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Approval:	Signature on File James W. Neton, Associate Director for Science	Approval Date:	08/24/2012

☐ New ☐ Total Rewrite ☒ Revision ☐ Page Change

FOR DOCUMENTS MARKED AS A TOTAL REWRITE, REVISION, OR PAGE CHANGE, REPLACE THE PRIOR REVISION AND DISCARD / DESTROY ALL COPIES OF THE PRIOR REVISION.

PUBLICATION RECORD

EFFECTIVE DATE	REVISION NUMBER	DESCRIPTION
04/20/2004	00	New technical basis document for the Nevada Test Site–Occupational Environmental Dose. First approved issue. Initiated by Eugene M. Rollins.
12/08/2006	00 PC-1	Approved page change revision as a result of biennial review. Adds acronyms and abbreviations on pages 4 and 5. Updates required language on page 6 in the Introduction. Adds Purpose and Scope sections on page 7. Added instructions to dose reconstructors on pages 44 and 45 in Section 4.5. As a result of formal internal review, deletes Section 4.5.2 from page 45. Completes references on page 47. This revision results in no change to the assigned dose and no PER is required. Training required: As determined by the Task Manager. Initiated by Eugene M. Rollins. Approval: Signature on File 12/05/2006 Eugene M. Rollins, Document Owner Signature on File 10/24/2006 John M. Byrne, Task 3 Manager Signature on File 10/24/2006 Edward F. Maher, Task 5 Manager Signature on File 11/15/2006 Kate Kimpan, Project Director Brant A. Ulsh Signature on File for 12/08/2006 James W. Neton, Associate Director for Science
05/27/2008	01	Approved revision initiated to apply only to post 1962 employment periods and all covered employees including those identified by job classification as a drillback operator prior to 1965. Also applies to covered employees involved with any of the 10 underground tests that resulted in unexpected release of radioactive materials. Revised Table 4.2.1.2.2.-2 (new Table 4-3) to reflect 2,000 hr/yr and 2400 $m^{3/}$yr breathing rate. Deleted Tables 4.2.1.2.2-3 and 4.2.1.2.2-4. Deleted Sections 4.2.1.2.3 and 4.2.1.2.4. Added Sections 4.2.1.2.3 and 4.1.2.4 to introduce revised method for assigning environmental ambient inhalation intakes. Replaced Section 4.2.2 with ambient ingestion intake section. Revised Section 4.4 to include dose from radon and other alpha and beta emitting radionuclides and to include discussion of radon exposure in Gravel Gerties. Added Section 4.4.3. Revised Section 4.4.4, Table 4.21 to increase radon WLM for G-Tunnel prior to 1984 and to maximize radon exposures for unidentified work locations. Added discussion of radon in Gravel Gerties. Added Section 4.4.5 to provide method for assigning inhalation and ingestion intakes for underground workers. Added Section 4.5.2 to provide instructions to dose reconstructors for assigning inhalation and ingestion ambient environmental intakes for all workers at the NTS. Occupational environmental dose will increase for all claims not previously worked using ORAUT-OTIB-0018 or ORAUT-OTIB-0002. Also, cases of lung, ET1, and ET2 will need to be re-evaluated for increased radon exposures. Attachments A and B (formerly Rollins 2007, SRDB Ref ID: 37805)

EFFECTIVE DATE	REVISION NUMBER	DESCRIPTION
		were added to support the changes discussed in Section A.6.0 which included enriching the near-field environment with refractory elements. The 01-C version only added refractory elements back to the depleted refractory near-field environment given by the Hicks data (i.e., multiplied the Hicks refractory values by 2). The 01-D version enriched the near-field refractory environment by multiplying the Hicks refractory values by 4. Assumption of the enriched refractory near-field environment resulted in larger fission and activation product inhalation and ingestion correction factors (Tables 4-8 and 4-12) and increased number of affected organs and the resultant doses (Tables 4-9 and 4-14). Incorporates formal internal and NIOSH review comments. Constitutes a total rewrite of the document. Training required: As determined by the Task Manager. Initiated by Eugene M. Rollins.
04/15/2010	02	Revision initiated to incorporate definitions and directions for dose reconstruction for nonpresumptive cancers that are excluded from the 1963 through 1992 Special Exposure Cohort. Added Section 4.1.2. Added years 2002 through 2008 ambient external doses to Tables 4-15 and 4-16. Incorporates formal internal and NIOSH review comments. Updated ORAUT references. Training required: As determined by the Objective Manager. Initiated by Eugene M. Rollins.
08/24/2012	03	Revision extended measured ambient radiation doses in Tables 4-15 and 4-16 to 2010 in Section 4.3.1. Also in Section 4.3.1, clarification was added to ensure that ambient dose was not to be assigned for years when no data was available; in particular, prior to 1958. Instructions clarified that for unmonitored workers likely to be exposed prior to 1958, the 50% coworker dose in Table 6-11 of ORAUT-TKBS-0008-6 (the NTS External TBD) should be assigned. Incorporates formal internal and NIOSH review comments. Training required: As determined by the Objective Manager. Initiated by Eugene M. Rollins.

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ACRONYMS AND ABBREVIATIONS

Bq	becquerel
CFR	Code of Federal Regulations
Ci	curie
cm	centimeter
d	day
DOD	U.S. Department of Defense
DOE	U.S. Department of Energy
EEOICPA	Energy Employees Occupational Illness Compensation Program Act
EPA	U.S. Environmental Protection Agency
ET	Extra-thoracic region
ft	foot
g	gram
GSD	geometric standard deviation
GZ	Ground Zero
HPD	Health Protection Department
hr	hour
HT	airborne tritium
HTO	tritiated water vapor
ICRP	International Commission on Radiological Protection
IMBA	Integrated Modules for Bioassay Analysis
in.	inch
IREP	Interactive RadioEpidemiological Program
keV	kiloelectron-volt, 1,000 electron-volts
kg	kilogram
km	kilometer
L	liter
lb	pound
LLI	lower large intestine
LLNL	Lawrence Livermore National Laboratory
LN	lymph node
LOS	line of sight
m	meter
MeV	megaelectron-volt, 1 million electron-volts
mg	milligram
mi	mile
min	minute
mL	milliliter
mR	milliroentgen
mrem	millirem
NIOSH	National Institute for Occupational Safety and Health

NTS	Nevada Test Site
pCi	picocurie
PERM	passive environmental radon monitor
POC	probability of causation
RBM	red bone marrow
RDC	radon daughter concentration
REECo	Reynolds Electrical & Engineering Company
RPISU	radon progeny integrating sampling units
SEC	Special Exposure Cohort
SI	small intestine
SRDB Ref ID	Site Research Database Reference Identification (number)
Sv	sievert
TBD	technical basis document
TH	thoracic
TLD	thermoluminescent dosimeter
TRU	transuranic
TTR	Tonopah Test Range
ULI	upper large intestine
U.S.C.	United States Code
WL	working level
WLM	working level month
yr	year
µCi	microcurie
µm	micrometer
§	section or sections

4.1 INTRODUCTION

Technical basis documents and site profile documents are not official determinations made by the National Institute for Occupational Safety and Health (NIOSH) but are rather general working documents that provide historic background information and guidance to assist in the preparation of dose reconstructions at particular sites or categories of sites. They will be revised in the event additional relevant information is obtained about the affected site(s). These documents may be used to assist NIOSH staff in the completion of the individual work required for each dose reconstruction.

In this document the word “facility” is used as a general term for an area, building, or group of buildings that served a specific purpose at a site. It does not necessarily connote an “atomic weapons employer facility” or a “Department of Energy [DOE] facility” as defined in the Energy Employees Occupational Illness Compensation Program Act [EEOICPA; 42 U.S.C. § 7384l(5) and (12)]. EEOICPA defines a DOE facility as “any building, structure, or premise, including the grounds upon which such building, structure, or premise is located … in which operations are, or have been, conducted by, or on behalf of, the Department of Energy (except for buildings, structures, premises, grounds, or operations … pertaining to the Naval Nuclear Propulsion Program)” [42 U.S.C. § 7384l(12)]. Accordingly, except for the exclusion for the Naval Nuclear Propulsion Program noted above, any facility that performs or performed DOE operations of any nature whatsoever is a DOE facility encompassed by EEOICPA.

For employees of DOE or its contractors with cancer, the DOE facility definition only determines eligibility for a dose reconstruction, which is a prerequisite to a compensation decision (except for members of the Special Exposure Cohort). The compensation decision for cancer claimants is based on a section of the statute entitled “Exposure in the Performance of Duty.” That provision [42 U.S.C. § 7384n(b)] says that an individual with cancer “shall be determined to have sustained that cancer in the performance of duty for purposes of the compensation program if, and only if, the cancer … was at least as likely as not related to employment at the facility [where the employee worked], as determined in accordance with the POC [probability of causation[1]] guidelines established under subsection (c) …” [42 U.S.C. § 7384n(b)]. Neither the statute nor the probability of causation guidelines (nor the dose reconstruction regulation, 42 C.F.R. Pt. 82) restrict the “performance of duty” referred to in 42 U.S.C. § 7384n(b) to nuclear weapons work (NIOSH 2010a).

The statute also includes a definition of a DOE facility that excludes “buildings, structures, premises, grounds, or operations covered by Executive Order No. 12344, dated February 1, 1982 (42 U.S.C. 7158 note), pertaining to the Naval Nuclear Propulsion Program” [42 U.S.C. § 7384l(12)]. While this definition excludes Naval Nuclear Propulsion Facilities from being covered under the Act, the section of EEOICPA that deals with the compensation decision for covered employees with cancer [i.e., 42 U.S.C. § 7384n(b), entitled “Exposure in the Performance of Duty”] does not contain such an exclusion. Therefore, the statute requires NIOSH to include all occupationally-derived radiation exposures at covered facilities in its dose reconstructions for employees at DOE facilities, including radiation exposures related to the Naval Nuclear Propulsion Program. As a result, all internal and external occupational radiation exposures are considered valid for inclusion in a dose reconstruction. No efforts are made to determine the eligibility of any fraction of total measured exposure for inclusion in dose reconstruction. NIOSH, however, does not consider the following exposures to be occupationally derived (NIOSH 2010a):

- Background radiation, including radiation from naturally occurring radon present in conventional structures
- Radiation from X-rays received in the diagnosis of injuries or illnesses or for therapeutic reasons

[1] The U.S. Department of Labor (DOL) is ultimately responsible under the EEOICPA for determining the POC.

4.1.1 Purpose

This technical basis document (TBD) discusses occupational environmental dose. Occupational environmental dose is the dose individuals received at the Nevada Test Site (NTS or Test Site) while outside operational facilities but on the site during work activities. These doses can be internal or external depending on the characteristics of the individual radionuclides. While inhalation of most radionuclides would cause a dose to various organs in the body, noble gases would primarily cause only an external dose because these inert radionuclides are not readily absorbed by the body. However, inhalation of radon will result in dose to the lungs and respiratory airways due to the subsequent deposition and decay of daughter products. Occupational environmental dose would be received by workers in the aboveground and underground working environments.

With the cessation of atmospheric testing at NTS in 1962, the greatest potential for environmental intakes of radioactive material in the aboveground environment results from the inhalation of radioactive particles that were resuspended from NTS soils into the atmosphere and from ingestion of soils that were previously contaminated by atmospheric nuclear weapons tests, reactor tests, and safety tests. The potential inhalation intakes can be estimated from air sampling data in the NTS annual environmental reports (see the Environmental Reports list in the References section) coupled with extensive soil contamination data gathered between 1983 and 1991 (McArthur and Kordas1983, 1985; McArthur and Mead 1987, 1988, 1989; McArthur 1991). Because the air monitoring data were limited to gross alpha and beta measurements, tritium, and isotopes of plutonium (e.g., ^{238}Pu, ^{239}Pu, ^{240}Pu), inhalation intakes of other relatively long-lived radionuclides that have been identified in NTS soils (e.g., ^{241}Am, ^{60}Co, ^{137}Cs, ^{90}Sr, ^{152}Eu, ^{154}Eu, ^{155}Eu) are scaled to those of plutonium based on their relative abundance in NTS soils. Ingestion intakes can be estimated by assuming consumption of contaminated NTS soils. To ensure that inhalation and ingestion intakes are not underestimated, the relative abundances of the long-lived radionuclides in NTS soils determined from the 1991 soil contamination data (McArthur 1991) were decay-corrected back to 1963. In addition, to ensure that intakes and resultant doses were not underestimated, correction factors were developed to account for potential exposures to short-lived fission and activation products based on test-specific data provided by Hicks (1981a,b,c,d). In addition, a correction factor was developed for inhalation intakes that accounts for the phenomenon of early resuspension (Anspaugh et al. 2002).

4.1.2 Special Exposure Cohort Petition (SEC) Information for the Nevada Test Site

The status of Special Exposure Cohort (SEC) petitions for NTS is:

Classes Added to the SEC

- *Department of Energy (DOE) employees or DOE contractor or subcontractor employees who worked at the Nevada Test Site from January 27, 1951 through December 31, 1962 for a number of work days aggregating at least 250 work days, either solely under this employment or in combination with work days within the parameters (excluding aggregate work day requirements) established for other classes of employees included in the SEC, and who were monitored or should have been monitored* (Leavitt 2006).

- *All employees of the Department of Energy, its predecessor agencies, and its contractors and subcontractors who worked at the Nevada Test Site from January 1, 1963 through December 31, 1992 for a number of work days aggregating at least 250 work days, occurring either solely under this employment or in combination with work days within the parameters established for one or more other classes of employees included in the SEC* (Sebelius 2010).

NIOSH has determined, and the Secretary of Health and Human Services has concurred, that in the absence of bioassay results for the worker, internal doses cannot be reconstructed between 1951 and 1962 inclusive for an Energy Employee (EE). Based on the SEC petition evaluation, internal dose is not to be reconstructed for work before 1963 unless a worker has specific bioassay results that can be directly related to an event or incident. Any bioassay results in the DOE files for NTS workers before 1963 should be assumed to be valid [HASL-300 (Harley 1976) procedures were used in the early bioassay program)]; therefore, these results can be used to evaluate internal dose. Much of the internal monitoring for individuals during the SEC period was event related. However, certain job classifications required routine monitoring. These included radiation safety personnel, industrial hygienist, and security personnel.

NIOSH has determined, and the Secretary of Health and Human Services has concurred that NIOSH lacks sufficient information that would allow it to adequately estimate internal exposures during the period 1963 through 1992 (Sebelius 2010). NIOSH believes that the cessation of nuclear testing, coupled with the implementation of the 1993 NTS internal technical basis document that demonstrates NTS compliance with 10 CFR Part 835, supports NIOSH's ability to bound internal dose for the evaluated class starting in 1993.

Dose reconstruction guidance in this technical basis document (TBD) for periods before January 1, 1993 is presented to provide a technical basis for partial dose reconstructions for nonpresumptive cancers not covered in the SEC classes through December 31, 1992. Although NIOSH found that it is not possible to bound total internal dose for the proposed classes, it intends to use internal and external monitoring data that might become available for an individual claim (and that can be interpreted using its existing dose reconstruction processes or procedures). Therefore, dose reconstructions for individuals employed at NTS during the period from 1951 through December 31, 1992, but who do not qualify for inclusion in the SEC, can be performed using these data as appropriate.

4.1.3 Scope

Section 4.2 discusses internal doses to aboveground workers from onsite releases to the air and resuspension of radioactive materials in soil, as well as from ingestion of contaminated soils. Section 4.3 describes external doses to workers from ambient radiation and releases of radioactive noble gases to air. Section 4.4 discusses internal dose to underground workers from inhalation of ambient concentrations of radioactive materials in the air, ingestion of contaminated materials, and exposure to radon. Section 4.5 provides instructions to dose reconstructors for assignment of environmental intakes. Section 4.6 presents attributions and annotations, which are indicated by bracketed callouts and used to identify the source, justification, or clarification of the associated information.

4.2 INTERNAL INHALATION AND INGESTION DOSE TO ABOVEGROUND WORKERS

Section 4.2.1 discusses the internal dose for workers outside the facilities, as determined from air concentrations resulting from ground-level releases. Section 4.2.2 discusses the internal dose from resuspension of radioactive materials in the soil.

4.2.1 Onsite Releases to Air

4.2.1.1 Source Description

4.2.1.1.1 Weapons Testing

NTS has been the primary location for the testing of nuclear explosives in the continental United States since 1951. Test programs have included atmospheric testing in the 1950s and early 1960s, earth-cratering experiments, and open-air nuclear reactor and rocket engine testing. Since the mid-

1960s, testing of nuclear devices has occurred underground in drilled vertical holes or in mined tunnels. No nuclear tests have been conducted since September 1992 (Black 1995).

In all, more than 900 nuclear tests have taken place at the Test Site as part of these programs. One result of these tests is that the surface soils in many parts of NTS contain measurable amounts of several long-lived radionuclides. Almost all of the more than 100 aboveground tests contaminated the soil near ground zero. In addition, several underground tests were cratering experiments that threw radioactive rock and soil hundreds of feet, and some deeper underground tests vented radioactive material to the surface. A few safety tests, in which a nuclear device was destroyed by conventional explosives, scattered plutonium (and in some cases uranium) over the nearby ground. Further, there was fallout of radioactive debris from many tests over the northern and eastern parts of the Test Site (McArthur 1991).

Radiation levels at NTS have been monitored regularly, and safety officials have identified and fenced off areas where the soil is heavily contaminated. In many other areas, radionuclide levels are not high enough to warrant closing the area, but they are still above background.

Atmospheric weapon and safety tests from 1951 and 1963 resulted in the release of about 1.2×10^{10} Ci to the atmosphere (DOE 1996). Much of this activity was from relatively short-lived radionuclides that decayed in a matter of days or weeks. The volatile radionuclides (such as radioiodines, noble gases, and tritium) were diluted in the atmosphere and transported off site. However, much of the nonvolatile, long-lived radionuclides settled in the soils at various locations (see Table 4-10 in Section 4.2.2). These contaminated soils continue to represent a potential inhalation pathway to workers from resuspension of soils by wind and such mechanical activities as cleanup and remediation (McArthur 1991).

In 1963, nuclear weapons testing was moved underground to prevent the release of radionuclides to the atmosphere and achieve containment. Releases of radioactive material after an underground test are generally categorized with terms that describe the volume of material released and the conditions of the release (DOE 1996):

- <u>Containment Failures:</u> Containment failures are unintentional releases of radioactive matter to the atmosphere due to failure of the containment system. A prompt massive release, or one that occurs soon after a test, is a venting.

- <u>Late-Time Seeps:</u> Late-time seeps are small releases that occur days or weeks after a test when gases diffuse through pore spaces in the overlying rock and are drawn to the surface by decreases in atmospheric pressure.

- <u>Controlled Tunnel Purging:</u> A controlled tunnel purge is an intentional release to enable either recovery of experimental data and equipment or reuse of part of the tunnel system.

- <u>Operational Release:</u> Operational releases are small consequential releases that occur when core or gas samples are collected, or when a drillback hole is sealed.

At present, processing of radioactive materials at NTS includes only laboratory analyses. Handling of these materials includes only the transport and storage of nuclear explosive devices and the operation of a radioactive waste management site for low-level radioactive and mixed wastes. Monitoring and evaluation of the activities indicate that the potential sources of onsite radiation exposure are releases from the following sources (Black 1995):

- Tritiated water vapor (HTO) from drainage containment ponds for E Tunnel in Area 12
- Onsite radioanalytical laboratories

- Area 3 and 5 waste facilities
- Other diffuse sources

The following sections describe effluent sources at NTS from information in Black (1995).

4.2.1.1.2 Ground Seepage of Noble Gases

Ground seepage can be enhanced when changes in ambient pressure pump small amounts of noble gases up through the overburden and into the atmosphere from the cavity created by a nuclear test. This process, sometimes referred to as *atmospheric pumping*, creates a diffuse source of radiological effluents. These area sources are rare and therefore not routinely monitored. The phenomenon is usually restricted to tests in the Pahute Mesa region of NTS. These seepages are from nuclear tests before 1993.

4.2.1.1.3 Tunnel Operations

Nuclear tests occurred in mined tunnel complexes in the Rainier Mesa region. Because some tunnels were sealed in the mid-1990s, small amounts of contaminated water continue to drain from only one tunnel (Section 4.2.1.1.4).

4.2.1.1.4 Containment Ponds

Water contaminated with radionuclides seeps from the tunnels in Area 12 and collects in containment ponds where some evaporates and some seeps into the soil. The only radiological contaminant that produces a measurable air emission from evaporation of the water is tritium (^3H) in the form of HTO.

4.2.1.1.5 Drillbacks

After underground nuclear tests, core samples have been taken from the cavity that was formed by the detonation for analysis and diagnosis. This core sampling is accomplished by drilling into the area of interest and recovering samples using special equipment. Radioactive material can escape to the atmosphere during these operations.

4.2.1.1.6 Laboratories

Reynolds Electrical & Engineering Company (REECo) conducted radiological analyses in Building 650, and Los Alamos National Laboratory conducted similar analyses in Building 701 at Mercury, Nevada. Because these facilities have processed primarily environmental samples, very little radioactivity has passed through them. However, there was potential for radionuclides to be discharged to the atmosphere through the hood ventilation system during sample processing, particularly of spiked samples or from loss of radioactive standards that contained heavy water, radioiodines, or noble gases.

4.2.1.1.7 Radioactive Waste Management Sites

Areas 3 and 5 contain sites for the disposal of low-level radioactive waste, and Area 5 contains sites for storage of transuranic and mixed transuranic wastes, as well as the Greater Confinement Disposal Test Unit and 12 accompanying boreholes (only a few contain waste). Disposal occurs in pits and trenches; concrete pads provide temporary storage of certain wastes. Area 5 is for packaged waste disposal only. The Waste Examination Facility houses a glovebox with high-efficiency particulate air filtration that is used to examine and repack transuranic (TRU) waste drums. No contamination has been released from glovebox operations to the environment. The drums, which have been sent to NTS from Lawrence Livermore National Laboratory (LLNL) in past years, are stored in the TRU Pad

Cover Building. Repacked drums will be sent to the Waste Isolation Pilot Plant. The facility is a diffuse source of radiological effluents. The only radioactive effluent that has been detected by the various samplers around the site is HTO in atmospheric moisture. The Area 3 low-level waste site is in a location where surface soil has been contaminated by deposited plutonium, and resuspension of this soil by wind and vehicular activity has resulted in detection of above-background levels of plutonium in nearby air samples.

4.2.1.2 Atmospheric Radionuclide Concentrations

In 1964, REECo established an environmental surveillance program at NTS to measure radiological conditions throughout the site without regard to nuclear tests (Glora and Brown 1964a). That is, the collected data were not to relate to specific tests but to general conditions of radiation. The short-term objective of the program was to minimize casual personnel exposure to radiation by locating and identifying localized radiological environmental conditions by type and quantity of contamination. The long-range objective of the program was to establish baseline environmental data that could provide a reference for comparison with subsequent test activities and radiation measurements (Glora and Brown 1964a).

The initial surveillance program included, over time, 12 permanent air-sampling stations in the most populated areas at NTS. The air samplers were low-volume Filter Queen samplers with 8- by 10-in. (Gelman Type E) glass-fiber filter papers. Operating times were determined by integrated electric timers with flow rates from calibrated rotometers. Typical flow rates varied from 3 to 6 ft^3/min; samples were collected weekly (Glora and Brown 1964b).

After the first reporting period (June 1964), positive-displacement Gast pumps, which were equipped with in-line total volume gas meters, replaced the Filter Queen samplers. In May and June 1965, the 8- by 10-in. Gelman filters were replaced with a new sampler that used 4-in. diameter Whatman 41 filter paper (Lewis, Glora, and Aoki 1965). The sampling rate of these samplers was about 3 ft^3/min. Therefore, the total volume of sampled air in a 7-day period was about 1×10^3 m^3. During this period, the number of sampling stations increased to 13 and caustic scrubbers were added for the detection of radioiodines.

Early particulate samples were typically analyzed only for gross alpha and gross beta. However, if gross beta concentrations exceeded 1×10^{-5} μCi/m^3, researchers conducted an analysis for ^{227}Ac (the most hazardous beta emitter present). Because no historical evidence exists that ^{227}Ac has been detected in air or soil samples, the assumption that unidentified beta emitters were ^{227}Ac would be unreasonable and inappropriate. Therefore, for dose reconstruction, ^{90}Sr is assumed to be the unidentified beta-emitting radionuclide with the highest internal dose impact (see Section 4.2.1.2.4 for details).

Environmental samples were analyzed for gross alpha and beta radioactivity by gas proportional counting. Air samples were analyzed with a Nuclear Chicago ULTRASCALER system with a 9- by 10-in. detection chamber to accommodate the 8- by 10-in. filters. All other samples were analyzed with a Beckman WIDEBETA system equipped with an automatic sample changer. This counting equipment and protocol allowed lower detection limits of about 2.9×10^{-9} and 1.6×10^{-8} μCi/m^3 for gross alpha and gross beta, respectively (Lewis, Glora, and Aoki 1965). By 1978, the lower detection limit for gross beta counting had been reduced to 1×10^{-10} μCi/m^3 (Lewis, Glora, and Aoki 1965).

Gross gamma screening was typically performed on samples that were prepared for gross beta counting. The screening was performed using a 5- by 5-in. NaI(Tl) detector that was coupled to a single-channel analyzer. Any samples that showed elevated gamma readings were sent to a multichannel analyzer for radionuclide identification (Lewis, Glora, and Aoki 1965). In the early 1970s, gamma spectroscopy was performed using germanium-lithium [Ge(Li)] detectors and 2,000-channel analyzers with a lower detection limit of 5×10^{-9} µCi/m^3 (Lantz 1978a). After cessation of atmospheric testing at NTS in 1963, fission products were frequently measured in the atmosphere but were typically correlated with foreign atmospheric weapons testing (see Section 4.2.1.2.4).

In 1971, weekly air samples from a given station were batched on a monthly basis and subjected to radiochemical analysis for ^{239}Pu. The procedure included acid dissolution with ion exchange recovery and electroplating onto stainless-steel discs, which was followed by alpha spectroscopy using solid-state surface barrier detectors (Lantz 1978a). This analysis provided a nominal minimum detection limit of 3×10^{-7} µCi/m^3 (Lantz 1978a). Routine analysis for ^{238}Pu started in 1989 (Wruble and McDowell 1990a).

In 1977, a separate sampler was designed for the collection of airborne tritium (HT) and HTO. The sampler was portable and capable of unattended operation for as long as 2 weeks in desert areas. A small electronic pump drew air into the sampler at about 0.5 L/min, and the HTO was removed from the air stream by a silica gel drying column. The dry air then passed through a catalytic converter with platinum to generate HTO from HT. Another drying column collected the vapor, in which a small volume of distilled water served as a trap for the HTO. Appropriate aliquots of condensed moisture were obtained by heating the silica gel. Counting via liquid scintillation techniques enabled the determination of HT and HTO activities. The typical minimum detection limit for this analysis was 3×10^{-7} µCi/m^3 (Lantz 1978a).

The number of air sampling stations increased over the years to a peak of 52 in 1989 (Wruble and McDowell 1990a). This number remained fairly constant until a gradual reduction began in 1995 (Black and Townsend 1996). This reduction occurred primarily because of a gradual strategy shift from environmental monitoring to demonstration of compliance with National Emission Standards for Hazardous Air Pollutants as approved by the U.S. Environmental Protection Agency (EPA; i.e., 40 CFR Part 61) (Black and Townsend 1997). Figure 4-1 shows the locations of the 48 sampling stations in 1997 (Black and Townsend 1998).

Since the late 1980s, the environmental surveillance program has routinely monitored atmospheric concentrations of tritium, ^{238}Pu, ^{239}Pu, and ^{240}Pu. These radionuclides were considered the most important to dose for workers and members of the public (Wruble and McDowell 1990a). In addition, since the mid-1960s, measurements have been reported for gross alpha and gross beta concentrations (Glora and Brown 1964b). The following sections discuss the history of these measurements in the NTS annual environmental reports (see References section) and their importance to dose reconstruction for unmonitored employees.

4.2.1.2.1 Annual Inhalation Intakes of Tritium

Atmospheric measurements at NTS began in 1977 with samplers in Areas 5, 10, and 23 (Lantz 1978a). The sampler near Building 650 (Mercury) was the control station. During that year, the samples in Area 10 (near the Sedan Crater) demonstrated the highest HTO concentrations with a

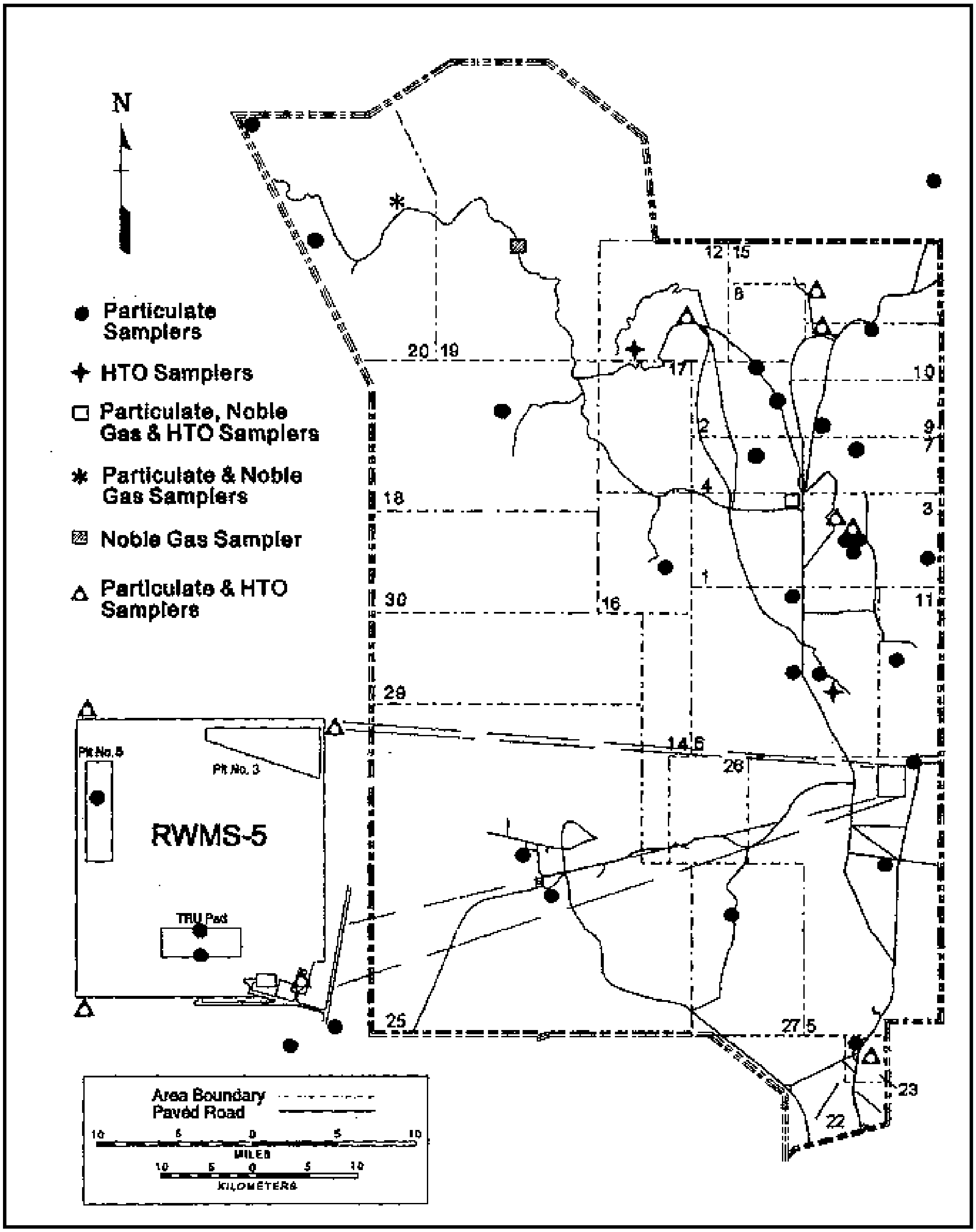

Figure 4-1. Air sampling stations in 1997.

high of 3.0×10^{-4} pCi/m^3 (Lantz 1978a). The number of sampling locations increased over the following years to include all the areas in Table 4-1. These locations provided representative samples from the most populated areas on the site (Black and Townsend 1998).

The tritium concentrations in Table 4-1 are typically the average of the maximum concentrations for a given area in a given year, unless these values were determined to be adversely affected by nearby point-source releases (e.g., Building 790 in 1983; Scoggins 1983, p. 29). If maximum values were not provided, the average concentrations were reported.

