

ORAU Team Dose Reconstruction Project for NIOSH Technical Basis Document for the Nevada Test Site – Occupational Internal Dose	Document Number: ORAUT-TKBS-0008-5 Effective Date: 09/30/2004 Revision No.: 00 Controlled Copy No.: _____ Page 1 of 82
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RECORD OF ISSUE/REVISIONS

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Draft	11/07/2003	00-A	Technical Basis Document for the Nevada Test Site – Occupational Internal Dose. Initiated by Eugene M. Rollins.
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ACRONYMS AND ABBREVIATIONS

ANSI	American National Standards Institute
BN	Bechtel Nevada, Inc.
BZ	breathing zone
CAM	continuous air monitor
cc	cubic centimeter
CFR	Code of Federal Regulations
cpm	counts per minute
CWT	chest wall thickness
DAC	Derived Air Concentration
DL	Decision Level
DOE	U.S. Department of Energy
dpm	disintegrations per minute
DU	depleted uranium
EPA	U.S. Environmental Protection Agency
FWHM	Full-Width-Half-Maximum
g	gram
GFP	gross fission product
G-M	Geiger-Mueller
HEPA	high-energy particulate air (filter)
HEU	highly enriched uranium
IMBA	Integrated Modules for Bioassay Assessment
K	potassium
keV	kilo electron volts
Kr	krypton
L	liter
LANL	Los Alamos National Laboratory
LASL	Los Alamos Scientific Laboratory
LLD	Lower Limit of Detection
LLI	lower large intestine
LLNL	Lawrence Livermore National Laboratory
LRL	Lawrence Radiation Laboratory
m ³	cubic meter
MDA	Minimum Detectable Activity; Minimum Detectable Amount
mL	milliliter
mm	millimeter
MPBB	Maximum Permissible Body Burden
MPC	Maximum Permissible Concentration
Nal	sodium iodide

NAS	National Academy of Sciences
NBS	National Bureau of Standards
nCi	nanocurie
NTS	Nevada Test Site
pCi	picocurie
RAS	retrospective air sampler
REEC _o	Reynolds Electrical & Engineering Company, Inc.
RSN	Raytheon Services Nevada
TBD	technical basis document
ULI	upper large intestine
U.S.C.	United States Code
WBC	Whole-Body Counter
WEF	Waste Examination Facility
μCi	microcurie

5.1 INTRODUCTION

Technical basis documents (TBDs) and Site Profile Documents are general working documents that provide guidance concerning the preparation of dose reconstructions at particular sites or categories of sites. They will be revised in the event additional relevant information is obtained about the affected site(s). These documents may be used to assist the National Institute for Occupational Safety and Health (NIOSH) in the completion of the individual work required for each dose reconstruction.

In this document the word “facility” is used as a general term for an area, building, or group of buildings that served a specific purpose at a site. It does not necessarily connote an “atomic weapons employer facility” or a “Department of Energy facility” as defined in the Energy Employee Occupational Illness Compensation Program Act of 2000 [(42 U.S.C. Sections 7384l (5) and (12))].

The Nevada Test Site (NTS) is unique in the U.S. Department of Energy (DOE) Complex because it has not hosted ongoing production of materials as other DOE facilities have. The NTS was, and is, an outdoor testing and research facility rather than a manufacturing or processing site. When the NTS began atmospheric testing in 1951, its radiological safety programs were assumed to be consistent with programs at Los Alamos National Laboratory (LANL; previously known as Los Alamos Scientific Laboratory, LASL) and Lawrence Livermore National Laboratory (LLNL; previously known as Lawrence Radiation Laboratory, LRL).

Reynolds Electrical & Engineering Company, Inc. (REECo), the NTS general contractor from 1952 to 1995, became responsible for onsite radiological safety activities in 1955 and began onsite laboratory analyses, including bioassay samples, in 1958. LANL and the military were the primary parties responsible for radiological safety oversight prior to 1955.

LANL conducted bioassay procedures from 1955 to 1958. In 1958, REECo began onsite bioassay. By 1961, REECo was conducting routine bioassay for ^3H , ^{239}Pu , gross fission products (GFP), and gamma emitters. Bechtel Nevada (BN), which became the site contractor in 1995, currently holds bioassay responsibility. In 2001, BN disbanded the onsite analytical laboratory and contracted the services to an outside laboratory.

In 1967, REECo established an onsite whole-body counting capability at NTS. Prior capability existed at other locations, but this information is not well documented. Records include lung, chest, thyroid, and wound counting results in support of specific tests and projects. The whole-body counting capability at NTS was maintained until 1999 when BN decommissioned the system.

Initially NTS adopted the contamination and internal limits published in the National Bureau of Standards Handbook 52 issued March 20, 1953 (REECo 1961). Authorized external limits could vary by test and job assignment based on the type and size of the test.

The information in this TBD applies only to the NTS. It does not apply to cases in which the worker was at non-NTS locations (e.g., Pacific, Mississippi, Colorado, New Mexico, Alaska, or other Nevada locations, such as the Tonopah Test Range, Central Nevada Test Area, and the Project Shoal site near Fallon) during testing. Reconstruction of doses for NTS workers who were present at tests at those locations will be addressed separately.

Historic Monitoring Perspective

At NTS, the primary emphasis was external dosimetry, particularly during atmospheric testing. This emphasis was based on animal data showing that the internal dose received during the life span of

the animal due to inhalation was small in comparison to the external dose. The rule-of-thumb for atmospheric testing was that external exposure was the controlling factor. If the external dose was controlled, the internal dose would be low. The following paragraphs describe the NTS historical perspective for internal dose (REECo unknown a):

*"During Atmospheric Testing Era at the Nevada Test Site, early animal data showed that the internal dose due to inhalation was small in comparison to the external exposure. For example, In WT-396, "Biological Injury from Particle Inhalation," [see Smith, Boddy, and Goldman 1952] reached the conclusion, **that total internal dose due to the emission of beta particles was less than one percent of the external dose.**"*

*UCLA [the University of California at Los Angeles] studied the question of internal exposure and reported their results in WT-1172, "Evaluation of the Acute Inhalation Hazard from Radioactive Fall-out Materials by Analysis of Results from Field Operations and Controlled Inhalation Studies in the Laboratory" [Taplin, Meredith, and Kade 1958]. The UCLA scientists concluded that "from consideration of physical factors alone (such as strength and type of detonation, particle-size distribution, decay rates, meteorological conditions, air-borne radioactivity levels, and percentages of radioactivity from 0.1 to 5.0 micron size range), **the acute external beta-gamma radiation hazard is at least 1000 times greater than that from inhalation.**" The rule-of-thumb for atmospheric testing was that the external exposure was the controlling factor. If you controlled for the external dose, the internal doses would be low.*

Tritium does not follow this rule-of-thumb, but tritium exposure was confined to the NTS tunnel environments. The Tunnel workers in this environment received quarterly bioassay tests. During the underground testing era, there was an active bioassay program. Personnel that had a potential for internal exposures were placed in the bioassay monitoring program.

Dose reconstructions performed for military personnel on the site during atmospheric testing determined that the rules-of-thumb were not valid for individuals in the line of the fallout or present in an area where recent or previous tests had been performed and resuspension of residual contamination was a possibility. Depending on work activities, internal exposure from resuspension of radioactive material can be an issue in some areas years after above-ground testing ended (NAS 2003). Potential intake from resuspension is discussed in Section 4.1.2 of this TBD, which addresses environmental doses.

Radionuclides of Concern

Technical Basis for Internal Dosimetry at the NTS (REECo 1993a) reflects radiological protection practices from about 1970 through the end of nuclear weapons testing in 1992, and is the best available source of internal dosimetry information for the nuclear weapons testing era (Arent and Smith 2003). Subsequent technical basis documents by Bechtel Nevada, Inc. (BN 2000, 2003a) cover the current needs of NTS and are not comprehensive on the support of nuclear weapons testing.

Each operation that occurred at NTS had a different set of radionuclides of concern at the time of the test and afterward. Radionuclides were identified for NTS locations and timeframes, including atmospheric, underground, and nuclear reactor/rocket development tests, and legacy contamination.

Radionuclides of concern for internal dosimetry can be identified by locations, test category or facility. This information is in Attachment 5D, Section 5D.4.1.

Iodine, krypton, xenon, and tritium were important radionuclides following nuclear tests. Krypton and xenon were primarily external hazards, whereas ^{131}I and ^3H were internal hazards (Glasstone 1971). Radionuclides that have resulted in recorded doses above established limits at NTS are ^3H , ^{131}I , ^{239}Pu , and ^{241}Am (Arent and Smith 2003).

At NTS, the primary elements of dosimetric concern are plutonium (^{238}Pu , ^{239}Pu , ^{240}Pu , ^{241}Pu , ^{242}Pu), uranium (^{233}U , ^{234}U , ^{235}U , ^{236}U , ^{238}U), americium (^{241}Am , ^{243}Am), curium (^{244}Cm), strontium (^{85}Sr , ^{90}Sr , ^{90}Sr - ^{90}Y), cesium (^{137}Cs), tritium (^3H), radium (^{226}Ra), and thorium (^{228}Th , ^{232}Th) (REECo 1993a).

Iodine was of concern after full criticality experiments (i.e., atmospheric testing and venting). Inhalation is assumed to be the most frequent mode of intake, so the lungs are organs of concern in all cases. Cesium and tritium irradiate the whole body relatively uniformly and do not contribute to one organ or tissue preferentially.

Bioassay Program Description

As noted in the Historical Perspective (REECo date unknown a), the dosimetry emphasis was on external monitoring. Early bioassays screens (e.g., nasal swabs, respirator swipes, and urine samples) were performed in the event that contamination was found or suspected. A positive nasal swab initiated the collection of a urine sample. As the bioassay program matured, urine samples were collected in a routine, random screening process. Routine bioassays at NTS included quarterly urine samples, annual whole-body count, and new/termination whole-body counts (REECo 1993a). Nonroutine bioassay types included job-specific and occurrence response. Examples of nonroutine bioassays include the following analyses:

- Tritium or gamma scan
- Gross fission product (beta)
- Specific radionuclides

Whole-body counts, lung counts, thyroid counts, and biological sampling were performed as soon as practicable after a suspected intake (REECo 1993a). Specific examples include:

- Lung counts following a suspected intake of thorium, uranium, or a transuranic
- Whole-body counts for detecting most gamma-emitting fission and activation products
- Thyroid counts for suspected radioiodine uptakes
- Urine bioassay for detection of pure beta emitters such as ^{89}Sr and ^{90}Sr
- Urine and feces sampling and lung and whole-body counting to detect and assess intakes of actinides

Facility descriptions and their specific routine bioassay programs (REECo 1993a) are summarized in Attachment 5D, Section 5D.4.1. Bioassay samples might be taken to document the fact that the worker was not internally contaminated as well as to confirm a suspected intake (REECo date unknown b).

At present, a large amount of the radiological work at NTS is performed through projects with a specified (often short) duration. Routine bioassay monitoring, therefore, is usually part of the project.

A baseline sample is collected as necessary, and post-work samples are collected at the conclusion of the project. If a project requiring bioassay monitoring is expected to last longer than 6 months, bioassay sampling on a quarterly to semiannual frequency throughout the duration of the project is recommended. An exception to this could be a ^3H project with high exposure potential with a relatively high frequency of monitoring required (e.g., weekly to monthly), because ^3H is not detected by typical field monitoring instruments (BN 2003a).

5.2 *IN VITRO* MINIMUM DETECTABLE ACTIVITIES AND COUNTING METHODS

Decision Levels and Minimum Detectable Activities

From BN (2003a), the DL is calculated as:

$$DL = \frac{1.65 \sqrt{Rb \times Ts \times \left(1 + \frac{Ts}{Tb}\right)} + 3}{K Ts}$$

where:

Rb = background count rate, Cb/Tb; Cb = background counts

Ts = sample count time

Tb = background count time

K = calibration factor in appropriate units such as counts in seconds per unit activity

According to BN (2003a), the DL equation is modified for unpaired blank and sample counting times, and was developed for an alpha probability of a Type I error (false positive) equal to 0.05. The DL is applied to an individual sample to determine if the sample count rate is different from the count rate of an appropriate blank.

The Minimum Detectable Activity or Amount (MDA) is an *a priori* value used to evaluate the laboratory's ability to detect an analyte in a sample. BN (2003a) defines the MDA as "the smallest amount (activity or mass) of an analyte in a sample that will be detected with a probability, beta, of non-detection (Type II error) while accepting a probability, alpha (Type I error), of erroneously deciding that a positive (non-zero) quantity of analyte is present in an appropriate blank sample. The MDA is computed using the same value of alpha as used for the DL [Decision Level]. The MDA depends on both alpha and beta. Measurement results are compared to the DL, not the MDA; the MDA is used to determine whether a program has adequate detection capability. The MDA will be greater than or equal to the DL." The MDA corresponding to the above DL, with a beta probability of a Type II (false negative) error equal to 0.05 is:

$$MDA = \frac{3 + \left(3.29 \sqrt{Rb \times Ts \times \left(1 + \frac{Ts}{Tb}\right)}\right)}{K Ts}$$

In REECO (1993a), the MDA equation was reported as:

$$MDA = [3 + (4.65(C_b)^{1/2})] / ETVR$$

where:

C_b = total counts collected in count time (T)

E = counting efficiency

T = count time
V = sample volume
R = fractional chemical recovery

Lower Limit of Detection (LLD) is defined in REEC_o (1993a) as a value selected above the MDA to reduce the probability of reporting false positive results. *Detection limit* is a general term related to the smallest amount of material detectable as a function of the measurement method and instrument background.

Bioassay codes used in the records are in the tables in Attachment 5D, Section 5D.2.1.

5.2.1 In Vitro Urine and Fecal Analysis

MDA Values for Urinalysis and Fecal Analysis

Table 5-1 lists current values of MDA for *in vitro* analyses of routine urine and fecal samples (BN 2003a). Because no specific beginning dates were found, the document publication date should be used as the effective date. Historic MDA values are included in Section 5.2.2 and Attachment 5D, Table 5D-6.

Table 5-1. 2003 target minimum detectable activities for *in vitro* bioassay sample analysis.

Parameter and analysis method	Sample type ^b and container	Minimum detectable amount ^a	Reporting units
Pu-238, Pu-239/240	Urine, 4-L or 500-mL plastic bottle ^c	0.006 ^c	pCi/sample
	Feces, 1-L plastic container	0.03	pCi/sample
Am-241	Urine, 4-L or 500-mL plastic bottle ^c	0.006 ^c	pCi/sample
	Feces, 1-L plastic container	0.03	pCi/sample
Th-230/232 ^(d)	Urine, 4-L or 500-mL plastic bottle	0.02	pCi/sample
	Feces, 1-L plastic container	0.05	pCi/sample
U-234, U-235, U-238	Urine, 4-L or 500-mL plastic bottle	0.04	pCi/sample
	Feces, 1-L plastic container	0.04	pCi/sample
Cm-244 ^d	Urine, 4-L or 500-mL plastic bottle	0.008	pCi/sample
Ra-226 ^d	Urine, 4-L or 500-mL plastic bottle	0.1	pCi/sample
Sr-90	Urine, 4-L or 500-mL plastic bottle	1	pCi/L
Gamma spectroscopy	Urine, 500-mL plastic bottle	100 (Cs-137)	pCi/L
H-3	Urine, 500-mL plastic bottle	0.005	μCi/L
Gross alpha/beta	Cotton swabs, smears	Alpha - 10 Beta - 100	pCi/sample

Source: BN (2003a)

- With the exception of Ra-226, Sr-90, Cs-137, H-3, and gross alpha/beta, upward adjustments for larger sample sizes can be allowed.
- Sample collection was 24 hr.
- Due to limits of the alpha spectroscopy methodology employed for these parameters, these MDAs, while preferred, should be considered goals toward which the contracted laboratory will work to achieve on each set of samples using reasonable processing parameters such as analyzing the entire 24-hr void sample, obtaining reasonable recoveries, extending count times, and reasonable background levels.
- Used in calibration activities

5.2.2 In Vitro Methods for Individual Radionuclides

REEC_o (1993a) states that periodic urine samples were collected and analyzed for ²³⁸Pu, ²³⁹Pu, elemental uranium, ²³⁴U, ²³⁵U, ²³⁸U, ²⁴¹Am, ⁸⁹Sr, ⁹⁰Sr, ³H, and gross fission products. Common *in vitro* radiobioassays included:

- Tritium in urine by liquid scintillation beta counting

- Strontium in urine by beta counting
- Gross fission products in urine by beta counting
- Uranium in urine by fluorometric analysis
- Plutonium and americium in urine by alpha spectroscopy
- Radium, uranium, plutonium, and americium in feces by alpha spectroscopy
- Gamma emitters in urine and feces by gamma spectroscopy

The following paragraphs discuss *in vitro* methods for specific radionuclides. The MDAs for these radionuclides are listed in Attachment 5D, Table 5D-6. The suggested priority for dose reconstructors with regard to MDAs and other detection thresholds is to use (1) the limit on dose report if available, (2) Table 5D-6 values, or (3) an appropriate published value from another DOE site or an applicable value referenced in the literature.

5.2.2.1 In Vitro Bioassay for Iodine

In 1961, the laboratory limit of sensitivity for ^{131}I was listed as 10 pCi/sample (REECo 1993b). In 1993, the ^{131}I LLD was listed as 100 pCi/L urine (REECo 1993a). Iodine was not listed as a routine bioassay in 1993 and the special bioassay program consisted of air monitoring. Bioassay records indicate if the air samples were collected using a charcoal canister or a filter

An NTS internal report, *Iodine, 1960-1963* (REECo date unknown c), includes method development information and post-test drilling air sampling results, isotopic ratios, thyroid doses by isotope, iodine thyroid dose related to external gamma dose, an *in vitro* blood testing protocol, percent contribution to dose among isotopes, relative thyroid doses from inhalation of curie-per-cubic-meter concentrations of each isotope, and a comparison of external whole-body dose and internal dose to the thyroid.

Following a nuclear detonation, several radioiodine isotopes are produced; those with mass numbers 131 through 135 are of significance from the standpoint of exposure to personnel. Table 5-2, derived from data presented in Bolles and Ballou (1956) provides the relative activity normalized to that of ^{131}I , at times ranging from 1 hr to 1 month after fission of ^{235}U . Hence, if the time of detonation and the activity of any one iodine isotope are known, the activity of the other iodine isotopes can be determined. Dose reconstructors should consult access records in the "Other Monitoring" section of the DOE file to determine likely exposure time after the test.

Table 5-2. Relative abundance of various radioiodides.

Nuclide	1 hr	2 hr	4 hr	10 hr	1 day	2 day	4 day	1 wk	2 wk	1 mo
I-131	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
I-132	10.1	5.9	4.5	4.2	3.4	1.9	1.7	1.4	0.4	--
I-133	22.2	14.8	14.2	13.4	9.5	6.2	1.1	--	--	--
I-134	344	142	68.1	0.9	--	--	--	--	--	--
I-135	84.4	59.1	43.6	16.6	4.5	0.3	--	--	--	--

The relative activity from each of these five radioiodine isotopes at early times after fission is a complex function of the ingrowth and decay of the various iodine isotopes and their fission chain precursors, and dependent to some extent on whether produced by thermal or fast fission and by fission of what nuclide. At times ranging to about 4 hr after fission, ^{134}I is by far the predominant radioiodine activity, accounting for almost all the activity from the five radioiodine species. The peak activity from this radionuclide occurs at about three-fourths of an hour after detonation, after which ingrowth of activity is no longer dominant and radioactive decay of this short-lived species ($T_{1/2} = 53$ min) becomes predominant. The peak activity from this nuclide is 26 MCi per KT of explosive yield, compared to 0.11 MCi from ^{131}I , which occurs about 5 hr after detonation (Holland 1964). A

similar, but less rapid effect is apparent for the other iodine isotopes in relation to ^{131}I . At times less than about 1 hr after detonation, the relative activity from the higher iodines is very much larger than at later times, largely as a result of the rapid ingrowth of ^{131}I , whose absolute activity peaks at about 5 hr after detonation, thereafter declining approximately according to its characteristic 8.05-day half life. Kathren (1964) contains a graphic representation of the relative activity from the five radioiodine isotopes of significance.

Dose reconstructors will need to consider iodine when the cancer identified with the claim is thyroid.

5.2.2.2 *In Vitro* Bioassay for Americium

The LASL (1954) method for americium in urine was based on the coprecipitation of americium with bismuth phosphate from a nitric acid solution of urine salts at a pH of 1.7. The bismuth phosphate was dissolved in 6N HCl and the americium was coprecipitated a second time with lanthanum fluoride. The precipitate was slurried onto a stainless-steel plate and counted with a low-background proportional alpha counter. Quantities of the order of 2 dpm/24-hr sample or 6×10^{-19} gram of americium could be determined by this method. The tolerance for americium in urine used at LASL was 7 dpm/24-hr sample. Samples were rechecked if the count was 2 dpm/24-hr sample or higher (LASL 1954).

In 1958, LASL used the same procedure for americium in urine and stated that thorium, plutonium, curium, actinium, and neptunium were carried through this determination. Quantities on the order of 0.5 dpm of americium were detected by this method (LASL 1958).

REECo documentation (REECo 1968-1991a) discussed an americium method with steps for separation by precipitation/oxidation, purification by anion exchange column, electrodeposition of ^{241}Am - ^{243}Am on a stainless-steel disc, and detection by alpha spectrometry during the period from 1981 through 1983. The procedure states that ^{241}Am is analyzed using ^{243}Am as a tracer. From 1982 to 1987, the detection limit was listed as 2×10^{-11} $\mu\text{Ci/mL}$ for ^{241}Am in urine (REECo 1977-1987).

In 1993, the LLD for ^{241}Am was listed as 0.03 pCi/L for urine and 0.03 pCi/g for feces, with a note that americium cannot be chemically differentiated from californium and curium (REECo 1993a). The routine bioassay was a quarterly urine sample and the special bioassay consisted of an additional urine sample and a lung count. REECo (1993a) stated that americium in urine was detected with an alpha counter (assumed to be an alpha spectrometer) with a "typical" MDA of 0.05 pCi/L. However, the documentation does not provide a definition for "typical." Attachment 5D, Table 5D-6 lists historical detection thresholds for various periods. Table 5-1 lists the current MDA.

5.2.2.3 *In Vitro* Bioassay for Plutonium

LASL used the following procedure for plutonium in urine in 1954:

The urine sample was ashed with nitric acid; plutonium was coprecipitated with bismuth phosphate, dissolved in hydrochloric acid, and then coprecipitated with lanthanum fluoride. The lanthanum fluoride precipitate was slurried on a stainless-steel plate and counted for alpha activity with a low-background proportional counter. Quantities of the order of 2 dpm or 2×10^{-11} gram of plutonium could be determined by this method. The tolerance for plutonium in urine in the LASL "Official Monitoring Handbook" was 7 dpm/24-hr sample. Samples were rechecked if the results were 2 dpm/24-hr sample or higher. (LASL 1954).

The 1958 LASL procedure for plutonium in urine was as follows: The urine sample was wet-ashed with nitric acid. After reduction to plutonium (III) with hydroxylamine hydrochloride, the plutonium was coprecipitated from acid solution with lanthanum fluoride, oxidized to plutonium (IV) with sodium nitrite, and separated from the residual urine salts and the lanthanum by extraction with thenoyltri fluoroacetone. The extracted plutonium was oxidized to plutonium (VI) by hypochlorite and electrodeposited on a stainless-steel disc with a 38.5-mm² plating area to give a thin, uniformly distributed film of plutonium. The discs were exposed to nuclear-track alpha plates and the number of tracks from the alpha particles were counted visually with a microscope. The electroplated discs were counted electronically if the results were needed immediately. The procedure separated plutonium from uranium, actinium and its progeny americium, curium, and thorium. Plutonium recoveries of 85 ±5% were routinely obtained. The detection limit at the 99% confidence level was approximately 0.05 dpm/sample (3.6×10^{-13} grams of plutonium). Samples showing 0.2 dpm/sample or more (1.5×10^{-12} grams of plutonium) were rescheduled (LASL 1958a).

