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Subject Expert(s): Brian P. Gleckler, and Mutty M. Sharfi

Approval:	Signature on File Brian P. Gleckler, Document Owner	Approval Date:	03/29/2011
Concurrence:	Signature on File John M. Byrne, Objective 1 Manager	Concurrence Date:	03/29/2011
Concurrence:	Signature on File Edward F. Maher, Objective 3 Manager	Concurrence Date:	03/29/2011
Concurrence:	Signature on File Kate Kimpan, Project Director	Concurrence Date:	03/29/2011
Approval:	Signature on File James W. Neton, Associate Director for Science	Approval Date:	04/01/2011

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PUBLICATION RECORD

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04/01/2011	01	This technical basis document was predominantly revised to address SC&A's issues with the document, as identified in SCA-TR-TASK1-0015. The changes that potentially affect the assessed doses for workers include: 1) the addition of an approach for assessing potential exposures to insoluble forms of tritium and 2) the modification of some of the default tritium urinalysis MDCs and/or reporting levels. In addition, a number of editorial changes were also made to this document. Incorporates formal internal and NIOSH review comments. Constitutes a total rewrite of the document. Training required: As determined by the Objective Manager. Initiated by Brian P. Gleckler.

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ACRONYMS AND ABBREVIATIONS

AMAD	activity median aerodynamic diameter
AEC	U.S. Atomic Energy Commission
CF	calibration factor
CFR	Code of Federal Regulations
cm	centimeter
cpm	counts per minute
DIL	derived investigation level
DOE	U.S. Department of Energy
DOL	U.S. Department of Labor
dpm	disintegrations per minute
DU	depleted uranium
EDE	effective dose equivalent
EEOICPA	Energy Employees Occupational Illness Compensation Program Act
g	gram
GE	General Electric Company
GEND	GE Neutron Devices
GENDD	GE Neutron Devices Department
GEPP	GE Pinellas Plant
GEXF	GE X-ray Division in Florida
GEXM	GE X-ray Division in Milwaukee, Wisconsin
HEPA	high-efficiency particulate air
hr	hour
ICRP	International Commission on Radiological Protection
keV	kiloelectron-volt, 1,000 electron-volts
HT	tritium gas
HTO	tritiated oxide
L	liter
LANL	Los Alamos National Laboratory
LOD	limit of detection
LSC	liquid scintillation counting
MDA	minimum detectable activity
MDC	minimum detectable concentration
MDL	minimum detection limit
MeV	megaelectron-volt, 1 million electron-volts
min	minute
mL	milliliter
mR	milliroentgen
MT	metal tritide
mrem	millirem
NIOSH	National Institute for Occupational Safety and Health
NIST	National Institute of Standards and Technology

OBT	organically bound tritium
POC	probability of causation
POPOP	1,4-bis[5-phenyloxazol-2-yl]benzene (a secondary scintillator)
PPO	2,5-diphenyloxazole (a primary scintillator)
RBE	relative biological effectiveness
ROI	region of interest
RTG	radioisotopically-powered thermoelectric generator
SRDB Ref ID	Site Research Database Reference Identification (number)
T	tritium
T_2	tritium gas
T_2O	tritium oxide
TBD	technical basis document
U.S.C.	United States Code
µCi	microcurie
µm	micrometer
§	section or sections

5.0 OCCUPATIONAL INTERNAL DOSE

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) define "performance of duty" for DOE employees with a covered cancer or restrict the "duty" to nuclear weapons work (NIOSH 2010).

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 2010):

- 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 Overview

This TBD is Part 5 of the Pinellas Plant's Site Profile. A site profile provides a summary of information about a site that is relevant to the dose reconstruction process.

The Pinellas Plant has been known by several names throughout its history. Those names include 908 Plant, Pinellas Peninsula Plant, GE X-ray Division-Florida (GEXF), GE Neutron Devices Department (GENDD), GE Neutron Devices (GEND), GE Pinellas Plant (GEPP), and the Pinellas Plant.

The General Electric Company built and operated the Pinellas Plant for DOE from its initial startup in January 1957 until June 1992. In June 1992, Martin Marietta Specialty Components, Inc. (MMSC) took over as the managing and operating contractor for the Pinellas Plant. In 1994, Lockheed merged with Martin Marietta and the managing and operating contractor for the Pinellas Plant was renamed Lockheed Martin Specialty Components (LMSC). The Pinellas Plant completed its war reserve fabrication of neutron generators at the end of September 1994, and began the transition from a defense mission to an environmental management mission. That transition included a number of decontamination and decommissioning activities that allowed the Plant to be turned over for commercial uses. LMSC continued as the managing and operating contractor until decontamination and decommissioning activities ended in 1997 (ORAUT 2011a).

