ORAU Team **NIOSH Dose Reconstruction Project** Technical Basis Document for the Nevada Test Site – Occupational Environmental Dose	Document Number: ORAUT-TKBS-0008-4 Effective Date: 04/20/2004 Revision No.: 00 Controlled Copy No.: _____ Page 1 of 54

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ACRONYMS AND ABBREVIATIONS

Bq	becquerel
CEDE	committed effective dose equivalent
Ci	curie
cm^2	square centimeter
d	day
DoD	U.S. Department of Defense
EPA	U.S. Environmental Protection Agency
ft	foot
GSD	geometric standard deviation
hr	hour
HT	airborne tritium
HTO	tritium oxide
in.	inch
kg	kilogram
km^2	square kilometer
L	liter
LANL	Los Alamos National Laboratory
lb	pound
m	meter
m^2	square meter
m^3	cubic meter
Mev	megavolt-electron, 1 million electron volts
mg	milligram
mi^2	square mile
min	minute
ml	milliliter
mR	milliroentgen
mrem	millirem
NTS	Nevada Test Site
pCi	picocurie
PERM	passive environmental radon monitor
REECo	Reynolds Electrical & Engineering Co., Inc.
RDC [RDCs]	radon daughter concentrations
RPISU	radon progeny integrating sampling unit
s	second

Sv   sievert

TLD   thermoluminescent dosimeter

WL   working level
WLM   working level month

yr   year

µCi   microcurie

4.1   INTRODUCTION

Occupational environmental dose is the dose received by individuals at the Nevada Test Site (NTS or Test Site) while outside onsite operational facilities 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 cause only an external dose because these inert radionuclides are not readily absorbed by the body. Job classifications of covered employees that may spend a majority of their time in the outdoor environment include carpenters, cement masons, drillers, electrical wiremen/linemen, ironworkers, laborers, operating engineers, painters, plumbers, pipefitters, sheet metal workers, and teamsters.

Section 4.2 discusses internal doses to unmonitored workers from on-site releases to the air and resuspension of radioactive materials in soil. Section 4.3 describes external doses to workers from ambient radiation and releases of radioactive noble gases into air. Section 4.4 discusses dose to tunnel workers from exposure to radon.

4.2   INTERNAL DOSE FROM ON-SITE ATMOSPHERIC RADIONUCLIDE CONCENTRATIONS

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. Section 4.2.3 discusses the uncertainties involved in these dose estimates.

4.2.1   <u>On-Site Releases to Air</u>

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.

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 above-ground 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. Finally, there was fallout of radioactive debris from many tests over the northern and eastern parts of the Test Site.

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 approximately 1.2 × 10^{10} Ci to the atmosphere. 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 the site. However, much of the nonvolatile, long-lived radionuclides settled back into the soils at various locations at NTS (see Table 4.2.2-1, Section 4.2.2 below). 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.

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

- *Containment Failures:* 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.*

- *Late-Time Seeps:* 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.

- *Controlled Tunnel Purging:* A controlled tunnel purge is an intentional release to allow either recovery of experimental data and equipment or reuse of part of the tunnel system.

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

At the present time, 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 and mixed wastes. Monitoring and evaluation of the various activities indicate that the potential sources of onsite radiation exposure are releases from the following sources:

- Evaporated tritiated water (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.

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 conducted 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, currently 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 which produces a measurable air emission from evaporation of the water is tritium (^{3}H) in the form of HTO.

4.2.1.1.5   Drillbacks

Following underground nuclear tests, core samples have been taken from the cavity formed by the nuclear 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

The Reynolds Electrical & Engineering Company (REECo) conducted radiological analyses in Building 650, and Los Alamos National Laboratory (LANL) conducted similar analyses in Building 701 at Mercury. Because these facilities have processed primarily environmental samples, very little radioactivity has passed through them. However, there was potential for some 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 containing 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 any waste). Disposal occurs in pits and trenches; concrete pads provide temporary storage of certain wastes. Area 5 is for packaged wastes disposal only. The Waste Examination Facility (WEF) houses a HEPA-filtered glovebox used to examine and repack TRU waste drums. No contamination has been released from glove box operations to the environment. The drums, sent to the NTS from LLNL in years past, are stored inside the TRU Pad Cover Building. Repacked drums will be sent to the WIPP. The facility is a diffuse source of radiological effluents. The only radioactive effluent detected by the various types of samplers surrounding 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 above-background levels of plutonium being detected in air samples collected nearby.

4.2.1.2   Atmospheric Radionuclide Concentrations

In 1964, REECo established an environmental surveillance program at NTS designed to measure radiological conditions throughout the site without regard to nuclear tests. That is, the data collected by the program was 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.

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 determined by calibrated rotometers. Typical flow rates varied from 3 to 6 cubic feet per minute; samples were collected weekly.

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 of 1965, the 8- by 10-in. Gelman filters were replaced with a new sampler that used 4-in.-diameter Whatman #41 filter paper. The sampling rate of these samplers was about 3 cubic feet per minute. 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 exist 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 purposes of 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 having 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}.

Gross gamma screening was typically performed on samples prepared for gross beta counting. The screening was performed using a 5- by 5-in. NaI(Tl) detector coupled to a single-channel analyzer. Any samples showing elevated gamma readings were sent to a multichannel analyzer for radionuclide identification. 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}. 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, followed by alpha spectroscopy using solid-state surface barrier detectors. 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.

In 1977, a separate sampler was designed for the collection of airborne tritium (HT) and tritiated water vapor (HTO). The sampler was portable and capable of unattended operation for up to 2 weeks in desert areas. A small electronic pump drew air into the sampler at about 0.5 liter per minute, and the HTO was removed from the air stream by a silica gel drying column. The dry air then passed through a catalytic converter containing 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 allowed for the determination of the HT and HTO activities. The typical minimum detection limit for this analysis was 3×10^{-7} µCi m^{-3}.

The total number of air sampling stations increased over the years to a peak of 52 in 1989. This number remained fairly constant until a gradual reduction began in 1995. 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). Figure 4.2.1.2-1 shows the locations of the 48 sampling stations in 1997.
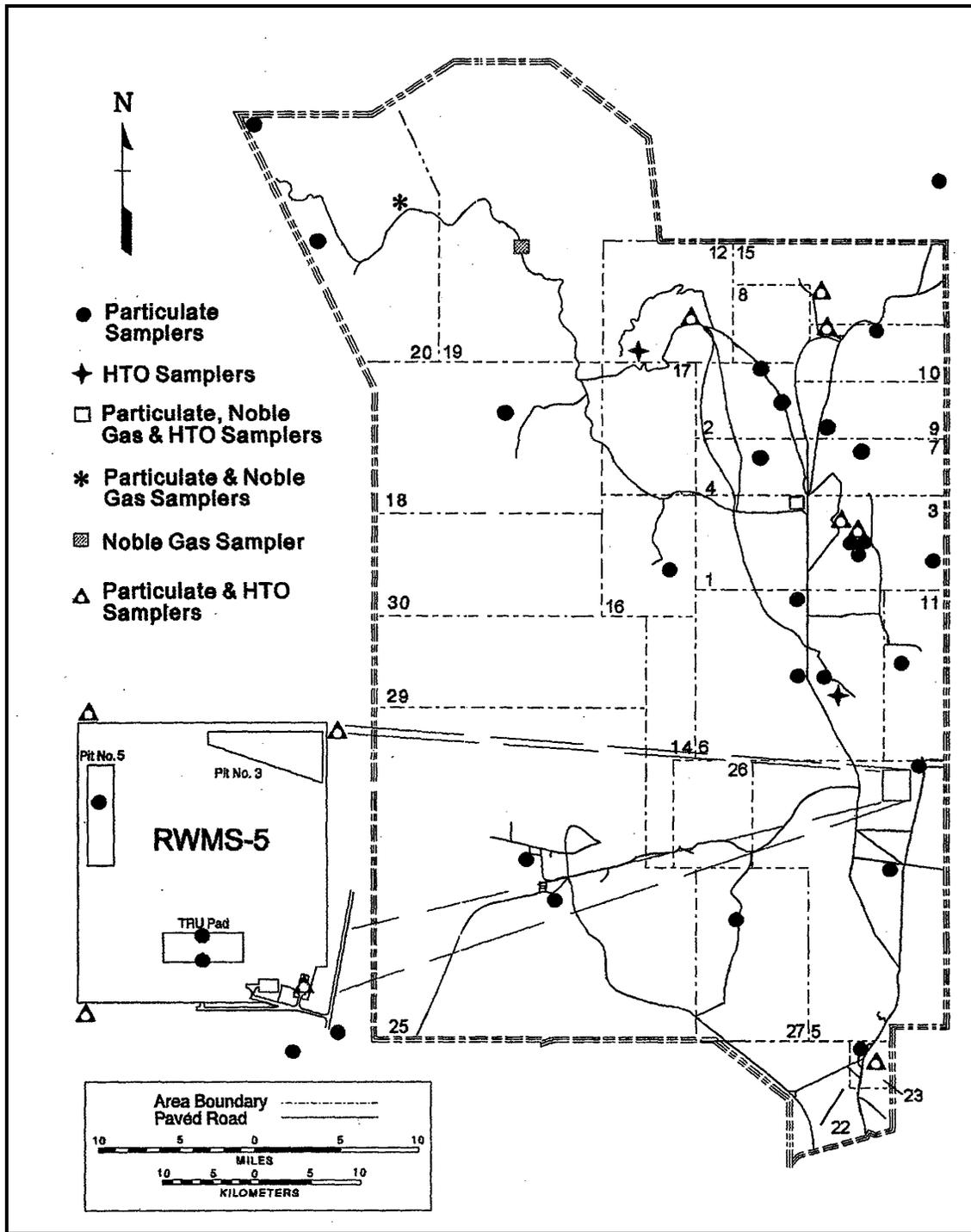
Figure 4.2.1.2-1. Air sampling stations in 1997.

Since the early 1970s, the environmental surveillance program has routinely monitored atmospheric concentrations of tritium and ^{238}Pu, ^{239}Pu, and ^{240}Pu. These were the radionuclides considered most important to dose for workers and members of the public. In addition, since the mid-1960s, measurements have been reported for gross alpha and gross beta concentrations. The following sections discuss the history of these measurements in NTS annual environmental reports (see

reference section for a listing) and their importance to dose reconstruction for unmonitored employees.

4.2.1.2.1 Tritium

Atmospheric measurements at NTS began in 1977 with samplers in Areas 5, 10, and 23. The sampler near Building 650 (Mercury) was designated as the control station. During that year, the samples in Area 10 (near the Sedan Crater) demonstrated the highest tritium (HTO) concentrations with a high of 3.0×10^{-4} pCi m^{-3}. The number of sampling locations increased over the following years to include all the areas listed in Table 4.2.1.2.1-1. These locations provided representative samples from the most populated areas on the site.

The tritium concentrations in Table 4.2.1.2.1-1 are typically the average of the maximum concentrations for a given area in a given year, except when these values were determined to be adversely affected by nearby point-source releases (e.g., Building 790 in 1983; Scoggins 1983, p. 29). In cases where maximum values were not provided, the average concentrations were reported.

In addition to the annual area tritium concentrations, Table 4.2.1.2.1-1 provides 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 purposes of dose reconstruction, these average and maximum site concentrations have been converted to annual organ dose using the the dose conversion factor for inhalation of 1.80×10^{-11} Sv Bq^{-1} (ICRP 1998) and by assuming submersion in the respective concentrations for 2,000 hr yr^{-1} and an inhalation rate of 2,400 m^3 yr^{-1}. 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 associated with the values in Table 4.2.1.2.1-1 were not provided in the annual environmental report until 2001. Based on the values provided 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 claimant-favorable assumption is made that the 50th-percentile expected intakes are those in Table 4.2.1.2.1-1 and that the 95th-percentile values would be enveloped by a factor of ±2 (i.e., ±100%).

