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RADIOLOGICAL ASSESSMENT REPORT
FOR
THE UNIVERSITY OF ROCHESTER ANNEX
400 ELMWOOD AVENUE
ROCHESTER, NEW YORK

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PREFACE AND EXECUTIVE SUMMARY

In the spring of 1943, the Manhattan Engineer District (MED) of the United States Army Corps of Engineers contracted with the University of Rochester to participate in efforts to produce an atomic bomb. The program at Rochester had two broad undertakings; one consisted of practical technical services and the other research. To house the program, B-Wing and C-Wing of the Annex Building at the University of Rochester Medical Center were constructed as were an underground tunnel from the Annex to the Medical Education Building, an underground vault, the Alpha Laboratory on the second floor of B-Wing and the low-level background facility. Major efforts in these facilities have been related to determination of the toxicities of radionuclides and a number of other substances. Research efforts using radionuclides are continuing at the university. All radionuclides with atomic numbers from 3 to 104 inclusive, as well as tritium, are licensed for use in the Annex Building at the University of Rochester.

At the request of the U.S. Department of Energy, a radiological assessment of the Annex Building and associated facilities was conducted by the Argonne National Laboratory Radiological Survey Group during April and May 1984, to determine the radiological condition of these buildings. The assessment included surveying the interiors of the building with four types of portable survey instruments; smearing for removable contamination; and air sampling to determine radon, thoron, actinon and tritium concentrations, and radon daughter Working Levels in several rooms on each floor of the Annex. Additionally, pressurized ion chambers were used to measure the ambient radiation field, both differential and integral with respect to time, at the 3-ft level; soil samples were collected and analyzed to determine the radionuclide concentrations within the first foot of soil; and samples taken from the drain lines, the storm drains, and the sewer systems were analyzed.

Contamination was found in 24 rooms, on the roofs of both B- and C-Wings and in the tunnel and vault. Surveying with portable instruments was complicated by the presence of radionuclides of various strengths in 19 rooms. These sources were used in conjunction with ongoing activities. Two rooms were inaccessible for survey due to the presence of stored furniture and equipment.

The measured concentrations of radionuclides in the air were below their respective criteria. All concentrations of radionuclides in the soil were at background levels.

Drain lines in B- and C-Wings were found to be contaminated with various radionuclides, including uranium, thorium, radium, and cesium. The sanitary sewer system was found to be contaminated with radioactive cesium, thorium, and uranium; the storm sewer system was found to be contaminated with radioactive uranium, radium, thorium, europium, silver, and cobalt.

The radiological assessment indicated that hoods and duct work in the Annex Building were contaminated in three instances. The incinerator in B-144 was contaminated with radioactive tin-indium, scandium, manganese, gadolinium, and zinc. Therefore, the flue attached to the incinerator must be considered contaminated. A blower on the roof of C-Wing was contaminated with natural thorium. The duct work attached to this blower appeared to be coming from C-12. The south vent on the B-Wing roof was contaminated. This vent appeared to be above B-301. The duct work connected to the contaminated blower and vent must also be assumed to be contaminated.

Although contamination was detected in the Annex Building, sewers, and at least part of the ventilation system, it is judged that there is no immediate radiological hazard.

The surveys were performed by the following Health Physics personnel of the Occupational Health and Safety Department, Argonne National Laboratory, Argonne, Illinois: W. H. Smith, C. M. Sholeen, K. F. Flynn, J. G. Ello, J. D. Thereon and D. W. Reilly in 1984; W. H. Smith, C. Boggs Mayes (now with DOE/CH), R. Rodriguez, and D. W. Reilly in 1981.

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INTRODUCTION

In the spring of 1943, the Manhattan Engineer District (MED) of the United States Army Corps of Engineers contracted with the University of Rochester to participate in efforts to produce an atomic bomb. The program at Rochester had two broad undertakings--one consisted of practical technical services and the other research. To house the program, construction began during the same year for the first and second floors of what is now known as B-Wing of the Annex Building to the University of Rochester Medical Center located at 400 Elmwood Avenue (see Fig. 1). The Medical Center is located at the southeastern edge of Rochester (see Fig. 2) on the Genesee River. The River Campus and the Medical Center are adjacent sections of the university campus (see Fig. 3). The A-Wing of the Annex had been built in 1942 by the Radiology Department and was equipped with a one million volt x-ray machine to examine castings for the armed forces (see Fig. 4).

The C-Wing was added in 1944 (see Fig. 5). After World War II, the U.S. Atomic Energy Commission (AEC) assumed responsibility for the contracts and facilities of the MED. Added during the period 1950 through 1964, were O-Wing of the Medical Education Building, an underground tunnel between C-Wing of the Annex and O-Wing of the Medical Education Building, the Alpha Laboratory on the second floor of B-Wing of the Annex, the third floor of B-Wing of the Annex, a low-level background facility adjacent to the Annex, an addition to C-Wing of the Annex, an underground storage vault in the tunnel, and OO-Wing of the Medical Education Building (see Fig. 4). Major efforts in these facilities have dealt with determination of the toxicities of radionuclides and a number of other substances.¹

The Alpha Laboratory in B-Wing was built in 1951 for studies of the radiotoxicology of ^{222}Rn , ^{210}Po , ^{239}Pu , and other alpha-emitting radionuclides. It was designed to protect investigators from the hazards of these elements. It was found that complete protection was afforded even when curie amounts of the radionuclides were used.²

In 1972, the Alpha Laboratory was partially decommissioned. By 1979, the decommissioning and decontamination activities were complete. The U.S. Department of Energy, Chicago Operations Office (DOE/CH) contracted with the Argonne National Laboratory (ANL) Radiological Survey Group (RSG) to conduct a radiological assessment of the laboratory. That survey was performed in June 1981.

Research efforts using radionuclides are continuing in the B- and C-Wings of the Annex. The university is licensed by the New York State Department of Health to have up to 3 curies of each of the isotopes with atomic numbers from 3 to 83 inclusive in any chemical or physical form. The total for these isotopes is not to exceed 40 curies. The license also permits up to 10 microcuries of each of the isotopes with atomic numbers from 84 to 104 inclusive in any chemical or physical form. The total for these isotopes is not to exceed 100 microcuries. In addition, the license permits larger amount of specific isotopes, various sources, and 40 curies of tritium in any chemical or physical form. Table 1 lists the isotopes that the university personnel indicated were in use at the annex during the 1984 survey; the list also includes isotopes found on the site or in samples during the survey and several of the isotopes studied in detail in previous years. Some of the isotopes studies in previous years were used in dramatic amounts (e.g. 11 Ci of ^{238}Pu on hand at one time).

The contract (DE-AC02-76EV03490) between the Department of Energy (DOE) and the University of Rochester is approaching its termination date. Under DOE regulations, a radiological assessment was necessary before deciding upon the disposition of the B- and C-Wings of the Annex, the tunnel, the vault and several buildings adjacent to the Annex. The ANL/RSG conducted the comprehensive radiological assessment during the period April 23, 1984, through May 25, 1984. The property surveyed included B- and C-Wings of the Annex, the low-level background facility, the garage, a metal-frame building adjacent to the Annex, the tunnel and vault below Elmwood Avenue, and the sewers and soil adjacent to the Annex (see Fig. 4). The results and conclusions based upon this survey are included in this report. Since the rooms on the second floor of B-Wing that constituted the former Alpha Laboratory were resurveyed at this time, the results of the 1981 survey of that facility are not included in this report.

RADIOLOGICAL SURVEY PROCEDURES

General

The comprehensive radiological assessment of B- and C-Wings of the the Annex Building and the adjacent buildings included six separate but related activities. The primary activity was the direct survey of the floors, walls, and overheads with portable instruments. The location and frequency of the remaining survey activities were determined on the basis of the results of the direct surveys and information from university personnel. Two types of smear surveys were performed, one using popcorn (expanded polystyrene) for low-energy beta emitters and a second using filter paper for all other contaminants. Three kinds of air samples were collected--the first to determine concentrations of radon daughters, thoron daughters and long-lived radionuclides; the second to determine the presence or absence of actinon daughters; and the third to determine the concentration of tritium. Water and sludge samples were collected from exterior sewers and interior drains. Soil samples were collected from the immediate vicinity and from remote undisturbed locations (to determine the background concentrations of radionuclides in local soil). Finally, the contamination on selected samples and items was identified by gamma spectral analysis. More details of these procedures and the instruments used are given below and in Appendix 1. The appendices include generic information on instruments and procedures that were required for this particular survey and for other radiological surveys.

Instrumentation

Four types of portable survey instruments were used to conduct the direct radiological surveys: (1) gas-flow proportional detectors with window areas of 100 cm² and 325 cm² (using Eberline PAC-4G-3 electronics) were used to monitor for alpha and/or beta-gamma radiation; (2) NaI crystal detectors, 5 cm (2 in)* in diameter by 2 mm thick (Eberline PG-2 with Eberline PRM-5-3 electronics), were used to monitor for low-energy x-ray and gamma radiation;

*All measurements were originally taken using the English system. The Systems International (SI) units are to the nearest approximation.

(3) NaI crystal detectors, measuring 2.5 cm (1 in) in diameter by 2.5 cm (1 in) thick (Eberline PRM-7 μ R meter) and calibrated with a ^{226}Ra standard source, were used to measure the ambient external penetrating radiation field in units of $\mu\text{R/h}$; and (4) an end-window Geiger-Mueller (GM) detector (Eberline HP-190 with a 7 mg/cm^2 window and Eberline 530 electronics), calibrated with a ^{226}Ra standard source, was used to measure the contact exposure rate (mR/h) of contaminated areas. All accessible floor areas were surveyed with the first three instruments. The walls were surveyed to a height of 2 m (7 ft) with the first two instruments. Overhead areas were surveyed on a random biased basis (biased to areas where contamination was deemed most likely to exist). Integrated measurements of the ambient penetrating radiation field (μR) were taken in selected locations with a Reuter-Stokes RSS-111 pressurized ionization chamber calibrated with an NBS-traceable ^{60}Co source. These instruments and associated calibration procedures are detailed in Appendices 1 and 2.

Contaminants were identified by gamma-spectral analyses of a blower, a sample of ash taken from the contaminated furnace, and the bench tops with high activity levels. These analyses were performed with a sodium iodide (NaI) or HyperPure Germanium (HPGe) detector coupled to a multichannel analyzer. This instrumentation is also described in Appendix 1.

Smear Surveys

- Filter Paper - Dry smears were taken at representative locations throughout each building with 4.25-cm-diameter filter papers (Whatman #1). A standard smear sample is obtained by applying moderate pressure with the tips of the first two fingers to the back of the filter paper and wiping the surface over an area of approximately 900 cm^2 . Smears were taken on original structures and components such as walls, floors, pipes, and vents. A smear of 100 cm^2 was taken from any area or object indicated by a portable survey instrument to have a higher than normal radiation level. A smear of 100 cm^2 was also taken if the surface was extremely dusty.

To expedite the counting of the numerous smear samples collected, two counting techniques were employed with two types of counters. A large-

area, thin-window, gas-flow proportional counter, sensitive to alpha and/or beta-gamma radiation, was used to make an initial count on groups of smears. For confirmatory counts on individual smears noted to be above the expected background level, a Nuclear Measurement Corporation Model PC-5 or 3A, internal gas-flow proportional counter (PC counter) with a thin aluminized Mylar window (referred to as Mylar spun top) was used.

Initial counts were made with the large-area counter on groups of ten smears at a time. Smears from any group indicating a reading above the instrument background were then counted individually in the PC counter. In addition, at least two smears randomly selected from each group of ten were counted in the PC counter. All smears of the areas or objects with elevated direct readings were counted individually in the PC counter. More detailed descriptions of the counters and the counting and calibration techniques are presented in Appendix 1.

Popcorn - As indicated in Table 1, tritium (^3H) had been used in some areas of the Annex. Tritium emits a low-energy beta particle (0.018 MeV). This low energy beta particle does not penetrate the mylar spun top of the PC counter. Therefore, popcorn (expanded polystyrene that resembles peanut shells) smears were taken at representative locations in the rooms that were designated for use of tritium. The popcorn was rubbed gently over an area of 100 cm^2 . The popcorn smear was dissolved in a Beckman Ready-Solv High Performance premix liquid scintillation cocktail. The vial containing the dissolved popcorn was counted in a liquid scintillation counter for four minutes. One window of the counter was adjusted for carbon-14, the other for tritium. Four-minute counts were taken of each vial, and four-minute background counts were taken of one vial containing clean popcorn dissolved in the scintillation cocktail and of another vial containing scintillation cocktail with nothing dissolved in it. (See Appendix 2 for conversion from cts/min to dis/min.)

Air Samples

- Radon Daughters and Long-Lived Activity - Air-particulate samples were collected using a commercial vacuum cleaner modified by ANL in order that it could be utilized as an air sampler by pulling air through filter medium (Hollingsworth-Vose HV-70 or LB-5211). A total volume of 26.7 m³ of air was sampled at a flow rate of 40 m³/h. A 10% portion (5 cm in diameter) of the filter medium was removed after collection and counted for both alpha and beta-gamma activity in a Nuclear Measurements Corporation PC counter. Concentrations of radon (²²²Rn) daughters, thoron (²²⁰Rn) daughters, and the presence of any long-lived airborne radionuclides were determined based on the result of several counts of each sample at specified intervals.

Air-particulate samples were also collected on Millipore membrane filter medium for 90 minutes at a flow rate of approximately 1 m³/h. A portion of each filter sample was used for alpha spectral analysis to determine the actinon (²¹⁹Rn) daughter to radon (²²²Rn) daughter ratio.

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

- Tritium - Airborne tritiated water vapor was collected using a cold trap method with a liquid scintillation counter (Nuclear Enterprises LSC-2). Air was drawn through a cold trap immersed in a dry ice methanol bath. The collection rate was approximately 1 m³/h for two to three hours. Water vapor from the air condensed in the cold trap. During sample collection, the relative humidity of the air was measured using a sling psychrometer. The wet bulb and dry bulb readings were recorded, and the amount of water in dry air (g/m³) was determined.³ A 1-ml aliquot of the condensate was mixed thoroughly with 10 ml of Beckman Ready-Solv (scintillation cocktail) in a 20-ml sample-counting vial. The sample was counted for four minutes in a liquid scintillation counter. Four-minute background counts were also taken of one vial containing only cocktail and of another vial containing distilled

water dissolved in the cocktail. Details of the tritium-sampling techniques and associated calculations are given in Appendix 3.

Miscellaneous Samples

Several samples were collected from items that indicated elevated levels of radioactivity with the portable survey instruments. An ash sample was removed from the incinerator in Room B-144 and three samples of bench top material were removed from two lab benches in Room B-301. In order to determine the relative radioactive contents of this bench top material, the university personnel supplied a piece of similar bench top material that had not been exposed to radioactive contamination.

The ash sample was ground with a mortar and pestle and sieved through a No. 30 standard testing sieve. The incinerator in B-144 is used to cremate the bodies of animals used in the experiments with radioactivity. Therefore, both the material identified as "sieved" and that identified as "rocks and dross" for the first entry in Table 2 were expected to contain the radioactive contaminants. A 5-g sample of the sieved material was prepared for radiochemical (fluorometric) analysis; a 100-g sample of the "rocks and dross" was prepared for gamma-spectral analysis.

A piece of the background bench top sample and the three samples from the benches in Room B-301 were sent in their entirety to the Analytical Chemistry Laboratory at Argonne National Laboratory for preparation and analysis.

Sewer and Drain Samples

Both storm and sanitary sewer systems were sampled from exterior access points. Since the present storm sewer system at one time served as the sanitary sewer system, it was necessary to sample both systems. Samples of sludge and water were removed from seven access points, and a water sample was removed from an additional point. Depending upon the amount of water at the access points, either 0.9 l (1 quart) or 4 l (1 gallon) of liquid were collected. The drains in four rooms and one hall drain also were sampled. Sludge and/or 0.9 l of water were removed from all drainline access points.

The sludge samples were sent in their entirety to the Analytical Chemistry Laboratory. These samples were dried and the dried residue was analyzed. The water samples were sent in their "as collected" state to the ANL Analytical Chemistry Laboratory. The suspended solids were filtered and the filtrate was evaporated to ascertain the dissolved solids content. The volume and weight of each sample are listed in Table 3.

Each fraction of the water sample, as well as the sludge solids, was analyzed for uranium (uranium fluorometric) and ^{226}Ra and ^{232}Th decay chains (gamma spectrometric). The sample identification numbers have three parts: (1) the site number (15), (2) the sample type (SS for sewer or drain sludge and SW for sewer or drain water), and (3) a consecutive number (4 through 18 for storm and sewer samples from exterior access points and 27 through 33 for drain samples from interior access points).

Soil Corings

Environmental soil samples (4-in-diameter by 12-in-deep corings) were taken from four selected locations near the Annex. Corings were taken at four additional sites away from the Annex (see Fig. 3) to determine background levels of radionuclides in the soils of the area.

The samples were collected using a 4-in-diameter, 6-in-long right-circular-cylinder cutting tool, commonly used to cut holes on golf course greens. Each soil core was taken in four segments for analysis. Starting from the surface, three 2-in segments were cut, bagged, and marked A, B, and C, respectively; the final 6-in segment was marked D. 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 the contaminants (or vice versa), and to reveal if any overburden or backfill had been added.

The soil samples were prepared for analyses by weighing each sample in its "as collected" state, drying it for approximately 48 hours at 80°C, and then reweighing the sample to determine dry weight. Each sample was then put into a mill jar (8.7 l) and milled for up to two hours. At no time were any rocks in these samples crushed, ground, or pulverized, since this would act to dilute, and hence lower, the reported concentrations of deposited

radioactive material in the soil. After sufficient milling, the sample material was sieved. Each fraction (rocks and dross vs. fines) was bagged and weighed separately (see Fig. 6). The weights of the samples are given in Table 2.

Weighed aliquots of the milled material from the soil samples were loaded into screwtop plastic containers; aliquots of 100 g, if available, were prepared for gamma-spectral analysis, and aliquots of 5 g for radiochemical (fluorometric) analysis. The soil samples were sent to the ANL Analytical Chemistry Laboratory for analysis. Every effort was made during sample preparation to prevent cross-contamination. Soil samples suspected of containing elevated levels of radioactivity were processed in equipment separate from soil samples considered to contain background levels. All processing equipment was scrubbed and air dried before introduction of the next sample.