Table 4-1. HTO atmospheric concentrations by year and area with estimated maximum and average annual organ dose.[a]

	Area												Site average (pCi/mL)	Site maximum (pCi/mL)	Site maximum annual dose (mrem)	Site average annual dose (mrem)
Year	**1**	**3**	**5**	**6**	**10**	**12**	**15**	**16**	**18**	**20**	**23**	**25**				
1977					3.0E-04						1.2E-06		1.5E-04	3.0E-04	6.9E-02	3.5E-02
1978			2.9E-03		1.3E-04	1.9E-05	1.9E-05				3.0E-05		6.2E-04	2.9E-03	6.7E-01	1.4E-01
1979													6.2E-04[b]	2.9E-03[b]	6.7E-01[b]	1.4E-01[b]
1980			2.6E-04								7.1E-05		1.7E-04	2.6E-04	6.0E-02	3.8E-02
1981			3.6E-03								9.8E-04		2.3E-03	3.6E-03	8.3E-01	5.3E-01
1982	7.5E-04		2.3E-03			4.4E-03	7.6E-04				5.0E-03		2.6E-03	5.0E-03	1.2E+00	6.1E-01
1983	8.2E-05		1.1E-04			8.9E-05	3.9E-03				6.5E-04		9.6E-04	3.9E-03	8.9E-01	2.2E-01
1984	1.7E-04		5.3E-04			3.3E-05	3.5E-05				5.0E-03		1.2E-03	5.0E-03	1.2E+00	2.7E-01
1985	1.4E-04		8.6E-05			4.7E-03	3.5E-05				4.7E-04		1.1E-03	4.7E-03	1.1E+00	2.5E-01
1986	1.2E-04		5.6E-05			6.6E-05	6.0E-05				8.4E-04		2.3E-04	8.4E-04	1.9E-01	5.3E-02
1987	4.5E-05		1.7E-04			6.3E-05	4.6E-04				2.6E-04		2.0E-04	4.6E-04	1.0E-01	4.6E-02
1988	4.4E-05		3.9E-05			2.0E-05	5.2E-04				3.2E-05		1.3E-04	5.2E-04	1.2E-01	3.0E-02
1989	1.7E-04		3.1E-05		1.3E-05	1.9E-05	8.2E-05				4.6E-06		5.3E-05	1.7E-04	3.9E-02	1.2E-02
1990	7.4E-06		2.2E-05		1.3E-05	4.0E-06	2.3E-05				7.5E-05	2.1E-05	2.4E-05	7.5E-05	1.7E-02	5.5E-03
1991	9.1E-06		2.8E-05		6.3E-06	8.4E-06	1.7E-05				4.3E-06	2.1E-06	1.1E-05	2.8E-05	6.5E-03	2.5E-03
1992	1.0E-05		3.5E-05		3.3E-06	6.2E-05	6.2E-05				6.1E-06	8.8E-05	3.8E-05	8.8E-05	2.0E-02	8.8E-03
1993	8.7E-06		2.8E-05		7.9E-06	3.7E-06	2.0E-05				4.0E-06	1.9E-06	1.1E-05	2.8E-05	6.5E-03	2.4E-03
1994	4.00E-06		4.70E-05	5.10E-06	2.20E-06	2.60E-06	3.70E-05				4.10E-06	2.90E-06	1.06E-05	2.81E-05	6.74E-03	2.55E-03
1995	3.7E-06	1.4E-06	2.1E-05	9.5E-07	7.0E-06	4.3E-06	1.0E-05				2.6E-06	1.4E-06	5.8E-06	2.1E-05	4.9E-03	1.3E-03
1996	3.1E-06	4.1E-06	1.6E-05	5.6E-05	2.6E-05	2.0E-05	1.4E-05				1.1E-06	1.1E-06	1.6E-05	5.6E-05	1.3E-02	3.6E-03
1997	2.2E-06	3.8E-06	7.5E-06	5.6E-05	2.9E-05	2.3E-05	2.0E-05				1.3E-06	1.3E-06	1.6E-05	5.6E-05	1.3E-02	3.7E-03
1998	3.4E-06		1.7E-05	1.8E-04	2.9E-05	5.5E-05	1.4E-05			4.6E-04			1.1E-04	4.6E-04	1.1E-01	2.5E-02
1999	2.0E-05		1.8E-05	4.2E-06	4.1E-05	2.8E-05	2.7E-05			7.5E-04			1.3E-04	7.5E-04	1.7E-01	2.9E-02
2000	4.1E-06		1.5E-05		5.1E-05	5.4E-05	1.9E-05			9.7E-04			1.9E-04	9.7E-04	2.2E-01	4.3E-02
2001	4.2E-06		3.6E-06	6.2E-06	4.2E-05	2.5E-05	3.8E-06	2.3E-06	1.3E-06	1.1E-03	7.1E-07	6.1E-06	1.1E-04	1.1E-03	2.5E-01	2.5E-02

a. Sources: Environmental Reports listed in the References section.
b. Value not provided in annual reports; assumed same as that for 1978.

In addition to the annual area tritium concentrations, Table 4-1 lists NTS site average and maximum concentrations, which represent the arithmetic averages of the concentrations of all the areas and the maximum of all the areas, respectively. For dose reconstruction, these average and maximum site concentrations have been converted to annual organ dose using the dose conversion factor for inhalation of 1.80×10^{-11} Sv/Bq (ICRP 1998) and by assuming submersion in the respective concentrations for 2,000 hr/yr and an inhalation rate of 2,400 m^3/yr.

In addition to the inhalation dose, a factor of 1.5 was included in the organ dose calculation to account for absorption through the skin.

The uncertainty that is associated with the values in Table 4-1 was not provided in the annual environmental reports until 2001. Based on the values for that year, the standard deviation would be typically less than 10% of the measured values. Because sampling methods were changed over time and the unavailability of uncertainty estimates for previous years, the assumption that the 50th-percentile expected intakes are those in Table 4-1 and the 95th-percentile values are twice the Table 4-1 values is favorable to claimants.

For dose reconstruction, the geometric standard deviation (GSD) of the values in Table 4-2 would be estimated by using the following expression for a lognormal distribution:

$$GSD = \left(\frac{95th\ percentile}{50th\ percentile}\right)^{\left(\frac{1}{1.645}\right)} \qquad (4\text{-}1)$$

If the assumption is made that the 95th-percentile values are twice the Table 4-1 values, the GSD can be shown using Equation 4-1 to be 1.52.

If the assumption that is favorable to claimants is used to estimate the 95th-percentile dose (i.e., add 100%) from the maximum annual doses in Table 4-1, the resultant annual organ doses would be less than 2 mrem/yr.

4.2.1.2.2 Annual Intakes of Plutonium

Routine isotopic atmospheric measurements of plutonium at NTS began in 1971 with samplers in 15 locations across the site (Lantz 1978a). Six additional sampling stations were added in 1978 (Lantz 1979). Equipment at fixed locations continuously sampled the ambient air to monitor for radioactive materials. These locations were chosen to provide representative samples from populated areas on the site and to monitor resuspension of low-fired plutonium spread by safety experiments before 1960 in Areas 2, 3, 4, 7, 9, and 10. Access, worker population, geographical coverage, presence of radioactivity, and availability of electric power were considerations in the site selection for air samplers (Black and Townsend 1997).

In 1988, efforts to monitor radioactive air emissions at NTS increased as a result of the requirements of DOE Order 5400.1 (DOE 1990). Known and potential effluent sources throughout NTS were assessed for their potential to contribute to public dose (Black and Townsend 1997).

The ^{239}Pu concentrations in Table 4-2 for 1989 through 2001 represent the average of the maximum concentrations for a given area in a given year. For cases in which maximum values were not provided (i.e., 1971 through 1988), the average of the average concentrations was listed. In addition to annual area concentrations, Table 4-2 lists NTS site average and maximum concentrations, which represent the arithmetic averages of the concentration of all the areas and the maximum of the area maximum or area averages of all the areas, respectively. Potential intakes associated with these concentrations can be calculated under the assumption that an unmonitored worker was occupationally exposed for 2,000 hours per year and had a breathing rate of 2,400 m3/yr. Table 4-3 lists these calculated intakes.

Some covered employees remained on the site continuously for weeks at a time. However, because most nonworking hours were spent indoors where ambient air particulate loadings would be much less than outdoor loadings and because of the conservative assumptions that were used to estimate the values in Table 4-3, adjustment of the tabular data is not required to ensure intakes are not underestimated for these individuals. In addition, employees who lived on the site during their work week would have been housed in Area 12 or Area 23 (Mercury). For most years, the values in Table 4-2 for these locations are less than the site average values. For dose reconstruction, the inhalation intake values in Table 4-3 can be assumed to be bounding; therefore, the organ dose from these intakes should be applied with a constant distribution.

It is assumed that plutonium could be any of absorption type S, Super S, or M depending on which type delivers the maximum organ dose. Because these doses are based on air monitoring results, evaluation for type Super S (ORAUT 2010b) is required for lung and thoracic cancers if there were no bioassay results for the employee.

4.2.1.2.3 Annual Intakes of Other Radionuclides

Extensive studies were performed in the 1980s to quantify residual contamination at NTS (McArthur and Kordas 1983, 1985; McArthur and Mead 1987, 1988, 1989; McArthur 1991). Table 4-4 lists the results of these studies (McArthur 1991). Table 4-5 lists the total areal depositions based on the inventory values in Table 4-4 divided by the areal size. The results in Table 4-5 are representative of areas of NTS that contain measurable levels of contamination. These areas actually represent only about one-third of the total area within the boundaries of NTS.

In addition, because the data in Table 4-4 are representative of soil contamination in 1991, the values in Table 4-5 were decay-corrected to the beginning of 1963 – the first year after the cessation of atmospheric testing.

4.2.1.2.4 Scaling Factors for Inhalation Intakes

Because the air sampling program did not provide isotopic analyses for all the radionuclides identified in NTS soils, scaling factors were developed to estimate potential intakes for these radionuclides based on their relative abundances in comparison with ^{239}Pu when the soil contamination data (McArthur 1991) have been decay-corrected to 1963. These area-specific ratios are listed in Table 4-6.

The scaling factors in Table 4-6 are used in conjunction with the ^{239}Pu intakes in Table 4-3 to determine potential environmental intakes of all the radionuclides important to dose (in Table 4-6) that have been identified as persistent in NTS soils. For dose reconstruction, maximum intakes of these radionuclides are calculated by selecting the maximum annual intake of plutonium (e.g., 0.381 Bq/yr derived for Area 9 in 1972) and multiplying this value by the maximum scaling factor for each of the radionuclides in Table 4-6.

4.2.1.2.5 Correction for Resuspension for Early Times After Atmospheric Tests

Anspaugh et al. (2002) stated that, based on empirical observations, concentrations of resuspended radionuclides in air have been noted to display a strong time dependence early after deposition and that this pathway could be important for reoccupation of contaminated property. Anspaugh et al. also stated that there has not been universal agreement that resuspension is an important pathway but that it is now generally accepted that there are a few instances in which the pathway could be the

Table 4-2. Atmospheric concentrations of ^{239}Pu for sampled areas (pCi/m^3).[a]

	Area										
Year	**1**	**2**	**3**	**4**	**5**	**6**	**7**	**9**	**10**	**11**	**12**
1971	3.7E-04	6.6E-04	1.7E-04		1.2E-04	1.7E-04		7.2E-04	2.1E-04	9.1E-05	1.6E-04
1972	1.5E-04	2.2E-04	3.7E-04		1.4E-04	1.3E-03		4.3E-03	2.9E-04	2.4E-04	7.9E-04
1973	8.7E-05	2.2E-04	2.1E-04		8.7E-05	1.2E-04		8.6E-04	4.4E-05	2.4E-04	4.3E-05
1974	8.0E-05	6.8E-05	1.3E-04		5.0E-05	7.9E-05		2.1E-04	5.7E-05	8.0E-05	5.8E-05
1975	2.9E-05	3.8E-05	1.5E-04		4.3E-05	6.3E-05		1.7E-04	5.7E-05	4.9E-05	3.3E-05
1976	2.9E-04	1.7E-04	1.0E-03		1.5E-04	4.4E-04		3.2E-03	2.1E-04	2.4E-04	2.8E-04
1977	2.7E-05	6.5E-05	2.6E-04		3.4E-05	5.5E-05		2.5E-04	4.6E-05	1.2E-04	2.1E-05
1978	4.7E-05	1.3E-04	2.8E-04		6.1E-05	9.9E-05		5.4E-04	2.9E-04	1.1E-04	6.2E-05
1979	7.4E-05	4.7E-04	1.7E-06		2.1E-05	4.0E-05	4.9E-05	5.2E-04	2.9E-04	3.7E-05	2.4E-05
1980	2.3E-05	2.6E-04	1.1E-06		2.9E-05	4.8E-05	4.5E-05	4.5E-04		4.4E-05	1.9E-05
1981	2.6E-05	6.4E-05	1.4E-04		2.5E-05	2.9E-05	3.3E-05	3.2E-04		4.6E-05	2.0E-05
1982	6.1E-05	4.0E-05	6.2E-05		2.0E-05	3.5E-05	7.4E-05	2.2E-04		5.3E-05	2.7E-05
1983	3.1E-05	6.5E-05	9.2E-05		2.3E-05	3.2E-05	3.3E-05	2.1E-04		1.8E-04	2.3E-05
1984	1.9E-05	1.0E-04	2.3E-04		3.8E-05	2.0E-05	3.8E-05	1.0E-03		5.6E-05	1.7E-05
1985	1.5E-05	5.8E-05	2.1E-04		3.1E-05	2.7E-05	2.7E-05	1.3E-03		5.2E-05	2.3E-05
1986	1.5E-04	6.2E-05	5.5E-04		2.8E-05	1.9E-04	4.8E-05	2.8E-04		2.5E-05	3.2E-05
1987	5.5E-05	2.3E-05	2.8E-03		1.7E-05	2.5E-05	1.6E-05	1.1E-04		3.2E-05	1.6E-05
1988	9.2E-05	2.8E-05	2.1E-04		2.8E-05	2.2E-05	1.5E-05	5.1E-05		1.2E-05	1.6E-05
1989	8.6E-04	5.8E-05	5.4E-04		5.0E-05	8.0E-05	2.7E-04	3.5E-04		5.4E-04	6.8E-04
1990	9.9E-05	3.2E-05	3.5E-04		3.1E-05	3.7E-05	4.1E-05	5.0E-04	4.3E-05	1.1E-04	1.0E-05
1991	3.1E-03	2.1E-05	3.8E-04		3.5E-05	7.2E-05	3.2E-05	3.5E-04	5.4E-07	2.2E-04	1.4E-05
1992	1.5E-04	1.1E-04	2.3E-03		8.0E-05	7.3E-05	3.9E-04	8.8E-04	2.4E-04	8.8E-05	3.8E-05
1993	2.0E-04	4.9E-05	4.0E-04		8.0E-05	9.0E-05	7.8E-05	2.7E-03	1.7E-04	4.2E-04	8.4E-04
1994											
1995	4.1E-05	1.1E-05	2.4E-04		1.5E-05	5.0E-05	1.2E-05	3.9E-04	3.8E-05	4.0E-05	1.1E-05
1996	7.5E-04		3.0E-04		1.6E-05	4.6E-05	4.5E-04	6.1E-04	5.8E-05	1.9E-05	3.7E-06
1997		1.1E-05							2.6E-05	1.1E-05	1.5E-06
1998	4.7E-04		6.7E-05		9.5E-05	1.1E-05	2.7E-05	7.4E-04		9.5E-06	
1999				5.9E-05							
2000	8.5E-05		2.7E-04	5.9E-05	2.1E-05	4.1E-04	3.4E-05	2.8E-03			
2001	4.5E-04	1.2E-04	1.2E-04	1.9E-04	6.8E-05	9.6E-05	1.7E-05	5.0E-04	8.5E-06		

Year	Area									Site average	Site maximum
	15	16	18	19	20	23	25	27	28		
1971		1.9E-04	8.0E-05	2.3E-04		9.4E-05		8.4E-05	3.9E-05	2.26E-04	7.21E-04
1972		1.7E-04	1.5E-04	2.7E-03		3.5E-04		1.7E-04	5.1E-05	7.63E-04	4.29E-03
1973		4.4E-05	3.8E-05	7.2E-05		5.6E-05		3.7E-05	3.3E-05	1.46E-04	8.59E-04
1974		6.0E-05		6.6E-05		8.5E-05	8.6E-05	7.2E-05	1.2E-04	8.64E-05	2.11E-04
1975		4.3E-05		3.4E-05		4.4E-05	5.7E-05	6.8E-05	7.0E-05	6.27E-05	1.68E-04
1976		1.6E-04		9.9E-05		5.7E-04	2.0E-04	1.4E-04	6.3E-05	4.82E-04	3.18E-03
1977		2.6E-05		2.3E-05		2.7E-05	3.3E-05	3.1E-05	2.3E-05	6.93E-05	2.55E-04
1978		5.1E-05		6.2E-05		5.8E-05	7.0E-05	9.4E-05	3.9E-05	1.33E-04	5.42E-04
1979	5.2E-05	2.1E-05		2.0E-05		2.2E-05	1.8E-05	1.6E-05	2.3E-05	9.98E-05	5.24E-04
1980	6.4E-05			9.5E-05		3.5E-05	3.1E-05	2.1E-05	1.6E-05	7.88E-05	4.52E-04
1981	9.3E-05	2.6E-05		2.3E-05		2.9E-05	2.4E-05	2.5E-05	1.6E-05	5.88E-05	3.22E-04
1982	4.2E-05	1.7E-05		2.2E-05		2.1E-05	1.7E-05			5.04E-05	2.15E-04
1983	2.2E-05	1.5E-05		3.0E-05		2.8E-05	3.0E-05			5.85E-05	2.14E-04
1984	3.2E-05	2.8E-05		2.5E-05		2.9E-05	3.2E-05	4.8E-05		1.15E-04	1.02E-03
1985	4.0E-05	2.1E-05		3.5E-05		2.2E-05	2.6E-05	3.1E-05		1.30E-04	1.33E-03
1986	3.9E-05	2.8E-05		2.2E-05		6.6E-05	2.1E-05	2.3E-05		1.03E-04	5.45E-04
1987	1.8E-05	1.4E-05		1.7E-05		1.6E-05	2.3E-05			2.13E-04	2.81E-03
1988	2.4E-05			1.4E-05		3.5E-05	1.5E-05	1.5E-05		4.15E-05	2.14E-04
1989	2.3E-05	1.1E-05		9.0E-06		1.2E-04	9.5E-06	4.8E-06		2.40E-04	8.55E-04
1990	8.2E-05	5.8E-06		3.9E-06		8.5E-05	1.1E-05	3.5E-05		9.17E-05	4.98E-04
1991	1.2E-04	2.9E-05		8.8E-06		1.3E-05	2.6E-05	7.7E-06		2.77E-04	3.10E-03
1992	1.1E-03	1.9E-05		8.9E-06		1.3E-05	1.8E-05	1.6E-04		3.52E-04	2.26E-03
1993	3.3E-04	1.3E-05		7.7E-06		6.3E-06	3.4E-06	1.1E-05		3.37E-04	2.70E-03
1994											
1995	1.6E-04	1.4E-05		1.0E-06		7.4E-06	4.5E-06			6.92E-05	3.90E-04
1996	2.4E-04	1.8E-06	3.5E-06		4.5E-06	6.3E-06	4.2E-06	2.6E-06		1.57E-04	7.50E-04
1997	2.9E-05	1.9E-07	2.0E-05		2.0E-05	6.0E-07	1.0E-06	1.6E-06		1.11E-05	2.90E-05
1998	6.3E-05		3.1E-05		3.3E-05	1.2E-05	6.3E-06			1.30E-04	7.35E-04
1999	1.1E-05		1.1E-05		2.0E-05		6.2E-06			2.14E-05	5.90E-05
2000	2.6E-04		1.0E-05		6.6E-06		1.1E-05			3.63E-04	2.83E-03
2001	1.4E-05	9.6E-06	1.7E-05		7.6E-06	2.0E-06	4.4E-06			1.02E-04	5.04E-04

a. Sources: Environmental Reports listed in the References section.

Table 4-3. Annual inhalation intakes from ^{239}Pu for sampled areas (Bq) (from Attachment A).

Year	Area										
	1	2	3	4	5	6	7	9	10	11	12
1971	0.0329	0.0587	0.0147		0.0105	0.0153		0.0640	0.0185	0.0081	0.0145
1972	0.0131	0.0199	0.0327		0.0124	0.1190		0.3810	0.0258	0.0216	0.0698
1973	0.0077	0.0198	0.0183		0.0078	0.0107		0.0763	0.0039	0.0216	0.0038

Year	Area										
	1	2	3	4	5	6	7	9	10	11	12
1974	0.0071	0.0060	0.0112		0.0044	0.0070		0.0187	0.0051	0.0071	0.0052
1975	0.0026	0.0034	0.0130		0.0038	0.0056		0.0149	0.0051	0.0044	0.0029
1976	0.0255	0.0153	0.0924		0.0133	0.0392		0.2824	0.0189	0.0210	0.0248
1977	0.0024	0.0058	0.0226		0.0030	0.0049		0.0225	0.0041	0.0107	0.0019
1978	0.0042	0.0118	0.0250		0.0054	0.0088		0.0481	0.0258	0.0095	0.0055
1979	0.0066	0.0413	0.0001		0.0018	0.0035	0.0044	0.0465	0.0258	0.0033	0.0021
1980	0.0020	0.0230	0.0001		0.0026	0.0043	0.0040	0.0401		0.0039	0.0017
1981	0.0023	0.0056	0.0124		0.0022	0.0026	0.0029	0.0286		0.0041	0.0018
1982	0.0054	0.0036	0.0055		0.0018	0.0031	0.0066	0.0191		0.0047	0.0024
1983	0.0028	0.0057	0.0082		0.0020	0.0029	0.0029	0.0190		0.0162	0.0020
1984	0.0017	0.0088	0.0202		0.0034	0.0018	0.0034	0.0906		0.0050	0.0015
1985	0.0013	0.0051	0.0189		0.0027	0.0024	0.0024	0.1181		0.0046	0.0020
1986	0.0131	0.0055	0.0484		0.0025	0.0165	0.0043	0.0249		0.0022	0.0028
1987	0.0048	0.0020	0.2495		0.0015	0.0022	0.0014	0.0098		0.0028	0.0014
1988	0.0082	0.0025	0.0190		0.0025	0.0020	0.0013	0.0045		0.0011	0.0014
1989	0.0759	0.0052	0.0478		0.0045	0.0071	0.0240	0.0311		0.0480	0.0604
1990	0.0088	0.0029	0.0308		0.0027	0.0033	0.0036	0.0442	0.0038	0.0095	0.0009
1991	0.2753	0.0018	0.0340		0.0031	0.0064	0.0029	0.0311	0.0000	0.0199	0.0013
1992	0.0133	0.0098	0.2007		0.0071	0.0065	0.0346	0.0781	0.0213	0.0078	0.0034
1993	0.0180	0.0044	0.0352		0.0071	0.0079	0.0069	0.2398	0.0151	0.0373	0.0746
1994											
1995	0.0036	0.0010	0.0216		0.0014	0.0045	0.0011	0.0346	0.0033	0.0036	0.0010
1996	0.0666	0.0000	0.0268		0.0015	0.0041	0.0400	0.0542	0.0051	0.0017	0.0003
1997		0.0010							0.0023	0.0010	0.0001
1998	0.0416		0.0060		0.0084	0.0010	0.0024	0.0653		0.0008	
1999											
2000	0.0075		0.0239	0.0052	0.0018	0.0366	0.0030	0.2509			
2001	0.0404	0.0011	0.0103	0.0166	0.0061	0.0086	0.0015	0.0447	0.0008		

Year	Area									Site average	Site maximum
	15	16	18	19	20	23	25	27	28		
1971		0.0171	0.0071	0.0201		0.0083		0.0075	0.0035	0.0200	0.0640
1972		0.0150	0.0129	0.2424		0.0309		0.0147	0.0045	0.0677	0.3810
1973		0.0039	0.0034	0.0063		0.0050		0.0033	0.0029	0.0130	0.0763
1974		0.0053		0.0059		0.0075	0.0076	0.0064	0.0104	0.0077	0.0187
1975		0.0038		0.0030		0.0039	0.0051	0.0060	0.0062	0.0056	0.0149
1976		0.0139		0.0088		0.0504	0.0178	0.0122	0.0056	0.0428	0.2824
1977		0.0023		0.0020		0.0024	0.0029	0.0028	0.0020	0.0062	0.0226
1978		0.0045		0.0055		0.0052	0.0062	0.0083	0.0035	0.0118	0.0481
1979	0.0046	0.0019		0.0017		0.0020	0.0016	0.0014	0.0020	0.0089	0.0465
1980	0.0057			0.0085		0.0031	0.0027	0.0019	0.0014	0.0070	0.0401
1981	0.0083	0.0023		0.0020		0.0025	0.0022	0.0022	0.0014	0.0052	0.0286
1982	0.0037	0.0015		0.0019		0.0019	0.0015			0.0045	0.0191
1983	0.0020	0.0013		0.0026		0.0025	0.0027			0.0052	0.0190
1984	0.0028	0.0025		0.0022		0.0025	0.0028	0.0043		0.0102	0.0906
1985	0.0036	0.0019		0.0031		0.0020	0.0023	0.0028		0.0115	0.1181
1986	0.0035	0.0025		0.0020		0.0058	0.0018	0.0020		0.0092	0.0484
1987	0.0016	0.0012		0.0015		0.0014	0.0020	0.0000		0.0189	0.2495
1988	0.0022			0.0012		0.0031	0.0013	0.0013		0.0037	0.0190
1989	0.0021	0.0010		0.0008		0.0104	0.0008	0.0004		0.0213	0.0759
1990	0.0073	0.0005		0.0003		0.0075	0.0010	0.0031		0.0081	0.0442
1991	0.0109	0.0026		0.0008		0.0012	0.0023	0.0007		0.0246	0.2753
1992	0.0977	0.0017		0.0008		0.0011	0.0016	0.0142		0.0312	0.2007
1993	0.0293	0.0012		0.0007		0.0006	0.0003	0.0010		0.0300	0.2398
1994										0.0000	0.0000
1995	0.0142	0.0012		0.0001		0.0007	0.0004			0.0061	0.0346
1996	0.0213	0.0002	0.0003	0.0000	0.0004	0.0006	0.0004	0.0002		0.0124	0.0666
1997	0.0026	0.0000	0.0018		0.0018	0.0001	0.0001	0.0001		0.0010	0.0026
1998	0.0056		0.0027		0.0029	0.0010	0.0006			0.0115	0.0653
1999	0.0010		0.0010		0.0018		0.0006			0.0011	0.0018
2000	0.0229		0.0009		0.0006		0.0010			0.0322	0.2509
2001	0.0012	0.0008	0.0015		0.0007	0.0002	0.0004			0.0090	0.0447

Table 4-4. Inventory of contaminated soil (Ci).

Area	Area (mi^2)	Am-241	Pu-238	Pu-239,240	Co-60	Cs 137	Sr-90	Eu-152	Eu-154	Eu-155
1	26.5	4.2	6.5	24	1.1	8.8	15	15	0.1	0.5
2	19.7	2.9	8.6	22	1.2	24	46	14		0.4
3	32.3	4.6	3.1	37	1	12	33	18	0.1	0.5
4	16	6.6	13	40	1.6	12	13	9.1		0.2
5	2.9	0.6	0.1	4.8	0.6	0.4	0.9	10	0.2	0
6	32.3	1.7	3.3	8.4	0.2	2.8	3.5			0
7	19.3	2.2	0.6	16	1	5.2	9.2	22	0.2	0.3
8	13.9	17	8	110	5.7	42	25	4.4		0.6
9	20	4.2	2.2	89	0.7	8.7	13	23	0.2	0.3
10	20	19	19	110	9.7	84	55	2.2	0.3	5
11	4	3.3	0.5	29	0	0.5	0.3			
12	39.6	5.7	8.5	39	1.2	20	17			
15	35.3	8	7.8	63	0.3	19	22			
16	14.3	0.7	1.5	3.7	0.1	2.9	3.7			
17	31.4	2.8	4.5	18	1.0	15	19			
18	27.3	19	5.6	100	0.7	10	17	1.1	0.1	0.8
19	148.3	21	32	140	1.1	36	31			
20	6.2	23	30	41	7.9	5.5	4.3	13	1.6	4.8
25	0.9					0.2	0.1	0.4		
26	0.2									
30	0.03	3.2	4.5	14	0.8	1.5	1.3	0.7	0.1	0.2

Source: McArthur (1991).

Table 4-5. Radionuclide areal soil deposition decay, corrected to 1963 (Bq/m^2)

Area	Am-241	Pu-238	Pu-239,240	Co-60	Cs 137	Sr-90	Eu-152	Eu-154	Eu-155
1	2.37E+03	4.37E+03	1.29E+04	2.35E+04	9.06E+03	1.57E+04	3.47E+04	4.89E+02	1.35E+04
2	2.20E+03	7.78E+03	1.60E+04	3.45E+04	3.32E+04	6.49E+04	4.35E+04	0.00E+00	1.45E+04
3	2.13E+03	1.71E+03	1.64E+04	1.76E+04	1.01E+04	2.84E+04	3.41E+04	4.01E+02	1.11E+04
4	6.16E+03	1.45E+04	3.57E+04	5.67E+04	2.05E+04	2.26E+04	3.48E+04		8.93E+03
5	3.09E+03	6.15E+02	2.37E+04	1.17E+05	3.76E+03	8.63E+03	2.11E+05	8.94E+03	0.00E+00
6	7.86E+02	1.82E+03	3.72E+03	3.51E+03	2.36E+03	3.01E+03			
7	1.70E+03	5.54E+02	1.19E+04	2.94E+04	7.35E+03	1.33E+04	6.98E+04	1.34E+03	1.11E+04
8	1.83E+04	1.03E+04	1.13E+05	2.33E+05	8.24E+04	5.00E+04	1.94E+04		3.08E+04
9	3.14E+03	1.96E+03	6.36E+04	1.98E+04	1.19E+04	1.81E+04	7.04E+04	1.30E+03	1.07E+04
10	1.42E+04	1.69E+04	7.86E+04	2.75E+05	1.15E+05	7.65E+04	6.74E+03	1.94E+03	1.79E+05
11	1.23E+04	2.23E+03	1.04E+05	0.00E+00	3.41E+03	2.09E+03			
12	2.15E+03	3.83E+03	1.41E+04	1.72E+04	1.38E+04	1.19E+04			
15	3.39E+03	3.94E+03	2.55E+04	4.82E+03	1.47E+04	1.73E+04			
16	7.31E+02	1.87E+03	3.70E+03	3.97E+03	5.53E+03	7.20E+03			
17	1.33E+03	2.55E+03	8.20E+03	1.81E+04	1.30E+04	1.68E+04			
18	1.04E+04	3.66E+03	5.24E+04	1.45E+04	9.99E+03	1.73E+04	2.47E+03	4.75E+02	2.09E+04
19	2.12E+03	3.85E+03	1.35E+04	4.21E+03	6.62E+03	5.81E+03			
20	5.54E+04	8.62E+04	9.45E+04	7.23E+05	2.42E+04	1.93E+04	1.28E+05	3.34E+04	5.53E+05
25					6.06E+03	3.09E+03	2.72E+04		
26									
30	1.59E+05	2.67E+05	6.67E+05	1.51E+06	1.36E+05	1.21E+05	1.43E+05	4.32E+04	4.76E+05

dominant one. Many observations have shown that the rate of resuspension decreases rapidly with time and, for accident situations, resuspension is only of importance (in comparison with the inhalation exposure from the initial cloud passage) over short periods. For this reason, Anspaugh et al. stated that the resuspension factor model has been widely used to predict the concentration of resuspended radionuclides early after initial deposition while the mass loading model (which uses measurements of dust loading in air and soil contamination data to predict air concentration of radionuclides) has generally been preferred for times long after deposition. However, Anspaugh et al. also stated that it is always preferable to rely on actual measurements that are performed over long periods (such as those in Section 4.2.1.2.2).

Table 4-6. Abundance of radionuclides in NTS soils in relation to ^{239}Pu decay, corrected to 1963.

Area	Am-241	Pu-238	Pu-239,240	Co-60	Cs 137	Sr-90	Eu-152	Eu-154	Eu-155
1	0.183	0.338	1.000	1.818	0.700	1.216	2.677	0.038	1.041
2	0.138	0.487	1.000	2.164	2.081	4.067	2.726		0.908
3	0.130	0.104	1.000	1.072	0.619	1.735	2.084	0.024	0.675
4	0.172	0.405	1.000	1.587	0.572	0.632	0.975		0.250
5	0.131	0.026	1.000	4.958	0.159	0.365	8.925	0.378	
6	0.212	0.490	1.000	0.944	0.636	0.810			
7	0.144	0.047	1.000	2.479	0.620	1.118	5.890	0.113	0.937
8	0.162	0.091	1.000	2.055	0.728	0.442	0.171		0.273
9	0.049	0.031	1.000	0.312	0.187	0.284	1.107	0.020	0.168
10	0.181	0.215	1.000	3.498	1.457	0.973	0.086	0.025	2.271
11	0.119	0.021	1.000		0.033	0.020			
12	0.153	0.272	1.000	1.220	0.978	0.848			
15	0.133	0.154	1.000	0.189	0.575	0.679			
16	0.198	0.505	1.000	1.072	1.495	1.945			
17	0.163	0.312	1.000	2.204	1.590	2.053			
18	0.199	0.070	1.000	0.278	0.191	0.331	0.047	0.009	0.400
19	0.157	0.285	1.000	0.312	0.491	0.431			
20	0.586	0.912	1.000	7.643	0.256	0.204	1.358	0.354	5.849
25									
26									
30	0.229	0.321	1.000	0.057	0.107	0.093	0.050	0.007	0.014
Maximum scaling factor	**0.586**	**0.912**	**1.000**	**7.64**	**2.08**	**4.07**	**8.93**	**0.378**	**5.85**
Scaled maximum intake (Bq/yr)	**0.223**	**0.347**	**0.381**	**2.91**	**0.792**	**1.55**	**3.40**	**0.144**	**2.23**

Anspaugh et al. (2002) presented several resuspension models that have been proposed but concluded that they can be over- or under-predictive at various times after deposition in comparison with empirical observations. However, with expanded data sets from the work of Hicks (1981e) and others in the 1980s, Anspaugh et al. proposed a resuspension model that more accurately describes the observed results over the entire timespan of the expanded data set for NTS:

$$S_f = [10^{-5}e^{-0.07t} + 6\times10^{-9}e^{-0.003t} + 10^{-9}]\times10^{\pm1}\ \text{m}^{-1} \tag{4-2}$$

A graphical depiction of Equation 4-2 is in Figure 4-2 from time *t* equal zero to 1,000 days after detonation. As shown in Figure 4-2, the resuspension factor S_f ranges from about 10^{-5} at times early after deposition, falls rapidly during the first 100 or so days to a value of about 10^{-8}, and then approaches a value of 10^{-9} after a few years. The factor of 10 at the end of the equation is a statement of uncertainty in the model.

If it is true that the mass loading approach is more predictive at time long after initial deposition and that the resuspension proposed by Anspaugh et al. is predictive of the observed results over the expanded dataset (including those developed in the 1980s), the factor of 10^{-9} could be taken to be the resuspension factor that would be predictive of the mass loading process that is thought to be more important during the times when air monitoring data are available (i.e., 1971 through 2001, see Section 4.2.1.2.2).

When Anspaugh et al.'s proposed resuspension model (Equation 4-2) is integrated over 180 to 545 days and compared with the result of the integral of the constant 10^{-9} over the same period, a factor is developed that can be used to correct the intakes derived from air sampling data (i.e., times long after initial deposition) for the early resuspension phenomenon that has been observed at NTS.