The Plant was built to manufacture neutron generators, a principal component in nuclear weapons. The neutron generators consisted of a miniaturized linear ion accelerator assembled with pulsed electric power supplies. The ion accelerator, or neutron tube, required ultraclean, high-vacuum technology; hermetic seals between glass, ceramic, glass-ceramic, and metal materials; and high-voltage generation and measurement technology. The Plant manufactured only neutron generators for its first 10 years of operation. It later manufactured other products including neutron detectors, radioisotopic thermoelectric generators (RTGs), high-vacuum switch tubes, specialty capacitors, and specialty batteries (Weaver 1990). As part of its program to promote commercial uses of the site, DOE sold most of the Plant to the Pinellas County Industry Council in March 1995 and leased back a portion through September 1997 to complete safe shutdown and transition activities (LMSC 1996).

5.1.2 Purpose

The purpose of this technical basis document (TBD) is to document the internal dosimetry program and practices at the Pinellas Plant, and to provide the technical basis to be used to evaluate the internal occupational radiation dose for EEOICPA claims.

5.1.3 Scope

This TBD provides supporting documentation to assist in the evaluation of occupational internal doses in accordance with the guidelines described in *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 doses at the Pinellas Plant throughout its entire history.

5.2 RADIOACTIVE SOURCE TERM AND LUNG ABSORPTION TYPES

5.2.1 Tritium

Tritium (also denoted as T, H-3, and ^{3}H) is a hydrogen atom with two neutrons. It is the heaviest of the three isotopes of hydrogen (protium [^{1}H], deuterium [^{2}H], and tritium [^{3}H]) and is the only radioactive hydrogen isotope. Tritium is a low-energy, beta-emitting radionuclide with a half-life of 12.28 years (Kocher 1981). The average and maximum beta particle energies are 5.7 keV and 18.6 keV, respectively (Kocher 1981). Between 1957 and 1993, annual tritium inventories at the Pinellas Plant ranged from 5.44 g (5.24×10^{4} Ci) to 53.27 g (5.14×10^{5} Ci) (Biedermann 1994). Tritium is not considered to be an external radiation hazard because the beta particles being emitted have too low of an energy to penetrate human skin. However, inside the body its radiation can cause damage to tissues and organs.

Four primary forms of tritium were present at the Pinellas Plant: tritiated water (HTO), tritium gas (HT or T_2), organically bound tritium (OBT), and metal tritides (MTs). One of the most common forms of organically bound tritium at the Pinellas Plant was contaminated pump oils. The MTs (primarily scandium tritide, erbium tritide, and titanium tritide) formed during production processes could have been released in the work environment as particulate aerosols. The gas was allowed to react with metal surfaces, thin metal coatings, and metal powders for various purposes. Powders were normally contained with vacuum systems, and metal systems normally remained intact (Burkhart 1995a, p. 2). Based on the available information about the Plant's metal tritide uses, only a small portion of the worker population had the potential to be exposed to a dispersible form of a metal tritide, and those scenarios were typically limited to accidents involving metal tritides. Titanium tritide was also used in the original tritium storage beds at the Pinellas Plant (Burkhart 1990a). In the original tritium storage beds, the titanium tritide was sealed inside a glass cylinder (Burkhart 1990a). Because the glass cylinders for the tritium storage beds broke on occasion, resulting in high levels of contamination, the original glass tritium storage beds were replaced with stainless-steel tritium storage beds in 1968 (Phillips 1975). The stainless-steel tritium storage beds utilized a different MT, DU tritide (Burkhart 1990a; Eichman1979; Phillips 1975). Because DU tritides were sealed within a stainless-steel construction, and because there are no known incidents of DU tritide contamination at the Pinellas Plant, exposures to DU and DU tritide were unlikely. Table 5-5, which is provided later in this document, includes several incidents involving the various tritium compounds that were present at the Pinellas Plant.

The potential tritium use areas at the Pinellas Plant are summarized in Table 5-1. There have been a number of tritium contamination incidents, as discussed in Section 5.8.

Both soluble and insoluble tritium compounds were present at the Pinellas Plant. Tritium compounds with the lung absorption properties of type M or S are considered to be insoluble tritium compounds. All other forms of tritium are considered to be soluble tritium compounds.

Table 5-1. Potential tritium radioactive material use areas.