In 1961, REEC_o urine analysis for plutonium included sample preparation, anion exchange, electrodeposition, autoradiograph, and alpha track counting (REEC_o 1961). The laboratory capability (limit of sensitivity) was 0.005 dpm/sample (REEC_o 1993b). Further documentation of the limit of sensitivity for ²³⁹Pu is described in Geiger and Whittaker (1961). In the evaluation of the procedure, 61 urine samples spiked with 0.027 pCi of ²³⁹Pu; 79 blank urine samples were analyzed. The average recovery was 84 ±18% (90% CL). The average blank was equivalent to 0.001 pCi of ²³⁹Pu. The sensitivity was arbitrarily stated as 0.005 pCi, which was five times the blank. An aliquot of 250 mL was used in the procedure; therefore, 0.02 pCi of ²³⁹Pu in a 24-hr sample could be easily detected. This is approximately 20% of the investigation limit and represents less than 5% of a maximum permissible body burden of ²³⁹Pu. The exposure time for this method was listed as 10,000 minutes (Geiger and Whittaker 1961).

In 1968, alpha spectrometry was used to count ²³⁹Pu in urine samples (REEC_o 1968-1991b) and the permissible bone burden for ²³⁹Pu was listed as 0.04 µCi (REEC_o 1993b). From 1982 to 1987, the ²³⁹Pu MDA was 5×10^{-11} µCi/mL and the ²³⁸Pu MDA was 2×10^{-10} µCi/mL (REEC_o 1977-1987).

REEC_o (1993a) stated that plutonium urine and fecal samples were analyzed by alpha spectrometry, which could not differentiate between ²³⁹Pu and ²⁴⁰Pu. Urine samples were also analyzed by gamma spectroscopy, which could differentiate between the two isotopes. Urine alpha spectroscopy had an MDA of 0.01 pCi/L, and the fecal sample MDA was 0.004 pCi/g. The gamma spectroscopy method had an MDA of 50 pCi/L for ²³⁹Pu. A "typical" MDA for a 1,500-mL urine sample counted for 1,000 minutes was 0.02 pCi/L. The ²³⁸Pu LLD was 0.01 pCi/L urine. Routine bioassay consisted of a quarterly urine analysis, and the special bioassay consisted of a combination of urine and fecal analysis plus a lung count. Attachment 5D, Table 5D-6 lists historical detection thresholds for various periods. The current MDA is listed in Table 5-1.

5.2.2.4 *In Vitro* Bioassay for Tritium

The 1954 LASL procedure for tritium in urine was as follows: The sample was prepared for counting in a vacuum line. Urine was dropped onto metallic calcium, and hydrogen and tritium were evolved. The gas flowing into the evacuated system was passed through liquid nitrogen-cooled traps to remove unreacted water and condensable gases. The gas was allowed to flow into a tube similar to a Geiger-Muller tube until a pressure of 15 cm of mercury was attained. Ethylene and argon were added to give a total pressure of 22 cm of mercury. The beta activity was counted with a scaling circuit having an input sensitivity of 0.25 volt. A tube similarly filled with inert hydrogen was counted simultaneously to determine the environmental background. The background count was subtracted from the sample count to obtain the true sample count. The method had an efficiency of approximately 40% and a

precision of $\pm 5\%$ in the range of 1 to 250 $\mu\text{Ci/L}$ of tritium. Samples with higher concentrations could be determined with appropriate dilutions. The tolerance for tritium in urine at LASL was 250 $\mu\text{Ci/L}$. Ten days was used as the biological half-life of tritium. This half-life could be decreased by increasing the fluid intake of the worker (LASL 1954). In 1958, the procedure was the same except the tolerance for tritium in urine used was 85 $\mu\text{Ci/L}$ (LASL 1958a).

LASL (1958b) stated the following regarding tritium urinalysis: An acute dose resulting in 124 $\mu\text{Ci/L}$ in the body fluids (including urine) will expose the person receiving it to 0.6 rem in the first 2 weeks following the exposure and, until completely eliminated (assuming a 12-day elimination half-life), expose the person to 1.08 rem to the whole body. The 1958 tolerance for tritium in urine was listed as 85 $\mu\text{Ci/L}$. In 1959, the Maximum Permissible Concentration (MPC) for tritium was listed as 1.2 mCi in the body, which yielded 0.1 rem per week. Assuming 43.4 L of water in the body, 1-mCi body burden of tritium would result in a urine assay of 23 $\mu\text{Ci/L}$.

Records from 1966 to 1968 show that tritium analysis was conducted with a liquid scintillation spectrometer that had an efficiency of 18% and an average background of 16 cpm (REECo 1993b). In 1971, the urine sample tritium "alert level" was listed as $1 \times 10^{-3} \mu\text{Ci/cc}$ (REECo 1993b). The detection limit for ^3H in urine was listed as $1 \times 10^{-6} \mu\text{Ci/mL}$ from 1982 to 1987 (REECo 1977-1987).

REECo (1993a) reported an MDA for tritium of 300 pCi/L using liquid scintillation counting to detect the weak beta emitted. Routine bioassay is a quarterly urine sample and the special bioassay is an additional urine sample. A "typical" MDA value of 470 pCi/L is listed for a 70-minute count. The current MDA is listed in Table 5-1. Attachment 5D, Table 5D-6 lists historical detection thresholds for various periods. On NTS record forms, tritium monitoring can be called "ACTIVITY" or "ACT," "EVERGREEN," "MINT," or "T".

Tritium monitoring started in 1958 with an MDA of 5 $\mu\text{Ci/L}$ used for urine samples (Arent and Smith 2004). Tritium was reported separately in the dosimetry records. Dose reconstructors should assign a tritium missed dose to the covered employee for years when the employee was monitored for tritium exposure and no tritium dose was reported or when the reported tritium dose was less than the calculated annual potential missed dose of 0.355 rem. This dose is based on the MDA of 5 $\mu\text{Ci/L}$ used at NTS. Individuals who were involved in tunnel work with job classifications of miner, mucker (muck machine operator), "bull gang" (underground laborer), shifter, tunnel walker, dinky locomotive operator and who held a Q-level clearance should be assigned tritium dose. No worker whose employment history is intermittent (employment intervals ≤ 5 months) could have obtained a Q clearance and, therefore, would not have been involved in tunnel reentry or emplacement of devices during events. Having a Q-level clearance and working in Area 1 or 12 is an indication to the dose reconstructor of the possibility of tritium exposure. All other workers should not be assigned tritium dose (Arent and Smith 2004).

Because of the ubiquitous distribution of tritium in the body, all organ doses will be identical to the annual dose. For the Interactive RadioEpidemiological Program (IREP), assume that all internal doses are "chronic" exposure rates. The radiation type for tritium is "electron < 15 keV." The dose distribution type is "constant." The annual dose is put in the parameter 1 column. Parameters 2 and 3 are not used in IREP calculations.

5.2.2.5 *In Vitro* Bioassay for Uranium

Environmental concentrations of uranium are highly dependent on geographic location (BN 2003b). The activity of uranium in urine samples will vary a great deal between individuals who work in different areas and, in a given area, will vary primarily as a function of the individual's primary water

source. Well water typically contains more uranium than water from public supplies. In addition, an individual's level of uranium excreted can vary significantly from day to day. Because the sources of uranium on the site are typically depleted uranium (DU) or highly enriched uranium (HEU), the ratio of ^{238}U to ^{234}U can be used as an indicator. DU has a $^{238}\text{U}:$ ^{234}U ratio range of about 3 to 10; HEU has a ratio of about 0.1 or smaller. Natural uranium in urine has a ratio of about 1, with ^{234}U activity often somewhat larger than that of ^{238}U (as much as a factor of 2 or 3 is not unusual). If the $^{238}\text{U}:$ ^{234}U ratio does not indicate a likely DU or HEU intake and there are no field indicators that an intake occurred, the result might be judged unlikely to have come from an occupational intake and no dose is calculated (BN 2003a).

The exception to DU and HEU being the source of uranium for exposures is at the Waste Examination Facility (WEF). Uranium at the WEF is likely to be the ^{233}U radionuclide, which can be present in some of the transuranic waste being characterized at the facility. The alpha particle detected for identifying ^{233}U is similar in energy to the ^{234}U alpha particle and alpha spectrometry is not capable of resolving the alpha particles from the two radionuclides. Therefore, ^{233}U detected in a sample is reported as ^{234}U , and a sample from personnel working at the WEF with an unexpectedly low $^{238}\text{U}:$ ^{234}U ratio (e.g., less than 0.3) potentially indicates an occupational intake.

In the case of positive ^{235}U results without a ^{234}U positive result in the sample, the positive ^{235}U result is considered to be a false positive because the ^{234}U is always a few times greater in activity concentration than the ^{235}U throughout the range of enrichment from DU to HEU. The only possible exception to this is if the uranium has been enriched above 6% ^{235}U weight percent from the laser isotopic separations (AVLIS) method (Rich et. al. 1988, as cited in BN 2003a). This type of enriched uranium currently is not known to exist at the NTS.

The LASL 1954 fluorophotometric method for uranium in urine was based on the intense yellow-green fluorescence (the principle line of which is reported to be at 555 μm) produced by traces of uranium fused in sodium fluoride. It was sensitive to concentrations of uranium from 10^{-5} to 5×10^{-10} gram per 0.25 gram of sodium fluoride, with a precision of $\pm 10\%$. The tolerance for normal uranium in urine at LASL was 100 $\mu\text{g/L}$ (LASL 1954). In 1958, the same method was sensitive to concentrations of uranium from 10^{-10} to 5×10^{-11} gram per 0.25 gram of sodium fluoride, with a precision of $\pm 10\%$ (LASL 1958a).

An ion exchange method for uranium alpha activity in urine was published by LASL in 1958. The uranium was concentrated from urine by coprecipitation with alkaline earth phosphates. The precipitate was dissolved in 8N hydrochloric acid and the complex uranium chloride anion was separated by passing the solutions through an anion exchange column. The uranium was eluted with 1N hydrochloric acid, which was evaporated and taken up in nitric acid, and plated directly on stainless-steel counting discs. The alpha activity was determined by counting in a low-background proportional counter. The recovery was approximately 90% in the range from 25 to 50 dpm/L. A set of nine analyses could be completed in 3 days with the equipment available at LASL. Rechecks were requested if the sample showed more than 50 dpm/L (LASL 1958a).

LASL also had an extraction method for uranium alpha activity for 1958. The urine was ashed with nitric acid and the salts were dissolved and made approximately 1N with nitric acid. The uranium was extracted from the acid solution of the salts with di-n-butyl ortho-phosphoric acid in carbon tetrachloride. The phosphoric acid was evaporated on platinum plates, fused, and alpha-counted with a low-background proportional counter. The method had an accuracy of $84 \pm 14\%$ with 1 to 10 dpm/L of enriched uranium. At higher concentrations the recovery approximated 100%. Approximately 25 analyses could be completed in 1 day by one worker with the equipment at LASL (LASL 1958a).

In 1961, REECO laboratory operations published a sensitivity limit for uranium of 0.03 dpm/sample (REECO 1961). HEU analysis involved coprecipitation of the uranium with alkaline earth phosphates from an ammonium hydroxide solution, filtration, destruction of organic material by wet-ashing, purification by extraction and anion exchange, electrodeposition, and alpha counting with a gas flow proportional detector (Geiger and Whittaker 1961).

A method for total uranium by fluorometry and alpha spectrometry and for ^{235}U (anion exchange, purification, electrodeposition, alpha spectrometry) was listed in REECO documents published from 1981 to 1983 (REECO 1968-1991a).

As described in REECO (1993a), ^{235}U and ^{238}U were measured in urine by alpha spectroscopy (MDAs of 0.01 and 0.02 pCi/L, respectively). "Typical" MDAs were listed as 0.1 pCi/L. These radionuclides were also measured by fecal sample analysis (MDA of 0.008 pCi/g). The routine bioassay was a quarterly urine sample and the special bioassay was an additional 24-hr urine sample and a prompt lung count. Elemental uranium was determined by a fluorometer, and the MDA was listed as 5 $\mu\text{g/L}$ for chemical analysis with a "method" MDA of 0.2 pCi/L. Attachment 5D, Table 5D-6 lists historical detection thresholds for various periods. Table 5-1 lists current MDAs for uranium isotopes.

BN (2003b) stated that approximately 60 nonoccupationally exposed adults residing in the southwestern portion of Nevada were sampled to determine the natural uranium background. The analyses showed that the distribution of uranium approximated lognormal but tended to be slightly skewed at high concentrations. Geometric means for ^{234}U , ^{235}U , and ^{238}U were 0.025, 0.0067, and 0.016dpm/L, respectively. The 99.9th percentile of the sample result distribution used to separate environmental from potential occupational exposures was observed to be 0.208 dpm/day for ^{234}U , 0.153 dpm/day for ^{238}U , and at 0.446 dpm/day (212 ng/L or 0.2 pCi/sample) for total uranium.

5.2.2.6 *In Vitro* Analysis for Strontium

Strontium (^{90}Sr) is present in the environment as a result of atmospheric weapons testing. There has been no strontium background study at NTS. Excretion due to environmental exposure is likely to be small (<MDA), but is highly variable due to varying diets, and might be as large as the current MDA. However, because the environmental component is not established, any detectable radiostrontium should be assumed to result from an occupational intake.

In 1961, REECO laboratory operations published a sensitivity limit for ^{90}Sr in urine of 25 pCi/sample (detected with an accuracy of $\pm 10\%$ at a 90% confidence level, counter background of 17 cpm, and a 60-minute counting period) (REECO 1993b). The analysis involved the following steps: sample preparation, solvent extraction, time delay for ^{90}Y buildup, solvent extraction of ^{90}Y , and beta count (REECO 1961). Note that this method did not account for any ^{89}Sr present at intake.

In 1993, gas-flow proportional beta counting was used to quantify ^{90}Sr in urine at NTS. The method had an MDA of 0.8 pCi/L, but did not differentiate between ^{89}Sr and ^{90}Sr . Strontium-90 decays to produce ^{90}Y , which is a beta emitter. Strontium was extracted from ^{90}Y and other contaminants. The first count included betas from ^{89}Sr plus ^{90}Sr . Then the ^{90}Y was allowed to grow in for 2 weeks and chemically separated and counted. The second count included only ^{90}Y . Strontium-90 was calculated from the ^{90}Y value, and then subtracted from the first count to obtain the ^{89}Sr value.

REECO (1993a) stated that "typical" MDAs for a 1,000-mL sample counted for 100 minutes were around 0.8 pCi/L. The routine bioassay for ^{90}Sr was a quarterly urine sample, and the special bioassay was a prompt urine sample. Attachment 5D, Table 5D-6 lists historical detection thresholds for various periods. Table 5-1 lists the current MDA.

5.2.2.7 *In Vitro* Analysis for Thorium

The LASL (1954) method for ^{230}Th in urine was as follows: The urine sample was ashed with nitric acid; the thorium was coprecipitated with bismuth phosphate, dissolved in hydrochloric acid, and then coprecipitated with lanthanum fluoride. The lanthanum fluoride precipitate was slurried on a stainless-steel plate and counted for alpha activity with a low-background proportional counter. Natural thorium, plutonium, some americium, curium, actinium, and neptunium were carried through this determination. Quantities of the order of 0.5 d/m of ^{230}Th could be determined with this method. In 1993, the ^{230}Th fecal MDA was 0.01 pCi/g (REECo 1993a). Attachment 5D, Table 5D-6 lists historical detection thresholds for various periods. Table 5-1 lists the current MDA.

5.2.2.8 *In Vitro* Analysis for Radium

Radium in urine was coprecipitated with barium as the sulfate. Polonium was removed by deposition on silver foil. The precipitate was slurried onto a stainless-steel plate and the alpha activity counted using a methane flow proportional counter. Uranium in amounts usually found in urine did not interfere. The recovery was approximately 96-97%. Approximately 10^{-13} g (0.88 dpm) of radium could be detected with this method if the sample was allowed to come to full equilibrium. It has been estimated that about 0.01% of the body radium [or about 10^{-11} g (22 dpm) at tolerance] was excreted daily in urine 6 months after exposure and that 0.0005% [or 5×10^{-13} g (1 dpm) at tolerance] was eliminated daily by long-standing chronic radium poisoning cases. Normal radium excretion of unexposed humans is on the order of 0.2 to 0.4 dpm/day at radioactive equilibrium. A 24-hr sample was usually collected for radium analysis (LASL 1958a). REECO (1993a) states that the radiobioassay of urine for ^{226}Ra had an MDA of 300 pCi/L, and the fecal sample MDA was from 0.04 to 0.4 pCi/g. Table 5D-6 in Attachment 5D lists historical detection thresholds for various periods. Table 5-1 lists current MDA value.

5.2.2.9 *In Vitro* Gross Fission Product Analysis

For interpreting results from the fission product urinalysis, strontium, barium, europium, zirconium, and niobium radionuclides concentrate primarily in the bone, with ^{90}Sr providing the largest dose rate. Cerium, lanthanum, and promethium concentrate primarily in the liver, with some concentration in the bone, with ^{144}Ce providing the largest dose rate. Cesium and ruthenium are assumed to be uniformly distributed in the whole body.

Note: If the worker had a whole-body count during any year and had a detectable fission product urinalysis result, determine intakes of ^{137}Cs , ^{106}Ru , or other gamma-emitting fission products from the whole-body count. Whole-body counting began at offsite locations in 1964 (Pan Am 1967). Onsite capability started in 1967 (Teasdale 1985). (See Section 5.3 for additional *in vivo* information and references.)

The LASL (1958a) method for gross beta activity in urine was as follows: The nuclides of ^{90}Sr - ^{90}Y , ^{140}Ba - ^{140}La , ^{144}Ce - ^{144}Pr , ^{89}Sr , and gross fission products were determined as alkaline phosphate precipitates and counted directly. The nuclide usually could be identified from the exposure history and decay characteristics. The recovery was $80 \pm 5\%$. If indicated, a gross gamma count was performed on 500 mL of urine using liquid scintillation techniques. LASL (1958a) states that if the gross beta (^{90}Sr - ^{90}Y , ^{140}Ba - ^{140}La) result is higher than 200 dpm/L, exposure should be suspected and investigated.

In 1961, REECO analyzed urine samples for GFP before performing specific analyses (Geiger and Whittaker 1961). Strontium, barium, lanthanide rare earths, and other fission products were

coprecipitated with alkaline earth phosphate from an alkaline solution. The precipitate was assayed without further chemical treatment for total beta emitters to obviate, in many cases, detailed radiochemical determination. If insignificant quantities of radioactive material were detected, no additional analyses were performed. Analyses for specific radionuclides could be requested if the gross beta activity exceeded the following control limits:

- Exposure to fission product mixture less than 3 months old – control limit was 1.0 pCi/mL.
- Exposure to fission product mixture greater than 3 months or age unknown – control limit was 0.1 pCi/mL.
- Anything above 20% of the control limit should have been recorded on the person's bioassay data card.

REECo (1961) stated that cesium was not recovered by this procedure (i.e., coprecipitating fission products on alkaline earth phosphates from an alkaline solution and counting the sample package for gross beta activity), but exposure to cesium without exposure to other fission products was not probable in most NTS areas. The GFP procedure provided a screening technique that resulted in a considerable saving of time and money, according to REECo records.

Specific analysis for ^{137}Cs in urine in 1961 involved sample preparation, cesium-phosphotungstate precipitation, and a specific gamma count (REECo 1961). NTS used gamma pulse height analyses to evaluate samples unless there was interference from higher energy gamma emitters, in which case NTS separated cesium as cesium-phosphotungstate followed by pulse height analysis. The sensitivity was listed as about 10 pCi per sample (REECo 1961). From 1982 to 1987, the detection limit for GFP (beta) in urine was 1×10^{-10} $\mu\text{Ci/mL}$ (REECo 1977-1987).

REECo (1993a) states the term *gross fission product* was used rather than *gross beta*. This analysis actually was used for all beta-emitting radionuclides except the alkali group (e.g., ^{134}Cs , ^{137}Cs , and ^{40}K). The intent was to eliminate naturally occurring ^{40}K from the sample because the concentration of potassium in urine varies widely. Detection for beta activity was done in a gas-flow proportional counter, and the MDA was listed as 3.0 pCi/L (250-mL sample counted for 100 minutes). The specific MDA for ^{137}Cs in urine was listed as 100 pCi/L (REECo 1993a).

Current guidance (BN 2003a) includes the following information for nonoccupational exposure. Cesium-137 is a product of fallout from weapons testing. Detection of background levels depends on the individual's diet, the primary source of ^{137}Cs being game meat (elk, deer, etc.). As with ^{90}Sr , background levels will generally result in small doses. Cesium sampling is performed quarterly. Dose reconstructors need not consider results less than the derived screening level for investigation of intake or assignment of dose. If a ^{137}Cs result is above the derived screening level, but there are no field indicators that an intake might have occurred, it is advisable to question the individual regarding the possibility of recent consumption of wild game meat. This information can support a decision on whether an occupational intake has occurred. Cesium-137 is a radionuclide of concern at NTS because of the nature of the work performed. It is appropriate to address only ^{137}Cs levels above the annual reported levels in NCRP (1987a) by evaluating the intake using the Integrated Modules for Bioassay Assessment (IMBA) computer program (see Table 5-9). Attachment 5D, Table 5D-25 identifies common radionuclides that are a significant part of fission products up to 1 year old.

Dose reconstructors should select fresh or aged fission products based on what is most claimant-favorable for the cancer diagnosed. The bioassay records do not indicate fresh or aged. Dose reconstructors should check the allowable limits for fission products identified as a dose concern.

They should review the “Other Monitoring” section of the case file for additional data, such as access logs and radiological control technician log books, and review test information to determine if they need to consider any fission products (Table 5D-25) unique to the test based on the cancer location. They might need to request additional records to determine aged or fresh fission products.

5.2.2.10 *In Vitro* Analysis for Gamma Emitters

In 1961, certain samples were examined with a gamma spectrometer to identify gamma-emitting components (REECo 1961). Quantitative estimates were made in some cases. Special chemical separations schemes were devised as needed based on the results of the initial pulse height analysis. Reportedly, almost any gamma component could be identified by a combination of pulse height analysis, chemical purification, and beta measurements. From 1982 to 1987, the detection limit for gamma emitters in urine was 5×10^{-8} $\mu\text{Ci/mL}$ (REECo 1977-1987).

All urine samples were counted by gamma spectroscopy in 1993 to determine the concentration of gamma-emitting radionuclides in the energy range from 40 to 2,000 keV (REECo 1993a). There was no special sample treatment except transfer to a 500-mL bottle. Counting was done in a fixed position with any of several coaxial high-purity germanium detectors and multichannel analyzer gamma spectroscopy systems. The MDAs varied according to the analyte and the specific detector system used. Table 5D-6 in Attachment 5D lists historical detection thresholds for various periods. Table 5-1 lists the current MDA value.

5.2.3 Correcting for Urinalysis Volume

The 1961 REECo procedures stated that urine samples were usually received in a kit of four bottles (Geiger and Whittaker 1961). When the solution in the four bottles was combined and mixed thoroughly, the sample represented an “equivalent 24-hour sample.” The purpose of this procedure was to provide a uniform method for obtaining homogenous aliquots for the determination of plutonium, uranium, and fission products.

