric analyses of the peripheral corings indicated background concentrations for all samples except one. The exception was the soil core taken at location 10-S9-D (the 6-in to 12-in level [see Fig. 5]). Gamma-spectral results indicated a ⁶⁰Co concentration of 30 ± 27 femtocuries* per gram of soil (see Table 3). This level is well below the recommended criteria of 100 picocurie per gram. Background soil concentrations were established from the soil corings taken at the Canoga Park site. The results of all the coring analyses for ¹³⁷Cs, ²³²Th decay chain, and ²²⁶Ra decay chain are presented in Table 4. These concentrations are within the range of naturally occurring concentrations for this area.

*One femtocurie (1 fCi) = 1×10^{-15} curie; one picocurie (1 pCi) = 1×10^{-12} curie.

Soil Borings

Subsurface investigations conducted in 1979 included borehole sampling and borehole logging at eight locations, six of which were in the vicinity of where the former reactor vessel and fuel storage vaults had been located (see Fig. 10). The depth of the boreholes ranged from 3 ft to 31 ft.

Analyses of the borehole soil samples and the borehole logging results during the 1979 assessment indicated ^{60}Co contamination in boreholes 10-S1, 10-S4, 10-S6, and 10-S17, as well as ^{152}Eu contamination in borehole 10-S6 (see Fig. 10 and Table 3). The depth at which ^{60}Co was detected ranged from the 1-ft level to the 15-ft level, and the activity levels ranged from 0.08 to 7.3 pCi/g. The depth at which ^{152}Eu occurred was 10 ft and 15 ft, and the activity ranged up to 0.8 ± 0.1 pCi/g. Although these activity levels were well within the soil contamination criteria established for this site, the detection of the radionuclides ^{60}Co and ^{152}Eu raised some concern that a "hot" spot could be the cause of the contamination e.g., previously irradiated stainless steel rods or bolts (objects containing ^{60}Ni) or stellite material (objects containing ^{59}Co). ^{60}Co is an activation product produced by a ^{60}Ni η -p reaction or a ^{59}Co η - γ reaction and ^{152}Eu is an activation product produced by a ^{151}Eu η - γ reaction. Since it was reported that "clean imported" material had been used as backfill and since the source of the ^{60}Co and ^{152}Eu could not be explained, it was decided to drill additional bore holes in the area of concern in order to resolve the question of whether there might be "hot" items in the backfill causing the soil contamination.

Thirteen additional boreholes were drilled, sampled, and logged during the fall of 1981 (see Fig. 11). Results of logging and soil analyses of these additional boreholes revealed the presence of ^{60}Co in the soil, but no evidence of "hot" spots. The maximum level of ^{60}Co , as reported from soil sample analyses, indicated 48.0 pCi/g from borehole number 10-S71 at the 11-ft level. Boreholes that exhibited ^{60}Co anomalies included 10-S70, 10-S71, 10-S74, and 10-S75. All were in the general area previously found to contain ^{60}Co anomalies (see Fig. 12). Two samples also indicated elevated concentrations of uranium--6 pCi/g at the 8-ft level in hole 10-S64, and 8 pCi/g at the 6-ft level in hole 10-S66. As shown in Figure 12 and Table 4, uranium was the only contaminant in these samples. The mass spectrometric analyses for sample 10-S66-6 (see Table 5)

indicates depleted uranium with uranium-236. Table 4 indicates 1.05 pCi/g of the radium-226 decay chain. This level of radium activity is natural for this area. Using this value to determine the contribution from the natural uranium and assuming the remainder is 0.2% depleted uranium, atom percents similar to those given in Table 5 are calculated. Since there is no standard that directly pertains to soil contamination from alpha emitters, these results are compared to the proposed average concentration of 40 pCi/g in 100 m² for uranium-238 and uranium-234 (from LA-UR-79-1864 Rev.). It should be noted that Table 5 indicates the presence of uranium-236 which is not covered in the above proposed standard. However, examination of the appropriate tables² indicates that the values of the 50-year dose commitment factors are similar (within 10%) for the various uranium isotopes if the solubility and the target organ are kept constant.

All the bore holes located in Figure 12 were logged in 60-cm (2-ft) increments. The ⁴⁰K peak was clearly evident at 1460 keV. In six holes, 10-S1, 10-S4, 10-S6, 10-S67, 10-S70 and 10-S71, ⁶⁰Co peaks could be seen at 1173 keV and 1332 keV. In one hole, 10-S71, the gamma-ray intensity increased from the 6-ft level to the bottom of the hole (see Fig. 13). Drilling was stopped in this hole when a rebar ($\frac{1}{2}$ -in-diameter steel reinforcing bar in concrete) was encountered. The logging of this hole exhibited the maximum intensity from ⁶⁰Co gamma-rays, and the maximum ⁶⁰Co concentrations (up to 48 pCi/g) in soil were found in samples removed from this hole (see Table 3). In the other holes, the ⁶⁰Co gamma-ray intensity was observed between 2 ft and 12 ft. In each case the intensity increased to a maximum between 6 ft and 12 ft and then decreased again, as demonstrated most dramatically in borehole 10-S4 (see Fig. 14). In only one hole (10-S67) ⁶⁰Co gamma-rays were detected during the logging but no ⁶⁰Co was detected in the soil removed from the hole (see Table 3). As seen in Figure 12, ⁶⁰Co was measured in the soil removed from four surrounding holes.

In all cases the contamination appeared to be dispersed in the soil and not the result of a previously irradiated object such as a stainless steel rod, bolt, or stellite material. The exact origin of the contamination was never determined.

CONCLUSIONS

Based on the results of the radiological assessment, the ANL-RSG arrived at the following conclusions:

1. Soil contaminated with the radionuclides ^{60}Co and ^{152}Eu of undetermined origin was detected in the southwest quadrant of the Building 010 site. ^{60}Co was also detected in one environmental sample taken from an area northwest of the site and in a borehole sample taken from the area that previously held the radioactive gas hold-up tanks. Uranium was detected in soil from a hole in the center of the building site and in a second hole southwest of the building site. In all cases, the radionuclide levels encountered in the soil were well below the criteria set by DOE for this site.
2. The direct instrument readings at the surface of the site were probably the result of natural radiation (terrestrial and celestial), as well as shine from the material being stored at the nearby RMDF facility. There was no evidence that the contaminated soil under the asphalt pad contributed detectable levels to the total background readings.

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ENERGY SYSTEMS GROUP
SANTA SUSANA FIELD LABORATORIES

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

Figure 3. Building 010, the SNAP-8 Site.

Figure 4. Two Steel Reinforced Concrete Vaults.

Figure 5. Location of Soil Corings on Building 010 Site.

Figure 6. Building 010 Site During Excavation.

Figure 7. Location of Background Soil Corings at Canoga Park Site.

Figure 8. Soil Processing Procedure.

Figure 9. Location of Building 010 Site Ambient External Penetrating Radiation Measurements.

Figure 10. Location of Soil Borings on Building 010 Site Taken November 1979.

Figure 11. Location of Soil Borings on Building 010 Site Taken Fall 1981.

Figure 12. Contamination in Boreholes and Borehole Samples.

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Figure 13. Logging of Borehole 10-S71.

Figure 14. Logging of Borehole 10-S4.

TABLE 1
SOIL-SAMPLE WEIGHTS
(grams)

Sample No.	Wet Weight	Dry Weight	Sieved Weight	Rocks and Dross
10-S1-1.0	529	502	350	150
10-S1-2.0	517	501	403	97
10-S1-3.0	525	506	411	93
10-S1-4.0	659	602	494	102
10-S1-5.0	508	442	340	98
10-S1-6.0	468	406	295	102
10-S1-7.0	439	400	338	60
10-S1-8.0	443	405	282	121
10-S1-9.0	386	348	227	116
10-S1-10.0	340	302	216	85
10-S1-11.0	539	495	352	143
10-S1-12.0	560	511	354	157
10-S1-12.8	500	457	283	171
10-S1-13.4	308	286	285	1
10-S1-14.3	292	274	270	4
10-S1-15.3	287	272	271	1
10-S1-17.5	430	395	212	182
10-S1-18.3	257	236	144	78
10-S1-19.7	443	416	172	229
10-S2-1.0	584	541	416	124
10-S2-2.0	516	500	386	112
10-S2-3.0	531	518	394	124
10-S2-7.0	381	339	304	32
10-S2-8.0	448	418	329	87
10-S2-8.7	424	400	253	147
10-S3-1.0	587	547	429	116
10-S3-2.0	621	571	420	149
10-S3-2.4	317	295	270	19
10-S3-3.3	195	183	178	3
10-S3-4.3	276	253	242	7
10-S3-5.3	256	241	238	1
10-S3-6.3	306	284	278	3
10-S3-7.3	261	241	233	4
10-S3-8.4	335	308	253	52
10-S3-9.5	344	318	302	13
10-S3-10.4	319	294	267	25
10-S3-11.2	192	181	179	1
10-S3-12.2	197	188	185	1
10-S3-16.2	306	290	257	29
10-S3-17.2	223	208	174	31
10-S3-19.2	177	171	164	4
10-S3-20.1	409	391	339	45

TABLE 1
(cont'd.)

Sample No.	Wet Weight	Dry Weight	Sieved Weight	Rocks and Dross
10-S4-1.0	579	548	361	186
10-S4-2.0	606	585	462	123
10-S4-3.0	514	479	404	73
10-S4-4.5	372	355	319	34
10-S4-5.8	489	462	453	8
10-S4-6.8	182	174	172	1
10-S4-7.0	140	133	129	1
10-S4-8.0	572	505	318	186
10-S4-9.0	526	477	269	204
10-S4-10.8	546	496	329	163
10-S4-11.3	291	274	239	27
10-S4-12.5	400	366	216	144
10-S4-13.4	302	280	203	71
10-S4-14.3	289	263	202	56
10-S4-15.3	318	295	258	35
10-S4-16.3	240	223	137	80
10-S4-17.2	288	269	205	61
10-S4-18.2	184	178	163	13
10-S4-19.2	249	236	221	12
10-S4-20.2	158	146	145	1
10-S4-21.3	502	474	401	71
10-S4-22.2	324	311	239	71
10-S4-23.1	138	133	118	11
10-S4-24.1	232	223	202	19
10-S4-25.2	122	118	102	12
10-S4-26.2	170	164	132	29
10-S4-27.1	134	130	114	13
10-S4-28.2	171	165	149	12
10-S4-29.1	98	96	81	11
10-S4-30.1	190	182	170	9
10-S4-31.1	181	173	156	15
10-S5-1.0	453	419	345	74
10-S5-2.0	388	363	290	70
10-S5-3.0	457	421	302	113
10-S5-4.9	574	526	370	153
10-S5-5.3	284	261	242	16
10-S5-7.3	125	118	116	1
10-S5-8.2	518	486	478	7
10-S5-9.2	91	87	86	1
10-S5-13.1	181	167	159	7
10-S5-16.2	652	615	405	209

TABLE 1
(cont'd.)

