DOE Review Release 08/27/2012

Document Title:	Document Number:	ORAUT-TKBS-0008-5
Nevada Test Site – Occupational Internal Dose	Revision:	03
	Effective Date:	08/17/2012
	Type of Document:	TBD
	Supersedes:	Revision 02

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Approval:	Signature on File James W. Neton, Associate Director for Science	Approval Date:	08/17/2012

☐ New ☒ Total Rewrite ☐ Revision ☐ Page Change

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PUBLICATION RECORD

EFFECTIVE DATE	REVISION NUMBER	DESCRIPTION
09/30/2004	00	Technical Basis Document for the Nevada Test Site – Occupational Internal Dose. First approved issue. Incorporates formal internal and NIOSH review comments. Initiated by Eugene M. Rollins.
07/20/2006	00 PC-1	Approved page change initiated to incorporate definitions and directions for dose reconstruction for non-presumptive cancers that are excluded from the 1951 through 1962 Special Exposure Cohort. Text was added or modified on pages 8-10 in Section 5.1. Approved issue of Rev 00 PC-1. No sections were deleted. This revision results in no change to the assigned dose and no PER is required. Training required: As determined by the Task Manager. Initiated by Eugene M. Rollins. Approval: Signature on File 07/17/2006 Eugene M. Rollins, TBD Team Leader Signature on File 07/17/2006 John M. Byrne, Task 3 Manager Signature on File 07/18/2006 Edward F. Maher, Task 5 Manager Signature on File 07/19/2006 Kate Kimpan, Project Director Signature on File 07/20/2006 James W. Neton, Associate Director for Science
05/28/2008	01	Approved Revision 01 to incorporate additional information regarding bioassay procedures used and expanded MDAs. Addresses comments from the Advisory Board's contractor, and includes guidance to the dose reconstructor in Section 5.6. Adds Attributions and Annotations Section 5.7. Incorporates formal internal and NIOSH review comments. Adds Sections 5.6.4 and 5.6.5. Updates boilerplate language in Section 5-1 and adds reference to OCAS-IG-003. Constitutes a total rewrite of the document. Training required: As determined by the Task Manager. Initiated by Eugene M. Rollins.
04/15/2010	02	Revision initiated to incorporate definitions and directions for dose reconstruction for nonpresumptive cancers that are excluded from the 1963 through 1992 Special Exposure Cohort. Text was added or modified on pages 11 and 12 in Section 5.1.2, and pages 51, and 52 in Sections 5.3 and 5.4. Added Sections 5.1.2 and 5.3. Incorporates formal internal and NIOSH review comment. Training required: As determined by the Objective Manager. Initiated by Eugene M. Rollins.

08/17/2012	03	Revision initiated to revise ratios of ^{238}Pu and ^{241}Am to that of ^{239}Pu to reflect the scaling factors in Table 4-6 of Revision 02 of ORAUT-TKBS-0008-4, *Nevada Test Site – Occupational Environmental Dose* technical basis document. Revised text in Sections 5.2.2.3 and 5.6.4.2 to reflect these ratios. Revised text in Section 5.6.3 to be consistent with text in Section 5.6.2 about methods available and appropriate for assignment during SEC periods. Updated appropriate references. Incorporates formal internal and NIOSH review comments. Constitutes a total rewrite of the document. Training required: As determined by the Objective Manager. Initiated by Eugene M. Rollins.

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ACRONYMS AND ABBREVIATIONS

BEEF	Big Explosive Experimental Facility
BN	Bechtel Nevada
Bq	becquerel
CA	contamination area
CAM	continuous air monitor
cfm	cubic feet per minute
CFR	Code of Federal Regulations
Ci	curie
cm	centimeter
cpm	counts per minute
cps	counts per second
CWT	chest wall thickness
DAC	derived air concentration
DL	decision level
DOD	U.S. Department of Defense
DOE	U.S. Department of Energy
DOL	U.S. Department of Labor
DPP	drift protection plug
dpm	disintegrations per minute
DR	dose reconstructor
DU	depleted uranium
EE	Energy Employee
EEOICPA	Energy Employees Occupational Illness Compensation Program Act
EG&G	Edgerton, Germeshausen, and Greer Corporation
FSN	Fenix & Scisson of Nevada
FWHM	full-width, half-maximum
g	gram
gal	gallon
GeLi	germanium-lithium
GFP	gross fission product
gK	0.000118 g K-40
GSP	gas seal plug
H&N	Holmes & Narver
HEPA	high-efficiency particulate air
HEU	highly enriched uranium
HNS	Helgeson Nuclear Services
hr	hour
ICRP	International Commission on Radiation Protection
IMBA	Integrated Modules for Bioassay Analysis
in.	inch
IREP	Interactive RadioEpidemiological Program
keV	kiloelectron-volt, 1,000 electron-volts
kg	kilogram

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
L X-ray	low-energy X-ray
m	meter
MCi	megacurie
mCi	millicurie
MDA	minimum detectable activity or amount
MeV	megaelectron-volt, 1 million electron-volts
Mg	milligram
Min	minute
mL	milliliter
mm	millimeter
MPBB	maximum permissible body burden
MPC	maximum permissible concentration
MPC_a	maximum permissible concentration in air
$MPCU_a$	maximum permissible concentration in air for an unknown alpha emitter
mR	milliroentgen
mrad	millirad
mrem	millirem
mS	millisiemen
MSA	Mine Safety Appliances Company
NBS	National Bureau of Standards
nCi	nanocurie
NCRP	National Council on Radiation Protection and Measurements
NIOSH	National Institute for Occupational Safety and Health
NRDS	Nuclear Reactor Development Station
NTS	Nevada Test Site
NV	DOE Nevada Operations Office
OBP	overburden plug
ORAU	Oak Ridge Associated Universities
Pan Am	Pan American Airways
pCi	picocurie
POC	probability of causation
PPE	personnel protection equipment
POC	probability of causation
PPG	Pacific Proving Grounds
radex	radiation exclusion
RadSafe	radiological safety
RAMATROL	Radioactive Material Control
RAS	retrospective air sampler
RCT	Radiological Control Technician

REECo	Reynolds Electrical & Engineering Company
RSN	Raytheon Services Nevada
SEC	Special Exposure Cohort
SMS	Site Monitoring Services
SNL	Sandia National Laboratories
SRDB Ref ID	Site Research Database Reference Identification (number)
TBD	technical basis document
TLD	thermoluminescent dosimeter
TRU	transuranic
TTR	Tonopah Test Range
U.S.C.	United States Code
WBC	whole-body count or counter
WEF	Waste Examination Facility
WLM	working level month
WSI	Wackenhut Services, Inc.
yr	year
µCi	microcurie
µg	microgram
µm	micrometer
µS	microsiemen
§	section or sections

5.1 INTRODUCTION

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

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

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

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

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

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

5.1.1 Purpose

The purpose of this document is to describe the Nevada Test Site (NTS) internal dosimetry systems and practices. The Oak Ridge Associated Universities (ORAU) Team will use this information as needed to evaluate potential internal occupational doses for EEOICPA claims. This document will be used in conjunction with other sections of the site profile.

5.1.2 Special Exposure Cohort Petition Information for the Nevada Test Site

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

Classes Added to the SEC

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

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

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

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

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

5.1.3 Scope

NTS is unique in the DOE Complex because it has not hosted ongoing production of materials as other DOE facilities have. NTS was, and is, an outdoor testing and research facility rather than a manufacturing or processing site. When 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). For each test the laboratory responsible for the device was the responsible organization during the test, providing the scientists and technicians associated with the device. NTS personnel provided support to these teams.

Reynolds Electrical & Engineering Company (REECo), the NTS general contractor from 1952 to 1995, became responsible for onsite radiological safety activities in 1955 (Smith 2003). LANL and the military were the primary parties responsible for radiological safety oversight before 1955 (Smith 2003).

LANL conducted bioassay procedures from 1955 to 1958 (Smith 2003). In 1958, REECo began onsite bioassay (Smith 2003). By 1961, REECo was conducting routine bioassay for ^{3}H, ^{239}Pu, gross fission products (GFP), and gamma emitters (Author unknown undated a). Bechtel Nevada (BN), was the site contractor from 1995 to 2006, and as such had responsibility for the bioassay program (BN 1998, 2000; McMahan and Ogurek 2003). In 2001, BN disbanded the onsite analytical laboratory and contracted the services to an outside laboratory (Smith 2003). The current contractor is National Security Technologies, LLC (DOE 2007).

In 1967, an onsite whole-body counting capability was established (Teasdale 1985). 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, purpose, and size of the test. For specific tests the information about authorized external limits for certain job titles would be found in the Radiation Safety Plan. The Site Research Database (SRDB) includes examples of these plans.

The primary internal dose considerations during the nuclear underground testing era at NTS included tritium in some tunnel environments, venting [fission products (primarily iodines and xenon/krypton)], and seepage (usually xenon/krypton) released over time (Author unknown ca. 2004). Iodine-131 was a radionuclide of concern up to 10 days after a test; ^{133}I and ^{135}I were dose concerns up to 1 day after a test (Allen et al. 1993). Xenon, an external dose concern that is discussed in ORAUT (2008), could have been present in significant quantities several days after a test.

The information in this TBD characterizes NTS. For some claims the NTS 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 may in some cases be developed using the information found in this TBD, if documentation for the test identifies REECo as responsible for Radiation Safety.

NTS contractors include BN; Lockheed Martin Nevada Technologies; Johnson Controls Nevada; and Wackenhut Services (WSI). Former NTS contractors included Holmes & Narver (H&N), 1956 to 1990;

Fenix & Scisson of Nevada (FSN), 1963 to 1990; Edgerton, Germeshausen, and Greer Corporation (EG&G), 1951 to 1995; REECo, 1953 to 1995; and Raytheon Services Nevada (RSN), 1990 to 1995.

This TBD contains supporting documentation to assist in the evaluation of occupational internal doses from the activities performed by the above contractors using the methodology in the Internal Dose Reconstruction Implementation Guideline (NIOSH 2002). NIOSH considers the available data and methods for performing internal dose reconstruction to be adequate for estimating with sufficient accuracy the internal radiation doses at NTS from 1993 to the present.

Attributions and annotations, indicated by bracketed callouts and used to identify the source, justification, or clarification of the associated information, are presented in Section 5.7.

5.1.4 Historical 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 (Author unknown undated a):

> *"During the 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,"* [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.*

Black and Potter (1986) stated that during the early period of nuclear weapons testing, external radiation exposure was more hazardous than internal exposure when local fallout (i.e., within a few hundred miles of a test) caused both. This was affirmed many times by advisory groups and published data. Black and Potter cited examples from published literature to illustrate this point.

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 aboveground testing ended (NRC 2003). ORAUT (2010a) addresses environmental doses and discusses potential intake from resuspension. NTS is a large site with much infrastructure. The vast majority of workers had no direct contact with radioactive materials (e.g., emplacing devices, retrieval of instrumentation and samples after an event, etc.). Environmental levels discussed in ORAUT (2010a) characterize the radiological exposure for the majority of the NTS workforce.

5.1.5 Radionuclides of Concern

Technical Basis for Internal Dosimetry at NTS (Allen et al. 1993) reflects radiological protection practices from about 1970 to 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 2004). Subsequent technical basis documents by BN (BN 2000; McMahan and Ogurek 2003) cover the needs of NTS after 1992 and are not comprehensive on the support of nuclear weapons testing.

NTS operations and facilities had different sets of radionuclides of concern at different times and locations on the site. 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 A, Section A.4. These radionuclide-of-concern lists are applicable to individuals identified with the specific operation or test, AND who had a potential for exposure based on their job categories. Dose reconstructors (DRs) can use these lists if a realistic assessment of the worker's dose is necessary to evaluate potential exposure.

Iodine, krypton, xenon, and tritium were important radionuclides following nuclear tests. Krypton and xenon were primarily external hazards, whereas ^{131}I and ^{3}H were internal hazards (Glasstone 1971). Radionuclides that have resulted in recorded doses above established limits at NTS are ^{3}H, ^{131}I, ^{239}Pu, and ^{241}Am (Arent and Smith 2004).

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 (^{3}H), radium (^{226}Ra), and thorium (^{228}Th, ^{232}Th) (Allen et al. 1993). 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. Associated internal dose with atmospheric testing has not been developed further with the establishment of the SEC for the Pacific Proving Grounds (PPG) and NTS (Leavitt 2006a,b).

5.1.6 Bioassay Program Description

As noted in the Historical Perspective (Author unknown undated b), the dosimetry emphasis was on external monitoring. Early bioassay 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 for workers identified by their supervisors as having the potential for internal exposure based on their assigned job tasks. Routine bioassays at NTS included quarterly urine samples, annual whole-body count (WBC), and new/termination WBCs (Allen et al. 1993). Table 5-1 lists the types of bioassay monitoring and radionuclides discussed in the REECo/BN Bioassay Worksheet. These should be considered the routine types of bioassays conducted at NTS and can be seen in the case documentation.

Table 5-1. Types of bioassay monitoring and radionuclides.

Monitoring	Radionuclides
Urine	H-3 (activity),[a] Pu-238, Pu-239,[a] gamma emitters,[a] GFP (gross beta),[a] Sr-90
Fecal	Pu-238, Pu-239
WBC	Gamma emitters[b]
Lung count	U-238, Pu-238, Pu-239, Am-241
Thyroid	Radioiodines
Nasal swabs	Pu-239
Wound counting	Pu-238, Pu-239, beta, gamma
Wound swabs	H-3, gross alpha, gross beta, gross gamma

a. Routine analysis by REECo starting in 1961. NOTE: REECo started analyzing nasal swabs (Pu-239) and urine samples (Pu-239, GFP) in 1958; LANL conducted nasal and urine sampling for Pu-239 in 1955 and 1956.

b. Routine analysis by Pan American Airways (Pan Am) from 1967 to 1973, and by REECo at NTS in 1973. Before 1967, several analyses were done by Helgeson Nuclear Services (HNS).

By instrument survey, air sampling, and surface swipe analysis procedures, health physics personnel identified and isolated work sites where personnel might have encountered internal dose hazards. Appropriate measures were taken to prevent workers at such sites from inhaling or ingesting radioactive contaminants. Urine samples were obtained to monitor the effectiveness of these measures. The data resulting from the analysis of such samples were not ordinarily used for estimation of internal dose. If internal contaminants were detected in sufficient quantities to indicate that dosimetry was possible, special procedures were initiated. If gamma emitters were involved, *in vivo* measurement procedures such as whole body (or specific organ) counting were preferred. If the internal contaminant was tritium, it was possible to make an initial estimate of the dose commitment from a single urine sample result. Dosimetry of any of the alpha or pure beta emitters was attempted on the basis of a carefully collected and analyzed series of urine samples (REECo ca. 1980s).

By 1993, the common *in vitro* radiobioassay conducted at NTS included the following (Allen et al. 1993):

- Tritium in urine – 5 mL of distillate from a 30-mL sample were quantified by liquid scintillation beta counting (70 minutes).
- Strontium in urine – the strontium precipitated as a carbonate from a 900-mL sample was determined by beta counting (100 minutes).
- GFPs in urine – a 250-mL sample was determined by gross beta counting the oxide from a basic oxalate (100 minutes).
- Uranium in urine – an aliquot of a 1.5-L sample was converted to fluoride and analyzed fluorometrically.
- Plutonium and americium in urine – a 2-L sample processed by the phosphate precipitate method, purified by anion exchange, followed by electrodeposition and counted by alpha spectroscopy (passivated implanted planar silicon detector system, 1,000 minutes).
- Gamma in urine – 50 mL were counted with a germanium-lithium (GeLi) gamma spectroscopy system for determination of gamma-emitting radionuclide activities.
- Gamma in feces – 500 g were counted with a GeLi gamma spectroscopy system for determination of gamma-emitting radionuclide activities. Fecal ash samples were analyzed by alpha spectroscopy for radium, uranium, plutonium, and americium.

Allen et al. (1993) lists routine bioassay programs for specific operations. General bioassay programs for drill-backs, mine-backs, atmospheric testing, and nuclear rocket operations were as follows:

- Drill-backs – Collection of urine samples after each drill-back; gamma spectroscopy for gross gamma-emitting radioactivity and for radionuclide identification; annual WBC.

- Mine-backs – Collection of urine samples based on air sampling results; gamma spectroscopy and tritium analysis; annual WBC.

- Atmospheric safety tests – Quarterly urine; gamma spectroscopy and analysis for plutonium, and americium; annual WBC.

- Atmospheric weapon tests – Quarterly urine; gamma spectroscopy and analysis for GFP, plutonium, and americium. Annual WBC; some personnel involved in atmospheric weapons tests also received periodic lung counts. Involvement means being part of the reentry team or directly involved with the testing activity in the restricted area.

- Nuclear rocket development tests – Quarterly urine; gamma spectroscopy and analysis for GFPs, plutonium, and tritium. Annual WBC.

Workers with the potential for an intake [e.g., Miners, Drillers, Radiological Control Technicians (RCTs; also called Radiation Safety Monitors] participated in the routine bioassay program. The results of the urine sample analysis were used to determine if further bioassay measurements such as thyroid, lung, or whole-body counting were indicated. The urine sample results were also used to determine if processing of the sample for specific beta/alpha particle emitters or fecal analysis was necessary. BN (1998) includes a list of routine urine and fecal analysis frequencies by radionuclide:

- Weekly – ^{3}H
- Monthly – ^{131}I, ^{239}Pu
- Quarterly – ^{60}Co, ^{90}Sr, ^{137}Cs, ^{226}Ra, ^{234}U, ^{238}Pu (fecal), ^{239}Pu (urine and fecal), ^{241}Am

Nonroutine bioassay types included job-specific and occurrence response. Examples of nonroutine bioassays include the following analyses:

- Tritium or gamma scan
- GFP (beta)
- Specific radionuclides

WBCs, lung counts, thyroid counts, and biological sampling were performed as soon as practicable after a suspected intake (Allen et al. 1993). Specific examples include:

- Lung counts after a suspected intake of thorium, uranium, or a transuranic (TRU) element

- WBCs 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

Attachment A, Section A.4, summarizes facility descriptions and their specific routine bioassay programs (Allen et al. 1993). Bioassay samples might have been taken to document that the worker was not internally contaminated as well as to confirm a suspected intake (Author unknown undated c).

At present, a large amount of the radiological work at NTS is through projects with a specified (often short) duration. Routine bioassay monitoring is usually part of a project. A baseline sample is collected as necessary, and postwork samples are collected at the conclusion of the project. If a project requiring bioassay monitoring is likely to last longer than 6 months, bioassay sampling on a quarterly to semiannual frequency for the duration of the project is recommended. An exception to this could be a ^{3}H project with high exposure potential and a relatively high frequency of required monitoring (e.g., weekly to monthly), because typical field monitoring instruments do not detect ^{3}H (McMahan and Ogurek 2003). While this was the bioassay program developed and documented by the contractor, internal monitoring was not performed for most workers (see Section 5.1.3). For many job categories, internal dose based on the site environmental inventory will provide an overestimate of potential internal exposure (i.e., environmental dose can be evaluated as both internal and external).

5.2 *IN VITRO* MINIMUM DETECTABLE ACTIVITIES AND COUNTING METHODS

Decision Levels and Minimum Detectable Activities
From McMahan and Ogurek (2003), the Decision Level (DL) is calculated as:

$$DL = \frac{1.65\sqrt{R_b \times T_s \times \left(1 + T_s/T_b\right)} + 3}{K T_s} \quad (5\text{-}1)$$

where:

R_b = background count rate, C_b/T_b; C_b = background counts
T_s = sample count time
T_b = background count time
K = calibration factor in appropriate units such as counts in seconds per unit activity

According to McMahan and Ogurek (2003), 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. McMahan and Ogurek (2003) 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 nondetection (Type II error) while accepting a probability, alpha (Type I error), of erroneously deciding that a positive (nonzero) 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. 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:

$$MDA = \frac{3 + \left(3.29\sqrt{R_b \times T_s \times \left(1 + T_s/T_b\right)}\right)}{K T_s} \quad (5\text{-}2)$$

In Allen et al. (1993), the MDA equation was reported as:

$$MDA = [3 + (4.65(C_b)^{1/2})] \div ETVR \quad (5\text{-}3)$$

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 Allen et al. (1993) 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.

The tables in Attachment A, Section A.2.1, list the bioassay codes used in the records.

5.2.1 *In Vitro* Urine and Fecal Analysis

MDA Values for Urinalysis and Fecal Analysis
Table 5-2 lists current values of MDA for *in vitro* analyses of routine urine and fecal samples (McMahan and Ogurek 2003). Because no specific beginning dates were found, the document publication date should be used as the effective date. Section 5.2.2 and Attachment A, Table A-7 contain historic MDA values.

5.2.2 *In Vitro* Methods for Individual Radionuclides

Allen et al. (1993) states that periodic urine samples were collected and analyzed for ^{238}Pu, ^{239}Pu, elemental uranium, ^{234}U, ^{235}U, ^{238}U, ^{241}Am, ^{89}Sr, ^{90}Sr, ^{3}H, and GFPs. *In vitro* radiobioassays included:

- Tritium in urine by liquid scintillation beta counting
- Strontium in urine by beta counting
- GFPs 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

Table 5-2. 2003 target MDAs for *in vitro* bioassay sample analysis.