As discussed in REECo (1993a), types of indirect bioassay samples include total 24-hour urine collection (including collection at work) and spot urine samples (single void). In general, urine data were normalized to total 24-hr excretion. Provided the sample was collected properly, a total or simulated 24-hr urine sample result was used without further normalization. A proper 12-hr result was normalized by doubling the result.

NTS collected plutonium samples for 24 hours. In effect, the collection procedure normalized them to 24-hour samples. Screenings for all other radionuclides were normalized to 24-hour samples in the bioassay records (Arent and Smith 2004).

5.2.4 Fecal Sample Analysis

Fecal sample analyses were performed at NTS as a special bioassay, usually if there were other indications that internal contamination had occurred. REECo (1993a) indicates that 5-g ash samples were analyzed by alpha spectroscopy for radium, uranium, plutonium, and americium. Alpha spectroscopy began in the late 1970s as a method of measurement. In the 1990s, some environmental projects to remediate plutonium-contaminated sites required periodic fecal bioassay.

REECo (1993a) states that the material measured in a fecal sample is the sum of excretion from the systemic body, translocation from the lungs, and unabsorbed ingested material accumulated over a certain period. This period could be difficult to specify because there can be considerable and

variable lag time in the gastrointestinal tract. To minimize this problem, fecal samples should be collected over a period that is long in comparison to the gastrointestinal tract lag time, a week for example, and the data should be evaluated with an accumulated feces excretion model. This is particularly important in the first week following an intake. A single isolated fecal sample should be assigned the period between voids, and if this is unknown, a period of 1 day should be used.

5.3 IN VIVO MDAS, COUNTING METHODS, AND REPORTING PRACTICES

During atmospheric testing (1951 to 1962), *in vivo* methods of analysis were in place but were secondary to other measurements. NTS did, however, have a bioassay program that included whole-body, thyroid, lung, and wound counting. In 1964, LASL identified whole-body counting for radionuclides that deliver whole-body doses. There is no description of where these counts took place; the first reference to an onsite capability is 1967 (Teasdale 1985). It is assumed that previous counts occurred at Los Alamos. It is also assumed that the University of California-Berkeley and LLNL performed counts at their facilities for their employees who spent time at NTS. Any records that pertain to REEC Co employees have been retrieved from other locations for inclusion in the REEC Co (now BN) record archives. Other reports/memoranda indicated that prior to 1967 whole-body counting was performed by contracted portable units on three occasions in a 2-year span. No information was provided indicating the type of job categories measured or if this was initial routine surveillance.

In general, the Whole-Body Counter (WBC) was used to measure amounts of individual gamma emitters in the body. The scanner measured radionuclides in specific parts of the body (e.g., ^{131}I in the thyroid). A mobile scanner was available for use in the field; however, under most circumstances the field instrument that was available was used.

Glasstone (1971) mentions a stationary WBC and scanners at the Southwestern Radiological Health Laboratory, which was used for U.S. Environmental Protection Agency/Public Health Service offsite monitoring. The only significant time this facility was used for NTS workers was during the YUBA incident (June 1963), but the information obtained was not used. Workers involved in that incident were sent to Donner Laboratory in California for whole-body counting.

Since 1967, whole-body counting has been performed at NTS. This is when a shadow shield-type counter was installed by Pan American Airways at the Nuclear Rocket Development Station as part of the Nuclear Space Propulsion Program (Teasdale 1985). This facility was transferred to NTS REEC Co dosimetry in 1974. Routine counting was performed for drillers, miners, and radiation monitoring personnel beginning in mid-1975. In 1977, NTS considered adding lung counting for low-energy X-ray detection. However, it was not cost-effective to upgrade the existing facility. Construction of a new facility began in March 1979, and the facility operated from February 1981 until shutdown in 1999. See Attachment D, Section 5D.3.1 for whole-body counting MDAs.

On the basis of this information, it is assumed chest counting at the new facility used Phoswich detectors. The first reference to chest counting was found in the 1983 REEC Co Standard Procedures, Chemical and Radiological Analysis (REEC Co 1968-1991a). *In vivo* count measurements over minimum reporting limits are included in the annual report and were microfilmed for the historical files. Individual hard copies of whole-body counts were included in the files provided to DOE.

As identified in REEC Co (1993a), the background subtraction algorithm used for these counts is designed to be invariant with respect to counting rates and the slope of the continuum. Identified photon peaks are corrected for natural background photon sources based on 24-hour environmental background counts. The background correction is not the traditional channel-by-channel subtraction,

but rather the subtraction of normalized 30-minute background radioactivity (nanocurie amounts) from the radioactivity (nanocurie amounts) in the individual's count.

The MDA is presently calculated as a function of the gross spectral counts and the full-width-half-maximum (FWHM) energy and is consistent with American National Standards Institute (ANSI) standard N13.30 (ANSI 1996). The MDA (assuming a 5% probability of either a type 1 or type 2 error) is defined as:

$$MDA = \frac{[C_1 + C_2 * (\sum_{c-w}^{c+w} Y_i)^{0.5}]}{(T_1 * E * Y * k)}$$

where:

Y_i = number of counts in gross spectrum
 Y = photon yield, 1.0 for "unknown" peaks
 c = centroid energy
 w = window function (area window 2), $w = 0.64 * FWHM$
 T_1 = count collection time
 E = detection efficiency at centroid energy of interest
 C_1 = reject MDA constant, user variable, set at 2.71
 C_2 = reject MDA sigma, user variable, set at 4.66
 k = unit conversion factor

5.3.1 Whole-Body Counting

Whole-body counting is not identified in the analysis capability at NTS from 1951 to 1973 but, as noted above, did occur prior to 1967. Helgeson Nuclear Services performed the first whole body counting at the NTS. It used a shadow shield that was transported in a semitrailer (Helgeson 1967). The shadow shield employed an 8" x 4" NaI crystal with a 7.6% resolution for ^{137}Cs . The crystal was placed 10.5" above a moving bed. The count time was 8 minutes (480 seconds). Helgeson estimated the sensitivity at two confidence levels – 50% and 99%. Table 5-3 summarizes these activities.

Table 5-3. Minimum detectable activities for the Helgeson shadow shield.

Isotope	Background	Photons		MDA - nCi	
		Energy (keV)	Intensity, I_γ photons/decay	50%	99%
Cs-137	125	662	0.851	1.0	3.0
Zr-95	95	724	0.81	0.7	2.1
		757	1.00		
Co-58	57	811	0.994	0.8	2.3
Zn-65	54	1,116	0.506	1.5	4.3
Co-60	35	1,173	1.00	0.4	1.3
		1,332	1.00		
K-40*	71	1,461	0.107	7.1 (8.3 gK)	21.4 (24.9 gK)

* 1 gK = 0.000118 g K-40 = 8.58138E-10 Ci = 0.858138 nCi/gK

The 99% MDA value was recommended. For the range from 662 keV to 1,461 keV, the maximum MDA is $2.55 \text{ nCi} \times I_\gamma$. For nuclides not listed in Table 5-3 that have primary gamma energies ≥ 100 keV, the MDA can be determined for the primary photon with an intensity, I_γ , by $3 \text{ nCi}/I_\gamma$. Other

radionuclides that dose reconstructors might need to address were calculated based on this formula and are listed in Table 5D-7.

A draft summary report provided by NTS indicated the completion of more than 300 whole-body counts prior to 1967 on this portable system. The calibration for this contracted system was performed by distributing 200 point sources (chips) through a standard man-sized masonite phantom. Scatter factors and efficiency constants were determined by counting ^{65}Zn , ^{40}K , ^{137}Cs , ^{106}Ru , ^{60}Co , ^{54}Mn , and mock ^{131}I . Detection consisted of a 3- × 3-in. NaI(Tl) crystal on the first visit and an 8- × 4-in. NaI(Tl) crystal on the remaining visits. The standard counting time was 10 minutes. Table 5-4 lists the results of these counts.

Table 5-4. Frequency (number of workers) of nuclide appearance and concentration in subjects counted before January 1967.

Nuclide	Body burden			
	≤ 0.02 μCi	≤ 0.05 μCi	≤ 0.1 μCi	≤ 0.5 μCi
Zr-Nb-95	98	43	16	6
Ta-182	102	17	3	1
Ru-103, -106	34	1	-	-
Ba-La-140	26	2	-	-
I-131	2	-	-	-

Source: Pan Am (1967)

In January 1967, Pan American acquired and installed the shadow shield WBC. The detection system consisted of a 3- × 3-in. NaI(Tl) crystal in conjunction with a Nuclear Data 512-channel analyzer. A count time of 20 to 40 minutes was used. Calibration of this system consisted of points at 0.36, 0.66, 1.12, and 1.33 MeV corresponding to photons emitted from mock ^{131}I , ^{137}Cs , ^{65}Zn , and ^{60}Co . The standards discussed above were placed in a standard man-sized masonite phantom for counting. The detection efficiency for nuclides appearing between these points was extrapolated from the results. No efficiencies were reported, but research at the time indicated that, depending on body weight (120- to 185-pound range), the count rate could change by a factor of 1.57 (Pan Am 1967). Table 5-5 lists results of the initial whole-body counts performed by this system.

Table 5-5. Frequency (number of workers) of nuclide appearance and concentration in subjects counted from January to August 1967.

Nuclide	MPBB and organ of concern ^a	Body burden (μCi)			
		≤ 0.02	≤ 0.05	≤ 0.1	≤ 0.5
I-133	0.3 / thyroid	3	3		
Te-I-132	0.3 / thyroid	3	1		
I-131	0.7 / thyroid	4	1		
Ru-103, -106	(b) / kidney	27	25	11	3
Zr-Nb-95	20 / total body	36	24	7	4
Ba-La-140	4.0 / bone	3	1		
Ta-182	7.0 / liver	1	-		

Source: Pan Am (1967)

- The MPBB and organ appear in handwritten notes on the original table. Documentation for the quantities and organs of concern has not been found.
- The handwritten notes indicate 20 μCi for Ru-103 and 3 μCi for Ru-106.

REECo standard operating procedures (REECo 1968-1991a) for chemical and radiological analysis describe the preliminary calibration of whole-body counts. The phantom consisted of polyethylene blocks stacked in a 3 × 5 × 5 arrangement with the source placed in one of the blocks in the center column. The source stack was moved over an x,y grid marked on the bed. By varying the position of

the source over the x,y,z coordinates, it was possible to simulate a source distributed throughout a phantom that would have been 5.9 in. thick, 22.4 in. wide, and 74.6 in. long. The average density of the phantom was 0.698 g/cm³ [Reserved]. Table 5-6 lists the sources used for this calibration.

Table 5-6. Preliminary calibration sources for 1977 through 1981.

Radionuclide	Energy (keV)	Branching ratio gamma/disintegration
Cr-51	320	0.09
Mn-54	835	1.00
Zn-65	1,115	0.49
Zn-65	511	0.34
Y-88	898	0.91
Y-88	1,836	1.00

Source: REEC Co (1968-1991a)

The efficiencies for counting gamma rays from this phantom are described by the equation:

$$\text{Efficiency}_{(\text{counts/disintegration})} = 0.0089e^{-0.000175 * E}$$

where E is energy in keV (REEC Co 1968-1991a).

Sensitivity was defined as activity that is detectable with a relative 2 sigma error of $\pm 20\%$. Background counts were 1,000 seconds. At a background count of 45 counts per second, the sensitivities for potassium and cesium are listed in Table 5-7.

Table 5-7. Whole-body count sensitivities.

Radionuclide	For 20-minute count (μCi)	For 40-minute count (μCi)
K-40 ^(a)	0.104	0.091
Cs-137 ^(b)	0.011	0.010

a. Source: REEC Co (1968-1991a).

b. Potassium activity in standard man is about 0.12 μCi .

c. The MPBB of Cs-137 in standard man is 30 μCi .

Potassium-40 activity in the body was routinely calculated and normalized to the 70-kg standard man. The results were hand-recorded on the printout of the whole-body count. Abnormal results, defined as those in excess of 200 grams potassium, standard man equivalent, were reviewed by a senior health physicist to determine the need for further action.

Starting in 1978, REEC Co Radiological Safety Division procedures described whole-body counting. Documentation indicates that from 1978 through 1980, whole-body counting used a 20.3- by 10.1-cm-thick NaI crystal housed in a shallow shield. (The detection energy range was 100 to 2,000 keV.) The system was connected to a Nuclear Data Analyzer, Model 2400, and the data printed through a teletypewriter. The monitored individual lay supine on the table during the counting procedure, which took 1,300 to 1,400 seconds. These counts were performed in street clothing; however, if the count level in the ⁴⁰K channels (12-160) appeared to be excessive (see Table 5-8), a recount was performed after street clothes were exchanged for a paper suit. In addition, the background of the bed was rechecked.

By 1981, the use of a paper suit became standard practice and the count time changed to 2,000 seconds. To provide lower background counts, a "vault" was constructed of pre-World War II steel plates covered with graded shielding consisting of 3 mm of lead, 0.6 mm of tin, and 0.25 mm of zinc.

Table 5-8. Examples of whole-body count “alert” levels.

Time (s)	Weight (lb)				
	155	175	195	215	235
	Permissible counts				
1,340	55,610	56,400	57,200	57,985	58,770
1,370	56,855	57,663	58,481	59,283	60,086
1,400	58,100	58,925	59,761	60,581	61,401
1,440	59,760	60,610	61,469	62,312	63,156

A 29.2-cm diameter by 10.1-cm thick NaI(Tl) detector was connected to a Canberra Series 30 multichannel analyzer. Radon daughter products in the air were reduced by a high-energy particulate air (HEPA) filtration system.

By 1983, two types of detection equipment were identified: (1) 29.2 cm by 10 cm thick NaI(Tl) crystal; and (2) Phoswich counting systems used to detect X- and gamma rays of less than 100 keV. These detectors consisted of a 12.7-cm diameter by 5-cm thick CsI(Na) crystal optically coupled to a 12.7-cm diameter by 3-mm thick NaI(Tl) crystal at the incident energy end. The crystals were hermetically sealed in a stainless-steel housing with a 0.25-mm beryllium entrance window. Electronics were modified to capture the signal from both parts of the Phoswich detector. Once-a-year calibration used a Bomab water phantom constructed of polyethylene that weighed 70 kg when filled with water and National Bureau of Standards (NBS)-traceable solutions of four monoenergetic gamma ray emitters. Phoswich detector system calibration used a tissue- and bone-equivalent torso phantom designed by LLNL. Tissue-equivalent plastic dosed with ^{241}Am , $^{238/239}\text{Pu}$, and DU was used to construct the lungs and liver. Curves were generated for the phantom composition of muscle, bone, and varying thicknesses of fat and muscle.

The system used in 1993 was calibrated for whole-body counting and used for whole-body, thyroid, and wound counting. The 70% relative efficiency detector was used routinely in this period with a 27% relative efficiency detector as a backup. The 70% relative efficiency detector is referred to as an “XtRa-extended range,” closed-end, coaxial detector with a diameter of 72.4 mm and length of 69.5 mm. The backup detector is a p-type, closed-end, coaxial detector with a diameter of 61.3 mm and length of 56 mm. Both detectors measured photons with energies in the calibration range of 50 keV to 2.5 MeV.

Workers were counted while sitting in a reclining chair. The chair back was about 30 degrees from vertical so the back and the seat yield an arc with the detector of 50 cm, providing a full view of the body trunk in relation to the detector. Whole-body counts were performed:

1. For new employees who were likely to be included in the routine bioassay program
2. For a current employee who changed to a job classification that required a routine bioassay program
3. Annually for employees who were currently on the routine whole-body count and bioassay program
4. For terminating employees
5. For employees who had a suspected intake of radioactive material, particularly gamma-emitting fission and activation products.

The routine time for the whole-body count was 20 minutes.

The suggested priority for dose reconstructors with regard to MDAs and other detection thresholds is to use (1) the limit on dose report if available, (2) Attachment 5D, Table 5D-7 values, or (3) an appropriate published value from another DOE site or an applicable value referenced in the literature.

Effect of Cesium on Whole-Body Counting

Fallout affected everyone in North America; body burdens of ^{137}Cs measurable in WBCs were common in the 1960s and 1970s. NCRP (1987b) provides mean body burdens of ^{137}Cs for the United States for the years most likely to produce interference with occupational whole-body count results. Table 5-9 lists those burdens. If whole-body count results show detection of only ^{40}K and ^{137}Cs and the ^{137}Cs result is less than the values listed in Table 5-9, the ^{137}Cs results can be assumed to be due to fallout. There are two exceptions:

- If other fission or activation product radionuclides are present in either the whole-body count or a recent urinalysis for ^{90}Sr or any of the iodines, it is claimant-favorable to assume the ^{137}Cs is from occupational sources.
- Depending on the level of ^{137}Cs in the whole-body count and the worker job description, the exposure might be based on the job.

5.3.2 Chest Counting

The first reference to lung or chest count was found in a REECO procedure dated July 1983 (REECO 1977-1987). This was performed with two - 5 in. diameter Phoswich detectors, which are used primarily to detect low-energy X-ray and photon emissions from “heavy” elements such as ^{239}Pu , ^{241}Am , etc. The total estimated sensitive area of the detectors was 500 cm^2 .

One complicating factor in the measurement of low-energy photon emissions from the lung is the absorption of photons in the tissue overlying the lung – adipose (fat), muscle, cartilage, and bone. The thickness of these tissues and, as a consequence, the attenuation can vary significantly from one individual to the next. This is particularly serious for detection of the 17-keV plutonium X-rays. At this energy, 6 mm of muscle can attenuate half of the transmitted X-rays. In recent years, sophisticated ultrasound measurement techniques have been applied for accurate determination of the effective chest wall thickness (CWT). In the early days of lung counting, height/weight relationships had been used to estimate the CWT, but these were crude and could easily have led to errors of a factor of 2 or more. The situation is made more difficult by the significance in attenuation properties of the three primary tissues of concern – adipose, muscle, and bone. Although bone is obviously denser than adipose or muscle, the attenuation differences between those tissues can be significant. It is, therefore, not sufficient for dose reconstructors to determine the CWT; they must also estimate the relative fraction of each tissue.

The Livermore torso phantom was fabricated to provide calibration information for transuranic nuclides in the lungs of individuals with a range of body statures and chest wall tissue compositions. The phantom set consists of the basic torso and three sets of chest plates or overlays. The purpose of the chest plate sets is to simulate tissue compositions or 100% muscle, 50% muscle – 50% adipose, and 13% muscle – 87% adipose. The phantom covers a CWT range of about 15 cm without chest plates to about 40 with the thickest overlays. There is some set-to-set variation.

The *International Calibration of Detectors Systems for the Measurement of Low Energy Photon Emitters In Vivo* (IAEA, 1991) contains background and efficiency information for the Phoswich detectors, using the Livermore phantom. In this comparison, ^{239}Pu detection was based on uranium L-X-rays, with a total intensity, I_x , of 0.0457 X-rays per disintegration. These measurements can be

Table 5-9. Mean body burdens of ¹³⁷Cs from fallout in the United States.

Year	NCRP determined body burden (nCi) ^a	Year	Calculated body burden (nCi) ^b
1953	0.27	1978	1.07
1954	1.1	1979	1.05
1955	2.2	1980	1.03
1956	4.3	1981	1.00
1957	5.1	1982	0.98
1958	6.5	1983	0.96
1959	8.1	1984	0.94
1960	6.8	1985	0.91
1961	4.6	1986	0.89
1962	6.0	1987	0.87
1963	11	1988	0.85
1964	19	1989	0.83
1965	16	1990	0.81
1966	9.7	1991	0.80
1967	5.6	1992	0.78
1968	3.5	1993	0.76
1969	2.7	1994	0.74
1970	2.7	1995	0.73
1971	2.7	1996	0.71
1972	2.7	1997	0.69
1973	2.7	1998	0.67
1974	1.6	1999	0.66
1975	1.1	2000	0.65
1976	1.6	2001	0.63
1977	1.1	2002	0.62
		2003	0.60
		2004	0.59

- From NCRP (1987b). NCRP only published Cs-137 data through 1977. Extrapolated the date range from 1977 to present.
- After 1978, the values have been calculated using the value for 1977 as the initial activity and applying the activity equation $A = A_0 e^{-(0.693/T_{1/2})t}$. Comparing any cesium bioassay results with these values is claimant-favorable because it does not take into account the Chinese atmospheric test or the accident at Chernobyl (i.e., any exposure for these events could be in the results for a worker, but will not be attributed to these events; it will be attributed to occupational exposure if the concentration is over the value found in Table 5-8 for the year the whole-body count was taken).

used to estimate the expected range of MDA values. Table 5-10 lists data for the measurements with ²³⁹Pu-loaded lungs. These are the results of one participating facility with higher than average backgrounds and lower than average reported detection efficiencies. However, these values reflect those that are could have been encountered in a routine measurement program. Table 5-11 lists background and efficiency data for ²³⁹Pu counting.

Table 5-12 lists background and efficiency data for ²⁴¹Am counting. The MDA values are markedly lower than those for ²³⁹Pu counting because:

- The reported counting efficiencies are about 40 – 200 times that for ²³⁹Pu.

Table 5-10. Uranium L X-ray intensity for decay of ^{239}Pu .

X-ray Energy - keV	X-ray Intensity – X-rays per disintegration
11.620	0.00106
13.442	0.00175
13.618	0.0157
15.400	0.00042
15.727	0.000296
16.410	0.0041
16.577	0.00046
17.068	0.00087
17.222	0.0157
17.454	0.00042
20.169	0.0038
20.487	0.00019
20.715	0.00018
20.844	0.0008
Total	0.045746

Source: LBNL (2004).

Table 5-11. Background and efficiency data for ^{239}Pu detection using Phoswich detectors (12-25 keV).

	CWT-cm	Normalized efficiency counts/cm ² /10 ⁶ photons	Efficiency* counts/photon	Efficiency** cpm/nCi	Estimated MDA - nCi (40 minute count)
Torso	1.51	4.07	0.00204	0.207	21
13% muscle – 87% adipose	2.12	2.73	0.00136	0.138	32
	2.79	1.68	0.00084	0.0852	51
	3.34	1.24	0.00062	0.0629	69
	40.3	0.84	0.00042	0.0426	102
50% adipose – 50% muscle	2.15	2.41	0.00120	0.1212	36
	28.0	1.57	0.000785	0.0796	55
	3.24	1.03	0.000515	0.0522	83
	3.99	0.68	0.00034	0.0345	126
100% muscle	2.11	2.02	0.00101	0.102	42
	2.75	1.18	0.00059	0.0599	73
	3.32	0.79	0.000395	0.0401	109
	3.96	0.45	0.000225	0.0228	191
Background	0.070 cpm/ cm ²				

* For 500 cm² detector

** L-xray intensity = 0.0457 L-xrays/disintegration

It has not been determined if lung counting was a viable bioassay method prior to construction of the whole-body counting facility from pre-World War II steel. However, safety reports from the atmospheric testing era mention chest/lung counting.

REECo (1993a) identified lung counts with nonroutine bioassay for the suspected intake of thorium from the use of special nuclear materials, uranium, or any of the transuranics, especially californium and curium. Lung counts for low-energy photon emitters were performed with four Canberra Low Energy Germanium detectors mounted on two adjustable support arms. Each detector had a window thickness of 0.5 mm and an active diameter of 50.5 mm, yielding a total active detection area of about 8,000 mm² for the array. The detector array measured photons from the lung area within the calibration energy range of 15 to 400 keV. With the individual in a reclining chair, one two-detector array was placed on each side of the chest. The placement minimized the CWT interference

Table 5-12. Background and efficiency data for ²⁴¹Am detection using phoswich detectors - (40-80 keV).