Location	Name	Activity
107	Tube Assembly	Vacuum tube manufacturing and coating
108	Tube Exhaust and Test	Vacuum tube evacuation and testing
109	Product Analysis	Magnetic and radioactive gas leak checking
128	Tube Test	
131	Final Tube Test	
132M	Fan Room	Stack effluent control and tritium recovery
157/158	Gas Analysis Laboratory	Hydrogen isotope analysis
182-C	Tube Assembly	Vacuum tube development and testing
182-G	Tube Exhaust	Vacuum tube development and testing
183	General Development	
191	CPE Hood Room	Ceramic product testing facility
	Radioanalytical Laboratory	Radiological Laboratory, tritium recovery
194	Engineering Environmental Testing	
200	Test areas	
800	Accelerator and Calibration	
1000	Waste Storage	

Source: DOE 1995, p. 5-3; Weaver 1993.

5.2.2 Plutonium

In terms of radioactive decay, ^{238}Pu and ^{239}Pu are alpha and X-ray emitting radionuclides with half-lives of 87.75 years and 24,131 years, respectively (Kocher 1981). The alpha particle emissions from these two radionuclides are the primary concerns in regards to internal dose. The photon (gamma ray and X-ray) and neutron radiation emitted from the sources containing plutonium are the primary concerns in regards to external dose. Gamma rays and neutrons are predominately emitted from the spontaneous fissions of plutonium, alpha-neutron reactions, and photon-neutron reactions.

The first plutonium that was received at the Pinellas Plant was a 7 g ^{239}Pu source, which was received in January 1957 (Author unknown undated a). The source was used for calibrating health physics monitoring equipment (Author unknown c. 1993a). Based on the available information regarding this source and its use, it was most likely an encapsulated plutonium-beryllium (Pu-Be) neutron source. The triply encapsulated plutonium oxide ($^{238}PuO_2$) heat sources that were used for the radioisotopic thermoelectric generators (RTGs) did not start arriving at the Pinellas Plant until November 1975 (Author unknown undated a). There were two different types of $^{238}PuO_2$ heat sources, 8.75 g sources and 10 g sources (GE 1982a). With the exception of 1975, no information was found regarding the annual inventories of $^{238}PuO_2$ heat sources. In November 1975, the site received seven $^{238}PuO_2$ heat sources (Author unknown undated a). By February 1991, all plutonium sources, with the exception of calorimeter sources and small instrument calibration check sources, were removed from the Pinellas Plant (MMSC 1992).

The potential plutonium use areas at the Pinellas Plant are summarized in Table 5-2.

Table 5-2. Plutonium radioactive material management areas.

Location	Name
400	RTG area
200	Test areas
800	Accelerator
1000	Waste storage

Based on the 1982 version of the Safety Analysis Report (SAR) for the RTG Facility, shipments of the $^{238}PuO_2$ heat sources were delivered directly to Building 400, where the RTG Facility is located (GE 1982a). When the shipping packages were to be opened, they were moved from the vault room to the source inspection hood where they were opened and surveyed for contamination (GE 1982a). Sources confirmed to be free of contamination were placed in a source storage container and returned to the vault room. If the unpacking survey showed contamination levels greater than 200 dpm, the source was to be immediately repackaged and returned to the supplier (GE 1982a; Huffman 1979). If the survey showed detectable contamination that was below 200 dpm, an effort would be made to decontaminate the source in accordance with procedures.

Plutonium sources were considered to be free of significant contamination if the removable contamination on a swipe of the entire source surface area was less than twice the statistical counting error associated with a 5-minute count and a 95% confidence level (GE 1982a). These criteria resulted in a control level of ≤ 6 dpm (GE 1982a). A letter dated January 1979 indicated that the occurrence of $^{238}PuO_2$ heat sources with contamination levels exceeding the limit of detection (LOD) was only about one in every 70 sources received or less than 6 per year (Huffman 1979). As of 1982, the SAR indicates that the circumstance of a contaminated $^{238}PuO_2$ heat sources needing to be returned to the supplier had not occurred. The SAR also states that the probability of a $^{238}PuO_2$ heat source leaking, although possible, is so small that it can be assumed that it will not occur. In another section of the SAR, it is stated that the "Gross failure of heat source encapsulation is not considered to be a credible event." The 1982 version of the SAR also states that "there is not, nor has there ever been, any plutonium contamination inside the facility nor released to the environment", which was reiterated in a 1989 memorandum (GE 1982a, Weaver 1989a).

Based on the information provided in the RTG Facility's SAR, plutonium intakes were extremely unlikely at the Pinellas Plant and the only probable plutonium intake scenario is a receipt inspection scenario involving a contaminated $^{238}PuO_2$ heat source. However, any potential intakes attributable to such a scenario would have been limited, since no contamination levels exceeding the 200 dpm limit appear to have ever been found.

5.2.3 Uranium

Depleted and natural uranium, which consist of ^{234}U, ^{235}U, ^{238}U and some of the radioactive progeny for these radionuclides, were present at the Pinellas Plant. In terms of radioactive decay, the uranium isotopes emit alpha particles and X-rays. However, some of the radioactive progeny emit beta particles and gamma rays.