Sample No.	Wet Weight	Dry Weight	Sieved Weight	Rocks and Dross
10-S6-1.0	955	887	617	269
10-S6-2.5	708	675	391	282
10-S6-3.5	833	797	606	191
10-S6-4.0	987	896	703	188
10-S6-5.0	844	737	464	270
10-S6-6.0	1022	901	656	244
10-S6-7.5	781	688	505	177
10-S6-8.5	833	743	519	220
10-S6-9.5	792	741	474	264
10-S6-10.0	904	798	324	472
10-S6-11.0	655	578	292	286
10-S6-13.0	586	540	359	181
10-S6-14.2	209	198	197	1
10-S6-15.3	230	213	212	1
10-S6-16.3	233	219	218	1
10-SB7-1.0	359	331	227	102
10-SB7-2.0	497	456	426	24
10-SB7-3.8	544	504	431	72
10-SB7-4.3	149	143	132	6
10-SB7-5.1	116	111	106	1
10-SB7-6.2	174	161	105	49
10-SB7-7.2	134	127	104	19
10-SB7-8.2	98	93	90	1
10-SB7-9.2	137	131	120	6
10-SB7-10.2	150	145	131	9
10-SB7-11.2	140	134	120	11
10-SB7-12.2	87	85	82	0
10-SB7-13.2	193	183	178	2
10-SB7-14.2	65	62	57	3
10-SB7-15.1	145	138	115	19
10-SB7-16.3	158	147	73	68
10-S17-1.0	441	410	234	169
10-S17-2.0	477	419	348	63
10-S17-3.0	561	508	284	215
10-S18-1.0	365	339	256	82
10-S18-2.6	369	342	240	100
10-S19-2.0	499	452	377	73
10-S19-3.2	193	172	149	19

TABLE 1
(cont'd.)

Sample No.	Wet Weight	Dry Weight	Sieved Weight	Rocks and Dross
10-S8-A	851	819	642	170
10-S8-B	1043	985	766	211
10-S8-C	1086	1027	855	167
10-S8-D	2950	2769	1985	775
10-S9-A	1070	1053	721	327
10-S9-B	1531	1498	1029	450
10-S9-C	1213	1184	922	254
10-S9-D	684	671	595	73
10-S10-A	1206	1154	715	433
10-S10-B	999	941	667	268
10-S10-C	1334	1220	852	360
10-S10-D	2465	2244	1431	804
10-S11-A	610	574	443	125
10-S11-B	887	846	582	258
10-S11-C	1031	990	719	266
10-S11-D	2847	2713	1672	1001
10-SB15-A	362	335	294	34
10-SB15-B	487	463	386	71
10-SB15-C	1124	1054	878	175
10-SB15-D	2500	2299	1754	534
10-SB16-A	522	496	438	51
10-SB16-B	374	363	329	28
10-SB16-C	749	726	639	79
10-SB16-D	2730	2599	2070	517
10-S64-1	265	249	229	10
10-S64-2	459	451	439	10
10-S64-3	414	406	395	9
10-S64-4	355	348	339	6
10-S64-5	562	488	424	57
10-S64-6	369	320	292	19
10-S64-7	350	306	277	24
10-S64-8	48	43	22	1
10-S64-9	184	166	143	16
10-S64-10	418	385	229	153
10-S64-12	373	338	254	80

TABLE 1
(cont'd.)

Sample No.	Wet Weight	Dry Weight	Sieved Weight	Rocks and Dross
10-S65-1	504	486	456	24
10-S65-2	540	490	428	56
10-S65-3	415	376	329	40
10-S65-4	477	453	439	11
10-S65-5	453	429	424	2
10-S65-6	374	353	319	28
10-S65-7	437	414	408	2
10-S65-8	527	488	444	37
10-S65-9	215	204	194	2
10-S65-10	456	427	419	3
10-S65-11	303	272	218	50
10-S65-12	416	385	358	22
10-S65-13	218	205	197	2
10-S65-14	494	457	349	107
10-S66-1	566	538	478	56
10-S66-4	201	188	85	76
10-S66-5	136	126	88	9
10-S66-6	78	71	55	6
10-S66-7	259	246	239	4
10-S66-8	329	314	307	3
10-S66-9	141	135	120	5
10-S66-10	367	344	338	1
10-S66-11	223	210	201	4
10-S66-12	323	305	299	2
10-S66-13	135	122	74	28
10-S66-14	131	126	119	1
10-S67-1	619	595	426	161
10-S67-2	537	509	467	35
10-S67-3	509	496	467	26
10-S67-4	541	523	471	47
10-S67-5	216	205	195	5
10-S67-6	237	210	191	15
10-S67-7	278	245	224	16
10-S67-8	439	405	344	57
10-S67-9	86	79	72	1
10-S67-10	162	150	109	37
10-S67-11	273	239	209	26
10-S67-12	226	197	164	29
10-S67-13	599	507	291	208
10-S67-14	524	464	321	137

TABLE 1
(cont'd.)

Sample No.	Wet Weight	Dry Weight	Sieved Weight	Rocks and Dross
10-S68-1	577	553	494	52
10-S68-2	479	468	440	19
10-S68-3	690	654	568	79
10-S68-4	580	539	454	77
10-S68-5	439	420	402	14
10-S68-6	455	436	408	24
10-S68-7	309	296	283	11
10-S68-8	434	415	371	39
10-S68-9	588	560	524	32
10-S68-10	560	515	422	83
10-S68-11	563	506	398	101
10-S68-12	489	438	337	97
10-S68-13	255	227	178	34
10-S68-14	390	349	227	122
10-S69-1	442	416	230	186
10-S69-2	548	524	405	115
10-S69-3	351	323	300	19
10-S69-4	650	604	461	132
10-S69-5	505	467	362	103
10-S69-6	534	495	415	77
10-S69-7	313	295	283	1
10-S69-8	625	583	457	120
10-S69-9	270	252	228	21
10-S69-11	134	126	108	15
10-S69-12	302	281	234	46
10-S70-1	507	487	368	118
10-S70-2	494	484	421	61
10-S70-3	451	429	374	54
10-S70-4	196	182	147	31
10-S70-5	247	225	209	15
10-S70-6	521	485	408	76
10-S70-7	394	374	323	48
10-S70-8	553	528	463	55
10-S70-9	391	371	353	16
10-S70-10	217	204	202	1
10-S70-11	155	145	142	1
10-S70-12	391	349	316	29
10-S70-13	551	489	394	95
10-S70-14	351	313	204	103
10-S70-15	424	381	308	70
10-S70-16	522	471	282	183

TABLE 1
(cont'd.)

Sample No.	Wet Weight	Dry Weight	Sieved Weight	Rocks and Dross
10-S71-1	452	441	333	108
10-S71-2	544	534	449	83
10-S71-3	467	457	429	28
10-S71-4	620	594	544	47
10-S71-5	414	401	239	157
10-S71-6	491	462	271	190
10-S71-7	526	482	329	151
10-S71-8	239	219	171	45
10-S71-9	132	121	99	19
10-S71-10	217	198	137	59
10-S71-11	330	300	249	50
10-S71-12	404	366	303	61
10-S72-1	449	422	344	75
10-S72-2	392	360	229	127
10-S72-3	546	479	300	176
10-S72-4	459	413	395	16
10-S72-5	380	348	298	46
10-S72-6	220	204	200	1
10-S72-7	293	272	266	1
10-S72-8	416	362	340	14
10-S72-9	477	425	245	175
10-S72-10	604	532	222	304
10-S73-1	544	523	220	301
10-S73-2	545	479	205	272
10-S73-3	633	555	175	376
10-S73-4	540	484	244	237
10-S73-5	615	535	179	352
10-S73-6	566	510	184	321
10-S73-7	383	349	170	175
10-S73-8	223	200	114	80
10-S73-9	599	537	320	212
10-S73-10	157	140	125	11
10-S74-1	339	323	227	92
10-S74-2	547	532	403	127
10-S74-3	602	564	367	194
10-S74-4	469	439	365	73
10-S74-6	111	105	70	32
10-S74-7	119	111	104	7

TABLE 1
(cont'd.)

Sample No.	Wet Weight	Dry Weight	Sieved Weight	Rocks and Dross
7-S75-1	437	422	190	231
7-S75-2	582	568	377	186
7-S75-3	582	566	375	186
7-S75-4	206	201	107	90
7-S75-5	266	249	121	126
7-S75-6	277	260	135	123
7-S75-7	537	489	217	270
1-S76-1	719	639	288	347
1-S76-2	563	527	413	111
1-S76-3	575	513	215	294
1-S76-4	590	518	200	313
1-S76-5	635	575	204	365
1-S76-6	464	425	218	201
1-S76-7	248	235	193	36
1-S76-8	407	379	203	172
1-S76-9	46	42	20	21
1-S76-12	110	103	85	7
1-S76-13	100	94	59	35
1-S76-15	139	129	122	1
1-S76-16	213	198	193	4

TABLE 2

DATA SHEET OF INSTRUMENT SURVEYS

Area	PAC-4G-3 Direct Readings ^a (dis/min-100 cm ²)		PRM-7 (μR/h) 1 meter	Ambient Background Radiation (μR/h at 1 m)	PRM-5-3 w/PG-2 (cts/min)	Comments
	Beta	Alpha				
Asphalt Pad	300- 1200	BKGD ^b	15-30	NS ^c	7000	General
	300- 1200	BKGD	15-30	17.2	7000	Location 1
	300- 1200	BKGD	15-30	15.8	7000	Location 2
	NRR ^d	NRR	160	NS	NRR	Borehole 10-S1

^aThe beta mode direct readings and alpha mode direct readings were taken with PAC instruments (see Appendix 1). The beta mode detects both electromagnetic and particle radiation. If an area indicated an instrument reading higher than background, a beta mode reading was obtained. The instrument was then switched to the alpha mode, and a reading of the alpha contamination was obtained. In the alpha mode, the instrument only responds to particles with high-specific ionization, such as alpha particles. The beta mode reading was compensated for any alpha contribution by subtracting the alpha mode readings.