Parameter and analysis method	Sample type[a] and container	Minimum detectable amount[b]	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

Parameter and analysis method	Sample type[a] and container	Minimum detectable amount[b]	Reporting units
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: McMahan and Ogurek (2003).

a. Sample collection was 24 hours.
b. 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.
c. 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-hour void sample, obtaining reasonable recoveries, extending count times, and reasonable background levels.
d. Used in calibration activities

The following paragraphs discuss *in vitro* methods for specific radionuclides. Attachment A, Table A-7 lists the MDAs for these radionuclides. NOTE: The actual MDA publication date is listed in this section whereas the date ranges for the MDAs in Table A-7 are extrapolated to cover missing periods. The suggested priority for DRs with regard to MDAs and other detection thresholds is to use (1) the limit on dose report if available, (2) Table A-7 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 (Author unknown undated a). In 1993, the ^{131}I LLD was listed as 100 pCi/L urine (Allen et al. 1993). BN (1998) lists the ^{131}I MDA as 99 pCi/L. 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.

5.2.2.2 *In Vitro* Bioassay for Americium

The McClelland (1955) method for detecting 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} g 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 (McClelland 1955).

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 (McClelland 1958; Dummer 1958). The LASL MDA value for americium was 9×10^{-1} pCi/24-hr sample (ORAUT 2009).

REECo documentation (REECo 1969–1987a) 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 stated that ^{241}Am was analyzed using ^{243}Am as a tracer and separated from sample matrices by precipitations such as $Y(OH)_3$, YF_3, and $Y_2(C_2O_4)_3$. Americium-241 was separated from rare earth elements by oxidation. Final purification was accomplished using an anion exchange column. Americium-241/243 was electrodeposited on a stainless-steel disc and the activity determined by alpha spectrometry. From 1982 to 1987, the detection limit was 2×10^{-11}

µCi/mL for ^{241}Am in urine (REECo 1968–1987a). The routine bioassay program was not designed for the control of ^{243}Am. NTS did not specifically analyze ^{243}Am in urine. If an employee has a history of work at the low-level waste sites, Waste Examination Facility (WEF), or the radiochemistry laboratory, DRs should evaluate ^{243}Am intake through WBC and/or lung counting results if an intake is suspected. Allen et al. (1993) lists no urine MDA for ^{243}Am; it does list MDA values for WBCs and lung counts for ^{243}Am.

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 (Allen et al. 1993). The routine bioassay was a quarterly urine sample and the special bioassay consisted of an additional urine sample and a lung count. Allen et al. (1993) 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." BN (1998) stated the listed MDA values are typical of MDAs achieved at NTS. Individual MDAs, which were calculated per sample, varied depending on the sample matrix for *in vitro* measurements or person for *in vivo* measurements, the length of the count, and the background of the specific instrument used for counting. Attachment A, Table A-7 lists limits of detection for urine and fecal analysis. Table 5-2 lists the current MDAs.

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} g 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 (McClelland 1955). A tolerance level of 1 dpm/24-hr sample was used for plutonium in urine for Project 56, a plutonium dispersal experiment conducted in Area 11 (AEC 1956).

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 thenoyl trifluoroacetone. The extracted plutonium was oxidized to plutonium (VI) by hypochlorite and electrodeposited on a stainless-steel disc with a 38.5-mm^2 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 was 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 $\pm$5% were routinely obtained. The detection limit at the 99% confidence level was approximately 0.05 dpm/sample (3.6×10^{-13} g of plutonium). Samples showing 0.2 dpm/sample or more (1.5×10^{-12} g of plutonium) were rescheduled (McClelland 1958; Dummer 1958). The LASL MDA values for plutonium (total alpha) were 0.4 pCi/24-hr sample (biphosphate/alpha counting) until January 1957 and 0.03 pCi/24-hr sample (aluminum nitrate/neutron track analysis) after January 1957 (ORAUT 2009).

In 1961, urinalysis for plutonium included sample preparation, anion exchange, electrodeposition, autoradiograph, and alpha track counting (REECo 1961). The laboratory capability (limit of sensitivity) was 0.005 dpm/sample (Author unknown undated a). Further documentation of the limit of sensitivity for ^{239}Pu is described in Geiger and Whittaker (1961). In the evaluation of the procedure, 61 urine samples spiked with 0.027 pCi of ^{239}Pu; 79 blank urine samples were analyzed. The average

recovery was 84 ±18% (90% CL). The average blank was equivalent to 0.001 pCi of ^{239}Pu. 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 ^{239}Pu in a 24-hour sample could be easily detected. This is approximately 20% of the investigation limit and represents less than 5% of a maximum permissible body burden (MPBB) of ^{239}Pu. The exposure time for this method was listed as 10,000 minutes (Geiger and Whittaker 1961).

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

Allen et al. (1993) stated that NTS used alpha spectrometry, which could not differentiate between ^{239}Pu and ^{240}Pu, to analyze plutonium urine and fecal samples. Urine samples were analyzed by gamma spectroscopy, which could differentiate between the two radionuclides. 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 ^{239}Pu. A "typical" MDA for a 1,500-mL urine sample counted for 1,000 minutes was 0.028 pCi/L. The ^{238}Pu LLD was 0.01 pCi/L urine. Routine bioassay consisted of a quarterly urinalysis, and the special bioassay consisted of a combination of urine and fecal analysis plus a lung count. Attachment A, Table A-7 lists limits of detection for urine and fecal analysis. Table 5-2 lists current MDAs for plutonium radionuclides.

Plutonium at NTS consists of a mix of ^{238}Pu, ^{239}Pu, ^{240}Pu, and ^{241}Pu. In addition, ^{241}Am would be present from the decay of ^{241}Pu. Therefore, when evaluating missed or confirmed doses from plutonium, DRs should include the other radionuclides associated with ^{241}Pu. In the 1980s and early 1990s, extensive surveys were conducted at NTS to determine the inventory of radionuclides in the soils as a result of atmospheric testing that occurred before 1963. ORAUT (2010a) summarizes the results of these surveys. These data can be used to estimate the quantities of ^{241}Am and ^{238}Pu in relation to $^{239+240}$Pu. A comparison of the total soil inventories over the 19 area surveys indicates that the ratio of ^{241}Am to ^{239}Pu is 0.59 and the ratio of ^{238}Pu to ^{239}Pu is 0.91 (see Table 4-6 of ORAUT 2010a). Therefore, when estimating total radionuclide intakes associated with intakes of ^{239}Pu, DRs should use these ratios to determine the accompanying intakes of ^{238}Pu and ^{241}Am. DRs should estimate these intakes when calculating potential missed dose and dose from confirmed intakes of ^{239}Pu when positive bioassay data for ^{238}Pu and ^{241}Am are not available.

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 ±5% in the range of 1 to 250 µCi/L of tritium. Samples with higher concentrations could be determined with appropriate dilutions. The tolerance for tritium in urine at LASL was 250 µCi/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 (McClelland 1955). The LASL MDA value for tritium was 1 µCi/L (ORAUT 2009). In 1958, the procedure was the same except the tolerance for tritium in urine used was 85 µCi/L (McClelland 1958).

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 (Author unknown undated a). In 1971, the urine sample tritium "alert level" was listed as 1×10^{-3} µCi/cm^3 (Author unknown undated a). The detection limit for ^{3}H in urine was listed as 1×10^{-6} µCi/mL from 1982 to 1987 (REECo 1968–1987a).

Tritium monitoring started in 1958 with an MDA of 5 µCi/L used for urine samples (Arent and Smith 2004). Allen et al. (1993) 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. Table 5-2 lists the current MDAs. Attachment A, Table A-7 lists limits of detection for urine and fecal analysis. On NTS record forms, tritium monitoring can be called "ACTIVITY" or "ACT," "EVERGREEN," "MINT," or "T".

Tritium was reported separately in the dosimetry records. DRs 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 based on the MDAs listed in Table A-7. After 1977, the tritium bioassay MDA is such that if all of the monitoring results are below the MDA, the dose is less than 0.001 rem/yr and the dose does not need to be in the dose reconstruction. Individuals assigned to areas with the potential for tritium exposure were on the routine tritium bioassay program. In addition, a select number of WSI security personnel were on the routine tritium bioassay because they could have been assigned to any location on NTS (Rollins 2004). The job title (ORAUT 2008), incident summary (Section A.4), and facility radionuclides of dose concern (Section A.2) discussions could assist the DR in determining the assignment of a tritium dose.

5.2.2.5 *In Vitro* Bioassay for Uranium

Environmental concentrations of uranium are highly dependent on geographic location (MJW 2003). 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}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). The DR should note that these ratios apply to solid-state material under laboratory conditions. In the field, water interferes with the chemistry and the natural ^{238}U:^{234}U ratios can vary because, when uranium enters ground water, there might be differences in the leaching rate between the two radionuclides.

The exception to DU and HEU being the source of uranium for exposures is at the WEF. Uranium at the WEF is likely to be the ^{233}U radionuclide, which can be present in some of the TRU 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}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 atomic vapor

laser isotope separation method (Rich et. al. 1988, as cited in McMahan and Ogurek 2003). This type of enriched uranium currently is not known to exist at NTS.

The LASL 1954 fluorophotometric method for uranium in urine was based on the intense yellow-green fluorescence (the principal 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} g per 0.25 g of sodium fluoride, with a precision of ±10%. The tolerance for normal uranium in urine at LASL was 100 µg/L (McClelland 1955). In 1958, the same method was sensitive to concentrations of uranium from 10^{-10} to 5×10^{-11} g per 0.25 g of sodium fluoride, with a precision of ±10% (McClelland 1958).

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 (McClelland 1958).

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 ±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 (McClelland 1958).

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).

Methods for total uranium by fluorometry and alpha spectrometry and for ^{235}U by anion exchange, purification, electrodeposition, and alpha spectrometry were listed in REECo documents published from 1981 to 1983 (REECo 1968–1987a).

As described in Allen et al. (1993), ^{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-hour urine sample and a prompt lung count. Elemental uranium was determined by a fluorometer, and the MDA was listed as 5 µg/L for chemical analysis with a "method" MDA of 0.2 pCi/L. Attachment A, Table A-7 lists historical detection thresholds for various periods. Table 5-2 lists current MDAs for uranium isotopes.

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 ±10% at a 90% confidence level, counter background of 17 cpm, and a 60-minute counting period) (Author unknown undated a). 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.

Allen et al. (1993) 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 A, Table A-7 lists limits of detection for urine and fecal analysis. Table 5-2 lists the current MDA.

5.2.2.7 *In Vitro* Analysis for Thorium

The McClelland (1955) 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 dpm of ^{230}Th could be determined with this method. In 1993, the ^{230}Th fecal MDA was 0.01 pCi/g (Allen et al. 1993). Attachment A, Table A-7 lists limits of detection for urine and fecal analysis. Table 5-2 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 radium in the body [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 (McClelland 1958). Allen et al. (1993) 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. Attachment A, Table A-7 lists limits of detection for urine and fecal analysis. Table 5-2 lists the current MDA.

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.

The McClelland (1958) method for gross beta activity in urine was as follows. The nuclides of ^{90}Sr-^{90}Y, ^{140}Ba-La, ^{144}Ce-Pr, ^{89}Sr, and GFPs 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. McClelland (1958) states that if the gross beta (^{90}Sr-^{90}Y, ^{140}Ba-La) result is higher than 200 dpm/L, exposure should be suspected and investigated. The LASL MDA value for fission products was 50 to 100 dpm/L (ORAUT 2009).

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. When this occurred, the DR would find a note on the bioassay record indicating what fission products were identified during the follow-up analysis.

REECo (1961) stated that this procedure (i.e., coprecipitating fission products on alkaline earth phosphates from an alkaline solution and counting the sample package for gross beta activity) did not recover cesium, 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 considerable savings of time and money, according to REECo records.

GFP analysis was generally reserved for quarterly urine samples. From 1970 to 1979, the procedure was designed to provide a urine screening or monitoring method by reclaiming radionuclides of concern such as ^{89}Sr, ^{90}Sr-^{90}Y, ^{95}Zr-Nb, ^{140}Ba-La, and ^{141}Ce. Radionuclides not reclaimed included ^{137}Cs and the iodines. From 1981 to 1987, the quarterly urine analysis was a screening method for measuring gross beta concentrations of radionuclides (i.e., presence or absence of beta-emitting radionuclides) after the chemical removal of potassium. Throughout the period, samples were forwarded to the laboratory for processing and planchets containing GFP activity were submitted for beta measurements (REECo 1970–1987). On rare occasions, strontium was analyzed separately. In addition, the gamma energy peak for ^{137}Cs was analyzed in gamma spectroscopy for some samples on an occasional basis (DeMarre 2005). From 1982 to 1987, the detection limit for GFP (beta) in urine was 1×10^{-10} µCi/mL (REECo 1968–1987a).

Allen et al. (1993) states that 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). Attachment A, Table A-7 lists limits of detection for urine and fecal analysis.

Cesium-137 is a product of fallout from weapons testing. As with ^{90}Sr, background levels will generally result in small doses. Cesium sampling is performed quarterly. Dose Reconstructors will need to consider ^{137}Cs as occupational exposure in the absence of evidence in the worker's record indicating the ^{137}Cs was from another source. Cesium-137 is a radionuclide of concern at NTS because of the nature of the work performed and is part of the environmental source term. It is appropriate to address ^{137}Cs levels by evaluating the intake using the Integrated Modules for Bioassay Analysis (IMBA) computer program

DRs should select fresh or aged fission products based on what is most favorable to claimants for the cancer diagnosed. The bioassay records do not indicate fresh or aged. Attachment A, Table A-30 identifies common radionuclides that are a significant part of fission products up to 1 year old. DRs should check the allowable limits for fission products identified as a dose concern. They should review the records provided by DOE for additional data, such as access logs and RCT logbooks, and review test information to determine if they need to consider any fission products (Table A-30) unique to the test based on the cancer location. They might need to request additional records to determine aged or fresh fission products.

NOTE: If the worker had a WBC 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 WBC. Whole-body counting began at offsite locations in 1964 (Church 1967). Onsite capability started in 1967 (Teasdale 1985). See Section 5.3 for additional *in vivo* information and references.

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 based on the results of the initial pulse height analysis. Almost any gamma component reported 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} µCi/mL (REECo 1968–1987a).

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 1968 to 1982, gross gamma analyses were conducted for urine samples. Samples were counted in a 9-in. by 9-in. sodium-iodide (NaI) well crystal. If counting rates exceeded an established level, the samples were placed in one of the 512-channel analyzer systems and a gamma spectrum was obtained. Samples counted in the well crystal had a crude spectrum available from a multichannel analyzer that accumulated data while the gross count rate was measured. While analysts could identify some radionuclides directly from this spectrum, summation effects in the well crystal severely restricted resolution of the more complex spectra; these samples had to be analyzed with the 512-channel system. The gross gamma counters were calibrated with a ^{137}Cs standard. The equivalent branching ratio was used in the conversion factor (disintegrations per count) and this was applied to the counting data regardless of the nuclides present. Gross gamma data from the well crystal were discriminated to represent the integral count rate from 100 keV to 4 MeV (from 1968 to 1979). The high-energy discrimination was necessary to limit the increase in count rate above 4 MeV caused by high-energy cosmic particles. In 1981 and 1982, gross gamma data were discriminated to represent the integral count rate from 100 keV to 2 MeV. During this period, the pulse height analyzers used to obtain gamma spectra were calibrated as 1 keV/channel with a zero energy threshold. The energy range of 50 keV to 2 MeV included emissions of most of the nuclides

encountered. Beginning in 1983, a GeLi detector system or intrinsic detector system was used to perform gamma spectra analyses for urine samples. The pulse height analyzers used to obtain gamma spectra were calibrated at 1 keV/channel with a zero energy threshold. The energy range of 50 keV to 2 MeV included emissions of most of the nuclides encountered (REECo 1968–1987b).

REECo procedures stated that, if gamma emitters were involved, *in vivo* measurement procedures such as whole-body (or specific organ) counting were preferred (REECo ca. 1980s). For routine urinalysis, the gamma scan was used as a screening tool. There was no additional sample preparation. If the scan result was less than or equal to 5,000 cpm, a spectral analysis via gamma spectroscopy occurred. If the scan result was less than 5,000 cpm, the analysis reported a ^{137}Cs equivalent concentration. Each positive radioiodine result was 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 indicated that thyroid counting of the individual was to be considered. DRs might need to evaluate gamma results above the detection limit based on the radionuclide of concern identified with the work area or activities performed and should assign such results to the radionuclide providing the most dose to the internal organ associated with the cancer type. If the DR cannot determine the work area or activities performed, assume the radionuclide providing the most dose to the internal organ associated with the cancer type.

In 1993, all urine samples were counted by gamma spectroscopy to determine the concentration of gamma-emitting radionuclides in the energy range from 40 to 2,000 keV (Allen et al. 1993). 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. The specific MDA for ^{137}Cs in urine was listed as 100 pCi/L (Allen et al. 1993). Attachment A, Table A-7, lists limits of detection for urine and fecal analysis. Table 5-2 lists the current MDA value.

5.2.3 Sample Collection and 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. The collection start and stop dates were recorded. When the interval was less than 1 day, the collection start and stop times were recorded. If the interval was more than 1 day, total hours were recorded. The quantity collected was also recorded.

Hartzell (1962) listed the quantities of urine samples required for routine laboratory analysis. For activity, a single void of at least 25 to 50 mL was required. For GFP, strontium, uranium, and plutonium, a 24-hour sample of at least 250 mL was required. To collect the 24-hour sample, a kit of three bottles (1,000 mL total) was issued from Building CP-2 (Area 6).

REECo procedures discussed how urine samples were received and processed from 1970 to 1987 (REECo 1970–1987). Those and later site procedures contained specific instructions and minimum samples sizes for the following radionuclides (REECo 1969–1987; Allen et al. 1993; BN 1998).

- GFP: 250 mL (1969–1993)
- Plutonium-239 (urine)
    - 1 L (February 1969–June 1985)

  - 900 mL (July 1985–July 1987)
  - 1.5-L (1990–1991) optimum sample size
  - 2 L (1993)
- Pu-239 (fecal): 1 g ash (1993–1998); also analyzed for radium, uranium, and americium (1993)
- Americium: 2 L (1993)
- Uranium: 1.5 L (1993)
- Strontium-89/90: 900 mL (1969–1993)
- Tritium: 20 mL (1990–1991); 5 mL of distillate from a 30-mL sample (1993)
- Gross alpha/beta: 250 mL (1990–1991)
- Gamma (urine): 500 mL desired, 50-mL minimum volume (1985–1993)
- Gamma (fecal): 500 g (1993)

In 1975, the employee's last name, initials, and social security number were recorded with each bioassay sample submitted. By 1985, the person's full first name and middle initial were required. Due to security concerns, only the event or program code associated with the project was entered on the laboratory analysis request form; the actual drill hole or tunnel number was not recorded. This could make it more difficult for DRs to associate an employee with a specific test. By 1990, the sample receiving/control and tracking procedures focused more on chain-of-custody forms and computer data entry functions.

As discussed in Allen et al. (1993), 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-hour excretion. If a 24-hour sample had a volume of less than 500 mL, NTS did not analyze it because this volume is too small to represent a true 24-hour collection. In such a case, the worker was required to provide another sample. Provided the sample was collected properly, a total or simulated 24-hour urine sample result was used without further normalization. A proper 12-hour result was normalized by doubling the result. BN (1998) updated the discussion as follows:

> *For most radionuclides, urine samples representative of excretion over a 24-hour period should be collected for quantitative intake and dose assessment. The concentration of materials in urine varies throughout the day, with the first void of the morning generally being the most concentrated. If an aliquot from a 24-hour sample is used for analysis, it is important that the sample be thoroughly mixed before removing the aliquot. For hard-to-detect nuclides, such as Pu, the analysis of the entire 24-hour sample is generally necessary to achieve the desired detection levels.*

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).

Allen et al. (1993) stated that NTS ascribed a time interval to each collected urine sample. The volume of the sample was measured and the interval was calculated assuming excretion of a constant volume of urine per day (1.4 L/day is the conventional value). Samples frequently spanned nonintegral periods (e.g., from Day 1.1 to Day 1.7), which gives an interval of 0.6 days ending 1.7 days after the intake. In such cases, the excretion function could normally be extrapolated to the samples. The excretion function was adapted to the sample by reviewing the expected excretion for the interval from Day 1.1 to Day 1.7.

An alternative method used at NTS was to normalize bioassay results to integral 1-day values. For example, the result from Day 1.1 to Day 1.7 was adjusted to give the results from Day 1 to Day 2. This method enabled comparison of expected and empirical results. The disadvantage with this method was that the normalization procedure could have introduced additional error. However, for most cases at NTS, the exact time of intake was not well known and the normalization error was rather insignificant. Therefore, NTS used the normalization method because it permitted the graphical evaluation of data.

Because special bioassays were initiated in response to known or suspected intake conditions, the intake-to-bioassay period was assumed to be relatively short. At NTS, the urine sample was assumed to have the concentration expected at 24 hours after intake (i.e., the "instantaneous" urinary excretion function was used) (Allen et al. 1993).

5.2.4 Fecal Sample Collection and Analysis

NTS performed fecal sample analyses as a special bioassay, usually if there were other indications that internal contamination had occurred. Allen et al. (1993) indicates that 5-g ash samples were analyzed by alpha spectroscopy for radium, uranium, plutonium, and americium. BN (1998) clarified the collection and analysis volumes: The MDA assumes that a 24-hour sample was collected, half was ashed (resulting in 5 g of material), and 1 g of the material was measured. 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.

Allen et al. (1993) 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 (e.g., a week), and the data should be evaluated with an accumulated feces excretion model. This is particularly important in the first week after 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.

Fecal samples might have been collected after an incident (Allen et al. 1993). Because fecal excretion patterns vary considerably, it was unreasonable to specify a minimal acceptable mass on a single sample, but 100-g wet mass was a good target value. Fecal samples were promptly screened for radioactivity and additional samples collected if radioactivity was detected. Incidents with significant potential for an intake warranted the collection of the first 3 to 7 days of fecal excretion. Because special bioassays were initiated in response to known or suspected intake conditions, intakes to bioassay periods were assumed to be relatively short. At NTS, the feces sample was assumed to consist of the material excreted in the first 24 hours after the incident (Allen et al. 1993).