The soil sample identification numbers have four parts: (1) the site number (15), (2) the sample-type identification (S for soil or SB for background soil), (3) a consecutive number (19 through 22 for background environmental soil samples and 23 through 26 for environmental soil samples), and (4) a depth indicator (A through D, as defined above).

ANALYSIS OF SURVEY RESULTS

General

Areas with elevated levels of radioactivity were found in 24 rooms, the roofs of B- and C-Wings, the tunnel, and the vault. In nine rooms, the activity was removable. In addition, radioactive sources used in the ongoing work with radioactive material were found in 19 rooms. Tritium was detected in three locations. Contamination was also found in the sewer and drain lines.

Instrument and Smear Survey

The measurements recorded with the gas-flow proportional-counter survey instruments were converted to surface contamination measurements according

to the following general procedures (also see Appendix 2). For gross readings taken in the beta mode, background and any alpha contribution were subtracted to determine the net beta-gamma count rate. The net count rate was then converted to disintegrations per minute (dis/min) by equating to normal uranium and normalized to a surface area of 100 cm². After subtraction of background, readings in the alpha mode also were converted to dis/min-100 cm² as equated to normal uranium. Two survey instruments were used. The first, identified as the floor monitor (FM), had a 325 cm² probe. The second, identified as a hand probe (HP), had a 100 cm² probe. 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.

Low-energy x-ray and gamma exposures were measured with the PG-2 detector attached to the PRM-5-3 electronics. The results are reported in counts per minute (cts/min) and include the instrument background of 400 to 2000 cts/min. The GM detector and μ R meter exposure-rate measurements include the instrument backgrounds of 0.01 - 0.03 mR/h and 3 - 15 μ R/h, respectively. The results of the instrument surveys are presented in Table 4.

Smear samples were counted for both alpha and beta-gamma activity, and the net count rates are converted to dis/min-100 cm² after subtracting the appropriate background. An integrated measurement of the ambient radiation field at select interior locations was taken with a pressurized ion chamber.

The instrument survey data and the smear results were reviewed with respect to the ANSI Draft Standard N13.12, "Control of Radioactive Surface Contamination of Facilities To Be Released for Uncontrolled Use"; the NRC's "Guidelines for Decontamination of Facilities and Equipment Prior To Release for Unrestricted Use or Termination of Licenses For By-Product, Source or Special Nuclear Material"; "The Surgeon General's Guidelines" as included in 10 CFR Part 712 Grand Junction Remedial Action Criteria; and 40 CFR 192 "Health and Environmental Protection Standards for Uranium Mill Tailings" (see Appendix 6).

The results of the surveys for specific areas are summarized in the following paragraphs.

C-Wing - All the rooms and halls in C-Wing were surveyed and smeared. The survey results are presented in Table 4 and the smear locations are shown in Figures 7 and 8. As shown in columns 2 and 3 of Table 4, the percent of the floor and wall areas accessible for survey varied from 10% to 100%. Contamination was found in ten of the rooms (see Table 5). The contaminated areas were all above criteria for unidentified radionuclides, with the possible exception of the contamination at location 104 under the wall between Rooms C-12 and C-16. The highest beta-gamma activity ($77 \text{ k dis/min-100 cm}^2$) was found in the pump room of Room C-12. The oily dirt in this area masked the alpha contamination that was later detected on the smear taken there. The highest alpha activity ($40 \text{ k dis/min-100 cm}^2$) was found on the floor of C-65. The greatest ambient external penetrating radiation field ($100 \text{ } \mu\text{R/h}$) was produced by a contaminated area on the floor of Room C-40. The highest low-energy x-ray and gamma radiation measurement (45 k cts/min) was detected in Room C-43 at Location 171. No loose activity was detected there.

The survey of some of the areas in C-Wing was complicated by the presence of radioactive sources used in current activities (see Fig. 9) in six rooms. The largest source was a $6000\text{-Ci } ^{60}\text{Co}$ source in Room C-34 (see Table 6). The radiation from this source was detected through the walls in the hall and the adjacent room. Radiation from ^{125}I sources in Room C-23 was also detected in the hall through the wall. After lead shielding was placed near the wall below the sources to supplement the lead above and on the sides of the sources, the hall activity dropped to background levels.

The roof of C-Wing was also surveyed. As seen in Figure 10, there were many ventilation stacks. All were surveyed with the portable instruments and ten were selected at random for a smear survey (see Fig. 11). Two areas of the roof had radioactivity slightly above background (see Table 5), though apparently below criteria. There was also an exhaust vent (see Fig. 12) that was contaminated. The contamination was identified as the natural thorium-232 decay chain. This identification

was based on the gamma-ray spectrum of the exhaust pipe measured with a portable ND-6 gamma spectrometer. The interior area of this vent was not accessible for smears. The vent appeared to be coming from Room C-12 or C-16. The contamination in this vent was above criteria for uncontrolled use.

B-Wing First Floor - The survey results are presented in Table 4 and the smear locations are shown in Figures 13 and 14. Two rooms had contaminated areas (see Table 5). There was a wall area in Room B-135 with relatively high levels of radioactivity (360 k dis/min-100 cm² beta-gamma, 270 k dis/min-100 cm² alpha) that was also removable (350 dis/min-100 cm² alpha and 440 dis/min-100 cm² beta-gamma). The furnace in Room C-144 (see Fig. 15) is used to cremate the bodies of laboratory animals used in continuing experiments to determine toxicities of radionuclides and other substances. There was a relatively high radiation field from a can of ashes removed from the furnace. When the can was moved from B-144 to B-146, a survey of the room became possible. The furnace was found to be contaminated (see Table 5).

Two radioactive sources were also found on this floor (see Table 6 and Figure 16). The ⁶⁰Co in B-113 could also be detected with the survey instruments in the hall and in Room B-115 (see Table 4). Unidentified sources were also found in a file cabinet in Room B-104.

B-Wing Second Floor - The survey results are presented in Table 4 and the survey locations are shown in Figures 17 and 18. Contaminated areas were found in eight rooms on floors, walls, and overhead light fixtures. The contamination levels in these areas were above criteria for uncontrolled use. The maximum activity came from a wall area in B-204 which had 71 k dis/min-100 cm² beta-gamma and 25 k cts/min low energy x and gamma-rays. When removable beta-gamma activity was found on the overhead smear from B-201, a more thorough survey of the overhead fixtures was performed; the results are shown in Table 5.

Five rooms on the second floor had radioactive sources (see Table 6 and

Figure 19). Most of the sources were unidentified, but x and gamma radiation was easily detected with the portable survey instruments.

• B-Wing Third Floor - The survey results are presented in Table 4 and the smear locations are shown in Figures 20 and 21. Radioactive contamination was found in four rooms. The bench top in B-301 exhibited relatively high levels of radioactivity. The gamma-ray spectrum measured with the portable ND-6 gamma spectrometer indicated that the radioactivity was the natural uranium and natural thorium decay chains. The top of the material was coated with a resin that fixed the activity; however, the bottom of the material was not coated and had removable activity (see Table 5). This radioactivity did not appear to be the result of a spill, but instead the result of concentrations of uranium and thorium in materials used to make the benchtop. There were areas with floor contamination in Room B-335, up to 480 k dis/min-100 cm² beta-gamma and 30 k cts/min low energy x and gamma radiation. The highest level of loose activity was 5500 dis/min-100 cm² beta-gamma.

Twenty-four rooms on the third floor were isolated from the rest of the facilities (see Fig. 21), university personnel requested that the ANL survey team not take smears in the area. However, all areas were surveyed with the instruments and found to exhibit only background levels of radioactivity.

Radioactive sources associated with ongoing activities were found in four rooms (see Table 6 and Figure 22). There were a variety of radioactive sources present that were easily detected. These sources contributed to the ambient background levels for other areas within these four rooms, as well as in adjacent rooms. Also, a monkey cage in Room B-328A was contaminated with unidentified beta and gamma radioactivity (see Table 5).

Rooms B-330 and B-332 could not be surveyed because of general clutter making it inaccessible. Therefore, no statement can be made regarding the radiological condition of these rooms.

• B-Wing Roof - There are four structures on the roof of B-Wing (see Fig. 23). The storage room (see Fig. 24) is a long, metal building with no radioactive contamination or sources. The roof of this room was not surveyed. The fan and storage room is a brick structure (see Fig. 1) that had no radioactive contamination or sources. The roof of this room was surveyed, and three of the vents were smeared (see Figs. 23 and 25). The roof above the north-south corridor of B-Wing (see Fig. 26) was surveyed and the three vents were smeared. Activity was detected on the south vent (5 k dis/min-100 cm² beta-gamma, see Table 5), but no removable activity was found. The origin of this vent stack was not determined, but it appeared to be above Room B-301. All other areas of the roof structures had background levels of radioactivity.

A member of the survey team climbed the ladder up the incinerator stack (see Fig. 24) to take smears of the exhausts. No loose contamination was detected. Because of the precarious nature of stack access, no survey of the stack was possible with the portable instruments. This stack appears to serve only the incinerator in Room B-144 (see Fig. 27).

• Tunnel and Vault - The stairs at the east end of C-Wing (see Fig. 9) lead to an underground tunnel between the Annex and O-Wing of the Medical Education Building (see Fig. 4). This tunnel was surveyed and the results are presented in Table 4. The smear locations are presented in Figure 20. Removable contamination was found at the floor/wall junction in the tunnel. The elevator, stairs, and store room at the entrance to the tunnel from C-Wing all exhibited background levels of radioactivity.

A storage vault is located in the middle of the tunnel (see Fig. 28). All radioactive sources were removed from the vault before it was surveyed. Twelve areas with elevated levels of radioactive contamination were detected in the vault (see Fig. 29). All levels of contamination were above the criteria for unidentified radionuclides. There was a concrete block shield in the northwestern corner of the vault. One of the bricks on the shield had 600 k dis/min-100 cm²

beta-gamma activity, 500 dis/min-100 cm² alpha activity, and 150 k cts/min low-energy x and gamma radiation. Removable activity was found at four locations on the floor (see Table 5).

- Auxiliary Buildings and Areas - Two buildings north of A-Wing (a metal-frame building and the low-level background facility) and two buildings (a garage and a pump house), a yard, and a fire escape north of B-Wing were also surveyed (see Fig. 30). The survey results are presented in Table 4 and the smear locations are presented in Figure 31. The presence of radioactive sources in the low-level background facility complicated the survey. These sources were in lead shields and the readings presented in Tables 4 and 6 are those measured through the shielding. The survey of the roof of the garage east of the low-level background facility was complicated by the radioactive chemicals on the window sill in Room B-113 of B-Wing. No residual contamination was detected in any of these buildings or areas above the local background encountered.

Radon Air Samples and Ambient External Radiation

A total of 25 "grab" air samples were collected in B- and C-Wings, the vault, the low-level background facility, and the fan/storage room on the B-Wing roof. The measured radon (²²²Rn) concentrations, thoron (²²⁰Rn) concentrations, and the milliWorking Level (mWL) determinations are presented in Table 7; the sample collection locations are identified in the figures as indicated in column 3 of the table. The detailed calculations used in evaluating the air samples are given in Appendix 3. The air samples revealed radon daughter levels from less than 0.06 to 14.6 mWL. These are all below the 20-mWL limit recommended in 40 CFR 192, "Health and Environmental Protection Standards for Uranium Mill Tailings," as the maximum annual average. The radon (²²²Rn) concentrations of less than 0.006 to 1.46 pCi/l are all below the 3-pCi/l limit for uncontrolled areas as stipulated in DOE 5480.1, Chapter XI (see Appendix 6); and the thoron (²²⁰Rn) concentrations of less than 0.8 to 9.5 fCi/l are also below the 10⁴-fCi/l limit for uncontrolled areas as stipulated in the same document. These levels are also below the 600 fCi/l limit for the thoron daughter

^{212}Pb for uncontrolled areas. These very low levels indicate that the porous surfaces of the concrete walls and floors are sealed and the air filtering system is good. No long-lived radioactivity was detected on any air sample.

A second air sample was collected on Millipore membrane filter paper in each of ten rooms and the vault (see Table 7, column 4). Alpha spectroscopy performed on these samples did not reveal any actinon (^{219}Rn) daughter activity on filter media.

The ambient external penetrating radiation levels were measured in five locations (see Table 8). In all cases, the ambient external gamma radiation levels were below the 20 $\mu\text{R/h}$ above background standard set in 40 CFR 192.12 for occupied buildings (see Appendix 6).

Tritium Survey

Personnel from the University of Rochester identified the rooms in which tritium was being used at the time of the survey. There were four rooms in C-Wing and two rooms on the second floor of B-Wing. Popcorn smears were taken in these rooms, and airborne water vapor was condensed in a cold trap for measurement of the tritium in the air.

- Popcorn Smears - No elevated activity was detected on any of the popcorn smears collected from the four rooms in C-Wing (see Fig. 32) or from the two rooms on the second floor of B-Wing (see Fig. 33). Smears were also taken in the vault (see Fig. 34). Two of the smears were taken from floor areas that were found to have elevated levels of radioactivity on the portable survey instruments. The popcorn smear 1039 and the filter paper smear 359 were taken from the same area. Neither smear indicated removable activity. The popcorn smear 1040 and the filter paper smear 355 were taken from the same area (see also Fig. 29 and Table 5). Smear 355 indicated 114 dis/min-100 cm^2 beta-gamma in the PC counter and the two windows on the liquid scintillation counter also indicated beta contamination. This activity was not identified. No other popcorn smear had any detectable activity.

Air Samples - Water vapor was condensed out of the air to measure the tritiated water vapor. Eight samples were collected; one in each room in which smears were taken (see Figs. 32 and 33), one in the vault, and another in the tunnel (see Fig. 34). Two rooms, C-23 and B-242, had detectable levels of tritium in the air (see Table 9). The concentration of tritium in the vault air sample, 0.85 pCi/l, was easily detected; therefore, an air sample was also collected in the tunnel downwind from the vault. It showed a lower level of tritium in the air, 0.30 pCi/l. All measured tritium concentrations were well below the 200-pCi/l limit for uncontrolled areas as stipulated in DOE 5480.1 Chapter XI.

Soil Samples

Environmental soil samples were taken from four undisturbed areas of the campus about one-half mile away from the Annex (see Fig. 3)--two from an area to the north (samples 15-SB19 and 15-SB20), and two from an area to the south (samples 15-SB21 and 15-SB22). The sample segments were used to determine the background concentrations of radionuclides in the soil at the University of Rochester (see Table 10).

Other environmental soil samples were taken from four locations in the immediate vicinity of the Annex (see Fig. 35). Soil in the area immediately north of the Annex could not be sampled because of the presence of asphalt pavement. Analyses indicated only background concentrations of radionuclides in all the soil sample segments (see Table 10).

Water and Sludge Samples

Fifteen sludge and water samples were removed from two access points of the storm sewer system and from six access points of the sanitary sewer system (see Fig. 35). Sludge sample 15-SS12 from the storm sewer line contained slightly elevated levels of uranium (4.4 pCi/g). There were elevated levels of radium (10.5 pCi/l) in the dissolved solids and uranium (37.1 pCi/g) in the suspended solids of one water sample (15-SW6) from the storm sewer line. There were elevated levels of the thorium-232 decay chain (4.22 pCi/g), europium-152 (5.01 pCi/g), silver-110m (0.93 pCi/g), cobalt-60

(0.77 pCi/g), and uranium (186 pCi/g) in the suspended solids from the other water sample (15-SW11).

Two of the six sludge samples from the sanitary sewer system had elevated concentrations of the radium-226 decay chain and uranium. Three of the six water samples had elevated concentrations of cesium-137 (up to 27 pCi/l) and/or the thorium-232 decay chain (up to 18.3 pCi/l) in the dissolved solids. One water sample had an elevated concentration of the radium-226 decay chain (5.19 pCi/g) in the suspended solids, and three had elevated concentrations of uranium (up to 13 pCi/g) in the suspended solids (see Table 10). Although these concentrations are below the water concentrations stipulated in DOE-5480.1 Chapter XI for uncontrolled areas, they are an indication that both sewer systems are contaminated, and according to the NRC Guidelines, must be assumed to be in excess of the limits for uncontrolled use. The measured concentrations of radionuclides in the dissolved solids from the storm and sanitary sewer systems are from either current ongoing contamination and/or the redissolution of contamination that had previously settled in the sewer systems.

Seven sludge and water samples were removed from five access points of the indoor drain lines. Uranium is known to have been used in Room C-12, and the sludge (15-SS27) and water (15-SS28) samples from the drain in this room (see Fig. 9) contained high levels of uranium (up to 1083 pCi/g, see Table 10). Each of the other five samples also had elevated concentrations of cesium-137, thorium-232 decay chain, and/or radium-226 decay chain. A uranium concentration of 1130 pCi/g was measured in sludge sample 15-SS31, which was removed from the drain in the center of Room B-202 (see Fig. 19). All the water samples (15-SW29 through 15-SW33) from the indoor drains had elevated concentrations of uranium (up to 61.6 pCi/g in the suspended solids and up to 19 pCi/l in the dissolved solids). These samples are located in Figures 9, 16, and 19.

Miscellaneous Samples

During the portable instrument survey, relatively high levels of radioactivity were detected in the ashes that had been removed from the incinerator in Room B-144 (see Fig. 16). The ashes (from the cremation of animals used in toxicity studies) were found to contain concentrations of

tin-113/indium-113, scandium-46, manganese-54, gadolinium-153 and zinc-65 (see Table 10).