For purposes of dose reconstruction, the geometric deviation (GSD) of the values in Table 4.2.1.2.2-1 would be estimated by using the following expression for a log-normal distribution:

$$ GSD = \left(\frac{95th\ percentile}{50th\ percentile} \right)^{\left(\frac{1}{1.65585} \right)} \tag{4-1} $$

If the assumption is made that the 95th-percentile values would be included in a 2 sigma of ±100%, then the geometric standard deviation can be shown using equation 4-1 to be 1.52.

If the claimant-favorable assumption is used to estimate the 95th-percentile dose (i.e., add 100%) from the maximum annual doses in Table 4.2.1.2.1-1, the resultant annual organ doses would still be less than 2 mrem yr^{-1}. For purposes of dose reconstruction, these small doses would be inconsequential to estimates of causation and should therefore not be included in dose reconstructions of unmonitored employees. Also, because these maximum annual dose estimates were based on empirical measurements at locations of known sources of tritium emission, it is considered unlikely that average ambient atmospheric concentrations of tritium prior to 1977 (when

data are not readily available), dose reconstructors should not assign missed environmental tritium dose for any unmonitored covered employee.

1

Table 4.2.1.2.1-1. Historical HTO atmospheric concentrations (pCi ml^{-1}) by year and Area with estimated maximum and average annual organ dose.

	Area at NTS												Site average (pCi ml^{-1})	Site maximum (pCi ml^{-1})	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 [a]	2.9E-03 [a]	6.7E-01 [a]	1.4E-01 [a]
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																
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. Value not provided in annual reports - assumed same as for 1978.

1

4.2.1.2.2 Plutonium

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. These locations were chosen to provide representative samples from the populated areas on the site as well as to monitor resuspension of low-fired plutonium spread by safety experiments before 1960 in Areas 2, 3, 4, 7, 9, and 10.

The ^{239}Pu and ^{238}Pu concentrations in Tables 4.2.1.2.2-1 and 4.2.1.2.2-3, respectively, are typically the average of the maximum concentrations for a given area in a given year. In cases where maximum values were not provided, the average of the average concentrations was reported. In addition to the annual area concentrations, Tables 4.2.1.2.2-1 and 4.2.1.2.2-3 provide NTS site average and maximum concentrations, which represent the arithmetic averages of the concentration of all the areas and the maximum 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^{-1} with a breathing rate of 2,400 m^3 yr^{-1}. Tables 4.2.1.2.2-2 and 4.2.1.2.2-4 present these estimated intakes.

It should also be noted that some covered employees remained onsite continuously for weeks at a time. However, because most of the non-working hours were spent indoors where ambient air particulate loadings would be greatly reduced from the outdoor loadings and because of the conservative assumptions used to estimate the values in Tables 4.2.1.2.2-2 and 4.2.1.2.2-4, adjustment of the tabular data is not required to assure intakes are not underestimated for these individuals.

Uncertainty data presented in NTS annual environmental reports show that, in most instances, the standard deviation was less than 50% of the measured value. However, in a few cases, the standard deviation was as high as 100%.

For purposes of dose reconstruction, the GSD of the values in Table 4.2.1.2.2-2 can be estimated by using equation 4-1 (see Section 4.2.1.2.1) for a log-normal distribution. Under the claimant-favorable assumption that the 50th-percentile expected intakes are those in Table 4.2.1.2.2-2 and that the 95th-percentile values would be enveloped by a factor of ±2 (i.e., ±100%), then the GSD for the values in Tables 4.2.1.2.2-2 and 4.2.1.2.2-4 would be 1.52 for all values.

For purposes of reconstructing potential unmonitored dose, the annual intakes in Tables 4.2.1.2.2-2 and 4.2.1.2.2-4 should be used only for individuals whose employment records indicate that they spent considerable time outdoors. Therefore, these intakes do not apply to miners or tunnel workers. In addition, reasonable adjustments to these values should be made whenever information indicates that the covered employee did not spend 100% of his workday in the outdoor environment. These individuals may have administrative job categories such as typist, secretary, janitor, or procedure-writer.

In addition, dose reconstructors should examine the values in Tables 4.2.1.2.2-2 and 4.2.1.2.2-4 and compare them with the values for resuspension intakes in Tables 4.2.2-2 and 4.2.2-3 for ^{238}Pu and $^{239, 240}$Pu because the tables in this section represent actual measured data, whereas the tables in Section 4.2.2 show calculated values based on claimant-favorable assumptions. Because both sets of tables represent the effects of the same primary pathway (i.e., resuspension), it would not be appropriate to use both sets of data to reconstruct a claimant dose. Although there is fairly good agreement between the two data sets, dose reconstructors should use the higher value for a given area when assessing intakes for unmonitored employees when the covered employee's work location

is well documented. If it is not known where a covered employee worked, the site average annual intakes for ^{238}Pu and $^{239,\ 240}$Pu provided in Section 4.2.2, Table 4.2.2-3 should be used.

It is assumed that plutonium could be either S or M absorption type.

1

Table 4.2.1.2.2-1. ^{239}Pu atmospheric concentrations (pCi m^{-3}) for sampled Areas.

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		

Table 4.2.1.2.2-1 (cont.). ^{239}Pu atmospheric concentrations (pCi m^{-3}) for sampled Areas.

Year	| Area 15	16	18	19	20	23	25	27	28	Site average	Site maximum
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

Table 4.2.1.2.2-2. Annual ^{239}Pu intakes (Bq) for sampled Areas.

Year	Area											
	1	2	3	4	5	6	7	9	10	11	12
1971	0.0329	0.0588	0.0147		0.0105	0.0153		0.0641	0.0185	0.0081	0.0145
1972	0.0132	0.0199	0.0327		0.0124	0.1191		0.3813	0.0258	0.0216	0.0699
1973	0.0077	0.0198	0.0183		0.0078	0.0107		0.0764	0.0039	0.0216	0.0038
1974	0.0071	0.0060	0.0112		0.0044	0.0070		0.0188	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	0.0393		0.2827	0.0189	0.0210	0.0248
1977	0.0024	0.0058	0.0227		0.0030	0.0049		0.0225	0.0041	0.0108	0.0019
1978	0.0042	0.0118	0.0251		0.0054	0.0088		0.0482	0.0258	0.0095	0.0055
1979	0.0066	0.0413	0.0001		0.0018	0.0035	0.0044	0.0466	0.0258	0.0033	0.0021
1980	0.0020	0.0230	0.0001		0.0026	0.0043	0.0040	0.0402		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	0.0203		0.0034	0.0018	0.0034	0.0907		0.0050	0.0015
1985	0.0013	0.0051	0.0189		0.0027	0.0024	0.0024	0.1182		0.0046	0.0020
1986	0.0132	0.0055	0.0484		0.0025	0.0165	0.0043	0.0249		0.0022	0.0028
1987	0.0048	0.0020	0.2498		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	0.0760	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	0.0443	0.0038	0.0095	0.0009
1991	0.2756	0.0018	0.0340		0.0031	0.0064	0.0029	0.0311	0.0000	0.0199	0.0013
1992	0.0133	0.0098	0.2009		0.0071	0.0065	0.0347	0.0782	0.0213	0.0078	0.0034
1993	0.0180	0.0044	0.0352		0.0071	0.0080	0.0069	0.2400	0.0151	0.0373	0.0747
1994											
1995	0.0036	0.0010	0.0216		0.0014	0.0045	0.0011	0.0347	0.0033	0.0036	0.0010
1996	0.0667		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	0.0417		0.0060		0.0085	0.0010	0.0024	0.0653		0.0008	
1999				0.0052							
2000	0.0075		0.0239	0.0052	0.0018	0.0366	0.0030	0.2511			
2001	0.0404	0.0011	0.0103	0.0166	0.0061	0.0086	0.0015	0.0448	0.0008		

Table 4.2.1.2.2-2 (cont.). Annual ^{239}Pu intakes (Bq) for sampled Areas.

Year	Area									Site average	Site maximum
	15	16	18	19	20	23	25	27	28		
1971	0.0172	0.0071	0.0201		0.0083		0.0075	0.0035	0.0201	0.0201	0.0641
1972	0.0150	0.0129	0.2427		0.0309		0.0148	0.0045	0.0678	0.0678	0.3813
1973	0.0039	0.0034	0.0064		0.0050		0.0033	0.0029	0.0130	0.0130	0.0764
1974	0.0053		0.0059		0.0076	0.0076	0.0064	0.0104	0.0077	0.0077	0.0188
1975	0.0038		0.0030		0.0039	0.0051	0.0060	0.0062	0.0056	0.0056	0.0149
1976	0.0139		0.0088		0.0505	0.0179	0.0122	0.0056	0.0428	0.0428	0.2827
1977	0.0023		0.0020		0.0024	0.0029	0.0276	0.0020	0.0078	0.0078	0.0276
1978	0.0045		0.0055		0.0052	0.0062	0.0084	0.0035	0.0118	0.0118	0.0482
1979	0.0019		0.0017		0.0020	0.0016	0.0014	0.0020	0.0089	0.0089	0.0466
1980			0.0085		0.0031	0.0027	0.0019	0.0014	0.0070	0.0070	0.0402
1981	0.0023		0.0020		0.0026	0.0022	0.0022	0.0014	0.0052	0.0052	0.0286
1982	0.0015		0.0019		0.0019	0.0015			0.0045	0.0045	0.0191
1983	0.0013		0.0026		0.0025	0.0027			0.0052	0.0052	0.0190
1984	0.0025		0.0022		0.0026	0.0028	0.0043		0.0102	0.0102	0.0907
1985	0.0019		0.0031		0.0020	0.0023	0.0028		0.0116	0.0116	0.1182
1986	0.0025		0.0020		0.0058	0.0018	0.0020		0.0092	0.0092	0.0484
1987	0.0012		0.0015		0.0014	0.0020			0.0203	0.0203	0.2498
1988			0.0000		0.0031	0.0013	0.0013		0.0036	0.0036	0.0190
1989	0.0010		0.0008		0.0104	0.0008	0.0004		0.0213	0.0213	0.0760
1990	0.0005		0.0003		0.0075	0.0010	0.0031		0.0082	0.0082	0.0443
1991	0.0026		0.0008		0.0012	0.0023	0.0007		0.0247	0.0247	0.2756
1992	0.0017		0.0008		0.0011	0.0016	0.0142		0.0313	0.0313	0.2009
1993	0.0012		0.0007		0.0006	0.0003	0.0010		0.0300	0.0300	0.2400
1994											
1995	0.0012		0.0001		0.0007	0.0004			0.0061	0.0061	0.0347
1996	0.0002	0.0003		0.0004	0.0006	0.0004	0.0002		0.0140	0.0140	0.0667
1997	0.0000	0.0018		0.0018	0.0001	0.0001	0.0001		0.0010	0.0010	0.0026
1998		0.0027		0.0029	0.0010	0.0006			0.0115	0.0115	0.0653
1999		0.0010		0.0018		0.0006			0.0019	0.0019	0.0052
2000		0.0009		0.0006		0.0010			0.0323	0.0323	0.2511
2001	0.0008	0.0015		0.0007	0.0002	0.0004			0.0090	0.0090	0.0448

Table 4.2.1.2.2-3. ^{238}Pu atmospheric concentrations (pCi m^{-3}) for sampled Areas.