5.3 *IN VIVO* MDAS, COUNTING METHODS, AND REPORTING PRACTICES

NTS had an *in vivo* bioassay program that included whole-body, thyroid, lung, and wound counting. 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.

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). This TBD assumes 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 employees of NTS contractors have been retrieved from other locations for inclusion in the record archives located in Las Vegas, NV. Other reports and memoranda indicated that before 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 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.

In 1967, a shadow shield-type counter was installed by Pan Am at the Nuclear Rocket Development Station as part of the Nuclear Space Propulsion Program (Teasdale 1985) for performing WBCs. This counter was transferred to NTS REECo dosimetry in 1974. Beginning in 1975, routine counting was performed for drillers, miners, and radiation monitoring personnel. 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 A, Section A.2.1 and Table A-9 for whole-body counting MDAs.

5.3.1 Whole-Body Counting

Helgeson Nuclear Services (HNS) performed the first WBCs at NTS. It used a shadow shield that was transported in a semitrailer (HNS 1967). The shadow shield employed an 8- by 4-in. 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). HNS estimated the sensitivity at two confidence levels: 50% and 99%. Table 5-3 summarizes these activities.

Table 5-3. MDAs for the HNS shadow shield.

		Photons		MDA–nCi	
Isotope	Background	Energy (keV)	Intensity, I_γ photons/decay	50%	99%
Cs-137	125	662	0.851	1.0	3.0
Zr-95	95	724 757	0.81 1.00	0.7	2.1
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 1,332	1.00 1.00	0.4	1.3
K-40*	71	1,461	0.107	7.1 (8.3 gK)	21.4 (24.9 gK)

Source: HNS (1967).
* 1 gK = 0.000118 g of 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 nCi × I_γ. For nuclides not listed in Table 5-3 that have primary gamma energies ≥100keV, the MDA can be determined for the primary photon with an intensity, I_γ, by 3 nCi/I_γ. Table A-9 lists other radionuclides that DRs might need to address and includes MDA estimates that were calculated based on this formula.

There were more than 300 WBCs performed before 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-in. by 3-in. NaI(Tl) crystal on the first visit and an 8-in. by 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.

	Body burden (µCi)			
Nuclide	**≤0.02**	**≤0.05**	**≤0.1**	**≤0.5**
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: Church (1967).

In January 1967, Pan Am acquired and installed the shadow shield WBC. The detection system consisted of a 3-in. by 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 (Church 1967). Table 5-5 lists results of the initial WBCs performed by this system.

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

	MPBB and organ	Body burden (µCi)			
Nuclide	**of concern[a]**	**≤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: Church (1967).

a. The MPBB and organ appear in handwritten notes on the original table. Documentation for the quantities and organs of concern has not been found.

b. The handwritten notes indicate 20 µCi for Ru-103 and 3 µCi for Ru-106.

The January 1977 through January 1981 REECo standard procedures for chemical and radiological analysis describe the WBC program (REECo 1969–1987). During the preliminary calibration of WBCs, the phantom consisted of polyethylene blocks stacked in a 3 by 5 by 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/cm^3 (REECo 1969–1987). 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: REECo (1969–1987).

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

$$Efficiency_{(\text{counts/disintegration})} = 0.0089e^{-0.000175E} \quad (5\text{-}4)$$

where *E* is energy in keV (REECo 1969–1987).

Sensitivity was defined as activity that is detectable with a relative 2-sigma error of ±20%. Background counts were 1,000 seconds. Table 5-7 lists the sensitivities for potassium and cesium at a background count of 45 cps.

Table 5-7. WBC 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

Source: REECo (1969–1987).
a. Potassium activity in standard man is about 0.12 µCi.
b. 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 WBC printout. Abnormal results, defined as those in excess of 200 g potassium, standard man equivalent, were reviewed by a senior health physicist to determine the need for further action (REECo 1978, p. 22).

REECo standard procedures for personnel dosimetry and records indicate that for 1978 and 1979, whole-body counting used a 20.3-in.-diameter by 10.1-cm-thick NaI crystal housed in a shallow shield (REECo 1978, p. 22; REECo 1979, p. 25). (The detection energy range was 100 to 2,500 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 ^{40}K channels appeared to be excessive, a recount was performed after street clothes were exchanged for a paper suit. In addition, the background of the bed was rechecked. There is no available documentation for 1980; it is assumed the 1979 equipment and procedures were used until 1981.

By 1981, a 29.2-cm-diameter by 10.1-cm-thick NaI(Tl) detector connected to a Canberra Series 30 multichannel analyzer was used (REECo 1981, pp. 7, 20). The use of paper suits 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. Radon daughter products in the air were reduced by a high-energy particulate air (HEPA) filtration system (Teasdale 1985).

From 1983 to 1987, two types of detection equipment were identified (REECo 1983, pp. 19, 21; REECo 1987, pp. 16–17):

1. A 29.2-cm by 10-cm-thick NaI(Tl) crystal used to scan for radionuclides with gamma energies in excess of 100 keV. The WBC was calibrated using a Bomar water phantom constructed of polyethylene that weighs 70 kg when filled with water. Four monoenergetic gamma-ray emitters were used as dose material for the solutions. An energy response curve was generated for the results using energies of ^{51}Cr, ^{54}Mn, ^{65}Zn, ^{137}Cs, and the high-energy peak of ^{88}Y.

2. Phoswich counting systems were used to detect X-rays and gamma rays of less than 100 keV. These detectors consisted of a 12.7-cm-diameter by 5-cm-thick CsI(Na) crystal that was 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 that contained ^{241}Am, $^{238/239}$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.

Employees were issued paper suits and assumed a supine position on the counting table within the vault. The detector moved from one end of the bed to the other over a 20-minute period (REECo 1983, pp. 19–20; REECo 1987, p. 16). There is no available documentation from 1988 to 1993; it is assumed the 1987 equipment and procedures were in use until 1993.

The system used in 1993 was calibrated for whole-body counting and used for whole-body, thyroid, and wound counting (Allen et al. 1993). 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 yielded an arc with the detector of 50 cm, providing a full view of the body trunk in relation to the detector.

WBCs were performed (Teasdale 1985, Allen et al. 1993):

- For new employees who were likely to be included in the routine bioassay program
- For a current employee who changed to a job classification that required a routine bioassay program
- Annually for employees who were currently on the routine WBC and bioassay program
- For terminating employees

- For employees who had a suspected intake of radioactive material, particularly gamma-emitting fission and activation products.

As identified in Allen et al. (1993), the MDA was calculated as a function of the gross spectral counts and the full-width, half-maximum (FWHM) energy and was consistent with American National Standards Institute standard N13.30 (HPS 1996). The MDA (assuming a 5% probability of either a type 1 or type 2 error) is defined as:

$$MDA = \frac{\left[C_1 + C_2 \times \left(\Sigma_{C-W}^{C+W} Y_i \right)^{0.5} \right]}{T_1 \times E \times Y \times k} \quad (5\text{-}5)$$

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

The suggested priority for DRs with regard to MDAs and other detection thresholds is to use (1) the limit on dose report if available, (2) Attachment A, Table A-9 values, or (3) an appropriate published value from another DOE site or an applicable value referenced in the literature (NIOSH 2002).

5.3.1.1 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. National Council on Radiation Protection and Measurements (NCRP) Report 94 (1988) provides mean body burdens of ^{137}Cs for the United States for the years most likely to produce interference with occupational WBC results. Because of the nature of the work and potential exposure at NTS, unless there is evidence that the ^{137}Cs result is not associated with employment at NTS, the result should be evaluated as occupational with no adjustment for the values.

5.3.2 Chest Counting

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

The first reference to lung or chest counting was found in a REECo standard procedure for personnel dosimetry and records dated July 1983 (REECo 1983, pp. 19–20). The procedure stated that for lung counts, the subject assumed a supine position, crystals were placed over the anterior lung region, and the counting time was 40 minutes. Lawrence Livermore Laboratory software to process lung counting data is mentioned in a 1981 REECo standard procedure for laboratory instruments and equipment (REECo 1968–1987a, p. 67).

Allen et al. (1993) identified lung counts with nonroutine bioassay for the suspected intake of thorium from the use of special nuclear materials, uranium, or any of the TRU elements, 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 mm^2 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 while maintaining proximity to the lung and bronchial region, thereby optimizing the number of photons measured.

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). Attachment A, Table A-10 lists the MDAs for chest lung counts.

The method used for lung counting at NTS in the early 1990s was based on published biometric relations for CWT measurement relations to weight and height (Allen et al. 1993). The chosen relation was:

$$CWT = 1.973(W/H) - 2.0038 \quad (5\text{-}6)$$

where:

CWT = chest wall thickness (cm)
W = weight (lb)
H = height (in.)

The Livermore torso phantom was fabricated to provide calibration information for TRU 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 of 100% muscle, 50% muscle to 50% adipose, and 13% muscle to 87% adipose. The phantom covers a CWT range of about 15 cm without chest plates to about 40 cm. with the thickest overlays. There is some set-to-set variation.

Direct measurements of plutonium in the body, typically via lung counts, are capable of detecting only very large intakes or extended chronic exposures. The site no longer performs lung counting (McMahan and Ogurek 2003).

5.3.3 Thyroid Counting

During weapons testing at NTS, exposure to radioiodines was a major concern for the first 100 hours after detonation. Thyroid counts were a concern for workers involved in drill-backs or reentry/mine-backs in this 100-hour timeframe. DRs can determine potential exposure within 100 hours of a test by reviewing employee access records in the files provided by DOE for the claim. 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 aboveground test or individuals on the ground after 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. The DR might find this referred to in employee records as a “prompt count.”

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 1968–1987b). 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 (Allen et al. 1993). The following paragraphs contain information on the ratios of the iodines in the period immediately after a test. DRs should use this for individuals directly involved in incidents such as Baneberry or reentry activities as indicated in the records provided by DOE (i.e. indication of potential for exposure based on bioassay or decontamination results for the worker).

5.3.3.1 Iodine

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

After a nuclear detonation, several iodine radionuclides are produced; those with mass numbers 131 through 135 are of significance from the standpoint of exposure to personnel. Table 5-8, derived from data cited in Kathren (1964), provides the relative activity normalized to that of ^{131}I, at times ranging from 1 hour to 1 month after fission of ^{235}U. Hence, if the time of detonation and the activity of one iodine radionuclide are known, the activity of the other iodine radionuclides can be determined. DRs should consult access records in the information provided by DOE to determine likely exposure time after the test.

Table 5-8. Relative abundance of radioiodines.

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 iodine radionuclides at early times after fission is a complex function of the ingrowth and decay of the radionuclides and their fission chain precursors, dependent to some extent on whether produced by thermal or fast fission and by fission of what nuclide. At times ranging to about 4 hours after fission, ^{134}I is by far the predominant radioiodine activity, accounting for almost all activity from the five radionuclides. The peak activity from this radionuclide occurs 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 minutes) becomes predominant. The peak activity from this nuclide is 26 MCi per kiloton of explosive yield, compared to 0.11 MCi from ^{131}I, which occurs about 5 hours after detonation (Holland 1964). A similar but less rapid effect is apparent for the other iodine radionuclides in relation to ^{131}I. At times less than about 1 hour after detonation, the relative activity from the higher iodines is much larger than at later times, primarily as a result of the rapid ingrowth of ^{131}I, whose absolute activity peaks about 5 hours 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 iodine radionuclides of significance.

A calculation based on the maximum iodine concentration for any event (i.e. Baneberry venting December 1970) provides an indication of the potential dose during a release. A concentration of 1.85×10^{-12} µCi/cm^3 was measured at Camp 12 on December 24, 1970, resulting in a calculated value of 3.1×10^{-12} µCi/cm^3 on December 18, 1970, the day of the venting. Assuming a breathing rate of 20 L/min and an exposure of 2 hours during the evacuation of the site, an intake of 7.44×10^{-6} µCi is reasonable and results in a dose of less than 1 mrem to the thyroid.

5.3.4 Wound Counting

Wound counting was performed (Allen et al. 1993) 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 germanium 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 WBC 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. McMahan and Ogurek (2003) 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

NTS air sampling records are workplace rather than intake monitoring records, and NTS does not maintain them in the same collection as bioassay records. Because air sampling records are associated with each facility, they are 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.

Radiological safety documentation describes historical air sampling. REECo prepared onsite radiation safety reports for each operation starting in 1957 (e.g., REECo 1957, 1958); these reports outline the support REECo provided to the testing groups [LANL, LLNL, and U.S. Department of Defense (DOD)/Defense Nuclear Agency (DNA)/Sandia National Laboratories (SNL)]. Additional documentation back to 1951 prepared by the laboratories or the Atomic Energy Commission is in a narrative form. The REECo reports briefly outline radiological support functions and describe specific support for each test until the end of the testing era in 1992. Topics discussed in the reports include:

- Test name, sponsor, location, type, date, and time.
- Number and location of remote telemetry stations and a summary of results.
- Details on initial reentry surveys, reentry and mine-back (for tunnel test), and post-test drilling. In the early 1980s, brief activity and operations logs were included.

- A final summary statement (e.g., No whole-body external or internal exposures exceeded the established control limits and no toxic gases or explosive mixtures were detected during the post-test drilling period.)

- A list of the radiation and toxic gas detection instruments in use.

Allen et al. (1993) 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 1988).

The principal workplace air monitoring device used at NTS was the retrospective air sampler (RAS) (Author unknown undated a). Retrospective air samplers (RASs) were used at the decontamination facility (Decon Pad) during post-test drilling, tunnel reentry, and other special operations. A typical RAS consisted of a portable or permanently installed air sampling system such as a vacuum pump with an attached meter to measure the sample flow rate or sample volumes. The ambient air sample was normally pulled through a 4-in. or 7-in. cellulose filter and a 4-in. activated charcoal cartridge at the rate of about 1 to 3 ft^3/min. The volume of air per unit time was dependent on the size of the pump, the number of samples collected by that pump at one time, and how loaded the samples were with material that resisted flow. Depending on the location of the sampler and the work being monitored, the filters were replaced at a frequency from every shift to every week. RASs provided a history of work environment airborne contamination levels to ensure that workers were not exposed to unexpected levels of airborne radioactive material.

Air samples collected from most work areas were counted at the REECo laboratory in Mercury, Nevada (Allen et al. 1993). Air samples from LANL work sites were counted by personnel at the LANL Health and Safety Laboratory in Mercury. Results from both laboratories were reported in microcuries per milliliter. Air samples from the Decon Pad (Area 6) were counted at the facility in a thin-window detector scaler system for beta- and alpha-emitting radioactivity; results were reported in microcuries per milliliter or in DAC fractions. Charcoal canisters were not used with Decon Pad air samples.

Another sampler used at NTS was the continuous air monitor (CAM), which consisted of a vacuum source, collection media, and a detector for alpha, beta, or beta-gamma radiation. When radiation was detected in excess of preset limits, audible and visual alarms were triggered in the affected areas to alert personnel of the release. CAMs were used at the Treatability Test Facility and on drill rig floors during LANL drill-backs. CAMs detected radioiodines, tritium, plutonium, or other radionuclides of concern (Allen et al. 1993).

5.4.1 Drilling Operations

A 1959 memorandum titled "Operational Guides for Above-Ground Drill Sites into Ground Zero Areas" (Wilcox 1959) stated that if surveys indicated significant beta-gamma concentrations, a 1-hour Staplex (8-in. by 10-in. glass-fiber filter) sample and nasal swabs were collected from each person possibly exposed. In addition, REECo published *Air Sampling for Radioactive Particulate Material*, which included the following air sampling information (Geiger 1959):

> *Information relative to hazards associated with internally deposited radioactive materials has been increasing continuously since 1953 when NBS Handbook 52 was published. In June 1959, NBS Handbook 69, containing the new recommendations of*

the NCRP, was published. Handbook 69 supersedes Handbook 52 as the guide for evaluating and controlling internal radiation hazards at NTS.

The recommended maximum permissible concentration in air for an unknown alpha emitter, $(MPCU)_a$, for continuous occupational exposure is 4 E-07 uCi/m^3, and for a 40-hour work week the recommended $(MPC)_a$ (excluding ^{210}Pb, ^{227}Ac, ^{228}Ra, and ^{241}Pu) for continuous exposure is 1 E-04 uCi/m^3. For a 40-hour work week this has been increased by a factor of three to 3 E-04 uCi/m^3.

For control purposes, the (REECo) Rad-Safe Division will determine concentration of airborne alpha emitters down to 10% of (MPC) for ^{239}Pu, i.e., down to 2 E-07 uCi/m^3, and concentration of beta emitters down to 10% of the $(MPCU)_a$ value, i.e., down to 3 E-05 uCi/m^3.

In order to measure $(MPC)_a$ concentrations within ± 10%, 90% of the time, a sample volume of 44 m^3 of air must be pulled through the filter. For the large filter (Staplex) samples the total volume should be 66 m^3. With these volumes collected on a glass fiber filter, the counting errors with the large area proportional detector for various concentrations of ^{239}Pu would be as follows:

Air concentration uCi/m^3	***Relationship to $(MPC)_a$ for Pu-239***	***Counting error at 90% confidence level***
2 E-06	*1*	*± 10%*
1 E-06	*0.5*	*± 17%*
5 E-07	*0.25*	*± 22%*
2 E-07	*0.10*	*± 54%*

Some typical sampling times for various samples are as follows:

Sampling equipment	***Flow rate***	***Typical sampling time***
Staplex 8" × 10" filter	*40 cfm*	*1 hr*
Staplex 4"-diameter filter	*20 cfm*	*1 hr 20 min*
Filter Queen	*4 cfm*	*6 hrs 30 min*
Gelman "Caddy"	*3 cfm*	*8 hrs 40 min*
Gelman Diaphragm	*0.37 m^3/hr*	*5 days*

The above sampling times, which were calculated based on ^{239}Pu, will provide plenty of sample for assay of long half-life beta emitters.

Continuous air sampling is preferred over "grab" type sampling. If the operation is likely to be such that air sampling is necessary only a few hours, the Staplex machine should be used. If the need for air sampling is for one day or more, either a Filter Queen or "Caddy" air sampler should be used; however, if neither of these air samplers is available, the Staplex may be used as a "second choice." If the history of the work location indicates that respiratory protection will probably be unnecessary for a 5-day period or more, the low volume Gelman Diaphragm pump is the preferred choice.

Radon and thoron may occasionally be a problem in tunnel areas. The most expeditious manner for ascertaining whether or not it is a problem is as follows: Collect a 5-minute air sample with a Staplex 8" × 10" filter using an air sampler pulling approximately 35 cfm. Immediately after the completion of the 5-minute sampling time, monitor the surface of the filter with an Eberline PAC-G survey meter. A reading of 25,000 cpm on a PAC-G that has been calibrated for 2" geometry corresponds with the

$(MPC)_a$ for a 40-hour work week. Anything above this will normally require respiratory protection. Sometimes the monitor needs to know if the alpha activity detected on a filter is radon and thoron or long half-life emitters without waiting for decay. A rough indication may be obtained as follows: Place the filter in a Glassine envelope. Compare the reading obtained with PAC-G with the filter in the envelope with that obtained with the filter outside the envelope. Radon and thoron alpha will be reduced to approximately 30%. If less than 30% of the alpha particles penetrate one thickness of the envelope, long half-life alpha emitters may be present in significant concentration. If no activity is detected through the envelope, the majority of the activity is long half-life alpha emitters.

If the long half-life alpha emitter is known to be ^{235}U instead of ^{239}Pu, more liberal $(MPC)_a$ values can be used and correspondingly shorter air sampling times may be adequate. For example, the $(MPC)_a$ for a 40-hour work week recommended for ^{235}U is 1 E-04 uCi/m^3 as compared with 2 E-06 uCi/m^3 for ^{239}Pu. In like manner, the $(MPC)_a$ for short half-life fission products may be more liberal. The $(MPCU)_a$ for an unknown beta emitter is 3 E-03 uCi/m^3 for a 40-hour work week if none of the following beta emitters are present in significant concentrations when compared with the $(MPC)_a$ values: ^{90}Sr, ^{129}I, ^{210}Pb, ^{227}Ac, ^{228}Ra, ^{230}Pa, ^{241}Pu, and ^{249}Bk. Since ^{90}Sr is the only fission product in this group of beta emitters the more liberal $(MPCU)_a$ value is more appropriate in the case of a normal mixture of short half-life fission products.

REECo (1976) noted differences between LANL and LLNL in performing drill-back operations conducted in 1966 and 1967. LANL used a pressurized recirculation containment system, whereas LLNL used a vent line system designed to trap radioactive effluent released from the drill casing. In the vent line system, air from around the drill casing was pulled through the vent line into a large container of activated charcoal by a positive displacement pump. Radioactive effluent, especially radioiodine, was absorbed on the charcoal and allowed to decay under controlled conditions. ORAUT (2008) contains additional information about post-test drilling.

An array of four air-sampling heads, one on each corner of the drill rig platform, provided continuous monitoring of the air on the platform. This array was activated at the start of drilling and operated continuously until closure of an abandonment valve. The results of this sampling gave an indication of the exposure, if any, experienced by workers on the platform during the drilling operation (Johnson et al. 1966). Documentation in 1963 first noted this type of air sampling on the drill platform, which continued through the end of the testing period in 1992.