High levels of radioactivity were also measured in the bench top material in Room B-301 (see Fig. 2.2 and Table 5). Two samples were obtained from university personnel during the survey. One was from the corner of a bench in Room B-301 that exhibited no activity above background (15-Benchtop-3), the other (15-Benchtop-2), was a piece of the same kind of material but with lower levels of radiation (considered background material). Later, the university personnel sent two additional samples (drill corings and fines), to ANL, one from the lab bench on the north wall of the room and the other from the lab bench at the east end of the room. Analysis by the Analytical Chemistry Laboratory of ANL indicated the concentrations of uranium were the same in all samples (see Table 10). The concentrations of the throrium-232 decay chain and the radium-226 decay chain in 15-Benchtop-34 and 15-Benchtop-35, were above the background levels but still very low. The PAC-4G-3 was also used to measure the activity of these samples before they were sent for analysis. The beta-gamma activities were 200 cts/min-100 cm² for 15-Benchtop-3, 50 cts/min-100 cm² for 15-Benchtop-34 and 100 cts/min-100 cm² for 15-Benchtop-35, above background. The alpha activity was background for all of the samples. These measured activities of the samples are all significantly below the 18 k cts/min-100 cm² beta-gamma and the 5 k cts/min-100 cm² alpha measured on the bottom side of the benchtop along the north wall during the radiological assessment of Room B-301.

Hoods and Duct Work

In three instances, surveys indicated contamination of the hoods and ducts. The first instance was the duct work attached to the incinerator in Room B-144. The portable instrument survey had revealed that the ashes from the incinerator and the inside of the incinerator contained radioactive material (see Table 5). The ash sample from the incinerator contained four radionuclides with concentrations ranging from 24 to 1400 pCi/g (see Table 10). Therefore, the incinerator must be considered as contaminated throughout, and the flues attached to the incinerator must also be considered as contaminated.

Secondly, it is known that radionuclides were used and stored in the several hoods in Room C-12 in C-Wing. Also, a contaminated blower was discovered during the portable instrument survey of the C-Wing roof. The ductwork attached to the blower appeared to be coming from Room C-12. Therefore, the hoods and ducts from C-12 must all be considered as being potentially contaminated.

Finally, the south vent on B-Wing was found to be contaminated during the survey of the B-Wing roof. The vent is approximately above Room B-301. The hoods and ducts connected to this vent must, therefore, also be considered potentially contaminated.

Although only three instances of contamination were found in the survey of the accessible points of the ventilation system, all the hoods and ductwork in B- and C-Wings must be considered as being potentially contaminated.

During the decontamination of the Alpha Laboratory on the second floor of B-Wing, radioactivity was found throughout the length of the Low-Risk Ventilation System, although none was found at the elbow or at the intake.² This same phenomenon has been found at ANL during decontamination and should be expected to occur in the ducts in the B- and C-Wings of the Annex.

ESTIMATED EXTENT OF CONTAMINATION

Any estimate of the total mass and volume of radioactively contaminated material that would be generated if remedial action were undertaken at the Annex is subject to many uncertainties. For example, one can only surmise as to the actual depth of contamination within concrete and brick. For the purposes of this report, it has been assumed that contamination on concrete, brick, or asphalt will require removal to a depth of 5 cm (2 in); contamination on wooden floors and steel will require removal to 2 cm (3/4 in), the standard thickness of such floors. These assumptions are believed to be conservative.

Estimates of the total activity of contaminated material are likewise subject to uncertainties because of survey limitations. Unless otherwise known, all readings of dis/min-100 cm² (as reported in Table 5) are equated to normal uranium infinitely-thin flat-plate standards. No corrections can

be accurately made for absorption by surface media and, hence, levels of activity in surface media are likely an underestimate.

Within the framework of these uncertainties and limitations, estimates of volume, mass, and activity for each type of material have been made and are presented in Table 11. The total would consist of an estimated 0.54 m^3 of material with a mass of 1200 kg and an activity of 48 μCi . This estimate includes only the contamination on floor and wall surfaces. However, the portable instrument survey and the analyses of drain samples have demonstrated that the drain lines are also contaminated. As indicated above, some of the hoods and ducts are contaminated. The incinerator in Room B-144 is contaminated, as are light fixtures in Room B-201 and a monkey cage in Room B-328A. The values of the volume, mass, and activity in Table 11 do not include estimates connected with any of this contamination. As indicated in Rooms C-12 and C-16, where low-energy x- and gamma-ray activities was detected at the common baseboard, modifications in the interior structure of the building have covered early contamination. In Room C-40, the contamination appeared to be below the tile. The extent of contamination in these two areas and of any contamination in other areas that was masked by construction modifications has not been estimated.

Finally, the activity in the bench tops in Room B-301 and B-316 has been considered as coming from elevated activity in the natural materials used to construct the bench top and are not included in Table 11.

DOSE AND POTENTIAL HAZARD EVALUATION

External Exposure

To assess the radiological hazard from external exposure to the radiation sources, a "conservative" (or worst-case) situation was assumed. Since commercial, rather than residential occupancy is involved, it was assumed that an individual would be exposed 40 hours per week to the maximum interior radiation levels.

The maximum interior radiation level observed was 3000 $\mu\text{R/h}$ in Room B-144 near the waste can containing ashes from the incinerator. The annual radiation dose from this source would be:

$$3000 \mu\text{R/h} \times 40 \text{ h/w} \times 50 \text{ w/y} \times 1 \text{ rem/R} = 6 \text{ rem/y.}$$

This radiation level changes when the can of ashes is moved. Additionally, the area is clearly unoccupied for large periods of time. Therefore, it is the judgement of the survey team that the contaminated areas of this site do not constitute an immediate radiological hazard in terms of external exposure, even though the worst-case dose (a seemingly unrealistic situation) is above the DOE 5480.1 Chapter XI limit of 5 rem per year for a person occupationally exposed (see Appendix 6).

Internal Exposure

To assess the potential for radiological hazard based on potential internal exposure, it was necessary to assume "conservative" but nevertheless plausible scenario whereby the radioactive contamination was assimilated internally. A hypothetical, yet conceivable, worst-case situation involving Room B-135 was developed. The maximum reading detected at Location 299 on the brick wall in Room B-135 was 3.6×10^5 dis/min-100 cm² beta-gamma equated to normal uranium. This equates to 1.62×10^5 pCi/100 cm² (2.22 dis/min-pCi).

The worst likely situation that could arise would probably involve the use of a concrete saw in a dry mode of operation. It will be assumed that a person is using a portable saw of this type to cut the wall for an electrical outlet. A typical saw blade used in cutting masonry would have a 30.5-cm (12-in) diameter, a 0.32-cm (1/8-in) thickness, and a 11.1-cm (4 3/8-in) cutting depth. The wobble of the blade would probably increase the width of the cut to a kerf of 0.40 cm. If the cut were 10 cm x 10 cm (4 in x 4 in), the perimeter would then be 41 cm (16 in). The overlap of the cuts upon the brick wall would extend the total span of the cuts to 89 cm (35 in). The area of the brick wall that the blade would consume is then 36 cm². It is further assumed that 50% of the brick that is displaced becomes airborne and respirable. Room B-135 is 4.9 m x 4.0 m (16 ft x 13 ft) with 3.0 m (10 ft) ceilings and has a total volume of 59 m³. It is assumed that the dust created becomes dispersed throughout the entire room. Only one person would probably be involved in this operation, and the job should require less than an hour. Assuming a person would inhale 1.2 m³ of

air per hour⁴ and would be involved in this job for a one-hour period, a total of 0.37 cm² of the brick surface would be inhaled. Based on this scenario, the following hazard levels (50-year dose commitment for one year of intake)⁵ were calculated.

Adult inhalation of aerosol:
(units: mrem/pCi x pCi/cm² x cm² = mrem)

Normal uranium:	Bone:	(.0075)(1620)(.37) = 4.5 mrem
	Total Body:	(.015)(1620)(.37) = 9.0 mrem
	Lung:	(0.508)(1620)(.37) = 300 mrem

More details of these calculations are given in Appendix 8. There are no regulations that set an allowable limit for radionuclides with both long radiological half-lives and long biological half-lives such as the uranium, radium, and thorium that were found on this site. DOE Order 5480.1 Chapter XI, defines dose commitment as "the dose equivalent (rem) received by specific organs during a period of one calendar year, that was the result of uptakes of radionuclides by a person occupationally exposed." It also sets the radiation protection standard for external and internal exposure of individuals and population groups in uncontrolled areas at 0.17 rem to the whole body, gonads or bone marrow and 0.5 rem to other organs. If the external and inhalation doses were to continue at the levels stated above for 50 years, the dose commitments for a single year would not be exceeded by the 50th year.

CONCLUSIONS

In light of the results of the comprehensive radiological assessment of the Annex and auxiliary facilities, the following conclusions can be made:

1. There is no immediate hazard from the elevated levels of radioactivity detected; however, some of these levels are above criteria.
2. The radon, thoron, actinon, long-lived particulates, and tritium in the air are all below criteria for unrestricted use.

3. Some ductwork has been identified as being contaminated. All ductwork must, therefore, be considered potentially contaminated.
4. Since several floor drains were found to exhibit elevated readings, and the samples had elevated concentrations of radionuclides, it must be concluded that the drain and sewer systems of the Annex are contaminated with radioactive material.
5. Since the samples collected from the storm and sewer systems outside the building also had elevated concentrations of radionuclides, these systems are also considered contaminated with radioactive material.
6. The grounds around the Annex have exhibited background concentrations of radionuclides.
7. Two rooms, B-330 and B-332, were inaccessible for survey due to the presence of stored furniture and equipment. Therefore, no comment about their radiological status can be made.
8. At the common baseboard for Room C-12 and C-16 and on the floor below the tile in Room C-40, contamination appeared to be masked by construction modifications. Other areas of the Annex must also be considered potentially contaminated where modifications may have masked the contamination.

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Figure 1. B-Wing of the Annex.

Figure 2. City of Rochester.

Figure 3. University of Rochester.

Figure 4. Annex and Environs.

Figure 5. C-Wing of the Annex.

Figure 6. Environmental Soil Processing.

Figure 7. C-Wing West End Survey Locations.

Figure 8. C-Wing East End Survey Locations.

Figure 9 C-Wing Sample Locations.

Figure 10. C-Wing Roof.

Figure 11. C-Wing Roof Survey Locations.

Figure 12. Contaminated C-Wing Vent.

Figure 13. B-Wing First Floor North/South Corridor Survey Locations.

Figure 15. Furnace in B-144.

Figure 16. B-Wing First Floor Sample Locations.

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Figure 17. B-Wing Second Floor North/South Corridor Survey Locations.

Figure 18. B-Wing Second Floor East/West Corridor Survey Locations.

Figure 19. B-Wing Second Floor Sample Locations.

Figure 20. B-Wing Third Floor North/South Corridor Survey Locations.

Figure 21. B-Wing Third Floor East/West Corridor Survey Locations.

Figure 22. B-Wing Third Floor Sample Locations.

Figure 23. B-Wing Roof and Roof Structures Survey Locations

Figure 24. B-Wing Roof Storage Room.

Figure 25. Vents on the Fan and Storage Room Roof.

Figure 26. Vents and Solar Panels on B-Wing Roof.

Figure 27. B-Wing Showing Incinerator Stack from Lot 7.

Figure 28. Tunnel Survey Locations.

Figure 29 Vault Survey Locations.

Figure 30. Auxiliary Buildings.

Figure 31. Auxiliary Buildings Survey Locations.

Figure 32. C-Wing Tritium Survey Locations.

Figure 33. B-Wing Second Floor Tritium Survey Locations.

Figure 34. Vault and Tunnel Tritium Survey Locations

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Figure 35. Outdoor Sample Locations.

TABLE 1. RADIONUCLIDES REPORTEDLY USED IN THE ANNEX

Isotope	Decay		Half-Life ^b	On Hand During Survey	Found on Site	Used in Past
	Mode ^a	Energy (MeV)				
^{110m} Ag	β	0.084 & 0.166	249.85 days		X	
²⁶ Al	β	1.2	7.2 E 5 years	X		
¹⁹⁸ Au	β	0.96	2.696 days	X		
⁷ Be	EC		53.44 days			X
¹⁴ C	β	0.16	5.730 years	X		
¹⁰⁹ Cd	EC		464 days	X		
¹¹⁵ Cd	β	0.58 & 1.11	53.46 hours	X		
^{115m} Cd	β	1.62	44.6 days	X		
⁵⁷ Co	EC		270.9 days	X		
⁶⁰ Co	β	0.32	5.271 years		X	
⁵¹ Cr	EC		27.704 days	X		
¹⁵² Eu	EC & β ⁺		13.6 years		X	
¹⁵² Eu	β	0.70 & 1.48	13.6 years		X	
⁵⁹ Fe	β	0.27 & 0.47	44.63 days	X		
⁶⁷ Ga	EC		3.261 days	X		

TABLE 1. - (cont'd.)

Isotope	Decay		Half-Life ^b	On Hand During Survey	Found on Site	Used in Past
	Mode ^a	Energy (MeV)				
¹⁵³ Gd	EC		241.6 days		X	
³ H	β	0.018	12.28 years	X		
¹⁸¹ Hf	β	0.41	42.39 days	X		
¹⁹⁷ Hg	EC		64.14 hours			X
²⁰³ Hg	β	0.21	46.60 days	X		
¹²⁵ I	EC		60.4 days	X	X	
¹³¹ I	β	0.19	8.040 days	X		
^{113m} In	IT		1.658 hours	X		
¹⁹² Ir	β	0.16 & 0.21	74.02 days			X
⁸⁵ Kr	β	0.69	10.72 years		X	
⁵⁴ Mn	EC		312.7 days	X		
^{92m} Nb	EC		10.15 days			X
³² P	β	1.71	14.29 days			X
²¹⁰ Pb	β	0.02 & 0.06	22.6 years			X
²¹⁰ Po	α	5.3	138.378 days			X

TABLE 1. - (cont'd.)

Isotope	Decay		Half-Life ^b	On Hand During Survey	Found on Site	Used in Past
	Mode ^a	Energy (MeV)				
²³⁸ Pu	α	5.5	87.75 years			X
²³⁹ Pu	α	5.1	24,131 years			X
²²⁶ Ra	α	4.8	1,600 years			X
⁸⁶ Rb	β	1.77	18.66 days	X		
²²² Rn	α	5.5	3.824 days			X
⁴⁶ Sc	β	0.36	83.83 days		X	
¹¹³ Sn- ¹¹³ In	EC		115.1 days		X	
⁹⁰ Sr	β	0.55	28.6 years			X
¹⁶⁰ Tb	β	0.57 & 0.87	72.3 years	X		
^{99m} Tc	β	0.29	2.13 E 5 years	X		
²²⁸ Th	α	5.3 & 5.4	1.913 E 9 years			X
²³² Th	α	4.0	1.405 E 10 years		X	X
Natural Th			1.405 E 10 years		X	
²⁰⁴ Tl	β	0.76	3.779 years			X
²³² U	α	5.3	72 years	X		
²³³ U	α	4.8	1.592 E 5 years	X		

TABLE 1. - (cont'd.)

Isotope	Decay		Half-Life ^b	On Hand During Survey	Found on Site	Used in Past
	Mode ^a	Energy (MeV)				
²³⁵ U	α	4.4	7.038 E 5 years	X		
²³⁸ U	α	4.2	4.468 E 9 years	X	X	
Natural U			4.468 E 9 years		X	
¹²⁷ Xe	EC		36.406 days			X
⁶⁵ Zn	EC		244.4 days	X	X	

^a α = Alpha, β = Beta, EC = Electron Capture, IT = Isomeric Transition, β^+ = Positron

^b From Radioactive Decay Data Tables - A Handbook of Decay Data for Application to Radiation Dosimetry and Radiological Assessments by David C. Kocher, 1981; DOE/TIC-11026; published by Technical Information Center, U.S. Department of Energy.

TABLE 2. MISCELLANEOUS AND SOIL SAMPLE WEIGHTS (gm)

Sample Number	Wet Weight	Dry Weight	Sieved Weight	Rocks and Dross
<u>Miscellaneous Samples</u>				
15-Ash-1		157	30	120
15-Benchtop-2		692		
15-Benchtop-3		46		
15-Benchtop-34		26		
15-Benchtop-35		23		
<u>Soil Samples</u>				
15-SB-19A	916	758	630	126
B	640	544	434	109
C	764	653	606	45
D	2469	2115	1726	377
15-SB20-A	771	642	589	52
B	528	458	408	49
C	916	817	635	180
D	2918	2584	2138	440
15-SB21-A	758	582	353	223
B	802	710	260	447
C	1009	949	220	727
D	4847	4396	1315	3067
15-SB22-A	675	512	215	293
B	725	615	273	339
C	869	770	388	378
D	3148	2676	1288	1379
15-S23-A	912	771	634	135
B	716	604	539	64
C	1210	1036	888	148
D	2160	1896	1256	639
15-S24-A	719	592	564	27
B	746	654	465	188
C	773	676	509	166
D	2628	2250	1585	655
15-S25-A	755	645	615	24
B	652	558	465	91
C	910	769	752	14
D	2491	2189	2004	184

TABLE 2. - (cont'd.)

Sample Number	Wet Weight	Dry Weight	Sieved Weight	Rocks and Dross
15-S26-A	1159	966	925	38
B	672	581	515	65
C	962	831	725	105
D	1508	1297	1075	215

TABLE 3. WATER AND SLUDGE SAMPLE VOLUME AND WEIGHTS

Sample Number	Volume (ℓ)	Total Dry Solids (g)
<u>Water Samples</u>		
15-SW-4	0.963	78.28
15-SW-6	3.710	0.960
15-SW-7	4.203	22.78
15-SW-10	4.273	0.549
15-SW-11	4.050	66.54
15-SW-14	4.005	0.320
15-SW-15	3.795	0.253
15-SW-18	3.593	14.75
15-SW-28	0.735	1.277
15-SW-29	0.529	1.756
15-SW-30	0.344	8.202
15-SW-32	0.969	1.616
15-SW-33	0.992	1.948
<u>Sludge Samples</u>		
15-SS-5		314.6
15-SS-8		196.8
15-SS-9		264.3
15-SS-12		556.9
15-SS-13		482.2
15-SS-16		203.4
15-SS-17		1204.1
15-SS-27		33.5
15-SS-31		62.5

TABLE 4. DATA SHEET OF AREA SURVEYS

Room or Area Number	Percent of Area Accessible to Survey		PAC-4G-3 Direct Readings ^a (dis/min-100 cm ²)		PRM-7 (μR/h) 1 meter	PRM-5-3 w/PG-2 (cts/min)	Comments
	Floor	Wall	Beta	Alpha			
<u>C-Wing</u>							
C-1	90	60	BKGD ^b	NA ^c	BKGD	1.3 k	Floor tile
C-5	60	10	BKGD	NA	BKGD	BKGD	
C-6	60	20	BKGD	NA	BKGD	BKGD	
C-7	60	10	59 k	BKGD	BKGD	3 k	Conduit around door 0.2 mR/h
C-7A & C-7B	60	10	BKGD	NA	BKGD	BKGD	
C-9	60	10	BKGD	NA	BLGD	BKGD	
C-10	80	20	10 k	1.3 k	BKGD	BKGD	Two floor areas
C-11	60	10	BKGD	NA	BKGD	BKGD	Except for de- pleted UO ₂
C-12	80	30	77 k	BKGD	BKGD	2.5 k	Floor, drains and sources 0.15 mR/h
C-13	60	10	BKGD	NA	BKGD	5 k	UO ₂ in C-11
C-15	60	10	BKGD	NA	BKGD	BKGD	
C-16	80	20	35 k	BKGD	BKGD	BKGD	Wall
C-17	60	10	BKGD	NA	BKGD	BKGD	

TABLE 4. - (cont'd.)