The period of integration was selected to begin at day 180 because the last atmospheric tests at NTS were in July 1962 and, therefore, about 180 days had passed before the beginning of 1963.

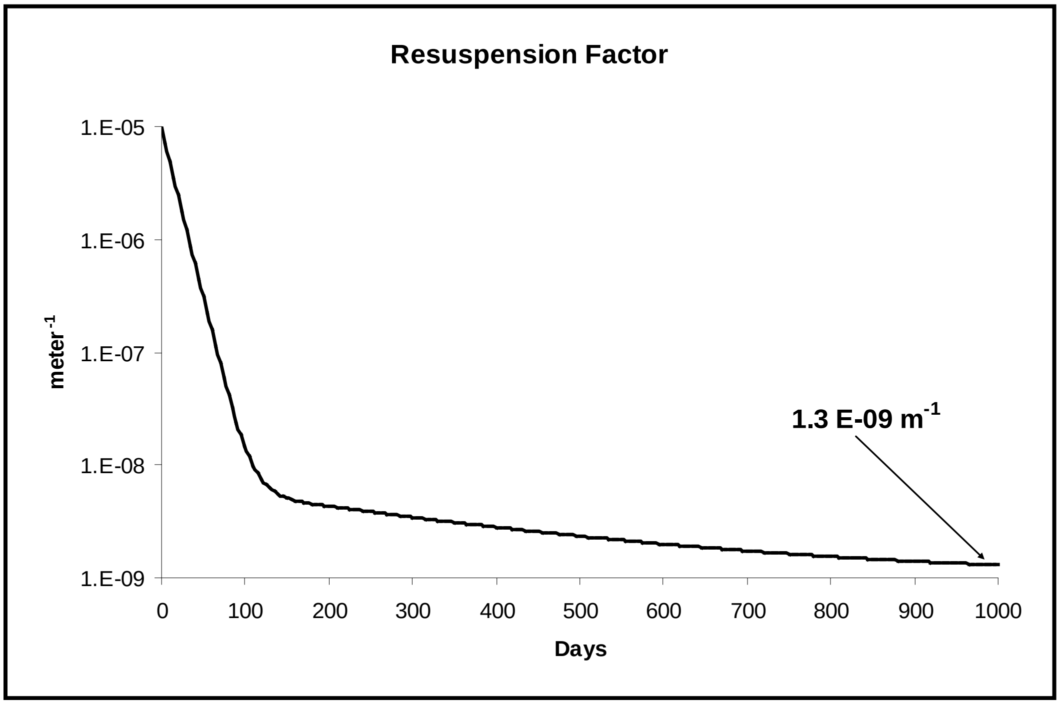

Figure 4-2. Resuspension factor as a function of time after initial deposition (from Attachment A).

For the Anspaugh et al. (2002) model, the early resuspension correction factor has been determined to be 3.12. In a similar manner, correction factors for 1964 and 1965 were determined to be 1.72 and 1.24, respectively. Thus, for dose reconstruction, the scaled maximum inhalation intakes in Table 4-6 should be multiplied by these factors to account for early resuspension from 1963 through 1965. If necessary, these intakes can be prorated for time less than a year for a best estimate if the worker was on the site for only a fraction of the year.

4.2.1.2.6 Correction for Inhalation Dose from Short-Lived Fission and Activation Products

Correction factors were developed to account for inhalation intakes of short-lived fission and activation products based on organ-specific dose from the ^{90}Sr intakes in Table 4-7 (from Attachment A). These corrections are listed in Table 4-8.

Table 4-7. Scaled inhalation intakes corrected for early resuspension (Bq/yr).

Year of intake	Am-241	Pu-238	Pu-239, 240	Co-60	Cs 137	Sr-90	Eu-152	Eu-154	Eu-155
1963	0.70	1.08	1.19	9.08	2.47	4.84	10.61	0.45	6.96
1964	0.38	0.59	0.65	4.98	1.35	2.65	5.81	0.25	3.81
1965	0.28	0.43	0.47	3.61	0.98	1.92	4.22	0.18	2.77
All subsequent years	**0.223**	**0.347**	**0.381**	**2.91**	**0.792**	**1.55**	**3.40**	**0.144**	**2.23**

The Integrated Modules for Bioassay Analysis (IMBA) computer program was used to determine the organ doses resulting from the scaled intakes of ^{90}Sr in Table 4-7 for a period of 10 years (i.e., 4.84 Bq for 1963, 2.65 Bq for 1964, 1.92 for 1965, and 1.55 for all subsequent years through 1972). These doses were then multiplied by the correction factors in Table 4-8 to determine the additional dose that should be added to account for potential dose from inhalation of short-lived fission and activation products. The organ-specific fission and activation doses greater than 0.001 rem are listed in Table 4-9. Other organs are not listed in Table 4-9 because all were less than 0.001 rem. These doses should be added to the input for the Interactive RadioEpidemiological Program (IREP) as a constant as 100% 30-to-250-keV photons. These doses are input as constants because they are based on bounding intakes (i.e., maximum) and they are entered as 30-to-250-keV photons as an assumption favorable to claimants.

Table 4-8. Organ-specific inhalation dose fission and activation product correction factors.

Organ	Fission and activation product correction factor by year									
	1963	1964	1965	1966	1967	1968	1969	1970	1971	1972
Skin, Thymus, Spleen, Muscle, Pancreas, Breast, Esophagus, Brain, Thyroid, Adrenals, SI, Skin, Uterus, Kidneys, Testes, Ovaries, Stomach, Gall bladder	730	364	242	182	145	121	104	90.7	80.6	72.5
ULI	458	179	99.2	64.0	45.0	33.4	25.9	20.6	16.8	14.0
Urinary bladder	335	149	91.3	63.6	47.5	37.2	30.1	24.9	21.0	18.0
Lungs	34,900	14,200	7,960	5,150	3,630	2,700	2,100	1,660	1,360	1,130
ET ET1 ET2 LN(TH) LN(ET)	1,570	827	598	492	438	412	412	412	412	412
LLI	420	142	70.8	42.4	28.2	20.1	15.1	11.7	9.4	7.6
Colon	390	148	79.4	50.0	34.5	25.2	19.3	15.2	12.3	10.2
Liver	9,260	4,620	1,540	1,190	988	858	769	706	661	629
Red bone marrow	37.9	18.2	12.8	10.4	10.4	10.4	10.4	10.4	10.4	10.4
Bone surfaces	78.5	40.1	28.1	22.4	22.4	22.4	22.4	22.4	22.4	22.4

Table 4-9. Inhalation dose from short-lived fission and activation products (rem).

Organ	Year									
	1963	1964	1965	1966	1967	1968	1969	1970	1971	1972
Liver	0.002									
Bone surface	0.001	0.002	0.002	0.002	0.002	0.002	0.002	0.002	0.002	0.002
LLI	0.001									
ET	0.002									
Lung	0.008	0.003	0.002	0.001						
ET1	1.635	0.474	0.248	0.164	0.146	0.138	0.138	0.138	0.138	0.138

4.2.2 Annual Intakes from Ingestion

To account for potential intakes from inadvertent ingestion of contaminated soil, the area-specific radionuclide soil deposition data in Table 4-5 were converted to volumetric data (i.e., Bq/g) by assuming a radionuclide relaxation depth of 2.3 cm and a soil density of 1.5 g/cm^3 (DOE 2003). The area-specific radionuclide soil concentrations are listed in Table 4-10.

If the assumption is made that the workers ingested 100 mg of soil each day [EPA (1989) recommends a value of 50 mg/d] and that full-time employment was 250 d/yr, annual ingestion can be calculated in a manner favorable to claimants. The area-specific annual ingestion rates are in Table 4-11.

For most radionuclides, Area 30 provided the highest areal deposition and resultant intakes. Area 30 is relatively small (150 km^2) and inaccessible, and is on the Western edge of NTS. It has rugged terrain and includes the northern reaches of Fortymile Canyon. In 1968, it was the site of Project BUGGY, the first nuclear row-charge experiment in the PLOWSHARE Program. As a result of the test, a trench 255 m long, 77 m wide, and 206 m deep was created. The test resulted in large quantities of vitrified glass. Because of the bias that is introduced when Area 30 is included, the maximum annual intakes in Table 4-11 have been provided without the Area 30 areal concentrations.

As with inhalation intakes, a method was developed to adjust ingestion doses for potential dose from short-lived fission and activation products that are no longer persistent in NTS soils in measurable amounts. The organ-specific fission and activation product correction factors were developed based on the relative contribution of 90Sr to the total ingestion dose. These correction factors are listed in Table 4-12.

Table 4-10. Radionuclide soil concentration by area (Bq/g).

Area	Am-241	Pu-238	Pu-239, 240	Co-60	Cs-137	Sr-90	Eu-152	Eu-154	Eu-155
1	0.069	0.127	0.375	0.682	0.263	0.456	1.005	0.014	0.391
2	0.064	0.226	0.463	1.001	0.963	1.882	1.262		0.420
3	0.062	0.050	0.475	0.509	0.294	0.824	0.989	0.012	0.321
4	0.179	0.420	1.036	1.644	0.593	0.655	1.010		0.259
5	0.090	0.018	0.686	3.401	0.109	0.250	6.122	0.259	
6	0.023	0.053	0.108	0.102	0.069	0.087	0.000	0.00	
7	0.049	0.016	0.344	0.852	0.213	0.384	2.024	0.039	0.322
8	0.530	0.297	3.280	6.741	2.389	1.450	0.562		0.894
9	0.091	0.057	1.844	0.575	0.344	0.524	2.042	0.038	0.311
10	0.411	0.491	2.279	7.973	3.321	2.217	0.195	0.056	5.177
11	0.357	0.065	3.005		0.099	0.060			
12	0.062	0.111	0.408	0.498	0.399	0.346			
15	0.098	0.114	0.740	0.140	0.426	0.502			
16	0.021	0.054	0.107	0.115	0.160	0.209			
17	0.039	0.074	0.238	0.524	0.378	0.488			
18	0.301	0.106	1.518	0.421	0.290	0.502	0.072	0.014	0.607
19	0.061	0.111	0.391	0.122	0.192	0.169			
20	1.607	2.500	2.741	20.946	0.701	0.559	3.723	0.969	16.031
25					0.176	0.090	0.789		
26									
30	4.620	7.749	19.340	43.835	3.953	3.493	4.143	1.252	13.804

Table 4-11. Area-specific and maximum annual ingestion rates (Bq/yr).

Area	Am-241	Pu-238	Pu-239, 240	Co-60	Cs-137	Sr-90	Eu-152	Eu-154	Eu-155
1	1.72	3.17	9.38	17.06	6.56	11.41	25.12	0.35	9.77
2	1.59	5.64	11.57	25.03	24.08	47.06	31.54		10.51
3	1.54	1.24	11.87	12.72	7.34	20.59	24.73	0.29	8.01
4	4.47	10.49	25.90	41.10	14.83	16.37	25.24		6.47
5	2.24	0.45	17.15	85.03	2.73	6.25	153.05	6.48	0.00
6	0.57	1.32	2.69	2.54	1.71	2.18			
7	1.23	0.40	8.59	21.29	5.33	9.61	50.59	0.97	8.05
8	13.24	7.43	81.99	168.52	59.73	36.25	14.05	0.00	22.34
9	2.27	1.42	46.11	14.38	8.60	13.10	51.04	0.94	7.76
10	10.29	12.27	56.98	199.31	83.02	55.42	4.88	1.41	129.41
11	8.93	1.61	75.12	0.00	2.47	1.51			
12	1.56	2.77	10.20	12.45	9.98	8.65			
15	2.45	2.85	18.49	3.49	10.64	12.56			
16	0.53	1.35	2.68	2.87	4.01	5.21			
17	0.97	1.85	5.94	13.09	9.44	12.19			
18	7.54	2.65	37.95	10.54	7.24	12.55	1.79	0.34	15.17
19	1.53	2.79	9.78	3.05	4.80	4.21			
20	40.17	62.49	68.51	523.64	17.54	13.98	93.06	24.23	400.77
25					4.39	2.24	19.73		
26									
30	115.50	193.72	483.50	1,095.89	98.83	87.33	103.56	31.30	345.10
Max[a]	**40.17**	**62.49**	**81.99**	**523.64**	**83.02**	**55.42**	**153.05**	**24.23**	**400.77**

a. Maximum value with Area 30 excluded.

To simplify the application of organ-specific ingestion dose from short-lived fission and activation products, the IMBA computer program was used to determine organ-specific annual doses for the ^{90}Sr intake of 55.42 Bq/yr for 1963 through 1972. These doses are listed in Table 4-13. The doses in

Table 4-13 are multiplied by the Table 4-12 organ-specific correction factors to provide the annual doses resulting from ingestion of short-lived fission and activation products. The annual organ-specific doses that were greater than 0.001 rem are listed in Table 4-14.

Table 4-12. Organ-specific ingestion fission and activation correction factors (from Attachment A).

Organ	Correction factor by year									
	1963	1964	1965	1966	1967	1968	1969	1970	1971	1972
Adrenals, breast Brain, skin, bladder Stomach, kidneys, muscle, pancreas, brain, esophagus, SI, liver, ovaries ET ET1 ET2 LM(ET) LN(TH), lungs, skin, spleen, testes, thymus, thyroid, uterus Gall bladder	416	219	155	123	106	95.0	88.1	84.0	82.0	81.8
ULI	514	184	95.2	58.2	39.3	18.6	16.3	14.6	13.4	12.4
Bone surface/RBM	3.8	3.1	2.7	2.5	2.3	2.3	2.3	2.3	2.3	2.3
LLI/colon	417	208	138	25.6	20.4	17.4	14.6	12.8	11.3	10.2

Table 4-13. Organ-specific annual ingestion doses (rem) for the ^{90}Sr intake of 55.42 Bq/yr.

Organ	1963	1964	1965	1966	1967	1968	1969	1970	1971	1972
Adrenals	2.11E-06	2.47E-06	2.64E-06	2.77E-06	2.89E-06	3.00E-06	3.08E-06	3.16E-06	3.22E-06	3.28E-06
Urinary bladder	5.68E-06	6.42E-06	6.67E-06	6.87E-06	7.04E-06	7.18E-06	7.31E-06	7.41E-06	7.50E-06	7.58E-06
Bone surface	1.58E-04	3.74E-04	5.47E-04	6.98E-04	8.32E-04	9.50E-04	1.06E-03	1.15E-03	1.23E-03	1.31E-03
Brain	2.11E-06	2.47E-06	2.64E-06	2.77E-06	2.89E-06	3.00E-06	3.08E-06	3.16E-06	3.22E-06	3.28E-06
Breast	2.11E-06	2.47E-06	2.64E-06	2.77E-06	2.89E-06	3.00E-06	3.08E-06	3.16E-06	3.22E-06	3.28E-06
Esophagus	2.11E-06	2.47E-06	2.64E-06	2.77E-06	2.89E-06	3.00E-06	3.08E-06	3.16E-06	3.22E-06	3.28E-06
Stomach	3.42E-06	3.78E-06	3.95E-06	4.09E-06	4.21E-06	4.31E-06	4.40E-06	4.47E-06	4.54E-06	4.59E-06
Small intestine	4.71E-06	5.07E-06	5.24E-06	5.38E-06	5.50E-06	5.60E-06	5.69E-06	5.76E-06	5.83E-06	5.88E-06
Upper large intestine	3.42E-05	3.43E-05	3.43E-05	3.43E-05	3.43E-05	3.43E-05	3.43E-05	3.43E-05	3.43E-05	3.43E-05
Lower large intestine	1.45E-04	1.45E-04	1.45E-04	1.45E-04	1.45E-04	1.45E-04	1.45E-04	1.45E-04	1.45E-04	1.45E-04
Colon	4.36E-03	2.26E-03	1.51E-03	1.14E-03	9.16E-04	7.71E-04	6.61E-04	5.84E-04	5.21E-04	4.71E-04
Kidneys	2.11E-06	2.47E-06	2.64E-06	2.77E-06	2.89E-06	3.00E-06	3.08E-06	3.16E-06	3.22E-06	3.28E-06
Liver	2.11E-06	2.47E-06	2.64E-06	2.77E-06	2.89E-06	3.00E-06	3.08E-06	3.16E-06	3.22E-06	3.28E-06
Muscle	2.11E-06	2.47E-06	2.64E-06	2.77E-06	2.89E-06	3.00E-06	3.08E-06	3.16E-06	3.22E-06	3.28E-06
Ovaries	2.11E-06	2.47E-06	2.64E-06	2.77E-06	2.89E-06	3.00E-06	3.08E-06	3.16E-06	3.22E-06	3.28E-06
Pancreas	2.11E-06	2.47E-06	2.64E-06	2.77E-06	2.89E-06	3.00E-06	3.08E-06	3.16E-06	3.22E-06	3.28E-06
Red marrow	1.08E-04	2.66E-04	3.87E-04	4.86E-04	5.69E-04	6.37E-04	6.94E-04	7.42E-04	7.81E-04	8.14E-04
Extrathoracic airways	2.11E-06	2.47E-06	2.64E-06	2.77E-06	2.89E-06	3.00E-06	3.08E-06	3.16E-06	3.22E-06	3.28E-06
Lungs	2.11E-06	2.47E-06	2.64E-06	2.77E-06	2.89E-06	3.00E-06	3.08E-06	3.16E-06	3.22E-06	3.28E-06
Skin	2.11E-06	2.47E-06	2.64E-06	2.77E-06	2.89E-06	3.00E-06	3.08E-06	3.16E-06	3.22E-06	3.28E-06
Spleen	2.11E-06	2.47E-06	2.64E-06	2.77E-06	2.89E-06	3.00E-06	3.08E-06	3.16E-06	3.22E-06	3.28E-06
Testes	2.11E-06	2.47E-06	2.64E-06	2.77E-06	2.89E-06	3.00E-06	3.08E-06	3.16E-06	3.22E-06	3.28E-06
Thymus	2.11E-06	2.47E-06	2.64E-06	2.77E-06	2.89E-06	3.00E-06	3.08E-06	3.16E-06	3.22E-06	3.28E-06
Thyroid	2.11E-06	2.47E-06	2.64E-06	2.77E-06	2.89E-06	3.00E-06	3.08E-06	3.16E-06	3.22E-06	3.28E-06
Uterus	2.11E-06	2.47E-06	2.64E-06	2.77E-06	2.89E-06	3.00E-06	3.08E-06	3.16E-06	3.22E-06	3.28E-06

Table 4-14. Organ-specific doses (rem) from ingestion of fission and activation products.

Organ	Year									
	1963	1964	1965	1966	1967	1968	1969	1970	1971	1972
Bladder	0.002	0.002	0.001							
Bone surface		0.001	0.002	0.002	0.002	0.003	0.003	0.003	0.003	0.003
Stomach	0.001									
SI	0.002	0.001								
ULI	0.015	0.006	0.003	0.002	0.001					
LLI	0.048	0.025	0.017	0.003	0.003	0.002	0.002	0.002	0.001	0.001

Organ	Year									
	1963	1964	1965	1966	1967	1968	1969	1970	1971	1972
Colon	0.028	0.015	0.010	0.002	0.002	0.001	0.001	0.001		

4.3 OCCUPATIONAL EXTERNAL DOSE

Workers incur external doses from ambient radiation levels and noble gases. Ambient radiation measurements were reported in NTS annual environmental reports (see References section) starting in 1967; no ambient radiation data were provided in the reports for 1968 to 1976.

4.3.1 Ambient Radiation

Before 1967, ambient radiation levels that were unaffected by weapons testing were not reported in the annual environmental reports. Although there were many radiation measurements between 1951 and 1967, most of these were to characterize the effects of weapons tests and, therefore, were not appropriate for use in estimating external environmental dose for unmonitored employees.

In 1967, ambient radiation levels were measured using Victoreen Model 239 indirect reading ionization chambers with effective ranges of 0 to 10 mR/hr. Five of these chambers were in small, semiprotective enclosures in NTS living areas (REECo 1968). As a backup for the ionization chambers, standard NTS film dosimeters were included at each sample location. However, all results from these dosimeters for 1967 were either zero or lost due to light or heat damage. Results from environmental NTS film dosimeters were subsequently discarded because of the likelihood of heat damage.

The ionization chambers were collected on a weekly basis and read on a Victoreen Minometer II reader. Corrections were made for background and for nonradiation-induced drift. Two sets of chambers were used, one at the sample locations for measurements and another stored fully charged in the laboratory. Each week these sets were exchanged, the fresh set being recharged and a record kept of the amount of "drift" while stored in the laboratory. A specially designed shock-proof box for transportation minimized accidental discharge from mechanical shock.

Readings from the five chambers in each location were averaged to obtain a mean value for each location each week. Readings significantly higher than others at a particular location were not used in compiling the data because the abnormal readings were most likely to be the result of shock or other malfunction and were not representative measurements. Because of the method of background subtraction discussed above, the data in the 1967 annual report were assumed to be due to manmade radiation in excess of natural background.

No results were reported between 1968 and 1976 in the NTS annual environmental reports; ambient radiation reporting was reestablished in 1977 using thermoluminescent dosimeters (TLDs) (Lantz 1978a). The dosimeters were CaF_2:Dy (TLD-200) 0.25- by 0.25- by 0.035-in. chips from Harshaw Chemical Company. A badge that consisted of at least two chips shielded by 0.047 in. of cadmium (1,030 mg/cm^2) inside a 0.050-in. black plastic (140 mg/cm^2) holder was placed about 1 m above the ground at each of 10 locations that coincided with air sampling stations. These sites were selected because of their proximity to workers. During that year, the natural background ambient radiation level was established at 0.26 mrem/d or about 95 mrem/yr.

In 1987, the Harshaw dosimeters were replaced with Panasonic Model UD-814 TLDs (Gonzalez 1988). These specifically designed environmental dosimeters contain three identical $CaSO_4$:Tm elements and one $Li_2B_4O_7$:Cu element. The lithium element is shielded with 14 mg/cm^2 of material to monitor beta particles in the environment. The three calcium elements are encapsulated in 1,000 mg/cm^2 of plastic and lead to monitor ambient gamma levels. The UD-814 TLDs have remained in use to the present time.

In subsequent years, the number of sampling locations increased to more than 150 and eventually covered all populated areas at NTS. Table 4-15 lists average ambient background radiation levels. Table 4-16 lists the annual measured ambient radiation including background and site average and maximum values. For dose reconstruction, because the values in Table 4-16 represent continuous exposure for an entire year, these values should be adjusted for occupational exposure (i.e., 2,000 hr/yr) and added to the dose of record for unmonitored employees. If the area in which the employee worked is known, the average value for that area should be used. If the area is not known, the site maximum value should be used to be favorable to claimants.

Table 4-15. Average ambient background radiations levels by year (mrem/yr).[a]

Year	Level	Year	Level	Year	Level
1964	(b)	1980	99	1997	95
1965	(b)	1981	110	1998	88
1966	(b)	1982	100	1999	91
1967	(b)	1983	100	2000	115
1968	(b)	1984	100	2001	114
1969	(b)	1985	100	2002	105
1970	(b)	1986	72	2003	112
1971	(b)	1987	102	2004	110
1972	(b)	1988	131	2005	119
1973	(b)	1989	106	2006	125
1974	(b)	1990	110	2007	121
1975	(b)	1991	112	2008	116
1976	(b)	1992	109	2009	120
1977	95	1994	93	2010	116
1978	95	1995	94		
1979	95	1996	91		

a. Sources: Environmental reports in the References section.
b. Not reported.

Table 4-16. Ambient radiation by area (mrem/yr).[a,b]

	Area													
Year	**1**	**2**	**3**	**4**	**5**	**6**	**7**	**8**	**9**	**10**	**11**	**12**	**13**	**14**
1967			318			194						205		
1968			285		183	175						190		
1969														
1970														
1971														
1972														
1973														
1974														
1975														
1976														
1977														
1978	130	167	200		138	110	110	120	150	320	140	150		
1979	119	147	190	140	133	104	125	115	140	168	155	191		
1980	123	156	199	155	133	106	130	135	145	192	165	196		
1981	120	155	218	150	166	113		125	150	180	160	187		
1982	126	149	188	145	141	108		125	140	209	140	189		
1983	115	141	181	143	144	103		95	135	201	140	174		
1984	97	149	162	116	111	86	327	125	102	166	112	155		
1985		141	78	101		107	347	94	114	225		180		
1986	93	126	150	92	92	77	318	94	96	120	107	106		
1987	146	189	184	164	191	126		120	149	181	133	168		
1988	155	209	207	164	177	134		150	179	203	158	194		
1989	140	217	205	132	141	106		126	151	179	153	139		

	Area													
Year	**1**	**2**	**3**	**4**	**5**	**6**	**7**	**8**	**9**	**10**	**11**	**12**	**13**	**14**
1990	134	167	187	142	142	109		122	148	170	132	140		
1991	128	168	194	146	165	116	132	124	132	150	130	190		
1992	123	167	173	141	150	104	145	128	136	157	158	170		
1993	147	178	200	158	198	134	165	153	156	172	184	201		
1994		62	62		89	84				70	102	97		
1995	102	136	150	121	212	87	176	102	104	179	124	130		
1996	102	131	151	110	225	105	164	98	99	176	124	135		
1997			131		86	88			81					
1998	106	132	137	110	134	100	201	127	101	168	121	148		
1999	108	133	135	122	110	105	191	129	99	160	122	152		
2000	127	155	161	129	134	122	217	147	190	187	144	176		
2001	121	299	151	120	136	107	214	143	115	180	140	163		
2002	113	272	168	341	122	98	174	135	106	167	131	162		
2003	180	260	257	334	145	104	159	209	189	188	133	169		
2004	162	251	239	323	148	106	157	204	190	185	130	166		
2005	159	246	236	320	148	108	159	205	191	182	132	166		
2006	154	243	228	309	144	108	156	202	186	182	133	167		
2007	155	242	228	306	147	112	158	199	190	184	138	171		
2008	148	227	219	298	138	107	150	189	179	174	129	159		
2009	151	229	218	283	140	109	152	191	180	175	134	164		
2010	146	216	250	269	144	98	149	184	173	170	132	156		

	Area																	
Year	**15**	**16**	**17**	**18**	**19**	**20**	**21**	**22**	**23**	**24**	**25**	**26**	**27**	**28**	**29**	**30**	**Average**	**Maximum**
1967									142				175				207	318
1968				186					131				172				189	285
1969																		*320*[c]
1970																		*320*[c]
1971																		*320*[c]
1972																		*320*[c]
1973																		*320*[c]
1974																		*320*[c]
1975																		*320*[c]
1976																		*320*[c]
1977																		*320*[c]
1978	129			173	185				150		162		140	120			155	320
1979	121		150	145	155	160		115	117		152		130	120		125	140	191
1980	134		165	163	185	207		70	98		163		135	130		205	152	207
1981	138		155	168	174	163		75	126		175		150			140	152	218
1982	134		150	173	176	200		75	95		150		135	135		140	146	209
1983	124		135	151	158	206		125	163		149		140			185	148	206
1984	108		129	124	132	157		58	154		129					173	137	327
1985			115	123	131	127					103					153	143	347
1986	81			124	134	160		55	71		106		89			135	115	318
1987	135		151	174	246	204		68	96		129		139				155	246
1988	155		172	212	223	211		84	123		163						172	223
1989	162		160	168	178	189		78	116		146		118				150	217
1990	124		164	174	178	192		83	177		139		146				148	192
1991	122		165	175	173	189		81	79		143		154	110		155	144	194
1992	134		152	168	189	185		79	75		132		143			179	145	189
1993	145		177	195	192	213		91	96		158		169			195	167	213
1994	143				141	131		58	61		115		121			140	98	143
1995	109		129	137	149	153		58	70		117		121				127	212
1996	103		137	144	153	150		55	56		114		124			165	128	225

	Area																	
Year	**15**	**16**	**17**	**18**	**19**	**20**	**21**	**22**	**23**	**24**	**25**	**26**	**27**	**28**	**29**	**30**	**Average**	**Maximum**
1997	102				150	153		83	63		127		135			170	114	170
1998	99			135	151	156		76	58		113		123			165	127	201
1999	108			137	153	151		75	57		111		126			170	126	191
2000	117			160	185	182		92	81		139		158			205	153	217
2001	112			167	159	131		90	74		135		145			193	147	299
2002	103			144	172	157		81	68		122		128				146	341
2003	116	133		146	165	334			65		127		113				176	334
2004	111	137		145	158	309			93		123		112				172	323
2005	114	144		146	164	305			116		125		115				174	320
2006	117	143		149	161	290			76		125		115				169	309
2007	116	145		151	165	287			98		128		117				172	306
2008	111	137		144	161	272			68		124		114				162	298
2009	115	133		148	160	255			63		117		116				162	293
2010	111	140		142	157	247			68		116		114				152	269

a. Sources: Environmental Reports list in the References section.
b. Blank cells indicate data not available.
c. Ambient radiation not provided for these years; assumed equal to value for 1978.

Some covered employees remained on site continuously for weeks at a time. However, because most of the nonworking hours were spent indoors where elevated ambient radiation would be reduced by shielding from the building and because of the conservative assumptions used to estimate the values in Table 4-16, adjustment of the tabular data is not required to ensure ambient exposures are not underestimated for these individuals.

Although not explicitly stated in the NTS annual environmental reports, the overall uncertainty of measurements from modern TLDs (e.g., Panasonic UD-814) has been determined to be less than 20% (ORAUT 2010a). However, for earlier measurement methods, the overall uncertainty could have been greater. Therefore, for dose reconstructions that are favorable to claimants, the assumption is made that the 95th-percentile values of the expected values (50th percentile) in Tables 4-15 and 4-16 are enveloped by a factor of ±2 of the overall error of 50% (i.e., ±100%). Equation 4-1 (Section 4.2.1.2.1) would then estimate the GSD for the expected values at 1.52.

Assignment of dosimeters to all persons at NTS began in April 1957 and ended in December 1992 (ORAUT 2010a). During this time, readings from control badges at Mercury were used to subtract ambient background dose from the personnel dosimeters (ORAUT 2010a). For this reason, from April 1957 to December 1992, ambient background external radiation dose should not be included in dose reconstructions. Because no documentation is available prior to 1967 (see Tables 4-15 and 4-16), ambient external doses should not be assigned under any circumstances from 1951 through 1956. After 1992, ambient background external dose should be assigned to all unmonitored workers in accordance with the guidelines for Tables 4-15 and 4-16.

4.3.2 <u>Releases of Noble Gases</u>

Ground seepage can increase when changes in ambient pressure pump small amounts of noble gases (primarily ^{85}Kr and ^{133}Xe) up through the overburden and into the atmosphere from the cavity created by a nuclear test. This process, sometimes referred to as *atmospheric pumping*, creates a diffuse source of radiological effluents.

In 1982, REECo assumed responsibility for six noble gas sampling stations EPA previously ran and replaced them with new samplers. These sampling units were housed in a metal toolbox with three metal air bottles attached with quick disconnect hoses. A vacuum was maintained on the first bottle, which caused a steady flow of air to be collected in the other two bottles. The flow rate was approximately 0.5 cm^3/min. The two collection bottles were exchanged weekly and yielded a sample volume of about 3×10^5 cm^3 (Scoggins 1983).

The noble gases were separated and collected from the atmospheric sample by a series of cryogenic gas chromatographic techniques. Water and carbon dioxide were removed at room temperature and krypton and xenon were collected on charcoal at liquid nitrogen temperatures. These gases were transferred to a molecular sieve where they were separated from remaining gases and each other. The krypton and xenon were transferred to separate scintillation vials and counted on a liquid scintillation counter. The lower limits of detection for krypton and xenon are 4×10^{-6} and 10×10^{-6} pCi/m^3, respectively (Scoggins 1983).

Krypton-85

The original six samplers were in Areas 1, 5, 12, 15, 23, and 25. These samplers showed no statistical difference in atmospheric concentrations of ^{85}Kr during 1982 and 1983. The average concentration of all stations for these years was 24.6 pCi/m^3 (0.91 Bq/m^3), which was established as NTS background. This background measurement shows good agreement with the global background of 27 pCi/m^3 (1.0 Bq/m^3) (WMO 2002).

In 1983, REECo measured a statistically significant elevation in ^{85}Kr concentrations in Area 20 of 4.0 pCi/m^3 (0.15 Bq/m^3) above background, which was subsequently related to seeps from nuclear cavities on Pahute Mesa (Scoggins 1984). These elevated concentrations continued until after the cessation of nuclear testing in 1992. Table 4-17 lists the average background and the Area 20 elevated concentrations from 1983 through 1992.

Table 4-17. ^{85}Kr average background and the Area 20 net concentrations and annual organ doses from 1983 through 1992.[a]

Year	NTS background (pCi/m^3)	Area 20 camp (pCi/m^3)	Net elevated (pCi/m^3)	Source[b] elevated concentration (pCi/m^3)	Source[b] elevated concentration (Bq/m^3)	Other organ[c] dose[d] at source[b] (mrem/yr)	Skin dose[e] at source[b] (mrem/yr)
1983	27	46	19	502	18.6	0.0029	0.18
1984	25	29	4	106	3.9	0.0006	0.04
1985	28	39	11	290	10.8	0.0017	0.10
1986	30	58	28	739	27.4	0.0043	0.26
1987	26	39	13	343	12.7	0.0020	0.12
1988	25	29	4	106	3.9	0.0006	0.04
1989	23	27	4	106	3.9	0.0006	0.04
1990	29	37	8	211	7.8	0.0012	0.07
1991	24	32	8	211	7.8	0.0012	0.07
1992	26	30	4	106	3.9	0.0006	0.04
1993	27	28	1	26	1.0	0.0002	0.01

a. Sources: Environmental reports in the References section.
b. Source receptor location assumed to be 500 m from U-20a emplacement hole.
c. Includes gonads, breast, lung, red marrow, bone surfaces, thyroid, remainder, and effective whole body.
d. Assumes 2,000 hr/yr exposure and dose conversion factor of 2.2×10^{-16} Sv/Bq/s/m^3 (Eckerman and Ryman 1993).
e. Assumes 2,000 hr/yr exposure and dose conversion factor of 1.32×10^{-14} Sv/Bq/s/m^3 (Eckerman and Ryman 1993).