Year	Area 1	2	3	4	5	6	7	9	10	11	12
1989	2.15E-05	1.60E-05	1.69E-05		1.43E-05	1.21E-05	2.60E-05	1.60E-05	0.00E+00	1.40E-05	2.00E-05
1990	2.23E-05	2.15E-05	1.43E-05		1.26E-05	1.40E-05	1.54E-05	1.99E-05	1.75E-05	1.21E-05	1.72E-05
1991	1.33E-05	1.30E-05	1.39E-05		1.12E-05	7.77E-06	1.46E-05	2.43E-05	1.53E-05	1.44E-05	4.50E-06
1992	1.23E-05	3.95E-06	7.80E-05		8.09E-06	6.02E-06	3.50E-06	1.30E-05	5.00E-06	5.60E-06	1.40E-05
1993	5.25E-06	2.64E-05	8.51E-06		4.18E-06	3.33E-06	1.40E-05	4.80E-05	7.60E-06	7.80E-06	9.70E-06
1994	1.23E-04	4.48E-06	1.08E-05		2.18E-06	1.92E-06	1.90E-06	2.60E-05	2.40E-06	3.50E-06	1.20E-06
1995	5.45E-06	1.90E-06	4.08E-06		1.51E-06	3.37E-06	8.10E-06	4.60E-06	1.30E-06	5.30E-07	9.50E-07
1996	7.60E-06	1.49E-06	8.07E-06		1.86E-06	1.52E-06	5.00E-06	9.20E-06	9.50E-07	2.50E-07	1.00E-06
1997	7.90E-07	4.00E-07	3.75E-07	6.90E-06	1.03E-07	4.47E-07	7.60E-08	4.20E-06	1.15E-06	3.90E-08	6.00E-07
1998	1.70E-06	2.85E-07	1.25E-06		3.52E-07	1.40E-07		3.30E-06	3.20E-06	2.90E-07	
1999	2.70E-07	2.60E-07	7.95E-07	8.30E-06		2.30E-07		1.40E-05	3.50E-06		
2000	9.89E-06	9.65E-06	8.11E-06	2.91E-05	8.47E-06	9.05E-06	5.41E-06	4.67E-05	1.27E-05		
2001	1.12E-05	5.54E-06	1.11E-05	4.62E-05	5.96E-06	9.15E-06	4.78E-06	8.36E-06	1.40E-05		

Table 4.2.1.2.2-3 (cont.). ^{238}Pu atmospheric concentrations (pCi m^{-3}) for areas sampled.

Year	Area 13	15	16	18	19	20	23	25	27	Site average	Site maximum
1989		2.07E-05	3.90E-05		1.75E-05	1.80E-05	1.55E-05	2.90E-05	2.20E-05	1.9E-05	3.9E-05
1990		1.36E-05	1.13E-05		1.80E-05	1.02E-05	1.63E-05	1.39E-05	4.62E-06	1.5E-05	2.2E-05
1991		1.09E-05	1.17E-05		1.77E-05	1.41E-05	1.27E-05	1.45E-05	8.04E-06	1.3E-05	2.4E-05
1992		1.50E-05			1.90E-06	6.80E-06	1.12E-06	2.95E-06	9.60E-06	1.2E-05	7.8E-05
1993		7.20E-06	7.70E-07		1.90E-06	1.10E-06	1.24E-06	3.05E-06	1.70E-06	8.9E-06	4.8E-05
1994		8.80E-06	9.60E-07							1.6E-05	1.2E-04
1995		5.40E-06			2.75E-07	3.50E-06	3.53E-07	5.10E-07		2.8E-06	8.1E-06
1996	3.10E-06	4.70E-06	2.10E-07	1.90E-07		3.90E-06	1.45E-07	7.80E-07	1.10E-06	2.8E-06	9.2E-06
1997	7.70E-07	3.00E-07	1.50E-08	1.50E-07			1.90E-08	6.80E-08		9.6E-07	6.9E-06
1998	4.10E-07	2.50E-07						1.90E-07		1.0E-06	3.3E-06
1999	1.93E-06									3.7E-06	1.4E-05
2000		5.64E-06		6.62E-06		5.66E-06		1.09E-05		1.3E-05	4.7E-05
2001		7.59E-06	2.12E-06	1.09E-05		7.58E-06	2.58E-06	4.77E-06		1.0E-05	4.6E-05

Table 4.2.1.2.2-4. ^{238}Pu intakes (Bq) for sampled Areas.

Year	Area										
	1	2	3	4	5	6	7	9	10	11	12
1989	1.9E-03	1.4E-03	1.5E-03		1.3E-03	1.1E-03	2.3E-03	1.4E-03		1.2E-03	1.8E-03
1990	2.0E-03	1.9E-03	1.3E-03		1.1E-03	1.2E-03	1.4E-03	1.8E-03	1.6E-03	1.1E-03	1.5E-03
1991	1.2E-03	1.2E-03	1.2E-03		1.0E-03	6.9E-04	1.3E-03	2.2E-03	1.4E-03	1.3E-03	4.0E-04
1992	1.1E-03	3.5E-04	6.9E-03		7.2E-04	5.4E-04	3.1E-04	1.2E-03	4.4E-04	5.0E-04	1.2E-03
1993	4.7E-04	2.3E-03	7.6E-04		3.7E-04	3.0E-04	1.2E-03	4.3E-03	6.8E-04	6.9E-04	8.6E-04
1994	1.1E-02	4.0E-04	9.6E-04		1.9E-04	1.7E-04	1.7E-04	2.3E-03	2.1E-04	3.1E-04	1.1E-04
1995	4.8E-04	1.7E-04	3.6E-04		1.3E-04	3.0E-04	7.2E-04	4.1E-04	1.2E-04	4.7E-05	8.4E-05
1996	6.8E-04	1.3E-04	7.2E-04		1.7E-04	1.4E-04	4.4E-04	8.2E-04	8.4E-05	2.2E-05	8.9E-05
1997	7.0E-05	3.6E-05	3.3E-05	6.1E-04	9.2E-06	4.0E-05	6.8E-06	3.7E-04	1.0E-04	3.5E-06	5.3E-05
1998	1.5E-04	2.5E-05	1.1E-04		3.1E-05	1.2E-05		2.9E-04	2.8E-04	2.6E-05	
1999	2.4E-05	2.3E-05	7.1E-05	7.4E-04		2.0E-05		1.2E-03	3.1E-04		
2000	8.8E-04	8.6E-04	7.2E-04	2.6E-03	7.5E-04	8.0E-04	4.8E-04	4.2E-03	1.1E-03		
2001	1.0E-03	4.9E-04	9.9E-04	4.1E-03	5.3E-04	8.1E-04	4.2E-04	7.4E-04	1.2E-03		

Table 4.2.1.2.2-4 (cont.). ^{238}Pu intakes (Bq) for sampled Areas.

Year	Area									Site average	Site maximum
	13	15	16	18	19	20	23	25	27		
1989		1.8E-03	3.5E-03		1.6E-03	1.6E-03	1.4E-03	2.6E-03	2.0E-03	1.7E-03	3.5E-03
1990		1.2E-03	1.0E-03		1.6E-03	9.1E-04	1.4E-03	1.2E-03	4.1E-04	1.3E-03	2.0E-03
1991		9.7E-04	1.0E-03		1.6E-03	1.3E-03	1.1E-03	1.3E-03	7.1E-04	1.2E-03	2.2E-03
1992		1.3E-03			1.7E-04	6.0E-04	1.0E-04	2.6E-04	8.5E-04	1.0E-03	6.9E-03
1993		6.4E-04	6.8E-05		1.7E-04	9.8E-05	1.1E-04	2.7E-04	1.5E-04	7.9E-04	4.3E-03
1994		7.8E-04	8.5E-05							1.4E-03	1.1E-02
1995		4.8E-04			2.4E-05	3.1E-04	3.1E-05	4.5E-05		2.5E-04	7.2E-04
1996	2.8E-04	4.2E-04	1.9E-05	1.7E-05		3.5E-04	1.3E-05	6.9E-05	9.8E-05	2.5E-04	8.2E-04
1997	6.8E-05	2.7E-05	1.3E-06	1.3E-05		0.0E+00	1.7E-06	6.0E-06		8.1E-05	6.1E-04
1998	3.6E-05	2.2E-05						1.7E-05		9.2E-05	2.9E-04
1999	1.7E-04									3.3E-04	1.2E-03
2000	0.0E+00	5.0E-04		5.9E-04		5.0E-04		9.7E-04		1.1E-03	4.2E-03
2001	0.0E+00	6.7E-04	1.9E-04	9.7E-04		6.7E-04	2.3E-04	4.2E-04		8.4E-04	4.1E-03

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4.2.1.2.3   Gross Alpha

As discussed in Section 4.2.1.2, measurement of gross alpha in NTS atmosphere started in 1964 at 12 locations. However, after a 5-day waiting period for decay of naturally occurring radon-thoron daughter products, long-lived alpha activity was infrequently detected. For this reason, and because of the high degree of uncertainty associated with alpha results, statistical summaries of the data were considered meaningless and therefore not reported in the annual environmental reports until 1971. In addition, the slight increases identified during these early years correlated well with fallout from foreign weapon tests and were therefore not attributed to NTS activities.

In 1970, REECo stated that plutonium results for air samples were uniformly low (i.e., on the order of 1 pCi m^{-3}). However, in the same report, REECo identified the area adjacent to the 9-300 crater (located in Area 9) as a source of slightly elevated, and rising, plutonium atmospheric concentration due to known plutonium fields that resulted from safety tests in the area before 1960.

Isotopic analysis of plutonium began in 1970 and, although data were provided for ^{239}Pu, the annual environmental reports did not provide gross alpha data from 1974 to 1996. Because of the correlation of positive alpha activity with foreign weapon tests and the high statistical uncertainty associated with the early data, dose reconstructors should not use the data in Tables 4.2.1.2.3-1 and 4.2.1.2.3-2 to adjust environmental dose. These data are provided only for completeness.

4.2.1.2.4   Gross Beta

As discussed in Section 4.2.1.2, measurement of gross beta in the NTS atmosphere started in 1964 at 12 locations. The gross beta concentrations in Table 4.2.1.2.4-1 are typically the average of the maximum concentrations for a given area in a given year. In cases where maximum values were not provided, the average of the average concentrations was reported. In addition to the annual area concentrations, Table 4.2.1.2.4-1 provides site average and maximum concentrations, which represent the arithmetic averages and maximums, respectively, of the concentrations of all site areas. Potential intakes associated with these concentrations can be calculated under the assumption that an unmonitored worker was occupationally exposed for 2,000 hr yr^{-1} with a breathing rate of 2,400 m^3 yr^{-1}. Table 4.2.1.2.4-2 presents these intakes.

For purposes of dose reconstruction, the GSD of the values in Table 4.2.1.2.4-2 can be estimated using equation 4-1 for a log-normal distribution. Uncertainty data in NTS annual environmental reports show that, in most instances, the standard deviation was less than 50% of the measured value. However, in a few cases, the standard deviation was as high as 100%.

Under the claimant-favorable assumption that the 50th-percentile expected intakes are those in Table 4.2.1.2.3-2 and that the 95th-percentile values would be enveloped by a factor of ±2 (i.e., ±100%), then the GSD for the values in Tables 4.2.1.2.4-2 and 4.2.1.2.4-4 is 1.52 for all values.

For purposes of reconstructing potential unmonitored dose, the annual intakes in Table 4.2.1.2.4-2 should be used only for individuals whose employment records indicate that they spent considerable time outdoors. Therefore, these intakes do not apply to miners or tunnel workers.

In addition, dose reconstructors should examine the values in Table 4.2.1.2.4-2 and compare them with the values for resuspension intakes in Tables 4.2.2-2 and 4.2.2-3 for ^{90}Sr (the unidentified radionuclide with the highest internal dose impact [i.e., the highest CEDE of all NTS beta emitters]). However, gross beta intakes in Table 4.2.1.2.4-2 during the early years (1967 through 1970) should

not be used for adjusting internal dose because of the well-documented correlation of these elevated data with foreign weapons testing.

Beginning in the early 1970s, the average and maximum annual intakes for ^{90}Sr based on the measured data in Table 4.2.1.2.4-2 show good agreement with the calculated values for ^{90}Sr in Table 4.2.2-3. However, because the tables in this section show actual measured data, whereas the tables in Section 4.2.2 show calculated values based on claimant-favorable assumptions (with the exception of the data before 1970) the annual intakes in Table 4.2.1.2.4-2 should be used to adjust internal dose to ^{90}Sr for unmonitored employees.