^bBKGD = Background. The following are the instrument background readings:

Instrument

PAC-4G-3 Beta Mode: 200-700 cts/min-100 cm² Alpha Mode: 0-5 cts/min-100 cm²
 PRM-5-3 500-1500 cts/min
 μR/h meter reading at 1 meter above the pad 15-30 μR/h.

^cNS = Not Selected; indicating the area was not selected for Reuter Stokes RSS-111 ambient background radiation sample. Locations of the ambient background radiation samples were chosen on a selected basis.

^dNRR = No Reading Recorded.

TABLE 3

INDUCED RADIOACTIVITY CONCENTRATION OF SOIL SAMPLES

Sample No.	(pCi/g $\pm\sigma^a$)		Sample No.	(pCi/g $\pm\sigma^a$)	
	$^{60}\text{Co}^c$	$^{152}\text{Eu}^c$		$^{60}\text{Co}^c$	$^{152}\text{Eu}^c$
SOIL BORINGS					
10-S1-7	0.44 $\pm$ 0.03	BDL ^b	10-S70-3	0.12 $\pm$ 0.02	BDL
10-S1-8	0.12 $\pm$ 0.02	BDL	10-S70-4	0.89 $\pm$ 0.09	BDL
10-S1-9	0.28 $\pm$ 0.03	BDL	10-S70-5	1.36 $\pm$ 0.14	BDL
10-S1-10	2.82 $\pm$ 0.20	BDL	10-S70-11	6.27 $\pm$ 0.63	BDL
10-S1-12.8	0.08 $\pm$ 0.03	BDL			
			10-S71-5	1.65 $\pm$ 0.16	BDL
10-S4-2	0.50 $\pm$ 0.04	BDL	10-S71-6	0.51 $\pm$ 0.05	BDL
10-S4-3	3.50 $\pm$ 0.25	BDL	10-S71-7	3.31 $\pm$ 0.33	BDL
10-S4-5.8	0.29 $\pm$ 0.03	BDL	10-S71-8	5.95 $\pm$ 0.60	BDL
			10-S71-9	4.47 $\pm$ 0.45	BDL
10-S6-5	0.08 $\pm$ 0.04	BDL	10-S71-10	11.10 $\pm$ 1.10	BDL
10-S6-8.5	0.08 $\pm$ 0.03	BDL	10-S71-11	48.00 $\pm$ 5.00	BDL
10-S6-10	7.32 $\pm$ 0.07	0.65 $\pm$ 0.04	10-S71-12	11.00 $\pm$ 1.00	BDL
10-S6-11	6.24 $\pm$ 0.13	BDL			
10-S6-13	0.56 $\pm$ 0.07	BDL	10-S74-4	0.46 $\pm$ 0.05	BDL
10-S6-15.3	1.99 $\pm$ 0.09	0.80 $\pm$ 0.10	10-S74-6	0.83 $\pm$ 0.08	BDL
			10-S74-7	0.46 $\pm$ 0.05	BDL
10-S17-1	0.07 $\pm$ 0.04	BDL	10-S75-7	5.13 $\pm$ 0.51	BDL
SOIL CORINGS					
10-S9-D	0.03 $\pm$ 0.03	BDL			

^aOne standard deviation due to counting statistics.

^bBDL = Below Detectable Limits.

^cSince the half-life of ^{60}Co is 5.27 years and ^{152}Eu is 13.6 years, the concentrations in this table are those measured on the following dates:

for hole 1: April 1980

for hole 4: May 1980

for holes 6, 9, and 17: June 1980

for holes 70, 71, 74, and 75: September 1982

TABLE 4

GAMMA-SPECTRAL AND URANIUM-FLUOROMETRIC
ANALYSES OF SOIL SAMPLES

Sample No.	Gamma Spectra, pCi/g $\pm\sigma$ ^a			Uranium Fluorometric ^b	
	¹³⁷ Cs	²³² Th Decay Chain	²²⁶ Ra Decay Chain	$\mu\text{g/g}\pm 10\%$	pCi/g $\pm 10\%$
SOIL CORINGS					
10-S8-A	BDL ^c	1.24 $\pm$ 0.16	0.79 $\pm$ 0.07		
10-S8-B	BDL	1.57 $\pm$ 0.18	0.78 $\pm$ 0.08		
10-S8-C	BDL	1.43 $\pm$ 0.19	0.81 $\pm$ 0.09		
10-S8-D	BDL	1.36 $\pm$ 0.15	0.82 $\pm$ 0.07	2.1 $\pm$ 0.2	1.4 $\pm$ 0.1
10-S9-A	0.27 $\pm$ 0.05	1.07 $\pm$ 0.21	0.52 $\pm$ 0.09		
10-S9-B	0.23 $\pm$ 0.04	1.44 $\pm$ 0.15	0.74 $\pm$ 0.07		
10-S9-C	0.08 $\pm$ 0.03	1.62 $\pm$ 0.13	0.77 $\pm$ 0.06		
10-S9-D	BDL	1.35 $\pm$ 0.13	0.67 $\pm$ 0.06		
10-S10-A	0.29 $\pm$ 0.03	1.53 $\pm$ 0.12	0.72 $\pm$ 0.06		
10-S10-B	0.14 $\pm$ 0.03	1.36 $\pm$ 0.13	0.71 $\pm$ 0.05		
10-S10-C	BDL	1.34 $\pm$ 0.13	0.82 $\pm$ 0.06		
10-S10-D	BDL	1.22 $\pm$ 0.27	1.21 $\pm$ 0.09		
10-S11-A	1.04 $\pm$ 0.06	0.87 $\pm$ 0.22	0.85 $\pm$ 0.09		
10-S11-B	0.57 $\pm$ 0.05	1.28 $\pm$ 0.16	0.70 $\pm$ 0.08		
10-S11-C	0.36 $\pm$ 0.04	1.28 $\pm$ 0.20	1.04 $\pm$ 0.08		
10-S11-D	0.07 $\pm$ 0.03	1.41 $\pm$ 0.13	0.92 $\pm$ 0.07	3.0 $\pm$ 0.3	2.0 $\pm$ 0.2
BACKGROUND SOIL CORING					
10-SB15-A	0.33 $\pm$ 0.04	0.64 $\pm$ 0.15	1.00 $\pm$ 0.08		
10-SB15-B	0.40 $\pm$ 0.04	1.00 $\pm$ 0.12	1.12 $\pm$ 0.06		
10-SB15-C	0.24 $\pm$ 0.05	0.92 $\pm$ 0.16	1.07 $\pm$ 0.08		
10-SB15-D	0.16 $\pm$ 0.04	0.83 $\pm$ 0.16	1.08 $\pm$ 0.08		
10-SB16-A	0.30 $\pm$ 0.07	0.57 $\pm$ 0.42	1.15 $\pm$ 0.09		
10-SB16-B	0.20 $\pm$ 0.04	1.45 $\pm$ 0.22	1.13 $\pm$ 0.10		
10-SB16-C	0.27 $\pm$ 0.04	0.77 $\pm$ 0.14	1.05 $\pm$ 0.07		
10-SB16-D	0.20 $\pm$ 0.05	1.02 $\pm$ 0.19	0.96 $\pm$ 0.10		

TABLE 4
(cont'd.)

Sample No.	Gamma Spectra, pCi/g±σ ^a			Uranium Fluorometric	
	137Cs	232Th	226Ra	µg/g±10%	pCi/g±10%
		Decay Chain	Decay Chain		
SOIL BORINGS					
10-S1-1	<0.03	1.05±0.07	0.72±0.05		
10-S1-2	<0.03	1.23±0.09	0.68±0.05		
10-S1-3	<0.03	1.06±0.07	0.38±0.03		
10-S1-4	<0.03	1.35±0.09	0.74±0.05		
10-S1-5	<0.03	1.50±0.11	1.12±0.08		
10-S1-6	<0.03	1.44±0.10	0.99±0.07		
10-S1-7	0.03±0.02	1.30±0.09	1.06±0.07		
10-S1-8	<0.03	1.29±0.09	1.00±0.07		
10-S1-9	<0.03	1.40±0.10	1.08±0.08		
10-S1-10	0.42±0.03	1.12±0.08	1.19±0.08		
10-S1-11	<0.03	1.42±0.10	1.01±0.07		
10-S1-12	<0.03	1.33±0.09	0.97±0.07		
10-S1-12.8	<0.03	1.53±0.11	1.82±0.13		
10-S1-13.4	<0.03	0.85±0.06	0.47±0.03		
10-S1-14.3	<0.03	0.69±0.05	0.53±0.04		
10-S1-15.3	<0.03	0.82±0.06	0.61±0.04		
10-S1-17.5	<0.03	1.10±0.08	3.46±0.24		
10-S1-18.3	<0.03	1.57±0.11	2.85±0.20		
10-S1-19.7	<0.03	1.73±0.12	2.04±0.17		
10-S2-1	<0.03	1.30±0.09	0.92±0.06		
10-S2-2	<0.03	1.12±0.08	0.50±0.04		
10-S2-3	<0.03	0.77±0.05	0.41±0.03		
10-S2-7	<0.03	1.47±0.10	0.86±0.06		
10-S2-8	<0.03	1.21±0.08	0.64±0.04		
10-S2-8.7	<0.03	1.48±0.10	0.78±0.05		
10-S3-1	<0.03	1.37±0.10	1.02±0.07		
10-S3-2	<0.03	1.20±0.08	0.94±0.07		
10-S3-2.4	<0.03	1.02±0.07	0.93±0.07		
10-S3-3.3	<0.03	1.18±0.08	0.92±0.06		
10-S3-4.3	<0.03	0.78±0.05	0.58±0.04		
10-S3-5.3	<0.03	0.76±0.05	0.56±0.04		
10-S3-6.3	<0.03	1.14±0.08	0.79±0.06		
10-S3-7.3	<0.03	0.94±0.07	1.16±0.08		
10-S3-8.4	<0.03	1.05±0.07	0.57±0.04		
10-S3-9.5	<0.03	1.10±0.08	0.96±0.05		
10-S3-10.4	<0.03	1.28±0.09	0.67±0.05		
10-S3-11.2	<0.03	1.20±0.08	0.98±0.07		
10-S3-12.2	0.04±0.02	0.75±0.05	0.74±0.05		
10-S3-16.2	<0.03	1.03±0.07	1.05±0.07		
10-S3-17.2	<0.03	1.63±0.11	1.25±0.09		
10-S3-19.2	<0.03	1.33±0.09	1.21±0.08		
10-S3-20.1	<0.03	1.44±0.10	1.32±0.09		

TABLE 4
(cont'd.)