The source of the elevated ^{85}Kr concentrations was attributed to the emplacement hole nearest the Area 20 sampler (U-20a), which is about 3,660 m (12,000 ft) to the south. To determine if these elevated concentrations could be of any consequence for unmonitored workers, upper bound concentrations of ^{85}Kr within 500 m of U-20a were estimated using Area 20 meteorological data station files and the CAP88-PC (ORAUT 2003) computer program. The program was used to develop atmospheric dispersion factors (X/Q) for various distances from the source of emissions. The ratio of the X/Q value at 500 m to that at 3,660 m was determined to be about 27, which means that the ^{85}Kr concentration 500 m from the source could, on average, be 27 times greater than that measured at the sampler location. The potential dose resulting from submersion in the plume of this noble gas is entirely external because the gas is not readily assimilated by the body. Therefore, these source concentrations were converted to potential organ doses using submersion dose conversion factors from Federal Guidance Report No. 12 (Eckerman and Ryman 1993).

As Table 4-17 indicates, the upper bound annual organ doses for all years are less than 1 mrem. For reconstructing environmental doses for unmonitored employees, these doses are inconsequential for determination of POC and, therefore, should be ignored. In addition, if a monitored employee was exposed to the calculated upper bound ^{85}Kr concentrations or significantly higher concentrations during work activities, these exposures would have been monitored by personal dosimeters and would, therefore, already be in the individual dosimetry record.

Uncertainty estimates for a number of ^{85}Kr measurements indicate, generally, a standard deviation of less than 20% of the measured value. Even at the 2-sigma level of ±40 percent, the annual organ doses in Table 4-17 are less than 1 mrem and, therefore, inconsequential for employee dose reconstruction.

Xenon-133

Although the vast majority of the ^{133}Xe measurements were below the lower limit of detection (i.e., 10×10^{-6} pCi/m^3), elevated atmospheric concentrations of ^{133}Xe were occasionally measured in Area 20. As with ^{85}Kr, these elevated concentrations were attributed to seepage from underground test cavities at the Pahute and Rainier Mesa tests (Fauver 1985). However, ^{133}Xe has a short half-life (5.25 days versus 10.7 years for ^{85}Kr). Therefore, these elevated concentrations were short-lived. In addition, these elevated ^{133}Xe concentrations were always less than the derived air concentrations (typically less than 3%) and would result in annual organ doses of less than 1 mrem. Therefore, potential doses to unmonitored employees from elevated concentrations of ^{133}Xe are inconsequential for dose reconstruction.

4.4 INTERNAL INHALATION AND INGESTION DOSE TO UNDERGROUND WORKERS

In the early 1980s, the Environmental Sciences Department of REECo recognized that the buildup of radon and radon daughter concentrations (RDCs) could pose a potential health problem in tunnels on Rainer Mesa and at other NTS locations. In 1984, to determine the concentrations of the RDCs and the effect of environmental conditions on the buildup of these concentrations, REECo conducted radon measurement surveys in G-, T-, and N-Tunnels (Fauvor 1987). This section discusses the results of these surveys and those in 1991 and 1992 (Lyons 1992a,b).

4.4.1 Underground Activities

Area 12, which is in the Nuclear or High Explosive Test Zone, occupies 104 km^2 (40 mi^2) at the northern boundary of NTS, known as Rainier Mesa. No atmospheric nuclear tests have occurred at this location; however, Area 12 was the site of the U.S. Atomic Energy Commission's first fully contained underground nuclear detonation, the RAINIER test, on September 19, 1957, in a horizontal tunnel about 487 m (1,600 ft) into the mesa and 274 m (900 ft) beneath the top of the mesa (ORAUT 2008). In the past several decades, a number of tunnels have been mined into Rainier Mesa, in which most of the U.S. Department of Defense (DOD) horizontal line-of-sight exposure experiments have occurred. The N-, P-, and T-Tunnel complexes, in particular, were developed extensively during the 1970s and 1980s. The tunnel experiments usually involved complex construction of large-diameter [up to 9 m (27 ft)], line-of-sight pipes and special closure mechanisms, blast and gas seal doors, stemming plugs, and the like. The G-Tunnel complex was originally established for nuclear testing but, since 1971, has been used only as an underground research facility (ORAUT 2008).

In addition to its use for nuclear testing purposes, N-Tunnel was the location of a Nonproliferation Experiment involving 1.3 million kg (2.9 million lb) of conventional explosives on September 22, 1993. DOD operates a high-explosives research and development tunnel in Area 12. This reusable test bed supports programs involving the detonation of conventional or prototype high explosives and munitions (ORAUT 2008).

The U1a Complex in Area 1 is an underground laboratory of horizontal tunnels about 0.5 mi long at the base of a vertical shaft about 960 ft beneath the surface. The vertical shaft is equipped with a mechanical hoist for personnel and equipment access. Another vertical shaft about 1,000 ft away provides cross ventilation, instrumentation, utility access, and emergency egress. The shaft was excavated in the 1960s, and a nuclear test was conducted in a horizontal tunnel mined from its base in 1990 (ORAUT 2008).

4.4.2 Radon Measurements

Radon-222, with a radioactive half-life of 3.8 days, occurs in the ^{238}U decay chain; ^{220}Rn, with a radioactive half-life of 54.5 seconds, occurs in the ^{232}Th chain. The decay daughters associated with ^{222}Rn include ^{218}Po, ^{214}Pb, ^{214}Bi, and ^{214}Po, and those associated with ^{220}Rn include ^{216}Po, ^{212}Pb, ^{212}Bi, and ^{208}Tl. Because radon is a noble gas and thus chemically inert, it migrates easily from the tunnel rock and soil, which contains naturally occurring trace quantities of uranium and thorium.

Radon progeny measurements in N- and T-tunnels were made in drifts mined in a rock formation known as Tunnel Bed Non-Welded Ash Fall Tuff. This rock is unconsolidated and only slightly fractured. The drift that was sampled in the inclined G-tunnel was mined in Grouse Canyon Welded Ash Fall Tuff, which is extremely fractured. Factors that affect concentrations of radon and radon progeny in air are ventilation rates, barometric pressure, relative humidity, temperature inversions, the degree of fracturing in the rock, and the amount of smoke and dust in the air. Factors such as barometric pressure and fractures affect the rate at which radon emanates from the rock, while others such as dust and ventilation rates affect the accumulation rate of progeny in the air. Concentrations fluctuate with changing seasons, weather conditions, and activities in the area being monitored (Fauver 1985).

The concentration of radon progeny (the major dose contributors) in air is measured in working levels (WL). This is the common unit for expressing radon progeny exposure rates. The WL was developed for use in uranium mines but is now used for environmental exposures. Numerically, the WL is any combination of short-lived decay products in 1 L of air that results in the emission of 1.3×10^5 MeV of potential alpha energy. When radon is in complete equilibrium with its short-lived decay products, 1 WL equals 100 pCi/L (i.e., 100 pCi/L each of ^{222}Rn and short-lived decay products ^{218}Po, ^{214}Pb, ^{214}Bi, and ^{214}Po) (NCRP 1988, p. 17). For ^{220}Rn and its decay products (^{216}Po, ^{212}Pb, ^{212}Bi, and ^{208}Tl), 1 WL is equal to 7.47 pCi/L. The advantage of the WL unit is that it enables comparison of different equilibrium levels and different concentrations of radon decay products. The degree of equilibrium is a critical factor for estimating inhalation exposure and is of equal importance to the radon concentration itself (NCRP 1988, p. 19). The WL unit considers this factor. However, because the dose conversion factors for ^{220}Rn progeny (^{212}Pb + ^{212}Po) are one-third the values for the ^{222}Rn progeny (ICRP 1981), the dose equivalent WL for ^{220}Rn is only one-third that of a ^{222}Rn WL.

The exposure of tunnel workers can be expressed in units of working-level months (WLM), which is an exposure rate of 1 WL for a working month of 170 hours (NCRP 1988, p. 17). For example, an exposure of 1 WLM would result from exposure to a concentration of 1 WL for 1 month or 0.5 WL for 2 months.

4.4.3 Underground Radon Concentrations

Measurements were taken in tunnels N, T, and G. The preliminary measurements indicated N- and T-Tunnel RDCs of about 0.01 WL under normal ventilation conditions. However, the data demonstrated that RDCs could rise to relatively high levels (i.e., 0.24 WL) when ventilation rates were significantly lower (Fauvor 1987, Figure 2, p. 7). The radon daughter concentrations in G-Tunnel were an order of magnitude higher than those in N- and T-Tunnels. The average RDC in the rock mechanics drift (the worst-case location in G-Tunnel) was 0.13 WL (ranging from 0.07 to 0.23 WL). Elevated RDCs in the rock mechanics drift of G-Tunnel seemed to be from a lower ventilation rate in conjunction with the more highly fractured nature of the welded tuff in which the incline drift was mined. By increasing the ventilation rate, a 60% reduction in RDCs from an average of 0.13 WL to an average of 0.05 WL was achieved (Table 4-18).

Table 4-18. Results of experiment to determine the effects of ventilation conditions on RDCs in the G-Tunnel inclined drift.[a]

Location	Ventilation conditions	Radon daughter integrated sample average (WL)
Rock mechanics drift at 0+52	Alternating	0.19
	Continuous	0.05
Average	Alternating	0.13
	Continuous	0.03

a. Source: Fauvor (1985, Table 3).

Lyons (1992a,b) reported additional radon measurements from 1991 and 1992. Although the G-Tunnel complex remained inactive in 1992, radon samples were taken to document potential radon WL in a worst-case scenario of complete ventilation failure throughout the complex. The maximum WL for radon daughters was 1.4. No samples were taken in 1992 in T-Tunnel, which was inactive that year.

Tables 4-19 and 4-20 list the results of extensive radon sampling in N- and P-Tunnel complexes in 1992 (from Lyons 1992a, Table 3, p. 8, and Lyons 1992b, Table 1, p. 8, respectively). The average concentrations in N- and P-Tunnels were 0.021 and 0.009 WL, respectively, and the maximum concentrations in N- and P-Tunnel were 0.038 and 0.017 WL, respectively.

4.4.4 Underground Worker Exposure to Radon

Although measurements were periodically performed in the tunnel complexes to ensure adequate worker protection, neither DOE nor its predecessor agencies attempted to quantify or record occupational exposures to radon and its daughters. Therefore, dose reconstructors should adjust the dose for lung, ET1, and ET2 cancers of any employee who was a miner or tunnel worker to account for radon exposure while working in the tunnel complexes. To quantify the exposure, Table 4-21 lists airborne ^{222}Rn RDCs based on measurement results that are favorable to claimants. The values in Table 4-21 pertain only to ^{222}Rn and its daughter products. The ^{220}Rn WL concentrations and resultant exposures for N- and P-Tunnels must be converted to annual alpha organ dose before entry in IREP.

For G-Tunnel workers, annual exposures (assuming a full year of underground activity) are assumed to be 1.92 WLM before 1984 and 0.60 WLM during and after 1984. This is because REECo did not recognize until 1984 that significant reductions in RDCs could be effected by leaving all ventilation fans running overnight. Until 1984, the practice was to shut down alternating fans each night, which resulted in higher average RDCs.

Table 4-19. Radon daughter concentrations for N-Tunnel in 1991 and 1992.

N-Tunnel location	Rn-222 (WL) Average	Rn-222 (WL) Maximum	Rn-220 (WL) Average	Rn-220 (WL) Maximum
January–June 1992				
Miner's lunchroom	0.005	0.007	0.01	0.016
Raytheon Alcove	0.003	0.004	0.005	0.008
Slow Alcove	0.005	0.007	0.008	0.009
24 Bypass Drift	0.005	0.006	0.015	0.015
24 LOS Drift at GZ	0.005	0.007	0.012	0.014
22 Bypass	0.006	0.007	0.013	0.015
July–December 1991				
Miner's lunchroom	0.005	0.007	0.011	0.018
Raytheon Alcove	0.009	0.03	0.017	0.06
21 LOS at 2 + 50	0.034	0.059	0.029	0.046
15 Assembly Drift	0.006	0.009	0.015	0.021
Slow Alcove	0.004	0.008	0.01	0.025
23 Fast Alcove	0.009	0.014	0.018	0.03
24 Bypass Drift	0.005	0.007	0.012	0.041
24 LOS Drift	0.005	0.01	0.015	0.036
Average	**0.008**	**0.013**	**0.014**	**0.025**

Table 4-20. Radon daughter concentrations for P-Tunnel in 1991 and 1992.

P-Tunnel location	Rn-222 (WL) Average	Rn-222 (WL) Maximum	Rn-220 (WL) Average	Rn-220 (WL) Maximum
January–June 1992				
01 Drift at Access Drift	0.001	0.001	0.002	0.005
01 Fast Alcove	0.001	0.001	0.004	0.006
02 Main Drift at 6 + 00	0.001	0.001	0.002	0.004
04 Reentry	0.001	0.002	0.004	0.005
04 LOS at 12 + 00	0.002	0.002	0.005	0.006
HPD Base Station	0.001	0.001	0.003	0.006
Miner's lunchroom	0.001	0.001	0.003	0.006
05 Cavity	0.001	0.001	0.003	0.004
July–December 1991				
01 Drift at Access Drift	0.003	0.006	0.004	0.007
01 Fast Alcove	0.003	0.005	0.003	0.006
02 Main Drift at 6 + 00	0.01	0.032	0.01	0.046
04 LOS at VP X-Cut	0.008	0.013	0.012	0.015
04 LOS Drift at GZ	0.005	0.006	0.014	0.02
04 LOS Test Ch.	0.003	0.004	0.005	0.006
04 Bypass at RE#1	0.007	0.013	0.008	0.011
IHD Alcove	0.003	0.009	0.004	0.015
LLNL Alcove	0.003	0.006	0.007	0.02
05 Cavity	0.003	0.006	0.006	0.015
Average	**0.003**	**0.006**	**0.006**	**0.011**

Because no radon measurement data were readily available for the underground portions of the U1a Complex, the assumption, which is favorable to claimants, was made that the RDCs in U1a would be similar to those in G-Tunnel (which is mined in highly fractured tuff) [1]. In addition, a reasonable assumption was made that the U1a Complex is well ventilated, similar to G-Tunnel. Dose reconstructors should use these average annual exposure values for miners and tunnel workers without identified tunnel locations.

Table 4-21. Annual ^{222}Rn exposures and uncertainties for internal dose reconstruction for miners and tunnel workers (Fauver 1985).

Tunnel complex	RDC concentration (WL)	Annual exposure (WLM)[a]	Uncertainty GSD[b]
G (before 1984)	0.16	1.92	1.52
G (1984 and later)	0.05	0.60	1.52
N	0.013	0.16	1.52
P	0.006	0.07	1.52
T	0.01	0.12	1.52
Unidentified[b] (before 1984)	0.16	1.92	1.52
Unidentified[b] (1984 and later)	0.05	0.60	1.52

a. Based on 170 hours of exposure per month for 12 months.
b. Use these values if underground work location is not known.

For dose reconstruction, the GSD of the values in Table 4-21 can be estimated using Equation 4-1 (Section 4.2.1.2.1). Under the assumption that the 50th-percentile expected annual exposures are those in Table 4-21 and that the 95th-percentile values are twice the values in Table 4-21, the GSD for all values in Table 4-21 is 1.52.

Gravel Gerties

As discussed in Part 2, Section 2.2.5 of this site profile (ORAUT 2008), underground bunkers were constructed in Area 5 for testing containment capabilities involving accidental explosion of high-explosive material in the underground bunker. Containment tests were conducted in the Gravel Gerties in 1957 and again in 1982. These tests resulted in successful designs that were used at the Pantex Plant in Texas and in the Device Assembly Facility in Area 6. Five Gravel Gerties were constructed in Area 6 from 1988 to 1992. However, with the cessation of testing in 1992, they have not been activated for device assembly.

Because the Area 5 Gravel Gerties were constructed for testing and the Area 6 Gravel Gerties were never activated, it is not likely that workers spent significant time inside the structures. However, if it can be determined from the dosimetry records or telephone interviews that a claimant worked in the Gravel Gerties for a significant time, it should be assumed that they would have been exposed to elevated levels of radon, and this radon exposure should be evaluated and included in the dose reconstruction.

Although radon measurements are not available for inside the NTS Gravel Gerties, because of the similarities of construction, application of the radon concentrations measured in the Pantex Gravel Gerties is assumed to be appropriate. Therefore, for a 12-month period for which it can be determined that a claimant worked in the NTS Gravel Gerties, a radon exposure of 0.072 WLM should be assigned (ORAUT 2007a, Section 5.3.3). This exposure can be prorated as appropriate for employment periods of less than a full year. These exposures should be applied as lognormal with a GSD of 3 (ORAUT 2007a, Section 5.3.3).

Uncertainty Associated with Radon Measurements

Two methods used to measure the concentration of radon and its progeny were the grab sample technique and the integration technique. Preliminary measurements were made by grab sampling for a general estimate of the concentrations. Lucas cells collected and counted grab samples of radon gas. The cells were evacuated in the laboratory and opened at the sampling location for a single intake of air. The filled cells were counted at the EPA Las Vegas Laboratory within 24 hours to minimize decay. The number of measurements was small due to limited availability of Lucas cells. Grab samples of radon daughters were collected and counted according to a technique described by Rolle (ANSI 1973), which utilized a single count of an air sample collected on a filter. Samples were collected on a 2-cm Whatman fiberglass filter at a rate of about 5 L/min and counted on an EDA Instruments Incorporated Radon Detector (Model RD200). The Rolle method allowed a choice of several different analysis regimes. The regime chosen for tunnel measurements was a 5-minute sampling time, a 6- minute decay time, and a 5- minute counting time. Using appropriate correction factors, the RDC in WL can be evaluated 11 minutes after sample collection. The values obtained using the Rolle method were periodically checked using the Kusnetz (Fauvor 1987, p. 3) method, which also used a single count of an air filter but allowed for a 40-to-90-minute decay time. These measurement techniques have good accuracy and a relative standard deviation of less than 15% (Fauvor 1987, p. 3).

Grab sampling is quick and convenient. However, as mentioned above, radon concentrations can fluctuate widely with time and location. To account for this variability, integrating monitoring instruments were used. A passive environmental radon monitor (PERM) was used to measure radon in picocuries per liter. A radon progeny integrating sampling unit (RPISU) measured the RDC in WL.

The PERM is an integrating radon monitor that employs electrostatic collection of radon daughter ions and uses a TLD as the radiation dose integration element. The lower limit of detection at the 95% confidence level is about 0.3 pCi/L for a 1-week sample with a relative standard deviation of 20% (Fauvor 1987, p. 4). The RPISU is a low-volume air sampler that draws air through a 0.65-µm-pore-size Millipore membrane filter. A Teflon disc TLD containing dysprosium-activated calcium fluoride (Harshaw TLD200) is positioned close to the collection surface of the filter and serves as the dose integrating element. The lower limit of detection at the 95% confidence level is about 0.0001 WL for a 1-week sampling period with a relative standard deviation of 10% (Fauvor 1987, p. 4). The RPISU technique is recommended for the measurement of RDCs by the DOE Technical Measurements Center, Division of Remedial Action Projects (Fauvor 1987, p. 4).

4.4.5 Underground Worker Exposure to Other Radionuclides

In addition to exposure to elevated levels of radon, underground workers could have been exposed to other sources of airborne contamination. Because the breathing air in the underground environment was drawn from the outside environment, underground workers would have been exposed to the same inhalation intakes as aboveground workers (see Sections 4.2.1.2.2 through 4.2.1.2.6).

4.5 INSTRUCTION TO DOSE RECONSTRUCTORS FOR ASSIGNMENT OF ENVIRONMENTAL INTAKES

4.5.1 Assigning External Ambient Dose

Starting in April 1957 and ending in 1992, all personnel entering NTS were monitored for external radiation exposure (ORAUT 2010a). Because unexposed control badges were processed with the personnel dosimeters and the readings from the control badges were subtracted from the dosimeter to obtain a net reading for determining exposure, ambient external dose should not be included in dose reconstruction before 1993. Before the institution of universal badging in April 1957 for employees who were not monitored, coworker doses should be assigned based on the values in ORAUT (2010a, Table 6-11) and no additional ambient dose should be assigned through 1992. Based on a review of the dosimetry literature of the site, no change in the handling of dosimeters occurred when universal badging was introduced; therefore, ambient dose is appropriately included in the coworker doses beginning in 1957 through 1992. After 1992, ambient external dose should be added to dose reconstruction for unmonitored workers in accordance with the guidelines for assigning dose from Tables 4-15 and 4-16.

External ambient doses for each year are available in Tables 4-15 and 4-16 and can be used for workers with continuous employment but nonexistent or incomplete monitoring records. For example, EG&G employees were likely to be stationed in Las Vegas and received dosimeters when they traveled to NTS. If the employee’s location cannot be clearly identified, it might be appropriate to assign ambient dose for intervals with no dosimeter exchanges. A second example includes REECo employees who were processed as new employees through NTS (Mercury) and assigned to work at Tonopah Test Range (TTR). These employees have at least one NTS dosimeter. Often no additional records are provided for these employees by Sandia National Laboratories, the responsible organization for dosimetry at TTR. For REECo employees assigned at TTR, ambient dose (ORAUT 2007b) can be assigned as appropriate.

4.5.2 <u>Assigning Internal Ambient Dose</u>

With the exception of cases that can be worked using the bounding assumptions in ORAUT-OTIB-0018 (ORAUT 2005), environmental inhalation and ingestion intakes listed in Tables 4-7 and 4-11, respectively, shall be applied starting in 1964.

To correct for exposure to short-lived fission and activation, the dose reconstructor should add annual doses that were greater than 0.001 rem from Tables 4-9 and 4-14 as 30-to-250-keV photons with a constant distribution. These doses shall be entered into the IREP analysis as constants because they can be considered bounding overestimates of the actual doses.

4.6 ATTRIBUTIONS AND ANNOTATIONS

Where appropriate in this document, bracketed callouts have been inserted to indicate information, conclusions, and recommendations provided to assist in the process of worker dose reconstruction. These callouts are listed here in the Attributions and Annotations section, with information to identify the source and justification for each associated item. Conventional References, which are provided in the next section of this document, link data, quotations, and other information to documents available for review on the Project's Site Research Database.

[1] Rollins, Eugene M. Oak Ridge Associated Universities Team. Division Manager. October 2007. The highest radon concentrations were measured in the G-Tunnel and occurred in fractured tuff. Therefore, with the lack of measured data in the U1a Complex, an assumption favorable to claimants was made that the radon concentrations in the U1a Complex were the same as the highest measured concentrations in the G-Tunnel complex.

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GLOSSARY

absorption

In external dosimetry, process in which radiation energy is imparted to material. In internal dosimetry, movement of material to blood regardless of mechanism.

absorption type

Categories for materials according to their rate of absorption from the respiratory tract to the blood, which replaced the earlier inhalation clearance classes. Defined by the International Commission on Radiological Protection, the absorption types are F: deposited materials that are readily absorbed into blood from the respiratory tract (fast solubilization), M: deposited materials that have intermediate rates of absorption into blood from the respiratory tract (moderate rate of solubilization), and S: deposited materials that are relatively in the respiratory tract (slow solubilization). Also called solubility type.

accuracy

The characteristics of an analysis or determination that ensures that both the bias and precision of the resultant quantity will remain within the specified limits.

alpha particle (α)

See *alpha radiation*.

alpha radiation

Positively charged particle emitted from the nuclei of some radioactive elements. An alpha particle consists of two neutrons and two protons (a helium nucleus) and has an electrostatic charge of +2.

ambient gamma radiation

Penetrating gamma radiation in the outside environment. Includes gamma radiation from natural cosmic and terrestrial sources as well as synthetic sources.

background radiation

Radiation from cosmic sources, naturally occurring radioactive materials including naturally occurring radon, and global fallout from the testing of nuclear explosives. Background radiation does not include radiation from source, byproduct, or Special Nuclear Materials regulated by the U.S. Nuclear Regulatory Commission. The average individual exposure from background radiation is about 360 millirem per year.

beta particle (β)

See *beta radiation*.

beta radiation

Charged particle emitted from some radioactive elements with a mass equal to 1/1,837 that of a proton. A negatively charged beta particle is identical to an electron. A positively charged beta particle is a positron.

breathing rate

Amount of air a person breathes in a specified time. In relation to health physics for workers, rates typically vary from light (1.2 cubic meters per hour) to heavy (1.7 cubic meters per hour) as defined by the International Commission on Radiological Protection.

buildup

Increase in flux or dose due to scattering in the medium.

confidence level
The interval about an estimate of a stated quantity within which the value of the quantity is expected to be with a specified probability. See *uncertainty*.

contamination
Radioactive material in an undesired location including air, soil, buildings, animals, and persons.

curie (Ci)
Traditional unit of radioactivity equal to 37 billion (3.7×10^{10}) becquerels, which is approximately equal to the activity of 1 gram of pure ^{226}Ra.

dose
In general, the specific amount of energy from ionizing radiation that is absorbed per unit of mass. Effective and equivalent doses are in units of rem or sievert; other types of dose are in units of roentgens, rads, reps, or grays.

dose conversion factor
Multiplier for conversion of potential dose to the personal dose equivalent to the organ of interest (e.g., liver or colon).

dose equivalent (*H*)
In units of rem or sievert, product of absorbed dose in tissue multiplied by a weighting factor and sometimes by other modifying factors to account for the potential for a biological effect from the absorbed dose. See *dose*.

dosimeter
Device that measures the quantity of received radiation, usually a holder with radiation-absorbing filters and radiation-sensitive inserts packaged to provide a record of absorbed dose received by an individual. See *film dosimeter* and *thermoluminescent dosimeter*.

dosimeter holder
Plastic holder for a dosimeter card, which typically includes one or more metallic filters that modify the response of the phosphor to radiation.

dosimetry
Measurement and calculation of internal and external radiation doses.

element
One of the known chemical substances in which the atoms have the same number of protons. Elements cannot be broken down further without changing their chemical properties.

environmental occupational dose
Dose received from radiation site-related activities (i.e., above normal background levels) while on a site, which is often recorded by monitoring stations in specific areas or along the boundaries of facilities.

error
Difference between the correct, true, or conventionally accepted value and the measured or estimated value. Sometimes used to mean estimated uncertainty. See *accuracy* and *uncertainty*.

exchange period (frequency)
Period (weekly, biweekly, monthly, quarterly, etc.) for routine exchange of dosimeters.

exposure

(1) In general, the act of being exposed to ionizing radiation. (2) Measure of the ionization produced by X- and gamma-ray photons in air in units of roentgens.

external dose

Dose received from radiation emitted by sources outside the body.

favorable to claimants

In relation to dose reconstruction for probability of causation analysis, having the property of ensuring that there is no underestimation of potential dose, which often means the assumption of a value that indicates a higher dose than is likely to have actually occurred in the absence of more accurate information. See *probability of causation*.

film

In the context of external dosimetry, radiation-sensitive photographic film in a light-tight wrapping. See *film dosimeter*.

film dosimeter

Package of film for measurement of ionizing radiation exposure for personnel monitoring purposes. A film dosimeter can contain two or three films of different sensitivities, and it can contain one or more filters that shield parts of the film from certain types of radiation. When developed, the film has an image caused by radiation measurable with an optical densitometer. Also called film badge.

filter

Material used in a dosimeter to adjust radiation response to provide an improved tissue equivalent or dose response.

gamma radiation

Electromagnetic radiation (photons) of short wavelength and high energy (10 kiloelectron-volts to 9 megaelectron-volts) that originates in atomic nuclei and accompanies many nuclear reactions (e.g., fission, radioactive decay, and neutron capture). Gamma photons are identical to X-ray photons of high energy; the difference is that X-rays do not originate in the nucleus.

gamma ray, particle, or photon (γ)

See *gamma radiation*.

geometric standard deviation (GSD)

In probability theory and statistics, the geometric standard deviation describes the spread of a set of numbers whose preferred average is the geometric mean.

Gravel Gertie

At NTS, a round room with 2-foot-thick concrete walls and a staging area with two staging bays connected to the round room. Tests were conducted in the Gravel Gertie with high explosives and uranium devices to measure fallout.

high explosive

Chemical compound or mechanical mixture that, when subjected to heat, impact, friction, shock, or other initiation stimulus, undergoes a rapid chemical change resulting in large volumes of highly heated gases that exert pressure in the surrounding medium. See *insensitive high explosive*.

holder

See *dosimeter holder*.

ingestion
Process of taking a substance into the body through the mouth.

intake
Radioactive material taken into the body by inhalation, absorption through the skin, injection, ingestion, or through wounds.

isotope
One of two or more atoms of a particular element that have the same number of protons (atomic number) but different numbers of neutrons in their nuclei (e.g., ^{234}U, ^{235}U, and ^{238}U). Isotopes have very nearly the same chemical properties. See *element*.

kiloelectron-volt (keV)
Unit of particle energy equal to 1,000 (1×10^3) electron-volts.

megaelectron-volt (MeV)
Unit of particle energy equal to 1 million (1×10^6) electron-volts.

monitor
Device that detects and usually records the presence of radioactivity or radioactive substances.

monitoring
Periodic or continuous determination of the presence or amount of ionizing radiation or radioactive contamination in air, surface water, groundwater, soil, sediment, equipment surfaces, or personnel (for example, bioassay or alpha scans). In relation to personnel, monitoring includes internal and external dosimetry including interpretation of the measurements.

neutron
Basic nucleic particle that is electrically neutral with mass slightly greater than that of a proton. There are neutrons in the nuclei of every atom heavier than normal hydrogen.

neutron radiation
Radiation that consists of free neutrons unattached to other subatomic particles emitted from a decaying radionuclide. Neutron radiation can cause further fission in fissionable material such as the chain reactions in nuclear reactors, and nonradioactive nuclides can become radioactive by absorbing free neutrons. See *neutron*.

nuclide
Stable or unstable isotope of any element. Nuclide relates to the atomic mass, which is the sum of the number of protons and neutrons in the nucleus of an atom. A radionuclide is an unstable nuclide.

occupational dose
Internal and external ionizing radiation dose from exposure during employment. Occupational dose does not include that from background radiation or medical diagnostics, research, or treatment, but does include dose from occupationally required radiographic examinations that were part of medical screening.

photon
Quantum of electromagnetic energy generally regarded as a discrete particle having zero rest mass, no electric charge, and an indefinitely long lifetime. The entire range of electromagnetic radiation that extends in frequency from 10^{23} cycles per second (hertz) to 0 hertz.

photon radiation
Electromagnetic radiation that consists of quanta of energy (photons) from radiofrequency waves to gamma rays.

probability of causation (POC)
For purposes of dose reconstruction for the Energy Employees Occupational Illness Compensation Program Act, the percent likelihood, at the 99th percentile, that a worker incurred a particular cancer from occupational exposure to radiation.

progeny
Nuclides that result from decay of other nuclides. Also called decay products and formerly called daughter products.

radiation
Subatomic particles and electromagnetic rays (photons) with kinetic energy that interact with matter through various mechanisms that involve energy transfer. See *ionizing radiation*.

radioactive
Of, caused by, or exhibiting radioactivity.

radioactivity
Property possessed by some elements (e.g., uranium) or isotopes (e.g., ^{14}C) of spontaneously emitting energetic particles (electrons or alpha particles) by the disintegration of their atomic nuclei. See *radionuclide*.

radionuclide
Radioactive nuclide. See *radioactive* and *nuclide*.

rem
Traditional unit of radiation dose equivalent that indicates the biological damage caused by radiation equivalent to that caused by 1 rad of high-penetration X-rays multiplied by a quality factor. The sievert is the International System unit; 1 rem equals 0.01 sievert. The word derives from roentgen equivalent in man; rem is also the plural.

roentgen
Unit of photon (gamma or X-ray) exposure for which the resultant ionization liberates a positive or negative charge equal to 2.58×10^{-4} coulombs per kilogram (or 1 electrostatic unit of electricity per cubic centimeter) of dry air at 0°C and standard atmospheric pressure. An exposure of 1 R is approximately equivalent to an absorbed dose of 1 rad in soft tissue for higher energy photons (generally greater than 100 kiloelectron-volts).

scattering
Change in direction of radiation by refraction or reflection, often accompanied by a decrease in radiation due to absorption by the refracting or reflecting material.

scintillation counter or detector
Combination of phosphor, a photomultiplier tube, and the associated electronic circuits for measuring the light emissions produced in the phosphor by ionizing radiation.

seep
At NTS, uncontrolled slow release of radioactive material with little or no energy. Seeps are not visible and can be detected only by measuring for radiation.

shielding

Material or obstruction that absorbs ionizing radiation and tends to protect personnel or materials from its effects.

sievert (Sv)

International System unit for dose equivalent, which indicates the biological damage caused by radiation. The unit is the radiation value in gray (equal to 1 joule per kilogram) multiplied by a weighting factor for the type of radiation and a weighting factor for the tissue; 1 Sv equals 100 rem.

thermoluminescent dosimeter (TLD)

Device for measuring radiation dose that consists of a holder containing solid chips of material that, when heated, release the stored energy as light. The measurement of this light provides a measurement of absorbed dose.

thermoluminescent dosimeter chip

Small block or crystal of lithium fluoride in a thermoluminescent dosimeter. A TLD-600 dosimeter contains a chip made from more than 95% ^{6}Li for neutron radiation detection, and a TLD-700 dosimeter contains a chip made from more than 99.9% ^{7}Li for photon and beta radiation detection. Also called crystals.

transuranic (TRU) elements

Elements with atomic numbers above 92 (uranium). Examples include plutonium and americium.

uncertainty

Standard deviation of the mean of a set of measurements. The standard error reduces to the standard deviation of the measurement when there is only one determination. See *accuracy*, *confidence level*, and *error*. Also called standard error.

unmonitored dose

Potential unrecorded dose that could have resulted because a worker was not monitored.

venting

At NTS, prompt, massive, and uncontrolled releases of radioactive material characterized as active releases under pressure, such as when radioactive material is driven out of the ground by steam or gas.

weighting factor

The ratio of the stochastic risk arising from tissue *T* to the total risk when the whole body is irradiated uniformly.

whole-body (WB) dose

Dose to the entire body excluding the contents of the gastrointestinal tract, urinary bladder, and gall bladder and commonly defined as the absorbed dose at a tissue depth of 10 millimeters (1,000 milligrams per square centimeter). Also called penetrating dose. See *dose*.

working level (WL)

Unit of concentration in air of the short-lived decay products of ^{222}Rn (^{218}Po, ^{214}Pb, ^{214}Bi, and ^{214}Po) and ^{220}Rn (^{216}Po, ^{212}Pb, ^{212}Bi, ^{212}Po) defined as any combination of the short-lived radioactive progeny of radon or thoron in 1 liter of air, without regard to the degree of equilibrium, that results in the ultimate emission of 130,000 MeV of alpha energy; 1 WL equals 2.083×10^{-5} joules per cubic meter.

working level month (WLM)

Unit of exposure to radon progeny defined as exposure for 1 working month (170 working hours) to a potential alpha energy concentration from of 1 WL; 1 WLM equals 1 WL times 170 hours, which is 0.00354 joule-hours per cubic meter. See *working level*.