Room or Area Number	Percent of Area Accessible to Survey		PAC-4G-3 Direct Readings ^a (dis/min-100 cm ²)		PRM-7 (μR/h) 1 meter	PRM-5-3 w/PG-2 (cts/min)	Comments
	Floor	Wall	Beta	Alpha			
C-19	90	10	BKGD	NA	BKGD	BKGD	
C-20	60	20	BKGD	NA	BKGD	BKGD	Except for Th(NO ₃) ₄
C-21	60	40	BKGD	NA	BKGD	BKGD	
C-23	60	20	BKGD	NA	40	BKGD	Except ¹²⁵ I
C-24 & C-24A	25	25	BKGD	NA	BKGD	BKGD	
C-25	60	10	BKGD	NA	BKGD	BKGD	
C-27	60	20	BKGD	NA	BKGD	BKGD	
C-31 & C-31A	60	10	BKGD	NA	BKGD	BKGD	
C-32 & C-32A	25	25	BKGD	NA	BKGD	BKGD	
C-34	60	10	4 k	BKGD	100	6 k	6 k Ci ⁶⁰ Co source
C-35 & C-35A	60	10	BKGD	NA	BKGD	BKGD	
C-37	90	80	BKGD	NA	BKGD	BKGD	

TABLE 4. - (cont'd.)

Room or Area Number	Percent of Area Accessible to Survey		PAC-4G-3 Direct Readings ^a (dis/min-100 cm ²)		PRM-7 (μR/h) 1 meter	PRM-5-3 w/PG-2 (cts/min)	Comments
	Floor	Wall	Beta	Alpha			
C-40	60	10	11 k	BKGD	100	8 k	Floor center 0.15 mR/h
C-40A & C-40B	60	10	BKGD	NA	BKGD	BKGD	
C-41	60	30	BKGD	NA	BKGD	BKGD	Except source in balance
C-43	40	10	13 k	BKGD	BKGD	45 k	Floor
C-43A	30	10	1.6 k	BKGD	BKGD	1 k	Wall under sink
C-50	25	40	BKGD	NA	BKGD	BKGD	
C-52	25	40	BKGD	NA	BKGD	BKGD	
C-54	70	80	BKGD	NA	BKGD	BKGD	
C-56	80	80	BKGD	NA	BKGD	BKGD	
C-57	80	20	35 k	BKGD	BKGD	30 k	Floor, drains and wall
C-58	75	80	BKGD	NA	BKGD	BKGD	
C-60	25	60	BKGD	NA	BKGD	BKGD	
C-62	40	60	700	BKGD	BKGD	BKGD	Southwest wall

TABLE 4. - (cont'd.)

Room or Area Number	Percent of Area Accessible to Survey		PAC-4G-3 Direct Readings ^a (dis/min-100 cm ²)		PRM-7 (μR/h) 1 meter	PRM-5-3 w/PG-2 (cts/min)	Comments
	Floor	Wall	Beta	Alpha			
C-64	25	35	BKGD	NA	BKGD	BKGD	
C-65	60	30	47 k	40 k	BKGD	1 k	Floor under hood
C-68	60	60	BKGD	NA	BKGD	BKGD	
C-69	80	80	800	BKGD	BKGD	BKGD	Ceramic floor tile
C-71	60	70	4 k	BKGD	100	22 k	Th(NO ₃) ₄ in C-20
C-72	25	25	BKGD	NA	BKGD	BKGD	
<u>C-Wing Halls</u>							
A	90	90	BKGD	NA	BKGD	BKGD	
B	100	100	BKGD	NA	BKGD	BKGD	
C	100	100	BKGD	NA	BKGD	BKGD	
D	100	100	BKGD	NA	BKGD	BKGD	
E	100	100	BKGD	NA	BKGD	BKGD	Except for sources in C-23 & C-34
Elevator Closet	20	30	BKGD	NA	BKGD	BKGD	

TABLE 4. - (cont'd.)

Room or Area Number	Percent of Area Accessible to Survey		PAC-4G-3 Direct Readings ^a (dis/min-100 cm ²)		PRM-7 (μR/h) ~ 1 meter	PRM-5-3 w/PG-2 (cts/min)	Comments
	Floor	Wall	Beta	Alpha			
Roof	100	-	13 k	270	80	20 k	0.3 mR/h, blower and two areas
Tunnel	100	100	58 k	1 k	15	10 k	Wall and floor
Store Room	40	20	BKGD	NA	BKGD	BKGD	
Elevator	100	100	BKGD	NA	BKGD	BKGD	
Stairway	100	100	BKGD	NA	BKGD	BKGD	
Elevator Control Room	100	80	BKGD	NA	BKGD	BKGD	
Vault	90	80	600 k	1 k	BKGD	150 k	Maxima from 12 areas
<u>B-Wing First Floor</u>							
B-102	40	30	BKGD	NA	BKGD	BKGD	
B-103	60	30	BKGD	NA	BKGD	BKGD	
B-104	50	20	35 k	BKGD	300	50 k	Source in file cabinet
B-105	90	80	BKGD	NA	BKGD	BKGD	
B-106	20	30	BKGD	NA	BKGD	BKGD	

TABLE 4. - (cont'd.)

Room or Area Number	Percent of Area Accessible to Survey		PAC-4G-3 Direct Readings ^a (dis/min-100 cm ²)		PRM-7 (μR/h) 1 meter	PRM-5-3 w/PG-2 (cts/min)	Comments
	Floor	Wall	Beta	Alpha			
B-107 & B-107A	90	90	BKGD	NA	BKGD	BKGD	
B-108	90	70	BKGD	NA	BKGD	BKGD	
B-109	90	70	BKGD	NA	BKGD	BKGD	
B-110	60	15	BKGD	NA	BKGD	BKGD	
B-112	80	60	BKGD	NA	BKGD	BKGD	
B-113	60	40	11 k	BKGD	2000	> 500 k	⁶⁰ Co source
B-115	60	30	BKGD	NA	30	15 k	East wall from B-113
B-115A	60	40	BKGD	NA	BKGD	BKGD	
B-130	80	60	BKGD	NA	BKGD	BKGD	
B-131	40	30	BKGD	NA	BKGD	BKGD	
B-132	70	90	BKGD	NA	BKGD	BKGD	
B-133	40	30	BKGD	NA	BKGD	BKGD	
B-134	50	70	BKGD	NA	BKGD	BKGD	
B-135	60	30	360 k	270 k	BKGD	10 k	West wall
B-136	100	100	BKGD	NA	BKGD	BKGD	

TABLE 4. - (cont'd.)

Room or Area Number	Percent of Area Accessible to Survey		PAC-4G-3 Direct Readings ^a (dis/min-100 cm ²)		PRM-7 (μR/h) 1 meter	PRM-5-3 w/PG-2 (cts/min)	Comments
	Floor	Wall	Beta	Alpha			
B-137 & B-137A	90	90	BKGD	NA	BKGD	BKGD	
B-138	100	100	BKGD	NA	BKGD	BKGD	
B-139	90	90	BKGD	NA	BKGD	BKGD	
B-140	100	100	BKGD	NA	BKGD	BKGD	
B-141	80	70	BKGD	NA	BKGD	BKGD	
B-142	90	100	BKGD	NA	BKGD	4 k	Wall tile
B-143	60	40	BKGD	NA	BKGD	BKGD	
B-144	30	20	47 k	BKGD	3000	250 k	Waste can of ashes and furnace
B-145, B-145A & B-145B	40	20	BKGD	NA	BKGD	BKGD	
B-146	100	100	BKGD	NA	BKGD	BKGD	
B-147	60	20	BKGD	NA	BKGD	BKGD	
B-148	80	60	BKGD	NA	BKGD	BKGD	
B-151	80	90	BKGD	NA	BKGD	BKGD	

TABLE 4. - (cont'd.)

Room or Area Number	Percent of Area Accessible to Survey		PAC-4G-3 Direct Readings ^a (dis/min-100 cm ²)		PRM-7 (μR/h) 1 meter	PRM-5-3 w/PG-2 (cts/min)	Comments
	Floor	Wall	Beta	Alpha			
B-151A	50	60	BKGD	NA	BKGD	BKGD	
B-152	80	80	BKGD	NA	BKGD	BKGD	
B-153	50	60	BKGD	NA	BKGD	BKGD	
B-154	80	90	BKGD	NA	BKGD	BKGD	
B-155	70	40	BKGD	NA	BKGD	BKGD	
B-157	70	40	BKGD	NA	BKGD	BKGD	
B-158	50	80	BKGD	NA	BKGD	BKGD	
B-159	80	80	BKGD	NA	BKGD	BKGD	
B-160	80	80	BKGD	NA	BKGD	BKGD	
<u>Corridors</u>							
Elevator	100	100	BKGD	NA	BKGD	BKGD	
Short East-West	100	100	BKGD	NA	BKGD	BKGD	From B-134 to B-142
North- South	100	100	BKGD	NA	BKGD	BKGD	
Main East-West	100	100	240	BKGD	40	15 k	Between B-113 & B-115

TABLE 4. - (cont'd.)

Room or Area Number	Percent of Area Accessible to Survey		PAC-4G-3 Direct Readings ^a (dis/min-100 cm ²)		PRM-7 (μR/h) 1 meter	PRM-5-3 w/PG-2 (cts/min)	Comments
	Floor	Wall	Beta	Alpha			
Main East-West (cont'd.)			BKGD	NA	20	4 k	Through door of B-142
			1.5 k	NA	60	15 k	Through door of B-144
<u>B-Wing, Second Floor</u>							
B-200	60	40	BKGD	NA	BKGD	BKGD	
B-201	100	90	600	BKGD	BKGD	BKGD	Overhead light fixture
B-201A	100	90	11 k	BKGD	BKGD	BKGD	Floor
B-201B	60	40	8 k	BKGD	BKGD	BKGD	Floor #480
B-202	60	40	7 k	500	BKGD	BKGD	Floor in front of hood with sources
B-204	70	40	71 k	BKGD	BKGD	25 k	Wall
B-205	85	40	6 k	BKGD	BKGD	BKGD	Over sink
B-207	100	100	BKGD	NA	BKGD	BKGD	
B-208	70	40	BKGD	NA	BKGD	> 500 k	⁸⁵ Kr in hood
B-209	60	40	BKGD	NA	BKGD	BKGD	

TABLE 4. - (cont'd.)

Room or Area Number	Percent of Area Accessible to Survey		PAC-4G-3 Direct Readings ^a (dis/min-100 cm ²)		PRM-7 (μ R/h) 1 meter	PRM-5-3 w/PG-2 (cts/min)	Comments
	Floor	Wall	Beta	Alpha			
B-210	60	30	1 k	BKGD	BKGD	BKGD	Floor area in cooler
B-211	100	100	BKGD	NA	BKGD	BKGD	
B-212	80	60	BKGD	NA	200	100 k	Source in Pb cave
B-213	60	20	BKGD	NA	500	250 k	Four sources
B-230	0	0	-	-	-	-	Closet inaccessible
B-231	100	80	BKGD	NA	BKGD	BKGD	
B-232	4	5	BKGD	NA	BKGD	BKGD	
B-233	80	60	BKGD	NA	BKGD	BKGD	
B-234 & B-234A	60	30	BKGD	NA	BKGD	BKGD	
B-235	80	40	BKGD	NA	30	13 k	Sources north-west corner cabinet
B-237	60	30	BKGD	NA	BKGD	BKGD	
B-238A, B & C	60	20	BKGD	NA	BKGD	BKGD	

TABLE 4. - (cont'd.)

Room or Area Number	Percent of Area Accessible to Survey		PAC-4G-3 Direct Readings ^a (dis/min-100 cm ²)		PRM-7 (μR/h) 1 meter	PRM-5-3 w/PG-2 (cts/min)	Comments
	Floor	Wall	Beta	Alpha			
B-238D, E & F	70	30	BKGD	NA	BKGD	BKGD	
B-238G	70	60	BKGD	NA	BKGD	BKGD	
B-238H	60	20	BKGD	NA	BKGD	BKGD	
B-239	70	20	BKGD	NA	BKGD	BKGD	
B-241	60	20	BKGD	NA	BKGD	BKGD	
B-242	70	40	BKGD	NA	BKGD	BKGD	
B-243	60	20	BKGD	NA	BKGD	BKGD	
B-244	80	50	7 k	BKGD	BKGD	BKGD	North floor by door
B-244A	70	40	BKGD	NA	BKGD	BKGD	
B-245	60	40	360	BKGD	BKGD	3 k	Brick walls
B-246	100	100	BKGD	NA	BKGD	BKGD	
B-247	70	30	BKGD	NA	BKGD	BKGD	
B-248	100	100	BKGD	NA	BKGD	BKGD	
<u>Corridor</u> North- South	100	90	BKGD	NA	BKGD	BKGD	

TABLE 4. - (cont'd.)

Room or Area Number	Percent of Area Accessible to Survey		PAC-4G-3 Direct Readings ^a (dis/min-100 cm ²)		PRM-7 (μR/h) 1 meter	PRM-5-3 w/PG-2 (cts/min)	Comments
	Floor	Wall	Beta	Alpha			
East-West	100	100	BKGD	NA	BKGD	BKGD	
<u>B-Wing, Third Floor</u>							
B-300	90	80	BKGD	NA	BKGD	BKGD	Elevated BKGD from hood in 301
B-300A	60	30	BKGD	NA	25	2.5 k	Tile wall
B-300B	50	20	BKGD	NA	BKGD	BKGD	
B-301	60	30	30 k	13 k	20	2.5 k	After sources removed for survey
B-302	40	10	BKGD	NA	BKGD	BKGD	
B-303	60	20	BKGD	NA	45	300 k	Sources in southeast corner
B-304	80	60	BKGD	NA	BKGD	BKGD	
B-305	80	30	BKGD	NA	BKGD	15 k	Sources in southeast corner
B-306	80	60	BKGD	NA	BKGD	BKGD	
B-307	100	100	BKGD	NA	BKGD	BKGD	

TABLE 4. - (cont'd.)

Room or Area Number	Percent of Area Accessible to Survey		PAC-4G-3 Direct Readings ^a (dis/min-100 cm ²)		PRM-7 (μR/h) 1 meter	PRM-5-3 w/PG-2 (cts/min)	Comments
	Floor	Wall	Beta	Alpha			
B-308	60	20	BKGD	NA	BKGD	BKGD	
B-309	40	30	BKGD	NA	BKGD	BKGD	
B-310	60	20	BKGD	NA	BKGD	BKGD	
B-311	60	20	BKGD	NA	BKGD	BKGD	Except for U and Th Chemicals
B-312	60	30	BKGD	NA	BKGD	BKGD	
B-314	60	30	BKGD	NA	BKGD	BKGD	
B-316	70	40	71 k	BKGD	BKGD	BKGD	Area on bench
B-318	100	90	BKGD	NA	BKGD	BKGD	
B-320	90	80	BKGD	NA	BKGD	BKGD	
B-321 A & B	90	90	BKGD	NA	BKGD	BKGD	
B-322	90	90	BKGD	NA	BKGD	BKGD	
B-323	90	90	BKGD	NA	BKGD	BKGD	
B-324	60	50	BKGD	NA	BKGD	BKGD	
B-325	90	90	BKGD	NA	BKGD	BKGD	

TABLE 4. - (cont'd.)

Room or Area Number	Percent of Area Accessible to Survey		PAC-4G-3 Direct Readings ^a (dis/min-100 cm ²)		PRM-7 (μR/h) 1 meter	PRM-5-3 w/PG-2 (cts/min)	Comments
	Floor	Wall	Beta	Alpha			
B-326	90	90	BKGD	NA	BKGD	BKGD	
B-326A	100	100	BKGD	NA	BKGD	BKGD	
B-327 & B-327A	90	90	BKGD	NA	BKGD	BKGD	
B-328	90	90	BKGD	NA	BKGD	BKGD	
B-328A	90	80	5 k	NRR ^d	BKGD	10 k	Monkey cage
B-329 & B-329D	90	90	BKGD	NA	BKGD	BKGD	
B-330	0	0	-	-	-	-	Inaccessible
B-331	95	95	BKGD	NA	BKGD	BKGD	
B-332	0	0	-	-	-	-	Inaccessible
B-333	90	90	BKGD	NA	BKGD	BKGD	
B-334	90	90	BKGD	NA	BKGD	BKGD	
B-335	70	20	4 k	BKGD	BKGD	30 k	Three areas
B-336	90	90	BKGD	NA	BKGD	BKGD	
B-337	90	90	BKGD	NA	BKGD	BKGD	
B-338	90	80	BKGD	NA	BKGD	BKGD	

TABLE 4. - (cont'd.)