The major use of DU was for the tritium storage beds that were first used in 1968 (Phillips 1975). Fifty grams of DU metal was used for the particulate uranium metal tritide in each of the tritium storage beds (Ward 1973, p. 29). Because the uranium in the tritium storage beds was sealed in stainless steel canisters, the uranium was considered to be a containerized source and would have posed little to no internal dose hazard. There was no indication that the uranium ever leaked from the storage beds at the Pinellas Plant. Given that particulate uranium metal is pyrophoric, any uranium metal leaking from the tritium storage beds would have ignited and resulted in a uranium fire incident at the site. Of the reported incidents for the Pinellas Plant, none were uranium release or uranium fire incidents. The depleted uranium (mainly ^{238}U) inside the tritium storage beds presents no significant external radiation hazard, due to the low specific activity and the nonpenetrating radiation emitted.

The major use of natural uranium was the use of borosilicate glass that was doped with natural uranium (1.5% by weight) in the form of U_3O_8 (Weaver 1992). Because the uranium would have been encapsulated in the glass prior to its arrival at the Pinellas Plant, the glass was considered to be a sealed source and would have posed little to no internal dose hazard.

5.2.4 Nickel-63

Nickel-63 is a low-energy, beta-emitting radionuclide with a half-life of 100.1 years. The average and maximum beta particle energies are 17.13 keV and 65.87 keV, respectively (Kocher 1981).

The information regarding the Pinellas Plant's use of ^{63}Ni, includes documents from the GE X-ray Division (a.k.a. GEXM) site. Because many of the same nuclear weapons-related activities were performed at the Pinellas Plant and GEXM sites, and because many items were obtained from the same vendors, use of ^{63}Ni and the form that it was obtained in were likely the same for both sites.

Nickel-63 was electroplated onto a nickel mesh inside a sealed glass tube (a *krytron*) by U.S. Radium, and averaged 0.3 µCi per spark gap (Weaver ca. 1995; Jech 1963). Krytrons are cold-cathode, gas-filled tubes intended for use as very high-speed switches, which have been used for igniting the exploding-bridge wire detonators and slapper detonators in nuclear weapons. The ^{63}Ni is used in conjunction with the keep-alive electrode, where the beta particles being emitted by the ^{63}Ni make the ionization inside the krytron easier. The available information indicates that the Pinellas Plant was not involved with the process of electroplating the ^{63}Ni to the keep-alive electrodes for the spark gap bodies, and only received the spark gap bodies containing the electroplated ^{63}Ni from U.S. Radium. It is not known if the electrodes plated with ^{63}Ni were already sealed in the glass tubes before they arrived at the Pinellas Plant or if the Pinellas Plant was creating and sealing the electrodes in the glass tubes. Given that one of the Pinellas Plant's areas of expertise was glass formulation, the Pinellas Plant was likely sealing the electroplated ^{63}Ni electrodes into the glass tubes to create the krytron.

No internal dose monitoring was conducted for ^{63}Ni based on the operations with devices and survey data of work areas and parts (Weaver ca. 1995). Pinellas Plant information indicates that hypothetical worst-case doses were calculated and placed in an "Internal dosimetry technical notes file" (Weaver ca. 1995). However, this file has not been found. Given that each spark gap only contained approximately 0.3 µCi (11,100 Bq) of ^{63}Ni and given that the worst-case organ dose coefficient for ^{63}Ni is 5.6×10^{-9} Sv/Bq (2.07×10^{4} rem/Ci) (ORDOSE 2003), inhaling the total radioactivity in a single spark gap body would only result in a maximum committed (50-year) organ dose of approximately 6.2 mrem. Therefore, it is unlikely that any workers at the Pinellas Plant received a significant internal dose from ^{63}Ni, and potential ^{63}Ni exposures do not need to be assessed for Pinellas Plant workers. As a result, ^{63}Ni is not discussed any further in this TBD.

5.2.5 Carbon-14

Carbon-14 is a low-energy, beta-emitting radionuclide with a half-life of 5,730 years. The average and maximum beta particle energies are 49.47 keV and 156.48 keV, respectively (Kocher 1981).