Table 4.2.1.2.3-1. Gross alpha atmospheric concentrations (pCi m^{-3}) for sampled Areas.

Year	Area

Year	1	2	3	4	5	6	7	9	10	11	12	13
1971	3.3E-01	3.7E-01			3.3E-01	4.5E-01		3.5E-01	2.3E-01	2.9E-01	2.7E-01	
1972	2.3E-01	2.0E-01	2.0E-01		1.8E-01	4.4E-01		3.7E-01	1.6E-01	1.8E-01	2.4E-01	
1973	3.9E-02	4.1E-02	3.9E-02		3.5E-02	4.1E-02		1.2E-01	9.6E-02	3.7E-02	4.2E-02	
1996	3.1E-03	4.0E-03	4.4E-03	3.3E-03	3.4E-03	3.2E-03	5.0E-03	6.5E-03	3.2E-03	3.4E-03	2.8E-03	4.8E-03
1997	1.5E-03	1.6E-03	1.9E-03	1.6E-03	1.8E-03	1.7E-03	1.5E-03	2.2E-03	1.6E-03	1.4E-03	1.1E-03	2.0E-03
1998	1.9E-03	1.6E-03	1.8E-03	1.5E-03	2.0E-03	1.8E-01	1.6E-03	1.9E-03	1.7E-03	1.5E-03		1.6E-03
1999	2.4E-03	2.3E-03	2.5E-03	2.4E-03	2.3E-03	2.2E-03	2.0E-03	4.6E-03	2.4E-03			2.1E-03
2000	1.5E-02	1.4E-02	2.6E-01	1.5E-02	1.5E-02	1.7E-02	1.6E-02	1.2E-01	1.5E-02			
2001	1.9E-02	1.4E-02	2.4E-02	1.6E-02	1.4E-02	1.9E-02	1.6E-02	1.4E-02	1.4E-02			

Table 4.2.1.2.3-1 (cont.). Gross alpha atmospheric concentrations (pCi m^{-3}) for sampled Areas.

Year	Area									Site average	Site maximum
Year	15	16	18	19	20	23	25	27	28		
1971		2.9E-01	2.9E-01	3.1E-01	2.7E-01	4.1E-01		2.9E-01	4.5E-01	2.3E-01	4.5E-01
1972		6.7E-01	1.8E-01	3.8E-01	2.8E+00	1.9E-01		2.3E-01	2.7E-01	3.3E-01	2.8E+00
1973		3.4E-02	5.3E-02	1.3E-01	4.0E-02	4.5E-02		3.0E-02	4.7E-02	4.1E-02	1.3E-01
1996	5.4E-03	4.3E-03	3.4E-03		3.0E-03	3.2E-03	4.2E-03	3.2E-03		3.5E-03	6.5E-03
1997	1.5E-03	1.4E-03	1.5E-03		1.5E-03	1.6E-03	1.7E-03	1.3E-03		1.4E-03	2.2E-03
1998	1.6E-03		1.8E-03		1.7E-03	1.8E-03	1.6E-03			9.6E-03	1.8E-01
1999	2.5E-03		2.1E-03		2.1E-03		2.3E-03			1.6E-03	4.6E-03
2000	1.5E-02		1.7E-02		1.8E-02		1.5E-02			2.6E-02	2.6E-01
2001	1.8E-02	1.5E-02	1.6E-02		1.3E-02	1.6E-02	1.5E-02			1.2E-02	2.4E-02

Table 4.2.1.2.3-2. Gross alpha annual intakes (Bq) for sampled Areas.

Year	Area											
	1	2	3	4	5	6	7	9	10	11	12	13
1971	3.0E+01	3.3E+01			2.9E+01	4.0E+01		3.1E+01	2.0E+01	2.5E+01	2.4E+01	
1972	2.0E+01	1.8E+01	1.8E+01		1.6E+01	3.9E+01		3.3E+01	1.4E+01	1.6E+01	2.2E+01	
1973	3.5E+00	3.6E+00	3.5E+00		3.1E+00	3.6E+00		1.1E+01	8.5E+00	3.3E+00	3.7E+00	
1996	2.8E-01	3.6E-01	3.9E-01	2.9E-01	3.0E-01	2.8E-01	4.4E-01	5.8E-01	2.8E-01	3.0E-01	2.5E-01	4.3E-01
1997	1.3E-01	1.4E-01	1.7E-01	1.4E-01	1.6E-01	1.5E-01	1.3E-01	2.0E-01	1.4E-01	1.2E-01	9.8E-02	1.8E-01
1998	1.7E-01	1.4E-01	1.6E-01	1.3E-01	1.8E-01	1.6E+01	1.4E-01	1.7E-01	1.5E-01	1.3E-01		1.4E-01
1999	2.1E-01	2.0E-01	2.2E-01	2.1E-01	2.0E-01	2.0E-01	1.8E-01	4.1E-01	2.1E-01			1.9E-01
2000	1.4E+00	1.2E+00	2.3E+01	1.3E+00	1.4E+00	1.5E+00	1.4E+00	1.0E+01	1.4E+00			
2001	1.7E+00	1.3E+00	2.2E+00	1.4E+00	1.2E+00	1.7E+00	1.4E+00	1.3E+00	1.2E+00			

Table 4.2.1.2.3-2 (cont.). Gross alpha annual intakes (Bq) for sampled Areas.

Year	Area									Site average	Site maximum
	15	16	18	19	20	23	25	27	28		
1971		2.5E+01	2.6E+01	2.7E+01	2.4E+01	3.6E+01		2.5E+01	4.0E+01	2.1E+01	4.0E+01
1972		5.9E+01	1.6E+01	3.4E+01	2.5E+02	1.7E+01		2.0E+01	2.4E+01	2.9E+01	2.5E+02
1973		3.0E+00	4.7E+00	1.1E+01	3.6E+00	4.0E+00		2.7E+00	4.2E+00	3.7E+00	1.1E+01
1996	4.8E-01	3.8E-01	3.0E-01		2.6E-01	2.8E-01	3.7E-01	2.8E-01		3.1E-01	5.8E-01
1997	1.3E-01	1.2E-01	1.3E-01		1.3E-01	1.4E-01	1.5E-01	1.2E-01		1.3E-01	2.0E-01
1998	1.4E-01		1.6E-01		1.5E-01	1.6E-01	1.4E-01			8.6E-01	1.6E+01
1999	2.2E-01		1.9E-01		1.9E-01		2.0E-01			1.4E-01	4.1E-01
2000	1.3E+00		1.5E+00		1.6E+00		1.4E+00			2.3E+00	2.3E+01
2001	1.6E+00	1.3E+00	1.4E+00		1.1E+00	1.4E+00	1.4E+00			1.0E+00	2.2E+00

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Table 4.2.1.2.4-1. Gross beta atmospheric concentrations (pCi m^{-3}) for sampled Areas.

Year	Area 1	2	3	4	5	6	7	9	10	11	12
1966			2.1E+00		3.2E+00	1.7E+00		2.2E+00			
1967	2.3E-01	3.4E+00	3.7E+00		2.2E+00	3.3E+00		3.0E+00	3.1E+00	2.5E+00	2.6E+00
1968	5.5E+01	4.7E-01	5.1E-01		7.5E-01	1.9E+00		7.1E+00	1.2E+00	6.3E-01	1.2E+00
1969	4.1E+01	1.1E+01	2.1E+01		4.0E+00	1.0E+00		1.3E+01	3.7E+00		1.2E+01
1970	8.7E-01	6.7E-01	7.1E-01		8.7E-01			8.4E-01	7.5E-01	7.8E-01	8.3E-01
1971											
1972											
1973											
1974	1.3E+01	1.1E-02	9.7E-02		9.7E-02	1.1E-01		1.2E-01	9.6E-02	9.6E-02	1.3E-01
1975	9.1E-02	8.3E-02	9.4E-02		1.1E-01	1.0E-01		1.0E-01	8.5E-02	8.9E-02	1.1E-01
1976	9.3E-02	3.0E-02	3.8E-02		6.3E-02	4.9E-02		5.7E-01	1.8E-01	9.5E-02	4.0E-02
1977	1.7E-01	1.7E+00	1.6E-01		2.0E-01	2.6E-01		1.6E-01	1.2E-01	1.6E-01	1.6E-01
1978	1.1E-01	6.8E-02	8.9E-02		1.0E+01	9.8E-02		1.0E-01		9.7E-02	1.0E-01
1979	3.7E-02	3.5E-02	3.3E-02		3.6E-02	3.5E-02	3.1E-02	3.4E-02		3.4E-05	3.3E-02
1980	3.6E-02	4.0E-02	4.4E-02		3.8E-02	3.7E-02	2.4E-02	3.8E-02		3.9E-02	3.4E-02
1981	1.5E-01	1.6E-01	1.6E-01		1.6E-01	1.6E-01	1.7E-01	1.6E-01		1.7E-01	1.6E-01
1982	2.2E-02	2.2E-02	2.3E-02		2.4E-02	2.3E-02	2.4E-02	2.3E-02		1.7E-02	2.4E-02
1983	1.8E-02	1.8E-02	1.8E-02		1.9E-02	1.8E-02	1.7E-02	1.8E-02		1.9E-02	1.6E-02
1984	1.8E-02	1.8E-02	1.9E-02		2.2E-02	1.9E-02	1.8E-02	1.8E-02		1.7E-02	1.7E-02
1985	1.7E-02	1.6E-02	1.7E-02		2.1E-02	1.6E-02	1.6E-02	1.6E-02		1.7E-02	1.5E-02
1986	4.6E-02	5.2E-02	4.8E-02		5.6E-02	4.6E-02	4.7E-02	5.1E-02		4.8E-02	4.9E-02
1987	3.6E-02	3.9E-02	2.8E-02		2.1E-02	2.6E-02	2.4E-02	5.3E-02		1.9E-02	2.4E-02
1988	2.0E-02	2.6E-02	1.9E-02		2.2E-02	1.8E-02	1.8E-02	1.8E-02		1.9E-02	1.9E-02
1989	3.8E-02	4.5E-02	3.6E-02		1.9E-01	3.2E-02	5.8E-01	6.7E-02		3.3E-02	8.7E-02
1990	3.4E-02	3.2E-02	4.1E-02		4.2E-02	5.3E-02	4.5E-02	3.7E-02	3.3E-02	4.0E-02	
1991	3.4E-02	7.7E-02	3.4E-02		4.2E-02	3.5E-02	3.4E-02	4.0E-02	3.3E-02	3.2E-02	4.5E-02
1992	3.8E-02	7.8E-02	4.6E-02		4.2E-02	3.7E-02	4.0E-02	4.6E-02	3.8E-02	3.8E-02	2.2E-01
1993	5.5E-02	3.5E-02	3.7E-02		4.3E-02	4.7E-02	3.3E-02	3.6E-02	3.4E-02	3.5E-02	3.2E-02
1994											
1995	4.0E-02	4.3E-02	3.8E-02		4.0E-02	3.8E-02	4.5E-02	5.1E-02	4.6E-02		3.7E-02
1996			3.3E-02		3.5E-02	3.6E-02	3.5E-02	3.4E-02	3.4E-02	5.3E-02	2.9E-02
1997	1.9E-02	2.0E-02	1.9E-02	1.9E-02	2.2E-02	1.4E-02	1.8E-02	1.9E-02	2.0E-02	1.8E-02	1.6E-02
1998	2.0E-02	1.7E-02	2.0E-02		2.0E-02	1.8E-02	1.9E-02	1.6E-02	1.8E-02	1.8E-02	
1999	2.2E-02	2.2E-02	2.1E-02	2.3E-02	2.1E-02	2.2E-02	2.1E-02	2.1E-02	2.2E-02		
2000	5.1E-02	5.5E-02	5.4E-02	4.8E-02	3.7E-02	4.7E-02	4.0E-02	5.7E-02			
2001	4.1E-02	4.1E-02	4.2E-02	2.7E-02	4.2E-02	3.9E-02	4.3E-02	3.9E-02	3.9E-02		

Table 4.2.1.2.4-1 (cont.). Gross beta atmospheric concentrations (pCi m^{-3}) for sampled Areas.