Sample No.	Gamma Spectra, pCi/g $\pm\sigma^a$			Uranium Fluorometric	
	^{137}Cs	^{232}Th Decay Chain	^{226}Ra Decay Chain	$\mu\text{g/g}\pm 10\%$	pCi/g $\pm 10\%$
SOIL BORINGS - (cont'd.)					
10-S4-1	<0.03	1.22 $\pm$ 0.08	0.95 $\pm$ 0.07		
10-S4-2	<0.03	0.96 $\pm$ 0.07	0.68 $\pm$ 0.05		
10-S4-3	<0.03	1.40 $\pm$ 0.10	0.80 $\pm$ 0.06		
10-S4-4.5	<0.03	0.94 $\pm$ 0.07	0.43 $\pm$ 0.03		
10-S4-5.8	<0.03	0.86 $\pm$ 0.06	0.52 $\pm$ 0.04		
10-S4-6.8	<0.03	1.16 $\pm$ 0.08	0.81 $\pm$ 0.06		
10-S4-7	<0.03	1.31 $\pm$ 0.09	0.84 $\pm$ 0.06		
10-S4-8	<0.03	1.46 $\pm$ 0.10	1.88 $\pm$ 0.13		
10-S4-9	<0.03	1.41 $\pm$ 0.10	1.63 $\pm$ 0.11		
10-S4-10.8	<0.03	1.67 $\pm$ 0.12	1.84 $\pm$ 0.13		
10-S4-11.3	<0.03	1.15 $\pm$ 0.08	1.20 $\pm$ 0.08		
10-S4-12.5	<0.03	1.84 $\pm$ 0.13	1.94 $\pm$ 0.14		
10-S4-13.4	<0.03	2.04 $\pm$ 0.14	2.08 $\pm$ 0.15		
10-S4-14.3	<0.03	1.44 $\pm$ 0.10	1.44 $\pm$ 0.10		
10-S4-15.3	0.07 $\pm$ 0.03	1.56 $\pm$ 0.11	1.08 $\pm$ 0.08		
10-S4-16.3	<0.03	1.84 $\pm$ 0.13	2.78 $\pm$ 0.19		
10-S4-17.2	<0.03	1.58 $\pm$ 0.11	1.47 $\pm$ 0.10		
10-S4-18.2	<0.03	1.30 $\pm$ 0.09	1.02 $\pm$ 0.07		
10-S4-19.2	<0.03	0.78 $\pm$ 0.05	0.87 $\pm$ 0.06		
10-S4-20.2	<0.03	1.13 $\pm$ 0.08	0.69 $\pm$ 0.05		
10-S4-21.3	<0.03	1.77 $\pm$ 0.12	1.44 $\pm$ 0.10		
10-S4-22.2	<0.03	1.29 $\pm$ 0.09	1.32 $\pm$ 0.09		
10-S4-23.1	<0.03	1.13 $\pm$ 0.08	1.07 $\pm$ 0.07		
10-S4-24.1	<0.03	1.16 $\pm$ 0.08	0.89 $\pm$ 0.06		
10-S4-25.2	<0.03	1.41 $\pm$ 0.10	0.80 $\pm$ 0.06		
10-S4-26.2	<0.03	1.49 $\pm$ 0.10	1.11 $\pm$ 0.08		
10-S4-27.1	<0.03	1.50 $\pm$ 0.10	0.82 $\pm$ 0.06		
10-S4-28.2	<0.03	0.95 $\pm$ 0.07	1.45 $\pm$ 0.10		
10-S4-29.1	<0.03	1.26 $\pm$ 0.09	0.91 $\pm$ 0.06		
10-S4-30.1	<0.03	1.29 $\pm$ 0.09	0.95 $\pm$ 0.07		
10-S4-31.1	<0.03	2.27 $\pm$ 0.16	1.58 $\pm$ 0.11		
10-S5-1	<0.03	1.45 $\pm$ 0.10	0.91 $\pm$ 0.06		
10-S5-2	<0.03	1.30 $\pm$ 0.09	1.03 $\pm$ 0.07		
10-S5-3	<0.03	0.97 $\pm$ 0.07	0.84 $\pm$ 0.06		
10-S5-4.9	<0.03	1.44 $\pm$ 0.10	1.00 $\pm$ 0.07		
10-S5-5.3	<0.03	1.18 $\pm$ 0.08	1.38 $\pm$ 0.10		
10-S5-7.3	<0.03	1.22 $\pm$ 0.09	0.54 $\pm$ 0.04		
10-S5-8.2	<0.03	1.41 $\pm$ 0.10	0.83 $\pm$ 0.06		
10-S5-9.2	<0.03	1.26 $\pm$ 0.09	0.95 $\pm$ 0.07		
10-S5-13.1	<0.03	1.91 $\pm$ 0.13	0.98 $\pm$ 0.07		
10-S5-16.2	<0.03	0.05 $\pm$ 0.03	0.76 $\pm$ 0.05		

TABLE 4
(cont'd.)

Gamma Spectra, pCi/g±σ ^a					
Sample No.	¹³⁷ Cs	²³² Th	²²⁶ Ra	Uranium Fluorometric ^b	
		Decay Chain	Decay Chain	μg/g±10%	pCi/g±10%
SOIL BORINGS - (cont'd.)					
10-S6-1.0	BDL	1.22±0.14	0.76±0.08		
10-S6-2.5	0.05±0.04	1.46±0.21	0.73±0.09		
10-S6-3.5	BDL	0.95±0.17	0.42±0.07		
10-S6-4.0	BDL	1.53±0.12	0.88±0.05		
10-S6-5.0	BDL	1.62±0.12	0.70±0.05		
10-S6-6.0	BDL	1.81±0.22	0.91±0.09		
10-S6-7.5	BDL	1.19±0.15	0.88±0.06		
10-S6-8.5	BDL	1.38±0.12	0.68±0.05		
10-S6-9.5	BDL	1.45±0.09	0.58±0.04		
10-S6-10.0	0.37±0.03	1.39±0.15	0.74±0.05	1.9 ±0.2	1.3 ±0.1
10-S6-11.0	0.97±0.06	1.44±0.23	0.80±0.08	2.1 ±0.2	1.4 ±0.1
10-S6-13.0	0.17±0.03	1.47±0.23	0.65±0.10		
10-S6-14.2	BDL	0.75±0.12	0.43±0.05		
10-S6-15.3	BDL	0.94±0.25	0.61±0.12		
10-S6-16.3	BDL	1.10±0.22	0.35±0.08		
10-S17-1.0	BDL	1.36±0.18	0.58±0.08		
10-S17-2.0	BDL	0.94±0.13	0.70±0.07		
10-S17-3.0	0.05±0.04	1.24±0.17	0.70±0.08		
10-S18-1.0	<0.03	1.47±0.10	1.06±0.07		
10-S18-2.6	<0.03	1.41±0.10	1.00±0.07		
10-S19-2.0	BDL	1.25±0.13	0.55±0.05		
10-S19-3.2	BDL	1.03±0.32	1.13±0.11		
10-S64-1	<0.03	1.31±0.13	1.35±0.14	3.7 ±0.4	2.5 ±0.3
10-S64-2	<0.03	1.06±0.11	0.71±0.36	1.5 ±0.2	1.0 ±0.1
10-S64-3	<0.03	0.88±0.09	0.46±0.07	1.4 ±0.1	1.0 ±0.1
10-S64-4	<0.03	1.08±0.11	0.67±0.07	1.3 ±0.1	0.9 ±0.1
10-S64-5	<0.03	1.19±0.12	0.99±0.10	2.1 ±0.2	1.4 ±0.1
10-S64-6	<0.03	1.88±0.19	1.25±0.12	2.5 ±0.2	1.7 ±0.2
10-S64-7	<0.03	1.44±0.14	1.75±0.18	2.7 ±0.3	1.8 ±0.2
10-S64-8	BDL	1.25±0.12	1.53±0.15	14.4 ±1.4	6.0 ±0.6 ^d
10-S64-9	<0.03	1.33±0.13	1.16±0.12	2.6 ±0.3	1.8 ±0.2
10-S64-10	<0.03	1.65±0.16	1.31±0.13	2.1 ±0.2	1.4 ±0.1
10-S64-12	<0.03	1.76±0.18	1.05±0.10	1.9 ±0.2	1.3 ±0.1

TABLE 4
(cont'd.)