The use of ^{14}C at the Pinellas Plant is only indicated in the gaseous effluent release reports (GE 1980, 1981, 1982b, 1983, 1984a) and in an environmental assessment (DOE 1983). The gaseous effluent release reports indicate that ^{14}C was used between 1979 and 1983 (GE 1980, 1981, 1982b, 1983, 1984a). Based on the reported gaseous effluent releases for those years, ^{14}C was used in much smaller quantities than tritium. A comparison of the annual quantities of gaseous effluents released indicates that the curies of tritium being processed were over 100,000 times greater than the curies of ^{14}C being processed. A 1983 environmental assessment indicated that small quantities of ^{14}C labeled-solvents were used in a laboratory testing operation (DOE 1983). No other documentation was found to indicate whether or not there were any other uses of ^{14}C. No documentation was found that indicates what chemical forms of ^{14}C were used. Given that ^{14}C use was much less than tritium use at the Pinellas Plant, and given that the worst-case organ dose coefficients for ^{3}H and ^{14}C in the Radiological Toolbox computer program (ORDOSE 2003) are within an order of magnitude of each other, it is unlikely that ^{14}C was a significant internal dose concern at the Pinellas Plant. Therefore,

internal doses due to ^{14}C exposures do not need to be assessed for Pinellas Plant workers unless ^{14}C exposure information is provided in the worker's dosimetry records. As a result, ^{14}C is not discussed any further in this TBD.

5.2.6 Krypton-85

Because ^{85}Kr is a noble gas, it is not a significant internal dose concern. Therefore, internal doses due to ^{85}Kr exposures do not need to be assessed, and ^{85}Kr is not discussed any further in this TBD.

5.2.7 Miscellaneous Radionuclides

A wide variety of other radionuclides were used at the Pinellas Plant; however, the uses of these radionuclides were mostly limited to sealed and plated check sources, static meter sources, explosive meter sources, heat sources, calibration sources, thickness gauges, gas chromatograph sources, dew point measurement sources, and static eliminator sources (Author unknown undated b). Even though some of these sources contained significant quantities of radioactivity (Author unknown undated b), they were not considered to be potential sources for radionuclide intakes, unless a specific worker was involved in an incident where an intake pathway was created for one of these sources. Therefore, intakes and internal doses for other miscellaneous radionuclides do not normally need to be evaluated for Pinellas Plant workers, and these radionuclides are not discussed further in this TBD. Any potential intakes of radioactivity and subsequent doses due to an incident involving one of these radioactive sources will need to be evaluated on a case-by-case basis.

5.3 HISTORICAL MONITORING PRACTICES

The Pinellas Plant internal dosimetry program started with site operations in 1957. Contamination monitoring, air sampling, and bioassay monitoring were the three primary types of monitoring used at the Pinellas Plant to detect potential intakes of radioactive materials.

5.3.1 Contamination Monitoring Practices

Tritium and plutonium were the only radioactive materials that contamination monitoring was routinely performed for in the production areas. Outside of the production areas at the Pinellas Plant, contamination monitoring might have been performed for other radioactive materials; however, no documentation regarding other potential types of contamination monitoring has been found.

5.3.1.1 Tritium Contamination Monitoring

Contamination monitoring for tritium was performed on a routine basis from the beginning of operations at the Pinellas Plant (GE 1957–1973). Work areas and personnel were checked for contamination on a routine basis. Any significant personnel contamination that could have gone undetected from contamination surveys would most likely have been identified through the tritium bioassay program.

The 1957 through 1973 monthly health physics reports (GE 1957–1973) indicate that as early as 1959, areas greater than 2×10^{-5} µCi/in^2 (688 dpm/100 cm^2) are recommended for decontamination. In a 1969 health physics report, the control limit was reported as 440 dpm/100 cm^2 (GE 1957–1973). A circa 1966 smear survey procedure indicated that the uncontrolled area limit for tritium was 220 dpm/100 cm^2 and the controlled area limit was 440 dpm/100 cm^2 (GE ca. 1966). The monthly health physics reports also provide information on the maximum surface contamination levels. Between 1957 and 1973, the highest surface contamination level reported was in 1970, 4.4×10^6 dpm/100 cm^2 (10,000 times the control limit) (GE 1957–1973). The next highest value was reported in 1959, 1.4×10^6 dpm/100 cm^2 (3.3×10^{-2} µCi/in^2) (GE 1957–1973). The majority of the rest of the annual

maximum surface contamination levels reported are at least an order of magnitude lower than these two.

It was known early that wet swipes for contamination were more efficient than dry swipes. The water geometry for counting was used with the liquid scintillation counter for contamination monitoring. This same water geometry was used for HTO until about 1971 for the urine bioassays as well (GE 1967–1982). The procedure for collecting the smears involved pipetting 5 mL of distilled water containing the cotton ball swipe, taking the wet cotton ball to the area, and swiping the area of concern (usually of 100 to 1,000 cm^2). Upon return to the counting lab, 10 mL of water was added to the swipe container, and the sample was filtered through Whatman #1 filter paper. Two-tenths of a milliliter of sample was added to 16.5 mL of cocktail solution, and the sample was counted for 5 minutes. The minimum detectable concentration (MDC) was 0.002 µCi/area swiped (GE 1967–1982).