Year	15	16	18	19	20	23	25	27	28	Site average	Site maximum
					Area						
1966			2.2E+00		2.0E+00			1.5E+00		7.4E-01	3.2E+00
1967		7.8E+00	9.1E-01	4.8E+00	2.7E+00			2.6E+00	2.6E+01	3.4E+00	2.6E+01
1968		1.3E+00	8.9E-01	2.1E+00	3.5E+02	1.4E+00		6.9E-01	1.8E+00	2.1E+01	3.5E+02
1969		2.1E+01		1.6E+01	1.1E+00	1.1E+00		1.6E+00	1.8E+00	7.5E+00	4.1E+01
1970		7.5E-01	7.3E-01	7.6E-01	7.3E-01	1.1E+00		8.0E-01	9.5E-01	6.1E-01	1.1E+00
1971											
1972											
1973											
1974		1.6E-01		7.8E-02	4.9E-02	8.3E-01	2.0E-01	9.9E-02	1.3E-01	7.5E-01	1.3E+01
1975		9.2E-02		9.4E-02	9.1E-02	1.3E-01	9.2E-02	9.3E-02	1.2E-01	7.9E-02	1.3E-01
1976		9.9E-02		4.9E-02	3.2E-02	1.0E-01	5.1E-02	2.8E-02	3.4E-02	7.8E-02	5.7E-01
1977		1.9E-01		1.7E-01	2.0E-01	5.2E-02	1.7E-01	1.4E-01	1.9E-01	2.0E-01	1.7E+00
1978		9.3E-02		9.2E-02		1.7E-01	8.9E-02	1.0E-01	1.1E-01	5.9E-01	1.0E+01
1979	2.4E-02	3.3E-02		3.2E-02		5.9E-02	3.5E-02	3.3E-02	3.5E-02	2.6E-02	5.9E-02
1980	3.8E-02			6.0E-02	6.6E-02	3.4E-02	3.8E-02		3.7E-02	3.0E-02	6.6E-02
1981	1.6E-01			1.5E-01	1.5E-01	3.9E-02	1.7E-01	1.7E-01	1.7E-01	1.2E-01	1.7E-01
1982	2.3E-02	2.2E-02		2.0E-02	2.0E-02	1.6E-01	2.2E-02	2.2E-02	2.2E-02	2.6E-02	1.6E-01
1983	1.7E-02	1.7E-02		1.6E-02	1.6E-02	2.3E-02	1.8E-02		1.7E-02	1.4E-02	2.3E-02
1984	1.5E-02			1.6E-02	1.6E-02	1.8E-02	1.8E-02			1.2E-02	2.2E-02
1985	1.5E-02	1.5E-02		1.4E-02	1.4E-02	1.8E-02	1.6E-02			1.2E-02	2.1E-02
1986	4.8E-02	3.6E-02		5.0E-02	4.8E-02	4.7E-02	4.8E-02	4.8E-02		3.8E-02	5.6E-02
1987	2.7E-02	1.6E-02		3.6E-02	2.8E-02	2.0E-02	1.4E-02	1.6E-02		2.1E-02	5.3E-02
1988	1.9E-02			1.7E-02	1.9E-02	2.0E-02	1.9E-02	1.7E-02		1.4E-02	2.6E-02
1989	2.5E-01	1.1E-01		1.2E-01	1.4E-01	4.6E-02	3.6E-02	3.4E-02		9.2E-02	5.8E-01
1990	3.6E-02	3.4E-02		3.3E-02	3.2E-02	3.9E-02	3.7E-02	4.6E-02		3.1E-02	5.3E-02
1991	3.2E-02	3.2E-02		3.1E-02	3.2E-02	4.4E-02	3.8E-02	3.7E-02		3.3E-02	7.7E-02
1992	1.5E-01	3.4E-02		3.4E-02	3.6E-02	3.6E-02	3.9E-02	3.8E-02		4.9E-02	2.2E-01
1993	3.7E-02	3.8E-02		3.3E-02	3.2E-02	1.9E-02	3.7E-02	3.9E-02		3.1E-02	5.5E-02
1994											
1995	4.2E-02	3.7E-02	4.5E-02	2.8E-02	3.8E-02	4.0E-02	4.8E-02	3.7E-02		3.5E-02	5.1E-02
1996	3.4E-02	3.2E-02	3.3E-02		2.8E-02	3.5E-02	3.7E-02	3.3E-02		2.6E-02	5.3E-02
1997	1.8E-02	1.8E-02	1.8E-02		1.8E-02	2.7E-02	1.9E-02	1.7E-02		1.7E-02	2.7E-02
1998			1.9E-02		1.9E-02	2.0E-02	1.7E-02			1.2E-02	2.0E-02
1999	2.1E-02		2.0E-02		2.0E-02	1.9E-02	2.1E-02			1.5E-02	2.3E-02
2000	4.7E-02		7.0E-02		4.1E-02		4.8E-02			3.0E-02	7.0E-02
2001	3.5E-02	4.0E-02	3.7E-02		3.9E-02	5.4E-02	3.9E-02			3.0E-02	5.4E-02

Table 4.2.1.2.4-2. Gross beta annual intakes (Bq) for sampled Areas.

Year	Area										
	1	2	3	4	5	6	7	9	10	11	12
1966			1.8E+02		2.8E+02	1.5E+02		2.0E+02			
1967	2.0E+01	3.0E+02	3.3E+02		2.0E+02	2.9E+02		2.7E+02	2.7E+02	2.2E+02	2.3E+02
1968	4.9E+03	4.2E+01	4.5E+01		6.6E+01	1.7E+02		6.3E+02	1.1E+02	5.6E+01	1.0E+02
1969	3.6E+03	1.0E+03	1.9E+03		3.5E+02	9.1E+01		1.1E+03	3.3E+02		1.1E+03
1970	7.7E+01	6.0E+01	6.3E+01		7.8E+01			7.5E+01	6.6E+01	6.9E+01	7.3E+01
1971											
1972											
1973											
1974	1.1E+03	9.7E-01	8.6E+00		8.6E+00	9.7E+00		1.1E+01	8.5E+00	8.5E+00	1.1E+01
1975	8.0E+00	7.4E+00	8.4E+00		1.0E+01	8.9E+00		8.9E+00	7.6E+00	7.9E+00	9.8E+00
1976	8.3E+00	2.7E+00	3.4E+00		5.6E+00	4.4E+00		5.1E+01	1.6E+01	8.4E+00	3.6E+00
1977	1.5E+01	1.5E+02	1.4E+01		1.8E+01	2.3E+01		1.4E+01	1.1E+01	1.5E+01	1.4E+01
1978	9.6E+00	6.0E+00	7.9E+00		9.2E+02	8.7E+00		9.1E+00		8.6E+00	9.0E+00
1979	3.3E+00	3.1E+00	2.9E+00		3.2E+00	3.1E+00	2.8E+00	3.0E+00		3.0E-03	2.9E+00
1980	3.2E+00	3.6E+00	3.9E+00		3.4E+00	3.3E+00	2.1E+00	3.4E+00		3.5E+00	3.0E+00
1981	1.3E+01	1.4E+01	1.4E+01		1.4E+01	1.5E+01	1.5E+01	1.4E+01		1.5E+01	1.4E+01
1982	2.0E+00	2.0E+00	2.0E+00		2.2E+00	2.0E+00	2.1E+00	2.0E+00		1.5E+01	2.1E+00
1983	1.6E+00	1.6E+00	1.6E+00		1.6E+00	1.6E+00	1.5E+00	1.6E+00		1.7E+00	1.4E+00
1984	1.6E+00	1.6E+00	1.7E+00		2.0E+00	1.7E+00	1.6E+00	1.6E+00		1.5E+00	1.5E+00
1985	1.5E+00	1.4E+00	1.5E+00		1.9E+00	1.4E+00	1.4E+00	1.4E+00		1.5E+00	1.3E+00
1986	4.0E+00	4.6E+00	4.2E+00		4.9E+00	4.1E+00	4.2E+00	4.5E+00		4.3E+00	4.4E+00
1987	3.2E+00	3.5E+00	2.5E+00		1.9E+00	2.3E+00	2.1E+00	4.7E+00		1.7E+00	2.1E+00
1988	1.7E+00	2.3E+00	1.7E+00		2.0E+00	1.6E+00	1.6E+00	1.6E+00		1.7E+00	1.7E+00
1989	3.4E+00	4.0E+00	3.2E+00		1.7E+01	2.8E+00	5.2E+01	6.0E+00		2.9E+00	7.7E+00
1990	3.0E+05	2.8E+05	3.7E+05		3.7E+05	4.7E+05	4.0E+05	3.3E+05	2.9E+05	3.6E+05	
1991	3.0E+00	6.8E+00	3.1E+00		3.7E+00	3.1E+00	3.0E+00	3.6E+00	2.9E+00	2.8E+00	4.0E+00
1992	3.4E+00	6.9E+00	4.1E+00		3.7E+00	3.3E+00	3.6E+00	4.1E+00	3.4E+00	3.4E+00	2.0E+01
1993	4.8E+00	3.1E+00	3.3E+00		3.8E+00	4.2E+00	2.9E+00	3.2E+00	3.0E+00	3.1E+00	2.8E+00
1994											
1995	3.5E+00	3.8E+00	3.3E+00		3.6E+00	3.4E+00	4.0E+00	4.5E+00	4.0E+00		3.3E+00
1996			3.0E+00		3.1E+00	3.2E+00	3.1E+00	3.0E+00	3.0E+00	4.7E+00	2.6E+00
1997	1.7E+00	1.7E+00	1.7E+00	1.7E+00	1.9E+00	1.3E+00	1.6E+00	1.7E+00	1.7E+00	1.6E+00	1.4E+00
1998	1.8E+00	1.5E+00	1.8E+00		1.8E+00	1.6E+00	1.7E+00	1.4E+00	1.6E+00	1.6E+00	
1999	2.0E+00	2.0E+00	1.8E+00	2.0E+00	1.9E+00	2.0E+00	1.9E+00	1.9E+00	2.0E+00		
2000	4.5E+00	4.9E+00	4.8E+00	4.2E+00	3.3E+00	4.2E+00	3.6E+00	5.0E+00			
2001	3.6E+00	3.6E+00	3.8E+00	2.4E+00	3.8E+00	3.5E+00	3.8E+00	3.5E+00	3.5E+00		

Table 4.2.1.2.4-2 (cont.). Gross beta annual intakes (Bq) for sampled Areas.