Room or Area Number	Percent of Area Accessible to Survey		PAC-4G-3 Direct Readings ^a (dis/min-100 cm ²)		PRM-7 (μR/h) 1 meter	PRM-5-3 w/PG-2 (cts/min)	Comments
	Floor	Wall	Beta	Alpha			
B-339	90	90	BKGD	NA	BKGD	BKGD	
B-340	90	90	BKGD	NA	BKGD	BKGD	
B-341	90	90	BKGD	NA	BKGD	BKGD	
B-342	90	90	BKGD	NA	BKGD	BKGD	
B-344	80	60	BKGD	NA	BKGD	BKGD	
B-351	90	90	BKGD	NA	BKGD	BKGD	
B-352	90	90	BKGD	NA	BKGD	BKGD	
B-353	90	90	BKGD	NA	BKGD	BKGD	
B-354	90	90	BKGD	NA	BKGD	BKGD	
B-355	90	90	BKGD	NA	BKGD	BKGD	
B-356	90	90	BKGD	NA	BKGD	BKGD	
B-357	90	90	BKGD	NA	BKGD	BKGD	
B-358	90	90	BKGD	NA	BKGD	BKGD	
<u>Corridors</u>							
Main North-South	100	100	BKGD	NA	BKGD	BKGD	From B-300 to B-311
Entrance	100	100	BKGD	NA	BKGD	BKGD	By B-314

TABLE 4. - (cont'd.)

Room or Area Number	Percent of Area Accessible to Survey		PAC-4G-3 Direct Readings ^a (dis/min-100 cm ²)		PRM-7 (μR/h) 1 meter	PRM-5-3 w/PG-2 (cts/min)	Comments
	Floor	Wall	Beta	Alpha			
East-West	90	90	BKGD	NA	BKGD	BKGD	West end
Circular	100	100	BKGD	NA	BKGD	BKGD	Around iso- lated area
East-West	90	90	BKGD	NA	BKGD	BKGD	In isolated area
<u>Stairwells</u>							
East	100	100	BKGD	NA	BKGD	BKGD	From first floor to fourth floor
West	100	100	BKGD	NA	BKGD	BKGD	From first floor to fourth floor
<u>Roof Building and B-Wing Roof</u>							
Big Building	70	40	BKGD	NA	BKGD	BKGD	Storage?
Southwest Building	30	10	BKGD	NA	BKGD	BKGD	
North-South Wing	80		5 k	BKGD	BKGD	BKGD	South vent
East-West Wing	100		BKGD	NA	BKGD	BKGD	

TABLE 4. - (cont'd.)

Room or Area Number	Percent of Area Accessible to Survey		PAC-4G-3 Direct Readings ^a (dis/min-100 cm ²)		PRM-7 (μR/h) 1 meter	PRM-5-3 w/PG-2 (cts/min)	Comments
	Floor	Wall	(dis/min-100 cm ²)				
			Beta	Alpha			
<u>Low-Level Background Facility</u>							
LB-1	40	20	BKGD	NA	BKGD	BKGD	
LB-2 & LB-3	40	20	BKGD	NA	BKGD	BKGD	Except sources in Pb pigs
LB-4	40	20	100 k	BKGD	NRR	NRR	1.3 mR/h ra- dioactive source
Roof	100		BKGD	NA	BKGD	BKGD	
<u>Other Auxiliary Buildings and Areas</u>							
Metal Frame Building	10	30	BKGD	NA	BKGD	BKGD	West of LB Facility
Garage	40	30	BKGD	NA	BKGD	BKGD	North of B-115A
Garage Roof	100	-	12 k	BKGD	300	50 k	Samples on window sill in B-113
Fire Escape Ladder	100	100	BKGD	NA	BKGD	BKGD	East of garage
Pump Room & Fenced Yard	100	100	BKGD	NA	BKGD	BKGD	East of garage

FOOTNOTES FOR TABLE 4

^aThe beta mode direct readings and alpha mode direct readings were taken with PAC-4G-3 instruments. The beta mode detects both electromagnetic and particulate radiation. If an area indicated a higher count rate than the area 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 readings are compensated for any alpha contribution by subtracting the alpha-mode reading from the beta-mode reading. The area background is subtracted in both the alpha- and beta-mode readings to obtain the net readings.

^bBKGD = Background. The following are the normal area background readings for each instrument.

	<u>Beta Mode</u>	<u>Alpha Mode</u>
Floor Monitor	350-500 cts/min-325 cm ²	0-50 cts/min-325 cm ²
PAC-4G-3	150-400 cts/min-100 cm ²	0-5 cts/min-100 cm ²
PC-5 Counter	50.1-54.3 cts/min	1.0-1.2 cts/min
PC-3A Counter	43.0-49.0 cts/min	0.05-0.3 cts/min
10-Wire	402-469 cts/min	1.2-6.6 cts/min
PRM-5-3	400-2000 cts/min	
PRM-7	3-15 μ R/h	
GM End Window Detector read 0.01 to 0.03 mR/h at floor level.		

^cNA = Nonapplicable. No contamination was detected above background in the beta mode; therefore, no alpha mode or contact GM End Window survey was necessary.

^dNRR = No reading recorded.

TABLE 5. - (cont'd.)

Room Number	Area (ft ²)	PAC-4G-3 Direct Readings ^a (dis/min-100 cm ²)		End Window (mR/h)	μ R Meter (μ R/h)	PRM-5-3 (cts/min)	Smears (dis/min-100 cm ²)	Comments
		Beta-Gamma	Alpha					
C-57 (cont'd.)		15 k	BKGD	NRR	BKGD	2.5 k	BKGD	Drain #1, Location 22
		1.2 k	BKGD	NRR	BKGD	BKGD	NST	Drain #2
	4	1.9 k	BKGD	NRR	BKGD	BKGD	NST	Floor near drain #2
	4	1.9 k	BKGD	NRR	BKGD	BKGD	BKGD	Floor, Location 21
	5	11 k	BKGD	NRR	BKGD	BKGD	NST	Floor
C-62	2	700	BKGD	NRR	BKGD	BKGD	BKGD	Southwest wall, Location 215
C-65	0.06	47k	40 k	NRR	BKGD	1 k	NST	Floor
Roof of C-Wing	10	240	BKGD	BKGD	BKGD	2.5 k	NST	Southwest area, Location 851
	4	BKGD	BKGD	BKGD	BKGD	2.5 k	NST	North area, Location 852
		13 k	270	0.3	80	20 k	NST	Blower ²³² Th, Location 853
<u>Tunnel and Vault</u>								
Tunnel		58 k	1 k	NRR	15	10 k	BKGD	Wall and floor, Location 370
Vault	1	15 k	BKGD	NRR	BKGD	BKGD	$\beta\gamma = 111$	Floor, Location 354
	1	7 k	BKGD	NRR	BKGD	BKGD	$\beta\gamma = 114$	Floor, Location 355
	2	14 k	BKGD	NRR	BKGD	BKGD	$\beta\gamma = 396$	Floor, Location 356
	1	11 k	BKGD	NRR	BKGD	BKGD	$\beta\gamma = 274$	Floor, Location 358
	1	3 k	BKGD	NRR	BKGD	BKGD	BKGD	Floor, Location 357
	1	7 k	BKGD	NRR	BKGD	BKGD	BKGD	Floor, Location 359
		600 k	500	NRR	BKGD	150 k	BKGD	Bricks, Location 362
	5	480 k	300	NRR	BKGD	150 k	NST	Behind shield
	2	9 k	BKGD	NRR	BKGD	25 k	BKGD	Floor, Location 366
	2	72 k	1 k	NRR	BKGD	BKGD	NST	Shelf

TABLE 5. ACTIVITY FROM CONTAMINATION

Room Number	Area (ft ²)	PAC-4G-3 Direct Readings ^a (dis/min-100 cm ²)		End Window (mR/h)	μ R Meter (μ R/h)	PRM-5-3 (cts/min)	Smears (dis/min-100 cm ²)	Comments
		Beta-Gamma	Alpha					
C-Wing								
C-7	2	59 k	BKGD ^b	0.2	BKGD	3 k	$\beta\gamma^c=99.5$	Conduit, Location 43
C-10	2	10 k	1.3 k	BKGD	BKGD	BKGD	$\alpha = 35.5$	Floor near sinks, Location 105
	4	9 k	0.4 k	BKGD	BKGD	BKGD	$\beta\gamma=23.6$ NST ^d	
C-12	10	7 k	BKGD	NRR ^e	BKGD	BKGD	NST	Floor pump room
	2	77 k	BKGD	0.15	BKGD	2.5 k	$\alpha = 25.8$	Floor pump room
		11 k	BKGD	NRR	BKGD	BKGD	$\beta\gamma = 30.6$	Location 93
		6 k	BKGD	NRR	BKGD	BKGD	NST	Drain #1
		3 k	BKGD	NRR	BKGD	BKGD	BKGD	Drain #2, Location 99
		BKGD	NA ^f	NA	BKGD	BKGD	BKGD	Drain #3, Location 101
						5 k	BKGD	Common baseboard with C-16, Location 104
C-16		BKGD	NRR	NRR	BKGD	2.3 k	NST	Common baseboard with C-12
	10	35 k	BKGD	NRR	BKGD	BKGD	BKGD	North wall, Location 112
C-40	2	11 k	BKGD	0.15	100	8 k	BKGD	Floor center, Location 147
C-43	4	13 k	BKGD	NRR	BKGD	45 k	BKGD	Floor, Location 171
C-43A		1.6k	BKGD	NRR	BKGD	1 k	BKGD	Wall under sink, Location 167
C-57	10	35 k	BKGD	NRR	BKGD	30 k	BKGD	South wall, Location 19

TABLE 5. - (cont'd.)

Room Number	Area (ft ²)	PAC-4G-3 Direct Readings ^a (dis/min-100 cm ²)		End Window (mR/h)	μ R Meter (μ R/h)	PRM-5-3 (cts/min)	Smears (dis/min-100 cm ²)	Comments
		Beta-Gamma	Alpha					
Vault	1	36 k	BKGD	NRR	BKGD	10 k	NST	Wood
(cont'd.)	9	1 k	BKGD	NRR	BKGD	BKGD	BKGD	Steel plate, Location 369
<u>B-Wing, First Floor</u>								
B-135	10	360 k	270 k	NRR	BKGD	10 k	$\alpha = 350$ $\beta\gamma = 440$	Wall, Location 299
B-144		47 k	BKGD	NRR	3000	250 k	BKGD	Waste can with ashes, Location 395
		NRR	BKGD	NRR	100	25 k	NST	Furnace face (east wall)
		NRR	BKGD	NRR	NRR	150 k	NST	Inside furnace to chimney, Locations 834 & 835
		71 k	BKGD	NRR	NRR	NRR	NST	Sample 15-Ash-1
		BKGD	NA	NA	BKGD	BKGD	$\beta\gamma = 70$	West wall
<u>B-Wing, Second Floor</u>								
B-201		BKGD	NA	NA	NRR	BKGD	$\beta\gamma = 125$	Overhead, Location 490
		600	BKGD	NRR	BKGD	BKGD	$\alpha = 6$ $\beta\gamma = 1174$	Light fixture, Location 815
B-201A	10	11 k	BKGD	NRR	BKGD	BKGD	BKGD	Floor, Location 478
B-201B	10	8 k	BKGD	NRR	BKGD	BKGD	BKGD	Floor, Location 480
B-202	3	7 k	500	NRR	BKGD	BKGD	BKGD	Floor, Location 532
B-204		71 k	BKGD	NRR	BKGD	25 k	BKGD	Wall, Location 523

TABLE 5. - (cont'd.)

Room Number	Area (ft ²)	PAC-4G-3 Direct Readings ^a (dis/min-100 cm ²)		End Window (mR/h)	μ R Meter (μ R/h)	PRM-5-3 (cts/min)	Smears (dis/min-100 cm ²)	Comments
		Beta-Gamma	Alpha					
B-205		6 k	BKGD	NRR	BKGD	BKGD	BKGD	Spot over sink, Location 479
B-210		1 k	BKGD	NRR	BKGD	BKGD	BKGD	Floor, Location 511
B-244		7 k	BKGD	NRR	BKGD	BKGD	NST	Floor
<u>B-Wing, Third Floor</u>								
B-301		5 k	1 k	NRR	BKGD	BKGD	BKGD	Window sill, Location 658
		NRR	NRR	NRR	NRR	NRR	$\alpha=10$	Overhead, Location 654
		39 k	BKGD	NRR	20	2.5 k	BKGD	Top of bench top, Location 659
		30 k	13 k	NRR	NRR	2.5 k	$\alpha=83$ $\beta\gamma=164$ $\alpha=71$ $\beta\gamma=123$	Bottom of bench top, Location 849
								Bottom of bench top, Location 850
B-316	1	71 k	BKGD	NRR	BKGD	BKGD	BKGD	Bench, Location 665
B-328A		5 k	NRR	NRR	BKGD	10 k	NST	Monkey cage
B-335		480 k	BKGD	NRR	BKGD	30 k	$\beta\gamma=150$	Floor, southwest corner, Location 729
		59 k	BKGD	NRR	BKGD	BKGD	NST	Floor, southwest corner
		59 k	BKGD	NRR	BKGD	BKGD	$\beta\gamma=5500$	Floor, northeast corner, Location 734

FOOTNOTES FOR TABLE 5

^aSee Footnote a in Table 4.

^bBKGD = Background. See Footnote b in Table 4

^c α =Alpha

$\beta\gamma$ =Beta-Gamma

(The beta-gamma readings are compensated for any alpha contamination by subtracting the alpha reading. The background count rate is subtracted in both readings.)

^dNST = No Smear Taken.

^eNRR = No Reading Recorded.

^fNA = Not Applicable. No contamination was detected above background in the beta mode; therefore, no alpha mode or contact GM end window survey was necessary.

TABLE 6. ACTIVITY FROM SOURCES

Room Number	PAC-4G-3 Direct Readings ^a (dis/min-100 cm ²)		End Window (mR/h)	μ R Meter (μ R/h)	PRM-5-3 (cts/min)	Comments
	Beta-Gamma	Alpha				
<u>C-Wing</u>						
C-11	NRR ^b	NRR	NRR	NRR	> 5 k	Depleted U
C-12	NRR	NRR	NRR	NRR	200 k	Sources in hoods
C-20	> 4 k	NRR	> 0.1	NRR	> 22 k	Th(NO ₃) ₄
C-23	> 700	NRR	> 0.4	40	> 6 k	¹²⁵ I (Pb shield added)
C-34	4 k	BKGD ^c	0.05	100	6 k	6000 Ci ⁶⁰ Co
C-41	NRR	NRR	NRR	NRR	NRR	Source in balance
<u>B-Wing, First Floor</u>						
B-104	35 k	BKGD	NRR	300	50 k	Source in file cabinet
B-113	11 k	BKGD	NRR	2000	> 500 k	⁶⁰ Co
<u>B-Wing, Second Floor</u>						
B-202	23 k	BKGD	NRR	NRR	NRR	Sources in hood in north-east corner
B-208	NRR	NA ^a	NA	BKGD	> 500 k	⁸⁵ Kr in hood
B-212	BKGD	NA	NA	200	100 k	Sources in Pb cave
B-213	BKGD	NA	NA	300	100 k	Bottle on floor
	BKGD	NA	NA	500	250 k	Source on bench
	BKGD	NA	NA	35	15 k	Source in hood
	BKGD	NA	NA	100	50 k	Waste box

TABLE 6. - (cont'd.)

Room Number	PAC-4G-3 Direct Readings ^a (dis/min-100 cm ²)		End Window (mR/h)	μ R Meter (μ R/h)	PRM-5-3 (cts/min)	Comments
	Beta-Gamma	Alpha				
B-235	BKGD	NRR	NRR	30	13 k	In corner cabinet
<u>B-Wing, Third Floor</u>						
B-301	NRR	NRR	NRR	1000	NRR	Column
B-303	BKGD	NRR	NRR	45	300 k	In southeast corner
B-305	BKGD	NA	NA	BKGD	15 k	In southeast corner
B-311	BKGD	NA	NA	25	200 k	U & Th chemicals
<u>Low-Level Background Facility</u>						
LB2 & LB3	NRR	NRR	NRR	NRR	NRR	Sources in Pb pigs
LB4	100 k	BKGD	1.3	NRR	NRR	Sources

^aSee Footnote a in Table 4.

^bNRR = No Reading Recorded.

^cBKGD = Background. (See Footnote b in Table 4.)

^dNA = Not Applicable. No contamination was detected above background in the beta mode; therefore, no alpha or contact GM end window survey was necessary.

TABLE 7. RADON CONCENTRATIONS IN AIR SAMPLES

Sample Number	Room Number	Figure Number ^a	Actinon ^{219}Rn	Radon ^{222}Rn (pCi/l)	Radon Daughter (mWL) ^b	Thoron ^{220}Rn (fCi/l)
1	C-56	9	NS ^c	0.071	0.71	6.3
2	C-62	9	NS	0.191	1.91	1.4
3	C-71	9	NS	0.034	0.34	3.4
4	C-72	9	NS	0.125	1.25	3.4
5	C-12	9	BDL ^d	0.124	1.24	0.8
6	C-20	9	BDL	0.109	1.09	1.7
7	C-31	9	NS	0.095	0.95	2.7
8	C-23	9	NS	0.086	0.86	1.7
9	C-40	9	BDL	0.046	0.46	4.6
10	Vault	29	BDL	0.670	6.70	9.5
11	LB-4	31	NS	1.460	14.60	9.3
12	B-110	16	NS	0.450	4.50	3.6
13	B-135	16	BDL	0.130	1.30	3.6
14	B-139	16	NS	0.048	0.48	<0.8
15	B-144	16	BDL	0.145	1.45	2.6
16	B-152	16	NS	<0.006	<0.06	<0.8
17	B-339	22	NS	0.012	0.12	<0.8
	Corridor					
18	B-301	22	BDL	0.015	0.15	4.7
19	B-314	22	BDL	<0.006	<0.06	<0.8
20	B-332	22	BDL	<0.006	<0.06	<0.8
21	4th Floor	23	NS	0.107	1.07	2.7
22	B-238	19	BDL	0.120	1.20	8.5
23	B-201	19	BDL	0.122	1.22	1.7
24	B-213	19	NS	0.093	0.93	3.7
25	B-243	19	NS	0.090	0.90	1.7

^aIndicates figure showing location at which measurement was made.