X-ray

Electromagnetic radiation (photons) produced by bombardment of atoms by accelerated particles. X-rays are produced by various mechanisms including bremsstrahlung and electron shell transitions within atoms (characteristic X-rays). Once formed, there is no difference between X-rays and gamma rays, but gamma photons originate inside the nucleus of an atom.

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A.1 Introduction

With the cessation of atmospheric testing at NTS in 1962, the greatest potential for environmental intakes of radioactive material results from the inhalation of radioactive particles that were resuspended from NTS soils into the atmosphere and from ingestion of soils contaminated by atmospheric nuclear weapons tests, reactor tests, and safety tests. The potential inhalation intakes can be estimated from air sampling data in the NTS annual environmental reports (see References section) coupled with extensive soil contamination data from between 1983 and 1991 (McArthur and Kordas 1983, 1985; McArthur and Mead 1987, 1988, 1989; McArthur 1991). Because the air monitoring data were limited to gross alpha and beta measurements, tritium, and isotopes of plutonium (e.g., ^{238}Pu, ^{239}Pu, and ^{240}Pu), inhalation intakes of other relatively long-lived radionuclides that have been identified in NTS soils (e.g., ^{241}Am, ^{60}Co, ^{137}Cs, ^{90}Sr, ^{152}Eu, ^{154}Eu, and ^{155}Eu) are scaled to those of plutonium based on their relative abundance in NTS soils. Ingestion intakes can be estimated by assuming consumption of contaminated NTS soils. To ensure that inhalation and ingestion intakes are not underestimated, the relative abundances of long-lived radionuclides in NTS soils that were determined from the 1991 soil contamination data (McArthur 1991) were decay-corrected to 1963. In addition, to ensure that intakes and resultant doses are not underestimated, correction factors are developed to account for potential exposures to short-lived fission and activation products based on test-specific data from Hicks (1981a,b,c,d). In addition, a correction factor is developed for inhalation intakes which accounts for the phenomenon of early resuspension (Anspaugh et al. 2002).

This approach has been developed to address the unique character of NTS, which is a large outdoor testing facility where potential exposure to radioactive materials was primarily based on residual contamination from atmospheric testing. For most employees of the primary contractor, occupational

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internal doses can be equated to the doses they potentially received from the resuspension of residual radioactivity from the atmospheric testing (i.e., environmental internal dose). For workers who were primarily indoors (cafeterias, administrative facilities, and maintenance shops) this dose is likely to represent an overestimate of internal exposure. The environmental internal dose is the occupational internal dose, regardless of the recorded external dose. If an internal exposure was suspected, bioassay was performed. Managing radioactive material in the form of devices was episodic and limited to a few workers (e.g., radiation safety and industrial hygiene personnel, miners, and experimenters). These workers are identified on the rosters that were published before the event, and these workers are likely to have bioassay results in the DOE records.

The following paragraphs discuss:

- Air sampling data from the NTS annual environmental reports from 1971 through 2001 and the calculated annual inhalation ^{239}Pu intakes.
- NTS soil contamination data that were gathered and reduced in the late 1980s (McArthur 1991); soil contamination data that are decay-corrected to 1963.
- Methods used to develop inhalation intake scaling factors for radionuclides identified in NTS soils in comparison with intakes of plutonium that were calculated from air sampling data.
- The method used to account for higher air concentrations due to resuspension for 1963 (the first year after suspension of atmospheric testing).
- The methods used to develop the correction factors that were used to account for organ dose from inhalation of short-lived radionuclides that no longer persist in the NTS environment in measurable quantities.
- The method used to calculate ingestion intakes from oral consumption of contaminated NTS soils; the method used to account for ingestion dose from short-lived fission and activation products.
- Step-by-step instructions to the dose reconstructor for application of ambient environmental inhalation and ingestion intakes.

Attachment B provides a discussion that includes the results of 30 years of inhalation and ingestion intakes.

A.2 Air Sampling Data and Intake Estimates

Routine isotopic atmospheric measurements of plutonium at NTS began in 1971 with samplers in 15 locations across the site. Six additional sampling stations were added in 1978. Equipment at fixed locations continuously sampled the ambient air to monitor for radioactive materials. These locations were chosen to provide representative samples from the populated areas on the site and to monitor resuspension of low-fired plutonium spread by safety experiments before 1960 in Areas 2, 3, 4, 7, 9, and 10. Access, worker population, geographical coverage, presence of radioactivity, and availability of electric power were considerations in the site selection for air samplers (Black and Townsend 1997).

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In 1988, efforts to monitor radioactive air emissions at NTS were increased as a result of the requirements of DOE Order 5400.1 (DOE 1990). Known and potential effluent sources throughout NTS were assessed for their potential to contribute to public dose (Black and Townsend 1997).

The ^{239}Pu concentrations in Table A-1 for 1989 through 2001 represent the average of the maximum concentrations for a given area in a given year. If maximum values were not provided (i.e., 1971 through 1988), the average of the average concentrations was reported. In addition to the annual area concentrations, Table A-1 lists NTS site average and maximum concentrations, which represent the arithmetic averages of the concentration of all the areas and the maximum of the area maximum or area averages of all the areas, respectively. Potential intakes associated with these concentrations can be calculated under the assumption that an unmonitored worker was occupationally exposed for 2,000 hr/yr with a breathing rate of 2,400 m^3/yr. Table A-2 lists these calculated intakes.

Table A-1. Atmospheric concentrations of ^{239}Pu for sampled areas (pCi/m^3).[a]

Year	Area 1	2	3	4	5	6	7	9	10	11	12
1971	3.7E-04	6.6E-04	1.7E-04		1.2E-04	1.7E-04		7.2E-04	2.1E-04	9.1E-05	1.6E-04
1972	1.5E-04	2.2E-04	3.7E-04		1.4E-04	1.3E-03		4.3E-03	2.9E-04	2.4E-04	7.9E-04
1973	8.7E-05	2.2E-04	2.1E-04		8.7E-05	1.2E-04		8.6E-04	4.4E-05	2.4E-04	4.3E-05
1974	8.0E-05	6.8E-05	1.3E-04		5.0E-05	7.9E-05		2.1E-04	5.7E-05	8.0E-05	5.8E-05
1975	2.9E-05	3.8E-05	1.5E-04		4.3E-05	6.3E-05		1.7E-04	5.7E-05	4.9E-05	3.3E-05
1976	2.9E-04	1.7E-04	1.0E-03		1.5E-04	4.4E-04		3.2E-03	2.1E-04	2.4E-04	2.8E-04
1977	2.7E-05	6.5E-05	2.6E-04		3.4E-05	5.5E-05		2.5E-04	4.6E-05	1.2E-04	2.1E-05
1978	4.7E-05	1.3E-04	2.8E-04		6.1E-05	9.9E-05		5.4E-04	2.9E-04	1.1E-04	6.2E-05
1979	7.4E-05	4.7E-04	1.7E-06		2.1E-05	4.0E-05	4.9E-05	5.2E-04	2.9E-04	3.7E-05	2.4E-05
1980	2.3E-05	2.6E-04	1.1E-06		2.9E-05	4.8E-05	4.5E-05	4.5E-04		4.4E-05	1.9E-05
1981	2.6E-05	6.4E-05	1.4E-04		2.5E-05	2.9E-05	3.3E-05	3.2E-04		4.6E-05	2.0E-05
1982	6.1E-05	4.0E-05	6.2E-05		2.0E-05	3.5E-05	7.4E-05	2.2E-04		5.3E-05	2.7E-05
1983	3.1E-05	6.5E-05	9.2E-05		2.3E-05	3.2E-05	3.3E-05	2.1E-04		1.8E-04	2.3E-05
1984	1.9E-05	1.0E-04	2.3E-04		3.8E-05	2.0E-05	3.8E-05	1.0E-03		5.6E-05	1.7E-05
1985	1.5E-05	5.8E-05	2.1E-04		3.1E-05	2.7E-05	2.7E-05	1.3E-03		5.2E-05	2.3E-05
1986	1.5E-04	6.2E-05	5.5E-04		2.8E-05	1.9E-04	4.8E-05	2.8E-04		2.5E-05	3.2E-05
1987	5.5E-05	2.3E-05	2.8E-03		1.7E-05	2.5E-05	1.6E-05	1.1E-04		3.2E-05	1.6E-05
1988	9.2E-05	2.8E-05	2.1E-04		2.8E-05	2.2E-05	1.5E-05	5.1E-05		1.2E-05	1.6E-05
1989	8.6E-04	5.8E-05	5.4E-04		5.0E-05	8.0E-05	2.7E-04	3.5E-04		5.4E-04	6.8E-04
1990	9.9E-05	3.2E-05	3.5E-04		3.1E-05	3.7E-05	4.1E-05	5.0E-04	4.3E-05	1.1E-04	1.0E-05
1991	3.1E-03	2.1E-05	3.8E-04		3.5E-05	7.2E-05	3.2E-05	3.5E-04	5.4E-07	2.2E-04	1.4E-05
1992	1.5E-04	1.1E-04	2.3E-03		8.0E-05	7.3E-05	3.9E-04	8.8E-04	2.4E-04	8.8E-05	3.8E-05
1993	2.0E-04	4.9E-05	4.0E-04		8.0E-05	9.0E-05	7.8E-05	2.7E-03	1.7E-04	4.2E-04	8.4E-04
1994											
1995	4.1E-05	1.1E-05	2.4E-04		1.5E-05	5.0E-05	1.2E-05	3.9E-04	3.8E-05	4.0E-05	1.1E-05
1996	7.5E-04		3.0E-04		1.6E-05	4.6E-05	4.5E-04	6.1E-04	5.8E-05	1.9E-05	3.7E-06
1997		1.1E-05							2.6E-05	1.1E-05	1.5E-06
1998	4.7E-04		6.7E-05		9.5E-05	1.1E-05	2.7E-05	7.4E-04		9.5E-06	
1999				5.9E-05							
2000	8.5E-05		2.7E-04	5.9E-05	2.1E-05	4.1E-04	3.4E-05	2.8E-03			
2001	4.5E-04	1.2E-04	1.2E-04	1.9E-04	6.8E-05	9.6E-05	1.7E-05	5.0E-04	8.5E-06		
Year	Area 15	16	18	19	20	23	25	27	28	Site average[b]	Site maximum[c]
1971		1.9E-04	8.0E-05	2.3E-04		9.4E-05		8.4E-05	3.9E-05	2.26E-04	7.21E-04
1972		1.7E-04	1.5E-04	2.7E-03		3.5E-04		1.7E-04	5.1E-05	7.63E-04	4.29E-03
1973		4.4E-05	3.8E-05	7.2E-05		5.6E-05		3.7E-05	3.3E-05	1.46E-04	8.59E-04
1974		6.0E-05		6.6E-05		8.5E-05	8.6E-05	7.2E-05	1.2E-04	8.64E-05	2.11E-04
1975		4.3E-05		3.4E-05		4.4E-05	5.7E-05	6.8E-05	7.0E-05	6.27E-05	1.68E-04
1976		1.6E-04		9.9E-05		5.7E-04	2.0E-04	1.4E-04	6.3E-05	4.82E-04	3.18E-03
1977		2.6E-05		2.3E-05		2.7E-05	3.3E-05	3.1E-05	2.3E-05	6.93E-05	2.55E-04
1978		5.1E-05		6.2E-05		5.8E-05	7.0E-05	9.4E-05	3.9E-05	1.33E-04	5.42E-04
1979	5.2E-05	2.1E-05		2.0E-05		2.2E-05	1.8E-05	1.6E-05	2.3E-05	9.98E-05	5.24E-04

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	Area										
Year	**1**	**2**	**3**	**4**	**5**	**6**	**7**	**9**	**10**	**11**	**12**
1980	6.4E-05			9.5E-05		3.5E-05	3.1E-05	2.1E-05	1.6E-05	7.88E-05	4.52E-04
1981	9.3E-05	2.6E-05		2.3E-05		2.9E-05	2.4E-05	2.5E-05	1.6E-05	5.88E-05	3.22E-04
1982	4.2E-05	1.7E-05		2.2E-05		2.1E-05	1.7E-05			5.04E-05	2.15E-04
1983	2.2E-05	1.5E-05		3.0E-05		2.8E-05	3.0E-05			5.85E-05	2.14E-04
1984	3.2E-05	2.8E-05		2.5E-05		2.9E-05	3.2E-05	4.8E-05		1.15E-04	1.02E-03
1985	4.0E-05	2.1E-05		3.5E-05		2.2E-05	2.6E-05	3.1E-05		1.30E-04	1.33E-03
1986	3.9E-05	2.8E-05		2.2E-05		6.6E-05	2.1E-05	2.3E-05		1.03E-04	5.45E-04
1987	1.8E-05	1.4E-05		1.7E-05		1.6E-05	2.3E-05			2.13E-04	2.81E-03
1988	2.4E-05			1.4E-05		3.5E-05	1.5E-05	1.5E-05		4.15E-05	2.14E-04
1989	2.3E-05	1.1E-05		9.0E-06		1.2E-04	9.5E-06	4.8E-06		2.40E-04	8.55E-04
1990	8.2E-05	5.8E-06		3.9E-06		8.5E-05	1.1E-05	3.5E-05		9.17E-05	4.98E-04
1991	1.2E-04	2.9E-05		8.8E-06		1.3E-05	2.6E-05	7.7E-06		2.77E-04	3.10E-03
1992	1.1E-03	1.9E-05		8.9E-06		1.3E-05	1.8E-05	1.6E-04		3.52E-04	2.26E-03
1993	3.3E-04	1.3E-05		7.7E-06		6.3E-06	3.4E-06	1.1E-05		3.37E-04	2.70E-03
1994											
1995	1.6E-04	1.4E-05		1.0E-06		7.4E-06	4.5E-06			6.92E-05	3.90E-04
1996	2.4E-04	1.8E-06	3.5E-06		4.5E-06	6.3E-06	4.2E-06	2.6E-06		1.57E-04	7.50E-04

	Area									Site	Site
Year	**15**	**16**	**18**	**19**	**20**	**23**	**25**	**27**	**28**	**average[b]**	**maximum[c]**
1997	2.9E-05	1.9E-07	2.0E-05		2.0E-05	6.0E-07	1.0E-06	1.6E-06		1.11E-05	2.90E-05
1998	6.3E-05		3.1E-05		3.3E-05	1.2E-05	6.3E-06			1.30E-04	7.35E-04
1999	1.1E-05		1.1E-05		2.0E-05		6.2E-06			2.14E-05	5.90E-05
2000	2.6E-04		1.0E-05		6.6E-06		1.1E-05			3.63E-04	2.83E-03
2001	1.4E-05	9.6E-06	1.7E-05		7.6E-06	2.0E-06	4.4E-06			1.02E-04	5.04E-04

a. Source: NTS environmental reports in References section.
b. Values represent the arithmetic average of the area average concentrations for 1971 through 1988 and the arithmetic average of the area maximum concentrations for 1989 through 2001.
c. Values represent the maximum of the average area concentrations for 1971 through 1988 and the maximum of the maximum area concentrations for 1989 through 2001.

Table A-2. Annual inhalation intakes from ^{239}Pu for sampled areas (Bq).

	Area										
Year	**1**	**2**	**3**	**4**	**5**	**6**	**7**	**9**	**10**	**11**	**12**
1971	0.0329	0.0587	0.0147		0.0105	0.0153		0.0640	0.0185	0.0081	0.0145
1972	0.0131	0.0199	0.0327		0.0124	0.1190		0.3810	0.0258	0.0216	0.0698
1973	0.0077	0.0198	0.0183		0.0078	0.0107		0.0763	0.0039	0.0216	0.0038
1974	0.0071	0.0060	0.0112		0.0044	0.0070		0.0187	0.0051	0.0071	0.0052
1975	0.0026	0.0034	0.0130		0.0038	0.0056		0.0149	0.0051	0.0044	0.0029
1976	0.0255	0.0153	0.0924		0.0133	0.0392		0.2824	0.0189	0.0210	0.0248
1977	0.0024	0.0058	0.0226		0.0030	0.0049		0.0225	0.0041	0.0107	0.0019
1978	0.0042	0.0118	0.0250		0.0054	0.0088		0.0481	0.0258	0.0095	0.0055
1979	0.0066	0.0413	0.0001		0.0018	0.0035	0.0044	0.0465	0.0258	0.0033	0.0021
1980	0.0020	0.0230	0.0001		0.0026	0.0043	0.0040	0.0401		0.0039	0.0017
1981	0.0023	0.0056	0.0124		0.0022	0.0026	0.0029	0.0286		0.0041	0.0018
1982	0.0054	0.0036	0.0055		0.0018	0.0031	0.0066	0.0191		0.0047	0.0024
1983	0.0028	0.0057	0.0082		0.0020	0.0029	0.0029	0.0190		0.0162	0.0020
1984	0.0017	0.0088	0.0202		0.0034	0.0018	0.0034	0.0906		0.0050	0.0015
1985	0.0013	0.0051	0.0189		0.0027	0.0024	0.0024	0.1181		0.0046	0.0020
1986	0.0131	0.0055	0.0484		0.0025	0.0165	0.0043	0.0249		0.0022	0.0028
1987	0.0048	0.0020	0.2495		0.0015	0.0022	0.0014	0.0098		0.0028	0.0014
1988	0.0082	0.0025	0.0190		0.0025	0.0020	0.0013	0.0045		0.0011	0.0014
1989	0.0759	0.0052	0.0478		0.0045	0.0071	0.0240	0.0311		0.0480	0.0604
1990	0.0088	0.0029	0.0308		0.0027	0.0033	0.0036	0.0442	0.0038	0.0095	0.0009
1991	0.2753	0.0018	0.0340		0.0031	0.0064	0.0029	0.0311	0.0000	0.0199	0.0013
1992	0.0133	0.0098	0.2007		0.0071	0.0065	0.0346	0.0781	0.0213	0.0078	0.0034
1993	0.0180	0.0044	0.0352		0.0071	0.0079	0.0069	0.2398	0.0151	0.0373	0.0746
1994											
1995	0.0036	0.0010	0.0216		0.0014	0.0045	0.0011	0.0346	0.0033	0.0036	0.0010
1996	0.0666	0.0000	0.0268		0.0015	0.0041	0.0400	0.0542	0.0051	0.0017	0.0003

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1997		0.0010							0.0023	0.0010	0.0001
1998	0.0416		0.0060		0.0084	0.0010	0.0024	0.0653		0.0008	
1999											
2000	0.0075		0.0239	0.0052	0.0018	0.0366	0.0030	0.2509			
2001	0.0404	0.0011	0.0103	0.0166	0.0061	0.0086	0.0015	0.0447	0.0008		

	Area									Site	Site
Year	**15**	**16**	**18**	**19**	**20**	**23**	**25**	**27**	**28**	**average**	**maximum**
1971		0.0171	0.0071	0.0201		0.0083		0.0075	0.0035	0.0200	0.0640
1972		0.0150	0.0129	0.2424		0.0309		0.0147	0.0045	0.0677	0.3810
1973		0.0039	0.0034	0.0063		0.0050		0.0033	0.0029	0.0130	0.0763
1974		0.0053		0.0059		0.0075	0.0076	0.0064	0.0104	0.0077	0.0187
1975		0.0038		0.0030		0.0039	0.0051	0.0060	0.0062	0.0056	0.0149

	Area									Site	Site
Year	**15**	**16**	**18**	**19**	**20**	**23**	**25**	**27**	**28**	**average**	**maximum**
1976		0.0139		0.0088		0.0504	0.0178	0.0122	0.0056	0.0428	0.2824
1977		0.0023		0.0020		0.0024	0.0029	0.0028	0.0020	0.0062	0.0226
1978		0.0045		0.0055		0.0052	0.0062	0.0083	0.0035	0.0118	0.0481
1979	0.0046	0.0019		0.0017		0.0020	0.0016	0.0014	0.0020	0.0089	0.0465
1980	0.0057			0.0085		0.0031	0.0027	0.0019	0.0014	0.0070	0.0401
1981	0.0083	0.0023		0.0020		0.0025	0.0022	0.0022	0.0014	0.0052	0.0286
1982	0.0037	0.0015		0.0019		0.0019	0.0015			0.0045	0.0191
1983	0.0020	0.0013		0.0026		0.0025	0.0027			0.0052	0.0190
1984	0.0028	0.0025		0.0022		0.0025	0.0028	0.0043		0.0102	0.0906
1985	0.0036	0.0019		0.0031		0.0020	0.0023	0.0028		0.0115	0.1181
1986	0.0035	0.0025		0.0020		0.0058	0.0018	0.0020		0.0092	0.0484
1987	0.0016	0.0012		0.0015		0.0014	0.0020	0.0000		0.0189	0.2495
1988	0.0022			0.0012		0.0031	0.0013	0.0013		0.0037	0.0190
1989	0.0021	0.0010		0.0008		0.0104	0.0008	0.0004		0.0213	0.0759
1990	0.0073	0.0005		0.0003		0.0075	0.0010	0.0031		0.0081	0.0442
1991	0.0109	0.0026		0.0008		0.0012	0.0023	0.0007		0.0246	0.2753
1992	0.0977	0.0017		0.0008		0.0011	0.0016	0.0142		0.0312	0.2007
1993	0.0293	0.0012		0.0007		0.0006	0.0003	0.0010		0.0300	0.2398
1994										0.0000	0.0000
1995	0.0142	0.0012		0.0001		0.0007	0.0004			0.0061	0.0346
1996	0.0213	0.0002	0.0003	0.0000	0.0004	0.0006	0.0004	0.0002		0.0124	0.0666
1997	0.0026	0.0000	0.0018		0.0018	0.0001	0.0001	0.0001		0.0010	0.0026
1998	0.0056		0.0027		0.0029	0.0010	0.0006			0.0115	0.0653
1999	0.0010		0.0010		0.0018		0.0006			0.0011	0.0018
2000	0.0229		0.0009		0.0006		0.0010			0.0322	0.2509
2001	0.0012	0.0008	0.0015		0.0007	0.0002	0.0004			0.0090	0.0447

Some covered employees remained on the site continuously for weeks at a time. However, because most of the nonworking hours were spent indoors where ambient air particulate loadings would be much less than the outdoor loadings and because of the conservative assumptions that were used to estimate the values in Table A-2, adjustment of the tabular data is not required to ensure intakes are not underestimated for these individuals. In addition, employees who lived on the site during their work week would have been housed in Area 12 or Area 23 (Mercury). For most years the values in Table A-1 for these locations are less than the site average values.

It is assumed that plutonium could be any of absorption type S, Super S, or M, depending on which type delivers the maximum organ dose.

A.3 Soil Contamination and Decay Corrections

Extensive studies were performed in the 1980s to quantify residual contamination at NTS (McArthur and Kordas 1983, 1985; McArthur and Mead 1987, 1988, 1989; McArthur 1991). Table A-3 lists the

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results of these studies (McArthur 1991). Table A-4 lists the total areal depositions based on the inventory values in Table A-3 divided by the respective areal size. The results in Table A-4 are representative of areas of NTS that have been shown to contain measurable levels of contamination. These areas actually represent only about one-third of the total area within the boundaries of NTS.

Because the data in Table A-3 are representative of soil contamination in 1991, these values were decay-corrected to the beginning of 1963 – the first year after the cessation of atmospheric testing. Table A-5 lists the decay-corrected areal soil deposition.

Table A-3. Inventory of contaminated soil (Ci).

Area	Area (mi^2)	Am-241	Pu-238	Pu-239,240	Co-60	Cs 137	Sr-90	Eu-152	Eu-154	Eu-155
1	26.5	4.2	6.5	24	1.1	8.8	15	15	0.1	0.5
2	19.7	2.9	8.6	22	1.2	24	46	14		0.4
3	32.3	4.6	3.1	37	1	12	33	18	0.1	0.5
4	16	6.6	13	40	1.6	12	13	9.1		0.2
5	2.9	0.6	0.1	4.8	0.6	0.4	0.9	10	0.2	0
6	32.3	1.7	3.3	8.4	0.2	2.8	3.5			0
7	19.3	2.2	0.6	16	1	5.2	9.2	22	0.2	0.3
8	13.9	17	8	110	5.7	42	25	4.4		0.6
9	20	4.2	2.2	89	0.7	8.7	13	23	0.2	0.3
10	20	19	19	110	9.7	84	55	2.2	0.3	5
11	4	3.3	0.5	29	0	0.5	0.3			
12	39.6	5.7	8.5	39	1.2	20	17			
15	35.3	8	7.8	63	0.3	19	22			
16	14.3	0.7	1.5	3.7	0.1	2.9	3.7			
17	31.4	2.8	4.5	18	1.0	15	19			
18	27.3	19	5.6	100	0.7	10	17	1.1	0.1	0.8
19	148.3	21	32	140	1.1	36	31			
20	6.2	23	30	41	7.9	5.5	4.3	13	1.6	4.8
25	0.9					0.2	0.1	0.4		
26	0.2									
30	0.03	3.2	4.5	14	0.8	1.5	1.3	0.7	0.1	0.2

Source: McArthur (1991).

Table A-4. Radionuclide areal soil deposition (Bq/m^2).

Area	Am-241	Pu-238	Pu-239,240	Co-60	Cs 137	Sr-90	Eu-152	Eu-154	Eu-155
1	2.26E+03	3.50E+03	1.29E+04	5.93E+02	4.74E+03	8.09E+03	8.09E+03	5.39E+01	2.70E+02
2	2.10E+03	6.24E+03	1.60E+04	8.70E+02	1.74E+04	3.34E+04	1.02E+04	0.00E+00	2.90E+02
3	2.03E+03	1.37E+03	1.64E+04	4.42E+02	5.31E+03	1.46E+04	7.96E+03	4.42E+01	2.21E+02
4	5.89E+03	1.16E+04	3.57E+04	1.43E+03	1.07E+04	1.16E+04	8.13E+03	0.00E+00	1.79E+02
5	2.96E+03	4.93E+02	2.36E+04	2.96E+03	1.97E+03	4.43E+03	4.93E+04	9.85E+02	0.00E+00
6	7.52E+02	1.46E+03	3.72E+03	8.85E+01	1.24E+03	1.55E+03	0.00E+00	0.00E+00	0.00E+00
7	1.63E+03	4.44E+02	1.18E+04	7.40E+02	3.85E+03	6.81E+03	1.63E+04	1.48E+02	2.22E+02
8	1.75E+04	8.22E+03	1.13E+05	5.86E+03	4.32E+04	2.57E+04	4.52E+03	0.00E+00	6.17E+02
9	3.00E+03	1.57E+03	6.36E+04	5.00E+02	6.21E+03	9.29E+03	1.64E+04	1.43E+02	2.14E+02
10	1.36E+04	1.36E+04	7.86E+04	6.93E+03	6.00E+04	3.93E+04	1.57E+03	2.14E+02	3.57E+03
11	1.18E+04	1.79E+03	1.04E+05	0.00E+00	1.79E+03	1.07E+03	0.00E+00	0.00E+00	0.00E+00
12	2.06E+03	3.07E+03	1.41E+04	4.33E+02	7.22E+03	6.13E+03	0.00E+00	0.00E+00	0.00E+00
15	3.24E+03	3.16E+03	2.55E+04	1.21E+02	7.69E+03	8.90E+03	0.00E+00	0.00E+00	0.00E+00
16	6.99E+02	1.50E+03	3.70E+03	9.99E+01	2.90E+03	3.70E+03	0.00E+00	0.00E+00	0.00E+00
17	1.27E+03	2.05E+03	8.19E+03	4.55E+02	6.82E+03	8.64E+03	0.00E+00	0.00E+00	0.00E+00
18	9.94E+03	2.93E+03	5.23E+04	3.66E+02	5.23E+03	8.90E+03	5.76E+02	5.23E+01	4.19E+02
19	2.02E+03	3.08E+03	1.35E+04	1.06E+02	3.47E+03	2.99E+03	0.00E+00	0.00E+00	0.00E+00

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20	5.30E+04	6.91E+04	9.45E+04	1.82E+04	1.27E+04	9.91E+03	3.00E+04	3.69E+03	1.11E+04
25	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.17E+03	1.59E+03	6.35E+03	0.00E+00	0.00E+00
26	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
30	1.52E+05	2.14E+05	6.67E+05	3.81E+04	7.14E+04	6.19E+04	3.33E+04	4.76E+03	9.52E+03

Table A-5. Radionuclide areal soil deposition (Bq/m^2) decay, corrected to 1963.

Area	Am-241	Pu-238	Pu-239,240	Co-60	Cs 137	Sr-90	Eu-152	Eu-154	Eu-155
1	2.37E+03	4.37E+03	1.29E+04	2.35E+04	9.06E+03	1.57E+04	3.47E+04	4.89E+02	1.35E+04
2	2.20E+03	7.78E+03	1.60E+04	3.45E+04	3.32E+04	6.49E+04	4.35E+04	0.00E+00	1.45E+04
3	2.13E+03	1.71E+03		1.76E+04	1.01E+04	2.84E+04	3.41E+04	4.01E+02	1.11E+04
4	6.16E+03	1.45E+04	3.57E+04	5.67E+04	2.05E+04	2.26E+04	3.48E+04		8.93E+03
5	3.09E+03	6.15E+02	2.37E+04	1.17E+05	3.76E+03	8.63E+03	2.11E+05	8.94E+03	0.00E+00
6	7.86E+02	1.82E+03	3.72E+03	3.51E+03	2.36E+03	3.01E+03			
7	1.70E+03	5.54E+02	1.19E+04	2.94E+04	7.35E+03	1.33E+04	6.98E+04	1.34E+03	1.11E+04
8	1.83E+04	1.03E+04	1.13E+05	2.33E+05	8.24E+04	5.00E+04	1.94E+04		3.08E+04
9	3.14E+03	1.96E+03	6.36E+04	1.98E+04	1.19E+04	1.81E+04	7.04E+04	1.30E+03	1.07E+04
10	1.42E+04	1.69E+04	7.86E+04	2.75E+05	1.15E+05	7.65E+04	6.74E+03	1.94E+03	1.79E+05
11	1.23E+04	2.23E+03	1.04E+05	0.00E+00	3.41E+03	2.09E+03			
12	2.15E+03	3.83E+03	1.41E+04	1.72E+04	1.38E+04	1.19E+04			
15	3.39E+03	3.94E+03	2.55E+04	4.82E+03	1.47E+04	1.73E+04			
16	7.31E+02	1.87E+03	3.70E+03	3.97E+03	5.53E+03	7.20E+03			
17	1.33E+03	2.55E+03	8.20E+03	1.81E+04	1.30E+04	1.68E+04			
18	1.04E+04	3.66E+03	5.24E+04	1.45E+04	9.99E+03	1.73E+04	2.47E+03	4.75E+02	2.09E+04
19	2.12E+03	3.85E+03	1.35E+04	4.21E+03	6.62E+03	5.81E+03			
20	5.54E+04	8.62E+04	9.45E+04	7.23E+05	2.42E+04	1.93E+04	1.28E+05	3.34E+04	5.53E+05
25					6.06E+03	3.09E+03	2.72E+04		
26									
30	1.59E+05	2.67E+05	6.67E+05	1.51E+06	1.36E+05	1.21E+05	1.43E+05	4.32E+04	4.76E+05

A.4 Scaling Factors for Inhalation Intake Estimates

Because the air sampling program did not provide isotopic analyses for all identified radionuclides in NTS soils, scaling factors were developed to estimate potential intakes for these radionuclides based on their relative abundances in comparison with ^{239}Pu with soil contamination data (McArthur 1991) decay-corrected to 1963. These area-specific ratios are listed in Table A-6.

The scaling factors in Table A-6 were used in conjunction with the ^{239}Pu intakes in Table A-2 to determine potential environmental intakes of all radionuclides important to dose (Table A-6) that have been identified as persistent in NTS soils. For dose reconstruction, maximum intakes of these radionuclides were calculated by selecting the maximum annual intake of plutonium (i.e., 0.4131 Bq/yr derived for Area 9 in 1972) and multiplying this value by the maximum scaling factor for each of the radionuclides in Table A-6.

A.5 Correction for Resuspension for Early Times After Atmospheric Tests

Anspaugh et al. (2002) stated that, based on empirical observations, concentration of resuspended radionuclides in air have been noted to display a strong time dependence early after deposition and that this pathway might be important for reoccupation of contaminated property. Anspaugh et al. also stated that there has not been universal agreement that resuspension is an important pathway but that it is now generally accepted that there are a few instances in which the pathway could be the dominant one. Many observations have shown that the rate of resuspension decreases rapidly with time, and that for accident situations resuspension is only of importance (in comparison with the inhalation exposure from the initial cloud passage) over short periods. For this reason, Anspaugh et al. (2002) stated that the resuspension factor model has been widely used to predict the concentration of resuspended radionuclides early after initial deposition while the mass loading model (which uses measurements of dust loading in air and soil contamination data to predict air concentration of

Table A-6. Abundance of radionuclides in NTS soils relative to ^{239}Pu decay, corrected to 1963.