Year	Area 15	16	18	19	20	23	25	27	28	Site average	Site maximum
1966			2.0E+02		1.7E+02			1.3E+02		6.6E+01	2.8E+02
1967		7.0E+02	8.0E+01	4.2E+02	2.4E+02			2.3E+02	2.3E+03	3.0E+02	2.3E+03
1968		1.1E+02	7.9E+01	1.9E+02	3.1E+04	1.3E+02		6.1E+01	1.6E+02	1.9E+03	3.1E+04
1969		1.9E+03		1.4E+03	9.9E+01	9.9E+01		1.4E+02	1.6E+02	6.7E+02	3.6E+03
1970		6.7E+01	6.5E+01	6.8E+01	6.5E+01	9.7E+01		7.1E+01	8.5E+01	5.4E+01	9.7E+01
1971										5.4E+01[a]	9.7E+01[a]
1972										5.4E+01[a]	9.7E+01[a]
1973										5.4E+01[a]	9.7E+01[a]
1974		1.4E+01		6.9E+00	4.4E+00	7.4E+01	1.7E+01	8.8E+00	1.2E+01	6.7E+01	1.1E+03
1975		8.2E+00		8.4E+00	8.1E+00	1.2E+01	8.2E+00	8.3E+00	1.1E+01	7.0E+00	1.2E+01
1976		8.8E+00		4.4E+00	2.8E+00	9.1E+00	4.5E+00	2.5E+00	3.0E+00	6.9E+00	5.1E+01
1977		1.7E+01		1.5E+01	1.8E+11	4.6E+00	1.5E+01	1.3E+01	1.7E+01	1.8E+01	1.5E+02
1978		8.3E+00		8.1E+00		1.5E+01	7.9E+00	9.1E+00	9.5E+00	5.2E+01	9.2E+02
1979	2.1E+00	2.9E+00		2.8E+00		5.3E+00	3.1E+00	2.9E+00	3.1E+00	2.3E+00	5.3E+00
1980	3.4E+00			5.3E+00	5.9E+00	3.0E+00	3.4E+00		3.3E+00	2.7E+00	5.9E+00
1981	1.4E+01			1.3E+01	1.3E+01	3.4E+00	1.5E+01	1.5E+01	1.5E+01	1.1E+01	1.5E+01
1982	2.0E+00	2.0E+00		1.8E+00	1.8E+00	1.5E+01	2.0E+00	2.0E+00	2.0E+00	2.3E+00	1.5E+01
1983	1.5E+00	1.5E+00		1.4E+00	1.4E+00	2.0E+00	1.6E+00		1.5E+00	1.3E+00	2.0E+00
1984	1.4E+00			1.4E+00	1.4E+00	1.6E+00	1.6E+00			1.1E+00	2.0E+00
1985	1.4E+00	1.3E+00		1.2E+00	1.2E+00	1.6E+00	1.4E+00			1.1E+00	1.9E+00
1986	4.3E+00	3.2E+00		4.4E+00	4.3E+00	4.2E+00	4.2E+00	4.3E+00		3.4E+00	4.9E+00
1987	2.4E+00	1.4E+00		3.2E+00	2.5E+00	1.8E+00	1.2E+00	1.4E+00		1.9E+00	4.7E+00
1988	1.6E+00			1.5E+00	1.7E+00	1.8E+00	1.7E+00	1.5E+00		1.3E+00	2.3E+00
1989	2.2E+01	9.8E+00		1.1E+01	1.2E+01	4.1E+00	3.2E+00	3.0E+00		8.2E+00	5.2E+01
1990	3.2E+05	3.0E+05		2.9E+05	2.8E+05	3.5E+05	3.3E+05	4.1E+05		2.7E+00	4.7E+00
1991	2.8E+00	2.8E+00		2.7E+00	2.8E+00	3.9E+00	3.3E+00	3.3E+00		2.9E+00	6.8E+00
1992	1.3E+01	3.0E+00		3.0E+00	3.2E+00	3.2E+00	3.5E+00	3.4E+00		4.4E+00	2.0E+01
1993	3.3E+00	3.4E+00		2.9E+00	2.8E+00	1.7E+00	3.3E+00	3.5E+00		2.8E+00	4.8E+00
1994										2.8E+00[b]	4.8E+00[b]
1995	3.7E+00	3.3E+00	4.0E+00	2.5E+00	3.4E+00	3.6E+00	4.3E+00	3.3E+00		3.1E+00	4.5E+00
1996	3.0E+00	2.8E+00	2.9E+00		2.5E+00	3.1E+00	3.2E+00	2.9E+00		2.3E+00	4.7E+00
1997	1.6E+00	1.6E+00	1.6E+00		1.6E+00	2.4E+00	1.6E+00	1.5E+00		1.5E+00	2.4E+00
1998			1.7E+00		1.6E+00	1.8E+00	1.5E+00			1.1E+00	1.8E+00
1999	1.9E+00		1.8E+00		1.8E+00	1.7E+00	1.9E+00			1.3E+00	2.0E+00
2000	4.2E+00		6.2E+00		3.7E+00		4.3E+00			2.6E+00	6.2E+00
2001	3.1E+00	3.6E+00	3.3E+00		3.4E+00	4.8E+00	3.4E+00	3.1E+00	3.6E+00	2.7E+00	4.8E+00

a. Value not provided in annual report - assumed same as for 1970.
b. Value not provided - assumed same as for 1993.

1

4.2.2 Annual Intakes from Resuspension

The resuspension of radionuclides after initial deposition on the ground surface is an exposure pathway that could, in certain situations, represent an important pathway to employees at NTS. Historical measurements at NTS and other locations have shown that the rate of resuspension decreases very rapidly with time, and that for conditions immediately following a test, resuspension is only of importance (compared to the inhalation exposure from the initial plume passage) over short periods. For cleanup and property-release situations, resuspension could be the dominant pathway, especially for transuranic radionuclides or for other radionuclides that do not readily cross biological barriers. Historically, the concern about resuspension has been with isotopes of plutonium, which do not readily cross biological boundaries but stay in the lung for long periods if inhaled (Anspaugh 2002, p. 675).

Airborne concentrations of radionuclides resuspended from contaminated soils can be estimated based on the following expression (Till and Meyer 1983, p. 5-30):

$$CA = RF \times CS \qquad \text{eq. (4-2)}$$

where

> CA = air concentration, Bq m^{-3}
> CS = radionuclide areal soil deposition, Bq m^{-2}
> RF = resuspension factor, m^{-1}

Extensive studies were performed in the 1980s to quantify residual contamination at NTS (McArthur 1983, 1985, 1987, 1988, 1989, 1991). Table 2-6 of the Site Description (Section 2) for this Technical Basis document lists the results of these studies. Table 4.2.2-1 lists the total areal depositions based on the inventory values in Table 2-6 (see Section 2 of this document) divided by the respective areal size.

Historically, there have been several models used to estimate the resuspension of radionuclides from contaminated soil. However, the most widely used model is the resuspension-factor model shown by equation 4-2. Efforts to estimate the resuspension factor, RF, at NTS began as early as 1957 and have culminated with a predictive model by Anspaugh (2002). This model (shown below as equation 4-3) was derived from an expanded databank of evaluated measurements (Beck 1980; Hicks 1982) and has attempted to describe empirical measurements over the entire span of the data set (i.e., early, intermediate, and late periods). The model also includes an explicit statement of uncertainty, a factor of 10, which is large but is indicative of the dispersion in the data sets. The model is as follows:

$$SF = [10^{-5} e^{(-0.07t)} + 6 x 10^{-9} e^{(-0.003t)} + 10^{-9}] x 10^{\pm 1} m^{-1} \qquad \text{eq. (4-3)}$$

Figure 4.2.2-1 shows the time-dependent nature of the resuspension factor as a function of time after initial deposition. As shown, after about 3 years the resuspension factor approaches a value of about

1

Table 4.2.2-1. Radionuclide areal soil deposition (Bq m^{-2}).

Area	Area (m^2)	Am-241	Pu-238	Pu-239,240	Co-60	Cs 137	Sr-90	Eu-152	Eu-154	Eu-155
1	6.9E+07	2.3E+03	3.5E+03	1.3E+04	5.9E+02	4.7E+03	8.1E+03	8.1E+03	5.4E+01	2.7E+02
2	5.1E+07	2.1E+03	6.2E+03	1.6E+04	8.7E+02	1.7E+04	3.3E+04	1.0E+04		2.9E+02
3	8.4E+07	2.0E+03	1.4E+03	1.6E+04	4.4E+02	5.3E+03	1.5E+04	8.0E+03	4.4E+01	2.2E+02
4	4.1E+07	5.9E+03	1.2E+04	3.6E+04	1.4E+03	1.1E+04	1.2E+04	8.1E+03		1.8E+02
5	7.5E+06	3.0E+03	4.9E+02	2.4E+04	3.0E+03	2.0E+03	4.4E+03	4.9E+04	9.9E+02	
6	8.4E+07	7.5E+02	1.5E+03	3.7E+03	8.8E+01	1.2E+03	1.5E+03			
7	5.0E+07	1.6E+03	4.4E+02	1.2E+04	7.4E+02	3.8E+03	6.8E+03	1.6E+04	1.5E+02	2.2E+02
8	3.6E+07	1.7E+04	8.2E+03	1.1E+05	5.9E+03	4.3E+04	2.6E+04	4.5E+03		6.2E+02
9	5.2E+07	3.0E+03	1.6E+03	6.4E+04	5.0E+02	6.2E+03	9.3E+03	1.6E+04	1.4E+02	2.1E+02
10	5.2E+07	1.4E+04	1.4E+04	7.9E+04	6.9E+03	6.0E+04	3.9E+04	1.6E+03	2.1E+02	3.6E+03
11	1.0E+07	1.2E+04	1.8E+03	1.0E+05	0.0E+00	1.8E+03	1.1E+03			
12	1.0E+08	2.1E+03	3.1E+03	1.4E+04	4.3E+02	7.2E+03	6.1E+03			
15	9.1E+07	3.2E+03	3.2E+03	2.5E+04	1.2E+02	7.7E+03	8.9E+03			
16	3.7E+07	7.0E+02	1.5E+03	3.7E+03	1.0E+02	2.9E+03	3.7E+03			
17	8.1E+07	1.3E+03	2.0E+03	8.2E+03	4.5E+02	6.8E+03	8.6E+03			
18	7.1E+07	9.9E+03	2.9E+03	5.2E+04	3.7E+02	5.2E+03	8.9E+03	5.8E+02	5.2E+01	4.2E+02
19	3.8E+08	2.0E+03	3.1E+03	1.3E+04	1.1E+02	3.5E+03	3.0E+03			
20	1.6E+07	5.3E+04	6.9E+04	9.4E+04	1.8E+04	1.3E+04	9.9E+03	3.0E+04	3.7E+03	1.1E+04
25	2.3E+06					3.2E+03	1.6E+03	6.3E+03		
26	5.2E+05									
30	7.8E+05	1.5E+05	2.1E+05	6.7E+05	3.8E+04	7.1E+04	6.2E+04	3.3E+04	4.8E+03	9.5E+03
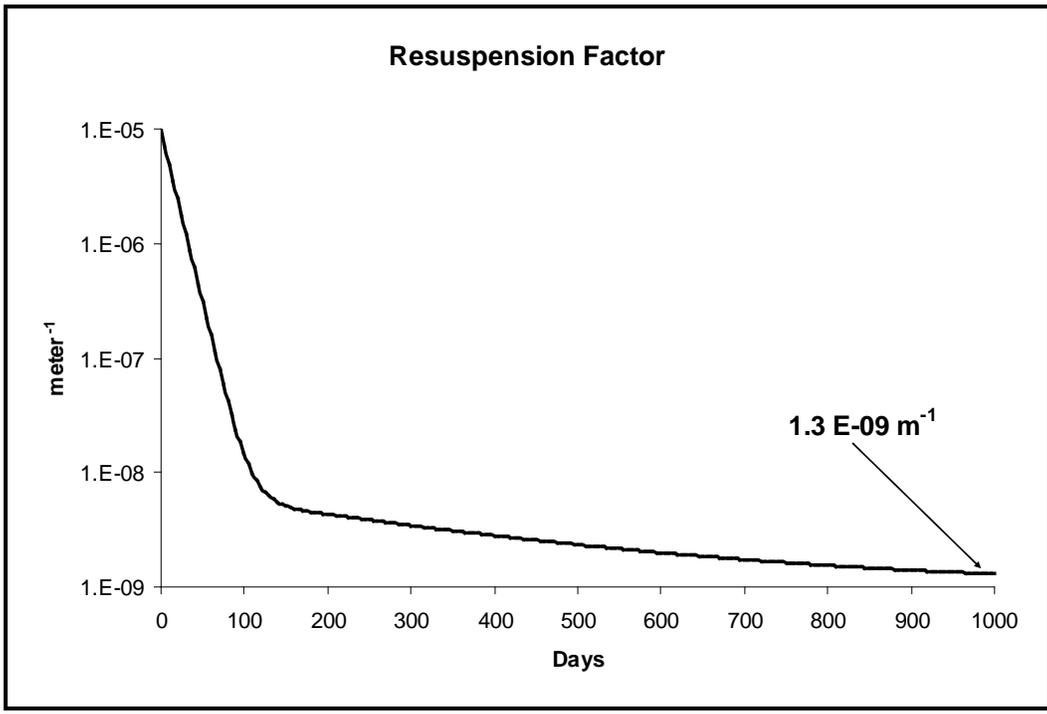
Figure 4.2.2-1. Resuspension factor as a function of time after initial deposition.