Sample No.	Gamma Spectra, pCi/g $\pm\sigma$ ^a				Uranium Fluorometric $\mu\text{g/g}\pm\sigma$	pCi/g $\pm\sigma$ ^b
	¹³⁷ Cs	²³² Th	²²⁶ Ra	Decay Chain		
		Decay Chain	Decay Chain			
SOIL BORINGS - (cont'd.)						
10-S65-1	<0.03	0.90 $\pm$ 0.09	0.65 $\pm$ 0.06		2.0 $\pm$ 0.2	1.4 $\pm$ 0.1
10-S65-2	<0.03	1.66 $\pm$ 0.17	0.94 $\pm$ 0.09		2.1 $\pm$ 0.2	1.4 $\pm$ 0.1
10-S65-3	<0.03	1.20 $\pm$ 0.12	1.02 $\pm$ 0.10		2.1 $\pm$ 0.2	1.4 $\pm$ 0.1
10-S65-4	<0.03	1.04 $\pm$ 0.10	0.55 $\pm$ 0.06		1.6 $\pm$ 0.2	1.1 $\pm$ 0.1
10-S65-5	<0.03	1.08 $\pm$ 0.11	0.75 $\pm$ 0.08		1.8 $\pm$ 0.2	1.2 $\pm$ 0.1
10-S65-6	<0.03	1.09 $\pm$ 0.11	0.82 $\pm$ 0.08		1.6 $\pm$ 0.2	1.1 $\pm$ 0.1
10-S65-7	BDL	1.34 $\pm$ 0.13	0.85 $\pm$ 0.08		1.8 $\pm$ 0.2	1.2 $\pm$ 0.1
10-S65-8	BDL	1.20 $\pm$ 0.12	0.91 $\pm$ 0.09		1.6 $\pm$ 0.2	1.1 $\pm$ 0.1
10-S65-9	<0.03	0.97 $\pm$ 0.10	0.76 $\pm$ 0.08		1.3 $\pm$ 0.1	0.9 $\pm$ 0.1
10-S65-10	<0.03	1.18 $\pm$ 0.12	0.95 $\pm$ 0.10		1.4 $\pm$ 0.1	1.0 $\pm$ 0.1
10-S65-11	BDL	1.30 $\pm$ 0.13	1.67 $\pm$ 0.17		3.4 $\pm$ 0.3	2.3 $\pm$ 0.2
10-S65-12	<0.03	1.18 $\pm$ 0.12	0.74 $\pm$ 0.07		1.4 $\pm$ 0.1	1.0 $\pm$ 0.1
10-S65-13	<0.03	1.52 $\pm$ 0.15	0.99 $\pm$ 0.10		1.3 $\pm$ 0.1	0.9 $\pm$ 0.1
10-S65-14	BDL	1.00 $\pm$ 0.10	0.81 $\pm$ 0.08		1.5 $\pm$ 0.1	1.0 $\pm$ 0.1
10-S66-1	<0.03	0.93 $\pm$ 0.09	0.62 $\pm$ 0.06		1.2 $\pm$ 0.1	0.8 $\pm$ 0.1
10-S66-4	<0.03	2.49 $\pm$ 0.25	1.15 $\pm$ 0.12		1.6 $\pm$ 0.2	1.1 $\pm$ 0.1
10-S66-5	BDL	1.20 $\pm$ 0.12	1.14 $\pm$ 0.11		2.2 $\pm$ 0.2	1.5 $\pm$ 0.2
10-S66-6	<0.03	1.27 $\pm$ 0.13	1.05 $\pm$ 0.10		18.8 $\pm$ 1.9	7.9 $\pm$ 0.8 ^d
10-S66-7	<0.03	1.09 $\pm$ 0.11	0.84 $\pm$ 0.08		2.0 $\pm$ 0.2	1.4 $\pm$ 0.1
10-S66-8	<0.03	1.34 $\pm$ 0.13	1.17 $\pm$ 0.12		2.5 $\pm$ 0.2	1.7 $\pm$ 0.2
10-S66-9	<0.03	1.24 $\pm$ 0.12	1.51 $\pm$ 0.15		2.9 $\pm$ 0.3	2.0 $\pm$ 0.2
10-S66-10	<0.03	1.57 $\pm$ 0.16	1.20 $\pm$ 0.12		1.7 $\pm$ 0.2	1.2 $\pm$ 0.1
10-S66-11	<0.03	1.54 $\pm$ 0.15	1.24 $\pm$ 0.12		2.0 $\pm$ 0.2	1.4 $\pm$ 0.1
10-S66-12	BDL	1.83 $\pm$ 0.18	1.62 $\pm$ 0.16		1.9 $\pm$ 0.2	1.3 $\pm$ 0.1
10-S66-13	<0.03	2.42 $\pm$ 0.24	4.64 $\pm$ 0.46		6.7 $\pm$ 0.7	4.6 $\pm$ 0.5
10-S66-14	<0.03	1.28 $\pm$ 0.13	0.80 $\pm$ 0.08		1.4 $\pm$ 0.1	1.0 $\pm$ 0.1
10-S67-1	0.08 $\pm$ 0.03	1.39 $\pm$ 0.14	0.60 $\pm$ 0.06		1.2 $\pm$ 0.1	0.8 $\pm$ 0.1
10-S67-2	<0.03	1.16 $\pm$ 0.12	0.78 $\pm$ 0.08		1.2 $\pm$ 0.1	0.8 $\pm$ 0.1
10-S67-3	0.04 $\pm$ 0.02	1.13 $\pm$ 0.11	0.59 $\pm$ 0.06		0.7 $\pm$ 0.1	0.5 $\pm$ 0.1
10-S67-4	<0.03	0.94 $\pm$ 0.09	0.61 $\pm$ 0.06		1.0 $\pm$ 0.1	0.7 $\pm$ 0.1
10-S67-5	<0.03	0.82 $\pm$ 0.08	0.50 $\pm$ 0.05		1.2 $\pm$ 0.1	0.8 $\pm$ 0.1
10-S67-6	BDL	1.42 $\pm$ 0.14	0.99 $\pm$ 0.10		1.9 $\pm$ 0.2	1.3 $\pm$ 0.1
10-S67-7	<0.03	1.33 $\pm$ 0.13	1.07 $\pm$ 0.11		2.1 $\pm$ 0.2	1.4 $\pm$ 0.1
10-S67-8	BDL	1.08 $\pm$ 0.11	0.77 $\pm$ 0.08		1.6 $\pm$ 0.2	1.1 $\pm$ 0.1
10-S67-9	<0.03	1.11 $\pm$ 0.11	0.95 $\pm$ 0.10		1.5 $\pm$ 0.1	1.0 $\pm$ 0.1
10-S67-10	0.52 $\pm$ 0.05	1.49 $\pm$ 0.15	0.64 $\pm$ 0.06		1.7 $\pm$ 0.2	1.2 $\pm$ 0.1
10-S67-11	<0.03	1.36 $\pm$ 0.14	1.06 $\pm$ 0.11		2.1 $\pm$ 0.2	1.4 $\pm$ 0.1
10-S67-12	0.14 $\pm$ 0.04	1.45 $\pm$ 0.14	1.20 $\pm$ 0.12		2.3 $\pm$ 0.2	1.6 $\pm$ 0.1
10-S67-13	<0.03	1.16 $\pm$ 0.12	1.18 $\pm$ 0.12		2.2 $\pm$ 0.2	1.5 $\pm$ 0.1
10-S67-14	BDL	1.23 $\pm$ 0.12	0.87 $\pm$ 0.09		1.8 $\pm$ 0.2	1.2 $\pm$ 0.1

TABLE 4
(cont'd.)

Sample No.	¹³⁷ Cs	Gamma Spectra, pCi/g±σ ^a		Uranium Fluorometric ^b	
		²³² Th	²²⁶ Ra	µg/g±10%	pCi/g±σ10%
		Decay Chain	Decay Chain		
SOIL BORINGS - (cont'd.)					
10-S68-1	BDL	1.01±0.10	0.61±0.06	1.2 ±0.1	0.8 ±0.1
10-S68-2	BDL	1.06±0.11	0.52±0.05	0.9 ±0.1	0.6 ±0.1
10-S68-3	BDL	1.10±0.11	0.54±0.05	1.1 ±0.1	0.8 ±0.1
10-S68-4	BDL	1.22±0.12	0.67±0.07	1.4 ±0.1	1.0 ±0.1
10-S68-5	BDL	0.89±0.09	0.58±0.06	1.0 ±0.1	0.7 ±0.1
10-S68-6	BDL	0.91±0.09	0.46±0.05	1.2 ±0.1	0.8 ±0.1
10-S68-7	BDL	0.98±0.10	0.60±0.06	1.0 ±0.1	0.7 ±0.1
10-S68-8	BDL	0.98±0.10	0.60±0.06	1.0 ±0.1	0.7 ±0.1
10-S68-9	BDL	1.04±0.10	0.52±0.05	1.2 ±0.1	0.8 ±0.1
10-S68-10	BDL	1.13±0.11	1.09±0.11	2.0 ±0.2	1.4 ±0.1
10-S68-11	BDL	1.21±0.12	1.63±0.16	2.8 ±0.3	1.9 ±0.2
10-S68-12	BDL	1.17±0.12	1.46±0.15	1.9 ±0.2	1.3 ±0.1
10-S68-13	BDL	1.66±0.17	2.54±0.25	3.8 ±0.4	2.6 ±0.3
10-S68-14	BDL	0.89±0.09	1.25±0.12	1.6 ±0.2	1.1 ±0.1
10-S69-1	<0.03	1.51±0.15	1.17±0.12	2.2 ±0.2	1.5 ±0.2
10-S69-2	<0.03	1.08±0.11	0.67±0.07	3.2 ±0.3	2.2 ±0.2
10-S69-3	BDL	1.21±0.12	1.10±0.11	1.7 ±0.2	1.2 ±0.1
10-S69-4	<0.03	1.11±0.11	0.67±0.07	1.5 ±0.1	1.0 ±0.1
10-S69-5	<0.03	1.38±0.14	0.67±0.07	1.2 ±0.1	0.8 ±0.1
10-S69-6	BDL	1.21±0.12	0.74±0.07	0.9 ±0.1	0.6 ±0.1
10-S69-7	<0.03	1.65±0.16	0.83±0.08	1.0 ±0.1	0.7 ±0.1
10-S69-8	<0.03	1.82±0.18	1.10±0.11	1.8 ±0.2	1.2 ±0.1
10-S69-9	BDL	1.63±0.16	0.96±0.10	1.6 ±0.2	1.1 ±0.1
10-S69-11	<0.03	1.71±0.17	0.79±0.08	1.2 ±0.1	0.8 ±0.1
10-S69-12	<0.03	1.72±0.17	1.28±0.13	1.9 ±0.2	1.3 ±0.1
10-S70-1	BDL	1.26±0.13	0.70±0.07	1.3 ±0.1	0.9 ±0.1
10-S70-2	BDL	0.98±0.10	0.56±0.06	0.8 ±0.1	0.5 ±0.1
10-S70-3	<0.03	1.15±0.12	0.74±0.07	1.5 ±0.1	1.0 ±0.1
10-S70-4	BDL	1.26±0.13	0.83±0.08	1.6 ±0.2	1.1 ±0.1
10-S70-5	BDL	1.30±0.13	0.90±0.09	1.7 ±0.2	1.2 ±0.1
10-S70-6	BDL	0.99±0.10	0.73±0.07	1.5 ±0.1	1.0 ±0.1
10-S70-7	BDL	0.96±0.10	0.54±0.05	1.1 ±0.1	0.8 ±0.1
10-S70-8	BDL	0.94±0.09	0.62±0.06	1.2 ±0.1	0.8 ±0.1
10-S70-9	BDL	1.10±0.11	0.63±0.06	1.4 ±0.1	1.0 ±0.1
10-S70-10	BDL	0.85±0.08	0.88±0.09	1.3 ±0.1	0.9 ±0.1
10-S70-11	BDL	1.07±0.11	0.93±0.09	1.2 ±0.1	0.8 ±0.1
10-S70-12	BDL	1.27±0.13	1.48±0.15	3.1 ±0.3	2.1 ±0.2
10-S70-13	BDL	1.28±0.13	1.52±0.15	2.5 ±0.3	1.7 ±0.2
10-S70-14	BDL	1.63±0.16	2.81±0.28	3.7 ±0.4	2.5 ±0.3
10-S70-15	BDL	1.18±0.12	1.68±0.17	2.6 ±0.3	1.8 ±0.2
10-S70-16	BDL	1.69±0.17	2.74±0.27	4.5 ±0.4	3.1 ±0.3

TABLE 4
(cont'd.)