In the appendix (McClendon and Eubank 1964) titled “A Typical Detailed Safety Support Plan for Underground Nuclear Tests,” REECo provided air sampling procedures on the drill platform. This report covered Operation Storax, which occurred in 1962 and 1963. The appendix stated that a low-volume air sampler equipped with a Whatman-41 prefilter and a carbon cartridge was positioned at breathing level on each drill platform at the beginning of the drilling operation. The prefilters were changed when they measured 300 mR/hr or, in any case, at the end of 8 hours. In addition, a high-volume air sampler (similarly equipped and positioned) was used on each drill platform for 5 minutes of each half-hour. Air sampling started at the beginning of the drilling operation and continued until the drill hole was capped. All air sample prefilters were checked periodically for alpha radiation. During the first 30 hours after the event, the charcoal cartridge and the prefilter from each air sampler were screened individually on a gross gamma counting system to determine if any samples contained significant activity. The detection limit of the system (a 3-in. by 3-in. thallium-coated NaI crystal) was approximately 500 pCi (1-minute count). When gross gamma screening indicated significant activity, sufficient multichannel and gross-gamma decay data were collected to provide decay curves and positive identification of significant radionuclides, isotopic ratios, etc. Sample results were reported as microcuries per cubic meter at mid-sampling time for each nuclide detected (other than radioxenons).

5.4.2 Tunnel Operations

In 1961, REECo (1961) published specific air sampling guidelines for B and E Tunnel operations as follows:

- B Tunnel: A 5-minute air sample was collected with a Staplex (8" × 10" glass fiber filter) approximately one, three, and five hours after the beginning of the shift. The sample was measured for alpha activity immediately and one hour after collection (radon-thoron check). These samples were not transmitted to the lab.

  A Gelman diaphragm pump air sampler near the work areas was checked for operation each shift and the filter was exchanged each Friday.

- E Tunnel: A 5-minute air sample was collected with a Staplex (8" × 10" glass fiber filter) each shift. The sample was measured for alpha activity immediately and one hour after collection (radon- thoron check). These samples were not transmitted to the lab.

  The Filter Queen sampler was checked for operation each shift and the filter was exchanged each Friday.

In both tunnels, if significant beta-gamma concentrations were indicated by portable instrument survey or radon-thoron checks, a 1-hour Staplex (8-in. by 10-in. glass-fiber filters) sample was obtained and nasal swabs were collected from each person possibly exposed. In 1962, B and E Tunnel-specific air sampling guidelines were revised such that 30-minute air samples were collected rather than 5-minute samples. In addition, after the 1-hour alpha check, the samples were transmitted to the Area 12 Change House counting facility (Wilcox 1962).

Characterization of the tunnel environments occurred on a routine basis. Reports titled *Current Tunnel Survey, Radiological Conditions and Reentry Requirements* (Bicker 1975) are available for the period from 1969 to 1980. The reports provide information for each tunnel by individual drift locations. They include exposure rates, note if airborne radiation is present (yes/no), and specify personnel protection equipment (PPE) for workers and visitors. The cover letters for these reports state that air samples were collected at representative vents before reentry (going in a tunnel that has been inactive for a period, not a post-test reentry).

There are detailed tunnel air sampling reports covering the period from 1991 to 1993. These reports give ambient air sampling results and, if a test occurred during the reporting period, air sampling results obtained during post-test reentry including isotopics. Tunnel air sampling is described in Lyons (1992), which stated, "Of the NTS operations air sampling programs in the tunnel complexes, the initial mining and event reentry and recovery operations represent the only real airborne radioactive inhalation potentials to personnel." Weekly Drierite samples for tritium analysis were collected in the active tunnel complexes to document any changes in normal background levels or reentry drifts as they were advanced toward ground zero areas. Underground water sources were considered primary transporters of tritium from old event areas.

Fission product beta/gamma emitters are the radionuclides of greatest interest for NTS operations (Lyons 1992). An air sample was collected in any work area where the potential for airborne contamination might have existed. These areas were identified using past air sample results from similar operations or from the type of work performed in the facility. The REECo laboratory in Mercury analyzed most air samples collected on a routine basis using gamma spectra analysis to determine beta/gamma radionuclide concentrations. The laboratory also analyzed Drierite samples. Gross alpha and beta/gamma concentrations were calculated at facilities where routine detailed radionuclide analysis was not required. It was REECo policy to perform air sampling in active work areas

containing contaminated debris or in areas where mining and drilling operations had the potential to encounter radioactive materials in fissures, cavities, or water. The sample program continued until the job was complete. The gamma spectra results for routine air samples were reviewed by REECo health physicists. Results indicating the presence of fission products were investigated and engineering and administrative controls were evaluated to prevent generation of airborne material during subsequent operations.

Routine air samples consisted of a Whatman 41 cellulose fiber prefilter and a Mine Safety Appliances Company (MSA)-activated charcoal cartridge housed in a 4-in.-diameter sample head. Several types of calibrated pumps with airflow calibrated liquid petroleum gas meters determined total air volume samples. Air sample data were reported in microcuries per milliliter. Tritium water vapor samples were drawn from ambient air in areas where elevated tritium levels were suspected or to collect background information for future reference. The collection system consisted of a 10/20 mesh of Drierite housed in a 10.75-in. by 1.75-in.-diameter Lucite column. Samples were drawn through a 0.188-in. aperture at flow rates of 3 to 4 L/min as measured with a calibrated rotometer. Collection times ranged from 3 to 6 hours to attain an 80% saturation level. These data were also reported in microcuries per milliliter (Lyons 1992).

For iodine air sampling, bioassay records indicate if the samples were collected with a charcoal canister or a filter (Arent and Smith 2004).

5.4.3 Examples of Air Sampling Results

The Report of the Test Manager on 56 Project – NTS details air sampling activities during support of plutonium dispersal experiments in Area 11 (AEC 1956). Personnel from Sandia, LANL, and EG&G were involved in Project 56 field operations. REECo established air sampling stations for routine collection of airborne material on a filter medium at each location where personnel worked above specified contamination levels. Stations were located at the camera trailer, the field decontamination station, the field control point, the J-12 count shack on Yucca Lake, and the Rad-Safety building (Area 6, CP-2). A tolerance level of 8.78 dpm/m^3 was used for air concentrations at these locations. At certain times, air samples were collected from the rooms of the field decontamination station and around ground zero sites. REECo maintained an access log for personnel who entered the contamination area (CA). The sampling found air concentrations as high as 72,000 dpm/m^3 in rooms where personnel decontamination occurred. Sample recovery air sampling results as high as 14,600 dpm/m^3 were measured and surface contamination levels greater than 2×10^6 cpm were recorded at ground zero. Maximum personnel contamination levels (outer coveralls) were measured above 20,000 cpm alpha.

Operation Plumbbob On-Site Rad-Safety Report (REECo 1957) stated that 1945 air samples were collected for the test series conducted from April 24 to October 7, 1957, which included a series of balloon, tower, surface, rocket, shaft, and tunnel tests. REECo collected air samples for long-lived alpha activity at designated locations (e.g., Warehouse 6, CP-2, Area 2, Gate 385, Area 13, Well 5B). In addition, REECo provided air sampling support to Project 57, which was a surface plutonium dispersal test (a safety test with zero yield conducted on April 24, 1957 in Area 13). In addition to the environmental fallout trays, REECo assisted Sandia Corporation in collecting samples from impactor stations on the day of the test. Continuous air samples were taken around the Decontamination Building and in the ground zero area and analyzed in the Rad-Safe Laboratory. Daily radiological surveys were made inside the Decontamination Building and in the surrounding “clean” areas using alpha survey instruments, filter paper swipes, and air samplers. An initial post-detonation survey of the ground zero area indicated extensive alpha contamination, but no significant beta-gamma activity. Table 5-9 lists results of the air samples obtained on April 30, 1957, at locations 500 and 1,000 ft from ground zero with a 15-knot wind blowing from the southeast.

Table 5-9. Air sample results from 1957 surface plutonium dispersal test.

Location	Long-lived alpha (dpm/m^3)	
	500 ft	1,000 ft
North	498	686
East	200	12.7
South	217	268
West	304	183

REECo collected air samples on the north side of the Decontamination Building from April 24 to April 30, 1957. Decontamination Building air samples had an activity of less than 32 dpm/m^3. Daily surveys and swipes in the Decontamination Building indicated no significant alpha contamination.

REECo (1957) contains a description of the decontamination of an Area 9 balloon test site and specific air sampling results from decontamination activities that began on September 3, 1957. Radioactive fallout from the Area 2 Smokey test (August 31, 1957) contaminated the Area 9 balloon site, where preparations were in progress for a subsequent test. Decontamination in the immediate vicinity of the balloon site ground zero was necessary to protect the workers. REECo developed a decontamination plan including radiological controls that included the use of heavy equipment to remove soil and replace it with clean fill. Personnel included two equipment operators, six truck drivers, seven laborers, two foremen, three radiation monitors, and a superintendent. Air samples were collected before, during, and after each phase of the decontamination to determine airborne activity. The highest concentrations of airborne contamination existed during loader operations (5.3×10^4 dpm/m^3 beta-gamma at a sampling station approximately 100 ft downwind; the activity was extrapolated to collection time). Table 5-10 lists additional air sampling data.

Table 5-10. Additional air sampling data, Area 9 balloon test site.

Condition of area	Beta plus gamma at collection time (dpm/m^3)
Before start of decontamination, no one in area	1.64E+04
During scraper operation before wetting down	1.33E+04
During scraper operation after wetting down	0.92E+04
During loader operation, very dusty	5.3E+04[a]
After decontamination, no one in area	1.32E+04

a. Respirators required above 2.2E+04 dpm/m^3 at NTS.

Operation Hardtack Phase II On-Site Rad-Safe Report (REECo 1958) listed air sampling reporting levels for airborne particulates as 2×10^{-6} µCi/m^3 (alpha) and 1×10^{-3} µCi/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.

Radiological Safety for Underground Nuclear Explosions (Wilcox and Coogan 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. The REECo report stated that nonradioactive material was sufficient in the "dump" to prevent accumulation of significant radiation levels. Debris beta-gamma ratios during the period (1 to 7 months after the detonation) were variable from 2:1 to 10:1.

Wilcox and Coogan (1960) stated that the liberal application of water and natural water seepage reduced airborne radioactivity in the mining work area. Where natural seepage was not enough to keep areas moist, airborne radioactive particulate concentrations were occasionally as high as 4.5×10^{-2} µCi/m^3 beta, but airborne concentrations never exceeded 1×10^{-3} µCi/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 the presence of high concentrations of airborne radioactive

material and that accompanying high levels of radon-thoron daughter products made measurements difficult.

According to Wilcox and Coogan (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 dry, and a large vacuum system removed cored material. In dry drilling operations, filtering the vacuum system exhaust reduced air activity, 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. The handling of "dried" core samples often resulted in significant airborne contaminants. Air hoods and respirators were used to prevent worker exposure to internal radiation.

Re-entry Problems Associated with Radiation from Underground Nuclear Detonations (Brown 1963) discussed radiological conditions for atmospheric versus underground testing. This document stated that inhalation exposure problems characterized the entire post-test work environment, 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 document reported the ejection of coarse (greater than 12 µm) and fine (less than 1 µm) particulate to several thousand feet.

Table 5-11 lists examples of historical limits for air samples for years from 1950 to 1987 (REECo 1985; Author unknown undated c; Geiger and Whittaker 1961). Threshold levels include tolerance, MPC, reporting, exposure, detection, alert, and sensitivity levels. Attachment A, Section A.4.4 discusses respiratory protection and contamination control procedures.

Table 5-11. Historic air sampling limits (Author unknown undated a).

Year	Limit	Note
1957	MPC in air above background of mixture of unknown radionuclides was 1 E-09 µCi/mL (beta and gamma) and 5 E-12 µCi/mL (alpha)	When radionuclides were known, MPC was 3 times above values based on exposures of 40-hour week
1958	Air tolerance for uranium was 50 µg/m^3 or approximately 70 dpm/m^3	LANL action level
1958	Laboratory reporting levels for airborne radionuclides were: 2 E-06 µCi/m^3 (alpha) 1 E-03 µCi/m^3 (beta)	
1959	MPC in air at NTS above background were: 2 E-12 µCi/mL (alpha) 3 E-9 µCi/mL (beta and 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 <0.00000001E-08 µCi/m^3	
1964	Air sample alert level was 1 E-05 µCi/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 µCi/cm^3 for radon. Thoron levels were negligible.
1965	MPC for Pu-239 (in air) was 2 E-12 µCi/cm^3 for 40-hour week	

Year	Limit	Note
1966–1967	Alert level in air: 1 E-14 µCi/cm^3 (alpha) 1 E-11 µCi/cm^3 (beta)	Alert levels based on MPC of unknown radionuclides in air over 168 hours were compared with sample activity results.
1968	Kr-85 could be measured down to 2 E-07 µCi/cm^3 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	

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 favorable to claimants to assume the result is real.

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, the assumption provided in the *Internal Dose Reconstruction Implementation Guideline* (NIOSH 2002) should be used for results near or at the reporting levels when no uncertainty is reported in the DOE claim file; 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)]. The reported value should be used if actual standard deviations or other indications of error are reported with a bioassay measurement result.

5.5.3 Less-Than Values

At NTS, the term *less-than* refers to data that are reported as less than some reporting level (Allen et al. 1993). For example, a plutonium urine bioassay might be reported as <1 E-12 (less than 1×10^{-12})

µCi/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 5×10^{-4} µCi/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.

5.6 DOSE RECONSTRUCTION RECOMMENDATIONS

5.6.1 Evaluation of Potential Internal Worker Exposure

NTS was unique in the DOE weapons complex in that no production processes occurred at this site. Chronic exposure would for most workers be based on ambient levels of radiation. Devices were assembled from materials produced elsewhere and field tested at NTS. The source term was specific to each testing category (atmospheric, tunnel, safety, etc.) and in some cases quantities of specific radionuclides and ratios remain classified. Residual soil concentrations at various NTS locations are available in the environmental section of this TBD (ORAUT 2010a).

As indicated in this TBD, during atmospheric testing the emphasis was on measurement of external radiation. Internal monitoring was based on the worker's assigned responsibilities and/or the worker being identified as potentially contaminated. Therefore, only a worker who was suspected of contamination would have received the initial screening provided by nasal swab. Nasal swabs were taken on a regular basis, particularly for anyone involved in reentries, decontamination activities, or other activities requiring respiratory protection. If a nasal swab result indicated contamination, the individual received bioassay (usually urinalysis). For most cases, the DR will review the worker's duties and job history and will base their assignment of internal dose on one or a combination of the following:

- The bioassay results in the worker's DOE records (Attachment A),
- Environmental measurements (ORAUT 2009, 2010a), and/or
- Air sampling limits after 1992 (ORAUT 2005a).

Internal dose is assigned based on current revisions of Project guidance (NIOSH 2002; ORAUT 2004a, 2005a, 2007).

5.6.2 Workers with Pre-1963 Employment

A class of employees from NTS has been added to the SEC for the period January 27, 1951 through December 31, 1962. For some claims it is required that a partial dose reconstruction be conducted. As a result of the SEC class for NTS, NIOSH has determined, and the Secretary of Health and Human Services has concurred, that in the absence of bioassay results for the worker, internal doses cannot be reconstructed between 1951 and 1962 inclusive for an EE. Based on the SEC petition evaluation, internal dose is not to be reconstructed for work before 1963 unless a worker has specific bioassay results that can be directly related to an event or incident. Any bioassay results found in the DOE files for NTS workers before 1963 should be assumed to be valid, [HASL-300 (Harley 1976) procedures were used in the early bioassay program], therefore these results can be used to evaluate internal dose. If internal monitoring exists for an individual in the SEC timeframe, it should be assumed to be a nonroutine sample and associated with some discrete event.

Some of the standard techniques used for process facilities in the DOE complex cannot be employed at NTS because internal monitoring was limited and exposure was episodic. Based on the bioassay results for the worker, both acute and/or chronic intakes should be evaluated and the best fit assigned when a best estimate is required. NTS bioassay results would normally be identified with an acute

event (test or venting) associated with the worker's employment period, duties, and or access records included in the DOE file. However, based on the records for certain workers indicating involvement in multiple tests, a chronic exposure may be appropriate.

Only the measured isotope can be used. For example, if there is a urine sample result for plutonium, the DR should not apply ratios for uranium and fission products to that intake. Only the plutonium should be estimated. If the bioassay report includes other analyses, such as gross gamma, a worst-case gamma emitter can be assumed.

Results from urine, chest counts, fecal, and WBCs found in the worker's records should be evaluated for intakes. Other analyses such as nasal swabs are a qualitative measure used to evaluate the potential for exposure. For NTS, the dose assignments based on bioassay results are consistent with project guidance (NIOSH 2002; ORAUT 2007).

For miners and other who are identified as working underground (i.e., tritium results found in the case records), the results can be evaluated. Radon and/or thoron assignment in accordance with Section 4.4.4 of the environmental chapter of this TBD (ORAUT 2010a) may also be considered depending on the cancer type (Allen 2007).

5.6.3 Workers with Employment From 1963 through 1992

NIOSH has determined, and the Secretary of Health and Human Services has concurred, that NIOSH lacks sufficient information that would allow it to adequately estimate internal exposures during the period 1963 through 1992. NIOSH believes that the cessation of all nuclear testing, coupled with the implementation of the 1993 NTS internal technical basis document that demonstrates NTS compliance with 10 CFR Part 835, supports NIOSH's ability to bound internal dose for the evaluated class starting in 1993. Although HHS has added this class to the SEC, NIOSH intends to use any available internal and external data for the recommended period (that can be interpreted using existing NIOSH processes and/or procedures) for the purpose of partial dose reconstructions.

Some of the standard techniques used for process facilities in the DOE complex cannot be employed at NTS because internal monitoring was limited and exposure was episodic. Based on the bioassay results for the worker, both acute and/or chronic intakes should be evaluated and the best fit assigned when a best estimate is required. NTS bioassay results would normally be identified with an acute event (test or venting) associated with the worker's employment period, duties, and or access records included in the DOE file. However, based on the records for certain workers indicating involvement in multiple tests, a chronic exposure may be appropriate.

Only the measured isotope can be used. For example, if there is a urine sample result for plutonium, the DR should not apply ratios for uranium and fission products to that intake. However, in the event that ^{239}plutonium results are identified in the absence of results for ^{241}americium and ^{238}plutonium, the absent result shall be accounted for by assuming the ratios to ^{239}plutonium provided in Section 5.2.3. If the bioassay report includes other analyses, such as gross gamma, a worst-case gamma emitter can be assumed.

Results from urine, chest counts, fecal, and WBCs found in the worker's records should be evaluated for intakes. Other analyses such as nasal swabs are a qualitative measure used to evaluate the potential for exposure. For NTS, the dose assignments based on bioassay results are consistent with project guidance (NIOSH 2002; ORAUT 2007).

For miners and others who are identified as working underground (i.e., tritium results found in the case records), the results can be evaluated. Ambient environmental radon and/or thoron assignment

in accordance with Section 4.4.4 of the environmental chapter of this TBD (ORAUT 2010a) may also be considered depending on the cancer type (Allen 2007).

5.6.4 Approach to Internal Dose Reconstruction for NTS

Starting April 1, 1957, all persons entering NTS were required to wear a film badge with the correct monthly color coding (Smith 2003). Because all personnel were issued film badges, and later thermoluminescent dosimeters (TLDs), the presence of external monitoring results does not equate to the worker entering radiation exclusion (radex) areas (i.e., areas of elevated alpha surface contamination measured by portable alpha survey meter). This means any distinction in the internal dose assigned has to be made based on the available information about the job duties of the individual and/or positive external dose results in the records. For example, you can assign environmental dose to workers not directly involved in post-test activities (e.g., cement truck driver, cafeteria worker, housekeeper, etc). This can be done starting in 1963.

After 1992, for individuals with job classifications where environmental internal dose assignment is not appropriate, the efficiency approach based on air sampling limits may be used to overestimate cases (ORAUT 2005a) with employment after 1992.

Note: For respiratory cancers (e.g., Lung, ET1, and ET2) when underground work has been identified, ambient environmental radon exposures should be applied (whether monitored or unmonitored) according to Section 4.4.4 of the environmental section of this TBD (ORAUT 2010a). Radon dose is assigned as working level month (WLM) in the radon section of the Interactive RadioEpidemiological Program (IREP) input sheet. Ensure that exposure to thoron (^{220}Rn) is also addressed. Thoron levels are provided for two locations in ORAUT (2010a). If the individual worked in other locations, thoron levels can be assigned using the thoron:radon ratio in these tables (typically the thoron:radon ratio is 2:1, other values can be used as determined by the DR). NIOSH (2008) provides a method for determining the dose from thoron. This dose is assigned as alpha dose in the standard IREP dose section using a lognormal distribution with the same geometric standard deviation as radon (entered into the radon section of the IREP sheet). Radon and thoron exposures should be prorated based on the estimated time underground and employment periods. This should all be discussed in the report.

The DR must be diligent in investigating the type of work the individual performed, where the work was performed, and what nuclear testing was being performed while the individual was at the site. This information will be used to determine the most appropriate intake type (chronic vs. acute), and what radionuclides need to be evaluated (e.g., cesium, strontium, radon, iodine, etc.).

In general as an efficiency method, the assumption of a chronic exposure will often work (especially when calculating missed dose), because it will approximate a series of acute intakes (i.e., there were enough tests that some individuals could have had an almost constant potential for intakes during some years). DRs should review the compiled list of events that occurred during the testing program (DOE 2000) to aid in this determination. However, there is still room for judgment, particularly in the case of a single positive result, and where there is a higher likelihood of the result being associated with a particular test. This information can be compared to the employee's period of employment to determine if the EE was potentially exposed during an event. During events the number of EEs associated with the test would have been extremely small.