^bWorking Level (WL) is defined as any combination of short-lived radon-decay products in one liter of air that will result in the ultimate emission of 1.3×10^5 MeV of potential alpha energy. The numerical value of the WL is derived from the alpha energy released by the total decay through RaC' of the short-lived radon-daughter products, RaA, RaB, and RaC at radioactive equilibrium with 100 pCi of ^{222}Rn per liter of air. A milliWorking Level (mWL) is 10^{-3} WL.

^cNS = Not Selected, indicating that the room or area was not selected for sampling with the positive displacement pump. The filter from the positive displacement pump is counted on the alpha spectrometer in order to distinguish actinon (^{219}Rn) daughters from radon (^{222}Rn) daughters.

^dBDL = Below Detectable Limits.

TABLE 8. AMBIENT BACKGROUND RADIATION MEASUREMENTS

Sample Number	Room Number	Figure Number ^a	Reuter Stokes RSS-111			PRM-7 (μR/h)
			Time (h)	Activity (μR)	Rate (μR/h)	
1	C-12	9	3.33	32	9.6	12-14
2	C-40	9	3 33	36	10.8	8-10
3	B-135	16	4.42	42	9.5	8
4	B-301	22	4.00	68	17.0	13
5	B-201	19	7.50	77	10.3	8

^aIndicates figure showing location at which measurement was made.

TABLE 9. TRITIUM AIR SAMPLES

Sample Number	Room Number	Collection Time (min)	Moisture Content (g H ₂ O/m ³ air)	Tritium Air Concentration	
				(μ Ci/m ³)	(pCi/l)
T-1	C-35	191	8.00	BDL ^a	BDL
T-2	C-23	188	7.50	7.1×10^{-5}	0.071
T-3	C-7	120	7.50	BDL	BDL
T-4	C-16	120	8.50	BDL	BDL
T-5	Vault	120	8.25	8.5×10^{-4}	0.85
T-6	Tunnel	120	6.50	3.0×10^{-4}	0.30
T-7	B-238C	140	8.25	BDL	BDL
T-8	B-242	135	8.25	4.8×10^{-4}	0.48

^aBDL = Below Detectable Limits. The number of counts on the liquid scintillation detector was at or below the number accumulated from a sample of distilled water in the scintillation cocktail. The concentration is, therefore, < 0.04 pCi/l.

TABLE 10

GAMMA SPECTRAL AND URANIUM FLUOROMETRIC ANALYSES OF SAMPLES

Sample Number	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 $\pm 10\%^b$
BACKGROUND ENVIRONMENTAL SOIL					
15-SB19-A	0.36 $\pm$ 0.04	0.44 $\pm$ 0.06	0.49 $\pm$ 0.05	1.0 $\pm$ 0.1	0.7 $\pm$ 0.1
15-SB19-B	0.27 $\pm$ 0.04	0.49 $\pm$ 0.07	0.56 $\pm$ 0.06	1.2 $\pm$ 0.1	0.8 $\pm$ 0.1
15-SB19-C	0.11 $\pm$ 0.03	0.62 $\pm$ 0.06	0.60 $\pm$ 0.06	1.0 $\pm$ 0.1	0.7 $\pm$ 0.1
15-SB19-D	<0.02	0.59 $\pm$ 0.06	0.47 $\pm$ 0.05	1.0 $\pm$ 0.1	0.7 $\pm$ 0.1
15-SB20-A	0.53 $\pm$ 0.05	0.38 $\pm$ 0.08	0.52 $\pm$ 0.05	0.9 $\pm$ 0.1	0.6 $\pm$ 0.1
15-SB20-B	0.24 $\pm$ 0.03	0.48 $\pm$ 0.05	0.54 $\pm$ 0.05	0.9 $\pm$ 0.1	0.6 $\pm$ 0.1
15-SB20-C	0.06 $\pm$ 0.02	0.56 $\pm$ 0.06	0.51 $\pm$ 0.05	1.0 $\pm$ 0.1	0.6 $\pm$ 0.1
15-SB20-D	<0.02	0.32 $\pm$ 0.06	0.41 $\pm$ 0.04	0.9 $\pm$ 0.1	0.6 $\pm$ 0.1
15-SB21-A	0.08 $\pm$ 0.03	0.57 $\pm$ 0.06	0.54 $\pm$ 0.05	1.2 $\pm$ 0.1	0.8 $\pm$ 0.1
15-SB21-B	0.17 $\pm$ 0.03	0.61 $\pm$ 0.06	0.64 $\pm$ 0.06	1.2 $\pm$ 0.1	0.8 $\pm$ 0.1
15-SB21-C	0.04 $\pm$ 0.02	0.59 $\pm$ 0.06	0.60 $\pm$ 0.06	1.3 $\pm$ 0.1	0.9 $\pm$ 0.1
15-SB21-D	<0.02	0.53 $\pm$ 0.05	0.73 $\pm$ 0.07	1.3 $\pm$ 0.1	0.9 $\pm$ 0.1
15-SB22-A	0.24 $\pm$ 0.04	0.63 $\pm$ 0.06	0.82 $\pm$ 0.08	1.3 $\pm$ 0.1	0.9 $\pm$ 0.1
15-SB22-B	0.25 $\pm$ 0.04	0.66 $\pm$ 0.07	0.77 $\pm$ 0.08	1.3 $\pm$ 0.1	0.9 $\pm$ 0.1
15-SB22-C	0.08 $\pm$ 0.03	0.74 $\pm$ 0.07	0.77 $\pm$ 0.08	1.3 $\pm$ 0.1	0.9 $\pm$ 0.1
15-SB22-D	0.03 $\pm$ 0.02	0.62 $\pm$ 0.06	0.65 $\pm$ 0.07	1.2 $\pm$ 0.1	0.8 $\pm$ 0.1
ENVIRONMENTAL SOIL					
15-S23-A	0.17 $\pm$ 0.03	0.51 $\pm$ 0.06	0.69 $\pm$ 0.07	1.3 $\pm$ 0.1	0.9 $\pm$ 0.1
15-S23-B	0.13 $\pm$ 0.04	0.56 $\pm$ 0.06	0.75 $\pm$ 0.08	1.4 $\pm$ 0.1	1.0 $\pm$ 0.1
15-S23-C	0.12 $\pm$ 0.04	0.59 $\pm$ 0.06	0.77 $\pm$ 0.08	1.4 $\pm$ 0.1	1.0 $\pm$ 0.1
15-S23-D	<0.02	0.72 $\pm$ 0.07	0.80 $\pm$ 0.08	1.4 $\pm$ 0.1	1.0 $\pm$ 0.1
15-S24-A	0.44 $\pm$ 0.04	0.63 $\pm$ 0.06	0.65 $\pm$ 0.07	1.3 $\pm$ 0.1	0.9 $\pm$ 0.1
15-S24-B	0.50 $\pm$ 0.05	0.56 $\pm$ 0.06	0.70 $\pm$ 0.07	1.2 $\pm$ 0.1	0.8 $\pm$ 0.1
15-S24-C	0.59 $\pm$ 0.06	0.60 $\pm$ 0.06	0.82 $\pm$ 0.08	1.4 $\pm$ 0.1	1.0 $\pm$ 0.1
15-S24-D	<0.02	0.86 $\pm$ 0.09	0.70 $\pm$ 0.07	1.3 $\pm$ 0.1	0.9 $\pm$ 0.1
15-S25-A	0.02 $\pm$ 0.01	0.38 $\pm$ 0.06	0.68 $\pm$ 0.07	1.6 $\pm$ 0.2	1.1 $\pm$ 0.1
15-S25-B	0.16 $\pm$ 0.04	0.65 $\pm$ 0.07	0.69 $\pm$ 0.07	1.6 $\pm$ 0.2	1.1 $\pm$ 0.1
15-S25-C	0.15 $\pm$ 0.04	0.40 $\pm$ 0.06	0.60 $\pm$ 0.06	1.2 $\pm$ 0.1	0.8 $\pm$ 0.1
15-S25-D	0.06 $\pm$ 0.02	0.40 $\pm$ 0.06	0.45 $\pm$ 0.05	1.0 $\pm$ 0.1	0.7 $\pm$ 0.1
15-S26-A	0.14 $\pm$ 0.03	0.29 $\pm$ 0.06	0.40 $\pm$ 0.04	1.2 $\pm$ 0.1	0.8 $\pm$ 0.1
15-S26-B	0.07 $\pm$ 0.03	0.29 $\pm$ 0.06	0.49 $\pm$ 0.05	1.2 $\pm$ 0.1	0.8 $\pm$ 0.1
15-S26-C	0.16 $\pm$ 0.04	0.46 $\pm$ 0.05	0.62 $\pm$ 0.06	1.3 $\pm$ 0.1	0.9 $\pm$ 0.1
15-S26-D	0.09 $\pm$ 0.04	0.50 $\pm$ 0.05	0.67 $\pm$ 0.07	1.4 $\pm$ 0.1	1.0 $\pm$ 0.1

TABLE 10 - (cont'd.)

Sample Number	Gamma Spectra, pCi/g $\pm\sigma^a$				
	^{137}Cs	^{232}Th Decay Chain	^{226}Ra Decay Chain	Uranium Fluorometric	
				$\mu\text{g/g}\pm 10\%$	pCi/g $\pm 10\%^b$
MISCELLANEOUS SAMPLES					
15-Ash-1	BDL ^c	BDL	BDL	3.6 $\pm$ 0.4	2.4 $\pm$ 0.2
15-Bench top-2	<0.02	<0.04	<0.02	0.5 $\pm$ 0.05	0.3 $\pm$ 0.03
15-Bench top-3	0.12 $\pm$ 0.04	<0.04	<0.02	0.5 $\pm$ 0.05	0.3 $\pm$ 0.03
15-Bench top-34	< .02	< .04	0.44 $\pm$ 0.04	0.5 $\pm$ 0.05	0.3 $\pm$ 0.03
15-Bench top-35	0.02 $\pm$ 0.01	0.24 $\pm$ 0.05	0.16 $\pm$ 0.02	0.5 $\pm$ 0.05	0.3 $\pm$ 0.03
SEWER AND DRAIN SLUDGE					
15-SS5	0.25 $\pm$ 0.04	0.49 $\pm$ 0.06	0.54 $\pm$ 0.05	2.9 $\pm$ 0.3	2.0 $\pm$ 0.2
15-SS8	0.18 $\pm$ 0.04	0.61 $\pm$ 0.06	9.2 $\pm$ 0.9	3.7 $\pm$ 0.4	2.5 $\pm$ 0.3
15-SS9	0.09 $\pm$ 0.04	0.37 $\pm$ 0.07	1.3 $\pm$ 0.1	14.5 $\pm$ 1.4	9.9 $\pm$ 1.0
15-SS12	1.2 $\pm$ 0.1	0.87 $\pm$ 0.09	1.1 $\pm$ 0.1	6.4 $\pm$ 0.6	4.4 $\pm$ 0.4
15-SS13	0.02 $\pm$ 0.01	0.45 $\pm$ 0.05	0.53 $\pm$ 0.05	0.9 $\pm$ 0.1	0.6 $\pm$ 0.1
15-SS16	<0.02	0.34 $\pm$ 0.07	0.53 $\pm$ 0.05	1.8 $\pm$ 0.2	1.2 $\pm$ 0.1
15-SS17	<0.02	0.39 $\pm$ 0.08	0.48 $\pm$ 0.05	2.5 $\pm$ 0.3	1.7 $\pm$ 0.2
15-SS27	<0.02	3.6 $\pm$ 0.4	0.24 $\pm$ 0.02	300 $\pm$ 30	205 $\pm$ 20
15-SS31	0.66 $\pm$ 0.07	6.7 $\pm$ 0.7	32 $\pm$ 3	1650 $\pm$ 165	1130 $\pm$ 113
SEWER AND DRAIN WATER - SUSPENDED SOLIDS					
15-SW4	0.02 $\pm$ 0.01	0.55 $\pm$ 0.06	0.69 $\pm$ 0.07	2.1 $\pm$ 0.2	1.4 $\pm$ 0.1
15-SW6	0.34 $\pm$ 0.03	<0.04	<0.02	54.3 $\pm$ 5.4	37.1 $\pm$ 3.7
15-SW7	<0.02	<0.04	0.49 $\pm$ 0.05	1.7 $\pm$ 0.2	1.2 $\pm$ 0.1
15-SW10 _d	0.19 $\pm$ 0.04	<0.04	5.19 $\pm$ 0.52	19.0 $\pm$ 1.9	13.0 $\pm$ 1.3
15-SW11	0.22 $\pm$ 0.04	4.22 $\pm$ 0.42	0.46 $\pm$ 0.05	273.0 $\pm$ 27.3	186.0 $\pm$ 18.6
15-SW14	0.26 $\pm$ 0.05	<0.04	<0.02	18.1 $\pm$ 1.8	12.4 $\pm$ 1.2
15-SW15	<0.02	<0.04	<0.02	5.4 $\pm$ 0.5	3.7 $\pm$ 0.4
15-SW18	<0.02	0.15 $\pm$ 0.03	1.15 $\pm$ 0.12	1.8 $\pm$ 0.2	1.2 $\pm$ 0.1
15-SW28	0.10 $\pm$ 0.03	1.31 $\pm$ 0.13	<0.02	1585.0 $\pm$ 158.5	1083.0 $\pm$ 108.3
15-SW29	0.83 $\pm$ 0.08	3.43 $\pm$ 0.34	0.28 $\pm$ 0.03	90.2 $\pm$ 9.0	61.6 $\pm$ 6.2
15-SW30	<0.02	2.11 $\pm$ 0.21	0.24 $\pm$ 0.02	20.3 $\pm$ 2.0	13.9 $\pm$ 1.4
15-SW32	<0.02	<0.04	2.03 $\pm$ 0.20	10.4 $\pm$ 1.0	7.1 $\pm$ 0.7
15-SW33	9.37 $\pm$ 0.94	<0.04	<0.02	22.2 $\pm$ 2.2	15.2 $\pm$ 1.5

TABLE 10. - (cont'd.)

Gamma Spectra, pCi/ℓ±σ ^a					
Sample Number	¹³⁷ Cs	²³² Th	²²⁶ Ra	Uranium Fluorometric	
		Decay Chain	Decay Chain	μg/ℓ±10%	pCi/ℓ±10% ^b
SEWER AND DRAIN WATER - DISSOLVED SOLIDS					
15-SW4	2.90±0.58	13.1 ±2.6	<0.2	0.84 ± 0.08	0.57 ± 0.06
15-SW6	<0.2	<0.4	10.5 ±2.2	2.7 ± 0.3	1.8 ± 0.2
15-SW7	<0.2	<0.4	<0.2	0.048± 0.005	0.033± 0.003
15-SW10	<0.2	<0.4	<0.2	0.076± 0.008	0.052± 0.005
15-SW11	<0.2	<0.4	<0.2	0.35 ± 0.04	0.24 ± 0.02
15-SW14	27 ±6	<0.4	<0.2	0.091± 0.009	0.062± 0.006
15-SW15	<0.2	18.3 ±3.6	<0.2	0.045± 0.005	0.031± 0.003
15-SW18	<0.2	<0.4	<0.2	0.54 ± 0.05	0.37 ± 0.04
15-SW28	9.1±1.8	<0.4	<0.2	447.0 ±44.7	305 ±30.5
15-SW29	<0.2	<0.4	<0.2	2.0 ± 0.2	1.4 ± 0.1
15-SW30	<0.2	<0.4	<0.2	28.0 ± 2.8	19.0 ± 1.9
15-SW32	11.0±2.2	<0.4	<0.2	0.17 ± 0.02	0.12 ± 0.01
15-SW33	<0.2	<0.4	<0.2	0.48 ± 0.05	0.33 ± 0.03

^aOne standard deviation due to counting statistics.

^bANL conversion factor of 0.683 pCi/ μg from Appendix 5.

^cBDL = Below Detectable Limit. The continuum from the major contaminants mask the concentrations of these isotopes. As of September 20, 1984, there was:

Isotope	pCi/g $\pm\sigma$
¹¹³ Sn- ¹¹³ In	1400 $\pm$ 140
⁴⁶ Sc	1300 $\pm$ 130
⁵⁴ Mn	620 $\pm$ 62
¹⁵³ Gd	360 $\pm$ 72
⁶⁵ Zn	24 $\pm$ 4

^dThis sample also contains these activities at count time October 1, 1984, 1000 hours:

Isotope	pCi/g $\pm 1\sigma$
¹⁵² Eu	5.01 $\pm$ 1.0
^{110m} Ag	0.93 $\pm$ 0.19
⁶⁰ Co	0.77 $\pm$ 0.15

TABLE 11

ESTIMATED VOLUME, MASS, AND ACTIVITY OF MATERIAL
THAT COULD BE GENERATED BY REMEDIAL ACTION

Type of Material	<u>Estimated Volume</u>		<u>Estimated Mass</u>		<u>Estimated Activity</u> μCi
	m^3	ft^3	kg	lbs (avoir)	
Concrete ($\rho=2.35$)	0.26	9.0	610	1340	17.4
Brick ($\rho=2.2$)	0.16	5.7	350	770	18.4
Wood ($\rho=0.5$)	0.054	1.9	27	60	1.7
Steel ($\rho=7.8$)	0.020	0.7	160	350	0.5
Asphalt ($\rho=1.2$)	0.046	1.6	55	120	10.0
Total	0.54 m^3	18.9 ft^3	1202 kg	2640 lb	48.0 μCi

APPENDICES

Appendices 1 through 5 and 7 contain detailed descriptions of the array of instruments and computational and analytical procedures typically employed by ANL's Radiological Survey Group in its comprehensive radiological assessments. Although the specific instruments and techniques used in a given survey depend on the conditions encountered and the information sought, descriptions of the entire array have been included here for completeness. The exact instruments and methods used in the survey reported in this document are specified in appropriate discussions in the text.

Appendix 6 contains excerpts from numerous regulations, standards, and guidelines relative to radiological conditions and exposure to radiation. Not all these necessarily apply at each site surveyed. Again, however, all have been included for completeness. The pertinent regulations, standards, and guidelines for this survey are cited in the text.