Sample No.	Gamma Spectra, pCi/g $\pm\sigma^a$				Uranium Fluorometric	
	^{137}Cs	^{232}Th Decay Chain	^{226}Ra Decay Chain		$\mu\text{g/g}\pm 10\%$	pCi/g $\pm 10\%$
SOIL BORINGS - (cont'd.)						
10-S71-1	<0.03	1.05 $\pm$ 0.10	0.69 $\pm$ 0.07		1.2 $\pm$ 0.1	0.8 $\pm$ 0.1
10-S71-2	<0.03	0.85 $\pm$ 0.08	0.39 $\pm$ 0.04		0.9 $\pm$ 0.1	0.5 $\pm$ 0.1
10-S71-3	<0.03	1.00 $\pm$ 0.10	0.31 $\pm$ 0.03		1.3 $\pm$ 0.1	0.9 $\pm$ 0.1
10-S71-4	<0.03	1.58 $\pm$ 0.16	0.51 $\pm$ 0.05		1.1 $\pm$ 0.1	0.8 $\pm$ 0.1
10-S71-5	<0.03	1.13 $\pm$ 0.11	0.38 $\pm$ 0.04		0.9 $\pm$ 0.1	0.6 $\pm$ 0.1
10-S71-6	<0.03	1.01 $\pm$ 0.10	0.80 $\pm$ 0.08		1.4 $\pm$ 0.1	1.0 $\pm$ 0.1
10-S71-7	0.59 $\pm$ 0.06	1.13 $\pm$ 0.11	0.78 $\pm$ 0.08		1.8 $\pm$ 0.2	1.2 $\pm$ 0.1
10-S71-8	0.53 $\pm$ 0.05	1.11 $\pm$ 0.11	0.96 $\pm$ 0.10		1.8 $\pm$ 0.2	1.2 $\pm$ 0.1
10-S71-9	0.58 $\pm$ 0.06	1.11 $\pm$ 0.11	0.98 $\pm$ 0.10		1.6 $\pm$ 0.2	1.1 $\pm$ 0.1
10-S71-10	0.57 $\pm$ 0.06	0.99 $\pm$ 0.10	0.89 $\pm$ 0.09		1.6 $\pm$ 0.2	1.1 $\pm$ 0.1
10-S71-11	0.74 $\pm$ 0.07	0.72 $\pm$ 0.07	1.10 $\pm$ 0.11		1.6 $\pm$ 0.2	1.1 $\pm$ 0.1
10-S71-12	0.50 $\pm$ 0.05	0.78 $\pm$ 0.08	0.84 $\pm$ 0.08		1.5 $\pm$ 0.1	1.0 $\pm$ 0.1
10-S72-1	<0.03	0.06 $\pm$ 0.03	<0.02		1.2 $\pm$ 0.1	0.8 $\pm$ 0.1
10-S72-2	<0.03	1.36 $\pm$ 0.14	1.12 $\pm$ 0.11		2.2 $\pm$ 0.2	1.5 $\pm$ 0.1
10-S72-3	<0.03	1.37 $\pm$ 0.14	2.05 $\pm$ 0.20		4.3 $\pm$ 0.4	2.9 $\pm$ 0.3
10-S72-4	<0.03	1.24 $\pm$ 0.12	0.99 $\pm$ 0.10		2.0 $\pm$ 0.2	1.4 $\pm$ 0.1
10-S72-5	<0.03	0.94 $\pm$ 0.09	0.72 $\pm$ 0.07		1.3 $\pm$ 0.1	0.9 $\pm$ 0.1
10-S72-6	0.03 $\pm$ 0.02	0.88 $\pm$ 0.09	0.67 $\pm$ 0.07		1.3 $\pm$ 0.1	0.9 $\pm$ 0.1
10-S72-7	<0.03	1.11 $\pm$ 0.11	0.78 $\pm$ 0.08		1.3 $\pm$ 0.1	0.9 $\pm$ 0.1
10-S72-8	BDL	1.71 $\pm$ 0.17	2.13 $\pm$ 0.21		2.9 $\pm$ 0.3	2.0 $\pm$ 0.2
10-S72-9	<0.03	1.35 $\pm$ 0.14	2.17 $\pm$ 0.22		3.5 $\pm$ 0.4	2.4 $\pm$ 0.2
10-S72-10	<0.03	1.39 $\pm$ 0.14	2.77 $\pm$ 0.28		4.6 $\pm$ 0.5	3.1 $\pm$ 0.3
10-S73-1	<0.03	1.49 $\pm$ 0.15	0.99 $\pm$ 0.10		1.3 $\pm$ 0.1	0.9 $\pm$ 0.1
10-S73-2	<0.03	1.59 $\pm$ 0.16	1.65 $\pm$ 0.16		2.2 $\pm$ 0.2	1.5 $\pm$ 0.1
10-S73-3	<0.03	1.65 $\pm$ 0.16	1.29 $\pm$ 0.13		2.2 $\pm$ 0.2	1.5 $\pm$ 0.1
10-S73-4	<0.03	2.10 $\pm$ 0.21	1.18 $\pm$ 0.12		2.3 $\pm$ 0.2	1.6 $\pm$ 0.2
10-S73-5	BDL	1.52 $\pm$ 0.15	2.66 $\pm$ 0.27		4.4 $\pm$ 0.4	3.0 $\pm$ 0.3
10-S73-6	BDL	1.37 $\pm$ 0.14	1.70 $\pm$ 0.17		4.6 $\pm$ 0.5	3.1 $\pm$ 0.3
10-S73-7	BDL	1.62 $\pm$ 0.16	1.36 $\pm$ 0.14		2.7 $\pm$ 0.3	1.8 $\pm$ 0.2
10-S73-8	BDL	1.82 $\pm$ 0.18	1.56 $\pm$ 0.16		2.6 $\pm$ 0.3	1.8 $\pm$ 0.2
10-S73-9	BDL	1.29 $\pm$ 0.13	1.54 $\pm$ 0.15		2.2 $\pm$ 0.2	1.5 $\pm$ 0.2
10-S73-10	BDL	1.53 $\pm$ 0.15	1.68 $\pm$ 0.17		2.5 $\pm$ 0.2	1.7 $\pm$ 0.2
10-S74-1	<0.03	0.89 $\pm$ 0.09	0.82 $\pm$ 0.08		1.3 $\pm$ 0.1	0.9 $\pm$ 0.1
10-S74-2	<0.03	1.23 $\pm$ 0.12	0.45 $\pm$ 0.04		1.1 $\pm$ 0.1	0.8 $\pm$ 0.1
10-S74-3	<0.03	1.07 $\pm$ 0.11	0.38 $\pm$ 0.04		1.2 $\pm$ 0.1	0.8 $\pm$ 0.1
10-S74-4	<0.03	1.02 $\pm$ 0.10	0.76 $\pm$ 0.08		1.4 $\pm$ 0.1	1.0 $\pm$ 0.1
10-S74-6	<0.03	1.18 $\pm$ 0.12	0.86 $\pm$ 0.09		1.2 $\pm$ 0.1	0.8 $\pm$ 0.1
10-S74-7	<0.03	0.98 $\pm$ 0.10	0.46 $\pm$ 0.05		1.4 $\pm$ 0.1	1.0 $\pm$ 0.1

TABLE 4
(cont'd.)

Gamma Spectra, pCi/g±σ ^a					
Sample No.	¹³⁷ Cs	²³² Th	²²⁶ Ra	Uranium Fluorometric	
		Decay Chain	Decay Chain	μg/g±10%	pCi/g±10% ^b
SOIL BORINGS - (cont'd.)					
10-S75-1	<0.03	1.31±0.13	0.80±0.08	1.4 ±0.1	1.0 ±0.1
10-S75-2	BDL	1.03±0.10	0.56±0.06	2.5 ±0.2	1.7 ±0.2
10-S75-3	<0.03	1.27±0.13	0.73±0.07	1.0 ±0.1	0.7 ±0.1
10-S75-4	0.05±0.02	1.05±0.10	0.67±0.07	1.1 ±0.1	0.8 ±0.1
10-S75-5	0.04±0.02	1.14±0.11	0.67±0.07	1.4 ±0.1	1.0 ±0.1
10-S75-6	0.05±0.02	0.99±0.10	0.45±0.04	1.2 ±0.1	0.8 ±0.1
10-S75-7	0.57±0.06	1.14±0.11	0.92±0.09	1.6 ±0.2	1.1 ±0.1
10-S76-1	<0.03	1.42±0.14	0.84±0.08	1.8 ±0.2	1.2 ±0.1
10-S76-2	<0.03	1.16±0.12	0.62±0.06	1.6 ±0.2	1.1 ±0.1
10-S76-3	<0.03	1.90±0.19	1.51±0.15	2.4 ±0.2	1.6 ±0.2
10-S76-4	<0.03	1.94±0.19	1.59±0.16	3.1 ±0.3	2.1 ±0.2
10-S76-5	BDL	1.77±0.18	1.72±0.17	3.0 ±0.3	2.0 ±0.2
10-S76-6	BDL	1.87±0.19	1.97±0.20	2.8 ±0.3	1.9 ±0.2
10-S76-7	BDL	1.27±0.13	1.10±0.11	1.6 ±0.2	1.1 ±0.1
10-S76-8	BDL	1.67±0.17	1.59±0.16	2.3 ±0.2	1.6 ±0.2
10-S76-9	<0.03	1.59±0.16	1.26±0.13	1.6 ±0.2	1.1 ±0.1
10-S76-12	<0.03	1.64±0.16	1.31±0.13	2.1 ±0.2	1.4 ±0.1
10-S76-13	BDL	1.26±0.13	0.99±0.10	1.9 ±0.2	1.3 ±0.1
10-S76-15	<0.03	1.90±0.19	1.34±0.13	2.6 ±0.3	1.8 ±0.2
10-S76-16	<0.03	1.12±0.11	0.94±0.09	2.1 ±0.2	1.4 ±0.1
BACKGROUND SOIL BORING					
1-SB7-1	BDL	1.59±0.18	0.75±0.09		
1-SB7-2	BDL	1.53±0.14	0.87±0.07		
1-SB7-3.8	BDL	1.29±0.17	0.83±0.08		
1-SB7-4.3	BDL	0.97±0.18	0.68±0.08		
1-SB7-5.1	BDL	1.92±0.22	0.51±0.10		
1-SB7-6.2	BDL	1.01±0.26	1.58±0.12		
1-SB7-7.2	BDL	1.13±0.12	0.63±0.07		
1-SB7-8.2	BDL	0.95±0.21	0.81±0.10		
1-SB7-9.2	BDL	1.63±0.17	0.86±0.07		
1-SB7-10.2	BDL	1.56±0.20	0.72±0.10		
1-SB7-11.2	BDL	1.14±0.15	0.84±0.08		
1-SB7-12.2	BDL	1.17±0.19	0.61±0.08		
1-SB7-13.2	BDL	0.92±0.29	0.69±0.11		
1-SB7-14.2	BDL	0.36±0.28	0.50±0.11		
1-SB7-15.1	BDL	1.33±0.13	0.91±0.07		
1-SB7-16.3	BDL	1.83±0.31	0.99±0.14		