Area	Am-241	Pu-238	Pu-239, 240	Co-60	Cs 137	Sr-90	Eu-152	Eu-154	Eu-155
1	0.183	0.338	1.000	1.818	0.700	1.216	2.677	0.038	1.041
2	0.138	0.487	1.000	2.164	2.081	4.067	2.726		0.908
3	0.130	0.104	1.000	1.072	0.619	1.735	2.084	0.024	0.675
4	0.172	0.405	1.000	1.587	0.572	0.632	0.975		0.250
5	0.131	0.026	1.000	4.958	0.159	0.365	8.925	0.378	
6	0.212	0.490	1.000	0.944	0.636	0.810			
7	0.144	0.047	1.000	2.479	0.620	1.118	5.890	0.113	0.937
8	0.162	0.091	1.000	2.055	0.728	0.442	0.171		0.273
9	0.049	0.031	1.000	0.312	0.187	0.284	1.107	0.020	0.168
10	0.181	0.215	1.000	3.498	1.457	0.973	0.086	0.025	2.271
11	0.119	0.021	1.000		0.033	0.020			
12	0.153	0.272	1.000	1.220	0.978	0.848			
15	0.133	0.154	1.000	0.189	0.575	0.679			
16	0.198	0.505	1.000	1.072	1.495	1.945			
17	0.163	0.312	1.000	2.204	1.590	2.053			
18	0.199	0.070	1.000	0.278	0.191	0.331	0.047	0.009	0.400
19	0.157	0.285	1.000	0.312	0.491	0.431			
20	0.586	0.912	1.000	7.643	0.256	0.204	1.358	0.354	5.849
25									
26									
30	0.229	0.321	1.000	0.057	0.107	0.093	0.050	0.007	0.014
Maximum scaling factor	**0.586**	**0.912**	**1.000**	**7.64**	**2.08**	**4.07**	**8.93**	**0.378**	**5.85**
Scaled maximum intake, Bq/yr	**0.223**	**0.347**	**0.381**	**2.91**	**0.792**	**1.55**	**3.40**	**0.144**	**2.23**

radionuclides) has generally been preferred for times long after deposition. However, Anspaugh et al. also stated that it is always preferable to rely on actual measurements that are performed over long periods (such as those in this attachment).

Anspaugh et al. (2002) presented several resuspension models that have been proposed but concluded that they can be over- or under-predictive at various times after deposition in comparison with empirical observations. However, with expanded datasets from Hicks (1981e) and others in the 1980s, Anspaugh et al. proposed a resuspension model that more accurately describes the observed results over the entire timespan of the expanded dataset for NTS:

$$S_f = [10^{-5}e^{-0.07t} + 6\times10^{-9}e^{-0.003t} + 10^{-9}]\times10^{\pm1}\ \mathrm{m}^{-1} \qquad \text{(A-1)}$$

A depiction of Equation A-1 is provided in Figure A-1 from time *t* equal zero to 1,000 days after detonation. As Figure A-1 shows, the resuspension factor S_f ranges from about 10^{-5} early after deposition, falls rapidly during the first 100 or so days to a value of about 10^{-8}, and then approaches a value of 10^{-9} after a few years. The factor of 10 at the end of the equation is a statement of uncertainty in the model. If the mass loading approach is more predictive at times long after initial deposition and the resuspension proposed by Anspaugh et al. (2002) is predictive of the observed results over the expanded dataset (including those developed in the 1980s), the factor of 10^{-9} could be taken to be the resuspension factor that would be predictive of the mass loading process that is thought to be more important during the times when air monitoring data are available (i.e., 1971 through 2001; see Section A.2). When the Anspaugh et al. proposed resuspension model (i.e., Equation A-1) is integrated from 180 to 545 days and compared to the result of the integral of the constant 10^{-9} over the same period, a factor is developed that can be used to correct the intakes derived from air sampling data (i.e., times long after initial deposition) for the early resuspension phenomenon that has been observed at NTS. The period of integration was selected to begin at day 180 because the last atmospheric tests at NTS were in July 1962 and, therefore, about 180 days had passed before the beginning of 1963 (the starting date for reconstruction of internal dose at NTS in the absence of bioassay results for the employee). For the Anspaugh et al. model, the early resuspension correction factor has been determined to be 3.12. In a similar manner, correction factors for 1964 and 1965 were determined to be 1.72 and 1.24, respectively. Therefore, for dose reconstruction, the scaled maximum inhalation intakes in Table A-6 should be multiplied by these factors to account for early resuspension from 1963 through 1965. These resultant intakes are listed in Table A-7. If necessary, these intakes can be prorated for time less than a year for a best estimate if the worker was on the site for only a fraction of the year.

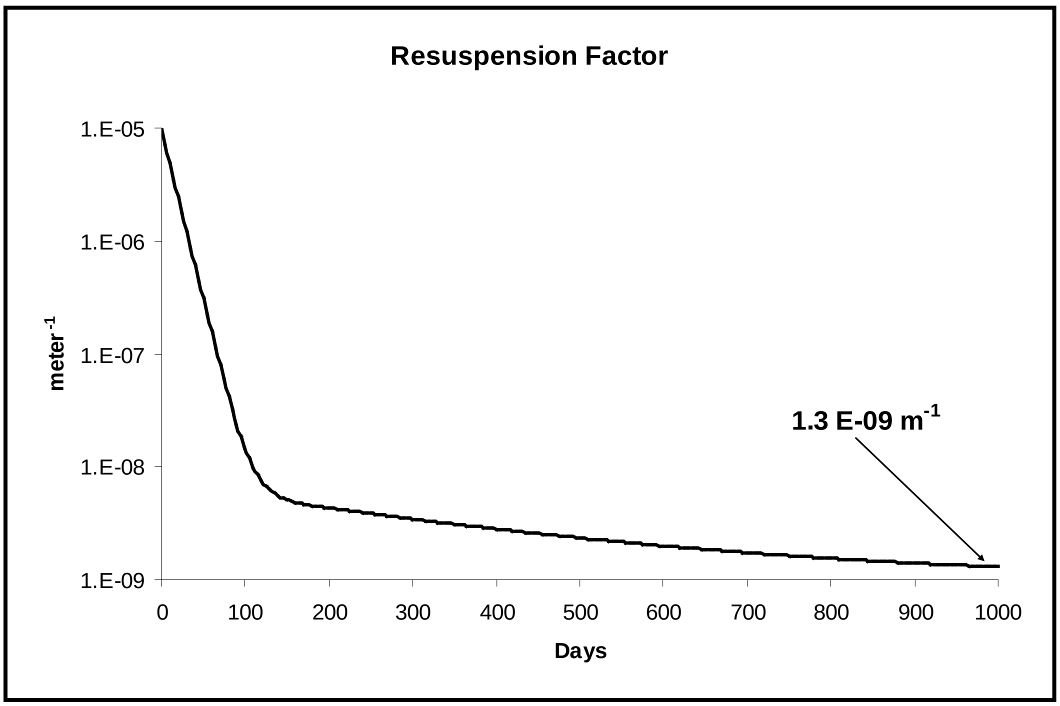

Figure A-1. Resuspension factor as a function of time after initial deposition.

Table A-7. Scaled inhalation intakes corrected for early resuspension in 1963 (Bq/yr).

Year of intake	Am-241	Pu-238	Pu-239, 240	Co-60	Cs 137	Sr-90	Eu-152	Eu-154	Eu-155
1963	0.70	1.08	1.19	9.08	2.47	4.84	10.61	0.45	6.96
1964	0.38	0.59	0.65	4.98	1.35	2.65	5.81	0.25	3.81
1965	0.28	0.43	0.47	3.61	0.98	1.92	4.22	0.18	2.77
All subsequent years	0.223	0.347	0.381	2.91	0.792	1.55	3.40	0.144	2.23

A.6 Corrections for Inhalation Dose from Short-lived Fission and Activation Products

During the early and mid-1960s, workers could have been exposed to fallout from atmospheric testing that contained short-lived fission and activation products (^{144}Ce, ^{106}Ru, etc.) that have not persisted in NTS soils in measurable quantities. For dose reconstruction, two methods are described to account for possible exposure to these radionuclides.

Potential inhalation dose from short-lived fission and activation products can be estimated using data developed by Hicks (1981e). Hicks provided estimates of ground deposition of 152 fission products and 25 neutron-induced nuclides from nuclear weapons tests. The data for these calculations were for the last atmospheric tests NTS conducted in 1962 as a function of time after detonation. Spreadsheets were developed to multiply the relative abundance of each of these radionuclides by its associated International Commission on Radiological Protection (ICRP) Publication 68 (ICRP 1995) inhalation organ dose conversion factor (Bunker 1999) to determine the relative importance of each of the 177 radionuclides to total organ dose. With the exception of ^{90}Sr, the inhalation dose conversion factors were for the absorption type (i.e., S, M, or F) that produced the largest dose to the organ of interest. For ^{90}Sr, only absorption type F was used. In addition, because ICRP Publication 68 organ dose factors represent the 50-year committed dose, the IMBA computer program was used to develop the annual organ-specific dose conversion factor for ^{241}Am, which was used in the calculations to better represent the relative importance of ^{241}Am, which delivers dose to the affected organ over long periods. Similarly, annual organ-specific dose conversion factors were developed for ^{90}Sr to provide a more accurate time-dependant correction factor.

The development of the short-lived fission and activation product correction factors based on the Hicks data must be adjusted for fractionation. Fractionation is a phenomenon due to chemical and physical separation of the radionuclides in the fireball in first few minutes after detonation. Within the first minute after detonation, the vaporized soil components condense with other refractory elements and begin to fall to the surface. The volatile elements (except krypton and xenon) and their progeny condense in 6 to 8 minutes and begin falling to the surface (Hicks 1981e). Because the Hicks data were developed to estimate offsite levels of fallout and resultant dose, fractionation effects were simulated in these data by the removal of a fraction of the refractory nuclides from the calculated abundances. In general, air drops were assumed to be unfractionated and offsite fallout from surface and cratering tests was assumed to have 0.4 of the refractory elements. For all other types of tests, offsite fallout was assumed to have 0.5 of the refractory elements present. Therefore, the refractory elements (beryllium, sodium, manganese, iron, cobalt, copper, yttrium, zirconium, niobium, barium, rare earths, thorium, uranium, neptunium, plutonium, americium, and curium) in the Hicks data must be adjusted to produce the best estimate of their enriched abundances in the onsite environment to which workers could have been exposed. Adjustment factors for each radionuclide were determined from data in Hicks (1984); this report provided relative abundances of radionuclides assuming no fraction, 50% fraction, and 90% fraction of refractory elements. From these data, ratios were developed for the 50% fractionation case (Table A-8). These ratios were used to deplete the refractory elements in the far-field (i.e., offsite) environment to estimate doses to offsite individuals. Therefore, to enrich the near-field (i.e., onsite) environment, the inverse of these ratios was applied to the Hicks SMALL BOY data (see below). These inverse ratios were applied twice because the Hicks SMALL BOY data were provided to estimate fallout in the offsite environment. The first application results in the data that represent no fractionation while the second application results in data that are enriched with refractory elements.

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In Table A-8, the radionuclides with ratios less than 1.0 represent the radionuclides that condense as refractory elements while the radionuclides with ratios greater than 1.0 are assumed to behave as volatile elements. Although strontium is a refractory element, for these calculation, the assumption is that it behaves like a volatile element because its fission precursors are noble gases (i.e., volatile elements).

Table A-8. Far-field refractory enrichment ratios.

Radio-nuclide	Ratio	Radio-nuclide	Ratio	Radio-nuclide	Ratio	Radio-nuclide	Ratio	Radio-nuclide	Ratio	Radio-nuclide	Ratio
Be-7	0.68	Br-82	1.36	Tc-101	1.36	Cd-118	1.36	I-132	1.36	Pr-145	0.68
Na-24	0.68	Se-83	1.36	Mo-102	1.36	In-118	1.36	Te-133m	1.36	Ce-146	0.68
Mn-54	0.68	Br-83	1.36	Tc-102m	1.36	Cd-119	1.36	Te-133	1.36	Pr-146	0.68
Fe-55	0.68	Kr-83m	1.37	Tc-102	1.36	In-119m	1.36	I-133	1.36	Pr-147	0.68
Fe-59	0.68	Br-84	1.36	Ru-103	1.36	In-119	1.36	Xe-133m	1.36	Nd-147	0.68
Cu-64	0.68	Kr85m	1.36	Rh-103m	1.36	Sn-121	1.36	Xe-133	1.36	Nd-149	0.68
Cu-67	0.68	Kr-87	1.36	Tc-104	1.36	Sn-123m	1.37	Te-134	1.36	Pm-149	0.68
W-181	1.36	Kr-88	1.36	Ru-105	1.36	Sn-123	1.36	I-134	1.36	Pm-150	0.68
W-185	1.35	Rb-88	1.36	Rh-105m	1.36	Sn-125	1.36	I-135	1.36	Nd-151	0.68
W-187	1.36	Rb-89	1.36	Rh-105	1.36	Sb-125	1.36	Xe-135m	1.36	Pm-151	0.68
W-188	1.36	Sr-89	1.36	Ru-106	1.36	Sb-126	1.36	Xe-135	1.36	Pm-152	0.68
U-237	0.68	Sr-90	1.36	Rh-106	1.36	Sn-127	1.36	Cs-136	1.36	Sm-153	0.68
U-239	0.68	Sr-91	1.19	Rh-107	1.36	Sb-127	1.36	Cs-137	1.36	Sm-155	0.68
U-240	0.68	Y-91m	1.19	Pd-107m	1.36	Te-127	1.36	Ba-137m	1.36	Eu-155	0.68
Np-239	0.68	Y-91	1.19	Pd-109	1.36	Sn-128	1.36	Xe-138	1.36	Sm-156	0.68
Np240m	0.69	Sr-92	0.68	Ag-109m	1.36	Sb-128m	1.37	Cs-138	1.36	Eu-156	0.68
Np-240	0.68	Y-92	0.68	Pd-111m	1.37	Sb-128	1.37	Cs-139	1.36	Eu-157	0.68
Am-241	0.68	Sr-93	0.68	Pd-111	1.36	Sn-129m	1.36	Ba-139	1.36	Eu-158	0.68
Cm-242	0.68	Y-93	0.68	Ag-111m	1.36	Sn-129	1.36	Ba-140	1.16	Eu-159	0.68
Ge-75	1.36	Y-94	0.68	Ag-111	1.36	Sb-129	1.36	La-140	1.16	Gd-159	0.68
Ge-77	1.36	Y-95	0.68	Pd-112	1.36	Te-129m	1.36	Ba-141	0.90	Tb-161	0.68
As-77	1.36	Zr-95	0.68	Ag-112	1.36	Te-129	1.36	La-141	0.90		
Se-77m	1.36	Nb-95	0.68	Ag-113	1.36	Sb-130m	1.36	Ce-141	0.90		
Ge-78	1.36	Zr-97	0.68	Ag-115	1.36	Sb-130	1.36	Ba-142	0.68		
As-78	1.36	Nb-97m	0.68	Cd-115m	1.36	I-130	1.36	La-142	0.68		
As-79	1.37	Nb-97	0.68	Cd-115	1.36	Sb-131	1.36	La-143	0.68		
Se-79m	1.36	Nb-98	0.68	In-115m	1.37	Te-131m	1.36	Ce-143	0.68		
Br-80	1.36	Mo-99	0.68	Cd-117	1.36	Te-131	1.36	Pr-143	0.68		
Se-81m	1.36	Tc-99m	0.68	In-117m	1.36	I-131	1.36	Ce-144	0.68		
Se-81	1.36	Mo-101	1.36	In-117	1.36	Te-132	1.36	Pr-144	0.68		

The fission and activation product correction factor was developed based on the relative contribution of ^{90}Sr to total organ dose. Strontium-90 was chosen because Hicks (1984) provided time-dependant abundances for this radionuclide and ^{90}Sr continues to persist in the NTS environment. Therefore, by determining the relative importance of organ dose from ^{90}Sr to the total dose from all 177 radionuclides, a multiplication factor was developed, by which the various organ doses from the ^{90}Sr intakes (Table A-7) can be multiplied, to account for short-lived fission and activation products. For example, using the Hicks data for STORAX SMALL BOY, which was the next-to-last atmospheric test at NTS on July 14, 1962 (STORAX LITTLE FELLER I was the last atmospheric test on July 17, 1962), it was determined that the relative importance of ^{90}Sr dose to the lung to the total dose from all 177 radionuclides varied, in a mostly linear fashion, from 0.00284 to 0.0589 from 1 to 365 days after detonation, respectively (see Figure A-2). Using the trend line function, an expression for the relative importance of ^{90}Sr dose y to total dose as a function of the number of days after detonation x was developed:

$$y = 0.0001x + 0.0074 \qquad \text{(A-2)}$$

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To ensure favorability to claimants, the relative importance of ^{90}Sr lung dose to the total lung dose from all fission and activation products was examined for several tests including TEAPOT TURK in 1955 and LITTLE FELLER I in 1962 (Hicks 1981a,b,c,d). The slope of the trend lines that predict the relative importance of ^{90}Sr dose was determined to be 0.0001*x* for STORAX SMALL BOY, 0.0002*x* for STORAX LITTLE FELLER I (Figure A-3) and TEAPOT TURK (see Figure A-4). Because the slope of the trend line is directly proportional to the relative importance of the ^{90}Sr dose to total dose (i.e., the larger the slope, the larger the relative importance of ^{90}Sr dose), the tests with the smallest slopes result in the highest

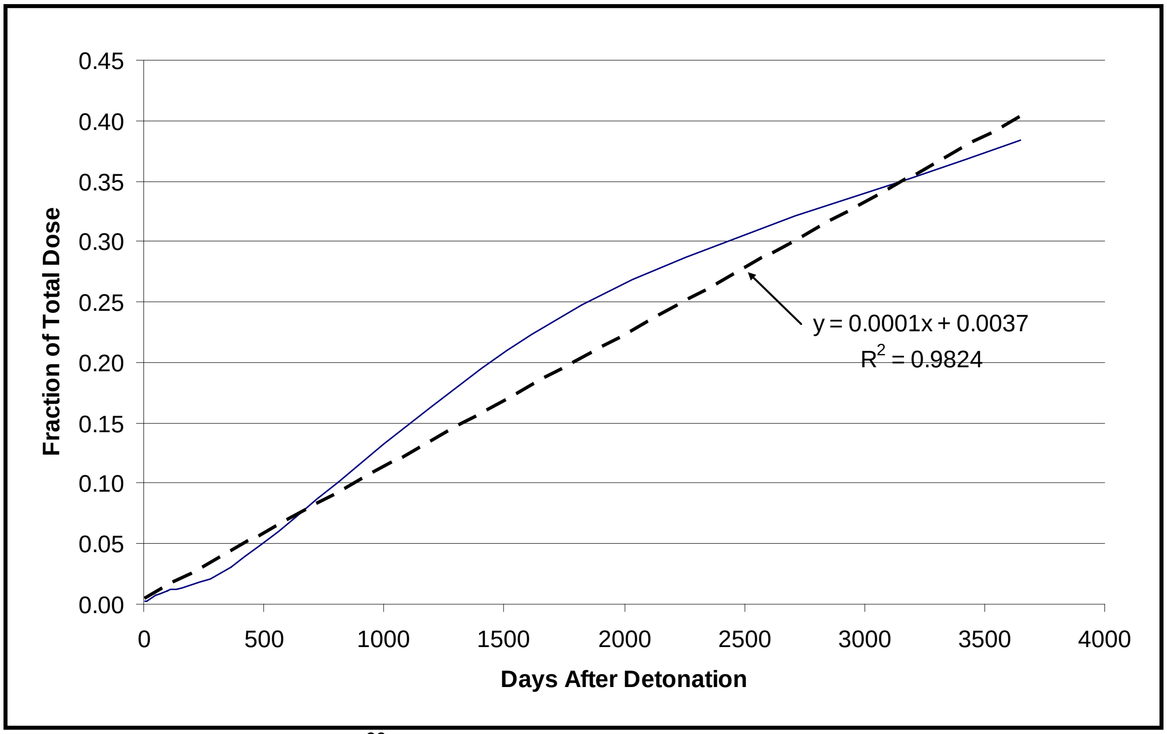

Figure A-2. SMALL BOY – ^{90}Sr fraction of total dose lungs.

multiplicative correction factors for fission and activation products. Therefore, to ensure the organ dose from short-lived fission and activation products is not underestimated, the Hicks (1984) data for the STORAX SMALL BOY test were selected to be used to determine the fission and activation product dose correction factor. This assumption is justified by the fact that the test was very near the last atmospheric test (i.e., STORAX LITTLE FELLER I) and would therefore have been the test most likely to produce the short-lived fission and activation product intakes for workers at NTS after 1962 (the period for which organ dose from environmental intakes is calculated).

Integrating Equation A-2 for SMALL BOY from 0 to 365 days and dividing the result by 365 (the value that represents the integrated total dose for 1 year), it was determined that for the first year after detonation the lung dose from ^{90}Sr represented 0.0000738 or about 0.00738% of the dose from all 177 radionuclides. Therefore, the inverse of the value would produce a factor of 13,600 that the lung dose from the ^{90}Sr intake (Table A-7) could be multiplied by to account for inhalation dose from short-lived fission and activation products. Similar integrations were performed for subsequent years through 1972.

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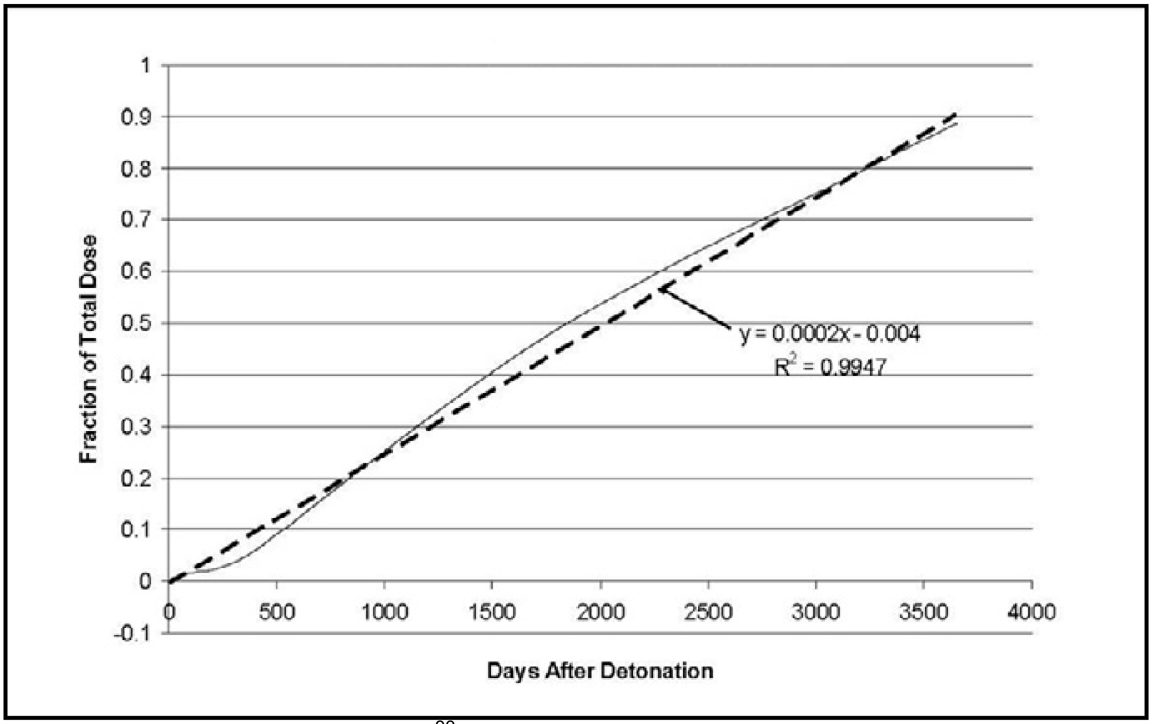

Figure A-3. LITTLE FELLER I – ^{90}Sr fraction of total dose to lungs.

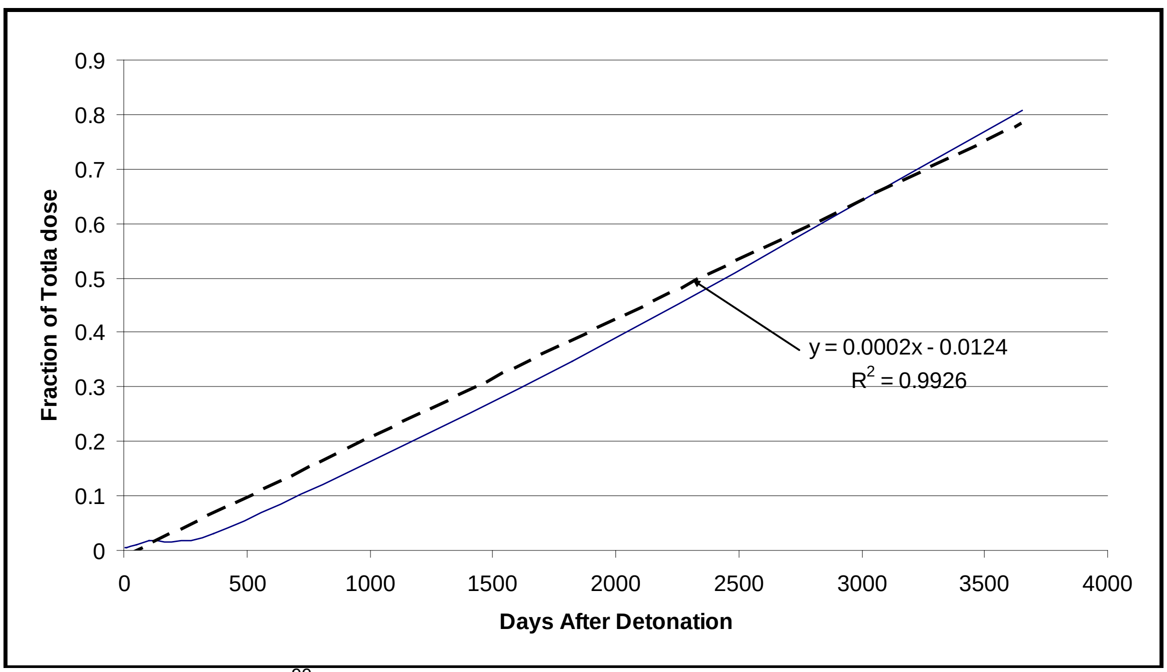

Figure A-4. TURK – ^{90}Sr fraction of total dose to lungs.

Correction factors to account for inhalation intakes of short-lived fission and activation products have been developed for all organs using the Hicks data for STORAX SMALL BOY (Table A-9). These correction factors were based on the relationships that Figures A-5 through A-11 show.

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The IMBA computer program was used to determine the organ doses from the scaled intakes in Table A-7 for a period of 10 years. These doses were multiplied by the correction factors in Table A-9 to determine the additional dose that should be added to account for potential dose from inhalation of short-lived fission and activation products. The organ-specific fission and activation doses are listed in Table A-10.

Table A-9 Organ-specific inhalation dose fission and activation product correction factors.

Organ	Fission and activation product correction factor									
Year	1963	1964	1965	1966	1967	1968	1969	1970	1971	1972
Skin Adrenals Thymus SI Spleen Skin Muscle Uterus Pancreas Kidneys Breast Testes Esophagus Ovaries Brain Stomach Thyroid Gall bladder	730	364	242	182	145	121	104	90.7	80.6	72.5
ULI	458	179	99.2	64.0	45.0	33.4	25.9	20.6	16.8	14.0
Urinary bladder	335	149	91.3	63.6	47.5	37.2	30.1	24.9	21.0	18.0
Lungs	34,900	14,200	7,960	5,150	3,630	2,700	2,100	1,660	1,360	1,130
ET ET1 ET2 LN(TH) LN(ET)	1,570	827	598	492	438	412	412	412	412	412
LLI	420	142	70.8	42.4	28.2	20.1	15.1	11.7	9.4	7.6
Colon	390	148	79.4	50.0	34.5	25.2	19.3	15.2	12.3	10.2
Liver	9,260	4,620	1,540	1,190	988	858	769	706	661	629
Red bone marrow	37.9	18.2	12.8	10.4	10.4	10.4	10.4	10.4	10.4	10.4
Bone surfaces	78.5	40.1	28.1	22.4	22.4	22.4	22.4	22.4	22.4	22.4

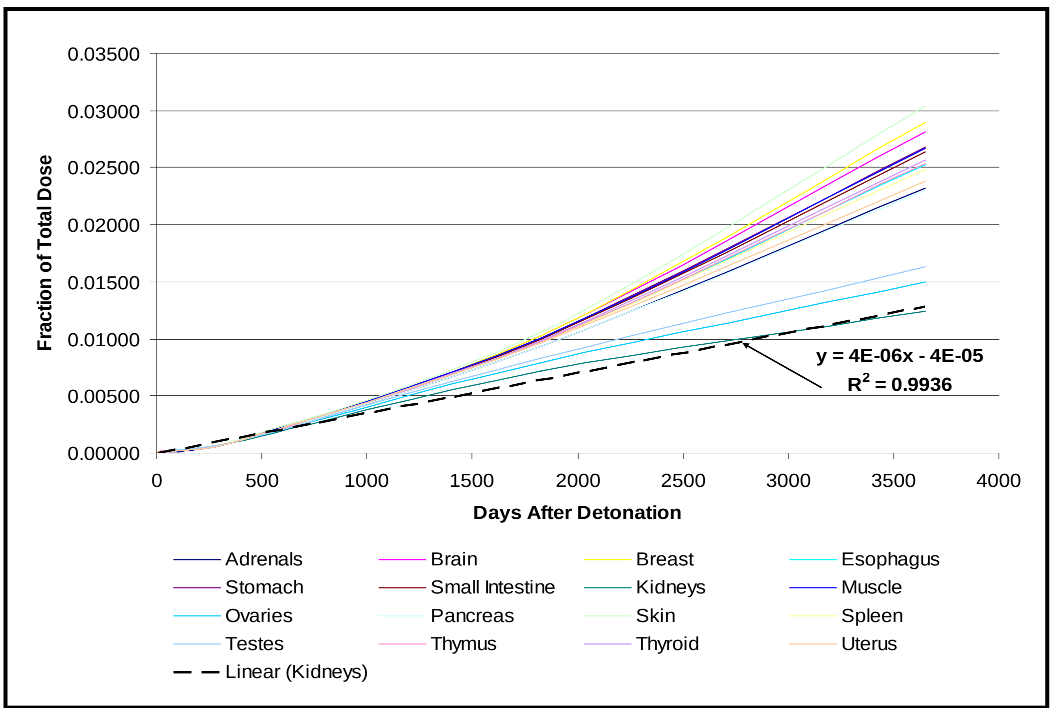

Figure A-5. Strontium-90 fraction of total dose.

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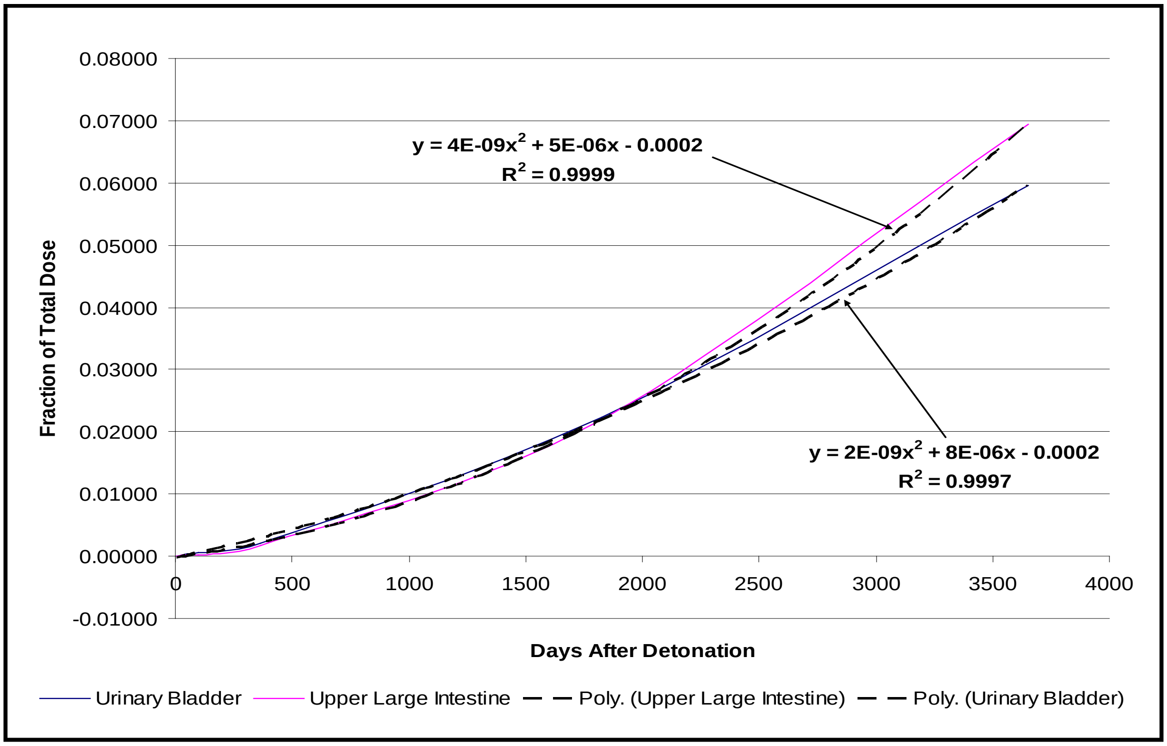

Figure A-6. Strontium-90 fraction of total dose.

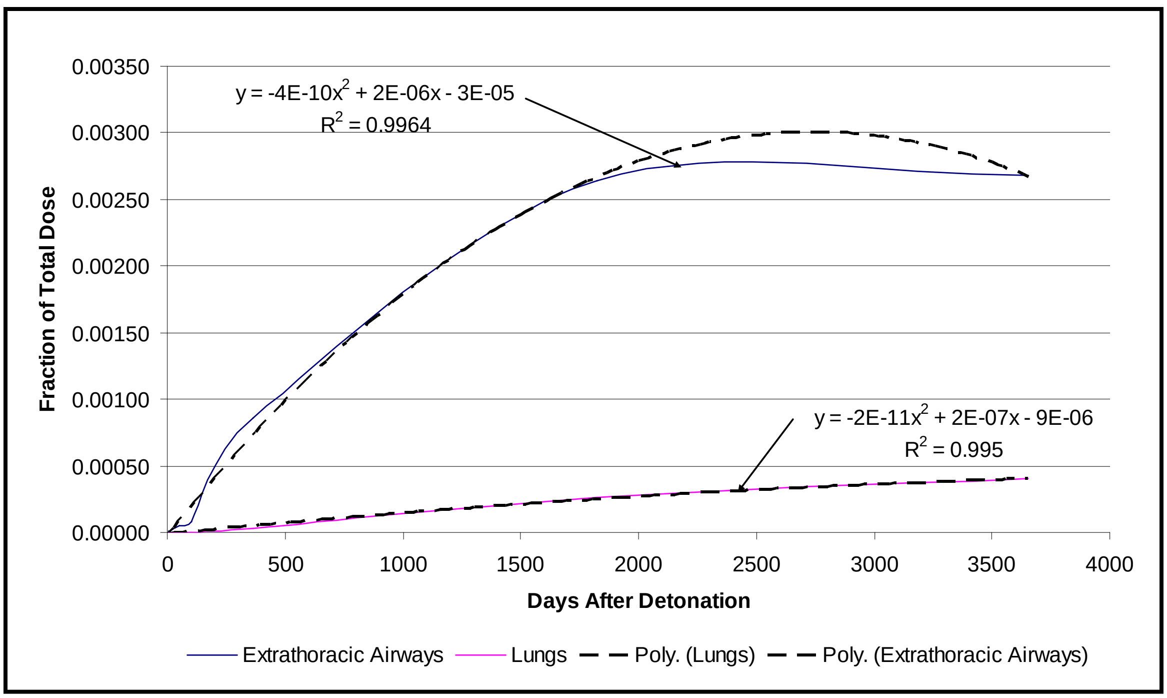

Figure A-7. Strontium-90 fraction of total dose.