By 1989, contamination monitoring occurred daily in what were considered contamination areas. This included Laboratories 158B, 182, and 108. Weekly surveys included Laboratories 158B, 157, 182, 182G, 108, 132B, 109, 128, and 131. Monthly surveys included the "Pure Zone"; Laboratories 107, 114 (X-ray), 138, 161, 158A, and 191-N (CPE laboratory); hallways; Buildings 200, 400, 800, and 1000; and the eating areas. Other areas were surveyed less frequently (GE 1989a; LMSC 1995).

5.3.1.2 Plutonium Contamination Monitoring

In the RTG areas, the Pinellas Plant routinely monitored the work areas for possible plutonium contamination from the beginning of plutonium source use for the RTGs in 1975. This occurred despite the fact that the sources were triply encapsulated and were held to more stringent standards than other Federal regulatory agencies required for sealed sources of radiation.

Areas were checked for contamination with a portable alpha meter, a PAC 4S proportional detector. Because this detector was also sensitive to gamma radiation, users were instructed to place a sheet of paper over the detector to absorb the alphas to positively indicate the presence of alpha radiation.

As indicated in Section 5.2.2, no areas were ever found to have plutonium contamination at the Pinellas Plant.

5.3.2 Air Sampling Practices

Tritium and plutonium were the only radioactive materials that air sampling was routinely performed for in work areas. Nonroutine air sampling might have been performed for other radioactive materials; however, no documentation regarding other potential types of air sampling has been found.

5.3.2.1 Tritium Air Sampling

Monitoring for airborne radioactivity was performed on a routine basis from the beginning of operations at the Pinellas Plant (GE 1957–1973). Because tritium was the only known source of contamination at the plant during 1957 and because this monitoring appears to have been performed using a Kanne ion chamber with a glass wool filter on the intake (GE 1957–1973, page 2), the early airborne radioactivity monitoring is assumed to have been limited to gaseous forms of tritium. In 1957, the maximum permissible concentration for airborne radioactivity was 7×10^{-5} µCi/mL (GE 1957–1973, p. 5).

Fixed-room monitors were located in all areas where there was a potential for release of tritium. The monitoring systems consisted of a 22 L Kanne ionization chamber that was connected to a picoammeter and an alarm panel. In 1973, there were 40 sampling ports and 20 monitors. The air blowers had backups and an alarm system if they were not working properly. Some Kanne chamber

systems had multiple or sequential sampling systems when more than two ports were attached. This allowed sequential sampling for 20-second intervals for each port. A Kanne ionization chamber monitor was capable of detecting tritium below the U.S. Atomic Energy Commission (AEC) 40-hr level of 2×10^{-5} µCi/mL, which is the same as the current derived air concentration agreed to by DOE and the U.S. Nuclear Regulatory Commission (Author unknown 1987; Ward 1973). The MDC has been calculated to be as low as 1.3×10^{-5} µCi/mL (Ward 1971). The alarm was set at about four times the AEC/DOE/NRC limit of 2×10^{-5} µCi/mL for tritium oxide (i.e., at about 80 µCi/m^3 or 8×10^{-5} µCi/mL). The operation of each fixed monitoring system was response checked monthly using a small gamma source and hand-held smoke generator. Areas monitored included Laboratories 108, 109, 157, 158, 182, and 132. Pinellas Plant health physicists decided whether chart recorders were to be used, such as in cases where suspected higher-than-normal HTO concentrations were anticipated.

Portable tritium gas monitors were used as temporary monitors in areas where fixed -room monitor probes were not located (Author unknown 1987; Ward 1973). Portable samplers using silica gel collection media, or silica gel stations were also set up in some areas.

5.3.2.2 Plutonium Air Sampling

Real-time air monitoring for ^{238}Pu and ^{239}Pu employed Eberline models alpha-2 or alpha-3 room monitors in Building 400 (eight locations) using Whatman microfibre #934-AH glass or # 41 ashless 5.5-cm-diameter filters. The following areas were monitored (Burkhart and Richardson 1986):

- The receiving room where newly arrived heat sources were unpacked;
- The gloveboxes where the heat sources were handled (inside and outside the glovebox lines);
- The defect analysis glovebox area where final units were disassembled;
- The storage vault where heat sources, partially assembled units, and completed units awaiting shipment were stored;
- The exhaust plenum [with high-efficiency particulate air (HEPA) filter] and ventilation exhaust stacks.