	CWT - cm	Normalized efficiency counts/cm ² / 10 ⁶ photons	Efficiency* counts/photon	Efficiency** cpm/nCi	Estimated MDA - nCi (40 minute count)
Torso	1.51	158	0.079	63.0	0.11
13% muscle – 87% adipose	2.12	146	0.073	58.2	0.11
	2.79	136	0.068	54.2	0.12
	3.34	127	0.0635	50.6	0.13
	40.3	116	0.058	46.2	0.14
50% adipose – 50% muscle	2.15	142	0.0710	56.6	0.12
	28.0	131	0.0655	52.2	0.13
	3.24	123	0.0615	49.0	0.14
	3.99	112	0.056	44.6	0.15
100% muscle	2.11	144	0.072	57.4	0.12
	2.75	132	0.066	52.6	0.13
	3.32	123	0.0615	49.0	0.14
	3.96	110	0.055	43.8	0.15
Background	0.163 cpm/ cm ²				

* For 500 cm² detector

** 59.5 keV gamma intensity = 0.359

- The photon intensity for the ²⁴¹Am gamma ray is 7.8 times that for the uranium X-rays emanating from the plutonium decay.

while maintaining proximity to the lung and bronchial region, thereby optimizing the number of photons measured.

The chest wall is a variable shield and the CWT must be determined for accurate quantitative radioactivity measurements for low-energy gamma rays. The differing densities and thicknesses of muscles and adipose can result in significant differences in attenuation corrections. The method used at NTS in the early 1990s was based on published biometric relations for CWT measurement relations to weight and height. The chosen relation was

$$\text{CWT} = 1.973 (\text{W/H}) - 2.0038$$

where:

CWT = chest wall thickness in centimeters

W = weight in pounds

H = height in inches

Data were analyzed using Canberra Industries software (ABACOS Plus) from physical data entered for the individual being counted. The software used a modified peak analysis technique that determined the areas of photo peaks in the spectrum after subtracting the underlying continuum background. It used two consecutive methods for determining peak location – a library-driven peak search, using a radionuclide library file, followed by a sliding-peak analysis to locate spectral peaks not included in the library. The routine time for a lung count was 1,000 seconds (16.67 minutes). The MDAs for chest lung counts are listed in Attachment 5D, Table 5D-8.

The current NTS contractor states that direct measurements of plutonium in the body, typically via lung counts, are capable of detecting only very large intakes or extended chronic exposures. Because such occurrences are rare, maintaining routine lung counting capability was not cost-effective (BN 2003a).

5.3.3 Thyroid Counting

During weapons testing at NTS, exposure to radioiodines was a major concern for the first 100 hr after detonation. Radioiodines were produced directly from fission and as daughter products of other fission nuclides. This was particularly important for individuals involved in cloud sampling after an above-ground test or individuals on the ground following the fallout pattern.

REECo (1961) states that a determination of ^{131}I in human and animal thyroid was made by direct gamma counting. In the field, this entailed taking a background count at 25 cm from the thigh of the technician or the worker and then taking a count of the worker thyroid at 25 cm.

REECo procedures from 1968, 1972, and 1975 described a thyroid count performed after a urinalysis that indicated an individual had been exposed to radioiodine (REECo 1968-1991b). A 2-in. NaI(Tl) crystal detector was placed next to the individual's larynx and a 5-minute gross gamma count was performed. Background was based on measurement of the machine operator's thigh; in later years the thigh of the individual being counted was used. Nasal swabs were sometimes obtained in an effort to determine if the nose and mouth were contributing gamma activity. Results were hand-calculated using appropriate correction factors and decay constants. A record of each measurement was maintained as the individual returned for follow-up data acquisition. In 1993, thyroid counting was performed with the whole-body counting system by positioning the detector over the appropriate area and counting for 40 minutes (REECo 1993a). Section 5.2.2.1 includes information on the ratios of the various iodines in the period immediately following a shot. Dose reconstructors should use this for individuals directly involved in incidents such as Baneberry or reentry activities as indicated in the "other monitoring" section of the records provided by DOE.

5.3.4 Wound Counting

Wound counting was performed (REECo 1993a) with the whole-body counting system by positioning the detector over the appropriate area 4 in. from the wound and counting for 40 minutes. The whole-body counting system was calibrated for these measurements by placing the Ge coaxial detector 4 in. away and perpendicular to the thyroid phantom or the simulated wound. A vial containing the same concentrated multiradionuclide solution used in the whole-body counting calibration was placed in the thyroid phantom or at a location to simulate a wound and counted for 40 minutes.

BN (2000) discusses wound activity levels and states that, when contamination is detected in the area of a wound, measurements should be made with a wound monitor. BN (2003a) states that a wound counter is maintained on the site as an investigative tool for situations in which immediate indications of potential intakes through wounds might be needed.

This background information is provided for historic purposes only; the programs used to determine internal and external dose for the NIOSH project cannot convert wound dose to dose to a specific organ.

5.4 PERSONAL AIR SAMPLING DATA

Air sampling records were considered workplace monitoring records rather than intake monitoring records, and were not maintained in the same collection as bioassay records. Air sampling records were associated with each facility and were stored as facility records. Boxes containing workplace monitoring records can be retrieved from storage. However, locating a specific set of air sampling records would be very time-consuming and should be considered only as a last resort. Correlation

between air sample concentrations in given rooms or work locations and a specific person would be difficult.

REECo (1993a) stated that, in general, air monitoring devices were positioned to provide samples representative of the worker's breathing zone. However, some work conditions, such as mining and drilling, might have required location of the air monitor intake in the area of highest expected concentration "to ensure that airborne radioactivity possibly breathed by the worker is not overlooked." Workplace air monitoring was required in occupied areas that had the potential to exceed 10% of any Derived Air Concentration (DAC) value listed in DOE Order 5480.11 (DOE 1989).

The principal workplace air monitoring device used at NTS was the retrospective air sampler (RAS). RASs were used at the decontamination facility (Decon Pad), during post-test drilling, tunnel reentry, and other special operations. REECo (1993a) describes specifics for collection and detection of air samples. Air samples were counted at the REECo laboratory in Mercury (A-23), the REECo Decon Pad in Area 6, and at the LANL laboratory in Mercury. Continuous air monitors (CAMs) were used at the Treatability Test Facility and on the drill rig floors during LANL drill-backs. The CAMs detected radioiodines, tritium, plutonium, or other radionuclides of concern. For iodine air sampling, bioassay records indicate if the air samples were collected with a charcoal canister or a filter (Arent and Smith 2004).

The *Operation Plumbbob On-Site Rad Safety Report* (REECo 1957) stated that 1945 air samples were collected for each atmospheric and underground test. As a specific example, the appendix describing decontamination of an A-9 balloon site where air samples were evaluated indicated the highest concentrations of airborne contamination existed during the loader operation (5.3×10^4 d/m per m^3 at a sampling station approximately 100 feet downwind). Respirators were required above 22.2×10^4 d/m per m^3 beta/gamma concentration in air. More air sample results are available in that report.

Operation Hardtack Phase II On-Site Rad-Safe Report (REECo 1958) listed air sampling reporting levels for airborne particulate as 2×10^{-6} $\mu\text{Ci}/m^3$ (alpha) and 1×10^{-3} $\mu\text{Ci}/m^3$ (beta). Airborne particulate material was normally collected on fiber filters using Staplex or Filter Queen sampling devices. An annular impactor for alpha in dust was used when immediate analysis was required.

A 1959 memorandum entitled "Operational Guides for Above-Ground Drill Sites into Ground Zero Areas" (REECo 1959) stated that if significant beta-gamma concentrations are indicated, a 1-hr Staplex sample will be collected and nasal swabs will be collected from each person possibly exposed.

Radiological Safety for Underground Nuclear Explosions (REECo 1960) described mining and drilling operations. REECo performed mining, which consisted mainly of high-explosive blasting, removal of broken rock, and reshoring of the reopened tunnel, at the request of LRL. Radioactive debris was dumped with the mine tailings. This report stated that nonradioactive material was sufficient in the "dump" to prevent significant radiation levels from accumulating. Debris beta-gamma ratios during the period (1 to 7 months after the detonation) were variable from 2:1 to 10:1.

In addition, REECo (1960) stated that airborne radioactivity in the mining work area was reduced by the liberal application of water and by natural water seepage. Where natural seepage was not enough to keep areas moist, airborne radioactive particulate concentrations were occasionally as high as 4.5×10^{-2} $\mu\text{Ci}/m^3$ beta, but airborne concentrations never exceeded 1×10^{-3} $\mu\text{Ci}/m^3$ beta in naturally moist portions of the tunnel. Natural emitters were excluded by a 5-day decay period or by annular impactor sampling. The report noted that high concentrations of airborne radioactive material

were encountered and measurements were made difficult by accompanying high levels of radon-thoron daughter products.

According to REECO (1960), core drilling in ground zero areas was performed by the E. J. Longyear Drilling Company and REECO at the direction of LRL and LASL. Drilling operations in the LRL tunnels used water; drilling in LASL shafts was performed dry, and cored material was removed by a large vacuum system. In dry drilling operations, air activity was reduced by filtering the vacuum system exhaust, but activity levels occasionally exceeded MPC values, at which point workers wore respiratory protection devices. The vacuum system exhaust was downwind from the drillers' work area when possible. Handling of "dried" core samples often resulted in significant airborne contaminants. Air hoods and respirators were used to prevent worker exposure to internal radiation.

"Operation Guides for Tunnel Areas" (REECO 1962) mentioned workplace area samples, radon/thoron check methods, and nasal swabs. *Operation Storax On-Site Radiological Safety Report* (REECO 1964) mentioned air sample collection on the drill platform (high and low volume) at breathing level. "Dynamic Environmental Sampling Program" (REECO 1963) discussed air sampling during drill-backs.

Re-entry Problems Associated with Radiation from Underground Nuclear Detonations (Brown 1963) discussed radiological conditions for atmospheric versus underground testing. This paper stated that the entire post-test work environment was characterized by inhalation exposure problems, but that radiation monitors controlled most of the hazards. It also stated that there were no external or internal exposures during the previous fiscal year (1962).

Hazards to Personnel Re-Entering NTS following Nuclear Reactor Tests (NRDL 1968) noted a concern of inhalation of particulate by reentry personnel. The report stated that coarse ($>12\ \mu\text{m}$) and fine ($<1\ \mu\text{m}$) particulate was ejected to several thousand feet.

Table 5-13 lists examples of historic limits for air samples for specific years from 1950 to 1987 compiled from references (REECO date unknown b, Geiger and Whittaker 1961, 1985). Threshold levels include tolerance, MPC, reporting, exposure, detection, alert, and sensitivity levels. Respiratory protection and contamination control procedures are outlined in Attachment 5D, Section 5D.4.3.

5.5 INTERFERENCES AND UNCERTAINTY

5.5.1 Contamination of Samples

Because levels in samples of activity significant in excreta, especially urine, were generally below detectability on workplace personnel detectors, contamination of samples from the worker's hands or clothing is a possibility. Hanford found a decrease in detectable plutonium bioassay results after switching to home collection. Laboratory contamination and mix-up of samples in the laboratory are also a possibility, although laboratory quality control procedures and performance of test samples were designed to minimize this source of contamination.

A contaminated sample will probably show up as an obvious outlier in the dataset for a given worker. If the dataset shows an unusually high urinalysis result for a radionuclide other than tritium, and if follow-up samples were not consistent with the high result, the high result can be considered an outlier. However, if the result is not obviously an outlier, it is claimant-favorable to assume the result is real.

Table 5-13. Historic air sampling limits.

Year	Limit	Note
1957	MPC in air above background of mixture of unknown radionuclides was 1 E-09 $\mu\text{Ci}/\text{mL}$ (beta and gamma) and 5 E-12 $\mu\text{Ci}/\text{mL}$ (alpha)	When radionuclides were known, MPC was 3 times above values based on exposures of 40-hr week
1958	Air tolerance for uranium was 50 $\mu\text{g}/\text{m}^3$ or approximately 70 dpm/ m^3	LANL action level
1958	Laboratory reporting levels for airborne radionuclides were: 2 E-06 $\mu\text{Ci}/\text{m}^3$ (alpha) 1 E-03 $\mu\text{Ci}/\text{m}^3$ (beta)	
1959	MPC in air at NTS above background were: 2 E-12 $\mu\text{Ci}/\text{mL}$ (alpha) 3 E-9 $\mu\text{Ci}/\text{mL}$ (beta & gamma)	
1961	Sensitivity for I-131 concentration in air was approximately 10 pCi/sample	I-131 concentration in air was determined by collection on activated charcoal and specific gamma counting.
1964	Minimum detection limit for I-131 was $< 1 \times 10^{-8}$ $\mu\text{Ci}/\text{m}^3$	
1964	Air sample alert level was 1 E-05 $\mu\text{Ci}/\text{m}^3$ gross alpha and beta.	If gross beta in sample exceeded this level, analysis for actinium (Ac-227) was made. Radon and thoron samples were collected in Area 15 shaft down to 1,000 ft; levels were below MPC of 3 E10-8 $\mu\text{Ci}/\text{cc}$ for radon. Thoron levels were negligible.
1965	MPC for Pu-239 (in air) was 2 E-12 $\mu\text{Ci}/\text{cc}$ for 40-hr week	
1966-1967	Alert level in air: 1 E-14 $\mu\text{Ci}/\text{cc}$ (alpha) 1 E-11 $\mu\text{Ci}/\text{cc}$ (beta)	Alert levels based on MPC of unknown radionuclides in air over 168 hr were compared with sample activity results.
1968	Kr-85 could be measured down to 2 E-07 $\mu\text{Ci}/\text{cc}$ Xe-133 and -135 could be measured down to 8 pCi/ m^3 .	Kr and Xe could be measured with 1-L ion chamber used when only one noble gas or mixtures of known composition were present.
1970	Alert level for alpha detection in air samples from unknown radionuclides was 1 pCi/ m^3 . Detection limit in air samples was 0.0019 pCi/ m^3 .	
1971	Alert level for radioactivity in air was 15 pCi/ m^3	
1985	Kr-85 LLD in air was 4 pCi/ m^3 Xenon LLD in air was 8 pCi/ m^3	

For *in vivo* measurements, contamination can occur as external to the body or, in the case of chest counting, as external to the lung. If a follow-up *in vivo* count obtained the same day or within a few days shows a dramatic decrease in activity or no detectable activity, external contamination can be assumed. Radon progeny and medical diagnostic or therapeutic procedures involving radionuclides can cause interference to *in vivo* measurements, especially for NaI detectors. However, unless the count was invalidated or noted as being influenced by such interference, use the results as recorded.

5.5.2 Uncertainty

Uncertainties for bioassay measurements were not stated in the records. For results near or at the reporting levels, use the assumption provided in *Internal Dose Reconstruction Implementation Guide* (NIOSH 2002); that is, the standard deviation is 0.3 times the MDA or reporting level. For results greater than 3 times the MDA or reporting level, the standard deviation can be assumed to be 0.1 times the result, based on quantification level [as cited in NIOSH (2002)]. If actual standard

deviations or other indications of error are reported with a bioassay measurement result, use the reported value.

REECo (1993a) cites NCRP (1987a) and Traub and Robinson (1987) for discussion of uncertainties. Uncertainties in intake estimates were used to select the best model and action, which should be taken to reduce errors to the extent reasonably achievable and give the most accurate intake estimate possible.

5.5.3 Less-Than Values

At NTS, the term *less-than* refers to data that are reported as less than some reporting level (REECo 1993a). For example, a plutonium urine bioassay might be reported as $<1 \text{ E-12}$ (less than 1×10^{-12}) $\mu\text{Ci/mL}$ or less than the LLD. Less-than data can be used as a constraint on the iteration; the predictions of a model should agree with the less-than data. For example, if a model predicts a urine concentration of $5 \times 10^{-4} \mu\text{Ci/mL}$ and the measured concentration is less than the LLD, the expectation, empirical, and results are in agreement as long as the LLD is higher than the expectation value. Less-than data should not be used for residual plots, test runs, or least squares fitting procedures.

5.5.4 Determination of Worker Exposure

NTS was unique in the DOE weapons complex. Devices were assembled from materials produced elsewhere. Once assembled, the devices were detonated or in some cases during the safety tests, blown up. The source term was specific to each testing category (atmospheric, tunnel, etc.) and in some cases quantities of specific radionuclides and ratios remain classified. The open literature contains no information about the ratios of plutonium and uranium for these tests.

As indicated in this TBD, during atmospheric testing the emphasis was on measurement of external radiation. Internal monitoring was based on the worker being identified as contaminated; therefore, a worker who was not suspected of contamination would not have received the initial screening provided by nasal swab. Nasal swabs were taken on a regular basis, particularly for anyone involved in decontamination activities. If contamination was indicated, the individual received bioassay (usually urinalysis).

- For an initial internal evaluation of a radiation worker diagnosed with cancer in a nonmetabolic organ, use the guidance in *Maximum Internal Dose Estimates for Certain DOE Complex Claims* (ORAU 2004a).
- For a radiation worker with thyroid cancer whose records indicate that there was involvement in reentry after a test or in venting or working drill-backs prior to 1963, evaluate iodines based on the information in Section 5.2.2.1. A reentry time after the test can be estimated from the access records and used in this determination.
- For a radiation worker with other metabolic cancers, review Attachment 5D to determine what other radionuclides would affect the development of the cancer and perform the evaluation based on the radionuclides that provided 90% of the dose. Attachment 5D provides information on the radionuclides present during various test categories.
- For a nonradiation worker (i.e., unmonitored for external and internal radiation exposure), base the internal evaluation on the assigned dose developed from *Technical Basis Document for the Nevada Test Site – Occupational Environmental Dose* (ORAU 2004b). Nonradiation

workers include but are not limited to office workers, most laborers after the atmospheric testing, janitors, cement workers, housekeepers, and cafeteria workers.

- [Reserved – Guidance for monitored workers with no or few bioassay measurements is under development. Example from Hanford TBD follows.]

I. Workers with No Confirmed Intakes

A. Special consideration for Pu Am and Th – If intake suspected, but not confirmed, DR can use more sensitive urine results from later time to determine worst case intake at time when analysis was not as sensitive. Use MDA from later urine analysis or if there are many sample all showing no detection, then use $0.5 \times \text{MDA}$.

B. Worst Case Chronic Intakes – For workers with many results over a long time, but no confirmed intakes, a maximum chronic intake can be determined using the MDA of the last sample as the upper bound of excretion assuming chronic intake for the entire exposure period. The MDA (not the decision level) should be used.

II. Unmonitored Workers (robust radiological safety program – not much chance of missing a large intake.)

A. No bioassay record and no evidence of dosimeter issue, then internal dose ~ environmental intake only

B. If wore dosimeter, then the internal dose is less than a monitored worker with no bioassay result above reporting level. Estimate upper bound using radionuclides of concern and MDAs.

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ATTACHMENT 5D OCCUPATIONAL INTERNAL DOSE FOR MONITORED WORKERS

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ACRONYMS AND ABBREVIATIONS

AMAD	activity-median aerodynamic diameter
CA	contamination area
cc	cubic centimeter
Ci	curie
cpm	counts per minutes
DOD	U.S. Department of Defense
DOE	U.S. Department of Energy
dpm	disintegration per minute
DPP	drift protection plug
DU	depleted uranium
EG&G	Edgerton, Germeshausen, and Grier Corporation
g	gram
GFP	Gross Fission Product
H	time of detonation
HEPA	high-energy particulate air (filter)
HEU	highly enriched uranium
H&N	Holmes & Narver
hr	hour
ICRP	International Commission on Radiation Protection
IMBA	Integrated Modules for Bioassay Assessment
keV	kilo (thousand) electron volts
kt	kiloton
L	liter
LANL	Los Alamos National Laboratory
LASL	Los Alamos Scientific Laboratory
LLD	lower limit of detection
LLNL	Lawrence Livermore National Laboratory
LOS	line of sight
LRL	Lawrence Radiation Laboratory
m	meter
MDA	Minimum Detectable Activity; Minimum Detectable Amount
MDI	Minimal Detectable Intake
mL	milliliter
mrem	millirem
MSA	Mine Safety Appliances Company
nCi	nanocurie
NTS	Nevada Test Site
OBP	overburden plug

OCAS	Office of Compensation Analysis and Support
P	purge or controlled ventilation
pCi	picocurie
PPE	personnel protection equipment
R	release
RCT	radiological control technician
REEC _o	Reynolds Electrical & Engineering Company, Inc.
RSN	Raytheon Services Nevada
SNL	Sandia National Laboratories
TTR	Tonopah Test Range
WBC	Whole-Body Counter
WEF	Waste Examination Facility
μCi	microcurie

5D.1 OCCUPATIONAL INTERNAL DOSE

The bioassay program monitored internal uptake of radionuclides using both *in vitro* analysis of urine and feces and *in vivo* monitoring using whole-body and chest counting.

5D.2 BIOASSAY CODES AND *IN VITRO* MINIMUM DETECTABLE ACTIVITIES AND DETECTION LEVELS

5D.2.1 Codes Used in Bioassay Records

Employer codes and job titles for NTS contractors [Reynolds Electrical & Engineering Company (REECo), Edgerton, Germeshausen, and Greer Corporation (EG&G), Holmes & Narver (H&N), and Raytheon Services Nevada (RSN)] are available to dose reconstructors on the O Drive maintained by the Office of Compensation Analysis and Support (OCAS) of the National Institute for Occupational Safety and Health. The computerized bioassay records contain the codes listed in Tables 5D-1 through 5D-5 (DeMarre 2003).

Table 5D-1. Codes for analyte.

For heading "an_desc_co"	For heading "analyze_de"
01	003-H (tritium)
02	ALPHA
03	BETA
04	GAMMA (gross)
05	GFP (gross fission products)
06	239-PU
07	235-U
08	241-AM
09	238-PU
10	PGAMMA (gamma, specific isotopics)
11	239-PU
22	89-SR
23	90-SR
74	CO2 (carbon dioxide)
75	H2O (water)

These codes were not identified with a particular period. The measurements of alpha, beta, gamma, and gross fission products were just a "gross measurement," to determine if additional measurements were needed (DeMarre 2004). No specific radionuclides were identified. This was a screening tool. For example, a proportional counter determined the counts of alpha and beta for nasal swipes. The proportional counter does not identify isotopes. If swipes were taken while working on a safety test (post test), they were used as an indicator that plutonium might be present in the sample. For gamma, when a multichannel analyzer was used, specific isotopes could be cited if they were detectable. For example, one could occasionally detect ruthenium.

The main concerns were tritium, iodines, and plutonium (Arent and Smith 2003). If iodine was present, it was easily identified. In the case of the YUBA test (1963), even the G-M survey meter detected iodine present in workers leaving the controlled area. Because NTS did not have a Whole-Body Counter (WBS) at this time, these individuals were sent to Donner Laboratory (California) for a full workup (whole body and urine). The iodine was easily detectable in urine and in the whole body. Due to security concerns, the word "tritium" was not used.

The terms "ACTIVITY or "ACT," "MINT," "EVERGREEN," and "T" were code words for tritium. "PRODUCT" was a code word for ²³⁹Pu. Synonyms for uranium included "oralloy" for enriched

uranium, “turballoy” for natural uranium, and “D-38” for depleted uranium. “LT” or “-” in the bioassay records means less than the detection limit. The code “-99” means not detected (Arent and Smith 2004). Table 5D-2 lists body part codes and abbreviations that are specific to different periods, as listed.

Table 5D-2. Codes for body parts.