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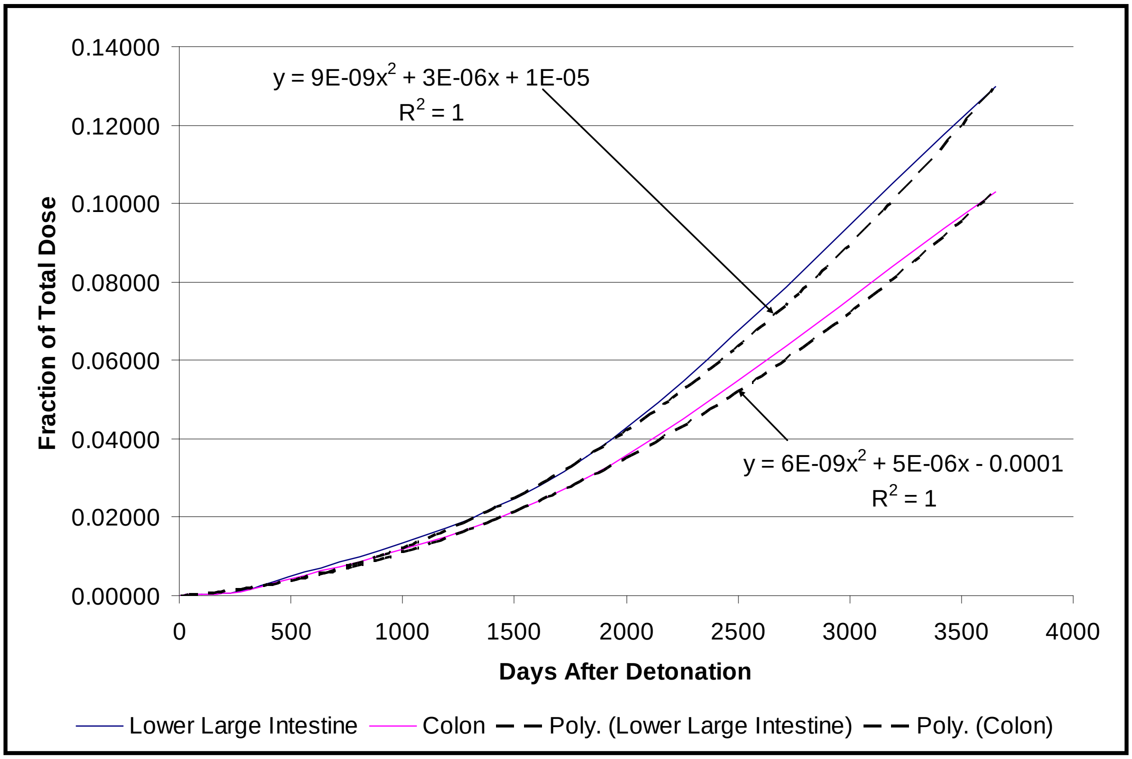

Figure A-8. Strontium-90 fraction of total dose.

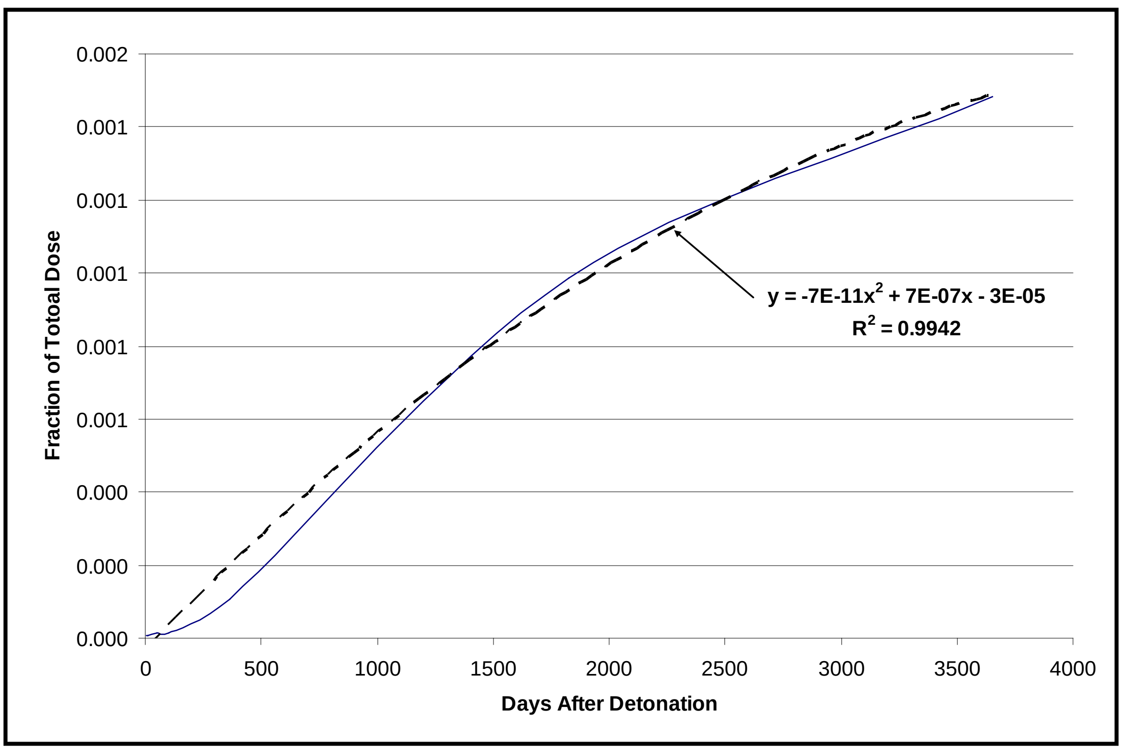

Figure A-9. Strontium-90 fraction of total dose to the liver.

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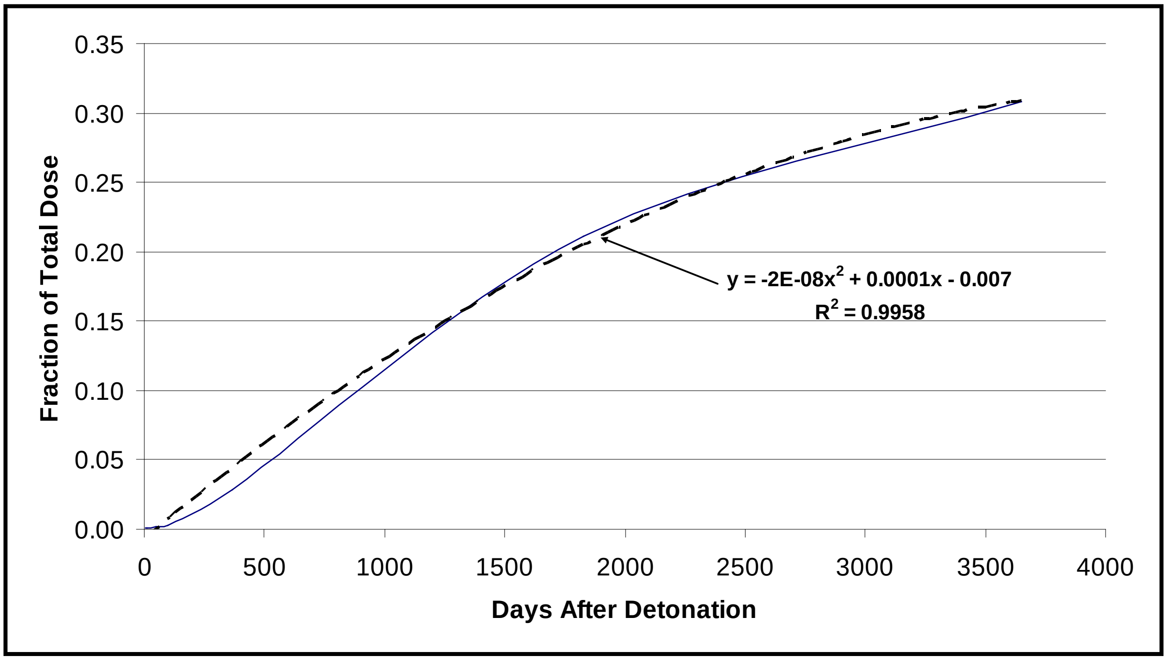

Figure A-10. Strontium-90 fraction of total dose to the red marrow.

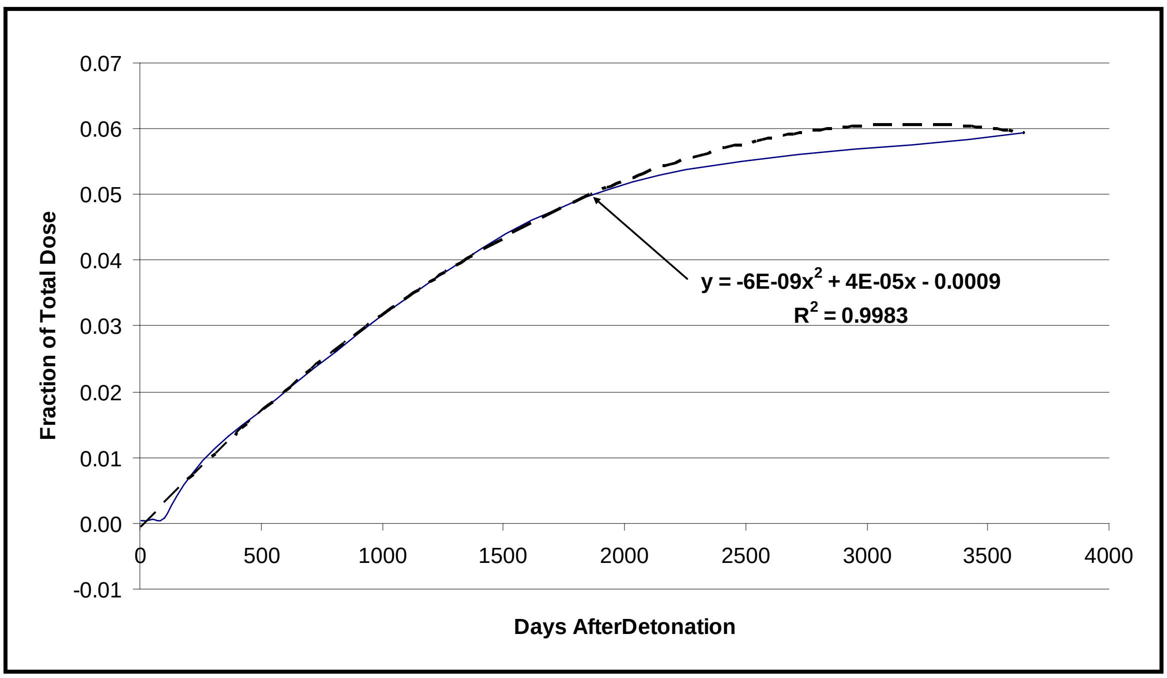

Figure A-11. Strontium-90 fraction of total dose to bone surface.

Table A-10. Inhalation dose from short-lived fission and activation products (rem).

Year	1963	1964	1965	1966	1967	1968	1969	1970	1971	1972
Adrenals	1.52E-04	6.76E-05	3.92E-05	2.89E-05	2.47E-05	2.17E-05	1.94E-05	1.74E-05	1.57E-05	1.42E-05
Bladder	1.86E-04	6.99E-05	3.45E-05	2.26E-05	1.81E-05	1.51E-05	1.29E-05	1.12E-05	9.73E-06	8.53E-06
Brain	1.52E-04	6.76E-05	3.92E-05	2.89E-05	2.47E-05	2.17E-05	1.94E-05	1.74E-05	1.57E-05	1.42E-05
Breast	2.08E-07	3.57E-08	1.65E-08	1.37E-08	1.18E-08	1.01E-08	8.70E-09	7.50E-09	6.49E-09	5.58E-09
Gall bladder	2.08E-07	3.57E-08	1.65E-08	1.37E-08	1.18E-08	1.01E-08	8.70E-09	7.50E-09	6.49E-09	5.58E-09
Heart wall	1.52E-04	6.76E-05	3.92E-05	2.89E-05	2.47E-05	2.17E-05	1.94E-05	1.74E-05	1.57E-05	1.42E-05
Kidney	1.52E-04	6.76E-05	3.92E-05	2.89E-05	2.47E-05	2.17E-05	1.94E-05	1.74E-05	1.57E-05	1.42E-05
Liver	1.93E-03	8.57E-04	3.71E-04	2.70E-04	2.33E-04	2.07E-04	1.85E-04	1.67E-04	1.52E-04	1.39E-04
Muscle	1.52E-04	6.76E-05	3.92E-05	2.89E-05	2.47E-05	2.17E-05	1.94E-05	1.74E-05	1.57E-05	1.42E-05
Ovaries	1.52E-04	6.76E-05	3.92E-05	2.89E-05	2.47E-05	2.17E-05	1.94E-05	1.74E-05	1.57E-05	1.42E-05
Pancreas	1.52E-04	6.76E-05	3.92E-05	2.89E-05	2.47E-05	2.17E-05	1.94E-05	1.74E-05	1.57E-05	1.42E-05
Testes	1.52E-04	6.76E-05	3.92E-05	2.89E-05	2.47E-05	2.17E-05	1.94E-05	1.74E-05	1.57E-05	1.42E-05
Thyroid	1.52E-04	6.76E-05	3.92E-05	2.89E-05	2.47E-05	2.17E-05	1.94E-05	1.74E-05	1.57E-05	1.42E-05
RBM	4.08E-04	7.00E-04	6.65E-04	6.09E-04	5.59E-04	5.18E-04	4.84E-04	4.57E-04	4.35E-04	4.16E-04
Bone surface	1.23E-03	2.03E-03	2.00E-03	1.91E-03	1.83E-03	1.77E-03	1.72E-03	1.69E-03	1.66E-03	1.64E-03
Stomach	1.69E-04	7.22E-05	4.14E-05	3.02E-05	2.57E-05	2.26E-05	2.01E-05	1.81E-05	1.63E-05	1.47E-05
SI	1.85E-04	7.67E-05	4.36E-05	3.15E-05	2.68E-05	2.35E-05	2.09E-05	1.87E-05	1.69E-05	1.53E-05
ULI	4.70E-04	1.36E-04	5.98E-05	3.59E-05	2.73E-05	2.19E-05	1.81E-05	1.52E-05	1.29E-05	1.11E-05
LLI	1.38E-03	3.26E-04	1.24E-04	6.66E-05	4.74E-05	3.63E-05	2.90E-05	2.37E-05	1.98E-05	1.67E-05
Skin	1.52E-04	6.76E-05	3.92E-05	2.89E-05	2.47E-05	2.17E-05	1.94E-05	1.74E-05	1.57E-05	1.42E-05
Spleen	1.52E-04	6.76E-05	3.92E-05	2.89E-05	2.47E-05	2.17E-05	1.94E-05	1.74E-05	1.57E-05	1.42E-05
Thymus	1.52E-04	6.76E-05	3.92E-05	2.89E-05	2.47E-05	2.17E-05	1.94E-05	1.74E-05	1.57E-05	1.42E-05
Uterus	1.52E-04	6.76E-05	3.92E-05	2.89E-05	2.47E-05	2.17E-05	1.94E-05	1.74E-05	1.57E-05	1.42E-05
ET	2.22E-03	6.97E-04	3.78E-04	2.60E-04	2.32E-04	2.19E-04	2.17E-04	2.15E-04	2.14E-04	2.13E-04
Lung	7.67E-03	2.96E-03	1.55E-03	1.08E-03	8.84E-04	7.46E-04	6.40E-04	5.52E-04	4.80E-04	4.18E-04
Colon	7.82E-04	2.08E-04	8.57E-05	4.85E-05	3.57E-05	2.78E-05	2.26E-05	1.86E-05	1.57E-05	1.34E-05
ET1	1.63E+00	4.74E-01	2.48E-01	1.64E-01	1.46E-01	1.38E-01	1.38E-01	1.38E-01	1.38E-01	1.38E-01
ET2	5.83E-04	2.24E-04	1.30E-04	9.60E-05	8.63E-05	8.10E-05	7.92E-05	7.78E-05	7.66E-05	7.55E-05
LN(ET)	3.27E-04	1.50E-04	9.15E-05	7.03E-05	6.34E-05	5.95E-05	5.77E-05	5.63E-05	5.51E-05	5.40E-05
LN(TH)	3.27E-04	1.50E-04	9.15E-05	7.03E-05	6.34E-05	5.95E-05	5.77E-05	5.63E-05	5.51E-05	5.40E-05
Esophagus	1.52E-04	6.76E-05	3.92E-05	2.89E-05	2.47E-05	2.17E-05	1.94E-05	1.74E-05	1.57E-05	1.42E-05

The application of the early fission and activation product correction factor through 1972 intakes is favorable to claimants because the Hicks (1984) data verifies that after 10 years, more than 90% of the total dose for all organs is delivered by ^{241}Am, ^{137}Cs, and/or ^{90}Sr. Further, as times after detonation become greater, the relative importance of ^{241}Am, ^{137}Cs, and ^{90}Sr becomes greater and, because the organ doses from these radionuclides are already accounted for by the scaled intakes (Table A-7), the dose from these radionuclides is doubled.

A.7 Ingestion Pathway

To account for potential intakes from inadvertent ingestion of contaminated soil, the area-specific radionuclide soil deposition data in Table A-5 were converted to volumetric data (i.e., Bq/mg) by assuming a radionuclide relaxation depth of 2.3 cm and a soil density of 1.5 g/cm^3 (DOE 2003). The area-specific radionuclide soil concentrations are listed in Table A-11.

Table A-11. Radionuclide soil concentration by area (Bq/g).

Area	Am-241	Pu-238	Pu-239,240	Co-60	Cs 137	Sr-90	Eu-152	Eu-154	Eu-155
1	0.069	0.127	0.375	0.682	0.263	0.456	1.005	0.014	0.391
2	0.064	0.226	0.463	1.001	0.963	1.882	1.262		0.420
3	0.062	0.050	0.475	0.509	0.294	0.824	0.989	0.012	0.321
4	0.179	0.420	1.036	1.644	0.593	0.655	1.010		0.259
5	0.090	0.018	0.686	3.401	0.109	0.250	6.122	0.259	
6	0.023	0.053	0.108	0.102	0.069	0.087	0.000	0.00	

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Area	Am-241	Pu-238	Pu-239,240	Co-60	Cs 137	Sr-90	Eu-152	Eu-154	Eu-155
7	0.049	0.016	0.344	0.852	0.213	0.384	2.024	0.039	0.322
8	0.530	0.297	3.280	6.741	2.389	1.450	0.562		0.894
9	0.091	0.057	1.844	0.575	0.344	0.524	2.042	0.038	0.311
10	0.411	0.491	2.279	7.973	3.321	2.217	0.195	0.056	5.177
11	0.357	0.065	3.005		0.099	0.060			
12	0.062	0.111	0.408	0.498	0.399	0.346			
15	0.098	0.114	0.740	0.140	0.426	0.502			
16	0.021	0.054	0.107	0.115	0.160	0.209			
17	0.039	0.074	0.238	0.524	0.378	0.488			
18	0.301	0.106	1.518	0.421	0.290	0.502	0.072	0.014	0.607
19	0.061	0.111	0.391	0.122	0.192	0.169			
20	1.607	2.500	2.741	20.946	0.701	0.559	3.723	0.969	16.031
25					0.176	0.090	0.789		
26									
30	4.620	7.749	19.340	43.835	3.953	3.493	4.143	1.252	13.804

If the assumption is made that the workers ingested 100 mg of soil each day, which is favorable to claimants [EPA (1989) recommends a value of 50 mg/d], and that full-time employment was 250 d/yr, annual ingestion can be calculated. The area-specific annual ingestion rates are listed in Table A-12.

Table A-12. Area-specific and maximum annual ingestion rates (Bq/yr).

Area	Am-241	Pu-238	Pu-239,240	Co-60	Cs 137	Sr-90	Eu-152	Eu-154	Eu-155
1	1.72	3.17	9.38	17.06	6.56	11.41	25.12	0.35	9.77
2	1.59	5.64	11.57	25.03	24.08	47.06	31.54		10.51
3	1.54	1.24	11.87	12.72	7.34	20.59	24.73	0.29	8.01
4	4.47	10.49	25.90	41.10	14.83	16.37	25.24		6.47
5	2.24	0.45	17.15	85.03	2.73	6.25	153.05	6.48	0.00
6	0.57	1.32	2.69	2.54	1.71	2.18			
7	1.23	0.40	8.59	21.29	5.33	9.61	50.59	0.97	8.05
8	13.24	7.43	81.99	168.52	59.73	36.25	14.05	0.00	22.34
9	2.27	1.42	46.11	14.38	8.60	13.10	51.04	0.94	7.76
10	10.29	12.27	56.98	199.31	83.02	55.42	4.88	1.41	129.41
11	8.93	1.61	75.12	0.00	2.47	1.51			
12	1.56	2.77	10.20	12.45	9.98	8.65			
15	2.45	2.85	18.49	3.49	10.64	12.56			
16	0.53	1.35	2.68	2.87	4.01	5.21			
17	0.97	1.85	5.94	13.09	9.44	12.19			
18	7.54	2.65	37.95	10.54	7.24	12.55	1.79	0.34	15.17
19	1.53	2.79	9.78	3.05	4.80	4.21			
20	40.17	62.49	68.51	523.64	17.54	13.98	93.06	24.23	400.77
25					4.39	2.24	19.73		
26									
30	115.50	193.72	483.50	1095.89	98.83	87.33	103.56	31.30	345.10
Max[a]	**40.17**	**62.49**	**81.99**	**523.64**	**83.02**	**55.42**	**153.05**	**24.23**	**400.77**

a. Maximum value with Area 30 excluded.

For most radionuclides, Area 30 provided the highest areal deposition and resultant intakes. Area 30 is relatively small (150 km^2) and inaccessible, and is on the Western edge of NTS. It has rugged terrain and includes the northern reaches of Fortymile Canyon. In 1968, it was the site of Project

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BUGGY, the first nuclear row-charge experiment in the PLOWSHARE Program. As a result of the test, a trench 255 m long, 77 m wide, and 206 m deep was created. The test resulted in large quantities of vitrified sand. Because of the bias that is introduced when Area 30 is included, the maximum annual intakes in Table A-12 have been provided with Area 30 areal concentrations excluded. The Area 30 intakes should be used only if it can be determined that the worker was assigned to and primarily worked in Area 30.

To determine the relative importance of the ingestion pathway, a hypothetical ingestion scenario was used that assumed 30 years of the Table A-12 maximum intakes and used the IMBA computer program to determine organ doses from these intakes. The annual organ doses from ingestion of ^{238,239}Pu and ^{241}Am greater than 0.001 rem are listed in Table A-13. With the exception of the red bone marrow and bone surfaces, the annual doses from the Table A-12 maximum ingestion rates of ^{60}Co, ^{137}Cs, and ^{152,154,155}Eu were all less than 0.001 rem. For the red bone marrow and bone surfaces, the annual doses were all less than 0.002 rem.

Table A-13. Annual organ doses from ingestion of ^{238,239}Pu and ^{241}Am (rem).

Year	Liver	Ovaries	Testes	RBM	Bone surface	LLI
1963					0.002	0.001
1964	0.001				0.007	0.001
1965	0.002			0.001	0.011	0.001
1966	0.003			0.001	0.016	0.001
1967	0.004			0.002	0.020	0.001
1968	0.005			0.002	0.024	0.001
1969	0.005			0.002	0.028	0.001
1970	0.006			0.002	0.032	0.001
1971	0.007			0.003	0.035	0.001
1972	0.008			0.003	0.039	0.001
1973	0.008			0.003	0.043	0.001
1974	0.009			0.003	0.046	0.001
1975	0.010			0.003	0.050	0.001
1976	0.010			0.004	0.053	0.001
1977	0.011			0.004	0.057	0.001
1978	0.011			0.004	0.060	0.001
1979	0.012			0.004	0.063	0.001
1980	0.013	0.001	0.001	0.004	0.066	0.001
1981	0.013	0.001	0.001	0.004	0.070	0.001
1982	0.014	0.001	0.001	0.004	0.073	0.001
1983	0.014	0.001	0.001	0.004	0.076	0.001
1984	0.015	0.001	0.001	0.005	0.079	0.001
1985	0.015	0.001	0.001	0.005	0.082	0.001
1986	0.016	0.001	0.001	0.005	0.085	0.001
1987	0.016	0.001	0.001	0.005	0.088	0.001
1988	0.017	0.001	0.001	0.005	0.091	0.001
1989	0.017	0.001	0.001	0.005	0.094	0.001
1990	0.018	0.001	0.001	0.005	0.096	0.001
1991	0.018	0.002	0.002	0.005	0.099	0.001
1992	0.019	0.002	0.002	0.005	0.102	0.001
1993	0.019	0.002	0.002	0.005	0.105	0.001

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Year	Liver	Ovaries	Testes	RBM	Bone surface	LLI
1994	0.019	0.002	0.002	0.005	0.105	
1995	0.019	0.002	0.002	0.005	0.103	
1996	0.018	0.002	0.002	0.005	0.101	
1997	0.018	0.002	0.002	0.004	0.099	
1998	0.017	0.002	0.002	0.004	0.098	
1999	0.017	0.001	0.001	0.004	0.096	
2000	0.017	0.001	0.001	0.004	0.095	
2001	0.016	0.001	0.001	0.004	0.093	
2002	0.016	0.001	0.001	0.003	0.092	
2003	0.016	0.001	0.001	0.003	0.090	
2004	0.015	0.001	0.001	0.003	0.089	
2005	0.015	0.001	0.001	0.003	0.088	
2006	0.015	0.001	0.001	0.003	0.086	
2007	0.014	0.001	0.001	0.003	0.085	

As with the inhalation intakes discussed in Section A.5, ingestion doses need to be adjusted for potential dose from short-lived fission and activation products that are no longer persistent in NTS soils in measurable amounts. The organ-specific fission and activation product correction factors were developed (for the reasons discussed in Section A.5) based on the relative contribution of ^{90}Sr to the total ingestion dose using the Hicks (1981c) data from the STORAX SMALL BOY test in July 1962. As in Section A.5 for inhalation dose, organ-specific relationships were developed for the ^{90}Sr fraction of total dose. These relationships are shown in Figures A-12 through A-17. With the exception of fractional dose to bone surfaces and red bone marrow, these relationships were valid out to 10 years after detonation. For bone surfaces and red bone marrow, the time-dependant relationships were determined for the first year and for the last 9 years.

To account for fractionation, the refractory elements included in the STORAX SMALL BOY Hicks (1981c) data were multiplied by a factor of 2. Increasing the abundance of the refractory elements provides a more reasonable estimate of their relative contribution to total dose. As for the inhalation correction factors, these radionuclide- and time-dependant relative abundances are multiplied by their organ-specific ingestion dose conversion factors (Bunker 1999) to determine the radionuclide-specific, relative importance to total dose as a function of time after detonation. Because of the large difference between the 50-year committed dose conversion factor and the annual dose conversion factor for ^{241}Am, the IMBA computer program was used to calculate the annual ingestion dose conversion factors, which were then used rather than the 50-year committed dose conversion factors. The same method was used for ^{90}Sr annual dose conversion factors. To ensure that organ doses were not underestimated, the f1 factor for each radionuclide was chosen to provide the largest dose to the specific organ of interest.

The ingestion correction factors were evaluated over a 10-year period. This is because ^{144}Ce and ^{106}Ru continue to provide relatively large ingestion doses through 5 years after detonation. Therefore, to capture their contributions to total dose, the correction factor integration times were extended to 10 years if the relative contribution of ^{144}Ce and ^{106}Ru to total dose was less than 3% for all organs. In addition, at the end of 10 years, more than 95% of the total ingestion dose is delivered by ^{90}Sr and ^{137}Cs, their dose accounted for by the ingestion intakes in Table A-12.

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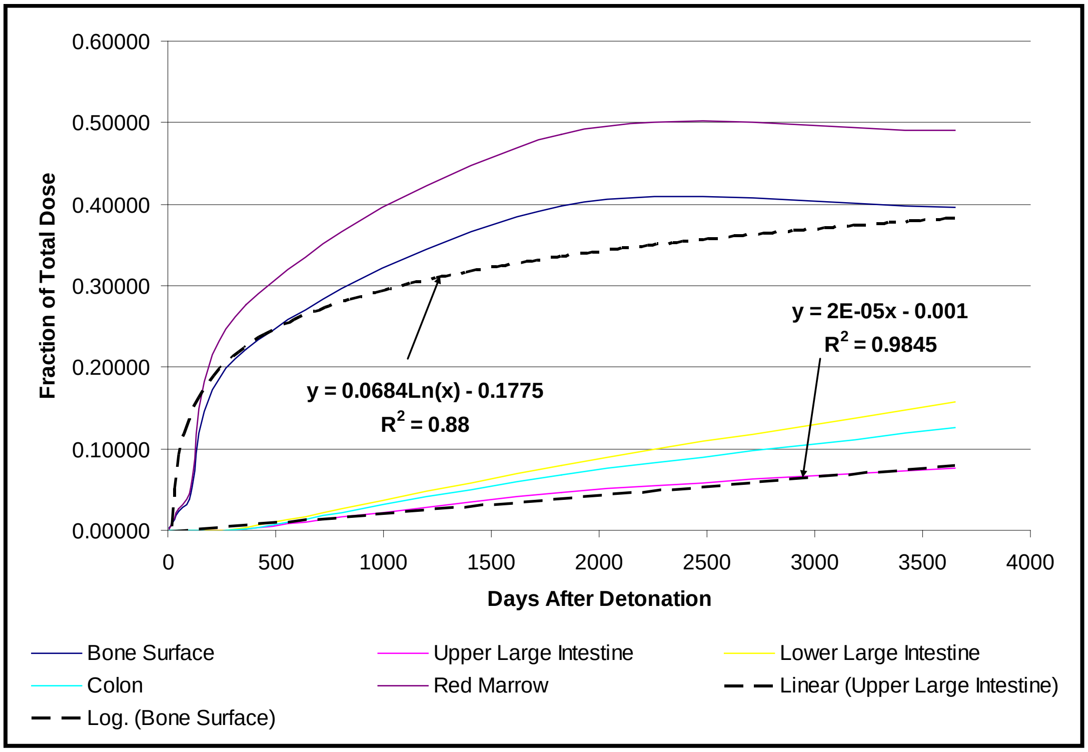

Figure A-12. Strontium-90 fraction of total ingestion dose.

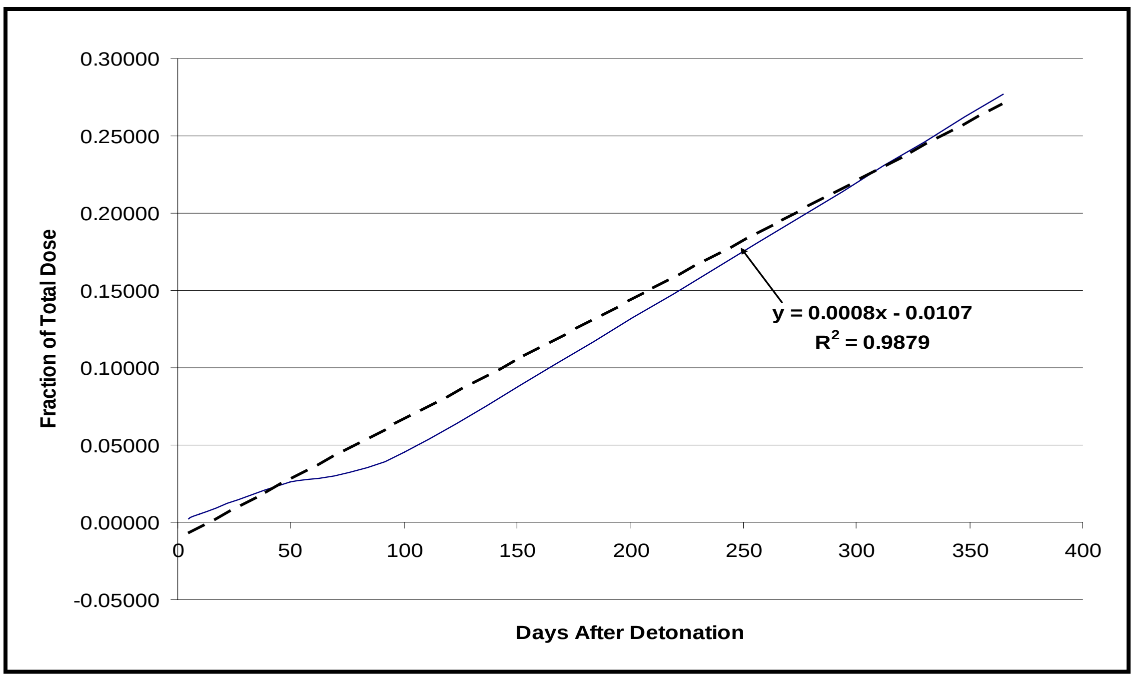

Figure A-13. Strontium-90 fraction of total ingestion dose to red bone marrow.