The filters for the receiving room, defect analysis box, and the storage vault were changed on Mondays, Wednesdays, and Fridays, and the air rotometers were checked at these times. Monthly changes occurred for HEPA filters and gloveboxes, and the air rotometers were also checked at this time. The air monitors were set at 10- or 15-cpm alarm points. They were checked with ^{239}Pu check sources on a monthly basis.

The Eberline alpha continuous air monitors were set at a fraction of the air concentration limits. They operated to monitor specifically for ^{238}Pu or ^{239}Pu and had background subtraction and alarming capabilities. Since the 1970s, Pinellas employed high-technology silicon barrier detector systems connected to a single-pulse analyzer, which enabled the monitoring of air for potential releases of plutonium (EIC 1970). The Pinellas Plant continued to upgrade the plutonium air monitoring systems over the years (Dixon 1988a,b).

Air sampling occurred at nine points (probably identical to the real-time air-sampling points listed above). These samplers operated continuously and were changed monthly. Following 3 days of decay, they were counted for gross alpha and composited quarterly for ^{238}Pu analysis. In the 1988 plutonium bioassay report, it was indicated that no plutonium had been found detected in the air, on heat sources, or on completed RTG units (Weaver 1989).

5.3.3 Bioassay Monitoring Practices

Tritium and plutonium were the only radioactive materials that bioassay monitoring was routinely performed for. Nonroutine bioassay monitoring might have been performed for other radioactive materials; however, no documentation regarding other potential types of bioassay monitoring has been found.

5.3.3.1 Records Interpretation

The only interpretation issue that is relatively common in the bioassay records has to do with the continuation sheets for the tritium urinalysis results, which are used when the number of urine samples for a given year exceed the allotted space on the primary datasheet. These continuation sheets sometimes do not include any information regarding the year that the bioassay results were collected. In those instances, the dates that are provided only include the month and day that each urine sample was collected. Fortunately, the continuation sheets appear to always follow the sheet with the data from the first part of the year. This can also be confirmed by comparing the end dates of the first sheet with the start dates of the second sheet. The start dates of the second sheet should pick up where the end dates of the first sheet left off.

5.3.3.2 Tritium Bioassay Monitoring

Since about 1986, the bioassay program at Pinellas was based upon ANSI standard N13.14-1983 (ANSI 1983). Participation was determined based on the recommendations in the ANSI standard and included:

- Anyone with the potential to receive 100 mrem/year from tritium;
- Declared pregnant workers likely to receive more than 50 mrem/gestation period;
- Minors, visitors, and members of the public likely to receive more than 50 mrem/year;
- All personnel who worked with or handled tritium-contaminated systems or equipment.

The routine sampling frequency was determined by the extent of possible exposure. In the earlier years from about 1957 to 1972, it appears that the frequency was usually weekly, but monthly or daily samples could have been taken. In later years, the frequency most likely followed the criteria as stated in the Pinellas internal dosimetry TBD (Burkhart 1995a; GE 1984ab, p. 2) as follows:

Daily or on each performance:
- Work on open neutron generator tubes or tube processing equipment;
- Maintenance on vacuum pumps, glove boxes, or exhaust systems including the Tritium Recovery system (TRS);
- Instances of area contamination (1984 version);
- Packaging and disposal of radioactive waste (1984 version).

Weekly:
- Operation of contaminated processing or analysis equipment;
- Decontamination of materials and facilities;
- Packaging and disposal of radioactive wastes;
- Mass spectrometers and tritide films (1984 version).

Monthly:
- Handling of processed tubes (slight potential of measurable exposure).

Pinellas required tritium bioassays for contractors and nonroutine work in tritium areas covered by work permits at the conclusion of the work. This occurred throughout the work history at the Pinellas Plant.

Beginning at least as early as 1966, urine samples to be analyzed for tritium were submitted on a daily basis at the end of a worker's shift, or at the beginning of the next shift for each individual worker. Spot samples were composited from each shifts' individual samples to be counted on the next shift. If the samples were weekly, daily samples were composited to give the weekly sample that was then sampled and counted.

A relative biological effectiveness (RBE) of 1.7 was used to calculate tritium exposures for the years before 1976. This was suggested by the National Bureau of Standards (NBS), ICRP, and National Council on Radiation Protection and Measurements-published references of that time. Calculations for 1976 and subsequent years were made using an RBE of 1.0 (Holliday ca. 1976). Exposures prior to 1976 were later adjusted to an RBE of 1.0. In the exposure records, most of the tritium dose records were adjusted from 1957 to 1975 by dividing by 1.7, as indicated in handwritten calculations in the claimants' DOE dose files.