1.3 × 10^{-9} m^{-1}. Because the vast majority of radionuclides in surface soils were deposited as a result on atmospheric testing that stopped in 1962, and because it is reasonable to assume that unmonitored employees would not be likely to be exposed to freshly deposited radionuclides, the value of 1.3 × 10^{-9} m^{-1} is assumed for estimating potential airborne concentrations to which employees could have been exposed.

Application of this resuspension factor with areal soil deposition values in Table 4.2.2-1 allows for estimation of airborne radionuclide concentrations at various locations at NTS. If the assumption is made that the standard breathing rate of an employee is 2,400 m^3 yr^{-1}, these concentrations can be used to estimate annual intakes. Table 4.2.2-2 lists these annual intakes.

Table 4.2.2-2. Average annual intakes[a] (Bq) for NTS Areas from soil resuspension.

Area	Am-241	Pu-238	Pu-239,240	Co-60	Cs 137	Sr-90	Eu-152	Eu-154	Eu-155
1	7.06E-03	1.09E-02	4.04E-02	1.85E-03	1.48E-02	2.52E-02	2.52E-02	1.68E-04	8.41E-04
2	6.56E-03	1.95E-02	4.98E-02	2.72E-03	5.43E-02	1.04E-01	3.17E-02		9.05E-04
3	6.35E-03	4.28E-03	5.11E-02	1.38E-03	1.66E-02	4.55E-02	2.48E-02	1.38E-04	6.90E-04
4	1.84E-02	3.62E-02	1.11E-01	4.46E-03	3.34E-02	3.62E-02	2.54E-02		5.57E-04
5	9.22E-03	1.54E-03	7.38E-02	9.22E-03	6.15E-03	1.38E-02	1.54E-01	3.07E-03	
6	2.35E-03	4.55E-03	1.16E-02	2.76E-04	3.86E-03	4.83E-03			
7	5.08E-03	1.39E-03	3.70E-02	2.31E-03	1.20E-02	2.12E-02	5.08E-02	4.62E-04	6.93E-04
8	5.45E-02	2.57E-02	3.53E-01	1.83E-02	1.35E-01	8.02E-02	1.41E-02	0.00E+00	1.92E-03
9	9.36E-03	4.90E-03	1.98E-01	1.56E-03	1.94E-02	2.90E-02	5.13E-02	4.46E-04	6.69E-04
10	4.23E-02	4.23E-02	2.45E-01	2.16E-02	1.87E-01	1.23E-01	4.90E-03	6.69E-04	1.11E-02
11	3.68E-02	5.57E-03	3.23E-01	0.00E+00	5.57E-03	3.34E-03			
12	6.42E-03	9.57E-03	4.39E-02	1.35E-03	2.25E-02	1.91E-02			
15	1.01E-02	9.85E-03	7.95E-02	3.79E-04	2.40E-02	2.78E-02			
16	2.18E-03	4.68E-03	1.15E-02	3.12E-04	9.04E-03	1.15E-02			
17	3.97E-03	6.39E-03	2.56E-02	1.42E-03	2.13E-02	2.70E-02			
18	3.10E-02	9.14E-03	1.63E-01	1.14E-03	1.63E-02	2.78E-02	1.80E-03	1.63E-04	1.31E-03
19	6.31E-03	9.62E-03	4.21E-02	3.31E-04	1.08E-02	9.32E-03			
20	1.65E-01	2.16E-01	2.95E-01	5.68E-02	3.95E-02	3.09E-02	9.35E-02	1.15E-02	3.45E-02
25					9.91E-03	4.95E-03	1.98E-02		
26									
30	4.75E-01	6.69E-01	2.08E+00	1.19E-01	2.23E-01	1.93E-01	1.04E-01	1.49E-02	2.97E-02

a. Geometric standard deviation is assumed to be 4.1 for all values.

For situations in which a covered employee spent a large amount of the working day outside, but the exact locations are not known, dose reconstructions should use the site average annual intake values in Table 4.2.2-3.

Table 4.2.2-3. Site average and maximum annual intakes[a] (Bq) for NTS from soil resuspension.

Radionuclide	Average intake	Maximum intake,
Am-241	1.31E-02	4.75E+00
Pu-238	1.39E-02	6.69E+00
Pu-239	7.93E-02	2.08E+01
Co-60	3.13E-03	1.19E+00
Cs-137	2.71E-02	2.23E+00
Sr-90	2.87E-02	1.93E+00
Eu-152	1.16E-02	1.54E+00
Eu-154	2.53E-04	1.49E-01
Eu-155	1.19E-03	3.45E-01

a. Geometric standard deviation is assumed to be 4.1 for all values.

For purposes of dose reconstruction, the GSD of the values in Table 4.2.2-2 can be estimated using equation 4-1. Under the assumption that the 50th-percentile expected intakes are those in Tables

4.2.2-2 and 4.2.2-3 and that the 95th-percentile values would be enveloped by a factor of ±10, then the GSD for the values in Tables 4.2.2-2 and 4.2.2-3 are 4.1 for all values.

For purposes of reconstructing potential unmonitored dose, the annual intakes in Tables 4.2.2-2 and 4.2.2-3 should be used only for individuals whose employment records indicate that they spent considerable time outdoors. Therefore, these intakes do not apply to miners or tunnel workers. In addition, reasonable adjustments to these values should be made whenever information indicates the covered employee did not spend 100% of his workday in the outdoor environment.

In addition, dose reconstructors should examine the values derived from measured data in Tables 4.2.1.2.2-2 and 4.2.1.2.2-4 (Section 4.2.2.2) and compare them with the values for resuspension intakes in Tables 4.2.2-2 and 4.2.2-3 for ^{238}Pu and $^{239, 240}$Pu because the tables in this Section 4.2.1.2 represent actual measured data, whereas the tables in this section show calculated values based on claimant-favorable assumptions. Because both sets of tables represent the effects of the same primary pathway (i.e., resuspension), it would not be appropriate to use both sets of data to reconstruct a claimant dose. Although there is fairly good agreement between the two data sets, dose reconstructors should use the higher value for a given area when assessing intakes for unmonitored employees when the covered employee's work location is well documented. If it is not known where a covered employee worked, the site average annual intakes for ^{238}Pu and $^{239, 240}$Pu provided in Section 4.2.2, Table 4.2.2-3 should be used to assess annual intakes for each year of employment..

With the exception of plutonium, the vast majority of the radionuclides in NTS soil resulted from atomic weapons tests and were thereby exposed to extreme heat. For purposes of calculating organ dose using the Integrated Modules for Bioassay Analysis program, internal dose reconstructors should assume biological clearance factors are Type S. Because much of the plutonium released was a result of safety tests, these radionuclides should not be considered high-fired. Therefore, for plutonium, dose reconstructors should use maximizing absorption factors for either Type S or Type M.

4.2.3   Uncertainty

There are several major factors that could affect the accuracy of the average annual intake estimates in Tables 4.2.2-2 and 4.2.2-3. The most important factors include the following:

- Uncertainty in and time-dependent nature of the resuspension factor
- Spatial variations in radionuclide soil concentrations
- Duration of exposures

Each of these factors could result in and over- or underestimation of annual intakes equal to several orders of magnitude. The following sections discuss each of these factors.

4.2.3.1   Time-Dependent Nature of the Resuspension Factor

As discussed in Anspaugh (2002), the uncertainty suggested in equation 4-3 is plus or minus one order of magnitude. This is a large uncertainty, but it is based on actual measurements that suggest the very complex nature and unpredictability of the resuspension process. In addition, the time-dependence of the resuspension factor suggested by equation 4-3 covers a range of almost four orders of magnitude. Therefore, if an unmonitored employee was exposed to freshly deposited fallout, the intake values in Tables 4.2.2-2 and 4.2.2-3 could represent significant underestimates of the actual intake. However, because of the rapid decrease in the resuspension factor with time, the likelihood of exposure of this type for periods greater than a few days would be very small. In addition, a claimant-favorable factor of 10 has been applied to the resuspension factor to minimize the

likelihood that airborne concentrations would be underestimated. In addition, the number of unmonitored employees likely to have been inadvertently exposed to fresh fallout would be small.

4.2.3.2   Spatial Variations in Radionuclide Soil Concentrations

The values of radionuclide areal soil concentration in Table 4.2.2-1 are average values for each particular area. The actual sample values for locations within each of the areas can vary by several orders of magnitude. In addition, McArthur (1991, p. 24) estimated the uncertainty of the inventory measurements at a factor of ±2. Therefore, the intake values estimated from these average soil concentrations could under- or overestimate the actual intake values depending on the location of the unmonitored employee within a particular area. However, it is reasonable to assume that radiological control practices would have prevented unmonitored employees from being inadvertently exposed in the more highly contaminated locations.

4.2.3.3   Duration of Exposures

The intake values in Tables 4.2.2-2 and 4.2.2-3 are based on the assumption that the unmonitored employee was exposed for an occupational year (i.e., 2,000 hr). This assumption, although claimant-favorable, is probably an overestimate of the actual time an unmonitored employee would likely have spent outdoors during a year. In addition, this assumption does not take into consideration any protection from airborne dust particles that could have been afforded by vehicles during transportation to and around work locations. Taken together, these factors could result in an overestimation of annual intake by a factor of 10 or more.

4.3   OCCUPATIONAL EXTERNAL DOSE

Workers incur external doses from the ambient radiation levels and from noble gases. Ambient radiation measurements were reported in NTS annual environmental reports (see reference section for document citations) starting in 1967; however, no ambient radiation data were provided in the annual environmental reports between 1968 and 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 completed between 1951 and 1967, most of these were to characterize the effects of weapons tests and were therefore 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^{-1}. Five of these chambers were in small, semiprotective enclosures at locations chosen in NTS living areas.

As a backup for the ionization chambers, standard NTS film dosimeters were included at each sample location. However, all results from these film 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 set at the sample locations for measurements and another set 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 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, however, the ambient radiation reporting was reestablished in 1977 using thermoluminescent dosimeters (TLDs). The dosimeters used were CaF_2:Dy (TLD-200) 0.25- by 0.25- by 0.035-in. chips from Harshaw Chemical Company. A badge consisting of at least two chips shielded by 0.047 in. of cadmium (1030 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^{-1} or about 95 mrem yr^{-1}.

In 1987, the Harshaw dosimeters were replaced with Panasonic Model UD-814 TLDs. 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 present time.

In subsequent years, the number of sampling locations increased to over 150 and eventually covered all populated areas at NTS. Table 4.3.1-1 lists average ambient background radiation levels.   Table 4.3.1-2 lists the annual ambient radiation measured with background included and site average and maximum values. For purposes of dose reconstruction, because the values in Table 4.3.1-2 represent continuous exposure for an entire year, these values should be adjusted for occupational exposure (i.e., 2,000 hr yr^{-1}), 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 in which the employee worked is not known, the claimant-favorable site maximum value should be used.

It should also be noted that some covered employees remained onsite continuously for weeks at a time. However, because most of the non-working 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 4.3.1-2, adjustment of the tabular data is not required to assure ambient exposures are not underestimated for these individuals.

Table 4.3.1-1. Ambient background radiations levels by year.

Year	mrem yr^{-1}
1964	(a)
1965	(a)
1966	(a)
1967	(a)
1968	(a)
1969	(a)
1970	(a)
1971	(a)
1972	(a)
1973	(a)
1974	(a)
1975	(a)
1976	(a)
1977	95
1978	95
1979	95
1980	99
1981	110
1982	100
1983	100
1984	100
1985	100
1986	72
1987	102
1988	131
1989	106
1990	110
1991	112
1992	109
1993	131
1994	93
1995	94
1996	91
1997	95
1998	88
1999	91
2000	115
2001	114

a. Not reported.