TABLE 4
(cont'd.)

^aOne standard deviation due to counting statistics.

^bANL conversion from Appendix 5.

^cBDL = Below Detectable Limits

^dA specific activity of 0.419 pCi/μg U was calculated for these samples from the data in Table 5. More details of the calculation are given in Appendix 5.

TABLE 5

URANIUM MASS SPECTROMETRIC ANALYSIS

Sample No.	Atom % U				
	233	234	235	236	238
10-S66-6	<0.0005	0.0012 ± 0.0005	0.2713 ± 0.0022	0.0045 ± 0.0005	99.723 ± 0.0003

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 cm² (AC-21), 100 cm², or 325 cm² (AC-22) in area, with a thin double-aluminized Mylar window (~ 0.85 mg/cm²).

Since this instrument has multiple high-voltage settings, it can be used to distinguish between alpha and beta-gamma contamination. This instrument was initially used in the beta mode. In that mode, the detector responds to alpha and beta particles 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.

The alpha voltage is set to 1600 V, and the input discriminator is set to 1.5 mV. The instrument is then calibrated in the alpha mode with four flat-plate, infinitely-thin, NBS-traceable ²³⁹Pu standards, and in the beta mode with a flat-plate, infinitely-thin, NBS-traceable ⁹⁰Sr-⁹⁰Y 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, making the window density approximately 7 mg/cm². At this density, the instrument is not sensitive to the majority of alpha emissions. 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. This instrument is calibrated with a ²²⁶Ra 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 (2 in) in diameter by 2 mm thick. This instrument is calibrated on three separate discriminators for three energy regions using ²³⁹Pu (17 keV), ²⁴¹Am (59.5 keV), and ²³⁵U (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 (1 in) in diameter by 2.5 cm (1 in) thick and can be used to measure fields of low-level radiation between 10 μ R/h and 5000 μ R/h. This instrument is used to survey ambient background radiation. It is held 1 m from the surface during the survey. This instrument is also calibrated with a standard ^{226}Ra source.

E. Integrating Radiation Meter

In addition to the PRM-7, a pressurized ionization chamber (Reuter Stokes Model RSS-111) was used at selected locations to determine the ambient radiation field in units of R/h. The RSS-111 has three output modes: (1) instantaneous exposure rate, (2) strip chart differential readout, and (3) integrated exposure. The chamber is mounted on a tripod, 3 ft ($\sim$ 1 m) above the surface and has a uniform energy response from about 0.2 MeV to about 4 MeV. A 2-hour 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 to count in the alpha mode; the other is used in the beta mode. Up to 10 samples can be counted simultaneously.

Any smear taken from a contaminated area was counted individually in a Nuclear Measurements Corporation gas-flow proportional counter (PC-5 or PC-3A). These instruments have been modified to contain a double-aluminized Mylar spun top (window). This top is placed over non-conducting media (e.g. paper smears) to negate the dielectric effect on the counter. These counters also use P-10 counting gas. Smears are counted in both the alpha and beta modes.

The PC counters are calibrated by adjusting the input discriminator with the high voltage set at 700 V until it begins to count an alpha source. The plateaus are run to establish the operating voltages for alpha and beta-gamma. The MS-2 input discriminator is set to 2 mV and again plateaus are run to establish the operating voltages.

III. AIR-SAMPLING DEVICE

Air samples were collected using a commercially available (ANL-modified Filter Queen) vacuum cleaner identified as the "Princess Model." Air was drawn through a filter medium at a flow rate of 40 m^3/h . The filters consist of 200- cm^2

APPENDIX 1
(cont'd.)

sheets of Hollingsworth-Vose (HV-70) or LB5211-9 mil filter paper. The collection efficiency at these flow rates for 0.3- μ m 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 membrane (0.5 to 0.8 μ m) filter paper for about one hour. An alpha spectrum can be measured from a section of this filter paper. The ratio of actinon (^{219}Rn) daughters (6.62 MeV α , AcC) to radon (^{222}Rn) daughters (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 (3 in) diameter by 7.6-cm (3 in) thick NaI(Tl) scintillation crystal is commonly used for determining a gamma spectrum. This instrument is calibrated with NBS-traceable gamma sources. Samples from contaminated areas were analyzed using this system, and the contamination radionuclides were identified.

Hyperpure Germanium detectors (ORTEC - 17% efficiency right-circular cylinders) can be used when more sophisticated gamma-ray analyses are 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 cm² (dis/min-100 cm²) and the derivation of those factors are listed below.

A. Conversion Factors

	PAC-4G-3 (51 cm ²)		PAC-4G-3 (100 cm ²)		Floor Monitor (FM-4G)	
	<u>Alpha</u>	<u>Beta</u>	<u>Alpha</u>	<u>Beta</u>	<u>Alpha</u>	<u>Beta</u>
To 100 cm ²	1.96	1.96	1.00	1.00	0.31	0.31
dis/min per cts/min for ⁹⁰ Sr- ⁹⁰ Y	-	2	-	2	-	2
dis/min per cts/min for ²³⁹ Pu	2	-	2	-	2	-
dis/min per cts/min for normal U	3.5	2.7	2.7	2.4	3.0	2.5
dis/min per cts/min for ²²⁶ Ra plus daughters	1.7	1.7	1.7	1.8	1.7	1.8

B. Derivation of Conversion Factors. Floor Monitor

Window Area: ~ 325 cm²

Conversion to 100 cm² = 0.31 times Floor Monitor readings

. PAC-4G-3

Window Area: ~ 51 cm²

Conversion to 100 cm² = 1.96 times PAC reading

Window Area: ~ 100 cm²

Conversion factor = 1.00 times PAC reading

APPENDIX 2
(cont'd)

. 2 π 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 ($\sim 0.85 \text{ mg/cm}^2$)} 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.

The PAC-4G-3 and PC counter were calibrated as described in Appendix 1. With a flat-plate, infinitely thin ^{226}Ra plus short-lived daughters standard used as a source of alpha emissions, the plate was counted in a 2π Internal Gas-Flow Counter (PC counter) with the source leveled to an apparent 2π geometry. The alpha counts per minute (cts/min) reading was found to be 1.86×10^4 cts/min, or $1.86 \times 10^4 \div 0.51^* = 3.65 \times 10^4$ 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 2.18×10^4 cts/min at contact. The conversion factor for cts/min to dis/min for the PAC instrument is $3.65 \times 10^4 \div 2.18 \times 10^4 = 1.7$ dis/min per cts/min alpha.

The same source was covered with two layers of conducting paper, each 6.31 mg/cm^2 , to absorb the alpha emissions. With the PC counter in the beta mode and the paper in good contact with the chamber, the count was found to be 1.17×10^4 or $1.17 \times 10^4 \div 0.50 = 2.35 \times 10^4$ dis/min beta. 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 1.36×10^4 cts/min. This indicates a conversion factor of $2.35 \times 10^4 \div 1.36 \times 10^4 = 1.7$ dis/min per cts/min beta-gamma. All three detectors (51 cm^2 , 100 cm^2 , and 325 cm^2) gave readings similar to those reported above for the alpha and beta-gamma modes.

Utilizing a $1.25 \text{ in} \times 1.25 \text{ in} \times 0.005 \text{ in}$ ($3.2 \text{ cm} \times 3.2 \text{ cm} \times 0.013 \text{ cm}$) normal uranium foil as a source of uranium alpha emissions, the foil was counted in a PC counter with the source leveled to an apparent 2π geometry. The same normal uranium source, covered with two layers of conducting paper in good contact with the chamber, each 6.31 mg/cm^2 to negate the alpha emissions, was counted for composite beta and gamma emissions in the PC counter. The source was leveled to an apparent 2π geometry; however, no provision was made for backscatter.

*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 $g \times bf \times sa \times waf$ is 0.51.

APPENDIX 2 (cont'd.)

The normal uranium source was also counted with the PAC instrument using all three detector areas in the alpha mode and covered with two layers of conducting paper in the beta mode. The conversion factors were calculated as for ^{226}Ra .

II. SMEAR COUNT

The conversion factors for cts/min-100 cm² to dis/min-100 cm² for smear counts are given below:

A. Conversion Equation (Alpha)

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

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 is used when determining alpha activity on a filter media.

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

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

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

B. Conversion Equation (Beta)

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

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 is used when determining beta activity on a filter media.

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

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

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

APPENDIX 3

RADON-DETERMINATION CALCULATIONS

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

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) for the alpha activity is 1.0.
- 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-minute decay 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) all are 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 minutes) based on the ThB half-life.
- Q. For all practical purposes, 36% of the ThC (alpha branch) and the ThC' decay at the same rate as ThB, which is 638.4 minutes.
- 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 all are 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 same rate as 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.