5.6.4.1 Single Positive Result

Bioassay was not performed on a routine basis for most individuals, it was typically performed when an intake for the individual was suspected or the bioassay was specified as part of the Radiation Safety Plan written for a specific event. Therefore, it is likely that a positive sample result is

associated with a recent intake. For a single result that can be associated with a particular event, it is best modeled as an acute intake. For NTS, the DR may need to associate the intake with an event if there is an indication that the individual was involved in reentry after the test rather than at the time of the test itself. In this case, the DR might need to put the exposure date no more than 1 week before the sample or select the date that best agrees with later negative data. There will often be no follow-up bioassay.

Note: "Bioassay" includes both *in vivo* and *in vitro* measurements. If only a baseline or a termination bioassay is included in the records, then the DR can consider the individual to be not monitored.

5.6.4.2 For Individuals with Urine Bioassay Results (Positive or <MDA)

Note: Bioassay results are very limited in the NTS records. Most will be from suspected intakes or a termination result. Be aware that often no follow-up bioassay was performed to help in the determination of assigning acute or chronic intake. Occupations that did receive routine bioassay by the 1960s are security guards, radiation monitors, and individuals included in immediate reentry after an event. The NTS Site Description includes a listing of job titles and duties that will help in the determination of chronic vs. acute exposures.

For each radionuclide (plutonium, uranium, strontium, tritium, etc.) but not gross gamma, gross alpha, or GFPs (gross beta) for which there are urine bioassay results:

- Review all site area(s) where the individual worked using his or her entire work history.

- Select the absorption type (F, M, S, or Super S) most favorable to the claimant for the radionuclide and organ of interest based on the work areas or job classification. If this cannot be determined, use the absorption type most favorable to the claimant listed for the entire site.

- For ^{238}Pu and, ^{239}Pu <MDA, determine the ^{239}Pu intake based on half of the MDA [standard missed dose calculation per ORAUT (2007)] and add intakes of the associated radionuclides, ^{238}Pu and ^{241}Am. These intakes are based on the NTS soil inventories of these radionuclides relative to ^{239}Pu. The intake of ^{238}Pu = 0.91 × ^{239}Pu and ^{241}Am = 0.59 × ^{239}Pu (see Table 4-6 of ORAUT 2010a). These ratios should also be used for positive ^{239}Pu results when results for ^{238}Pu and ^{241}Am are <MDA or not available. If lung counter results are provided in conjunction with the *in vitro* results, determine the projected chest count values and compare them to the measured results to assure that the calculated intakes are not in disagreement with the measured values.

- For uranium results <MDA, calculate missed dose based on MDA for total uranium and then, for purposes of dose calculation, assume the calculated intake (in units of activity) is 100% ^{234}U. Bioassay methods for uranium were developed at NTS, and uranium results may be found for some individuals. The type of uranium and the specific activity to apply will vary according to the duties of the individual and the timeframe when the individual was on site. The IMBA program includes defaults that can be used. For an underestimate, if lung counter results are provided in conjunction with the *in vitro* results, calculate the potential missed intake rates based on the lung counter MDAs to ensure that the intakes of ^{235}U and ^{238}U have not been overestimated using the urine data.

- Calculate ^{3}H doses per the technical information bulletin on tritium (ORAUT 2004b). Tritium is an issue for individuals in Area 12 B Tunnel (U12b). The presence of tritium in this area was identified in 1961 and individuals who worked there have been routinely monitored since then. The original source of the tritium was thought to be from 1958 test activities.

For unspecified *in vitro* bioassay results (alpha, beta, gamma, and fission products):

- For gross alpha results (positive), assume analysis was for ^{239}Pu if no other bioassay results specific to plutonium are available.
- For gross beta and GFP results, assume the analysis was for ^{90}Sr unless there is evidence that other radionuclides were present. Use the GFP MDAs applied as ^{90}Sr.
- For gross gamma results, assume analysis was for ^{137}Cs unless there is evidence that other nuclides are present. Use the gamma MDAs assigned as cesium.

5.6.4.3 For Individuals with Whole-Body Count Results

For WBC results all <MDA or no remarks:

- Use the NTS Chooser tool/aid to determine the nuclide favorable to the claimant based on the length of employment and organ identified with the cancer type using half of the published MDA.
- Determine the organ dose associated with each organ.
- Use a triangular distribution for missed dose.
- For specified fission products with positive WBCs, use standard project methods (ORAUT 2007).

5.6.4.4 For Individuals with Lung Count Results

Lung (or chest) counts were typically performed after a suspected intake of thorium, uranium, or a TRU element.

- If there are urine sample results for any of the above nuclides, use the lung counts in conjunction with the urine sample results to ensure that the two sets of measurements are not in disagreement.
- If there are no urine sample results and the results are all <MDA, assume the radionuclide of interest was ^{241}Am, used as a tracer for plutonium intakes, unless there is indication of other radionuclides. Calculate missed dose using the MDA for ^{241}Am and using an Associated Radionuclide of ^{241}Pu in IMBA. This is necessary because ^{241}Am is being produced by the decay of ^{241}Pu in the plutonium mixture and this ingrowth must be taken into account. Add intakes of associated radionuclides (^{239}Pu and ^{238}Pu). Calculate missed dose.

5.6.5 Examples of the Assignment of Internal Dose for Selected Job Titles

Support workers not directly involved in testing (e.g., cafeteria worker, clerk, housekeeper, bus driver, warehouseman, expediter, etc.) and who were not assigned underground duties. The assignment of internal dose based on environmental values (ORAUT 2010a) should be an overestimate of exposure. Bioassay records above the MDAs in the DOE records should be evaluated and included to the internal dose estimate. Claims that include this environmental dose can be completed as over or best estimates.

Workers directly involved with testing [e.g., drill crews or anyone with a job title indicating a full-time underground work assignment (full shift and a 40-hour workweek)]. ORAUT (2008) contains job

titles and identifies the potential for tunnel work. For respiratory tract cancers, include radon (ORAUT 2010a) and thoron (NIOSH 2008) for both over and best estimate cases. If the worker records include tritium monitoring, evaluate tritium as discussed in Section 5.1.3.2 and include it for all cancer types. Bioassay records found in the worker records should be evaluated and included in the internal dose estimate using a method appropriate for underestimate, best estimate or overestimate based on project guidance (ORAUT 2007).

Engineers, scientists, supervisors, waste management and safety personnel, and allied crafts. These workers could require a combination of environmental and air sampling. Some job categories might spend some time in the tunnels and some time in town or in an onsite office. For a favorable to claimant assignment of internal dose, DRs should assign these workers environmental dose. Worker bioassay records above the MDAs should be evaluated and included in the internal dose estimate. Workers who were involved in the assembling of devices and the analysis of debris from events will likely have positive external dosimetry results as well as bioassay results.

The records might indicate that workers were underground for several years and then moved to a job classification that put them above ground. For example, some RCTs and industrial hygiene personnel were assigned full-time underground for their entire term of service. Others were assigned full-time underground for limited periods (e.g., a quarter or a year). Still others might have gone underground only on special assignment (e.g., reentry support or special monitoring tasks) or to backfill for regularly assigned employees. It would be reasonable and favorable to claimants to apply environmental internal dose for all years when the length of time the worker actually worked in the tunnels cannot be determined.

A reasonable or minimizing approach would be to assign internal dose based on the bioassay results for the worker. Both acute and chronic intakes should be evaluated and the best fit assigned. Because exposure at NTS was episodic, the bioassay results would normally be identified with an acute event (test or venting) associated with the worker's employment period, duties, and or access records included in the DOE file. However, based on the records for certain workers indicating involvement in multiple tests, a chronic exposure may be appropriate.

5.6.6 Unusual or Unique Working Environments

The following circumstances indicated in the DOE records will require special treatment or evaluation:

5.6.6.1 Potential for Iodine Exposure

For a worker with thyroid cancer whose records indicate that there was involvement in reentry after a test or in a venting or working drill-backs before 1964, DRs should evaluate iodines based on the information in Section 5.3.3. From the records for the claim, a reentry time after the event for the worker should be estimated and used in the determination of the relative abundances of the iodines. Baneberry, the most significant venting to take place at NTS, provides a bounding concentration that can be used to develop an iodine dose estimate when the worker received no internal monitoring. Based on the maximum reported on site concentration from the Baneberry venting, no significant exposure occurred for the unmonitored worker (see Section 5.3.3).

5.6.6.2 Cleanup after Safety Tests

If a DR suspects but cannot confirm intake of plutonium, americium, and thorium, a more sensitive urine result from a later period can be used to determine the worst-case intake at a time when analysis was not as sensitive. The DR should use MDAs from later urinalyses or, if many samples show no detection, one-half the MDA. Plutonium is a consideration for workers on Project 56 (1955 and 1956) and/or Project 57 (1957), which were plutonium dispersal experiments. Reference to

"Plutonium Valley" or Area 11 or work at the Nellis Air Force Range during this period is an indication of involvement. Partial remediation of the Project 57 area was completed in 1981 and soils remediation is included in the ongoing environmental restoration program for both locations. Plutonium urine or fecal sampling in the record is a possibility for workers who participated in the original tests or cleanup activities up to the present time. Plutonium calculations should also include evaluation for Type Super S plutonium (ORAUT 2010c).

5.6.6.3 Nuclear Reactor Development Station

The dose estimation methods and models for nuclear reactor tests in the Naval Research Defense Laboratory (NRDL) report (NRDL 1968) have been evaluated for their usefulness and application to the NIOSH dose reconstruction process for personnel participating in reentry operations at NTS. The dose models in the NRDL report are applicable *only* to the relatively small population of NTS workers that were involved in reentry operation after a nuclear propulsion rocket test at the Nuclear Reactor Development Station (NRDS). The data are experiment specific and require knowledge of specific physical parameters.

The NRDL report provides methods of computing doses from both individual particles and infinite field contamination levels from the nuclear rocket tests. The model provides methods for calculating doses to the lungs, GI tract, skin, and gonads for particular exposure scenarios. If a worker who participated in NRDS nuclear rocket re-entries has a covered cancer in a location where a hot particle exposure has been documented, and the worker can be associated with a specific test, and the requisite physical parameters for the test are available from NTS records, DRs may consider using the models and methods in the NRDL report. If this information is not available for NRDS workers, internal exposures can be addressed through procedures developed for the project (NIOSH 2002) and the IMBA EXPERT codes when bioassay results are available in the dosimetry records. If the worker's dosimetry records include dose reconstruction using the NRDL methods, the resulting doses should be included in the IREP evaluation.

5.6.6.4 Incidents

Workers directly involved in accidental releases will require internal evaluation if they were in the area within 10 days of the release. This information would be in the DOE records. Table A-26 lists releases with identified radionuclides. Relative abundances can be determined according to methods in ORAUT (2010a). These workers might have been monitored or not monitored for internal exposure.

For a worker with a metabolic cancer and bioassay results, DRs should review Attachment A to determine what other radionuclides would affect development of the cancer and perform the evaluation based on the radionuclides that provided 90% of the dose. Attachment A contains information on radionuclides present during test categories and historical MDAs. Based on theoretical areal concentrations associated with testing, intakes can be assigned and doses calculated.

5.7 ATTRIBUTIONS AND ANNOTATIONS

All information requiring identification was addressed via references integrated into the reference section of this document.

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GLOSSARY

alert level

A urinalysis result that calls the attention of the observer to the fact that further action related 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. The following are alert levels for specific analyses:

- *gamma* – The alert level for routine quarterly urine samples was "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 were activity concentrations indicative of body burdens that are 10% of the MPBBs for continuous exposure given in *Health Physics*, volume 3, "Report of ICRP Committee II on Permissible Dose for Internal Radiation (ICRP 1959)," June 1960.

- *tritium* – The alert level for a single tritium sample was 1.53×10^{-2} µCi/cm^3. The dose "to infinity" to the fat-free soft tissue of the whole body indicated by a single exposure resulting in this concentration was 50 mrem.

- *GFP* – The alert level was 10^{-7} µCi/cm^3.

- *Plutonium* (or other transuranic elements) – The alert level was any detectable level.

- *Strontium* – The alert level was any detectable level.

atmospheric

A test conducted above ground or above water (i.e., in the open air). The last atmospheric test at NTS was in 1962.

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.

controlled release

A planned, filtered release frequently performed to reduce airborne radiation levels in the tunnel working environment.

controlled tunnel purging

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

crater

A nuclear device placed shallow enough underground to produce a throw-out of earth when exploded.

detonation

A single nuclear device explosion; one or more comprise a test.

drill-back

Directional drilling operation performed after a shaft test to sample fission product materials in the test cavity.

gas sampling

Operation usually performed after test activities have ceased to determine levels of noble gases present.

gross beta

Measure of the total beta activity.

H

Hour Detonation time (zero hour), the time the device was detonated.

hole

Alphanumeric designation given to the nuclear weapon test/detonation conducted at NTS. The alpha designator refers to the type of location (i.e., S = surface, U = underground). The numeric designator refers to the geographical area of the site. The remaining letters refer to the sequential ordering of the test/detonation. For most atmospheric tests, only the NTS geographical area was given.

kiloton (kt)

The energy of a nuclear explosion that is equivalent to the explosive power of 1,000 tons of TNT.

late-time seeps

[1] 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. [2] A slow release of gases from test detonation sites that could exist from a few hours to even weeks after all other operations in the area have ceased (Schoengold, DeMarre, and Kirkwood 1996).

lower limits of detection (LLD, circa 1980s)

[1] 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 s_o. After January 1, 1978, LLD = 3.29 s_o. The term s_o is the estimated standard error for the net sample activity (before 1993). [2] A value selected above the MDA to reduce the probability of reporting false positive results (Allen et al. 1993).

minimum detectable activity (or amount) (MDA)

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

monitored worker

A worker who participated in the internal dosimetry program at NTS. All persons entering NTS were monitored externally via film badges or thermoluminescent dosimeters.

operational release

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

Radex areas

Full Radex Area: radiological contamination is >100 mR/hr (gamma) measured 3 ft from the ground, or $>10^4$ cpm/55 cm^2 (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.

Limited Radex Area: radiological contamination is ≥ 10 mR/hr but <100 mR/hr (gamma) measured 3 ft from the ground, or ≥10^3 to ≤10^4 cpm/55 cm^2 (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, the first entry into an area or tunnel (as soon as safety limitations permit) immediately after 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 minimum level of a bioassay measurement result, which requires the measurement lab to provide prompt notification to Dosimetry (Allen et al. 1993).

safety experiment

[1] Experiment designed to confirm a nuclear explosion will not occur in case of an accidental detonation of the explosive associated with the device. Some safety tests were set off at the bottom of unstemmed drilled holes, producing a "Roman candle" effect (a test conducted underground in an unstemmed hole to minimize, but not eliminate, the release of radioactivity to the atmosphere). Because incandescent gases were released, this was sometimes referred to as a "Roman candle" effect. [2] Experiment designed to confirm a nuclear explosion will not occur in case of an accidental detonation of the explosive associated with the device (Schoengold, DeMarre, and Kirkwood 1996).

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.

shaft

A drilled or mined vertical hole in which a nuclear device is exploded. The hole depth and diameter were dictated by the projected yield and diagnostics. Typical hole depths ranged from 600 to 2200 feet, and typical hole diameters ranged from four to 10 feet.

stemming

Placement of materials such as grout, magnetite, bentonite, sand, and other materials to contain the nuclear explosion in a tunnel or shaft.

surface

A nuclear device placed on or close to the earth's surface.

test

Either a single underground nuclear explosion conducted at a test site, or two or more underground nuclear explosions conducted within an area delineated by a circle having a diameter of 2 km and conducted within a total period not to exceed 0.1 second (Threshold Test Ban Treaty).

tolerance level

The term for maximum permissible exposure.

ton

A measure of the amount of energy released when a nuclear device explodes stated in terms of the quantity of TNT that would produce the same amount of explosive energy. A ton is equivalent to 1 ton of TNT.

tower

A nuclear device mounted at the top of a steel or wooden tower and exploded in the atmosphere.

tunnel

A nuclear device exploded at the end of a long horizontal drift mined into a mountain or mesa in such a way that the burst point is placed deep within the earth.

uncontrolled release

A spontaneous release occurring after a test but before post-test drilling operations commence. Schoengold, DeMarre, and Kirkwood (1996) uses this term when referring to tunnel tests.

underground

Underground nuclear test conducted in a tunnel or at the bottom of a drilled hole or shaft. Some underground nuclear tests were not designed to contain all radioactivity (e.g., cratering tests or safety experiments).

venting

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

weapons effects

A nuclear test to evaluate civil or military effects of a nuclear detonation on various targets, such as military hardware.

weapons related

A nuclear detonation conducted to test a nuclear device intended for a specific type of weapon system.

yield

The total effective energy released in a nuclear explosion, usually expressed in terms of equivalent tonnage of TNT required to produce the same energy release in an explosion.

ATTACHMENT A
OCCUPATIONAL INTERNAL DOSE FOR MONITORED WORKERS
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TABLE OF CONTENTS

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. Internal monitoring at NTS was limited and was used only for specific job categories supporting events and after incidents.

The BN records archive has all available bioassay records for NTS contractors. The archive does not include records for national laboratory employees, which are held by the respective laboratories.

Records for REECo employees assigned to the Tonopah Test Range (TTR) under "Work for Others" are held by Sandia National Laboratory.

Internal dosimetry records during the period from 1955 through 1959 include nasal swabs and plutonium bioassays performed by LASL. Beginning in 1958, NTS urine bioassays included tritium, gross beta, gamma scan, and plutonium. In 1990, fecal sampling was available. Other nuclear testing data are included in individual test radiological safety reports, final operation radiological safety reports, monitor or RCT logbooks, area access registers, air sampling forms, laboratory request forms and follow-up analyses, and meteorological data (DeMarre 2004a,b). The term used in earlier periods for RCTs was *Radiation* or *Rad-Safe Monitors*.

A.1 BIOASSAY CODES AND *IN VITRO* MINIMUM DETECTABLE ACTIVITIES AND DETECTION LEVELS

A.1.1 Codes and Special Terminology Used in Bioassay Records

Employer codes and job titles for some NTS contractors [REECo, EG&G, H&N, and RSN] are available to DRs in the SRDB maintained by the Division of Compensation Analysis and Support of the National Institute for Occupational Safety and Health. Other NTS contractors include BN; Lockheed Martin Nevada Technologies, Inc.; Johnson Controls Nevada, Inc.; and WSI. Former NTS contractors included H&N (1956–1990), FSN (1963–1990), EG&G (1951–1995), REECo (1953–1995), and RSN (1990–1995). EG&G had offices and shops in Las Vegas, but some EG&G workers worked at NTS for extended periods. Some REECo workers were assigned to the TTR; Sandia National Laboratory is the custodian of TTR dosimetry records.

The computerized bioassay records contain the codes listed in Tables A-1 through A-5.

These codes were not identified with a particular period. The measurements of alpha, beta, gamma, and GFPs were just a "gross measurement" to determine if additional measurements were needed (DeMarre 2004a,b). 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. Swipes taken for a worker on a safety test (post test), were used as an indicator for the presence of plutonium in the sample. For gamma, when a multichannel analyzer was used, specific radionuclides could be cited if they were detectable. For example ruthenium could occasionally be detected.

The main internal concerns were tritium, iodines, and plutonium (Arent and Smith 2004). 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 WBC facility at the 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 ^{239}Pu. Synonyms for uranium included "Oralloy" for enriched uranium, "tuballoy" for natural uranium, and "D-38" for depleted uranium. "LT" or "-" in the bioassay records means less than the detection limit. The

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Table A-1. Codes for analyte (DeMarre 2003).

For heading "an_desc_co"	For heading "analyze_de"
01	003-H (tritium)
02	ALPHA
03	BETA
04	GAMMA (gross)
05	GFP
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)

code "-99" means not detected (Arent and Smith 2004). Table A-2 lists body part codes and abbreviations that are specific to different periods, as listed.

Table A-2. Codes for body parts (DeMarre 2003).

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

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For heading[a] "bdy_prt_cd"	For heading[b] "bdy_prt_pr"	For heading "bdy_prt_df"	Period
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 A-3 lists codes for internal radionuclides and specific operations 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.

Radionuclides 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 2004a,b).

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Table A-3. Codes for radionuclides and sample types (DeMarre 2003).

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	TRU elements	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

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 2004a,b).

Table A-4 lists codes for the type of sample to be analyzed, and Table A-5 lists the associated units.

Table A-4. Codes for sample types (DeMarre 2003).

sam_typ_co	Sample_typ
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

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sam_typ_co	Sample_typ
30	Lung
31	Whole body
60	Miscellaneous

Table A-5. Codes for units (DeMarre 2003).

Code	Unit	Description		Code	Unit	Description
10	dpm	disintegrations per minute		45	µCi/m^3	microcuries per cubic meter
11	dpm/cm^3	dpm per cubic cm		46	µCi/kg	microcuries per kilogram
12	dpm/g	dpm per gram		50	mCi	microcuries
13	dpm/L	dpm per liter		51	mCi/cm^3	millicuries per cubic centimeter
15	dpm/m^3	dpm per cubic meter		52	mCi/g	millicuries per gram
16	dpm/kg	dpm per kilogram		53	mCi/L	millicuries per liter
20	Bq	Becquerels		54	mCi/mg	millicuries per milligram
21	Bq/cm^3	Becquerels per cubic centimeter		55	mCi/m^3	millicuries per cubic meter
22	Bq/g	Becquerels per gram		56	mCi/kg	millicuries per kilogram
23	Bq/L	Becquerels per liter		60	cpm	counts per minute (c/m)
25	Bq/m^3	Becquerels per cubic meter		61	cpm/cm^3	c/m per cubic centimeter
26	Bq/kg	Becquerels per kilogram		62	cpm/g	c/m per gram
30	pCi	picocuries		63	cpm/L	c/m per liter
31	pCi/cm^3	picocuries per cubic centimeter		64	cpm/mg	c/m per milligram
32	pCi/g	picocuries per gram		65	cpm/m^3	c/m per cubic meter
33	pCi/L	picocuries per liter		66	cpm/kg	c/m per kilogram
35	pCi/m^3	picocuries per cubic meter		67	%	percent
36	pCi/kg	picocuries per kilogram		68	µg/g	micrograms per gram
40	µCi	microcuries		69	µg/L	micrograms per liter
41	µCi/cm^3	microcuries per cubic centimeter		70	µg	micrograms U (Reg: kinetic phosphorescence analysis)
42	µCi/g	microcuries per gram		71	µS/cm	microsiemens per centimeter
43	µCi/L	microcuries per liter		72	mS/cm	millisiemens per centimeter
44	µCi/mg	microcuries per milligram				

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. DRs should review data from the original forms, not the microfiche “deadbio” index.