Appendix 8 contains a generic discussion of the nature and sources of radiation, its potential danger to humans, and methods utilized to evaluate radiation exposures.

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 in mR/h with a ²²⁶Ra standard.

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 high voltages 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 (3 ft.) from the surface during the survey. This instrument is also calibrated with a standard ^{226}Ra source and checked daily with a ^{137}Cs - ^{137m}Ba source.

E. Integrating Radiation Meter

In addition to the PRM-7, a pressurized ion chamber (Reuter Stokes Model RSS-111) was used at selected locations to determine the ambient radiation field. The RSS-111 has three output modes: (1) instantaneous exposure rate, (2) strip chart differential readout, and (3) integrated exposure. 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 3-h period of operation is usually sufficient to obtain significant data.

II. SMEAR-COUNTING INSTRUMENTATIONA. Large Area Counter

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 ten samples can be counted simultaneously.

B. Proportional Counters

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. This counter also uses P-10 counting gas. Smears are counted in both the alpha and beta modes.

C. Calibration Procedures

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.

APPENDIX 1 (cont'd.)

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.

D. Liquid Scintillation Counter

In areas where low-energy beta emitting nuclides (e.g., ^3H and ^{14}C) are known to have been used, the smear survey is performed using expanded polystyrene. The polystyrene is dissolved in a liquid scintillation cocktail. The vial of liquid is counted in a Nuclear Enterprises, Inc., Liquid Scintillation Counter LSC-2 designed to measure picocurie quantities of these nuclides. Dual channels with associated single-channel analyzers provide capability for counting beta spectra at two energy regions simultaneously.

III. AIR-SAMPLING DEVICE

A. ANL Radon Progeny Pump (Filter Queen)

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 m³/h. The filters consist of 200-cm² sheets of Hollingsworth-Vose (HV-70) or LB5211-9 mil filter paper. The collection efficiency at these flow rates for 0.3- μm particles is about 99.9%.

B. Actinon Pump

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 μm) 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.

C. Tritium

A cold trap was set up with dry ice and methanol. An air sample was taken with a positive displacement pump drawing about 20 l/min through the cold trap for two to three hours. The water vapor from the air condenses on the sides of the cold trap. At the end of sampling, a 1 m ℓ aliquot of the condensed water vapor and 10 m ℓ of scintillation cocktail are mixed in a 20 m ℓ counting vial.

The windows on the Liquid Scintillation Counter (LSC-2) are set to count the beta particles from ^{14}C and ^3H . The samples in the vials are counted for four minutes after they have settled and cleared.

The relative humidity reading is measured with a sling psychrometer. The wet and dry bulb readings are recorded.

APPENDIX 1
(cont'd.)

IV. GAMMA SPECTRAL INSTRUMENTATION

A. Contamination Identification

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) lead shielded scintillation crystal is commonly used for determining a gamma spectrum. The crystal and lead shielding are located inside the radiological survey vehicle. This instrument is calibrated with NBS-traceable gamma sources. This system can be used to identify contaminant radionuclides by analyzing the gamma-rays emitted by samples from contaminated areas.

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).

B. Borehole Logging

The gamma-ray spectrum of a borehole is logged using the Bicron 5.1 cm (2 in) by 5.1 cm (2 in) NaI crystal and either the ND-66 or the ND-100. This crystal is specially designed to withstand the temperature changes that can be encountered in boreholes. Permanent records of the spectrum are produced by the teletype printer and paper tape punch or the Centronix Printer and Magnetic Tape Drive.

C. Portable Analyzer

A Nuclear Data portable multichannel analyzer, ND-6, utilizing a 5 cm diameter by 5 cm thick (2 in by 2 in) NaI(Tl) scintillation crystal is commonly used for determining the gamma-ray spectrum of items or areas that cannot be taken to the mobile laboratory. This instrument is battery operated and has both internal storage capabilities and a tape cassette. This instrument is also calibrated with NBS-traceable gamma-ray sources and is used to identify contaminant radionuclides from their emitted gamma-rays.

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	±.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 to 100 cm² = 1.00 times PAC reading.

. 2π Internal Gas-Flow Counter, PC counter

Geometry: Solid Steel Spun Top - 0.50

APPENDIX 2 (cont'd.)

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 counters 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 cts/min 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 normal uranium source was also counted with the PAC instruments 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 .

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

APPENDIX 2
(cont'd.)

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)

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

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 ²³⁹Pu (0.713) is used.

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

B. Conversion Equation (Beta)

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

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

A backscatter factor (bf) of 1.1 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 ⁹⁰Sr-⁹⁰Y (0.85) is used.

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

APPENDIX 2
(cont'd.)

III. TRITIUM SMEAR COUNT

The conversion equation for cts/min-100 cm² to dis/min-100 cm² for popcorn smear counts is given below:

$$C_e = \frac{A_e - B}{Y}$$

where: C_e = dis/min removable contamination
 A_e = measured counts/min of removable contamination
 B = measured background of clean popcorn dissolved in the scintillation cocktail
 Y = combined yield of the counter and scintillation cocktail as given in more detail in Appendix 3, Section VA.

For the Nuclear Enterprises, Inc., Liquid Scintillation Counter (LSC-2), the high voltage is set to 900 Volts. The A channel threshold is set at 360 and the window is set at 999 for ¹⁴C. The B gain is set at times ten (x10). The B channel threshold is set at 150 and the window is set at 825 for ³H.

APPENDIX 3

AIRBORNE RADIONUCLIDE CONCENTRATION CALCULATIONS

Calculations for determining radon concentrations in air samples collected with an Argonne National Laboratory-designed air sampler using HV-70 or LB-5211 filter media are summarized in Sections I through IV of this appendix; the basic assumptions and calculations used to derive the air concentrations also are included. Calculations for determining tritium concentrations in air samples collected by condensing water vapor from the air are summarized in Section V of this appendix.

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 collected on the filter media, (i.e., unattached fraction).
- 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 period. 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 (minutes).

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 (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 LB-5211 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

APPENDIX 3 (cont'd.)

sample rate is approximately 20 liters/minute for a sampling time of at least 90 minutes. The center portion of the sample is removed and counted in an alpha spectrometer which exhibits the 6.62 MeV AcC alpha emissions and the 7.69 MeV RaC' alpha emissions. If these two peaks are observed in the spectrum, then the following calculations are performed:

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

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 applications of the equations for determining all types of activity and their concentrations are given below:

Data	$f = 40 \text{ m}^3/60 \text{ min}$	$t = 40 \text{ min}$
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 filter paper, the above concentrations are then doubled to determine actual concentrations in the air. We assume that there is no unattached fraction for the long-lived activity.

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

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

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

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

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

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 :

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

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

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

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

APPENDIX 3 (cont'd.)

V. TRITIUM CONCENTRATIONS

It is assumed that all the tritium in the air is part of a water molecule and that all the water vapor is collected in the cold trap.

A. Yield

A secondary tritium standard is made by taking a primary standard of known concentration and diluting it with distilled water to make a secondary standard of approximately 5×10^4 dis/min. The activity of the secondary standard is calculated from the dilution ratio and by using the decay equation:

$$I = I_0 e^{-\lambda t}$$

where: I_0 = dis/min of the primary standard at the date indicated for this standard.

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

$t_{1/2}$ = half-life of ^3H or 12.28 years.

t = time interval from the primary standard date to the time of mixing the secondary standard.

I = dis/min of the secondary standard at the time it was mixed with distilled water.

The scintillation standard is mixed at the time of use by taking a 1 ml aliquot of the secondary standard and dissolving it in 10 ml of the scintillation cocktail. The activity of this scintillation standard is calculated using the decay equation above. Now, however,

I_0 = dis/min of the secondary standard at the time of mixing.

t = time interval from mixing the secondary standard to mixing the scintillation standard.

I = dis/min of the scintillation standard at the time it was mixed with the cocktail.

The combined yield of the counter and the scintillation cocktail are determined using the following equation.

$$Y = \frac{As - B}{I}$$

APPENDIX 3 (cont'd.)

where: Y = yield

As = measured counts of a tritium standard

B = measured background

I = dis/min of the standard in the scintillation cocktail at the time of calibration as calculated from the decay equation given above.

B. Humidity

The grams of water per cubic meter of dry air (ϵ) are determined from the composite humidity chart (see Fig. 3A).

C. Tritium

The activity of 1 mℓ of the condensate is then:

$$C_e = \frac{A_e - B}{Y}$$

where: C_e = dis/min-mℓ of the water vapor

A_e = measured counts of the water vapor sample

B = measured background of distilled water dissolved in the scintillation cocktail.

The concentration of tritium in the air is then:

$$C_a = \frac{C_e \times \epsilon}{\rho \times 2.22 \text{ dis/min-pCi}}$$

where: C_a = $\mu\text{Ci/m}^3$

ϵ = humidity ($\text{gmH}_2\text{O/m}^3$ dry air)

ρ = water density (1 gm/mℓ)

D. Example

The secondary standard was mixed April 5, 1984, in the liquid scintillation cocktail. The activity of the scintillation standard was calculated and the information recorded on the scintillation vial as indicated:

APPENDIX 3
(cont'd.)

Data on April 5, 1984: $I_o = 4.14 \times 10^4$ dis/min of the tritium scintillation standard at the time it was prepared.

Data on May 1, 1984: $A_s = 8854.75$ cts/min from the tritium standard

$B = 29.25$ cts/min background

Wet bulb = 56°F, Dry Bulb = 68°F

$A_e = 78$ cts/min from the water vapor sample

Source strength at the time the sample was collected:

$$I = 4.14 \times 10^4 \times \exp - \frac{0.693 \times 0.0712}{12.28} = 4.12 \times 10^4$$

Combined counter and scintillation cocktail yield:

$$Y = \frac{8854.75 - 29.25}{4.12 \times 10^4} = 0.214$$

Humidity: $\epsilon = 8.25$ g/m³

Concentration in a 1 ml aliquot of the condensate:

$$C_e = \frac{78 - 29.25}{0.214} = 228 \text{ dis/min-ml.}$$

Concentration in the air: $C_a = \frac{228 \times 8.25}{1 \times 2.22} = 847 \text{ pCi/m}^3$

$$847 \text{ pCi/m}^3 \times 10^{-3} \text{ m}^3/\ell = 0.85 \text{ pCi}/\ell.$$

Reference

1. Weast, R.C. (Editor-in-Chief). 1959. Handbook of Chemistry and Physics, 59th Edition (1978-1979) D-232, E-45, F11.

ANL-HP DWG. NO. 84-81

Figure 3A Absolute Humidity Chart.

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 from the sieved material. 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 grams of this are needed for radiochemical analysis for Pu, Am, and Th if these analyses are required. The bottles containing these weighed samples are marked with sample number and date, and

APPENDIX 4 (cont'd.)

this information is recorded in the soil-sample book. The rocks (and dross) and remaining soil are placed in storage.

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. Room air is drawn through the 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 sus-

APPENDIX 4 (cont'd.)

pended and dissolved solids are analyzed by appropriate radiochemical analytical techniques.

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 fraction from each soil sample is analyzed by high resolution gamma-ray spectroscopy using a germanium crystal detector coupled to a multichannel analyzer. This analysis allows for a quantitative determination of the 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 (using a ²³⁴Th spike for yield determination) followed by plating a thin source of the

APPENDIX 4
(cont'd.)

radiochemically separated thorium and determining the thorium isotopes (^{228}Th and ^{232}Th) by alpha spectroscopy.

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), radiothorium (^{228}Th decay chain), plutonium (^{238}Pu , $^{239,240}\text{Pu}$), 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 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-NCS-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	2.446×10^5	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 $(\text{wt } \%)_i =$

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

Specific activity for normal uranium:

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

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, uranium having the isotopic composition (atom %) 238(0.99268), 236(0.000007), 235(0.007233), 234(0.000064) and 233(0.000012) as determined by mass spectrometry would have the composition (weight %) of 238(.99278), 236(0.000007), 235(0.007142), 234(0.000063), and 233 (0.000012) and the following specific activity:

APPENDIX 5
(cont'd.)

$$\begin{aligned}
 0.99278 \times 0.746 &= 0.7406 \text{ dis/min-}\mu\text{g from } ^{238}\text{U} \\
 0.000007 \times 1.436 \times 10^2 &= 0.0010 \text{ dis/min-}\mu\text{g from } ^{236}\text{U}^* \\
 0.007142 \times 4.798 &= 0.0343 \text{ dis/min-}\mu\text{g from } ^{235}\text{U} \\
 0.000063 \times 1.387 \times 10^4 &= 0.8738 \text{ dis/min-}\mu\text{g from } ^{234}\text{U} \\
 0.000012 \times 2.140 \times 10^4 &= 0.2568 \text{ dis/min-}\mu\text{g from } ^{233}\text{U}^{**} \\
 &\quad 1.9065 \text{ dis/min-}\mu\text{g U total}
 \end{aligned}$$

corresponding to:

$$(1.9065 \text{ dis/min-}\mu\text{g}) / (2.22 \text{ dis/min-pCi}) = 0.859 \text{ pCi}/\mu\text{g U}$$

*The 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).

**The half-life for ^{233}U (1.5911×10^5 yr) and the corresponding specific activity (2.140×10^4 dis/min- μg) were taken from A. H. Jaffey, K. F. Flynn et. al., Phys. Rev. C 9 1991 (1974).

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.)

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 1

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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(cont'd.)

TABLE 1

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.

APPENDIX 6
(cont'd.)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,

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Division of Safety, Standards, and Compliance, U.S. Department of Energy, Washington, D.C. 20545.

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:

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(cont'd.)

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

(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.

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(cont'd.)

- (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.
- (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 involved 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.

APPENDIX 6
(cont'd.)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 Sites192.10 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 a Federal agency, or

(2) a license [issued 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 derived 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

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

^bLimits are to apply to only one nuclide present in the soil. If more than one is present, a weighted average should apply.

^cBased on a diet of a home gardener.

^dModified from LA-UR-79-1865-Rev. values to correct error.

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- VI. DOE 5480.1 Chg. 6, Chapter XI
(Excerpts have been retyped for purposes of this report.)

REQUIREMENTS FOR RADIATION PROTECTION

(Excerpts have been retyped for purposes of this report. Table XI-1 has been modified to reflect the fundamental definition of the Curie as a unit of radioactivity. The designation of uranium is changed to reflect the original usage of the terms normal and natural.)

1. PURPOSE This chapter establishes radiation protection standards and requirements for Department of Energy and Department of Energy contractor operations based upon the recommendations of the Environmental Protection Agency and the National Council on Radiation Protection and Measurement.
2. DEFINITIONS
 - a. Controlled Area. Any area to which access is controlled in order to protect individuals from exposure to radiation and radioactive materials.
 - b. Dose Commitment. The dose equivalent (rem) received by specific organs during a period of one calendar year, that was the result of uptakes of radionuclides by a person occupationally exposed.
4. REQUIREMENTS.
 - a. Occupationally Related Exposure of Individuals in Controlled Areas. Radiation exposures shall be limited to levels reasonably achievable within the standards prescribed below.
 - (1) Radiation Protection Standards for External and Internal Exposures

<u>Type of Exposure</u>	<u>Exposure Period</u>	<u>Dose Equivalent (Dose or Dose Commitment¹ rem)</u>
Whole body, head and trunk, gonads, lens of the eye ² , red bone marrow, active blood-forming organs.	Year	5 ³
	Calendar Quarter	3

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Unlimited areas of the skin (except hands and forearms). Other organs, tissues, and organ systems (except bone).	Year	15
	Calendar Quarter	5
Bone	Year	30
	Calendar Quarter	10
Forearms ⁴	Year	30
	Calendar Quarter	10
Hands ⁴ and Feet	Year	75
	Calendar Quarter	25

¹To meet the above dose commitment standards, operations must be conducted in such a manner that it would be unlikely that an individual would assimilate in a critical organ, by inhalation, ingestion, or absorption, a quantity of radionuclide or mixture of radionuclides that would commit the individual to an organ dose that exceeds the limits specified in the above table.

²A beta exposure below a maximum energy of 700 keV will not penetrate the lens of the eye; therefore, the applicable limit for these energies would be that for the skin (15 rem/year).

³In special cases, with the approval of EP-30, a worker may exceed 5 rem/year, provided his or her average exposure per year since age 18 will not exceed 5 rem per year. This does not apply to emergency situations.

⁴All reasonable effort shall be made to keep exposures of forearms and hands to the general limit for the skin.

(2) Procedural Requirements

(a) Restrictions

- 1 An individual under age 18 shall neither be employed in, nor allowed to enter, controlled areas in such a manner that he or she will receive doses of radiation in amounts exceeding one-tenth the standards in paragraph 4a(1) above.

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2 Dose to students under age 18 exposed to radiation during educational activities shall not exceed 0.1 rem/year. This exposure shall be considered a part of the 0.5 rem/year limit for workers under age 18 and not supplemental to it.

(b) Combining Internal and External Dose. Current year whole body internal dose commitment from radionuclides for which the whole body is the critical organ must be combined with the external whole body dose. Where both the external penetrating dose and internal dose to critical organ are known, they shall be combined for that organ.

b. 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.

(1) Radiation Protection Standards for External and Internal Exposure.

Type of Exposure	Annual Dose Equivalent or Dose Commitment ¹	
	Based on Dose to Individuals at Points of Maximum Probable Exposure	Based on Average Dose to a Suitable Sample of the Exposed Population ²
Whole body, gonads, or bone marrow	0.5 rem (or 500 mrem)	0.17 rem (or 170 mrem)
Other organs (or 1500 mrem)	1.5 rem (or 500 mrem)	0.5 rem

¹In keeping with Department of Energy policy on lowest practicable exposures, exposures to the public shall be limited to as small a fraction of the respective annual dose limits as is reasonably achievable.

²See Paragraph 5.4, Federal Radiation Council Report No. 1, for discussion on concept of suitable sample of exposed population.