For heading ^(a) “bdy_prt_cd”	For heading ^(b) “bdy_prt_pr”	For heading “bdy_prt_df”	Period
01	WB	Whole body	1945 to present
02	SK	Skin (PL 98-542)	1945 to present
03	LH	Left hand	1945 to present
04	RH	Right hand	1945 to present
05	FA	Forearms	1945 to present
06	EX	Extremity	1945 to present
07	EY	Eye	1945 to present
08	HD	Head	1945 to present
09	GN	Gonads	1945 to present
10	TH	Thyroid (PL 98-542 & 100-321)	1945 to present
11	BO	Bone (PL 98-542)	1945 to present
12	GI	GI tract	1945 to present
13	MU	Muscle	1945 to present
14	LU	Lung (PL 98-542)	1945 to present
15	PA	Pancreas (PL 98-542 & 100-321)	1945 to present
16	LI	Liver (PL 98-542 & 100-321)	1945 to present
17	AG	Adrenal gland	1945 to present
18	SP	Spleen	1945 to present
19	KI	Kidney (PL 98-542)	1945 to present
20	PR	Prostate	1945 to present
21	LF	Left foot	1945 to present
22	RF	Right foot	1945 to present
23	OT	Any tissues other than specific target organ	1945 to present
24	RW	Right wrist	1945 to present
25	LW	Left wrist	1945 to present
26	WW	Wrist	1945 to present
27	LL	Lower large intestine	1945 - 1986
28	UI	Upper large intestine	1945 - 1986
29	SI	Small intestine (PL 100-321)	1945 - 1986
30	SW	Stomach (PL 98-542 & 100-321)	1945 - 1986
31	BS	Bone surfaces	1945 to present
32	RM	Red marrow (PL 98-542 & 100-321)	1945 - 1986
33	BR	Breast (PL 100-321)	1945 - 1986
34	BF	Breast (female) (PL 98-542)	1945 - 1986
35	ES	Esophagus (PL 98-542 & 100-321)	1945 - 1986
36	CO	Colon (PL 98-542)	1945 - 1986
37	BL	Bladder (PL 98-542)	1945 - 1986
38	SA	Salivary gland (PL 98-542)	1945 - 1986
39	PH	Pharynx (PL 100-321)	1945 - 1986
40	BI	Bile duct (PL 100-321)	1945 - 1986
41	GA	Gall bladder (PL 100-321)	1945 - 1986
42	LY	Lymph gland(PL 100-321)	1945 - 1986
43	BN	Brain	1945 - 1986
44	LX	Larynx	1945 - 1986
45	MX	Maxillary sinus	1945 - 1986
46	FR	Frontal sinus	1945 - 1986
47	RU	Rectum	1945 - 1986
48	AO	All organs	1945 - 1986
49	HT	Heart	1945 - 1986
50	UR	Urethra	1945 - 1986
51	SC	Spinal cord	1945 - 1986
52	FT	Foot	1945 - 1986
53	AK	Ankle	1945 - 1986
54	HP	Hip	1945 - 1986

a. From 1987 to the present, this heading is “bp_code”.

b. From 1987 to the present, this heading is “bp_name”.

Table 5D-3 lists codes for internal radionuclides and sample types that may be found in excreta records. The following codes were used as general information. The code 001 is used for nuclear weapons testing operations from 1945 to 1962 in the database (these are the one-record-per-page printouts). In general 001 is the main code used for 1945 to 1962. This information is added in updates for specific test operations where summary annual exposure data were replaced with individual dosimeter entries.

Table 5D-3. Codes for radionuclides and sample types.

int_nuc_cd	int_nuc_df	int_nuc_cd	Int_nuc_df
	No data available	022	Te-132
001	Detonation fission products	023	Te-134
002	Reactor fission products	024	I - 131
003	Criticality accident	025	I - 132
004	X-ray generator	026	I - 133
005	Accelerator	027	I - 134
006	Reactor neutrons	028	I - 135
007	Neutron generator	029	Xe-133
008	Cyclotron	030	Xe-135
009	Activation products	031	Cs-137
010	Photo neutrons	032	Ba-140
011	Transuranics	033	Th-232
012	H-3	034	U-234
013	Na-24	035	U-235
014	Fe-59	036	U-238
015	Sr-89	037	Pu-236
016	Sr-90	038	Pu-238
017	Sr-y-90	039	Pu-239
018	Nb-95	040	Pu-240
019	Ru-103	041	Am-241
020	Ru-Rh-106	042	Cf-2xx
021	Te-131	043	Pu-239, Am-241

These codes were not specifically for the bioassay program. They were generated for the main database, principally for 1945-1962 data, so one could distinguish between the reactor program and the weapons test program. The codes provided a potential flexibility for the database. The database design was developed in 1979 and augmented in 1983 to create a relational database. One of the potentials envisioned by designers was to support epidemiological studies with the data (DeMarre 2004).

In general, there was no consistent listing of the test name on the bioassay request form that went to the laboratory. Sometimes a location identifier was listed (e.g., U12b for B Tunnel). Until 1994, there were unannounced tests, the names of which did not appear on the bioassay request forms (DeMarre 2004).

Table 5D-4 lists codes for the type of sample to be analyzed.

NTS maintains a record set called the “dead bioassay database” (Arent and Smith 2004). There are no codes in this database that include information from 1955 to 1963. In the microfiche copy of the bioassay data, the years are listed as two digits, (56, 57, etc.). The main purpose of the dead bioassay file/microfiche is to point to the raw data (note, reel, and frame citations). If an individual has a “deadbio” record, the microfiche page is followed by a copy of the original data forms for the data

cited. Dose reconstructors should review data from the original forms, not the microfiche “deadbio” index.

Table 5D-4. Codes for sample types.

sam_typ_co	Sample_type
06	Feces
09	Nasal smear
10	Urine
19	Wound swab
20	Tissue gross
21	Muscle
22	Blood
23	Skin
24	Liver
25	GI tract
26	Thyroid
27	Bone
28	Kidney
29	Milk
30	Lung
31	Whole body
60	Miscellaneous

Table 5D-5. Codes for units.

Code	Unit	Description
10	dpm	disintegrations per minute
11	dpm/cc	d/m per cubic cm
12	dpm/g	d/m per gram
13	dpm/L	d/m per liter
15	dpm/m ³	d/m per cubic meter
16	dpm/kg	d/m per kilogram
20	Bq	Becquerels
21	Bq/cc	Becquerels per cubic centimeter
22	Bq/g	Becquerels per gram
23	Bq/L	Becquerels per liter
25	Bq/m ³	Becquerels per cubic meter
26	Bq/kg	Becquerels per kilogram
30	pCi	picocuries
31	pCi/cc	picocuries per cubic centimeter
32	pCi/g	picocuries per gram
33	pCi/L	picocuries per liter
35	pCi/m ³	picocuries per cubic meter
36	pCi/kg	picocuries per kilogram
40	μCi	microcuries
41	μCi/cc	microcuries per cubic centimeter
42	μCi/g	microcuries per gram
43	μCi/L	microcuries per liter
44	μCi/mg	microcuries per milligram

Code	Unit	Description
45	μCi/m ³	microcuries per cubic meter
46	μCi/kg	microcuries per kilogram
50	mCi	microcuries
51	mCi/cc	millicuries per cubic centimeter
52	mCi/g	millicuries per gram
53	mCi/L	millicuries per liter
54	mCi/mg	millicuries per milligram
55	mCi/m ³	millicuries per cubic meter
56	mCi/kg	millicuries per kilogram
60	cpm	counts per minute (c/m)
61	cpm/cc	c/m per cubic centimeter
62	cpm/g	c/m per gram
63	cpm/L	c/m per liter
64	cpm/mg	c/m per milligram
65	cpm/m ³	c/m per cubic meter
66	cpm/kg	c/m per kilogram
67	%	percent
68	μg/g	micrograms per gram
69	μg/L	micrograms per liter
70	μg	micrograms U (Reg: KPA analysis)
71	μS/cm	microseimens per centimeter
72	mS/cm	milliseimens per centimeter

5D.2.2 In Vitro Analyses for Individual Radionuclides

Table 5D-6 lists limits of detection for urine and fecal analyses. The Minimum Detectable Activity or Amount (MDA) is an *a priori* value used to evaluate the laboratory’s ability to detect an analyte in a sample. *Lower Limit of Detection* (LLD) is defined in REEC0 (1993a) as a value selected above the MDA to reduce the probability of reporting false positive results. Limit of sensitivity is equivalent to the LLD. *Detection limit* is a general term related to the smallest amount of material detectable as a

function of the measurement method and instrument background. All urine samples were collected over a 24-hr period.

Table 5D-6. Limits of detection for urine and fecal analysis.

Radionuclide	Period	Urine MDA (pCi/L) ^(a)	Fecal MDA (pCi/g) ^(a)	Source
H-3	1958-1976	5 µCi/L	-	Arent and Smith 2004
H-3	1977-1987	1E-6 µCi/mL (detection limit) ^(b)	-	REECo 1977-1987
H-3	1988 ^(c) -1999 ^(d)	300	-	REECo 1993a
H-3	2000-2002	0.001 µCi/L	-	BN 2000
H-3	2003-present	0.005 µCi/L	-	BN 2003
Na-22	1970-present ^(d)	200	-	REECo 1993a
Na-24	1970-present ^(d)	100 ^(e)	-	REECo 1993a
Mn-54	1970-present ^(d)	100	-	REECo 1993a
Co-57	1970-present ^(d)	100	-	REECo 1993a
Fe-59	1970-present ^(d)	600	-	REECo 1993a
Co-60	1970-present ^(d)	50	-	REECo 1993a
Sr-85	1970-present ^(d)	200	-	REECo 1993a
Sr-90	1961-1969 ^(d)	25 pCi/sample (limit of sensitivity) ^(f)	-	REECo 1993b
Sr-90	1970-1999 ^(d)	0.8 ^(g)	-	REECo 1993a
Sr-90	2000-present	1	-	BN 2000, 2003
Zr-95	1970-present ^(d)	600	-	REECo 1993a
Nb-95	1970-present ^(d)	100	-	REECo 1993a
Tc-99m	1970-present ^(d)	100	-	REECo 1993a
Rh-101	1970-present ^(d)	100	-	REECo 1993a
Rh-102	1970-present ^(d)	100	-	REECo 1993a
Rh-102m	1970-present ^(d)	200	-	REECo 1993a
Ru-103	1970-present ^(d)	200	-	REECo 1993a
Ru-106	1970-present ^(d)	1000	-	REECo 1993a
Sb-122	1970-present ^(d)	100	-	REECo 1993a
Sb-124	1970-present ^(d)	100	-	REECo 1993a
Sb-125	1970-present ^(d)	300	-	REECo 1993a
I-131	1961-1969 ^(d)	10 pCi/sample (limit of sensitivity)	-	REECo 1993b
I-131	1970-present ^(d)	100	-	REECo 1993a
I-132	1970-present ^(d)	90	-	REECo 1993a
Te-132	1970-present ^(d)	100	-	REECo 1993a
Ba-133	1970-present ^(d)	200	-	REECo 1993a
Ba-133m	1970-present ^(d)	100 or 400 ^(h)	-	REECo 1993a
I-133	1970-present ^(d)	100	-	REECo 1993a
Cs-134	1970-present ^(d)	500	-	REECo 1993a
I-135	1970-present ^(d)	200	-	REECo 1993a
Cs-136	1970-present ^(d)	100	-	REECo 1993a
Cs-137	1961-1969 ^(d)	10 pCi/sample (limit of sensitivity)	-	Geiger and Whittaler 1961, REECo 1993b
Cs-137	1970-present ^(d)	100	-	REECo 1993a
Ce-139	1970-present ^(d)	100	-	REECo 1993a
Ba-140	1961-1969 ^(d)	10 pCi/sample (limit of sensitivity) ⁽ⁱ⁾	-	REECo 1993b
Ba-140	1970-present ^(d)	500	-	REECo 1993a
La-140	1970-present ^(d)	200	-	REECo 1993a
Ce-141	1970-present ^(d)	200	-	REECo 1993a
Ce-143	1970-present ^(d)	200	-	REECo 1993a
Ce-144	1970-present ^(d)	800	-	REECo 1993a
Nd-147	1970-present ^(d)	400	-	REECo 1993a
Eu-152	1970-present ^(d)	300	-	REECo 1993a
Eu-154	1970-present ^(d)	200	-	REECo 1993a

Table 5D-6 (Continued). Limits of detection for urine and fecal analysis.

Radionuclide	Period	Urine MDA (pCi/L) ^(a)	Fecal MDA (pCi/g) ^(a)	Source
Eu-155	1970-present ^(d)	500	-	REECo 1993a
Yb-169	1970-present ^(d)	600	-	REECo 1993a
Ta-182	1970-present ^(d)	300	-	REECo 1993a
W-187	1970-present ^(d)	300	-	REECo 1993a
Ir-192	1970-present ^(d)	100	-	REECo 1993a
Fission products- GFP (beta)	1977-1987	1E-10 μ Ci/mL (detection limit) ^(b)	-	REECo 1977-1987
Fission products- GFP	1988 ^(c) -present ^(d)	3 ^(d)	-	REECo 1993a
Gross alpha	1968-present	-	1E-07 μ Ci/g (detection limit) ^(k)	REECo 1993b
Gamma	1977-1987	5E-8 μ Ci/mL (detection limit) ^(b)	-	REECo 1977-1987
Gamma (Cs-137)	1988 ^(c) -present	100	-	BN 2000, 2003
Ra-226	1958-1969 ^(d)	0.88 dpm/sample (detection level) ^(l)	-	LASL 1958a
Ra-226	1970-present ^(d)	300	0.4 or 0.04 ^(h)	REECo 1993a
Ra-226	2000-present	0.1 pCi/sample	- ?	BN 2000, 2003
Th-228	1970-present ^(d)	-	0.01	REECo 1993a
Th-230	1954 -1992 ^(d)	0.5 dpm/sample (detection level) ^(m)	-	LASL 1954, REECo 1993b
Th-230	1970-present ^(d)	-	0.01	REECo 1993a
Th-230	2003-present	pCi/sample	0.04 pCi/sample	BN 2003
Th-232	1993-1999 ^(d)	-	0.01	REECo 1993a
Th-232	2000-present	0.02 pCi/sample	0.05 pCi/sample	BN 2000, 2003
U-234	1970-present ^(d)	0.02	-	REECo 1993a
U-234	2000-present	0.04 pCi/sample	0.04 pCi/sample	BN 2000, 2003
U-235 ⁽ⁿ⁾	1961-1969	0.03 dpm/sample (limit of sensitivity)	-	REECo 1961, 1993b
U-235	1970-present ^(d)	0.01	0.008	REECo 1993a
U-235	2000-present	0.04 pCi/sample	0.04 pCi/sample	BN 2000, 2003
U-238	1970-1999 ^(d)	0.02	0.008	REECo 1993a
U-238	2000-present	0.04 pCi/sample	0.04 pCi/sample	BN 2000, BN 2003
Elemental U	1970-present	5 μ g/L ^(o)	-	REECo 1993a
Pu-238	1982-1987	2E-10 μ Ci/mL (detection limit) ^(b)	-	REECo 1977-1987
Pu-238	1988 ^(c) -1999 ^(d)	0.01	-	REECo 1993a
Pu-238	2000-present	0.006 pCi/sample	0.03 pCi/sample	BN 2000, 2003
Np-239	1970-present	400	-	REECo 1993a
Pu-239 ^(p)	1954-1957 ^(d)	2 dpm/24 hr (detection limit) ^(q)	-	LASL 1954
Pu-239 ^(p)	1958-1960 ^(d)	0.05 dpm/sample (detection limit) ^(r)	-	LASL 1958a, REECo 1993b
Pu-239 ^(p)	1961-1976 ^(d)	0.005 dpm/sample (limit of sensitivity)	-	REECo 1961, 1993b
Pu-239	1977-1987	5E-11 μ Ci/mL (detection limit) ^(b)	-	REECo 1977-1987
Pu-239	1988 ^(c) -2000 ^(d)	0.01 (alpha spec) ^(s) ; 50 (gamma spec) ^(s)	0.004 (alpha spec)	REECo 1993a
Pu-239	2000-present	0.006 pCi/sample	0.03 pCi/sample	BN 2000, 2003
Pu-240	1970-present	0.01	0.004	REECo 1993a
Pu-240	2000-present	0.006	-	BN 2000, 2003
Am-241 ^(t)	1954-1957 ^(d)	2 dpm/sample (detection limit) ^(u)	-	LASL 1954
Am-241 ^(t)	1958-1981 ^(d)	0.5 d/m-24 hr sample (LLD) ^(v)	-	LASL 1958a, REECo 1961, 1993b
Am-241	1982-1987	2E-11 μ Ci/mL (detection limit) ^(b)	-	REECo 1977-1987
Am-241	1988 ^(c) -1999 ^(d)	0.03 ^(w)	0.03	REECo 1993a
Am-241	2000-2002	0.008 pCi/sample	0.03 pCi/sample	BN 2000
Am-241	2003-present	0.006 pCi/sample	0.03pCi/sample	BN 2003
Cm-244	2000-present	0.008 pCi/sample	-	BN 2000, 2003

a. MDA units are pCi/L for urine and pCi/g for feces unless noted otherwise.

b. Reference states detection limit or "less-than" value.

c. The start date for this value was extended back from 1993 to 1988 to account for data gaps.

- d. Only the period start date is confirmed (unless otherwise noted); the value is assumed to apply to the remainder of the period.
- e. 1.4×10^{-7} $\mu\text{Ci/mL}$ for blood sodium radiobioassay.
- f. With counter background of 17 cpm, a 60-minute counting period, 25 pCi of Sr-90/sample detected with an accuracy of $\pm 10\%$ at 90% confidence level.
- g. Method does not differentiate between Sr-89 and Sr-90.
- h. Both values are listed in REECo (1993a).
- i. REECo (1961) states the concentration of La-140 is estimated from the Ba-140 concentration and the history of the sample.
- j. Gas flow proportional counting for GFP.
- k. Assumed this was a fecal sample based on the units of $\mu\text{Ci/g}$.
- l. Reference states approximately 0.88 dpm of radium could be detected with the method if the sample was allowed to come to equilibrium.
- m. LASL 1954 states Th-230 could be detected down to 0.5 dpm.
- n. References state uranium in urine. This was assumed to be U-235.
- o. Measured by fluorometric analysis.
- p. References state plutonium in urine. This was assumed to be Pu-239.
- q. Reference states quantities of the order of 2 dpm of plutonium can be determined by the method.
- r. LASL (1958a) states the detection limit at the 99% confidence level is approximately 0.05 dpm/sample.
- s. Can't differentiate Pu-239 from Pu-240 with alpha spectrometry; can differentiate at 50 pCi/L with gamma spectrometry; results reported as Pu-239.
- t. Original references state americium in urine. This was assumed to be Am-241.
- u. Reference states quantities of the order of 2 dpm of americium can be determined by the method. The 1958 revision lowers the value to 0.5 dpm.
- v. REECo (1993b) states americium could be detected down to 0.5 dpm/sample. (dpm/sample = cpm/efficiency of counter; alpha counter at LASL were assumed to have an efficiency of 50%.)
LASL (1958b) states for lack of better data, see plutonium (0.5 dpm/24 hr sample lower limit of detection). Alpha proportional counting does not separate thorium, plutonium, curium, actinium, and neptunium.
- w. Cannot be chemically differentiated from californium, curium, or other isotopes of americium; results are reported as americium..

5D.3 IN VIVO MDAS AND REPORTING PRACTICES AT NTS

In vivo detection capabilities are discussed below. Error in estimated body content of radionuclides tends to increase as the estimate approaches the minimum detection limit of the *in vivo* measurement. The radionuclide content assessed from the results of a whole-body or lung count can be in error by as much as 100% or more from the true content of a low-energy, photon-emitting radionuclide. This is not significant in most cases because the MDA is a small fraction of the maximum permissible intake.

5D.3.1 Whole-Body Counting

Consistent with industry-wide improvements in this area, a variety of hardware and software applications have been used for whole-body counting. In 1993, individuals in the routine bioassay program received employee, annual, and termination whole-body counts. Employees who received routine bioassay included radiation protection technicians and field operation supervisors, chemists and laboratory technicians supporting the Analytical Services Department, industrial hygiene personnel trained as Radiation Workers, and workers in airborne or contamination areas or who shipped or disposed of radioactive material who were trained as Radiation Workers. Whole-body counting was used to detect intake of most gamma-emitting fission and activation products. It was not used for plutonium and americium bioassay due to their low-energy photon emissions. Some MDAs were reported on the forms for individual measurements; use these when they are available. Table 5D-7 lists default MDAs for various radionuclides.

Table 5D-7. Whole-body counting MDAs.

Radio-nuclide	1967 MDA (nCi)	1993 MDA (nCi)	Radio-nuclide	1967 MDA (nCi)	1993 MDA (nCi)	Radio-nuclide	1967 MDA (nCi)	1993 MDA (nCi)
Na-22	3.0	0.7	I-131	3.7	1.6(c)	Lu-174	NA	2.5(b)
Na-24	3.0	1.1	I-132	3.0	1.4(c)	Ta-182	NA	3.3(b)
Sc-46	3.0	1.1	I-133	3.5	1.4(c)	W-181	NA	4.6(b)
Mn-54	3.0	1.2	I-135	10	1.4(c)	W-187	13	4.6
Fe-59	5.3	2.3	Cs-134	3.1	1.5	Ir-192	3.6	0.9
Co-57	3.5	1.4	Cs-136	3.0	1.4	Ra-226 & progeny		50(4)(b,d)
Co-60	1.3	0.9	Cs-137	3.0	1.7	Ac-227	57	28(b,e)
Sr-85	3.0	1.8	Ba-133m	NA	1.1	Th -228	75	33(b,f)
Zr-95	2.1	1.9	Ba-133	5	1.8	Pa-231	NA	6.7(b)
Nb-95	3.0	1.1	Ba-140	12	4.8	Th-232	NA	5(b,g)
Mo-99	23	10(a)	La-140	3.1	0.4	U-233	NA	13(b)
Tc-99m	3.4	7.1	Ce-139	3.7	1.4(b)	U-235	5.6	3.4(b)
Ru-103	3.4	1.2	Ce-141	6.2	2.6(b)	U-237	14	4.8(b)
Ru-106	30	11	Ce-143	7.1	1.4(b)	Np-237	8.3	2.8(b,h)
Rh-104	4.1	1.4(b)	Ce-144	NA	12(b)	Np-239	13	5.6(b)
Rh-102m	6.5	7.9(b)	Nd-147	NA	5.5(b)	Am-241	NA	4.3(b)
Rh-102	3.2	1.0	Eu-152	11	4.2(b)	Am-234	NA	2.8(b)
Sb-122	4.2	2.1	Eu-154	7.4	3.2(b)	²⁴³ Cm	13	4.9(b)
Sb-124	3.1	1.6	Eu-155	14	4.1(b)			
Sb-125	10	3.6	Yb-169	NA	0.2			
Te-132	3.4	1.5(c)	Lu-174m	NA	3.7(b)			

Source: REEC (1993a)

- Only bioassay method identified for this radionuclide.
- In conjunction with lung count.
- In conjunction with thyroid count.
- 50 nCi for Ra-226; 4 nCi based on Pb-241 in equilibrium with Ra-226.
- Based on Th-227 in equilibrium with Ac-227.
- Based on Ra-224 in equilibrium with Th-228.
- Based on Ac-228 in equilibrium with Th-232.

- h. Protactinium-233 is used to measure Np-237 by whole-body count and the MDA for this procedure is 2.8 nCi of Pa-233 which, in equilibrium with Np-237, also represents 2.8 nCi of Np-237.
- i. "NA" means there was no gamma with energy greater than 100 keV on which to base the estimate.