Table A-6 lists special location terminology found in NTS bioassay records.

Table A-6. Special terminology (ORAUT 2008, 2010b).

Term	Explanation
A-23	Mercury, NV
A-24	Las Vegas
A-52	TTR
A-400	A-25 LANL term for NRDS area
A-401	A-25 LLNL term for Pluto/Ramjet area
A-6, CP-1	Test Control Room Building
A-6, CP-2	Radiological Safety (RadSafe) Building and dosimeter issue location
A-23, Building 1000	Badge office outside NTS main gate (Mercury)

Term	Explanation
A-23, Buildings 111, 155, and 650	RadSafe/dosimeter issue buildings
Alcove	Side excavation in tunnel drift
BP	Bypass (type of tunnel drift)
CP	Control Point or A-6
Downhole shaft	Vertical underground tunnel
Drift	Horizontal tunnel
GZ	Ground zero
Invert	Part of bottom of tunnel made level by adding materials.
LOS	Line of sight (drift or pipe)
Portal	Tunnel entry point
Shaft designation	Example "U20bb" U = underground, 20 = Area 20, bb = hole number
Tunnel designation	Example "U12b.10" U = underground, 12 = Area 12, b = B Tunnel, 10 = drift number 10

Tunnels are located in NTS areas (Wilcox and Coogan 1960; ORAUT 2008, 2010b):

- A-1 (U1a Complex, formerly known as the Low-Yield Nuclear Explosive Research Facility Complex)
- A-12 [B, C ("Saturn"), D, E, F, G, J, K, N, P, and T Tunnels]
- A-15 (15E Tunnel)
- A-16 (16A Tunnel)
- A-25 (X Tunnel, non-nuclear experiments only)

A.1.2 *In Vitro* Analyses for Individual Radionuclides

Table A-7 lists limits of detection for urine and fecal analyses. The MDA is an *a priori* value used to evaluate the laboratory's ability to detect an analyte in a sample. LLD is defined in Allen et al. (1993) 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-hour period.

Author unknown (ca. 1978) notes a change in the definition of LLD. The LLD was defined as the smallest amount of sample activity that was reported positive with a specified degree of confidence (i.e., 95% of the time). The LLD expression for data before January 1, 1978, was:

$$LLD = 2s_o \quad \text{(A-1)}$$

where s_o = the estimated standard error for the net sample activity.

The LLD expression for data changed after January 1, 1978, to:

$$LLD = 3.29s_o \quad \text{(A-2)}$$

Table A-7. 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 1968–1987a
H-3	1988[c]–1999[d]	300	-	Allen et al. 1993
H-3	2000-2002	0.001 µCi/L	-	BN 2000
H-3	2003–present	0.005 µCi/L	-	McMahan and Ogurek 2003
Na-22	1970–present[d]	200	-	Allen et al. 1993
Na-24	1970–present[d]	100[e]	-	Allen et al. 1993
Mn-54	1970–present[d]	100	-	Allen et al. 1993
Co-57	1970–present[d]	100	-	Allen et al. 1993
Fe-59	1970–present[d]	600	-	Allen et al. 1993
Co-60	1970–present[d]	50	-	Allen et al. 1993
Sr-85	1970–present[d]	200	-	Allen et al. 1993
Sr-90	1961–1969[d]	25 pCi/sample (limit of sensitivity)[f]	-	Author unknown undated a
Sr-90	1970–1999[d]	0.8[g]	-	Allen et al. 1993
Sr-90	2000–present	1	-	BN 2000, McMahan and Ogurek 2003
Zr-95	1970–present[d]	600	-	Allen et al. 1993
Nb-95	1970–present[d]	100	-	Allen et al. 1993
Tc-99m	1970–present[d]	100	-	Allen et al. 1993
Rh-101	1970–present[d]	100	-	Allen et al. 1993
Rh-102	1970–present[d]	100	-	Allen et al. 1993
Rh-102m	1970–present[d]	200	-	Allen et al. 1993
Ru-103	1970–present[d]	200	-	Allen et al. 1993
Ru-106	1970–present[d]	1000	-	Allen et al. 1993
Sb-122	1970–present[d]	100	-	Allen et al. 1993
Sb-124	1970–present[d]	100	-	Allen et al. 1993
Sb-125	1970–present[d]	300	-	Allen et al. 1993
I-131	1961–1969[d]	10 pCi/sample (limit of sensitivity)	-	Author unknown undated a
I-131	1970–present[d]	100	-	Allen et al. 1993
I-132	1970–present[d]	90	-	Allen et al. 1993
Te-132	1970–present[d]	100	-	Allen et al. 1993
Ba-133	1970–present[d]	200	-	Allen et al. 1993
Ba-133m	1970–present[d]	100 or 400[h]	-	Allen et al. 1993
I-133	1970–present[d]	100	-	Allen et al. 1993
Cs-134	1970–present[d]	500	-	Allen et al. 1993

Radionuclide	Period	Urine MDA (pCi/L)[a]	Fecal MDA (pCi/g)[a]	Source
I-135	1970–present[d]	200	-	Allen et al. 1993
Cs-136	1970–present[d]	100	-	Allen et al. 1993
Cs-137	1961–1969[d]	10 pCi/sample (limit of sensitivity)	-	Geiger and Whittaker 1961, Author unknown undated a
Cs-137	1970–present[d]	100	-	Allen et al. 1993
Ce-139	1970–present[d]	100	-	Allen et al. 1993
Ba-140	1961–1969[d]	10 pCi/sample (limit of sensitivity)[l]	-	Author unknown undated a
Ba-140	1970–present[d]	500	-	Allen et al. 1993
La-140	1970–present[d]	200	-	Allen et al. 1993
Ce-141	1970–present[d]	200	-	Allen et al. 1993
Ce-143	1970–present[d]	200	-	Allen et al. 1993
Ce-144	1970–present[d]	800	-	Allen et al. 1993
Nd-147	1970–present[d]	400	-	Allen et al. 1993
Eu-152	1970–present[d]	300	-	Allen et al. 1993
Eu-154	1970–present[d]	200	-	Allen et al. 1993
Eu-155	1970–present[d]	500	-	Allen et al. 1993
Yb-169	1970–present[d]	600	-	Allen et al. 1993
Ta-182	1970–present[d]	300	-	Allen et al. 1993
W-187	1970–present[d]	300	-	Allen et al. 1993
Ir-192	1970–present[d]	100	-	Allen et al. 1993
Fission products- GFP (beta)	1977–1987	1E-10 µCi/mL (detection limit)[b]	-	REECo 1968–1987a
Fission products- GFP	1988[c]–present[d]	3[j]	-	Allen et al. 1993
Gross alpha	1968–present	-	1E-07 µCi/g (detection limit)[k]	Author unknown undated a
Gamma	1977–1987	5E-8 µCi/mL (detection limit)[b]	-	REECo 1968–1987a
Gamma (Cs-137)	1988[c]–present	100	-	BN 2000, McMahan and Ogurek 2003
Ra-226	1958–1969[d]	0.88 dpm/sample (detection level)[l]	-	Dummer 1958
Ra-226	1970–present[d]	300	0.4 or 0.04[h]	Allen et al. 1993
Ra-226	2000–present	0.1 pCi/sample	- ?	BN 2000, McMahan and Ogurek 2003
Th-228	1970–present[d]	-	0.01	Allen et al. 1993
Th-230	1954–1992[d]	0.5 dpm/sample (detection level) [m]	-	McClelland 1955, Author unknown undated a
Th-230	1970–present[d]	-	0.01	Allen et al. 1993
Th-230	2003–present	pCi/sample	0.04 pCi/sample	McMahan and Ogurek 2003

Radionuclide	Period	Urine MDA (pCi/L)[a]	Fecal MDA (pCi/g)[a]	Source
Th-232	1993–1999[d]	-	0.01	Allen et al. 1993
Th-232	2000–present	0.02 pCi/sample	0.05 pCi/sample	BN 2000, McMahan and Ogurek 2003
U-234	1970–present[d]	0.02	-	Allen et al. 1993
U-234	2000–present	0.04 pCi/sample	0.04 pCi/sample	BN 2000, McMahan and Ogurek 2003
U-235[n]	1961–1969	0.03 dpm/sample (limit of sensitivity)	-	REECo 1961, Author unknown undated a
U-235	1970–present[d]	0.01	0.008	Allen et al. 1993
U-235	2000–present	0.04 pCi/sample	0.04 pCi/sample	BN 2000, McMahan and Ogurek 2003
U-238	1970–1999[d]	0.02	0.008	Allen et al. 1993
U-238	2000–present	0.04 pCi/sample	0.04 pCi/sample	BN 2000, McMahan and Ogurek 2003
Elemental U	1970–present	5 μg/L[o]	-	Allen et al. 1993
Pu-238	1982–1987	2E-10 μCi/mL (detection limit)[b]	-	REECo 1968–1987a
Pu-238	1988[c]–1999[d]	0.01	-	Allen et al. 1993
Pu-238	2000–present	0.006 pCi/sample	0.03 pCi/sample	BN 2000, McMahan and Ogurek 2003
Np-239	1970–present	400	-	Allen et al. 1993
Pu-239[p]	1954–1957[d]	2 dpm/24 hr (detection limit)[q]	-	McClelland 1955
Pu-239[p]	1958–1960[d]	0.05 dpm/sample (detection limit)[r]		Dummer 1958, Author unknown undated a
Pu-239[p]	1961–1976[d]	0.005 dpm/sample (limit of sensitivity)	-	REECo 1961, Author unknown undated a
Pu-239	1977–1987	5E-11 μCi/mL (detection limit)[b]	-	REECo 1968–1987a
Pu-239	1988[c]-2000[d]	0.01 (alpha spec)[s]; 50 (gamma spec)[s]	0.004 (alpha spec)	Allen et al. 1993
Pu-239	2000–present	0.006 pCi/sample	0.03 pCi/sample	BN 2000, 2003
Pu-240	1970–present	0.01	0.004	Allen et al. 1993
Pu-240	2000–present	0.006	-	BN 2000, McMahan and Ogurek 2003
Am-241[t]	1954–1957[d]	2 dpm/sample (detection limit)[u]	-	McClelland 1955
Am-241[t]	1958–1981[d]	0.5 dpm-24-hr sample (LLD)[v]	-	Dummer 1958, REECo 1961, Author unknown undated a
Am-241	1982–1987	2E-11 μCi/mL (detection limit)[b]	-	REECo 1968–1987a
Am-241	1988[c]–1999[d]	0.03[w]	0.03	Allen et al. 1993

Radionuclide	Period	Urine MDA (pCi/L)[a]	Fecal MDA (pCi/g)[a]	Source
Am-241	2000-2002	0.008 pCi/sample	0.03 pCi/sample	BN 2000
Am-241	2003–present	0.006 pCi/sample	0.03 pCi/sample	McMahan and Ogurek 2003
Cm-244	2000–present	0.008 pCi/sample	-	BN 2000, McMahan and Ogurek 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} µCi/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 Allen et al. (1993).
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 µCi/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. McClelland (1955) 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. Dummer (1958) states the detection limit at the 99% confidence level is approximately 0.05 dpm/sample.
s. Cannot 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. Author unknown (undated a) 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%.) McClelland (1958) 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.

The change accounted for the possibility of a beta (β) error [i.e., to define the situation with 95% confidence that, on the average, activity is reported when it is present (1-β), as well as not reported when it is not present (1-α)].

All results for samples collected after January 1, 1978, appear in the REECo Sample Tracking Data Management System as actual values [i.e., as they were calculated regardless of whether they were negative or positive (Author unknown ca. 1978)]. Off-station data were reported to two significant digits and only minimal detectable concentration (MDC) values were reported for less-than results. An MDC value is calculated for every sample and compared to the sample value. If the MDC value was greater than or equal to the sample value, an MDC was reported for off-station data. For on-station use, actual values with counting errors were reported to three significant figures on preliminary reports. In the data system, less-than values were stored as three significant digits in the counting error field. Table A-8 summarizes the method of data reporting.

Table A-8. Radiological data reporting.

Date	Result	Sample tracking data management system	Off station
Pre 1978	$<2\ s_o$	$<2\ s_o$	
	$>2\ s_o$	Result $\pm$ 2 s_o (2 significant figures)	
Post 1978	$<3.29\ s_o$	Result $\pm$ MDC	<MDC
	$>3.29\ s_o$	Result $\pm$ 2 s_o (3 significant figures)	Result $\pm$ 2 s_o (2 significant figures)

A.2 *IN VIVO* MDAs AND REPORTING PRACTICES

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.

A.2.1 Whole-Body Counting

The MDA values for 1967 are based on those reported by HNS (1967) for a 480-second count. The values reported by HNS are at the 99% confidence level. To facilitate comparison with other MDA values reported at the 99% confidence level, the reported values have been multiplied by the ratio of k_β at the 95% confidence level (0.645) to the value of k_β at the 99% confidence level (2.326 – 0.71). HNS reported values for ^{60}Co, ^{65}Zn, ^{95}Zr-^{95}Nb, and ^{137}Cs; the values for the remaining radionuclides were derived from the HNS data. A nominal photon MDA of 2.1 nCi is the basis for the 1967 values in Table A-7 for photons in the range 100 to 2,000 keV, with correction for the yield of the primary measurement photon, $Y_\gamma - MDA = 2.1/Y_\gamma$ nCi.

Specific efficiency data have not been found for the shadow shield counter with a 3-in. by 3-in. NaI crystal, so values for 1967 to 1977 using a 3-in. by 3-in. NaI crystal are based on values for a shadow shield counter of similar design (UK11.1) published by the IAEA (IAEA 1970). Since the decrease of detector efficiency with energy is essentially offset by the decrease in background with energy, a nominal MDA of 10 nCi is used for photon emitters in the range 0.1 to 2 MeV, with correction for the yield of the primary measurement photon, $Y_\gamma - MDA = 10/Y_\gamma$ nCi.

The counter used during the period from 1977 to 1980 featured use of an 8-in. by 10-in. NaI detector. The efficiency reported for this detector is determined from $E = 0.0089 \times e^{-0.000175E\gamma}$, where E_γ is in

keV. Figure A-1 shows the efficiency curves reported for that detector, and the 11.5-in. detector installed in 1981. Figure A-2 shows the counting efficiency in terms of counts per minute per photon nCi. Background values for these detectors were not reported. Background values for ^{137}Cs under similar conditions can be found in IAEA (1970). Six counting facilities reported values ranging from 60 to 135 cpm. Using a value of 135 cpm, a counting time of 1,300 second (21.7 minutes), and an efficiency of 7.9×10^{-3} counts per photon, the 662-keV photon MDA is 0.93 nCi. (For ^{137}Cs with a yield of 0.851 photon per disintegration, this becomes 1.1 nCi.) The values for 1978 to 1980 in Table A-9 are based on a nominal MDA value of 1 photon nCi, with correction for the yield of the primary measurement photon, $Y_\gamma - MDA = 1/Y_\gamma$ nCi.

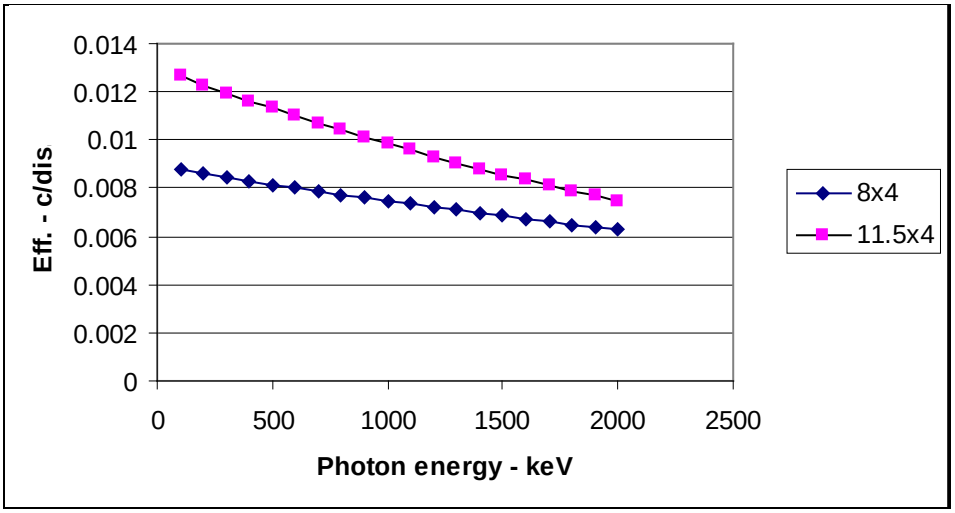

Figure A-1. NaI detector efficiency, counts per photon disintegration.

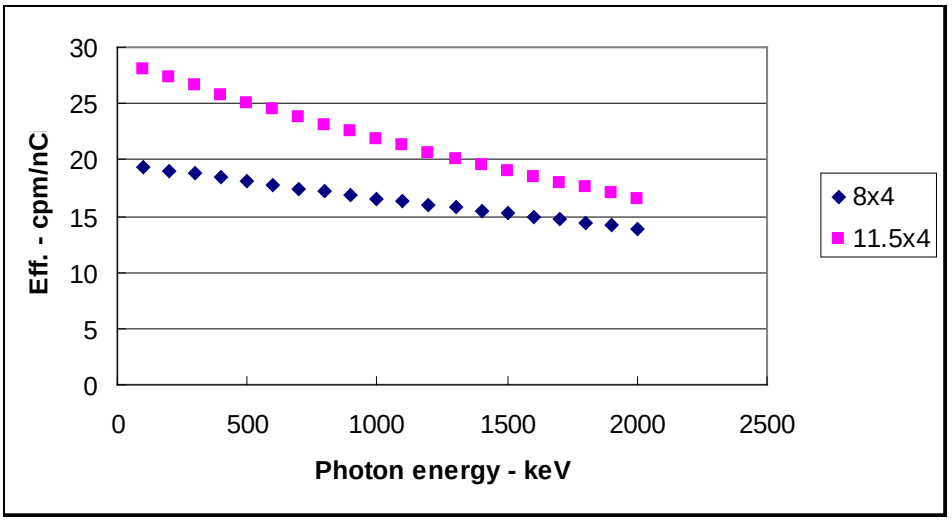

Figure A-2. NaI detector efficiency, cpm per photon nCi.

A new WBC facility began operation in 1981, with a shielded room, scanning bed, and 11.5-in. by 8-in. NaI detector (Teasdale 1985). The counting efficiency, E (counts per disintegration), can be determined from $E = (1.3 \times 10^{-2})\ (e^{-0.0002778\ E_\gamma})$, where E_γ is in keV. Little background information is available for such large crystals in IAEA (1970). However, it can be estimated from the value for the 8-in. by 4-in. detector value by using the ratio of detector face areas:

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$$Bkg_{11.5x4} = Bkg_{8x4} \times 2.07 = 279 \text{ cpm} \quad \text{(A-3)}$$

The resulting estimate for the 662-keV MDA is 2,365 γ/min, or 1.06 nCi. In this case, the reported increase in detector efficiency does not fully compensate for the calculated increase in background. For the period from 1981 to 1993, the MDA values in Table A-9 are based on a nominal photon MDA of 1.1 nCi, with correction for the yield of the primary measurement photon, $Y_\gamma - MDA = 1.1/Y_\gamma$ nCi.