- Z. 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.

APPENDIX 3
(cont'd.)

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)

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

t = time interval (minutes) from end of sampling period to counting interval (usually ~ 100 minutes)

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

$t_{1/2}$ = 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-m³)

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

f = sampling rate (m³/min = m³/h x 1 h/60 minutes)

t_s = length of sampling time (minutes)

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

$t_{1/2}$ = half-life of isotope or controlling parent (minutes).

III. ACTINON CORRECTION

Since the actinon (²¹⁹Rn) progeny (AcA, AcB & AcC) decays at the AcB half-life of 36 minutes, it cannot be distinguished from the radon (²²²Rn) 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 membrane (0.5 to 0.8 μ m) filter media. The sample rate is

APPENDIX 3 (cont'd.)

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:

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

Where: B_j = summation of the counts in 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 fractions 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	$f = 40 \text{ m}^3/60 \text{ min}$	$t_s = 40 \text{ min}$
at $t = 100 \text{ min}$		$A^s = 2000 \text{ dis/min}$
at $t = 360 \text{ min}$		$A = 140 \text{ dis/min}$
at $t = 7 \text{ days}$		$A = 5 \text{ dis/min}$

For long-lived activity:

$$A_o = A = 5 \text{ dis/min}$$

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

APPENDIX 3
(cont'd.)

For thoron:

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

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

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, 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.)

$$B_1 = 44$$

$$B_1/0.84 = 52 \text{ counts}$$

$$B_2 = 601 \text{ counts}$$

$$\text{Actinon} = 52/653 = 0.08$$

$$\text{Radon} = 601/653 = 0.92$$

$$C(\text{Rn}) = C(36) \times \text{Radon}\% = 669.7 \times 0.92 = 616.1 \text{ dis/min-m}^3$$

$$C = C(36) \times \text{Actinon}\% = 669.7 \times 0.08 = 53.6 \text{ dis/min-m}^3.$$

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

$$C \text{ actual} = C \text{ measured} \times \text{progeny correction factor}$$

$$C(\text{L}) = 0.19 \text{ dis/min-m}^3$$

$$C(\text{Tn}) = 7.6 \text{ dis/min-m}^3 \times 2 = 15.2 \text{ dis/min-m}^3$$

$$C(\text{An}) = 53.6 \text{ dis/min-m}^3 \times 2 = 107.2 \text{ dis/min-m}^3$$

$$C(\text{Rn}) = 616 \text{ dis/min-m}^3 \times 2 = 1232 \text{ dis/min-m}^3.$$

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

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

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

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

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

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 each 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 "wet 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 (approximately 48 hours). 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 (1½" x 1½" 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 µ mesh) 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). A 5-g sample of the sieved material is transferred to a 1-oz Nalgene bottle. One gram of this sample is used for the determination of uranium by laser fluorometry; 100 g are needed for radiochemical analysis of Pu, Am, and Th if these analyses are required. 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 rinsed three times with tap water and three times with demineralized water, 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 with warm air until absolutely dry. Room air is drawn through Mill jars with a hose which is attached to a fume hood or specially constructed drying box.

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 furnace. The first trap is immersed in an ordinary ice bath (0°C); the second trap is immersed in a CO₂-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, or less, fraction from each soil sample is analyzed by high resolution gamma-ray spectroscopy using a germanium crystal detector coupled to the appropriate electronics and a multichannel analyzer. This analysis allows for a quantitative determination of the ²²⁶Ra decay chain (via the 609 keV γ -ray of ²¹⁴Pb) and the ²³²Th decay chain (via the 911 keV γ -ray of ²²⁸Ac), as well as any other gamma emitting radionuclide (e.g., ¹³⁷Cs) 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 or other sample (using a ²³⁴Th spike for yield determination) followed by plating a thin source of the radiochemically separated thorium and determining the thorium isotopes (²²⁸Th and ²³²Th) by alpha spectroscopy.

Plutonium and/or americium analysis consists of an acid leach of the soil or other sample (using the appropriate spike for yield determination) followed by plating a thin source of the radiochemically separated plutonium or americium and determining the isotopes (²³⁸Pu, ^{239,240}Pu, ²⁴¹Am) 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 radiothorium (^{228}Th decay chain), plutonium (^{238}Pu , 239 , ^{240}Pu), and americium (^{241}Am) present in the contaminated material.

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

APPENDIX 5

CALCULATION OF 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 of ^{235}U and ^{238}U were taken from A. H. Jaffey et al., Phys. Rev. C 4 1889 (1971). The half-life for each isotope was taken from David C. Kocher, "Radioactive Decay Tables - A Handbook of Decay Data for Application to Radiation Dosimetry and Radiological Assessments" (1981). The percent abundances were taken from N. E. Holden, BNL-NCD-50605 (1977). Atomic weights were taken from the Handbook of Chemistry and Physics, 52nd Edition (1971). The specific activity of ^{234}U was calculated from the half-life.

Isotope	Specific Activity (dis/min- μg)	Half-life (years)	Abundance (atom %)	Atomic Weight (grams)	Abundance (wt %)
^{234}U	1.387×10^4	2.445×10^5	0.0054	234.0409	0.0053
^{235}U	4.798	7.038×10^8	0.720	235.0439	0.7110
^{238}U	0.746	4.4683×10^9	99.2746	238.0508	99.2837
			100.0000		100.0000

where $(\text{wt } \%)_i =$

$$\frac{(\text{atom } \%)_i (\text{atomic weight})_i}{\sum_j (\text{atom } \%)_j (\text{atomic weight})_j} = \frac{(\text{atom } \%)_i (\text{atomic weight})_i}{238.0289}$$

Specific activity for normal uranium:

$$\begin{aligned} 0.746 \times 0.99284 \times 2 &= 1.481 \quad \text{dis/min-}\mu\text{g from } ^{234}\text{U} \text{ \& } ^{238}\text{U} \\ 4.798 \times 0.00711 &= 0.034 \quad \text{dis/min-}\mu\text{g from } ^{235}\text{U} \\ &1.515 \quad \text{dis/min-}\mu\text{g for normal U} \end{aligned}$$

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

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 .

Calculation of the specific activity of other than normal mixtures of uranium isotopes are performed in a similar manner. For example, neutron activated reprocessed uranium having the isotopic composition 238(0.99723), 236(0.000045), 235(0.002713), 234(0.000012) as determined by mass spectrometry would have the following specific activity:

APPENDIX 5
(cont'd.)

$$\begin{aligned}
 0.99723 \times 0.746 &= 0.7439 \text{ dis/min-}\mu\text{g from } ^{238}\text{U} \\
 0.000045 \times 1.436 \times 10^2 &= 0.0065 \text{ dis/min-}\mu\text{g from } ^{236}\text{U}^a \\
 0.002713 \times 4.798 &= 0.0130 \text{ dis/min-}\mu\text{g from } ^{235}\text{U} \\
 0.000012 \times 1.387 \times 10^4 &= \underline{0.1664} \text{ dis/min-}\mu\text{g from } ^{234}\text{U} \\
 &0.9299 \text{ dis/min-}\mu\text{g U total}
 \end{aligned}$$

corresponding to:

$$(0.9299 \text{ dis/min-}\mu\text{g}) / (2.22 \text{ dis/min-pCi}) = 0.419 \text{ pCi}/\mu\text{g U}$$

^aThe half-life for ^{236}U (2.342×10^7 yr) and the corresponding specific activity (1.436×10^2 dis/min- μg) were taken from K. F. Flynn, et. al. J. Inorg. Nucl. Chem. 34 1121 (1972).

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 non-occupational MPC (Note 2) is 2×10^{-13} Ci/m ³ or less or for which the nonoccupational MPC (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 (Note 2) is 1×10^{-12} Ci/m ³ or less for which the nonoccupational MPC (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 cm² 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 cm², if (1) from measurements of a representative number n of sections it is determined that $1/n \sum_{i=1}^n S_i \geq L$, where S_i is the dis/min-100 cm² 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 cm² 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 cm².
- (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 ≤ 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.

APPENDIX 6
(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

APPENDIX 6
(Cont'd.)

ALTERNATE SURFACE CONTAMINATION LIMITS

* The levels may be averaged over one square meter provided the maximum activity in any area of 100 cm² 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 cm², if (1) from measurements of a representative number n of sections it is determined that $1/n \sum_{i=1}^n S_i \geq L$, where S_i is the dis/min-100 cm² 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 cm² 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 cm².
- (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 ≤ 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.

APPENDIX 6
(Cont'd.)

II.

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

APPENDIX 6
(cont'd.)

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

APPENDIX 6
(Cont'd.)

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 (nu- clides with decay modes other than alpha emission or spontaneous fission) except ⁹⁰ Sr and others noted above.	5000 dis/min-100 cm ² βγ	15,000 dis/min-100 cm ² βγ	1000 dis/min-100 cm ² βγ

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TABLE 3

ACCEPTABLE SURFACE CONTAMINATION LEVELS

^aWhere surface contamination by both alpha and beta-gamma emitting nuclides exists, the limits established for alpha and beta-gamma emitting nuclides should apply independently.

^bAs 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.

^cMeasurements 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.

^dThe maximum contamination level applies to an area of not more than 100 cm².

^eThe amount of removable radioactive material per 100 cm² 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.

^fThe 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.

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IV.

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

(Excerpts have been retyped for purposes of this report)

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.

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

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.

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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)

	Inhalation	Ingestion		External Radiation	All Pathways ^c
		Home Gardener	Full Diet		
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 ²⁴¹Am and ²³⁹,²⁴⁰Pu are available from EPA recommendations, and a soil limit for ²²⁶Ra 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.

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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

Type of Exposure	Annual Dose Equivalent or Dose Commitment	
	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

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

APPENDIX 7

EVALUATION OF RADIATION EXPOSURESINTRODUCTIONA. 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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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 operating nuclear reactors and accelerators, or conducting 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 subsequently 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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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 hydrogen-3 (tritium), 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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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 the properties of 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 the radioactive material 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 sources 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

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fossil fuels (natural 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 an exposed person's 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 reports issued by the 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). 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

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is presently subject to large uncertainties and, therefore, potential continual revision. The risk 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 individual 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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