The ^{137}Cs whole-body count sensitivity for 1968 to 1991 was 0.011 μCi for a 20-minute count and 0.010 μCi for a 40-minute count. The Maximum Permissible Body Burden of ^{137}Cs in standard man is 30 μCi .

5D.3.2 Chest Counting

Based on the 1993 internal technical basis document (REECo 1993a), lung counts occurred as soon as practicable after a suspected intake of thorium, uranium, or a transuranic. The first mention of lung counts in REECo procedures was in July 1983 (REECo 1977-1987). There is no earlier mention of chest counting in REECo documentation; however, "lung counts" are noted in some safety reports without specific information provided.

MDAs were generated individually for counts by processing software, with typical values listed in Table 5D-8.

Table 5D-8. 1993 MDAs for chest (lung) counting.^(a)

Radionuclide	MDA (nCi)	Radionuclide	MDA (nCi)
Tc-99m	0.02	Pa-231	0.1
Rh-101	0.02	Th-232 ^(e)	0.2
Rh-102m	0.05	U-233	0.5
Ce-139	0.03	U-234	7
Ce-141	0.03	U-235	0.04
Ce-143	0.05	U-236	0.04
Ce-144	0.2	U-238 ^(f)	9(0.4 or 0.04)
Nd-147	0.04	Np-237	0.05
Eu-152	0.03	Pu-238	2.8
Eu-154	0.04	Np-239	0.05
Eu-155	0.05	Pu-239	7.3
Lu-174m	0.05	Pu-240	2.9
Lu-174	0.04	Pu-241 ^(g)	200
W-181	0.05	Pu-242	40
Ta-182	0.04	Am-241	0.04
Ra-226 ^(b)	0.06	Cm-243	0.05
Ac-227 ^(c)	0.3	Cm-244	1.5
Th-228 ^(d)	0.7	Cf-252	4
Th-230	4		

Source: REECo (1993a)

- a. Values are based on 1,000-second count.
- b. Value based on Pb-214 in equilibrium with Ra-226 (MDA is approximately 1 nCi for Ra-226).
- c. Based on Th-227 in equilibrium with Ac-227.
- d. Based on Ra-224 in equilibrium with Th-228.
- e. Based on Ac-228 in equilibrium with Th-232.
- f. When U-238 is measured directly by lung count, the MDA is 9 nCi for Ra-226. When determined from Th-234, the MDA for U-238/Th-234 is 0.4 or 0.04 nCi assuming equilibrium [both values are listed in REECo (1993a); 0.04nCi is the value listed in Table 4.1].
- g. Based on Am-241 measurement.

5D.3.3 Thyroid Counts

Based on the 1993 internal technical basis document (REECo 1993), thyroid counts occurred as soon as practicable after a suspected radioiodine uptake. Processing software generated individual MDAs for counts, with typical values listed in Table 5D-9.

Table 5D-9. MDAs for thyroid counts.

Radionuclide	Procedure	MDA (nCi)
Te-132	0.004 prompt with special WBC	0.02
I-131	2.2E-4 (thyroid) prompt with special (0.006 µCi) WBC	0.03
I-132	0.15 µCi (thyroid) prompt with special (6.5 µCi) WBC	0.01
I-133	6.1E-4 prompt with special (0.01 µCi) WBC	0.04
I*135		0.07

Source: REECo (1993a)

REECo (1993a) states that radionuclides such as ¹³²Te, ¹³¹I, ¹³³I, and ¹³⁵I are not reliably detected by routine whole-body counts because of their short physical half-lives. Special bioassay programs, such as air monitoring and breathing zone samples (based on work situation and exposure potential), are used to detect intakes of ¹³²Te, followed by prompt special whole-body and thyroid counts for anyone suspected of being exposed to tellurium. Special whole-body and thyroid count bioassays are promptly conducted for anyone suspected of being exposed to radioiodines. "Prompt" is interpreted to mean as soon as possible.

Table 5D-21 lists tests and incidents for which dose reconstructors might need to address iodine as an acute intake. Based on records provided by DOE, the internal dose can be calculated from bioassay results. If no results were provided, the dose reconstructor should conclude that an acute intake evaluation is necessary based on information in the DOE access records indicating that the worker entered a controlled access area after a test. From these records, the dose reconstructor can determine the time and date of entry after the test, and compare this to the date that the test occurred; For a thyroid cancer, the relative amounts of various iodides over time are listed in Table 5-2. Without access records, base internal exposure on environmental levels. Compare the radionuclides listed in Table 5D-9 to the lists in Section 5D.4 and decide which radionuclides are most significant to dose based on the cancer location.

5D.4 OTHER NTS INFORMATION

5D.4.1 Radionuclides of Concern and Specific Bioassay Programs for NTS Facilities

The information in the following paragraphs is from the 1993 NTS Technical Basis (REECo 1993a), Chapter 7, Facility Descriptions and their Specific Routine Bioassay Programs. The radionuclides of concern were determined in the following way:

- The radionuclides in a facility or area were identified from personal interviews, survey reports, radioactive material accountability reports, knowledge of past and present operations, and the open literature.
- The radionuclides in each area or facility, whose radiotoxicity and exposure potential could combine to deliver 90% of the maximum dose to an individual, were considered to be the radionuclides of concern.

- The radionuclides present in sufficient quantities, with gamma or beta emissions with detectable energies, were used to identify the possible presence of other radionuclides more difficult to detect. The indicator radionuclides were used in the Internal Dosimetry Program for screening routine bioassay samples. Further bioassay samples might have been collected if the screening isotopes were present.

The results of the urine samples might have indicated further bioassay measurements such as thyroid, lung, or whole-body counting, fecal sampling, or processing the urine samples for beta and alpha particle emitters.

Drill-backs (Yucca Flat and Pahute Mesa)

Drill-back operations took place within days or weeks of an underground nuclear weapons test. The primary goal of the drill-back was to recover samples of the condensed fission and activation products remaining in the cavity created during the explosion. This was accomplished with a directional drilling technique to recover core samples from the cavity. Because of the short period after the test, it was possible to encounter high gas pressures that created a potential exposure to gaseous and particulate fission and activation products. Starting in 1963, engineering devices were used to prevent the escape of radioactive gases and particulates; however, there was a potential for release in some instances. The pathways for release and personnel exposure were (1) loss of containment during drilling or coring, and (2) resuspension of particulate fission or activation products during the coring operations.

Radionuclides for identification of a problem from containment loss and dose concern were ^{131}I , ^{133}I , and ^{137}Cs . Radionuclides for identification of a problem from resuspension and used to trigger searches in bioassays for the radionuclides of dose concern for resuspension are listed in Table 5D-10. Radionuclides of dose concern from drill-back resuspension are listed in Table 5D-11.

Table 5D-10. Drill-back resuspension and mine back containment loss radionuclides for identification versus time after test.

1 day	10 days	100 days	365 days	10,000 days
Mn-54	Mn-54	Mn-54	Mn-54	Co-60
Fe-59	Fe-59	Fe-59	Co-57	Sr-90
Co-57	Co-57	Co-57	Co-60	Sb-125
Co-60	Co-60	Co-60	Sr-90	Ba-133
Y-91	Sr-89	Sr-89	Y-91	Cs-137
Zr-95	Sr-90	Sr-90	Zr-95	
Zr-97	Y-91	Y-91	Nd-95	
Mo-99	Zr-95	Zr-95	Ru-106	
Tc-99m	Nd-95	Nd-95	Sb-125	
Ru-103	Mo-99	Ru-103	Ba-133	
Ru-106	Ru-103	Ru-106	Cs-135	
Sb-124	Ru-106	Sb-124	Ce-139	
Sb-125	Sb-124	Sb-125	Ce-144	
I-131	Sb-125	Ce-139	Pm-147	
I-133	I-131	Ce-141	Ta-182	
I-135	Te-132	Ce-143		
Cs-137	Ba-140	Ce-144		
Ba-140	La-140	Ta-182		
La-141	Ce-141			
Ce-143	Ce-144			
Ce-144	Nd-147			

Nd-147	Ta-182			
Pm-149				
Ta-182				

The routine bioassay for drill-backs was to collect quarterly urine samples after each drill-back operation. The urine samples were analyzed for gamma emitters by gamma spectroscopy for gross

Table 5D-11. Drill-back resuspension, reentry/mine back resuspension, and decontamination facility, isotopes of concern for dose versus time after test.

1 day	10 days	100 days	365 days	10,000 days
Zr-95	Sr-89	Sr-89	Sr-90	Sr-90
Zr-97	Y-91	Sr-90	Zr-95	Cs-137
Mo-99	Zr-95	Y-91	Ru-106	
Ru-106	Ru-103	Zr-95	Ce-139	
I-131	Ru-106	Ru-103	Ce-144	
Te-132	I-131	Ru-106	Pm-147	
I-133	Te-132	Ce-144		
I-135	Ce-141			
Ce-143	Ce-144			
Ce-144				

gamma-emitting radioactivity and for isotope identification. The primary isotopes used for identification of an intake were ^{131}I , ^{133}I , and ^{137}Cs and those listed in Table 5D-10. Annual whole-body counts were conducted on a routine basis for specific radiation workers and on a special basis following any situation where intake was considered likely. Specific radiation workers include RCTs, radiological field operations supervisors, chemists and laboratory technicians, specific industrial hygiene personnel, and specific radiation worker-trained personnel who worked in airborne radioactive material areas or contamination areas, or who ship or dispose of radioactive material. Specific drilling job categories included:

- Driller Operator Supervisor (Oil Field or Core Drill Type)
- Driller Operator (Oil Field or Core Drill Type)
- Rotary Drill Operator/Rotary Drill Helper
- Driller Helper (Oil Field or Core Drill Type)
- Derrickman (Oil Field Type)
- Motorman (Oil Field Type)
- Fishing Tool Engineer (Oil Field Type)
- Drill Helper Trainee

For the years before the establishment of engineering controls, the dose reconstructor will need to evaluate the potential for exposure, based on review of the access records provided in the claim file provided by DOE.

Reentry and Mine Back (A-1& A-12)

Reentry and mine back operations were similar to those for drill-backs. The major differences were that the operations took place in a confined underground environment, the time after a test that the cavity was entered, the horizontal method of drilling and, in some cases, the cavity was opened and personnel entered. There were four pathways for exposure during reentry and mine backs. The first pathway is based on a loss of containment in the drilling or coring operations during routine drilling or

by failure of containment equipment. The radionuclides of concern were gaseous fission and activation products or their particulate daughters. The second pathway was resuspension of particulate fission or activation products during the coring operations of the drill-back. The third exposure pathway was the possibility for gaseous fission or activation products to seep through fissures in the rock and reenter the working areas of the tunnel. The fourth intake possibility existed when the line-of-sight (LOS) pipe was opened to remove experiment equipment and samples. Specific tunnel job categories included:

- Miner
- Bull Gang (Underground Laborer)
- Mucker (Muck Machine Operator)
- Shifter
- Tunnel Walker
- Dinky Locomotive Operator

Radionuclides of dose concern from containment loss were ^3H , ^{131}I , ^{133}I , and ^{137}Cs . Radionuclides for identification of a problem from containment loss and resuspension are listed in Table 5D-10.

Radionuclides of dose concern from resuspension are listed in Table 5D-11. Radionuclides for identification of a problem from fissures and LOS pipe opening were ^3H , ^7Be , ^{59}Fe , ^{60}Co , ^{124}Sb , ^{131}I , ^{133}I , ^{137}Cs , and ^{182}Ta . Isotopes of dose concern from fissures and LOS pipe openings were ^3H , ^{60}Co , ^{124}Sb , ^{131}I , ^{133}I , and ^{137}Cs .

The routine method for bioassay for reentry and mine backs was to collect urine samples based on the results of air samples. Tunnel air sampling began in 1957 in locations with the potential for airborne exposure (Arent and Smith 2004). Air samplers operated continuously. RCTs checked and exchanged the filters each shift. Bioassay was done only if there was an indication of an effluent release (e.g., positive air sample) (Arent and Smith 2004). The urine samples were gamma counted and sampled for ^3H . Annual whole-body counts were conducted on a routine basis for specific radiation workers and on a special basis following any situation for which intake was considered likely.

Routine Tunnel Operations (A-1 and A-12)

Tritium was the isotope of dose concern; routine bioassay was to collect quarterly urine samples which were processed for ^3H . (No bioassay information was given in REECa 1993a.) See Section 5.2.2.4 for a discussion on assigning tritium dose.

Decontamination Facility (A-6)

Isotopes used for identification of a problem at this facility are listed in Table 5D-12. Isotopes of dose concern are listed in Table 5D-11. Routine bioassay was to collect quarterly urine samples and conduct annual whole-body counts. Urine samples were gamma counted and analyzed for GFP, ^{238}Pu , ^{239}Pu , ^{240}Pu , and ^{241}Am .

Test Treatability Facility (A-25)

The Test Treatability Facility was a pilot project to bench-test technologies to be used for decontamination of soils containing transuranic materials. The concentrations of radionuclides in the soils were not intended to exceed a few picocuries per gram of soil. Isotopes for identification of a problem and dose concern were ^{238}Pu , ^{239}Pu , ^{240}Pu , and ^{241}Am . Routine bioassay was to collect quarterly urine samples and conduct annual whole-body counts. Urine samples were counted on a gamma analyzer for isotopic identification and analyzed for plutonium and americium.

Atmospheric Weapon Safety Tests (A-3 and A-11)

Isotopes for identification of a problem, sample screening, and of dose concern were ^{238}Pu , ^{239}Pu , ^{240}Pu , and ^{241}Am . Routine bioassay was to collect quarterly urine samples and conduct annual whole-body counts. Urine samples were gamma counted and analyzed for plutonium and americium.

Atmospheric Weapons Test Areas (all areas except A-22, 23, and 27 due to test location or airborne dispersion)

Isotopes for identification of a problem and dose concern are listed in Table 5D-13. Routine bioassay was to collect quarterly urine samples and conduct annual whole-body counts. Some area monitoring

Table 5D-12. Decontamination facility, isotopes for identification versus time after test.

1 day	10 days	100 days	365 days	10,000 days
Sr-89	Sr-89	Sr-89	Sr-90	Sr-90
Sr-90	Sr-90	Sr-90	Y-91	Ba-133
Zr-93	Y-91	Y-91	Zr-93	Cs-137
Zr-95	Zr-93	Zr-93	Zr-95	Pu-239
Zr-97	Zr-95	Zr-95	Ru-106	Am-241
Mo-99	Mo-99	Ru-103	Ba-133	
Tc-99m	Ru-103	Ru-106	Cs-135	
Ru-103	Ru-106	Ba-133	Cs-137	
Ru-106	I-131	Cs-135	Ce-139	
I-131	Te-132	Cs-137	Ce-144	
Te-132	I-133	Ce-139	Pm-147	
I-133	Ba-133	Ba-140	Pu-239	
Ba-133	I-135	Ce-141	Am-241	
I-135	Cs-137	Ce-143		
Cs-137	Ba-140	Ce-144		
Ce-139	La-140	Pu-239		
Ba-140	Ce-141	Am-241		
La-140	Ce-144			
Ce-141	Nd-147			
Ce-143	Pu-239			
Ce-144	Am-241			
Nd-147				
Pm-149				
Pu-239				
Am-241				

Table 5D-13. Atmospheric weapons test areas, isotopes for identification and of concern for dose.

Co-60	Sb-125	Cs-137	Eu-155	Pu-239
Sr-90	Ba-133	Eu-152	Lu-174	Pu-240
Ru-101	Cs-134	Eu-154	Pu-238	Am-241
Ru-102m				

personnel also had periodic lung counts. Urine samples were analyzed for gamma emitting radionuclides, GFP, plutonium, and americium.

Low-Level Waste Site (A-3)

Isotopes for identification of a problem and dose concern are listed in Table 5D-14. Routine bioassay was to collect quarterly urine samples and conduct annual whole-body counts. Urine samples were gamma analyzed for GFP, plutonium, and americium.

Table 5D-14. Low-level waste site (A-3), isotopes for identification and of concern for dose.

Mn-54	Ru-103	Ce-144	U-233	Pu-240
Co-60	Ru-106	Ac-227	U-234	Pu-241
Sr-85	Cs-134	Th-228	U-238	Pu-242
Sr-90	Cs-137	Th-230	Pu-238	Am-241
Zr-95	Ba-140	Th-232	Pu-239	Am-243
Nb-95	Ce-141			

Low-Level Waste Site (A-5)

Isotopes for identification of a problem and dose concern are listed in Table 5D-15. Routine bioassay was to collect quarterly urine samples and conduct annual WBCs. Urine samples were analyzed for gamma emitters, sampled for ^3H , and analyzed for GFPs, Pu, and Am.

Table 5D-15. Low level waste site (A-5), isotopes for identification and of concern for dose.

H-3	Mo-99	Ce-141	Ac-227	U-238
Na-22	Ru-103	Ce-144	Th-228	Pu-238
Mn-54	Ru-106	Eu-152	Th-230	Pu-239
Co-57	Sb-124	Eu-154	Th-232	Pu-240
Co-60	Sb-125	Eu-155	U-233	Pu-241
Sr-85	Ba-133	Yb-169	U-234	Am-241
Sr-90	Cs-134	Ta-182	U-235	Pu-242
Zr-95	Cs-137	Ir-192	Np-237	Am-243
Nb-95	Ba-140	Ra-226		

Radiation Instrument Calibration Facilities

Isotopes for identification of a problem and dose concern are listed in Table 5D-16. Routine bioassay for personnel who regularly worked with calibration sources was to collect quarterly urine samples and conduct annual whole-body counts. Urine samples were analyzed for ^3H and plutonium. A special bioassay whole-body count and/or urine sample collection was performed if the loss of calibration source containment was detected by a source leak test or other methods. The follow-up action and type of analyses conducted was determined by the particular source in question.

Table 5D-16. Radiation instrument calibration facilities, isotopes for identification and of concern for dose.

H-3	Sr-90	Ra-226	Th-230	Pu-239
Co-60	Cs-137	Th-228	Pu-238	Am-241

Radiography Operations

The isotope of concern was ^{92}Ir . Bioassay (urine/whole-body count) was conducted if a loss of source containment was detected by swipes or other analysis. Urine samples were gamma counted and follow-up actions taken accordingly.

Well Logging Operations

Isotopes for identification of a problem and dose concern were ^{60}Co , ^{131}I , ^{137}Cs , ^{226}Ra , ^{228}Th , ^{238}Pu , and ^{241}Pu . Bioassay (urine/whole-body count) was conducted if there was a loss of source containment detected by swipes or other analysis. Urine samples were gamma counted or processed for Pu depending on the source type. Additional "Nuclides of Concern" listed in BN (2003) include the following well logging sources: ^{241}Am - ^7Be , ^{238}Pu - ^7Be , and ^{226}Ra - ^7Be .

Device Assembly Facilities (A-27)

Isotopes for identification of a problem and dose concern were ^3H , ^{235}U , ^{239}Pu , and ^{240}Pu . No bioassay information was listed.

Nuclear Rocket Development Areas (A-25)

Tests of nuclear reactors for use as propulsion units were conducted above ground into the 1960s. The fission and activation products from these tests were widely dispersed into the environment. Isotopes for identification of a problem and dose concern were ^{60}Co , ^{90}Sr , ^{137}Cs , and ^{152}Eu . Routine bioassay was to collect quarterly urine samples and conduct annual whole-body counts. Urine samples were gamma counted, sampled for ^3H , and analyzed for GFP and plutonium.

Radioactive Source Storage Areas

Isotopes for identification of a problem and dose concern are listed in Table 5D-17. There was no routine bioassay program in support of normal source storage facility operations. If source leak test or other means indicated a stored source was leaking or had leaked radioactivity, special whole-body counts and/or urine sample collection and analysis were conducted. Specific bioassays and analysis depended on the source leaking.

Table 5D-17. Radiation instrument calibration facilities, isotopes for identification and of concern for dose.

Na-22	Ni-63	Cs-137	Ra-226	Pu-238
Mn-54	Sr-90	Eu-152	Th-228	Pu-239
Co-57	Cd-109	Eu-154	U-235	Am-241
Co-60	Ba-133	Ir-192		

Radiochemistry and Counting Laboratories

Isotopes for identification of a problem and dose concern are listed in Table 5D-18. Routine bioassay was to collect quarterly urine samples and conduct annual whole-body counts only for those personnel routinely handling uncontained radioactive materials. Urine samples were gamma counted, sampled for ^3H , and analyzed for GFP and plutonium.

Table 5D-18. Radiochemistry and counting laboratories, isotopes for identification and of concern for dose.

H-3	Sr-85	Cd-109	Ce-144	Am-241
Na-24	Y-88	Sn-113	Eu-152	Pu-242
Co-57	Sr-90	Cs-137	Hg-20	Am-243
Co-60	Y-90	Ce-139	Pu-239	Cm-244

Radionuclides of dose concern for NTS are summarized in Table 5D-19. Table 5D-20 lists nuclides of concern at different NTS locations.

Section 4.2.1.2 of this TBD includes tables of atmospheric radionuclide concentrations by year that can be used in conjunction with this table.

Table 5D-19. Isotopes of concern for dose, summary list.

H-3	Mo-99	I-133	Lu-174	U-235
Na-22	Ru-101	Cs-134	Ta-182	U-238
Na-24	Ru-102m	Cs-137	Ir-192	Np-237
Mn-54	Ru-103	Ba-140	²⁰ Hg	Pu-238
Co-57	Ru-106	Ce-139	Ra-226	Pu-239
Co-60	Cd-109	Ce-141	Ac-227	Pu-240
Sr-85	Sn-113	Ce-144	Th-228	Pu-241
Y-88	Sb-124	Eu-152	Th-230	Pu-242
Sr-90	Sb-125	Eu-154	Th-232	Am-241
Y-90	I-131	Eu-155	U-233	Am-243
Zr-95	Ba-133	Yb-169	U-234	Cm-244
Nb-95				

Table 5D-20. Current nuclides of concern for NTS locations.