Table A-9. Whole-body counting MDAs.[a]

Radionuclide	Primary photon energies[b] (keV)	Y_γ Photons per disintegration	Estimated MDAs (nCi)				
			1967[c]	1968–1977[d]	1978–1980[e]	1981–May 1993[f]	June 1993[g]
Na-22	1,274	0.999	2.1	10	1.0	1.1	0.7
Na-24	1,369 2,754	1.00 0.999	2.1	10	1.0	1.1	1.1
Sc-46	889 1,121	1.00 1.00	2.1	10	1.0	1.1	1.1
Mn-54	835	1.00	2.1	10	1.0	1.1	1.2
Fe-59	1,099 1,292	0.565 0.432	3.8	18	1.8	2.0	2.3
Co-57	122	0.856	2.5	12	1.2	1.3	1.4
Co-60	1,173 1,333	1.00 1.00	2.0	10	1.0	1.1	0.9
Zn-65	1,116	0.506	4.0	20	2.0	2.2	-
Sr-85	514	0.96	2.2	10	1.0	1.1	1.8
Zr-95	724 757	0.442 0.540	3.8	19	1.9	2.1	1.9
Nb-95	766	1.00	2.1	10	1.0	1.1	1.1
Mo-99	(140)	0.89	2.4	11	1.1	1.2	10[h]
Tc-99m							7.1[i]
Rh-101							1.4[i]
Ru-103	497	0.909	2.3	11	1.1	1.2	1.2
Ru-106, Rh-106	(512) (622)	0.20 0.0993	11	50	5.0	5.5	11
Rh-102	475 556	0.384 0.960	2.2	10	1.0	1.1	1.0
Rh-102m	475 631	0.56	2.3	18	1.0	1.1	7.9[i]
Sb-122	564	0.710	3.0	14	1.4	1.5	2.1
Sb-124	603 1,691	0.980 0.487	2.2	10	1.0	1.1	1.6
Sb-125	428 464 601 636	0.300 0.105 0.179 0.113	5.3	33	3.0	3.3	3.6
Te-132	228	0.88	2.4	11	1.1	1.2	1.5[j]
I-131	364	0.817	2.6	12	1.2	1.3	1.6[j]
I-132	667 773	0.99 0.756	2.1	10	1.0	1.1	1.4[j]
I-133	530	0.870	2.4	11	1.1	1.1	1.4[j]

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Radionuclide	Primary photon energies[b] (keV)	Y_γ Photons per disintegration	Estimated MDAs (nCi) 1967[c]	1968–1977[d]	1978–1980[e]	1981–May 1993[f]	June 1993[g]
I-135, Xe-135m	(526) 1,132 1,260	0.805 0.227 0.289	2.6	12	1.2	1.3	1.4[j]
Cs-134	605 796	0.976 0.855	2.2	10	1.0	1.1	1.5
Cs-136	341 818	0.422 1.00	2.1	10	1.0	1.1	1.4
Cs-137	662	0.851	2.1	12	1.2	1.3	1.7
Ba-133	356	0.620	3.6	16	1.6	1.8	1.8
Ba-133m	276	0.178	12.	56	5.6	6.2	1.1
Ba-140							4.8
La-140	(1,596)	0.954	2.2	10	1.0	1.1	0.4
Ce-139	166	0.80	2.7	12	1.2	1.3	1.4[l]
Ce-141	145	0.482	4.4	21	2.1	2.3	2.6[l]
Ce-143	293	0.428	5.0	23	2.3	2.5	1.4[l]
Ce-144	134	0.111	19.	90	9.0	9.9	12[l]
Nd-147	531	0.131	16.	76	7.6	8.4	5.5[l]
Eu-152	122 344 1,408	0.286 0.265 0.210	7.1	35	3.5	3.9	4.2[l]
Eu-154	123 723 1,274	0.408 0.202 0.352	5.3	25	2.5	2.8	3.2[l]
Eu-155	-	-	NA	NA	NA	NA	4.1[l]
Yb-169	-	-	NA	NA	NA	NA	0.2
Lu-174	-	-	(k)	(k)	(k)	(k)	2.5[l]
Lu-174m	-	-	(k)	(k)	(k)	(k)	3.7[l]
Ta-182	1,121 1,189 1,221 1231	0.349 0.162 0.27 0.114	6.1	29	2.9	3.2	3.3[l]
W-181	-	-	(k)	(k)	(k)	(k)	4.6[l]
W-187	480 686	0.218 0.273	7.8	37	3.7	4.1	4.6
Ir-192	316 468	0.828 0.478	2.6	12	1.2	1.3	0.9
Ra-226 & progeny	-	-	(k)	(k)	(k)	(k)	504[l,l]
Ac-227, Th-227	(236) (256)	0.123 0.070	17.0	81	8.1	8.9	28[l,m]
Th-228, Ra-224	(241)	0.041	51.8	244	24.	26.	33[l,n]
Pa-231	-	-	(k)	(k)	(k)	(k)	6.7[l]
U-235	186	0.572	3.7	17	1.7	1.9	3.4[l]
Th-232, Ac-228	(911) (965) (969)	0.258 0.0499 0.158	8.5	39	3.9	4.3	5[l,o]

Radionuclide	Primary photon energies[b] (keV)	Y_γ Photons per disintegration	Estimated MDAs (nCi)				
			1967[c]	1968–1977[d]	1978–1980[e]	1981–May 1993[f]	June 1993[g]
U-233	-	-	(k)	(k)	(k)	(k)	13[i]
U-237	208	0.212	9.9	47	4.7	5.2	4.8[i]
Np-237, Pa-233	(854) (865)	0.34 0.157	6.4	29	2.9	3.2	2.8[i,p]
Np-239	106 228 278	0.272 0.108 0.144	7.8	37	4.7	5.2	5.6[i]
Am-241	-	-	(k)	(k)	(k)	(k)	4.3[i]
Am-243	-	-	(k)	(k)	(k)	(k)	2.8[i]
Cm-243	228 278	0.106 0.140	14.9	71	7.1	7.8	4.9[i]

a. Source: IAEA (1970).
b. Parenthesis denotes daughter photon energy used for detection.
c. HNS shadow shield, with 8-in. by 4-in. NaI detector, 8-minute count (HNS 1967).
d. Pan American or REECo shadow shield with 3-in. by 3-in. NaI detector, 20-minute count (Teasdale 1985).
e. REECo shadow shield with 8-in. by 4-in. NaI detector, 1,300-second count time.
f. REECo scanning bed with 11.5-in. by 4-in. NaI detector, 1,200-second count time (Teasdale 1985).
g. REECo germanium with chair geometry (Allen et al. 1993).
h. Only bioassay method identified for this radionuclide.
i. In conjunction with lung count.
j. In conjunction with thyroid count.
k. No gamma with energy greater than 100 keV with sufficient abundance on which to base the estimate.
l. 50 nCi for Ra-226; 4 nCi based on Pb-241 in equilibrium with Ra-226.
m. Based on Th-227 in equilibrium with Ac-227.
n. Based on Ra-224 in equilibrium with Th-228.
o. Based on Ac-228 in equilibrium with Th-232.
p. Pa-233 is used to measure Np-237 by whole-body count; 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.

Consistent with industry-wide improvements in this area, a variety of hardware and software applications have been used for whole-body counting. In 1993, workers in the routine bioassay program received employment, annual, and termination WBCs. Workers who received routine bioassays included RCTs 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. While an MDA was listed, 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; these should be used when they are available. The ^{137}Cs WBC sensitivity for 1968 to 1991 was 0.011 μCi for a 20-minute count and 0.010 μCi for a 40-minute count. The MPBB of ^{137}Cs in standard man is 30 μCi.

A.2.2 Chest Counting

Based on Allen et al. (1993), lung counts occurred as soon as practicable after a suspected intake of thorium, uranium, or a TRU element. The first mention of lung counts in REECo procedures was in July 1983 (REECo 1983). 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 A-10.

Table A-10. 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-237	0.04
Ce-144	0.2	U-238[f]	9(0.4)
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	Am-243	0.01

Source: Allen et al. (1993).

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 based on the detection of the 16-keV photon. When determined from Th-234, the MDA for U-238/Th-234 is 0.4 nCi based on the 63-keV photon from thorium, assuming equilibrium. [Note: Allen et al. (1993) lists 0.4 nCi in the text (p. 49) and 0.04 nCi in the summary table (Table 4.1)].
g. Based on Am-241 measurement.

A.2.3 Thyroid Counts

Based on Allen et al. (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 A-11.

Table A-11. 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: Allen et al. (1993).

Allen et al. (1993) states that radionuclides such as ^{132}Te, ^{131}I, ^{133}I, and ^{135}I 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 ^{132}Te, 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 A-26 later in this attachment lists tests and incidents for which DRs 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 DR 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 within 1 month of the test or was present during an incident or immediately after a test. From these records if available, the DR 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 5-8.

Without specific access information for a worker, the DR should base internal exposure on environmental levels. Radionuclides listed in Table A-11 should be compared to the lists in Section A.4 that most closely characterize the type of test the worker was supporting, and the DR must decide which radionuclides are most significant to dose based on the cancer location. The list of radionuclides of concern varies according to the types of event and time interval before the worker was present. The DOE records might contain access records for test support (e.g., reentry, mine-back, drill-back) including the times the worker entered and exited the controlled areas. If there are no specific access data, short-lived radionuclides will not be a concern. Exceptions to this will be workers who can be placed in the area where an uncontrolled venting occurred. Table A-26 lists releases from underground testing that DRs can review when determining if the worker was in the area during one of these releases.

In addition, DRs must consider the timeframe. DRs can determine when the worker was present from the "Other Monitoring" section in the DOE records. This section includes NTS Access Records for the worker's reentry into an area after an event. If there are no "Other Monitoring" data, short-lived radionuclides are not a concern. Exceptions are workers who can be placed in the area where an uncontrolled venting occurred. Table A-23 in Section A.3.1.19 lists releases from underground testing that DRs can review when determining if the worker was in the affected area during one of these releases.

A.3 AREA AND ASSOCIATED RADIONUCLIDES OF CONCERN

A.3.1 Radionuclides of Concern and Specific Bioassay Programs for Facilities

The information in the following paragraphs is from the 1993 NTS Technical Basis (Allen et al. 1993), Chapter 7, Facility Descriptions and their Specific Routine Bioassay Programs. The NTS Technical Basis 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 2004). 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. NOTE: Many radionuclides listed in the following paragraphs were not specifically measured by the bioassay program. Routine bioassay analysis was done for the following radionuclides:

- ^{241}Am: Lung count
- Tritium: Urine, wound swabs
- Gamma emitters: Urine, WBC, wound swab
- Gross alpha: Wound swab
- GFP (gross beta): Urine, wound count/swab
- Iodines: Thyroid count
- ^{90}Sr: Urine
- ^{238}U: Lung count
- ^{238}Pu: Urine, feces, lung count, wound count
- ^{239}Pu: Urine, feces, lung count, nasal swabs, wound count

A.3.1.1 Drill-Backs (Yucca Flat and Pahute Mesa, or Areas 1-10 and 18-20)

Drill-back operations took place within days or weeks of an underground nuclear weapons test. Initial reentry surveys in the radiological exclusion area up to the trailer park usually began within 1 hour after a test and the duration of the survey was typically about 1 hour depending on logistics (test location, weather, etc.) or whether there were identified problems. Initial reentry surveys were conducted by the REECo Rad-Safe Monitors. Typically, post-test drilling would begin within 1 to 4 days after a test, but could begin more than a week after a test due to a variety of factors (containment success, initial survey results, test location, test type, weather, schedule, etc.). The duration of the post-test drilling operation was typically 2 to 6 days, but due to the factors mentioned above, plus concerns such as circulation problems, sampling issues, hole integrity, etc., a drill-back could last 2 to 3 weeks or more. Specific information about a particular drill-back might be available in the DOE records. 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 1964, 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. Table A-12 lists radionuclides for identification of a problem from resuspension and used to trigger searches in bioassays for the radionuclides of concern for resuspension. Table A-13 lists radionuclides of concern from drill-back resuspension.

Table A-12. Drill-back resuspension and mine back containment loss radionuclides for identification versus time after test. (Allen et al. 1993)

1 day	10 days	100 days	365 days	10,000 days
Na-24	Na-24[a]			
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				

a. Added in response to Sanford Cohen & Associates comment.

Table A-13. Drill-back resuspension, reentry/mine back resuspension, and decontamination facility, isotopes of concern for dose versus time after test (Allen et al. 1993).

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	

1 day	10 days	100 days	365 days	10,000 days
I-133	Te-132	Ce-144		
I-135	Ce-141			
Ce-143	Ce-144			
Ce-144				

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 gamma-emitting radioactivity and for radionuclide identification. The primary isotopes used for identification of an intake were ^{131}I, ^{133}I, and ^{137}Cs, and those listed in Table A-12. Annual WBCs were conducted on a routine basis for specific radiation workers, and on a special basis after 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

Other historical job titles that might appear in the records include Rough Neck and Equipment Oiler. The use of engineering controls such as the blowout preventer was incorporated to contain radioactive gases in 1964. DRs will need to evaluate the potential for exposure based on review of the access records provided in the DOE claim file before 1964. ORAUT (2008) describes drill-back operations including potential exposures.

A.3.1.2 Reentry and Mine Back (Area-1 & Area-12)

Reentry and mine back operations were similar to those for drill-backs. The major differences were 1) the time after a test that the cavity was entered, 2) the horizontal method of drilling, and 3) the fact that operations took place in a confined underground environment. Additionally, in some cases, the cavity was opened and personnel entered.

Initial (surface) reentry surveys in the radiological exclusion area (from a preassigned location in the forward area (e.g., Gate 300) up to and including the tunnel portal yard and the Rainier Mesa Trailer Park) usually began within 1 to 2 hours of a test and the duration of the survey was typically about 1 hour depending on the test location or if problems were identified. Initial surface reentry surveys were conducted by REECo Rad-Safe Monitors. Once the initial (surface) reentry survey was completed, remote gas sampling was conducted and the initial underground reentry began.

The initial underground reentry usually began 1 day after the test and the duration varied depending on various factors (tunnel condition, containment success, radiological readings, etc.). Multiple reentries and concurrent work activities were typically conducted until the Test Director declared the

official end of the reentry phase. Teams with different missions, but with some overlapping duties, were involved in the underground reentry, including:

- Work Teams (miners and an RCT) assessed tunnel integrity and performed tunnel safety inspections, removed doors and panels, reinstalled train track, reestablished ventilation, conducted radiation surveys, and collected samples up to a preestablished location. This was the first team to enter the tunnel after the test.

- Initial Reentry Teams, Reentry Teams, Mine Rescue Teams (miners, Industrial Hygienists, RCTs, and other crafts as needed) advanced the reentry into the potentially contaminated area. These workers collected gas samples; performed radiation surveys; established utility services; opened crawl tube doors, panels, and valves; reestablished ventilation; installed security grates; and assessed tunnel integrity. These teams were typically the first to encounter radiological exposure and wore appropriate contamination control PPE (e.g., anticontamination clothing and self-contained breathing apparatus) as needed.

- Scientific Teams - Data Recovery Team, Special Assessment Team, Scientific Assessment Team (national laboratory personnel plus RCTs) collected experiments and data. Once the reentry teams surveyed and cleared the retrieval areas, these teams typically entered with minimal or no required contamination control PPE. Multiple entries were required to retrieve the required instrumentation.

Mining through barriers (e.g., gas seal plug, drift protection plug, overburden plug) could begin within 2 or 3 days of a test, but could be delayed a month or more after the test depending on conditions. National laboratory and DNA personnel continued their photography and recovery efforts as the mine-back progressed. Radiological survey support continued during each phase of the recovery period. The mission of the scientific teams was to enter the line-of-sight (LOS) pipe and collect experiments/ data from the test cells. The duration of the mine-back varied depending on test results and schedule.

Post-test drilling from the surface ground zero usually began 1 to 2 months after the tunnel test. Drilling was mentioned in the onsite radiological reports in the 1960s and early 1970s. It apparently did not occur after about 1974.

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 LOS pipe was opened to remove experiment equipment and samples. Underground operations and construction included drilling, mining and mucking, haulage and transportation, rock bolting, ground control and timbering, ventilation, grouting (applying cement as containment material), welding, electrical and utilities support, and barring down (inspection for loose ground). These operations used unique equipment including continuous Alpine mining machines, (load/haul/dump) mucking machines (small front-end loaders to remove mined material), dinky locomotives (train engines) and cars used to move personnel and materials (e.g., Moran cars for cement and grout, powder cars for explosives, high cars for elevated work), shotcrete machines [equipment to apply shotcrete or fibercrete, which is a cement-

like material imbedded with metal wire and applied to the tunnel back and ribs (ceiling and walls) for ground control], jackleg and Jumbo drills, and spaders (similar to a jackhammer).

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 ^{3}H, ^{131}I, ^{133}I, and ^{137}Cs. Radionuclides for identification of a problem from containment loss and resuspension are listed in Table A-12. Radionuclides of dose concern from resuspension are listed in Table A-13. Radionuclides for identification of a problem from fissures and LOS pipe opening were ^{3}H, ^{7}Be, ^{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 ^{3}H, ^{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 performed 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 ^{3}H. Annual whole-body counts were conducted on a routine basis for specific radiation workers and on a special basis after any situation for which intake was considered likely.

A.3.1.3 Routine Tunnel Operations (Areas 1, 12, 15, and 16)

Tritium was the isotope of dose concern; routine bioassay was to collect quarterly urine samples which were processed for ^{3}H. (Allen et al. 1993 contains no bioassay information.) See Section 5.2.2.4 for a discussion on assigning tritium dose.

A.3.1.4 Decontamination Facility (Area 6)

Table A-14 lists radionuclides used for identification of a problem at this facility. Table A-13 lists radionuclides of concern. In addition to the radionuclides listed in Table A-11, BN (2000) lists the following radionuclides of concern for this facility: ^{232}Th, ^{234}U, ^{235}U, ^{238}U, ^{239}Pu, and ^{241}Am. The routine bioassay method was to collect quarterly urine samples and conduct annual WBCs. Urine samples were gamma-counted and analyzed for GFP, ^{238}Pu, ^{239}Pu, ^{240}Pu, and ^{241}Am.

A.3.1.5 Test Treatability Facility (Area 25)

The Test Treatability Facility was a pilot project to bench-test technologies to be used for decontamination of soils containing TRU materials. The concentrations of radionuclides in the soils were not intended to exceed a few picocuries per gram of soil. Radionuclides 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.

Table A-14. 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				

A.3.1.6 Atmospheric Weapon Safety Tests (Area 3 and Area 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.

A.3.1.7 Atmospheric Weapons Test Areas (all Areas except Area 22, Area 23, and Area 27 due to test location or airborne dispersion)

Table A-15 lists radionuclides for identification of a problem and dose concern. Routine bioassay was to collect quarterly urine samples and conduct annual whole-body counts. Some area monitoring personnel also had periodic lung counts. Urine samples were analyzed for gamma-emitting radionuclides, GFP, plutonium, and americium.

Table A-15. Atmospheric weapons test areas, isotopes for identification and of concern for dose (1993–present).

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				

A.3.1.8 Low-Level Waste Site (Area 3)

Table A-16 lists radionuclides for identification of a problem and dose concern. In addition to these radionuclides, BN (2000) lists ^{235}U as a radionuclide of concern for this facility. 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 A-16. Low-level waste site (Area 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			

A.3.1.9 Low-Level Waste Site (Area 5)

Table A-17 lists radionuclides for identification of a problem and dose concern. Routine bioassay was to collect quarterly urine samples and conduct annual WBCs. Urine samples were analyzed for gamma emitters, sampled for ^{3}H, and analyzed for GFPs, plutonium, and americium.

Table A-17. Low-level waste site (Area-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		

A.3.1.10 Radiation Instrument Calibration Facilities (Area 6 and Area 23)

Table A-18 lists radionuclides for identification of a problem and dose concern. In addition to these radionuclides, BN (2000) lists ^{235}U as a radionuclide of concern for these facilities. 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 ^{3}H 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 A-18. 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

A.3.1.11 Radiography Operations (Area 23)

The isotope of concern was ^{192}Ir. In addition to ^{192}Ir, BN (2000) lists ^{60}Co and ^{137}Cs as radionuclides of concern for this operation. Bioassay (urine/WBC) 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.

A.3.1.12 Well Logging Operations (Areas 1–10 and Areas 18–20)

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/WBC) was conducted if there was a loss of source containment detected by swipes or other analysis. Urine samples were gamma-counted or processed for plutonium depending on the source type. Additional "Nuclides of Concern" listed in McMahan and Ogurek (2003) include the following well logging sources: ^{241}Am-^{7}Be, ^{238}Pu-^{7}Be, and ^{226}Ra-^{7}Be.

A.3.1.13 Nuclear Explosive and Device Assembly Facilities (Area 6 and Area 27)

Radionuclides for identification of a problem and dose concern were ^{3}H, ^{235}U, ^{239}Pu, and ^{240}Pu. In addition to these radionuclides, BN (2000) lists ^{234}U, ^{238}Pu, ^{241}Pu, and ^{241}Am as radionuclides of concern for these facilities. No specific bioassay information was listed in Allen et al. (1993), BN (2000) or McMahan and Ogurek (2003) technical basis documents.

A.3.1.14 Nuclear Rocket Development Areas (Area 25 and Area 26)

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. In addition to these radionuclides, BN (2000) lists ^{234}U, ^{235}U, and ^{238}U as radionuclides of concern for these areas. Routine bioassay was to collect quarterly urine samples and conduct annual WBCs. Urine samples were gamma-counted, sampled for ^{3}H, and analyzed for GFP and plutonium.

A.3.1.15 Radioactive Source Storage Areas (Area 6 and Area 23)

Table A-19 lists radionuclides for identification of a problem and dose concern. 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 WBCs and/or urine sample collection and analysis were conducted. Specific bioassays and analysis depended on the source leaking.

Table A-19. 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		

A.3.1.16 Radiochemistry and Counting Laboratories

Table A-20 lists radionuclides for identification of a problem and dose concern. In addition to these radionuclides, BN (2000) lists ^{226}Ra, ^{232}Th, ^{234}U, and ^{238}U as radionuclides of concern for these facilities. Routine bioassay was to collect quarterly urine samples and conduct annual WBCs only for those personnel routinely handling uncontained radioactive materials. Urine samples were gamma-counted, sampled for ^{3}H, and analyzed for GFP and plutonium.

Table A-20. 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

A.3.1.17 Other Facilities Identified After 1992 (the End of Testing)

McMahan and Ogurek (2003) included some facilities and radionuclides of concern that Allen et al. (1993) did not describe. The following paragraphs discuss these newer facilities and Site Monitoring Services (SMS)/Radioactive Material Control (RAMATROL):

- Big Explosive Experimental Facility (BEEF) (Area 4) – Most experiments at BEEF involve the use of high explosives. On occasion, depleted uranium or tritium could be used during an experiment. After detonation, particulates occur in the area. The nuclides of concern are ^{3}H, ^{234}U, and ^{238}U.

- Waste Examination Facility (Area 5) – The WEF processes TRU waste drums in preparation for shipment to the Waste Isolation Pilot Plant. The drums are primarily from LLNL, but a small number are from LANL and Rocky Flats. The TRU waste is in 55-gal drums and processed in a glovebox. There is a storage area for the drums called the TRU Pad. The potential for internal exposure can come from a failure of glovebox integrity. Nuclides of concern at WEF are ^{233}U, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Am, ^{243}Am, ^{244}Cm, and ^{252}Cf. Plutonium radionuclides and ^{241}Am have been identified as the predominant nuclides of concern.