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- (2) Monitoring Requirements. To assure that doses to the public are maintained as low as reasonably achievable consistent with dose standards set forth in paragraph 4b(1) above, effluents to the environment, and other parameters shall be monitored and documented in accordance with DOE 5484.1.
- (3) Concentration Guides.
- (a) Concentration Guides in Attachment XI-1, Table II, were derived for the most part from the dose standards for individuals in paragraph 4b(1) above (assume 168 hours of exposure per week). These guides shall be reduced by a factor of three when applied to a suitable sample of the population. Where transient exposures can be calculated, the concentration guides other than those in Attachment XI-1, Table II, may be used to evaluate compliance with the dose commitment standard.
 - (b) There may be situations where it is not feasible or desirable to evaluate the exposure of individuals and samples of exposed populations to effluents to assure compliance with standards in paragraph 4b(1) above. In those cases, effluent releases to uncontrolled areas shall be such that average concentrations of radioactivity at the point of release are within the concentration guides and are as low as reasonably achievable. The point of release shall be considered to be the point at which the effluents pass beyond the site boundary. Radioactivity concentrations may be averaged over periods up to 1 year.
- (4) Further Limitations on Effluent Discharges. In any situation in which the effluents discharged by one or more activities of the Department, Department contractors, or others cause exposure to approach the standards specified in subparagraph b(1) above, appropriate effluent discharge limits may be set for these operations. In such cases, the manager of the field organization may take the necessary corrective action if all activities concerned are within his or her area of responsibility. Otherwise, each case will be referred to EP-30 for appropriate action including, where appropriate, coordination with actions taken by the Nuclear Regulatory Commission under the Code of Federal Regulations, Title 10, Part 20.106(e).
- (5) Discharge to Sanitary Sewage Systems.
- (a) Effluents may be discharged to public sanitary sewage systems provided:

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- 1 The quantity of radioactivity released in any one month, if diluted by the average monthly quantity of water released by the installation, will not result in an average concentration exceeding the concentration guide in Attachment XI-1, Table I, Column 2.
- 2 The radiation protection standards in paragraph (1), above, are not exceeded.
- (b) Concentrations or quantities of radioactive materials greater than those specified in paragraphs 4(b) and (5)(a) 1 and 2, above, may be released to chemical or sanitary sewage systems owned by the Federal Government provided the standards in paragraph 4b(1) above are not exceeded in uncontrolled areas.

Attachment XI-1

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 Air (pCi/l)	Column 2 Water (pCi/l)	Column 1 Air (pCi/l)	Column 2 Water (pCi/l)
Hydrogen (1)	H	3	S	5000	1×10^8	200	3×10^6
			I	5000	1×10^8	200	3×10^6
Cobalt (27)	Co	60	S	300	1×10^6	10	5×10^4
			I	9	1×10^6	0.3	3×10^4
Silver (47)	Ag	110m	S	200	9×10^5	7	3×10^4
			I	10	9×10^5	0.3	3×10^4
Cesium (55)	Cs	137	S	60	4×10^5	2	2×10^4
			I	10	1×10^6	0.5	4×10^4
Europium (63)	Eu	152	S	10	2×10^6	0.4	8×10^4
			I	20	2×10^6	0.6	8×10^4
Lead (82)	Pb	212	S	20	6×10^5	0.6	2×10^4
			I	20	5×10^5	0.7	2×10^4
Radon (86)	Rn	220	S	300		10	
		222	I	100		3	
Radium (88)	Ra	226	S	0.05	400	0.003	30
			I	0.05	9×10^5	0.002	3×10^4

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Thorium (90) ¹ natural	S	0.06	2×10^4	0.002	2000
	I	0.06	6×10^5	0.002	2×10^4
Uranium (92) ² normal	S	0.14	4×10^4	0.006	1200
	I	0.12	1×10^6	0.004	4×10^4

¹The Curie is the activity of that quantity of radioactive material in which the number of disintegrations per second is 3.7×10^{10} . In DOE Order 5480.1, Chapter XI, the Curie was redefined to 3.7×10^{10} dis/sec from Th-232 plus 3.7×10^{10} dis/sec from Th-228. In order to make the above table consistent with the real definition of Curie, the concentrations for natural thorium were multiplied by a factor of 2.

²In this report, the terms natural uranium and normal uranium are used as they were commonly used in the early days of MED work. Natural uranium is uranium as found in nature in equilibrium with all its daughter products. Normal uranium is uranium that has been processed to separate it from its long-lived daughter products (i.e., the daughter products Th-230 through Pb-206).

In Doe Order 5480.1 Chapter XI the Curie is redefined as 3.7×10^{10} dis/sec from U-238 plus 3.7×10^{10} dis/sec from U-234 plus 1.7×10^9 dis/sec from U-235. This is also called natural uranium in these regulations. In order to make the above table consistent with the real definition of Curie, the concentrations for uranium were multiplied by a factor of 2.046. Also, the uranium is called normal uranium as the term is used elsewhere in this report and since the regulation implies that the uranium has been separated from its long-lived daughter products. This is consistent with the earliest use of these terms.

APPENDIX 7

ESTIMATED EXTENT OF CONTAMINATION

I. SOIL

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

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

Estimates of the extent of contamination are usually determined for averaged (Option 1) and maximum or worst-case (Option 2) conditions. Sample calculations for the extent of contamination in the Back Forty area of the Albany, Oregon Bureau of Mines Site are as follows:

$$\begin{aligned} \text{Volume (average)} &= 34,800 \text{ ft}^2 \text{ (area)} \times 3.6 \text{ ft (avg. depth)} = 125,000 \text{ ft}^3 \\ &= 3,550 \text{ m}^3 \end{aligned}$$

$$\begin{aligned} \text{Volume (maximum)} &= 34,800 \text{ ft}^2 \text{ (area)} \times 9 \text{ ft (max. depth)} = 314,000 \text{ ft}^3 \\ &= 8,880 \text{ m}^3 \end{aligned}$$

$$\text{Mass (average)} = 3,550 \text{ m}^3 \times 1,500 \text{ kg/m}^3 = 5.33 \times 10^6 \text{ kg}$$

$$\text{Mass (maximum)} = 8,880 \text{ m}^3 \times 1,500 \text{ kg/m}^3 = 1.33 \times 10^7 \text{ kg}$$

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

$$\begin{aligned} \text{Average: } &5.33 \times 10^6 \text{ kg} \times 14 \times 10^{-12} \text{ Ci/g} \times 10^3 \text{ g/kg} \times .05 \text{ (fraction)}^* \\ &= 0.004 \text{ Ci} \end{aligned}$$

$$\begin{aligned} \text{Maximum: } &1.33 \times 10^7 \text{ kg} \times 16 \times 10^{-12} \text{ Ci/g} \times 10^3 \text{ g/kg} \times .05 \text{ (fraction)}^* \\ &= 0.011 \text{ Ci.} \end{aligned}$$

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

APPENDIX 7 (cont'd.)

II. STRUCTURES

Any estimate of the total mass and volume of radioactively contaminated structural material that would be generated if remedial action were undertaken at a site is subject to many uncertainties. For the purposes of this report, it has been assumed that contamination on concrete, brick, or asphalt will require removal to a depth of 5 cm (2 in.); and contamination on wooden floors and steel will require removal to 2 cm (3/4 in.). These assumptions are believed to be conservative. The densities of the materials are taken to be 2.35 g/cm³ for concrete, 2.2 g/cm³ for brick, 0.5 g/cm³ for wood, 7.8 g/cm³ for steel, and 1.2 g/cm³ for asphalt. Estimates of the surface area contaminated are made during the survey.

It is assumed that the contamination can only be removed by removing the supporting material. The contaminated areas are summed according to host material (e.g., concrete, steel, or wood). The total surface activity was calculated from the activity (dis/min-100 cm²) as measured by the survey meter times the estimated surface area. The activity was then converted to μ Ci. No corrections were made for the depth of the contamination, therefore, the values are underestimated.

A sample calculation follows for two areas of concrete; an 8-ft² area with 7000 dis/min-100 cm² beta-gamma activity and a 2-ft² area with 77,000 dis/min-100 cm² beta-gamma activity.

$$\text{Surface area} = 10 \text{ ft}^2 = 0.93 \text{ m}^2$$

$$\text{Volume} = 0.93 \text{ m}^2 \times 0.05 \text{ m} = 4.6 \times 10^{-2} \text{ m}^3 = 1.6 \text{ ft}^3.$$

$$\text{Mass} = 4.6 \times 10^{-2} \text{ m}^3 \times 2350 \text{ kg/m}^3 = 108 \text{ kg}$$

$$\begin{aligned} \text{Activity} &= (7 \text{ k dis/min-100 cm}^2 \times 8 \text{ ft}^2 \\ &\quad + 77 \text{ k dis/min-100 cm}^2 \times 2 \text{ ft}^2) \times 929 \text{ cm}^2/\text{ft}^2 \\ &= 1.95 \times 10^6 \text{ dis/min} \\ &= 1.95 \times 10^6 / 2.22 \times 10^6 \text{ } \mu\text{Ci} = 0.88 \text{ } \mu\text{Ci} \end{aligned}$$

APPENDIX 8

EVALUATION OF RADIATION EXPOSURES

I. INTRODUCTION

A. Types of Radiation

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

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

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

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

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

B. Sources of Radiation

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

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

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on the other hand, are 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 or 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 flux density. These particles, together with photons from the gamma-emitting, naturally occurring radionuclides in the local environment,

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form the external penetrating component of the background environmental radiation field which provides a significant portion of the whole-body radiation dose to man.

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.

II. 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.

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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. 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 radionuclides 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).

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

III. 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 individual'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.

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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 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

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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.

IV. DOSE DETERMINATION CALCULATION

A. External Exposure

External penetrating radiation dose rates are measured on contact with an end-window beta-gamma Geiger-Mueller (GM) detector (7 mg/cm² window), and at 1 meter with a NaI crystal detector (1 in diameter by 1 in thick) manufactured by Eberline Corporation (PRM-7 μ R Meter). For the purpose of these calculations, the following conservative assumptions are made. First, it is assumed that the half-life of the contaminant is long and, therefore, the dose rate is constant with respect to time. Second, it is assumed that a person is stationary at the location of maximum dose for 40 hours per week for 50 weeks per year. For such a situation, the annual dose (A) for a 0.3 mR/h radiation field (about ten times normal background) would be:

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

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

B. Internal Exposure from Soil

The internal radiological hazard from inhalation/ingestion of contamination in soil is assessed by postulating hypothetical "worst case" scenarios. To this end two cases are considered. The first case is based on the situation whereby a child would eat 100 g per year of the contaminated soil. The second case assumes a home gardener would rototill the contaminated soil (dry) to a 1-ft depth for a working day (eight hours) once a year. For this latter case, a resuspension factor of 10^{-6} m^{-1} , a breathing rate of $9.6 \text{ m}^3/\text{working day}$ and a soil density of 1.5 g/cm^3 , are used. This means 4.39 g ($1 \text{ ft} \times 0.305 \text{ m/ft} \times 1.5 \text{ g/cm}^3 \times 10^6 \text{ cm}^3/\text{m}^3 \times 10^{-6} \text{ m}^{-1} \times 9.6 \text{ m}^3$) of soil are inhaled each year. In both cases it is assumed that the average concentration of contaminants in the soil is equal to the maximum measured value (a conservative assumption). All calculations are based on methods outlined in ORNL/NUREG/ TM-190, Vol. 3.⁽²⁾ These calculations approximate the ICRP-30 guidelines for hazard analysis.

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

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An example calculation is given below that is based on the above scenarios assuming a soil contamination level of 5 pCi/g of radium-226 EPA limit. This gives the following dose levels (50-year dose commitment):

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

$$\begin{aligned}\text{Bone:} & \quad 6.5 \times 10^{-2} \text{ mrem/pCi} \times 5 \text{ pCi/g} \times 100 \text{ g} = 32.5 \text{ mrem} \\ \text{Total Body:} & \quad 5.5 \times 10^{-3} \text{ mrem/pCi} \times 5 \text{ pCi/g} \times 100 \text{ g} = 2.75 \text{ mrem.}\end{aligned}$$

Inhalation (rototilling and breathing ensuing aerosol):

$$\begin{aligned}\text{Lung:} & \quad 1.1 \times 10^{-1} \text{ mrem/pCi} \times 5 \text{ pCi/g} \times 4.39 \text{ g} = 2.4 \text{ mrem} \\ \text{Bone:} & \quad 9.2 \times 10^{-2} \text{ mrem/pCi} \times 5 \text{ pCi/g} \times 4.39 \text{ g} = 2.0 \text{ mrem} \\ \text{Total Body:} & \quad 9.5 \times 10^{-3} \text{ mrem/pCi} \times 5 \text{ pCi/g} \times 4.39 \text{ g} = 0.2 \text{ mrem.}\end{aligned}$$

There is no regulation setting an allowable limit for radionuclides with both a long radiological half-life and a long biological half-life such as radium.

C. Internal Exposure from Building Materials

The internal radiological hazard from inhalation of contamination on a brick wall is assessed by postulating a hypothetical "worst-case" scenario. To this end, a case is considered whereby a person is using a portable concrete saw in the dry mode of operation to cut a brick wall for an electrical outlet. A typical saw blade used in cutting masonry would have a 30.5-cm (12-in) diameter, a 0.32-cm (1/8-in) thickness, and a 11.1-cm (4 3/8-in) cutting depth. The wobble of the blade would probably increase the width of the cut to a kerf of 0.4 cm. If the cut were 10 cm by 10 cm (4 in by 4 in), the perimeter would then be 41 cm (16 in). The overlap of the cuts upon the brick wall would extend the total span of the cuts to 89 cm (35 in). The area of the brick wall that the blade would consume is then 36 cm². It is assumed that 50% of the brick becomes airborne and respirable. The room has a total volume of 60 m³, and the dust created becomes dispersed throughout the entire room. One person is involved and the job requires one hour. The person inhales 1.2 m³ of air per hour. This means 0.36 cm² (36 cm² x 0.5 x 1.2 m³/h x 1 h/60 m³) of the brick surface would be inhaled. This calculation is also based on methods outlined in ORNL/NUREG/TM190, Vol. 3,² which approximates the ICRP 30 guidelines for hazard analysis.

An example calculation is given below that is based on the above scenario assuming a surface contamination of 3.6×10^5 dis/min-100 cm² normal uranium or 1.62×10^3 pCi/cm². This gives the following dose levels (50-year dose commitment):

Inhalation (cutting brick):

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(cont'd.)

Lung:	7.5×10^{-3}	mrem/pCi	$\times 1620$	pCi/cm ²	$\times 0.36$	cm ²	= 4.4 mrem.
Bone:	1.5×10^{-2}	mrem/pCi	$\times 1620$	pCi/cm ²	$\times 0.36$	cm ²	= 8.7 mrem.
Total Body:	5.1×10^{-1}	mrem/pCi	$\times 1620$	pCi/cm ²	$\times 0.36$	cm ²	= 300 mrem.

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 (cont'd.)

TABLE 1

FIFTY-YEAR DOSE COMMITMENT FACTORS (inhalation/ingestion)^a
 (mrem/pCi assimilated)

Nuclide ^b	Inhalation			Ingestion	
	Bone	Lung	Total Body	Bone	Total Body
²³⁸ U	7.1×10^{-3}	4.8×10^{-1}	1.5×10^{-2}	2.8×10^{-4}	2.1×10^{-5}
²³⁴ U	7.9×10^{-3}	5.4×10^{-1}	1.6×10^{-2}	3.1×10^{-4}	2.4×10^{-5}
²³⁰ Th	3.1×10^{-1}	5.3×10^{-1}	3.8×10^{-2}	1.2×10^{-3}	9.2×10^{-5}
²²⁶ Ra	4.9×10^{-2}	5.6×10^{-2}	4.7×10^{-3}	4.3×10^{-2}	3.4×10^{-3}
²¹⁰ Po	8.1×10^{-4}	4.6×10^{-2}	1.3×10^{-3}	5.2×10^{-4}	4.1×10^{-4}
²¹⁰ Pb	4.2×10^{-2}	6.2×10^{-3}	3.5×10^{-3}	2.1×10^{-2}	1.7×10^{-3}
Total Chain	4.2×10^{-1}	1.7	7.9×10^{-2}	6.6×10^{-2}	5.6×10^{-3}
²²⁶ Ra Chain	9.2×10^{-2}	1.1×10^{-1}	9.5×10^{-3}	6.5×10^{-2}	5.5×10^{-3}
²³⁸ U+ ²³⁴ U	1.5×10^{-2}	1.0	3.1×10^{-2}	5.9×10^{-4}	4.5×10^{-5}
²³⁵ U	7.2×10^{-3}	4.8×10^{-1}	1.5×10^{-2}	2.8×10^{-4}	2.2×10^{-5}
²³¹ Pa	9.6×10^{-1}	5.9×10^{-1}	1.4×10^{-1}	1.8×10^{-2}	2.1×10^{-3}
²²⁷ Ac	5.4×10^{-1}	1.0	1.0×10^{-1}	1.2×10^{-2}	1.3×10^{-3}
²²³ Ra	9.6×10^{-4}	4.6×10^{-2}	8.6×10^{-4}	1.2×10^{-3}	2.3×10^{-4}
²²⁷ Th	4.8×10^{-4}	6.9×10^{-2}	1.1×10^{-3}	2.7×10^{-5}	5.1×10^{-6}
Total Chain	1.5	2.2	2.6×10^{-1}	3.2×10^{-2}	3.7×10^{-3}
²³² Th	3.3×10^{-1}	4.5×10^{-1}	3.8×10^{-2}	1.3×10^{-3}	9.6×10^{-5}
²²⁸ Th	4.4×10^{-2}	7.2×10^{-1}	1.9×10^{-2}	4.5×10^{-4}	3.8×10^{-5}
²²⁸ Ra	2.9×10^{-2}	4.8×10^{-3}	2.5×10^{-3}	2.1×10^{-2}	1.7×10^{-3}
²²⁴ Ra	3.0×10^{-4}	8.8×10^{-3}	1.8×10^{-4}	4.0×10^{-4}	7.5×10^{-5}
²¹² Pb	2.6×10^{-5}	1.8×10^{-3}	2.9×10^{-5}	1.8×10^{-5}	2.6×10^{-6}
Total Chain	4.0×10^{-1}	1.2	6.0×10^{-2}	2.3×10^{-2}	1.9×10^{-3}

^aData taken from Reference 2.

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

APPENDIX 8
(cont'd.)REFERENCES

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