NTS facility and area	Radionuclides of concern			
Big Explosive Experimental Facility (BEEF)	U-234	U-238		
Nuclear Explosive Assembly Facilities (DAF and Area 27)	H-3	U-234	U-235	Pu-238
	Pu-239	Pu-240	Pu-241	Am-241
Routine tunnel operations (Areas 1 and 12)	H-3			
Decontamination Facility (Area 6)	Sr-90	Cs-137	Th-232	U-234
	U-235	U-238	Pu-239	Am-241
Legacy atmospheric weapons safety test areas	Pu-238	Pu-239	Pu-240	Am-241
Legacy atmospheric weapons test areas; legacy weapons test waste trenches and support facilities	Co-60	Sr-90	Ru-101	Ru-102m
	Ba-133	Cs-137	Eu-152	Eu-154
	Eu-155	Pu-238	Pu-239	Pu-240
	Am-241			
Tonopah Test Range and Area 13 safety test areas	Pu-238	Pu-239	Pu-240	Am-241
Tonopah Test Range DU munition tests	U-234	U-238		
Low-Level Waste Site, Area 3	Sr-90	Cs-137	Th-232	Pu-239
	Am-241	U-234	U-235	U-238
Low-Level Waste Site, Area 5	H-3	Sr-90	Cs-137	Th-232
	U-234	U-235	U-238	Pu-239
	Am-241			
Radiation instrument calibration facilities	H-3	Co-60	Sr-90	Cs-137
	Ra-226	U-235	Pu-238	Pu-239
	Am-241			
Radiography operations	Co-60	Cs-137	Ir-192	
Well logging operations	Co-60	Cs-137	Ra-226/Be-7	
	Pu-238/Be-7	Am-241/Be-7		
Nuclear Rocket Development Area (Area 25)	Co-60	Sr-90	Cs-137	Nb-95(a)
	U-234	U-235	U-238	
Legacy biokinetic test areas (Test Cell A, Area 25)	Am-241			
Radioactive source storage areas	Na-22	Mn-54	Co-57	Co-60
	Ni-63	Sr-90	Cd-109	Ba-133
	Cs-137	Eu-152	Eu-154	Ir-192
	Ra-226	Th-228	U-235	Pu-238
	Pu-239	Am-241		
Radiochemistry and counting laboratories	H-3	Na-24	Co-57	Co-60
	Sr-85	Y-88	Y-90	Sr-90
	Cd-109	Sn-113	Eu-152	Cs-137
	Ce-139	Ce-144	Hg-203	Ra-226
	Th-232	U-234	U-238	Pu-239
	Pu-242	Am-241	Am-243	Cm-244
Waste Examination Facility (WEF)	U-233	Pu-238	Pu-239	Pu-240
	Am-241	Am-243	Cm-244	Cf-252
Site Monitoring Services and RAMATROL	H-3	Sr/Y-90	Cs-137	U-234

	U-235	U-238	Pu-239	Am-241
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Source: BN 2003a

5D.4.2 Incidents

Based on guidance from NTS, the dose reconstructor should do the following when researching a claim:

1. Determine that the individual was working at NTS during a given operation.
2. Consult *Compilation of Local Fallout Data from Test Detonations 1945 – 1962* (DNA 1979) in relation to atmospheric tests and *Radiological Effluents Released from U.S. Continental Tests 1961-1992* (Schoengold, DeMarre, and Kirkwood 1996) in relation to underground tests to determine potential exposure to radioisotopes for a given test. *United States Nuclear Tests, July 1945 through September 1992* (DOE 2000) provides a list of tests by date or by test name. Other information contained in DOE (2000) includes operation name, test date and time, sponsor, location, hole number, latitude and longitude, surface elevation, test type, test purpose, and yield.
3. Establish if the claimant is linked to a release by job title or work location.

In addition, the dose reconstructor might find the information in Table 5D-21 [from *The Containment of Underground Nuclear Explosions* (OTA 1989)] and Schoengold, DeMarre, and Kirkwood (1996) useful in identifying underground tests in which problems occurred.

The following paragraphs summarize other specific incidents that are important from an internal dosimetry standpoint at NTS:

- The *Report of the Test Manager for 56 Project – NTS*, Part VI, Chapter 1, “Special Incidents,” describes an overexposure during a plutonium dispersion test (AEC 1956).
- The *Operation Plumbbob On-Site Radiological Report* discusses four instances of possible internal exposure (REECo 1957):
 1. Escape of radioactive gas from the Tower 2-A cab contaminated the working area, resulting in 12 workers being exposed.
 2. One worker removed his respirator while working in an area highly contaminated with alpha-emitting material.
 3. Several workers were exposed while removing and cutting a cable highly contaminated with alpha-emitting material.
 4. Four workers without respirators entered a tunnel that was highly contaminated with alpha-emitting materials.
- The Hardtack II report (DNA 1982) describes a tunnel incident in which an explosion occurred. Six REECo miners and two REECo radiation technicians were affected; no bioassay results were above permissible limits. The report also describes REECo tunnel decontamination methods (air supplied respiratory and full anticontamination clothing (clothing that might have consisted of coveralls, hoods, booties, and gloves), nasal swabs taken).

- A memo for B Tunnel, U12b (AEC 1961), describes the following tritium results for tests that occurred in November 1961: 42 workers were exposed above 3 rem (quarterly external limit) and 48 workers were exposed above 5 rem (yearly internal and external limit). In summary, 108 miners approached the quarterly exposure limit in 1961 due to tritium exposure in U12b. The original source of tritium was from 1958 activities (Arent and Smith 2004).
- The *Operation Storax On-Site Radiological Safety Report* (REECo 1964) discusses internal radiation hazards for 1962 and 1963. Urinalysis results showed that 280 workers received 50 mrem or more, and the maximum internal dosage was 2,435 mrem. Appendix D of this report, "Typical Detailed Safety Support Plan for Underground Nuclear Tests," describes radiological support to post-shot drilling. Air samples were collected on the drill platform (high and low volume) at breathing level. Nasal and urine samples were collected at the discretion of the shift supervisor.
- Appendix A of *General Re-Entry Procedure for Underground Nuclear Events* (LRL 1961) is the specific program protocol used for the YUBA test in 1963, during which an accidental release of radioactivity was detected on the site and an operational release of radioactivity was

Table 5D-20. Releases from underground tests.

Test name	Date	Type	Purpose & yield	Sponsor	Location	Isotopes identified in release	Release type & notes	Release (Ci)	Release summary
Platte	4/14/62	Tunnel	Weapons-related 1.85 kt	LLNL	U12k.01	^{40}K , ^{95}Zr , ^{95}Nb , ^{103}Ru , ^{106}Ru , ^{131}I , ^{133}I , ^{135}I , ^{132}Te , ^{140}Ba , ^{140}La , ^{141}Ce , ^{144}Ce	Test/prompt particle sampling release Accidental release of radioactivity detected offsite	Test release at R+12 hours: 1.9E+06	Venting occurred at tunnel portal, through fissures, and at sampling hole at H+1.5 seconds. Fissures were created on side of hill, and radial cracks formed on top of hill. Persistent cloud was produced containing appreciable quantities of radioactivity associated with particles.
Eel	5/19/62	Shaft	Weapons-related 4.5 kt	LLNL	U9m	^{95}Zr , ^{95}Nb , ^{103}Ru , ^{106}Ru , ^{105}Rh , ^{131}I , ^{133}I , ^{135}I , ^{132}Te , ^{140}Ba , ^{140}La , ^{141}Ce , ^{144}Ce	Test/prompt particle sampling release Accidental release of radioactivity detected offsite	Test release at R+12 hours: 1.9E+06	Venting, in form of geyser, occurred at H+10 seconds from satellite hole U9m-2 and continued steadily until H+19 minutes, 42 seconds. Similar venting occurred at H+15 seconds from satellite hole U9m-3 and lasted until H+21 minutes. Venting ceased with crater subsidence.
Des Moines	6/13/62	Tunnel	Weapons-related 2.9 kt	LLNL	U12j.01	^{103}Ru , ^{106}Ru , ^{105}Rh , ^{131}I , ^{133}I , ^{135}I , ^{132}Te , ^{140}Ba , ^{140}La	Test/prompt particle sampling release Accidental release of radioactivity detected offsite	Test release at R+12 hours: 11E+06	Venting began at H+0.2 seconds on top of hill at surface ground zero, then from sampling hole on face of hill, and finally through portal. Duration of release was about 5 minutes. Test vented from tunnel mouth with sufficient pressure and flow rate that radioactive debris was projected entirely across canyon and deposited on slope behind trailer shelter.
Baneberry	12/18/70	Shaft	Weapons-related 10 kt	LLNL	U8d	Gross fission products Isotopic analysis ^(f) : ^{99}Mo , ^{131}I , ^{132}I , ^{132}Te	Test release Accidental release of radioactivity detected offsite	Test release at R+12 hours: 6.7E+06	Venting occurred from fissure near surface ground zero at H+3.5 minutes. Effluent venting rate steadily decreased with time, but visible vapor continued to emanate from fissure for 24 hours after detonation.
Camphor	6/29/71	Tunnel	Weapons effects <20 kt	LLNL SNL DOD	U12g.10	Test: ^{133}Xe , ^{135}Xe Controlled: ^{131}I , ^{133}I , ^{135}I	Test and controlled releases Accidental release of radioactivity detected onsite only Containment failure ^(a)	Test release at R+12 hours: 360 ^(b)	Test releases occurred from cable building (on mesa) at H+1 hour, lasting for 30 minutes, and from portal at H+3.9 hours, lasting 4 days. Controlled release through ventilation system of tunnel complex began at 1034 hours on July 27, 1971, and lasted 3 days.
Diagonal Line	11/24/71	Shaft	Weapons effects <20 kt	LLNL DoD	U11g	$^{85\text{m}}\text{Kr}$, ^{87}Kr , ^{88}Kr , ^{131}I , ^{132}I , ^{133}I , ^{135}I , $^{131\text{m}}\text{Xe}$, ^{133}Xe , ^{135}Xe , $^{133\text{m}}\text{Xe}$, ^{135}Xe	Test and seepage Accidental release of radioactivity detected offsite by aircraft only Containment failure ^(a)	Test release and seepage at R+12 hours: 6,800	Test release (seepage) occurred from H+3.3 to H+20 hours. Low-level seepage continued for about 3 days, but all significant activity had been released by H+20 hours. Effluent was primarily ^{135}Xe (80-85%), $^{85\text{m}}\text{Kr}$, ^{87}Kr , ^{88}Kr , $^{131\text{m}}\text{Xe}$, ^{133}Xe , and $^{133\text{m}}\text{Xe}$, with trace quantities of ^{131}I , ^{132}I , ^{133}I , and ^{135}I detected. Minor levels of radioactivity were detected offsite by aircraft only.

Table 5D-21 (Continued). Releases from underground tests.

Test name	Date	Type	Purpose & yield	Sponsor	Location	Radionuclides identified in release	Release type & notes	Release (Ci)	Release summary
Hybla Fair	10/28/74	Tunnel	Weapons effects <20 kt	LLNL DOD	U12n.09	Xe-133, Xe-133m	Controlled tunnel purge (release detected onsite only)	Controlled release activity at time of release: 500 Controlled release activity at R+12 hours: 500	Release occurred from area between over burden plug (OBP) No. 1 and OBP No. 2 from November 13 to November 14, 1974. Stemming failed during test and noble gases seeped through or around OBP No. 2. All activity was successfully contained inside OBP No. 1. Effluent released during controlled ventilation of tunnel complex was activity contained between OBP No. 1 and OBP No. 2 only. Activity was 99% Xe-133; remainder was Xe-133m. Second release occurred from U12n.09 drift complex from November 20, 1974, to January 6, 1975. Activity released passed through HEPA and aerosol filter before being released through tunnel ventilation system. Effluent was 99% Xe-133 with some Xe-133m.
Hybla Gold	11/1/77	Tunnel	Weapons effects <20 kt	LANL DOD	U12e.20	Xe-133	Controlled tunnel purge (release detected onsite only)	Controlled release activity at time of release: 5.0E-03 Controlled release activity at R+12 hours: 5.0E-03	Activity was successfully contained inside drift protection plug until it had decayed to insignificant level. Controlled release occurred on November 29, 1977.
Riola	9/25/80	Shaft	Weapons-related 1.07 kt	LLNL	U2eq	Kr-85m, Kr-87, Kr-88, Xe-133, Xe-135, Xe-135m, tritium, and tritiated water	Test, seepage, and gas sampling Release detected offsite (test only) Containment failure ^a Gas sampling release at time of release, in curies: 9.8 H-3 in curies: 9.8 Kr-85 in curies: 1.5 E-04	Test release at R+12 hours: 960 (mixed fission products) Natural seepage at time of release: 2200 (tritium and tritiated water)	Test release and seepage from surface ground zero area occurred at H+10 hours and 59 minutes. Test release, consisting of xenons and kryptons, occurred through surface ground zero cracks and lasted until 1020 hours on September 26, 1980. Seepage continued until it was no longer positively quantified in March 1981. Seepage rate varied throughout period as it was affected by atmospheric pressure changes. Controlled gas sampling containment tank release occurred on December 6, 1982.
Miners Iron	10/31/80	Tunnel	Weapons effects <20 kt	LANL DOD	U12n.11	Xe-133, Xe-135	Controlled tunnel purge (release detected onsite only)	Controlled release activity at time of release: 3.0E-01 Controlled release activity at R+12 hours: 1.0E-01	Controlled release occurred from H+49.5 hours until H+67 hours. Prior to that time, seepage from stemming area into open part of LOS pipe had occurred. Effluent was 87% Xe-135 and 13% Xe-133. Activity was contained in LOS pipe until controlled ventilation of pipe was established. Release point was N Tunnel mesa vent hole.
Huron Landing (Simultaneous with Diamond Ace Test)	9/23/82	Tunnel	Weapons effects <20 kt	LLNL DOD	U12n.15	Kr-85m, Kr-88, Xe-133, Xe-133, Xe-135	Controlled tunnel purge (release detected onsite only)	Controlled release activity at time of release: 280 Controlled release activity at R+12 hours: 120	Controlled ventilation of tunnel occurred from H+27.8 hours until H+36 hours. Prior to that time, activity had been contained in OBP until ventilation to mesa could be established. Release point was N Tunnel mesa vent hole. Effluent was 86% Xe-135, 7% Kr-85m, 3% Xe-133, 3% Xe-133, and 1% Kr-88.
Mini Jade	5/26/83	Tunnel	Weapons effects <20 kt	LANL DOD	U12n.12	Xe-133, Xe-133m	Controlled tunnel purge (release detected onsite only)	Controlled release activity at time of release: 1.0 Controlled release activity at R+12 hours: 1.0	Controlled ventilation occurred from H+5.2 days until H+6.2 days. Prior to that, activity had been contained inside the drift protection plug (DPP) until ventilation to mesa had been established. Release point was N Tunnel mesa vent hole. Effluent was 89% Xe-133 and 11% Xe-133.

Table 5D-21 (Continued). Releases from underground tests.

Test name	Date	Type	Purpose & yield	Sponsor	Location	Radionuclides identified in release	Release type & notes	Release (Ci)	Release summary
Midas Myth	2/15/84	Tunnel	Weapons effects <20 kt	LANL DOD	U12t.04	N/A	Unexpected crater collapse	N/A	Unexpected crater collapse occurred 3 hours after test above test tunnel causing injuries to personnel (1 fatality) and damaging equipment trailers. All radioactive material was contained in vessel with no release to atmosphere or tunnel.
Agrini	3/31/84	Shaft	Weapons-related <20 kt	LLNL	U2ev	Kr-85m, Kr-87, Kr-88, Xe-133, Xe-133, Xe-135, Xe-135m	Test, controlled, and drill-back Accidental release of radioactivity detected onsite only Containment failure ^(a)	Test release at R+12 hours: 690 Controlled release activity at time of release: 3.0E-02 H-3: 2.8E-02 Xe-133: 2.8E-04 Ar-37: 1.6E-03 Drill-back release activity at time of release: 2.0E-03 Xenons: 2.0E-03	Releases occurred as follows: (1) seepage from crater from 1530 hours on March 31, 1984, to 1900 hours on April 1, 1984; (2) controlled, filtered release on June 13, 1984; and (3) ventilation line release at 0705 hours on April 5, 1984, during post-shot drilling operations.
Kappeli	7/25/84	Shaft	Weapons-related 20-150 kt	LLNL	U20am	Kr-85	Late-time seep (release detected onsite only)	Natural, late-time seepage at time of release: 12	Seepage began months after test and continued as follows: 9/24/84 - 12/31/84: 0.5 Ci of Kr-85 11/25/84 - 7/25/85: 3.6 Ci of Kr-85 7/25/85 - 7/25/86: 5.0 Ci of Kr-85
Tierra	12/15/84	Shaft	Weapons-related 20-150 kt	LLNL	U19ac	Xe-133, Xe-131m, Kr-85, Ar-37	Late-time seep (release detected onsite only)	Natural, late-time seepage at time of release: 600 Xe-133: 5.7E+02 Xe-131m: 4.0 Kr-85: 12 Ar-37: 9.0	Seepage occurred intermittently from December 26, 1984, to January 4, 1986.
Misty Rain	4/6/85	Tunnel	Weapons effects <20 kt	LLNL DOD	U12n.17	Xe-133, Xe-133, Xe-135	Controlled tunnel purge Controlled release of radioactivity detected offsite	Controlled release activity at time of release: 63 Controlled release activity at R+12 hours: 45	Controlled ventilation occurred from H+2.85 days until H+4 days. Prior to that, activity had been contained inside gas seal plug until ventilation could be reestablished. Release points were N Tunnel portal and N Tunnel mesa ventilation lines. Effluent was 72% Xe-133, 22% Xe-135, and 6% Xe-133.
Mill Yard	10/9/85	Tunnel	Weapons effects <20 kt	LANL DOD	U12n.20	Xe-133, Xe-135, Xe-135m	Controlled tunnel purge (release detected onsite only)	Controlled release activity at time of release: 5.9 Controlled release activity at R+12 hours: 4.6	Controlled ventilations occurred as follows: 1. Controlled ventilation from working point side of U12n.20 drift was conducted from H+1.9 days until H+2.5 days. Effluent was 80% Xe-135, 18% Xe-133, and 2% Xe-133m. 2. Controlled release occurred during ventilation of MILL YARD cavity from H+16 days until H+18 days. Effluent was 98% Xe-133 and 2% Xe-133m.
Diamond Beech	10/9/85	Tunnel	Weapons effects <20 kt	LLNL DOD	U12n.19	Xe-133, Xe-133m, Xe-135	Controlled tunnel purge (release detected onsite only)	Controlled release activity at time of release: 1.1 Controlled release activity at R+12 hours: 1.0	Controlled ventilations occurred as follows: 1. Ventilation of tunnel to portal side of U12n.19 DPP occurred from H+1.8 days until H+2.5 days. Effluent was 80% Xe-135, 11% Xe-133m, and 9% Xe-133. 2. Ventilation of U12n.19 main drift occurred from H+8 days until H+9 days. Effluent was 82% Xe-133 and 18% Xe-133m.
Mighty Oak	4/10/86	Tunnel	Weapons effects <20 kt	LLNL DOD	U12t.08	Kr-85, I-131, Xe-133	Controlled tunnel purge (offsite) Controlled release of radioactivity detected offsite	Controlled release activity at time of release: 3.6E+04 Controlled release activity at H+12 hours: 3.3E+04	Eight controlled ventilations occurred. ^(b)

Table 5D-21 (Continued). Releases from underground tests.

Test name	Date	Type	Purpose & yield	Sponsor	Location	Radionuclides identified in release	Release type & notes	Release (Ci)	Release summary
Labquark	9/30/86	Shaft	Weapons-related 20-150 kt	LLNL	U19an	Xe-133, Kr-85	Late-time seepage (onsite only)	Natural, late-time seepage activity at time of release: 16 Xe-133: 2.6 Kr-85: 13	Two late-time releases, due to seepage, occurred from October 25, 1986 to January 13, 1987.
Bodie	12/31/86	Shaft	Weapons-related 20-150 kt	LLNL	U20ap	Xe-133, Xe-133m, Xe-131m, Kr-85, Ar-37, Xe-135	Drill-back and late-time seepage (onsite only) ^(c)	Drill-back release activity at time of release: 1.2 Xe-133: 1.2 Xe-133m: 4.8 E-02 Xe-135: 9.8 E-04 Natural, late-time seepage activity at time of release: 50 Xe-133: 44 Xe-133m: 2.0 Xe-131m: 1.0 Kr-85: 2.0 Ar-37: 1.0	Five drill-back releases occurred from ventilation line from 2307 hours on December 20, 1986, until 0215 hours on December 21, 1986, for total release time of 24.3 minutes. Seepage occurred continuously from December 15, 1986, to January 20, 1987, and sporadically, depending on atmospheric pressure, until December 16, 1987.
Mission Ghost	6/20/87	Tunnel	Weapons effects <20 kt	LANL DOD	U12t.09	Kr-85	Controlled tunnel purge (onsite only) ^(c)	Controlled release at time of release: 3.0 Kr-85: 3.0	Activity was contained in cavity until ventilation was established on December 16, 1987. Release continued intermittently for about 3 weeks.
Releases from atmospheric tests 1951-1963								1.2 E+10	
Other releases from 108 tests from 1970-1988 ^(d)								5.5 E+03	

Table 5D-21 contains information for NTS underground tests. A similar table is planned for NTS atmospheric tests [Reserved]. The "isotopes identified in the release" are from offsite monitoring (see Schoengold, DeMarre, and Kirkwood 1996) and are not necessarily the isotopes of concern for workers.

- (a) Containment failures are normalized to 12 hr after the test.
- (b) The camphor failure includes Ci-140 from tunnel purging.
- (c) Bodie and Mission Ghost had drill-back releases.
- (d) Many operational releases are associated with tests that were not announced.
- (e) Isotopic analysis from the Baneberry Test Manager's Rad Safe Advisor Status Report (DOE 1970).
- (f) MIGHTY OAK eight controlled ventilations occurred as follows:

1. Controlled ventilation from the gas seal plug (GSP) to the DPP was performed from 0950 hours on April 22 to 0611 hours on April 23, 1986. At the time of release, 340 Ci of activity were released (calculated to be 316 Curies at P+12).
2. Controlled ventilation of the tunnel complex, work point side of the DPP, was performed from 1040 hours to 1440 hours on April 25. At the time of release, 3,400 Ci were released (calculated to be 3,200 Ci at P+12).
3. Controlled ventilation of the tunnel complex occurred from 1002 hours on April 28 to 0310 hours on April 29. At the time of release, 9,800 Ci were released (calculated to be 9,100 Ci at P+12).
4. Controlled ventilation of the tunnel complex occurred from 1034 hours to 1504 hours on April 29. At the time of release, 1,800 Ci were released (calculated to be 1,700 Ci at P+12).
5. Controlled ventilation of the tunnel complex occurred from 1422 hours to 1805 hours on April 30. At the time of release, 1,200 Ci were released (calculated to be 1,100 Ci at P+12).
6. Controlled ventilation of the tunnel complex occurred from 1011 hours to 1937 hours on May 1, 1986. At the time of release, 4,900 Ci were released (calculated to be 4,600 Ci at P+12).
7. Controlled ventilation of the tunnel complex occurred from 0946 hours on May 2 to 0450 hours on May 4. At the time of release, 9,000 Ci were released (calculated to be 8,400 Ci at P+12).
8. Controlled ventilation of the tunnel complex occurred from 1350 hours on May 5 to 1050 hours on May 19. At the time of release, 5,500 Ci were released (calculated to be 5,100 Ci at P+12).

Notes: The total release, at the time of release, was 36,000 Ci; at P+12, the total calculated activity was 33,000 Ci. The total release associated with MIGHTY OAK was assumed to be all ¹³³Xe, but during the ventilation period, 2.4 Ci of I-131 and 4.3 Ci of Kr-85 were also released. All ventilations of the tunnel were accomplished with the approval of the Test Controller.

H = Time of detonation (NCI 1997). For example, "H + 12" means 12 hours after detonation.

R = Time of release (Schoengold, DeMarre, and Kirkwood 1996).

P = Time of controlled ventilation ("purge") (Schoengold, DeMarre, and Kirkwood 1996).

detected off the site from U12b (B Tunnel). In summary, nine REEC Co personnel received thyroid doses in excess of 30 rads during YUBA (Arent and Smith 2004).