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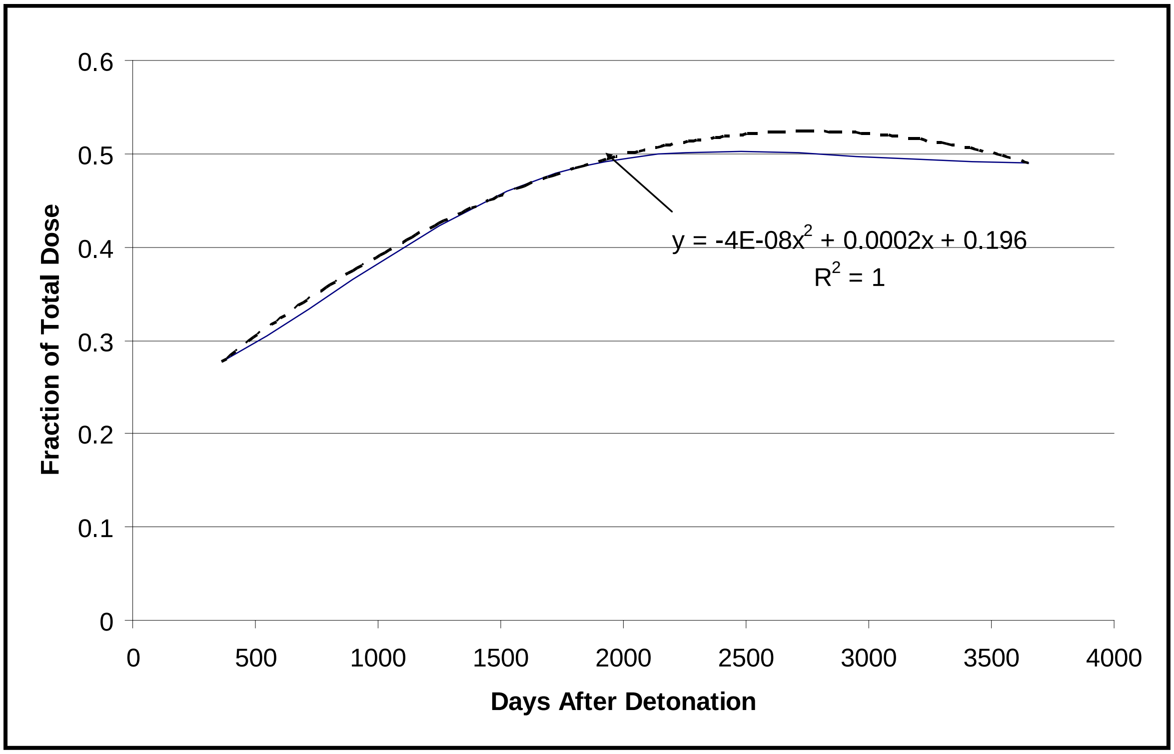

Figure A-14. Strontium-90 fraction of total ingestion dose to red bone marrow.

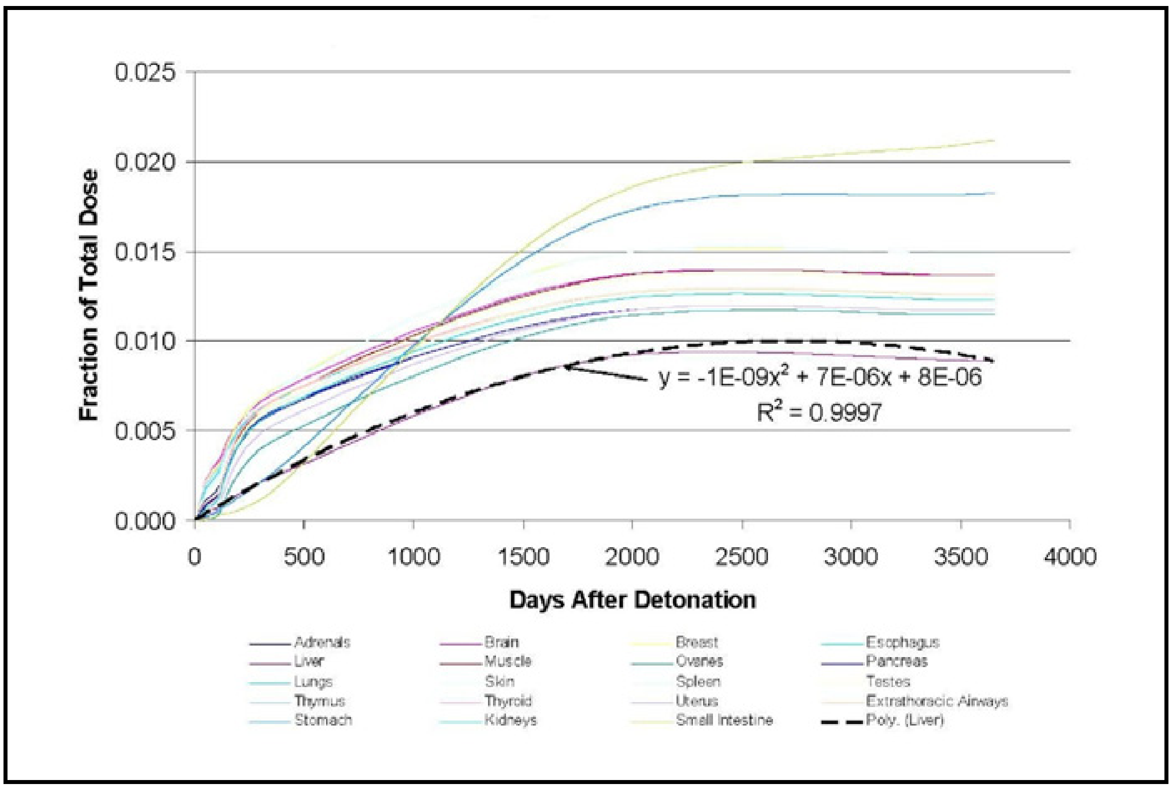

Figure A-15. Strontium-90 fraction of total ingestion dose.

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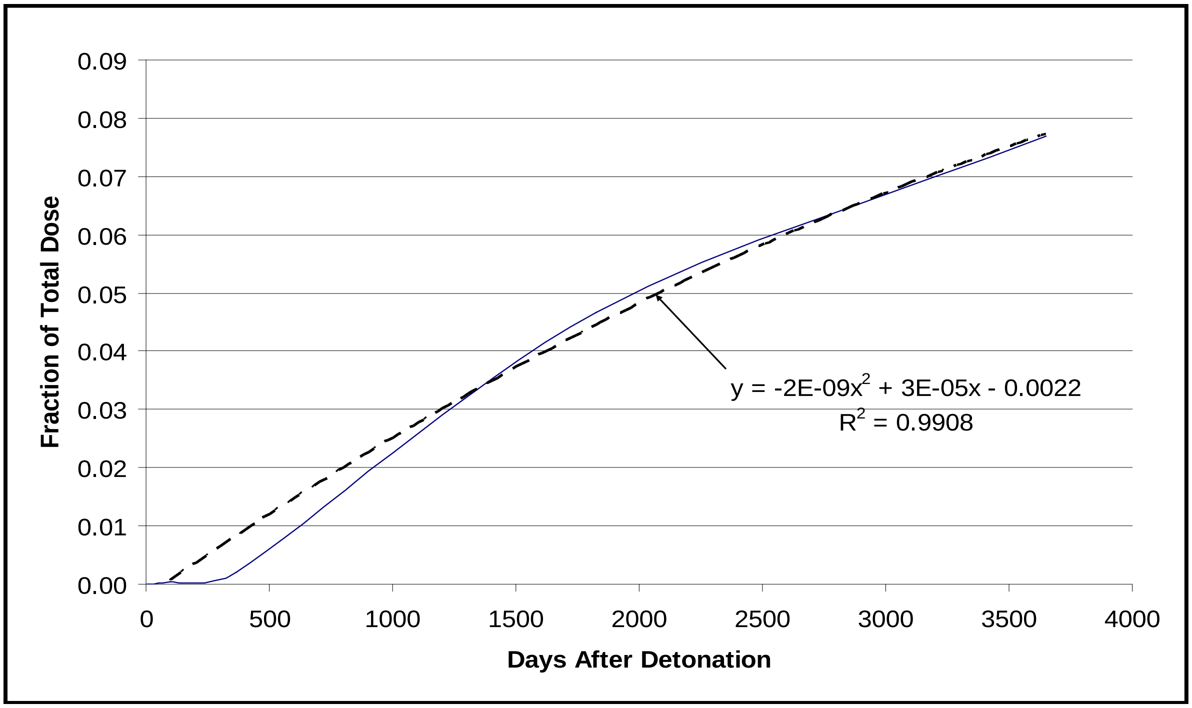

Figure A-16. Strontium-90 fraction of ingestion dose to the upper large intestine.

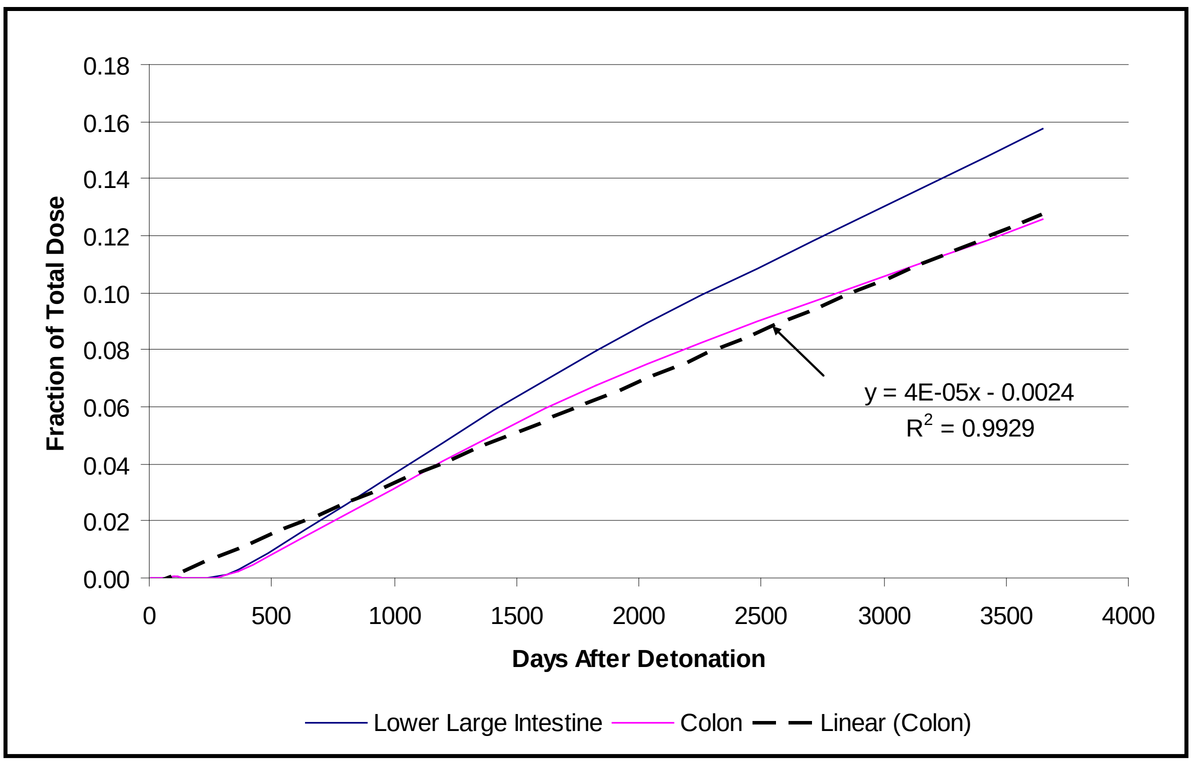

Figure A-17. Strontium-90 fraction of total ingestion dose.

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Using the relationships shown in Figures A-12 through A-17, short-lived fission and activation ingestion correction factors were developed (Table A-14). As with the inhalation fission and activation factors, these correction factors are multiplied by the organ-specific ^{90}Sr annual doses (from the intakes in Table A-12 [i.e., 55.42 Bq/yr]) to calculate additional ingestion dose from short-lived fission and products.

Table A-14. Organ-specific ingestion fission and activation correction factors.

Organ	Correction factor									
Year	**1963**	**1964**	**1965**	**1966**	**1967**	**1968**	**1969**	**1970**	**1971**	**1972**
Adrenals, breast Brain, skin, bladder Stomach, kidneys, muscle, pancreas, brain, esophagus, SI, liver, ovaries ET ET1 ET2 LM(ET) LN(TH), lungs, skin, spleen, testes, thymus, thyroid, uterus Gall bladder	416	219	155	123	106	95.0	88.1	84.0	82.0	81.8
ULI	514	184	95.2	58.2	39.3	18.6	16.3	14.6	13.4	12.4
Bone Surface/RBM	3.8	3.1	2.7	2.5	2.3	2.3	2.3	2.3	2.3	2.3
LLI/Colon	417	208	138	25.6	20.4	17.4	14.6	12.8	11.3	10.2

To simplify the application of organ-specific ingestion dose from short-lived fission and activation products, the IMBA computer program was used to determine the organ-specific annual doses for the ^{90}Sr intake of 55.42 Bq/yr for 1963 through 1972 (Table A-15). These doses are multiplied by the Table A-14 organ-specific correction factors to provide the annual doses from ingestion of short-lived fission and activation products (Table A-16).

Table A-15. Organ-specific annual ingestion doses (rem) for the ^{90}Sr intake of 55.42 Bq/yr.

Organ	1963	1964	1965	1966	1967	1968	1969	1970	1971	1972
Adrenals	2.11E-06	2.47E-06	2.64E-06	2.77E-06	2.89E-06	3.00E-06	3.08E-06	3.16E-06	3.22E-06	3.28E-06
Urinary bladder	5.68E-06	6.42E-06	6.67E-06	6.87E-06	7.04E-06	7.18E-06	7.31E-06	7.41E-06	7.50E-06	7.58E-06
Bone surface	1.58E-04	3.74E-04	5.47E-04	6.98E-04	8.32E-04	9.50E-04	1.06E-03	1.15E-03	1.23E-03	1.31E-03
Brain	2.11E-06	2.47E-06	2.64E-06	2.77E-06	2.89E-06	3.00E-06	3.08E-06	3.16E-06	3.22E-06	3.28E-06
Breast	2.11E-06	2.47E-06	2.64E-06	2.77E-06	2.89E-06	3.00E-06	3.08E-06	3.16E-06	3.22E-06	3.28E-06
Esophagus	2.11E-06	2.47E-06	2.64E-06	2.77E-06	2.89E-06	3.00E-06	3.08E-06	3.16E-06	3.22E-06	3.28E-06
Stomach	3.42E-06	3.78E-06	3.95E-06	4.09E-06	4.21E-06	4.31E-06	4.40E-06	4.47E-06	4.54E-06	4.59E-06
Small intestine	4.71E-06	5.07E-06	5.24E-06	5.38E-06	5.50E-06	5.60E-06	5.69E-06	5.76E-06	5.83E-06	5.88E-06
Upper large intestine	3.42E-05	3.43E-05	3.43E-05	3.43E-05	3.43E-05	3.43E-05	3.43E-05	3.43E-05	3.43E-05	3.43E-05
Lower large intestine	1.45E-04	1.45E-04	1.45E-04	1.45E-04	1.45E-04	1.45E-04	1.45E-04	1.45E-04	1.45E-04	1.45E-04
Colon	4.36E-03	2.26E-03	1.51E-03	1.14E-03	9.16E-04	7.71E-04	6.61E-04	5.84E-04	5.21E-04	4.71E-04
Kidneys	2.11E-06	2.47E-06	2.64E-06	2.77E-06	2.89E-06	3.00E-06	3.08E-06	3.16E-06	3.22E-06	3.28E-06
Liver	2.11E-06	2.47E-06	2.64E-06	2.77E-06	2.89E-06	3.00E-06	3.08E-06	3.16E-06	3.22E-06	3.28E-06
Muscle	2.11E-06	2.47E-06	2.64E-06	2.77E-06	2.89E-06	3.00E-06	3.08E-06	3.16E-06	3.22E-06	3.28E-06
Ovaries	2.11E-06	2.47E-06	2.64E-06	2.77E-06	2.89E-06	3.00E-06	3.08E-06	3.16E-06	3.22E-06	3.28E-06
Pancreas	2.11E-06	2.47E-06	2.64E-06	2.77E-06	2.89E-06	3.00E-06	3.08E-06	3.16E-06	3.22E-06	3.28E-06
Red marrow	1.08E-04	2.66E-04	3.87E-04	4.86E-04	5.69E-04	6.37E-04	6.94E-04	7.42E-04	7.81E-04	8.14E-04
Extrathoracic airways	2.11E-06	2.47E-06	2.64E-06	2.77E-06	2.89E-06	3.00E-06	3.08E-06	3.16E-06	3.22E-06	3.28E-06
Lungs	2.11E-06	2.47E-06	2.64E-06	2.77E-06	2.89E-06	3.00E-06	3.08E-06	3.16E-06	3.22E-06	3.28E-06

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Organ	1963	1964	1965	1966	1967	1968	1969	1970	1971	1972
Skin	2.11E-06	2.47E-06	2.64E-06	2.77E-06	2.89E-06	3.00E-06	3.08E-06	3.16E-06	3.22E-06	3.28E-06
Spleen	2.11E-06	2.47E-06	2.64E-06	2.77E-06	2.89E-06	3.00E-06	3.08E-06	3.16E-06	3.22E-06	3.28E-06
Testes	2.11E-06	2.47E-06	2.64E-06	2.77E-06	2.89E-06	3.00E-06	3.08E-06	3.16E-06	3.22E-06	3.28E-06
Thymus	2.11E-06	2.47E-06	2.64E-06	2.77E-06	2.89E-06	3.00E-06	3.08E-06	3.16E-06	3.22E-06	3.28E-06
Thyroid	2.11E-06	2.47E-06	2.64E-06	2.77E-06	2.89E-06	3.00E-06	3.08E-06	3.16E-06	3.22E-06	3.28E-06
Uterus	2.11E-06	2.47E-06	2.64E-06	2.77E-06	2.89E-06	3.00E-06	3.08E-06	3.16E-06	3.22E-06	3.28E-06

Table A-16. Organ-specific doses (rem) from ingestion of fission and activation products.

Organ	1963	1964	1965	1966	1967	1968	1969	1970	1971	1972
Adrenals	8.78E-04	6.12E-04	4.75E-04	4.09E-04	3.73E-04	3.49E-04	3.32E-04	3.20E-04	3.13E-04	1.40E-04
Bladder	2.36E-03	1.55E-03	1.15E-03	9.50E-04	8.45E-04	7.76E-04	7.31E-04	7.02E-04	6.85E-04	2.21E-04
Brain	8.78E-04	6.12E-04	4.75E-04	4.09E-04	3.73E-04	3.49E-04	3.32E-04	3.20E-04	3.13E-04	1.40E-04
Breast	8.78E-04	6.12E-04	4.75E-04	4.09E-04	3.73E-04	3.49E-04	3.32E-04	3.20E-04	3.13E-04	1.40E-04
Gall bladder	8.78E-04	6.12E-04	4.75E-04	4.09E-04	3.73E-04	3.49E-04	3.32E-04	3.20E-04	3.13E-04	1.40E-04
Heart wall	8.78E-04	6.12E-04	4.75E-04	4.09E-04	3.73E-04	3.49E-04	3.32E-04	3.20E-04	3.13E-04	1.40E-04
Kidney	8.78E-04	6.12E-04	4.75E-04	4.09E-04	3.73E-04	3.49E-04	3.32E-04	3.20E-04	3.13E-04	1.40E-04
Liver	8.78E-04	6.12E-04	4.75E-04	4.09E-04	3.73E-04	3.49E-04	3.32E-04	3.20E-04	3.13E-04	1.40E-04
Muscle	8.78E-04	6.12E-04	4.75E-04	4.09E-04	3.73E-04	3.49E-04	3.32E-04	3.20E-04	3.13E-04	1.40E-04
Ovaries	8.78E-04	6.12E-04	4.75E-04	4.09E-04	3.73E-04	3.49E-04	3.32E-04	3.20E-04	3.13E-04	1.40E-04
Pancreas	8.78E-04	6.12E-04	4.75E-04	4.09E-04	3.73E-04	3.49E-04	3.32E-04	3.20E-04	3.13E-04	1.40E-04
Testes	8.78E-04	6.12E-04	4.75E-04	4.09E-04	3.73E-04	3.49E-04	3.32E-04	3.20E-04	3.13E-04	1.40E-04
Thyroid	8.78E-04	6.12E-04	4.75E-04	4.09E-04	3.73E-04	3.49E-04	3.32E-04	3.20E-04	3.13E-04	1.40E-04
RBM	4.12E-04	2.66E-04	3.87E-04	4.86E-04	5.69E-04	6.37E-04	6.94E-04	7.42E-04	7.81E-04	8.14E-04
Bone surface	6.01E-04	1.31E-03	1.75E-03	2.09E-03	2.35E-03	2.56E-03	2.76E-03	2.94E-03	3.11E-03	3.05E-03
Stomach	1.42E-03	9.00E-04	6.79E-04	5.71E-04	5.13E-04	4.74E-04	4.48E-04	4.31E-04	4.21E-04	1.41E-04
SI	1.96E-03	1.18E-03	8.79E-04	7.29E-04	6.49E-04	5.96E-04	5.61E-04	5.39E-04	5.27E-04	1.43E-04
ULI	1.53E-02	5.87E-03	3.11E-03	1.96E-03	1.36E-03	7.26E-04	6.30E-04	5.64E-04	5.14E-04	1.34E-04
LLI	4.84E-02	2.50E-02	1.66E-02	3.47E-03	2.66E-03	2.23E-03	1.92E-03	1.68E-03	1.49E-03	1.34E-03
Skin	8.78E-04	6.12E-04	4.75E-04	4.09E-04	3.73E-04	3.49E-04	3.32E-04	3.20E-04	3.13E-04	1.40E-04
Spleen	8.78E-04	6.12E-04	4.75E-04	4.09E-04	3.73E-04	3.49E-04	3.32E-04	3.20E-04	3.13E-04	1.40E-04
Thymus	2.11E-06	2.47E-06	2.64E-06	2.77E-06	2.89E-06	3.00E-06	3.08E-06	3.16E-06	3.22E-06	3.28E-06
Uterus	8.78E-04	6.12E-04	4.75E-04	4.09E-04	3.73E-04	3.49E-04	3.32E-04	3.20E-04	3.13E-04	1.40E-04
ET	8.78E-04	6.12E-04	4.75E-04	4.09E-04	3.73E-04	3.49E-04	3.32E-04	3.20E-04	3.13E-04	1.40E-04
Lung	8.78E-04	6.12E-04	4.75E-04	4.09E-04	3.73E-04	3.49E-04	3.32E-04	3.20E-04	3.13E-04	1.40E-04
Colon	2.80E-02	1.45E-02	9.66E-03	2.07E-03	1.59E-03	1.33E-03	1.15E-03	1.01E-03	8.94E-04	8.06E-04
ET1	8.78E-04	6.12E-04	4.75E-04	4.09E-04	3.73E-04	3.49E-04	3.32E-04	3.20E-04	3.13E-04	1.40E-04
ET2	8.78E-04	6.12E-04	4.75E-04	4.09E-04	3.73E-04	3.49E-04	3.32E-04	3.20E-04	3.13E-04	1.40E-04
LN(ET)	8.78E-04	6.12E-04	4.75E-04	4.09E-04	3.73E-04	3.49E-04	3.32E-04	3.20E-04	3.13E-04	1.40E-04
LN(TH)	8.78E-04	6.12E-04	4.75E-04	4.09E-04	3.73E-04	3.49E-04	3.32E-04	3.20E-04	3.13E-04	1.40E-04
Esophagus	2.89E-04	2.07E-04	1.62E-04	1.41E-04	1.29E-04	1.21E-04	1.16E-04	1.13E-04	1.11E-04	5.02E-05

A.8 Instruction to Dose Reconstructors for Assignment of Environmental Intakes

With the exception of cases that can be worked with the use of the bounding assumptions from ORAUT-OTIB-0018 (ORAUT 2005), environmental inhalation and ingestion intakes in Tables A-7 and A-12, respectively, should be applied starting in 1963. In addition, for applicable years of employment and affected organs, the annual doses in Tables A-10 and A-16 should be applied to account for dose from inhalation and ingestion of short-lived fission and activation products. These intakes and resultant doses should be entered into the IREP analysis with a constant distribution because they are reasonable overestimates of the actual intakes and doses. It should be noted that doses that are less than 0.001 rem do not need to be entered into the IREP analysis because they would not affect the overall dose to the affected organ. The organs that have dose equal to or greater than 0.001 rem and the resultant dose are given in Tables 4-9 and 4-14. Only these doses need to be entered into the IREP analysis and they should be entered as 30-to-250-keV photons with a constant distribution.

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In the previous sections, methods were presented to account for organ dose from ambient environmental inhalation and ingestion intakes of radioactive materials in the NTS atmosphere and soils. To illustrate the importance of these pathways, the IMBA computer program was used to determine organ doses from 30 years of the inhalation intakes in Table A-7 and the ingestion intakes in Table A-12. These 30-year organ doses and the fractional contribution for each of the radionuclides persistent in NTS soils are listed in Table B-1. To correct for exposure to short-lived fission and activation, the dose reconstructor should add annual doses that were greater than 0.001 rem from Tables 4-9 and 4-14 as 30-to-250-keV photons with a constant distribution. The doses shown in table B-1 are for purposes of illustration only and should not be entered into the IREP analysis. These doses will be accounted for by the assignment of the intakes shown in Tables 4-7 and 4-11 (MAX) and repeated in Tables A-7 and A-12 (MAX) for all years of employment.

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Table B-1. Total annual organ doses greater than 0.0001 rem from 30 years of inhalation and ingestion intakes (rem).

Year	Radiation Type	Liver	Ovaries	Testes	RBM	Bone surface	LLI	ET	Lung	Colon	ET1	ET2	LN(ET)	LN(TH)
1963	Alpha	0.001				0.006	0.001	0.004	0.007	0.001	0.005	0.004		0.001
1964	Alpha	0.003			0.002	0.017	0.001	0.009	0.006	0.001	0.003	0.009	0.001	0.003
1965	Alpha	0.005			0.002	0.026	0.001	0.009	0.005	0.001	0.002	0.009	0.001	0.004
1966	Alpha	0.006			0.003	0.033	0.001	0.009	0.004	0.001	0.002	0.009	0.002	0.006
1967	Alpha	0.008			0.003	0.040	0.001	0.009	0.004	0.001	0.002	0.009	0.003	0.007
1968	Alpha	0.009			0.004	0.046	0.001	0.008	0.004	0.001	0.002	0.008	0.003	0.008
1969	Alpha	0.010			0.004	0.052	0.001	0.008	0.004	0.001	0.002	0.008	0.004	0.009
1970	Alpha	0.011			0.004	0.058	0.001	0.008	0.004	0.001	0.002	0.008	0.004	0.011
1971	Alpha	0.012			0.005	0.065	0.001	0.008	0.004	0.001	0.002	0.008	0.005	0.012
1972	Alpha	0.013	0.001	0.001	0.005	0.070	0.001	0.008	0.004	0.001	0.002	0.008	0.005	0.013
1973	Alpha	0.015	0.001	0.001	0.005	0.076	0.001	0.008	0.004	0.001	0.002	0.008	0.006	0.014
1974	Alpha	0.016	0.001	0.001	0.006	0.082	0.001	0.008	0.004	0.001	0.002	0.008	0.006	0.016
1975	Alpha	0.017	0.001	0.001	0.006	0.088	0.001	0.008	0.005	0.001	0.002	0.008	0.007	0.017
1976	Alpha	0.018	0.001	0.001	0.006	0.093	0.001	0.008	0.005	0.001	0.002	0.008	0.007	0.018
1977	Alpha	0.019	0.001	0.001	0.006	0.098	0.001	0.008	0.005	0.001	0.002	0.008	0.007	0.019
1978	Alpha	0.020	0.002	0.002	0.006	0.104	0.001	0.008	0.005	0.001	0.002	0.008	0.008	0.020
1979	Alpha	0.021	0.002	0.002	0.007	0.109	0.001	0.008	0.005	0.001	0.002	0.008	0.008	0.022
1980	Alpha	0.021	0.002	0.002	0.007	0.114	0.001	0.008	0.005	0.001	0.002	0.008	0.008	0.023
1981	Alpha	0.022	0.002	0.002	0.007	0.119	0.001	0.008	0.005	0.001	0.002	0.008	0.009	0.024
1982	Alpha	0.023	0.002	0.002	0.007	0.124	0.001	0.008	0.005	0.001	0.002	0.008	0.009	0.025
1983	Alpha	0.024	0.002	0.002	0.007	0.129	0.001	0.008	0.005	0.001	0.002	0.008	0.009	0.026
1984	Alpha	0.025	0.002	0.002	0.008	0.134	0.001	0.008	0.005	0.001	0.002	0.008	0.009	0.027
1985	Alpha	0.026	0.002	0.002	0.008	0.139	0.001	0.008	0.005	0.001	0.002	0.008	0.010	0.028
1986	Alpha	0.027	0.002	0.002	0.008	0.144	0.001	0.008	0.005	0.001	0.002	0.008	0.010	0.029
1987	Alpha	0.027	0.002	0.002	0.008	0.149	0.001	0.008	0.005	0.001	0.002	0.008	0.010	0.030
1988	Alpha	0.028	0.002	0.002	0.008	0.153	0.001	0.008	0.005	0.001	0.002	0.008	0.010	0.031
1989	Alpha	0.029	0.002	0.002	0.008	0.158	0.001	0.008	0.005	0.001	0.002	0.008	0.011	0.032
1990	Alpha	0.030	0.002	0.002	0.008	0.162	0.001	0.008	0.005	0.001	0.002	0.008	0.011	0.033
1991	Alpha	0.030	0.003	0.003	0.009	0.167	0.001	0.008	0.005	0.001	0.002	0.008	0.011	0.034
1992	Alpha	0.031	0.003	0.003	0.009	0.171	0.001	0.008	0.005	0.001	0.002	0.008	0.011	0.035
1993	Alpha	0.031	0.003	0.003	0.009	0.172		0.006	0.003			0.006	0.011	0.035
1994	Alpha	0.030	0.003	0.003	0.008	0.169		0.004	0.002			0.004	0.011	0.035
1995	Alpha	0.030	0.003	0.003	0.008	0.166		0.003	0.002			0.003	0.011	0.035
1996	Alpha	0.029	0.003	0.003	0.007	0.163		0.002	0.001			0.002	0.011	0.035
1997	Alpha	0.028	0.002	0.003	0.007	0.160		0.001	0.001			0.001	0.011	0.035
1998	Alpha	0.028	0.002	0.002	0.006	0.158							0.010	0.034
1999	Alpha	0.027	0.002	0.002	0.006	0.155							0.010	0.034
2000	Alpha	0.026	0.002	0.002	0.006	0.153							0.010	0.033

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Year	Radiation Type	Liver	Ovaries	Testes	RBM	Bone surface	LLI	ET	Lung	Colon	ET1	ET2	LN(ET)	LN(TH)
2001	Alpha	0.026	0.002	0.002	0.006	0.151							0.009	0.033
2002	Alpha	0.025	0.002	0.002	0.005	0.148							0.009	0.032
2003	Alpha	0.025	0.002	0.002	0.005	0.146							0.009	0.032
2004	Alpha	0.024	0.002	0.002	0.005	0.144							0.008	0.031
2005	Alpha	0.024	0.002	0.002	0.005	0.142							0.008	0.031
2006	Alpha	0.023	0.002	0.002	0.005	0.140							0.008	0.030
2007	Alpha	0.023	0.002	0.002	0.005	0.138							0.007	0.029
2008	Alpha	0.022	0.002	0.002	0.004	0.136							0.007	0.029
2009	Alpha	0.022	0.002	0.002	0.004	0.134							0.007	0.028
2010	Alpha	0.021	0.002	0.002	0.004	0.132							0.007	0.027
2011	Alpha	0.021	0.002	0.002	0.004	0.130							0.006	0.026
2012	Alpha	0.021	0.002	0.002	0.004	0.128							0.006	0.026
1963	Photons E>250 keV										0.002			
1964	Photons E>250 keV										0.001			
1965	Photons E>250 keV										0.001			
1963	Electrons E>15 keV										0.002			
1964	Electrons E>15 keV										0.001			
1965	Electrons E>15 keV													
1966	Electrons E>15 keV													
1967	Electrons E>15 keV					0.001								
1968	Electrons E>15 keV					0.001								
1969	Electrons E>15 keV					0.001								
1970	Electrons E>15 keV					0.001								
1971	Electrons E>15 keV					0.001								
1972	Electrons E>15 keV					0.002								
1973	Electrons E>15 keV					0.002								
1974	Electrons E>15 keV				0.001	0.002								
1975	Electrons E>15 keV				0.001	0.002								
1976	Electrons E>15 keV				0.001	0.002								
1977	Electrons E>15 keV				0.001	0.002								
1978	Electrons E>15 keV				0.001	0.002								
1979	Electrons E>15 keV				0.001	0.002								
1980	Electrons E>15 keV				0.001	0.002								
1981	Electrons E>15 keV				0.001	0.002								
1982	Electrons E>15 keV				0.001	0.002								
1983	Electrons E>15 keV				0.001	0.002								
1984	Electrons E>15 keV				0.001	0.002								
1985	Electrons E>15 keV				0.001	0.002								
1986	Electrons E>15 keV				0.001	0.002								

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Year	Radiation Type	Liver	Ovaries	Testes	RBM	Bone surface	LLI	ET	Lung	Colon	ET1	ET2	LN(ET)	LN(TH)
1987	Electrons E>15 keV				0.001	0.002								
1988	Electrons E>15 keV				0.001	0.002								
1989	Electrons E>15 keV				0.001	0.002								
1990	Electrons E>15 keV				0.001	0.002								
1991	Electrons E>15 keV				0.001	0.002								
1992	Electrons E>15 keV				0.001	0.002								
1993	Electrons E>15 keV					0.002								
1994	Electrons E>15 keV					0.002								
1995	Electrons E>15 keV					0.002								
1996	Electrons E>15 keV					0.001								
1997	Electrons E>15 keV					0.001								
1998	Electrons E>15 keV					0.001								
1999	Electrons E>15 keV					0.001								
2000	Electrons E>15 keV					0.001								