1

Table 4.3.1-2. Ambient radiation (mrem yr^{-1}) by Area.[a]

Year	Area 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		
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		

Table 4.3.1-2 (cont.). Ambient radiation (mrem yr^{-1}) by Area.

Year	15	16	17	18	19	20	21	22	23	24	25	26	27	28	29	30	Average	Maximum
																	Area	
1967									142				175				207	318
1968				186					131				172				189	285
1969																		320^b
1970																		320^b
1971																		320^b
1972																		320^b
1973																		320^b
1974																		320^b
1975																		320^b
1976																		320^b
1977																		320^b
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
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

a. Blank cells indicate data not available.
b. Ambient radiation not provided for these years - assumed equal to value for 1978.

1

It is likely that the elevated ambient radiation peaked in the early 1960s near the end of above-ground testing. Therefore, it would be reasonable and probably claimant-favorable to assign maximum annual environmental doses for the years 1963 through 1966 equal to the maximum value reported for 1967 (i.e., 318 mrem yr^{-1}). It should be noted that for 1967, the maximum value is about 50 percent greater than the average value for all areas where measurements were reported. Following this logic, maximum doses have been added to Table 4.3.1-2 (shown in italics) for 1969 through 1977 for which data were not readily available. These values were based on the maximum value for 1978 and should be used to adjust environmental dose for unmonitored employees for those years.

Although not explicitly stated in NTS annual environmental reports, the overall uncertainty of measurements made with modern TLDs (e.g., Panasonic UD-814) has been determined to be less than 20%. However, for earlier measurement methods, the overall uncertainty could have been greater. Therefore, for purposes of dose reconstruction, the claimant-favorable assumption is made that the 95th-percentile values of the expected values, 50th percentile, in Tables 4.3.1-1 and 4.3.1-2 are enveloped by a factor of ±2 of the overall error of 50% (i.e., ±100%) Then equation 4-1 (Section 4.2.1.2.1) would estimate the GSD for the expected values at 1.52.

4.3.2   Releases of Noble Gases

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 previously run by EPA and replaced them with new samplers. These sampling units are housed in a metal tool box with three metal air bottles attached with quick disconnect hoses. A vacuum is maintained on the first bottle, which causes a steady flow of air to be collected in the other two bottles. The flow rate is approximately 0.5 cm^3 min^{-1}. The two collection bottles are exchanged weekly and yield a sample volume of about 3×10^5 cm^3.

The noble gases are separated and collected from the atmospheric sample by a series of cryogenic-gas chromatographic techniques. Water and carbon dioxide are removed at room temperature and the krypton and xenon are collected on charcoal at liquid nitrogen temperatures. These gases are transferred to a molecular sieve where they are separated from any remaining gases and each other. The krypton and xenon are 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.

Krypton-85
The original six samplers were in Areas 1, 5, 12, 15, 23, and 25. These six 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. These elevated concentrations continued until after the cessation of nuclear testing in 1992. Table 4.3.1-3 lists the average background and the Area 20 elevated concentrations from 1983 through 1992.

The source of the elevated [85]Kr concentrations was attributed to the emplacement hole nearest to the Area 20 sampler (U-20a), which is about 3,660 m (12,000 ft) to the south. To determine whether

Table 4.3.1-3. [85]Kr average background and the Area 20 net concentrations and annual organ doses from 1983 through 1992.

Year	NTS background (pCi m[-3])	Area 20 Camp (pCi m[-3])	Net elevated (pCi m[-3])	Source[a] elevated concentration (pCi m)	Source[a] elevated concentration (Bq m[-3])	Other organ[b] dose[c] at source[a] (mrem yr[-1])	Skin dose[d] at source[a] (mrem yr[-1])
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. Source receptor location assumed to be 500 m from U-20a emplacement hole.
b. Includes gonads, breast, lung, red marrow, bone surfaces, thyroid, remainder, and effective whole body.
c. Assumes 2,000 hr yr[-1] exposure and dose conversion factor of 2.2×10^{-16} Sv per Bq s m[-3] (EPA 1993).
d. Assumes 2,000 hr yr[-1] exposure and dose conversion factor of 1.32×10^{-14} Sv per Bq s m[-3] (EPA 1993).

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 (Grossman 2003) and the CAP88-PC (EPA 1998) computer program. The program was used to develop atmospheric dispersion factors, X/Qs, for various distances from the source of emissions. The ration 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 considered to be 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 (EPA 1993).

As shown in Table 4.3.1-3, the upper-bound annual organ doses for all years are less than 1 mrem. For purposes of reconstructing environmental doses for unmonitored employees, these doses are inconsequential for determination of probability of causation 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 present in the individual dosimetry record.

Uncertainty estimates provided for a number of the [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 4.3.1-3 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. Like [85]Kr, these elevated concentrations were attributed to seepage from the underground test

cavities at the Pahute and Rainier Mesa tests. However, ^{133}Xe has a short half-life (5.25 d versus 10.7 yr 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   OCCUPATIONAL DOSE TO TUNNEL WORKERS FROM EXPOSURE TO RADON

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 locations at the Nevada Test Site. 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 (Favor 1987). This section discusses the results of these surveys and those conducted in 1991 and 1992 (Lyons 1992a,b).

4.4.1   Tunnel 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, named 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. 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's (DOD's) 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 purposes but, since 1971, has been used only as an underground research facility.

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.

The U1a Complex in Area 1 is an underground laboratory of horizontal tunnels about 0.5 mi in length 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 and instrumentation as well as 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.

4.4.2   Radon Measurements

Radon-222, with a radioactive half-life of 3.8 d, occurs in the ^{238}U decay chain; ^{220}Ra, with a radioactive half-life of 54.5 s, 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}Ra 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 daughter 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 sampled in the inclined G-Tunnel was mined in Grouse Canyon Welded Ash Fall Tuff, which is extremely fractured. Some factors that affect concentrations of radon and radon daughters 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 daughter products in the air. Concentrations fluctuate with changing seasons, changing weather conditions, and changing activities in the area being monitored.

The concentration of radon daughters (the major dose contributors) in air is measured in *working levels* (WL). This is the common unit for expressing radon decay product exposure rates. The WL was developed for use in uranium mines but is now also used for environmental exposures. Numerically, the WL is any combination of short-lived decay products in 1 L of air that will result 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^{-1} (i.e., 100 pCi L^{-1} each of ^{222}Rn and short-lived decay products ^{218}Po, ^{214}Pb, ^{214}Bi, ^{214}Po) (NCRP 1988, p. 17). For ^{220}Rn and its decay daughters of ^{216}Po, ^{212}Pb, ^{212}Bi, and ^{208}Tl, 1 WL is equal to 7.47 pCi L^{-1}. The advantage of the WL unit is that it allows 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 1998, p. 19). The WL unit considers this factor. However, because the dose conversion factors for ^{220}Rn daughters (Pb-212 + Po-212) are one-third the values for the daughters of ^{222}Rn (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 hr (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 Tunnel 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 can rise to relatively high levels (i.e., 0.24 WL) when ventilation rates were significantly lower (Favor 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 (see Table 4.4.3-1).

Lyons (1992a,b) reported additional radon measurements from 1991 and 1992. Although the G-Tunnel complex remained inactive during 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 during 1992 in T-Tunnel, which was inactive that year.

Table 4.4.3-1. Results of experiment to determine the effects of ventilation conditions on RDCs in the G Tunnel inclined drift.

Location	Ventilation conditions	Radon daughter grab sample average (WL)
Rock mechanics drift at 0+52	Alternating	0.13
	Continuous	0.05
Rock mechanics access drift at the heating block alcove	Alternating	0.13
	Continuous	0.04
Bottom of rock mechanics access drift	Alternating	0.13
	Continuous	0.04
EV6/EV5 Junction	Alternating	0.1
	Continuous	0.03
EV5 at HF36	Alternating	0.08
	Continuous	0.03
Average	Alternating	0.13
	Continuous	0.05

Source: Favor (1987, Table 3).

Tables 4.4.3-2 and 4.4.3-3 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.

Table 4.4.3-2. Radon daughter concentrations for N-Tunnel in 1991 and 1992.

	Rn-222 WL		Rn-220 WL	
N Tunnel location	Average	Maximum	Average	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**

4.4.4   Tunnel Workers Exposure to Radon

Although measurements were periodically performed in the tunnel complexes to ensure adequate worker protection, neither the U.S. Department of Energy nor its predecessor agencies attempted to quantify or record occupational exposures to radon and its daughters. Therefore, dose reconstructors should adjust the dose of any employee who was a miner or tunnel worker to account for radon

Table 4.4.3-3. Radon daughter concentrations for P-Tunnel in 1991 and 1992.

P-Tunnel location	Rn-222 WL		Rn-220 WL	
	Average	Maximum	Average	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**

exposure while working in the tunnel complexes. To quantify the exposure, Table 4.4.4-1 lists claimant-favorable airborne ^{222}Rn RDCs based on measurement results. It should be noted that the values provided in Table 4.4.1-1 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 prior to entry into IREP.

Table 4.4.1-1. Annual ^{222}Rn exposures and uncertainties for internal dose reconstruction for miners and tunnel workers.

Tunnel complex	RDC concentration (WL)	Annual exposure (WLM)[a]	Uncertainty GSD[b]
G (before 1984)	0.13	1.56	1.33
G (1984 and later)	0.05	0.60	1.33
N	0.013	0.16	1.33
P	0.006	0.07	1.33
T	0.01	0.12	1.33
U1a Complex[c]	0.05	0.60	1.33
Unidentified[d]	0.05	0.60	1.33

a. Based on 170 hr of exposure per month for 12 months.
b. Based on the assumption that one standard deviation equals ±30%.
c. Applicable to underground work only.
d. Use these values when underground work location is not known.

For Tunnel G workers, annual exposures (assuming a full year of underground activity) are assumed to be 1.56 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.

Because no radon measurement data were readily available for the underground portions of the U1a Complex, the claimant-favorable assumption was made that the RDCs in U1a would be similar to those in G Tunnel (which is mined in highly fractured tuff). In addition, a reasonable assumption was made that the U1a Complex is well ventilated, similar to G Tunnel after 1984. Dose reconstructors should use these average, well-ventilated, annual exposure values for miners and tunnel workers without identified tunnel locations.

For purposes of dose reconstruction, the GSD of the values in Table 4.4.4-1 can be estimated using equation 4-1. Under the assumption that the 50th-percentile expected annual exposures are those in Table 4.4.4-1 and that the 95th-percentile values would be enveloped by a factor equal to ±60 percent (i.e., two times a sigma value of ±30 percent), then the GSD for the values in Tables 4.4.4-1 are 1.33 for all values.

4.4.5   Uncertainty

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 hr 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 approximately 5 L min^{-1} and counted on an EDA Instruments Incorporated Radon Detector (Model #RD200). The Rolle method allowed for a choice of several different analysis regimes. The regime chosen for tunnel measurements was a 5-min sampling time, a 6-min decay time, and a 5-min counting time. Using appropriate correction factors, the RDC in WL can be evaluated 11 min after sample collection. The values obtained using the Rolle method were periodically checked using the Kusnetz (Favor 1987, p.3) method, which also utilized a single count of an air filter but allowed for a 40- to 90-min decay time. These measurement techniques have been found to have good accuracy and a relative standard deviation of less than 15% (Favor 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 pCi L^{-1}. 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 approximately 0.3 pCi L^{-1} for a 1-week sample with a relative standard deviation of 20% (Favor 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 TLD2OO) 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 approximately 0.0001 WL for a 1-week sampling period with a relative standard deviation of 10% (Favor 1987, p. 4).

The RPISU technique is recommended for the measurement of RDCs by the Technical Measurements Center of the U.S. Department o f Energy, Division of Remedial Action Projects (Favor 1987, p. 4).

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