- SMS and RAMATROL – Personnel assigned to SMS and RAMATROL provide radiological survey services mainly on NTS but also at other sites as assigned. These services include contamination surveys of equipment and materials being removed from NTS, contamination surveys of equipment and buildings determined to be excess to government inventory, and surveys of radiological shipments entering and departing NTS. In addition, these personnel perform surveys of Radioactive Material Areas and of areas where handling of swipe samples

and other potentially contaminated material occurs. The most common radionuclides these personnel encounter are aged fission products (e.g., ^{137}Cs and ^{90}Sr/^{90}Y), ^{3}H, uranium, ^{239}Pu, and ^{241}Am. SMS and RAMATROL personnel could encounter these nuclides as removable or fixed contamination, typically with removable contamination less than 200 dpm/100 cm^2 for beta emitters and 20 dpm/100 cm^2 for alpha emitters.

A.3.1.18 Summary Lists of Radionuclides of Concern

Table A-21 summarizes radionuclides of dose concern for NTS. Table A-22 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 A-21. Radionuclides 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	Hg-203	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 A-22. Current nuclides of concern for NTS locations.

NTS facility and area	Radionuclides of concern			
BEEF[a]	U-234	U-238	H-3	
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 munitions 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			

NTS facility and area	Radionuclides of concern			
	Ra-226	U-235	Pu-238	Pu-239
	Am-241			
Radiation instrument calibration facilities	H-3	Co-60	Sr-90	Cs-137
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[b]
	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
WEF	U-233	Pu-238	Pu-239	Pu-240
	Am-241	Am-243	Cm-244	Cf-252
SMS and RAMATROL	H-3	Sr/Y-90	Cs-137	U-234
	U-235	U-238	Pu-239	Am-241

Source: McMahan and Ogurek (2003).
a. BN (2000) also lists H-3 as a radionuclide of concern.
b. Identified by Ogurek (2004) as a radionuclide of concern.

A.3.1.19 Atmospheric Testing Era

The following information is included for completeness. Based on the determination of Health and Human Services, internal dose cannot be included for employment during the atmospheric testing era at either NTS or PPG (Leavitt 2006a,b).

Between June 1946 and October 1958, the United States used Enewetak and Bikini Atolls, part of the Marshall Islands approximately 2,500 nautical miles southwest of the Hawaiian Islands, as testing grounds for 67 nuclear devices. Some NTS workers might have been stationed in the Pacific for these tests. Materials associated with the nuclear tests conducted at or near Bikini and Enewetak Atolls include ^{210}Po, ^{228}Th, ^{230}Th, ^{232}Th, ^{233}U, ^{238}U, ^{241}Am, ^{242}Cm, and the fission product ^{137}Cs (Robison et al. 2001). Radionuclides detected in the marine (lagoon) environment (plankton, fish, invertebrate, seawater, and sediments) from studies conducted from 1954 through the early 1970s included ^{3}H, ^{14}C, ^{54}Mn, ^{55}Fe, ^{57}Co, ^{58}Co, ^{60}Co, ^{63}Ni, ^{65}Zn, ^{90}Sr, ^{95}Zr, ^{99}Tc, ^{102m}Rh, ^{106}Ru, ^{110m}Ag, ^{144}Ce, ^{147}Pm, ^{151}Sm, ^{155}Eu, ^{126}Sb, ^{137}Cs, ^{184}W, ^{207}Bi, ^{237}Np, ^{237}U, $^{239+240}$Pu, ^{241}Pu, and ^{244}Cm (Robison and Noshkin 1998). The limits found in NBS Handbooks 52 (NBS 1953) and 69 (NBS 1959) would have been in effect. SAIC (1985) lists the following radionuclides for Pacific testing: 148 fission products, 17 activation products, and 18 actinide elements. The 148 Pacific testing fission products are as follows:

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Table A-23. Fission products associated with above ground testing.

Se-79	Y-92	Mo-105	Sb-129	Cs-134m	Pr-144
Br-82	Sr-93	Tc-105	Te-129	I-135	Pr-144m
Br-83	Y-93	Ru-105	Te-129m	Xe-135*	Ce-145
Kr-83m*	Zr-93	Rh-105	I-129	Xe-135m*	Pr-145
Br-84	Nb-93m	Rh-105m	Sn-130	Cs-135	Ce-146
Br-85	Y-94	Ru-106	Sb-130m	Cs-136	Pr-146
Kr-85*	Y-95	Rh-106	I-130	Xe-137*	Ce-147
Kr-85m*	Zr-95	Pd-107	Sb-131	Cs-137	Pr-147
Rb-86	Nb-95	Rh-109	Te-131	Ba-137m	Nd-147
Kr-87*	Y-96	Pd-109	Te-131m	Xe-138*	Pm-147
RB-87	Zr-97	Ag-109m	I-131	Cs-138	Pr-148
Kr-88*	Nb-97	Ag-110m	Xe-131m*	Cs-139	Pm-148
Rb-88	Nb-97m	Ag-111	Sb-132	Ba-139	Pm-148m
Kr-89*	Nb-98m	Cd-113m	Sb-132m	Cs-140	Pm-149
Rb-89	Mo-99	Cd-115m	Te-132	Ba-140	Pm-151
Sr-89	Tc-99	Sn-123	I-132	La-140	Sm-151
Rb-90	Tc-99m	Sb-124	Sb-133	Ba-141	Eu-152
Rb-90m	Mo-101	Sn-125	Te-133	La-141	Sm-153
Sr-90	Tc-101	Sb-125	Te-133m	Ce-141	Eu-154
Y-90	Mo-102	Te-125m	I-133	Ba-142	Eu-155
Rb-91	Tc-102	Sn-126	Xe-133*	La-142	Eu-156
Sr-91	Tc-103	Sb-126	Xe-133m*	La-143	Tb-160
Y-91	Ru-103	Sb-127	Te-134	Ce-143	Ho-166m
Y-91m	Rh-103m	Te-127	I-134	Pr-143	
Sr-92	Tc-104	Te-127m	Cs-134	Ce-144	

* = gaseous fission product.

Table A-24. Activation products associated with above ground testing.

Be-7	Co-58	W-187
Na-24	Co-60	W-188
Mn-54	Cu-64	Au-198
Fe-55	Cu-67	Au-199
Fe-59	W-181	Pb-203
Co-57	W-185	

Table A-25. Actinide elements associated with above ground testing.

Th-229	**U-237**	**Pu-238**
Th-230	U-238	Pu-239
Th-234	U-240	Pu-240
U-233	Np-237	Pu-241
U-234	Np-239	Am-241
U-235	Np-240	Cm-242

A.3.2 Incidents

Based on guidance from NTS, the DR 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* (Hawthorne 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 there is a link between the claimant and a release by timeframe, work location, and job title using ORAUT (2008) and Section A.4.1 of this document.

In addition, the DR might find useful the information in Table A-26 [from *The Containment of Underground Nuclear Explosions* (OTA 1989)] and Schoengold, DeMarre, and Kirkwood (1996) 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 (see Table A-26):

- AEC (1956), Part VI, Chapter 1, "Special Incidents," describes an overexposure during a plutonium dispersion test. In January 1956, a work party was dispatched into the "hot" area to recover vital laboratory samples according to a plan that had been used successfully on earlier operations and for the first three tests of 56 Project – NTS. In performing this hazardous task, members of the party made three errors that resulted in four men receiving technical overexposures varying from 4.3 R to 28 R. An inquiry by the Test Director after the operation determined the causes of the overexposure. Routine Rad-Safe procedures in effect at NTS were amended to reduce the possibility of a reoccurrence of the errors committed.

- The *Operation Plumbbob On-Site Radiological Report* discusses four instances of possible internal exposure (REECo 1957):

  - Escape of radioactive gas from the Tower 2-A cab contaminated the working area, resulting in 12 workers being exposed.

  - One worker removed his respirator while working in an area highly contaminated with alpha-emitting material.

  - Several workers were exposed while removing and cutting a cable highly contaminated with alpha-emitting material.

  - Four workers without respirators entered a tunnel that was highly contaminated with alpha-emitting materials.

- The Hardtack II report (Ponton et al. 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, including air supplied respirators, the taking of nasal swabs, and full anticontamination clothing (clothing that might have consisted of coveralls, hoods, booties, and gloves).

Table A-26. Releases from underground tests.

Test name	Date	Type	Purpose & yield	Sponsor	Location	Radionuclides identified In release	Release type & notes	Release (Ci)	Release summary
Platte	4/14/62	Tunnel	Weapons-related 1.85 kt	LLNL	U12k.01	K-40, Zr-95, Nb-95, Ru-103, Ru-105, I-131, I-133, I-135, Te-132, Ba-140/La-140, Ce-141, Ce-144	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	Zr-95/Nb-95, Ru-103, Ru-106, Rh-105, I-131, I-133, I-135, Te-132, Ba-140/La-140, Ce-141, Ce-144	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.

Test name	Date	Type	Purpose & yield	Sponsor	Location	Radionuclides identified In release	Release type & notes	Release (Ci)	Release summary
Des Moines	6/13/62	Tunnel	Weapons-related 2.9 kt	LLNL	U12j.01	Ru-103, Ru-106/Rh-106, I-131, I-133, I-135, Te-132, Ba-140/La-140	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	GFPs Isotopic analysis[e]: Mo-99, I-131, I-132, Te-132	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: Xe-133, Xe-135 Controlled: I-131, I-133, I-135	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.

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Test name	Date	Type	Purpose & yield	Sponsor	Location	Radionuclides identified In release	Release type & notes	Release (Ci)	Release summary
Diagonal Line	11/24/71	Shaft	Weapons effects <20 kt	LLNL DOD	U11g	Kr-85m, Kr-87, Kr-88, I-131, I-132, I-133, I-135, Xe-131m, Xe-133, Xe-133m, Xe-135	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 Xe-135 (80-85%), Kr-85m, Kr-87, Kr-88, Xe-131m, Xe-133, and Xe-133m, with trace quantities of I-131, I-132, I-133, and I-135 detected. Minor levels of radioactivity were detected offsite by aircraft only.

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.

Test name	Date	Type	Purpose & yield	Sponsor	Location	Radionuclides identified In release	Release type & notes	Release (Ci)	Release summary
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-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.

Test name	Date	Type	Purpose & yield	Sponsor	Location	Radionuclides identified In release	Release type & notes	Release (Ci)	Release summary
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. Before 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. Before 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. Before 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.

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.

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Test name	Date	Type	Purpose & yield	Sponsor	Location	Radionuclides identified In release	Release type & notes	Release (Ci)	Release summary
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. Before 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.

Test name	Date	Type	Purpose & yield	Sponsor	Location	Radionuclides identified In release	Release type & notes	Release (Ci)	Release summary
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.

Test name	Date	Type	Purpose & yield	Sponsor	Location	Radionuclides identified in release	Release type & notes	Release (Ci)	Release summary
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.[f]
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	6/20/87	Tunnel	Weapons	LANL	U12t.09	Kr-85	Controlled tunnel	Controlled	Activity was contained in cavity

Test name	Date	Type	Purpose & yield	Sponsor	Location	Radionuclides identified in release	Release type & notes	Release (Ci)	Release summary
Ghost			effects <20 kt	DOD			purge (onsite only)[c]	release at time of release: 3.0 Kr-85: 3.0	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 A-26 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 (Schoengold, DeMarre, and Kirkwood 1996) and are not necessarily the isotopes of concern for workers.

a. Containment failures are normalized to 12 hour after the test.
b. The camphor failure includes Cs-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 (AEC 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 Xe-133, 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).

- A memorandum for B Tunnel, U12b (Johnson 1962), 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. Twenty-seven of these employees were subjected to internal radiation greater than 3,000 mrem/yr. The original source of tritium was from 1958 activities (Arent and Smith 2004).

  The *Operation Storax On-Site Radiological Safety Report* (McClendon and Eubank 1964) discusses internal radiation hazards for 1962 and 1963, which includes the YUBA test conducted at U12b.10 on June 5, 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-test 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 detected off the site from U12b (B Tunnel). In summary, nine REECo personnel received thyroid doses in excess of 30 rads during YUBA (Arent and Smith 2004).

- *Operation Whetstone, On-Site Radiological Report* (Eubank 1965) for 1964 and 1965 describes exposure to internal radiation up to NTSO-0524-01 limits (AEC 1961). Two post-test 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. In a collection of papers entitled *REECo Historical Radiation Records, Operation Whetstone (July 1964 – June 1965), #20 Merlin Event, U3CT* a report entitled *Report of Investigation of Thyroid overexposure - U3ct, Type "B" Incident* (REECo 1965) also describes the iodine incident resulting in doses of 27 and 31 rem (^{131}I and ^{133}I, respectively) to the thyroid for two employees involved in post-test drilling. In all, six employees received measurable thyroid radiation exposures. A Type B Investigation was conducted (REECo 1965).

- Extensive records for the 1970 Baneberry test are available in the site research database. These include summary reports from the DOE Nevada Operations Office (DOE-NV), REECo radiological safety reports and procedures, and records of monitoring and decontamination efforts (REECo 1973). On December 18, 1970, the Baneberry underground nuclear test at 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 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. After Baneberry, new containment procedures were adopted to prevent a similar occurrence.

- Evaluation of Protection, Bioassay, and Dose Assessment Programs for Internal Radiation Exposures at 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 (^{3}H) during the mid–1990s. The three situations included:

- 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. The activity of ^{239}Pu in fecal samples of 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. The bioassay data from E Tunnel workers were limited in the quantity and quality needed for making reasonable accurate estimates of their intakes and doses.

- The exposure of one worker in June 1995 to ^{239}Pu aerosols, believed to have been generated during the handling and characterization of high-specific-activity fragments found during the Double Track Site soils characterization project. French and Skrable (1995) estimated the worker at the Double Track Site had an effective ingestion intake of 3,260 pCi (^{239}Pu aerosol) from nonrespirable aerosols with a committed effective dose of 0.2 mrem, and an inhalation intake of 360 pCi of 1-µm activity median aerodynamic diameter class Y aerosols with a committed effective dose of 90 mrem.

- 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 (1995) concluded that considerable bioassay and air sampling data confirmed that the exposures were minimal with little dose consequence, and provided no further dose estimates. In 1995, 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 an EG&G scientist. This modification included breaching the sealed tube that contained three tritide disks (130 Ci tritium total). According to the *Annual Site Environmental Report for Calendar Year 1995* (Black and Townsend 1996), the maximum release into the building was 123 mCi.

A.3.3 Respiratory Protection Practices

The following paragraphs provide examples of NTS respiratory protection practices. Historical accounts of respirator usage are included to give the DR 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 1952) 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 57 was an atmospheric safety experiment that studied plutonium dispersal in NTS Area 11 from 1955 to 1956. AEC (1956) stated that respirators or “full-face assault masks” were required for persons entering the 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 (Nielsen 1957) 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 memorandum lists measured contamination levels on PPE (up to 10^6 alpha cpm) and includes individual nasal swipes results (sanitized). A second memorandum (Bybee 1957) details the August 30, 1957, reentry into U12c during which team members wore respirators and MSA 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" (Wilcox 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/55cm^2 in the main drift from Drift 01, and 2 × 10^3 c/m/55cm^2 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 Wilson 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."

Atomic Weapon Accidents – 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 × 10^3 dpm/55cm^2. Full-face masks equipped with dust, fume, and mist filters were worn, and nasal swabs were collected at the exit of the CA. If nasal swab results were >200 dpm (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} µc/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-hour 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 (Rainier 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 would 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 A-27 lists NTS respiratory protection levels for selected years (Author unknown undated a).

A.3.4 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 stated REECo collected the following samples: 4,230 nasal swabs, 109 facial swipes, and 1,945 air samples. In addition, 73 urine kits and 10,755 respirators were issued.

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Table A-27. Historical NTS respiratory protection action levels.

Year	Respiratory protection action levels
1957	No respirator ≤9.6E+3 dpm/m^3 hours (alpha) per quarter year
	No respirator (short periods) ≤100 dpm/m^3 (alpha)
	Respirator required >100 dpm/m^3 (alpha)
	No respirator ≤2.2E+4 dpm/m^3 (beta + gamma)
	Respirator required >2.2E_4 dpm/m^3 (beta + gamma)
	Note: Respirators using filters have an efficiency up to 99.9% for particles as small as 0.3 µm 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.
	4E10-11 µCi/cm^3 (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

The Defense Nuclear Agency (DNA) report on the 1957 Plumbbob Operations (Harris et al. 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.

Radiological Safety for Underground Nuclear Explosions (Wilcox and Coogan 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 × 10^3 dpm (beta, total both nostrils) or 2 × 10^2 dpm (alpha, total both nostrils) were requested to submit urine samples. These samples were processed in the laboratory to determine body burdens of GFPs or alpha emitters.

"Procedures to Limit Radiological Exposures of Miners" (Reeves 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 A-28 lists NTS contamination limits for selected years. Contamination limits were often used as indicators of when bioassay samples were to be taken.

Table A-28. 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 coveralls are >1E+3 dpm/55 cm^2. Decontaminate personnel: • Outer clothes: >500 cpm/55 cm^2 (alpha) or 7 mR/hr (beta + gamma) • Shoes: >500 cpm/55 cm^2 (alpha) or 7 mR/hr (gamma) • Skin or underclothing: >100 cpm/55 cm^2 (alpha) or 1 mR/hr (gamma) • Equipment: <7 mR/hr (gamma) or 500 cpm/55 cm^2 (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 decontamination. Access permits required to enter areas with beta-gamma levels >10 mR/hr or 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: Author unknown (undated a).

A.4 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. DRs should assume that acute exposure occurred within one work shift, when single bioassay results are included in the DOE records for the worker. 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 records provided by 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 A-26).

Note: If the file provided by DOE does not include bioassay, DRs should base the internal dose assigned to the worker on the environmental ambient.

Tables A-29 through A-31 provide information about the characteristics of the source term at NTS. The DR needs to remember that this source term changes over the years based on the test category.

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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 DR might have to address. The DR 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 A-29. Inhalation classes for radionuclides found at NTS.

Radionuclide(s)	Allen et al. 1993	BN 1998	BN 2003
Ac-227	D	-	-
Am-241	W	W	W
Am-243	W	-	-
Ba-133	D	-	-
Ba-140	D	-	-
Ce-139	Y	-	-
Ce-141	Y	-	-
Ce-143	Y	-	-
Ce-144	Y	-	-
Cs-134	D	-	-
Cs-137	D	D	D
Co-57	Y	-	-
Co-60	Y	W, Y	
Cm-244	W	-	W
Eu-152	W	-	-
Eu-154	W	-	-
Eu-155	W	-	-
I-131	D	D	D
I-132	D	-	-
I-133	D	-	-
Ir-192	Y	-	-
Lu-174	W	-	-
Mn-54	W	-	-
Mo-99	Y	-	-
Na-22	D	-	-
Na-24	D	-	-
Nb-95	Y	-	-
Np-237	W	-	-
Pu-238	W[a]	W, Y	W, Y
Pu-239	W[a]	W, Y	W, Y
Pu-240	W[a]	-	-
Pu-241	W[a]	-	-
Pu-242	W[a]	-	-
Ra-226	W	W	W
Rh-101	Y	-	-
Rh-102m	Y	-	-
Ru-103	Y	-	-
Ru-106	Y	-	-
Sb-123	W	-	-
Sb-125	W	-	-
Sr-85	Y	-	-
Sr-90 [(d)]	Y	D, Y	D, Y

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Radionuclide(s)	Allen et al. 1993	BN 1998	BN 2003
Ta-182	Y	-	-
Te-132	D	-	-
Th-228	W	-	-
Th-230	W	-	-
Th-232	W	-	W, Y
U-233	W	-	-
U-234[b]	W, Y[c]	D, W or Y	D, W or Y
U-235[b]	W, Y	D, W or Y	D, W or Y
U-238[b]	W, Y	D, W or Y	D, W or Y
Zr-95	D	-	-

a. Class W was assigned for all plutonium radionuclides based on Pu-240, which has the lowest annual limit on intake (Allen et al. 1993).
b. Class D was listed in the source documents and may be applicable to personnel working in laboratories with soluble forms of uranium.
c. Assumed based on information for U-235/238.
d. For X-10, guidance has been provided to the DRs indicating that Type Y (current Type S) should only be assigned when strontium titinate is identified, if this compound cannot be confirmed then Type D (current Type F) should be assumed.

Table A-30. Fission products up to 1 year old as identified in 1954 documentation.

Radionuclide	Half-life	Beta energy (MeV)
Strontium-89	53 days	1.463
Strontium-91	9.7 hours	2.665 - 0.62
Strontium-90 Yttrium-90	19.9 years 61 hours	0.61 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

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Radionuclide	Half-life	Beta energy (MeV)
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

Source: ORDOSE (2003).

Table A-31. 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

Source: ORAUT (2010a).

NOTE: The International Commission on Radiation Protection (ICRP) Publication 30 (ICRP 1979) respiratory tract model describes the relative rate of clearance from the pulmonary region of the lungs. There are three defined inhalation classes for materials: D (clearance half-time less than 10 days), W (10 to 100 days), and Y (greater than 100 days). (Allen et al. 1993 defines D as days, W as weeks, and Y as years.)

The ICRP Publication 66 (1994) respiratory tract model describes the relative speed of dissolution and translocation to blood of material within the respiratory tract and is related to the physical chemistry of the material. There are three defined absorption types: F (fast), M (moderate), and S (slow). For the purpose of this project, the DR should assume that D is equivalent to F, M is equivalent to W, and S is equivalent to Y.