- *Operation Whetstone, On-Site Radiological Report* (REEC Co 1965a) for 1964 and 1965 describes exposure to internal radiation up to NTS-SOP-0524 limits. Two post-shot drilling personnel received internal doses to the thyroid above the quarterly limit of 10 rem. One of these exceeded the annual limit (30 rem) with a dose of 31 rem.
- *Historical Radiation Records for Operation Whetstone (July 1964 – June 1965), #20 Merlin Event, U3ct* (REEC Co 1965b) describes an iodine incident resulting in doses of 27 and 31 rem to the thyroid (^{131}I and ^{133}I , respectively) for two employees involved in post-shot drilling. These levels were above permissible exposure limits. Six employees received measurable thyroid radiation exposures. A Type B Investigation was conducted (REEC Co 1965b).
- Extensive records for the 1970 Baneberry test are available on the OCAS O Drive. These include summary reports from the DOE Nevada Operations Office (DOE-NV), REEC Co radiological safety reports and procedures, and records of monitoring and decontamination efforts (REEC Co 1973). On December 18, 1970, the Baneberry underground nuclear test at the NTS released radioactivity to the atmosphere. The release or venting resulted in a cloud of radioactive dust about 10,000 ft above the surface. Levels of radioactivity measured off the NTS were below radiation guidelines. Approximately 86 employees were exposed to radiation from Baneberry, but none received exposure that exceeded the guideline for radiation workers. Following Baneberry, new containment procedures were adopted to prevent a similar occurrence.
- A report entitled *Evaluation of Protection, Bioassay, and Dose Assessment Programs for Internal Radiation Exposures at the NTS Particularly as Related to Three Exposure Situations* (French and Skrable 1995) describes internal uptakes at NTS E Tunnel (^{239}Pu), Nellis Air Force Range Double Tracks (^{239}Pu) at the Tonopah Test Range, and Building A-1 (Atlas Facility) in Las Vegas (^3H) during the mid-1990s. The three situations included (1) the exposure of 24 workers to ^{239}Pu aerosols while working in the E-Tunnel from June to August, 1994, possibly from removal of timbers from the walls and roof of the tunnel, which might have had high specific activity; (2) the exposure of one worker in June 1995 to ^{239}Pu aerosols believed to have been generated while handling and characterizing high-specific-activity fragments found during the Double Track Site soils characterization project; and (3) the exposure of workers in 1995 to tritium released from metal foils in Building A-1 at the Atlas Facility on Losee Road in North Las Vegas.
- French and Skrable noted that the bioassay data from the E-Tunnel workers were limited in both the quantity and quality needed for making reasonable accurate estimates of their intakes and doses.
- French and Skrable estimated the worker at the Double Track Site (^{239}Pu aerosol) had an effective ingestion intake of 3,260 pCi from nonrespirable aerosols with a committed effective dose of 0.2 mrem, and an inhalation intake of 360 pCi of 1 μm AMAD class Y aerosols with a committed effective dose of 90 mrem. The activity of ^{239}Pu in fecal samples of the E-Tunnel workers did not exceed 1 pCi and dose estimates ranged from 0 to 1,337 mrem based on the assumption of inhalation intakes of respirable aerosols (French and Skrable 1995).
- For the tritium exposure, French and Skrable concluded that considerable bioassay and air sampling data confirmed that the exposures were minimal with little dose consequence, and

no further dose estimates were provided. In 1995, the previous contractor (EG&G) discovered contamination in the same basement as a result of routine bioassay samples from workers. The source of contamination was later found to have resulted from a modification of a sealed-tube neutron generator by a scientist employed by that contractor. This modification included breaching the sealed tube that contained three tritide disks (130 Ci tritium total). According to the DOE NTS *Annual Site Environmental Report for Calendar Year 1995* (Black and Townsend 1996), the maximum release into the building was 123 millicuries.

5D.4.3 Respiratory Protection Practices at NTS

The following paragraphs provide examples of NTS respiratory protection practices. Historical accounts of respirator usage are included to give the dose reconstructor a chronological snapshot of the program that was in place during the early testing period. This is not a comprehensive review of respiratory protection practices at NTS.

The Operation Ranger Report (LASL 1951) stated that all persons entering a target area where background was higher than 3 mr/hr received respirators and that bulldozer operators received respirators with plastic hoods.

Project 56 was an atmospheric safety experiment that studied plutonium dispersal in NTS Area A-11 from 1955 to 1956. Documentation (AEC 1956) stated that respirators or "full-face assault masks" were required for persons entering the contamination area (CA). Nose swipes were required on leaving the CA; urinalysis was required for anyone who entered the CA at end of operations. AEC (1956) contains details on alpha air levels with workplace air sample results.

A memorandum for the Plumbbob Operation (LRL 1957a) includes a detailed step-by-step narrative for a Saturn Tunnel (U12c) reentry conducted on December 12, 1957. The reentry party was fitted with "airpacks." The memo lists measured contamination levels on personnel protection equipment (PPE; up to 10^6 alpha cpm) and includes individual nasal swipes results (sanitized). A second memorandum (LRL 1957b) details the August 30, 1957 reentry into U12c in which team members wore respirators and Mine Safety Appliances (MSA) Company air packs. Specific results included individual nose swipe results (sanitized), filters from respirators measured at 2,000 to 3,000 cpm (after 3 days, the count was the same), and filter paper on the air sampler at the blast door that measured $>100,000$ cpm after a 3-day decay. Urine samples were reportedly submitted for plutonium analysis, but the memorandum included no results.

A 1958 REECO memorandum entitled "Decontamination of Tunnel U12f" (REECO 1958), discussed PPE worn by the reentry teams. The first team wore air-supplied masks and full anticontamination clothing. Alpha contamination was measured at >500 c/m/55cm² in the main drift from Drift 01 and 2×10^3 c/m/55cm² in Drift 02 to the blast door. The second team wore full-face MSA masks with all-service canisters and anticontamination clothing. The third decontamination team wore Willson full-face masks and anticontamination clothing; nasal swabs were collected. The memorandum stated "air sampling and nasal swabs indicated no significant air activity during latter stages of decontamination and afterwards."

Alpha Emitter Dispersal & Decontamination (Wilcox and Coogan 1959) described what to expect with an atomic weapon accident. Recommendations were based on NTS experience with large land areas contaminated with alpha-emitting material. The CA was defined as $>2 \times 10^3$ d/m/55cm². Full-face masks equipped with dust, fume, mist filters were worn and nasal swabs were collected at the exit of the CA. If nasal swab results were >200 d/m (total both nostrils), workers contributed a urine sample. The report stated: "Airborne radioactive material was generally below permissible levels, but windy

conditions occasionally caused concentration of airborne activity to 4×10^{-9} $\mu\text{Ci}/\text{mL}$. (Resuspension of radioactive particulate was reduced by keeping the area moistened with water.) Sample analysis indicated that of the more than 1,000 separate exposures to airborne alpha-emitters, no person received significant internal exposure."

General Re-entry Procedure for Underground Nuclear Events (LRL 1961) described PPE worn during tunnel reentry. A McCaa 2-hr self-contained oxygen breathing apparatus and full Radex clothing were specified for the reentry party and rescue team. MSA all-service gas masks and full Radex clothing were designated for the surface radiological survey party (Ranier shaft). Appendix A of this document is the Re-entry Program for the YUBA test in 1963, in which an accidental release of radioactivity was detected on the site and an operational release of radioactivity was detected off the site from U12b (B Tunnel).

Palanquin Reentry and Recovery Safety Procedure (LRL 1965) stated that all reentry and recovery personnel will wear full protective clothing and respiratory protection (full-face mask with MSA Model "CMR" cartridge or equivalent). Workers were fit-tested with smoke tubes immediately before entry. Bioassays were to be performed if internal exposure to workers was suspected. "Spot" bioassay samples were to be taken from representative workers after reentry to verify the effectiveness of the control program.

Table 5D-22 lists NTS respiratory protection levels for selected years (REECo 1993b).

5D.4.4 Historical Practices and Contamination Levels

The Plumbbob On-Site Radiological Safety Report (REECo 1957) is an example of the magnitude of early radiological support; it states the following samples were collected:

The Defense Nuclear Agency (DNA) report on the 1957 Plumbbob Operations (DNA 1981) describes REECo activities, checkpoint during reentry, Radex area cards, and PPE in Radex areas (>100 mR/hr gamma). It also describes radiation safety support for the Project 57 internal alpha radiation hazard, and discusses REECo and Sandia surveys and PPE. Nose swipes and nasal swabs were analyzed by REECo. Urine samples were packaged and shipped off the site for analysis.

Table 5D-22. Historical NTS respiratory protection action levels.

Year	Respiratory protection action levels
1957	No respirator $\leq 9.6 \times 10^3$ dpm/m ³ hrs (alpha) per quarter year
	No respirator (short periods) ≤ 100 dpm/m ³ (alpha)
	Respirator required > 100 dpm/m ³ (alpha)
	No respirator $\leq 2.2 \times 10^4$ dpm/m ³ (beta + gamma)
	Respirator required $> 2.2 \times 10^4$ dpm/m ³ (beta + gamma)
	Note: Respirators using filters have an efficiency up to 99.9% for particles as small as 0.3 micron in diameter.
1959	Ultra-filter respirator can be worn to levels < 500 dpm/m ³ (Pu-239 alpha)
	Full-face mask with dust, fume, and mist canister required in levels from 500 to 10,000 dpm/m ³ (Pu-239 alpha)
	Air-supplied mask recommended when levels $> 10,000$ dpm/m ³ (Pu-239 alpha)
1968	0 to 100 dpm/m ³ (alpha)—no respiratory protection for short-term exposure.
	4×10^{-11} $\mu\text{Ci}/\text{cc}$ (beta, gamma)—respiratory protection required
	100 to 100,000 dpm/m ³ (alpha)—high-filtration, full-face respirator (99.9% effective)
	Greater than the above levels—self-contained breathing apparatus

Nasal swabs	4,230
Facial swipes	109
Air samples	1,945
Urine kits issued	73
Respirators issued	10,755

Radiological Safety for Underground Nuclear Explosions (REECo 1960) describes general hazards, nasal swabs and urine samples, respirators, and air monitoring in tunnels. Workers received anticontamination clothing (coveralls, head covers, gloves, shoe covers) and respirators. Workers leaving the area at the end of the shift were monitored and decontaminated if necessary. Air samples were obtained and nasal swabs were taken to determine possible internal exposures. Individuals whose nasal swabs measured $>3 \times 10^3$ d/m (beta, total both nostrils) or 2×10^2 d/m (alpha, total both nostrils) were requested to submit urine samples. These samples were processed in the laboratory to determine body burdens of gross fission products or alpha emitters.

Procedure to Limit Radiological Exposures of Miners (DOE 1962) stated the following practices were to be implemented for underground operations:

- Anticontamination clothing and respiratory equipment when conditions indicate they are necessary; impermeable clothing when the situation warrants.
- Urine samples for specific activities collected on a routine basis from personnel working in areas where it is possible to receive internal exposures; results kept on cumulative records and included in the daily estimated exposure.
- Film, dosimeter, and internal exposure results added together to determine the daily estimated dose of each person; forwarded to Superintendent before shift (rotate men).
- Low-pressure weather – various radioactive gases seep into work areas from cracks. Evacuate area.
- Increased and expanded vent system to clear radioactive gases and reduce chance for internal exposure.
- One-side drift engineering of new tunnels.
- Table 5D-23 lists NTS contamination limits for selected years. Contamination limits were often used as indicators of when bioassay samples were to be taken.

5D.5 REFERENCE TABLES FOR DETERMINING INTERNAL DOSE

At NTS, which is a large outdoor testing facility, there was never a process that carried over from year to year; these differences have been identified as test categories in this document. For determining dose, assume that acute exposure occurred within one work shift. More than 1,000 different tests have occurred at NTS. Very few workers have been directly involved in these tests and they have been identified in the “other monitoring” section of the DOE file for each claim. These are the workers who should be considered for potential iodine dose, tritium dose, or other unique internal dose consideration if their exposure can be linked to a specific release (see Table 5D-21).

Note: If the file provided by DOE does not include “other monitoring,” dose reconstructors should base the internal dose assigned to the worker on the environmental ambient.

The following tables provide information about the characteristics of the source term at NTS. The dose reconstructor needs to keep in mind that this source term changes over the years based on the test category. Information has been provided about the radionuclides of concern as the primary focus; however other radionuclides are present depending on the timeframe and test category that the dose reconstructor might have to address. The dose reconstructor should use professional judgment when including or deleting specific radionuclides in the source term based on the test category with which the worker was associated and the cancer type.

Table 5D-23. NTS historical contamination limits.

Year	Contamination limits
1957	Personnel: 1 mR/hr (gamma) and 100 cpm/55 cm ² (alpha) Protective clothing: 7 mR/hr (beta + gamma) and 500 cpm/55 cm ² (alpha) Respiratory devices: 1 mR/hr (beta + gamma) and 100 cpm/55 cm ² (alpha)
1958	Coveralls must be removed for eating if alpha levels on the coveralls are > 1 × 10 ³ dpm/55 cm ² . Decon personnel at >: - Outer clothes: 500 cpm/55 cm² (alpha) or 7 mR/hr (beta + gamma) - Shoes: 500 cpm/55 cm² (alpha) or 7 mR/hr (gamma) - Skin or underclothing: 100 cpm/55 cm² (alpha) or 1 mR/hr (gamma) - Equipment: <7 mR/hr (gamma) or 500 cpm/55 cm² (fixed alpha) Respiratory devices: < 1 mR/hr (beta + gamma) or 100 cpm/55 cm ² (alpha fixed and removable) Alpha contamination >5,000 cpm/55 cm ² requires field decon. Access permits required to enter areas with beta-gamma levels >10 mR/hr or with alpha levels >500 cpm/55 cm ² .
1959	Nasal swab results >200 dpm (total for both nostrils) required urine samples. Outer pair of coveralls removed to eat if contaminated to levels in excess of 500 cpm/55 cm ² .
1963	Laundry items >1 × 10 ⁴ dpm/55 cm ² (alpha) or 20 mrad/hr (beta-gamma) are disposed of or retained until radioactive decay reduces activity levels. Clothing and equipment are released for reissue after laundering if < 7 mrad/hr (beta-gamma) or 1,000 dpm/55 cm ² (alpha) for coveralls and 1 mrad/hr (beta-gamma) or 200 dpm/55 cm ² (alpha) on respirators.
1970	Anticontamination coveralls must be free of loose contamination and below 100 cpm alpha before reuse. Beginning in 1959, studies were conducted to determine effects of environmental forces on shifting and resuspension of deposited alpha contamination and soil penetration since initial deposition or fixation (by windrowing).

Source: REEC0 1993b

Table 5D-24. Solubility types for radionuclides found at NTS.

Radionuclides of concern	Solubility type
Americium-241/243	M
Cesium-137	F
Curium-244	M
Iodine-131/133	F
Plutonium-239	M or S
Radium-226	M
Strontium-85/90	F or S
Strontium-90/Yttrium-90	F or S
Thorium-228/232	F or S
Tritium	F
Uranium-238	F, M, or S

Radionuclides of concern	Solubility type
Other radionuclides identified at NTS	
Antimony-125	F or M
Cesium-134	F
Cerium-141/144	M
Colbalt-60	M or S
Europium-152/154/155	M
Iron-59	F
Manganese-54	F
Polonium-210	M
Niobium-95	M or S
Ruthenium-106	F or S
Zirconium-95	F or M

Table 5D-25. Fission products up to 1 year old as identified in 1959 documentation.

Radionuclide	Half-life	Beta energy (MeV)
Strontium-89	53 days	1.463
Strontium-91	9.7 hours	2.665 - 0.62
Strontium-90	19.9 years	0.61
Yttrium-90	61 hours	2.18
Yttrium-91	61 days	1.55
Yttrium-92	3.6 hours (?)	3.5
Yttrium-93	10 hours	3.1
Zirconium-95	65 days	0.84 – 0.371
Zirconium-97	17.0 hours	1.91
Niobium-95	35 days	0.16
Niobium-97	72.1 minutes	1.4
Molybdenum-99	67 hours	1.23 – 0.45
Ruthenium-103	39.8 days	0.22 – 0.7
Ruthenium-106	1.0 year	0.039
Rhodium-106	30 seconds	3.55 – 2.30
Rhodium-103	57 minutes	?
Rhodium-105	36.5 hours	0.570 – 0.25
Tellurium-132	77.7 hours	1.3 – 2.4
Iodine-131	8.14 days	0.815-0.250
Iodine-132	2.4 hours	2.2 – 0.9
Iodine-133	20.5 hours	1.3 – 0.4
Iodine-135	6.68 hours	1.4 0.5
Xenon-133	5.27 days	0.34
Xenon-135	9.13 hours	0.9
Cesium-137	33 years	0.51
Barium-137m	2.6 months	-
Barium-140	12.8 days	1.02 – 0.47
Lanthanum-140	40 hours	1.3 – 2.26
Lanthanum-141	3.7 hours	2.43 – 0.9
Cerium-141	33.1 days	0.58 – 0.442
Cerium-143	33 hour	1.39 – 0.71
Cerium-144	282 day	0.3 – 0.17
Praseodymium-144	17.5 minutes	2.97
Praseodymium-143	13.7 day	0.92
Neodymium-147	11.3 day	0.83 – 0.38
Promethium-147	2.6 year	0.22
Promethium-149	54 hour	1.05

Table 5D-26. Other common radionuclides not normally a part of fission products that might be present.

Radionuclide	Half-life	Beta energy (MeV)
Curium-242	162 day	6.1
Neptunium-239	2,433 day	0.715 – 0.33
Americium-243	10 year	5.27
Plutonium-241	14 year	0.02

Table 5D-27. Radionuclide activity ratios at formation (immediately after detonation).

Numerator	H-3	Sr-89	Sr-90	Y-91	Mo-99	Ru-103	Ru-106	I-131	Cs-135	Cs-136	Cs-137
H-3	1										
Sr-89		1	1.47E+2	7.95E-1	2.45E-1	3.86E-1	7.59	1.41E-1	7.87E+6	1.90E+1	1.01E+2
Sr-90		6.82E-3	1	5.42E-3	1.67E-4	2.64E-3	5.18E-2	9.64E-4	5.37E+4	1.30E-1	6.92E-1
Y-91		1.26	1.84E+2	1	3.08E-2	4.85E-1	9.54	1.78E-1	9.90E+6	2.39E+1	1.27E+1
Mo-99		4.07E+1	5.97E+3	3.25E+1	1	1.57E+1	3.09E+2	5.75	3.21E+8	7.75E+2	4.14E+3
Ru-103		2.59	3.80E+2	2.06	6.33E-2	1	1.97E+1	3.66E+1	2.04E+7	4.92E+1	2.63E+2
Ru-106		1.32E-1	1.93E+1	1.05E-1	3.23E-3	5.09E-2	1	1.86E-2	1.04E+6	2.51	1.34E+1
I-131		7.08	1.04E+3	5.63	1.73E-1	2.73	5.37E+1	1	5.57E+7	1.35E+2	7.18E+2
Cs-135		1.27E-7	1.86E-5	1.01E-7	3.12E-9	4.90E-8	9.64E-7	1.79E-8	1	2.41E-6	1.29E-5
Cs-136		5.26E-2	7.71	4.18E-2	1.29E-3	2.03E-2	6.10E-2	7.43E-3	4.14E+5	1	5.34
Cs-137		9.86E-3	1.45	7.85E-3	2.42E-4	3.81E-3	7.48E-2	1.39E-3	7.76E+4	1.87E-1	1

Table 5D-28. Specific activity of selected alpha emitters.

Radionuclide	Specific activity (mCi/mg)
Th-230	1.94E-2
U-232	2.14E+1
U-233	9.47E-3
U-234	6.18E-3
U-235	2.20E-6
U-236	6.34E-5
U-238	3.33E-7
Pu-236	5.31E+2
Pu-238	1.74E+1
Pu-239	6.13E-2
Pu-240	2.26E-1
Pu-242	3.90E-3
Am-241	3.24
Am-243	1.85E-1

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GLOSSARY

containment failure

Unintentional release of radioactive material to the atmosphere due to a failure of the containment system. Called *venting* if it is a prompt, massive release; or *seep* if it is a slow small release that occurs soon after the test.

alert level (circa 1980s)

The result that calls the attention of the observer to the fact that further action pertaining to the sample or the person from whom it came is required. The action might consist of a more informative level of analysis of the same sample, a request for additional samples, or a recommendation of *in vivo* counting of the person involved. Alert levels for specific analyses follow:

gamma

When the gamma count rate of a sample reaches 5,000 cpm, the sample is transferred to a gamma spectrometer for further analysis. Otherwise, a ^{137}Cs -equivalent activity concentration ($\mu\text{Ci/cc}$) is reported.

Each positive radioiodine result is noted and compared to a curve relating alert level activity concentrations to the time elapsed between exposure and urine sample void time. A result at or near the alert level indicates that thyroid counting of the individual is to be considered.

The alert level for routine quarterly urine samples is “any detectable” for ^{91}Y , ^{103}Ru , ^{106}Ru , ^{106}Rh , ^{141}Ce , ^{144}Ce , ^{144}Pr , and ^{154}Eu .

Alert levels for all other gamma emitters are activity concentrations indicative of body burdens that are 10% of the maximum permissible body burdens (MPBB) for continuous exposure given in *Health Physics* 3, “Report of ICRP Committee II on Permissible Dose for Internal Radiation (1959),” June 1960.

tritium

The alert level for a single tritium sample is $1.53 \times 10^{-2} \mu\text{Ci/cc}$. The dose to infinity to the fat-free soft tissue of the whole body indicated by a single exposure resulting in this concentration is 50 mrem.

GFP - alert level was $10^{-7} \mu\text{Ci/cc}$.

Pu (or other transuranic elements) and Sr – any detectable level is the alert level.

controllable area

An area in which remedial actions are feasible. Criteria for a controllable area as defined by DOE are “... areas where trained rad-safe monitors are available, where communications are effective (where the exposure of each individual can be documented), where people can be expected to comply with recommended remedial actions, and where remedial actions against uptake of radionuclides in the food chain are practicable.” This equates to a zone of approximately 125 miles from the test control point.

controlled tunnel purging

An intentional release to allow recovery of experimental data and equipment or reuse of part of the tunnel system.

late-time seeps

Small releases that occur days or weeks after a test when gases diffuse through pore spaces of the overlying rock and are drawn to the surface by decreases in atmospheric pressure.

lower limits of detectability (LLD, circa 1980s)

The smallest amount of a sample activity that will be reported positive with a specified degree of confidence (95%). Before January 1, 1978, LLD = 2 so. After January 1, 1978, LLD = 3.29 so. The term so is the estimated standard error for the net sample activity (before 1993). REEC0 1993a states the LLD is a value selected above the MDA to reduce the probability of reporting false positive results.

Minimum Detectable Activity (or Amount) (MDA)

An *a priori* value used to evaluate the laboratory's ability to detect an analyte in a sample (BN 2003).

operational release

Small consequential releases that occur when core or gas samples are collected or when the drill-back hole is sealed.

Radex Areas

In a Full Radex Area, radiological contamination is ≥ 100 mR/hr (gamma) measured 3 ft from the ground, or $> 10^4$ cpm/55 cm² (alpha surface contamination measured by portable alpha survey meter). No one was allowed into a Full Radex Area unless accompanied by a certified monitor who remained with or near them during the entire period. In a Limited Radex Area, radiological contamination is ≥ 10 mR/hr but < 100 mR/hr (gamma) measured 3 ft from the ground, or $\geq 10^3$ to $\leq 10^4$ cpm/55 cm² (alpha surface contamination measured by portable alpha survey meter). No one was allowed into a Limited Radex Area unless accompanied by a certified monitor who would initially survey the area and return periodically to check radiological conditions.

reentry

In the context of NTS operations over the entire period of weapons testing, the first entry into an area or tunnel immediately (as soon as safety limitations will permit) following a nuclear detonation. Reentry is under the control and responsibility of the Technical Director assigned by the Test Manager. Reentry does not cover subsequent cleanup operations or preparing for a future detonation.

reporting level (1993)

The reporting level for the period 1993-2003 is defined as the minimum level of a bioassay measurement result, which requires the measurement lab to provide prompt notification to Dosimetry (REEC0 1993a).

seep

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

tolerance level

The term for maximum permissible exposure before MPD was coined.

venting

Prompt, massive, uncontrolled releases of radioactive material. Ventings are characterized as active releases under pressure, such as when radioactive material is driven out of the ground by steam or gas.