Prior to 1987, a 5 mrem/µCi/L infinite dose factor, which was based on an effective half-life for tritium in the body of 12 days, was used for the tritium dose calculations (Holliday 1983). In November of 1986, the infinite dose factor was recalculated as 4.21 mrem/µCi/L, which was based on an effective half-life of 10 days for tritium in the body (GE 1986, Weaver 1994). The use of the new infinite dose factor appears to have been implemented in 1987, and was used for the tritium dose calculations throughout the remainder of the Pinellas Plant's history (GE 1986, Weaver 1994). Both of the infinite dose factors were based on the following equations (GE 1986).

$$DCF_{\infty} = \frac{V_{BW}\left(2.22 \times 10^{6}\ \frac{dpm}{\mu Ci}\right)\left(1440\ \frac{min}{day}\right)\left(5.7 \times 10^{-3}\ \frac{Mev}{disintegration}\right)\left(1.6 \times 10^{-6}\ \frac{erg}{Mev}\right)\left(1000\ \frac{mrem}{rem}\right)}{M_{BW}\ \lambda_E\ QF\left(100\ \frac{erg/g}{rad}\right)} \quad (5\text{-}1)$$

$$V_{BW} = \frac{M_{BW}\left(1 \times 10^{-3}\ \frac{L}{mL}\right)}{d_W} = 42 L \quad (5\text{-}2)$$

$$\lambda_E = \frac{\ln(2)}{T_{\frac{1}{2}E}} \quad (5\text{-}3)$$

where

DCF_{∞} - daily intake to infinite dose conversion factor [mrem/(µCi/day)]
M_{BW} - mass of total body water in standard man (42,000 g)
QF - quality factor for radiation type (1 rem/rad)
V_{BW} - volume of total body water in standard man (L)
d_W - density of water (1 g/ml)
λ_E - effective removal rate for tritium in the body (day^{-1})
$T_{½E}$ - effective half-life of tritium in the body (days)

For $T_{½E}$ = 12 days: $\lambda_E = 0.05776\ day^{-1}$ and $DCF_\infty = 5.048 \approx 5$ [mrem/(µCi/day)]

For $T_{½E}$ = 10 days: $\lambda_E = 0.06931\ day^{-1}$ and $DCF_\infty = 4.206 \approx 4.21$ [mrem/(µCi/day)]

Since 1986, the method for calculating internal tritium doses at the Pinellas Plant was based on ANSI Standard N13.14-1983 (ANSI 1983). Prior to that, the following equation was used to calculate the internal tritium doses (Holliday 1983).

$$D_\infty = DCF_\infty[(C_i e^{-\lambda_E(\Delta T-1)})] - (C_{i-1} e^{-\lambda_E T_0}) \quad (5\text{-}4)$$

where

D_∞ - infinite dose (rem)
DCF_∞ - daily intake to infinite dose conversion factor [5 mrem/(µCi/day)]
C_i - tritium concentration in most recent urine sample (µCi/L)
C_{i-1} - tritium concentration in prior urine sample (µCi/L)
ΔT - elapsed time between the sample collection dates (days)
T_0 - time after prior sample (1 day)
$T_{½E}$ - effective half-life of tritium in the body (12 days)
λ_E - effective removal rate for tritium in the body (0.05776 day^{-1})

The GEDOSE computer program was written in 1988 to process both external and internal dosimetry data and doses. The computer program's trigger for performing internal dose calculations was 0.1 µCi/L. If this level was exceeded by any urine sample submitted during a calendar year, an internal dose was calculated for the worker using all of that year's sample results. If none of a worker's urine sample results for a given year exceeded the trigger level, an internal dose of zero was assigned for that year (Burkhart 1995a).

5.3.3.3 Plutonium Bioassay Monitoring

The Safety Analysis Report for the Radioisotopic Thermoelectric Generator (RTG) Facility indicates that personnel assigned to work in the east portion of Building 400 (i.e., the RTG operations portion) were required to submit a preoperational urine sample and annual samples thereafter, because of the presence of the encapsulated plutonium sources at the site (GE1982a, Holiday 1983). The available plutonium bioassay data also confirms that this was being done.

For urine samples being analyzed for plutonium, the reported sample size was what was received. If the sample received was less than 800 mL, the entire sample was analyzed. If more than 800 mL was received, only an 800 mL aliquot was analyzed.

The criteria used by the Pinellas Plant to evaluate the plutonium bioassay results indicated that no plutonium intakes occurred at the site. In addition, the Pinellas Plant never assigned any internal plutonium doses based on its assessment of the plutonium bioassay data.

5.4 *IN VITRO* BIOASSAY DETECTION LEVELS AND ANALYSIS METHODS

Urine sampling was the only *in vitro* bioassay method employed at the Pinellas Plant.

5.4.1 Urinalysis Detection Levels

5.4.1.1 Tritium Urinalysis Detection Levels

Tritium urinalysis is only capable of detecting intakes of soluble forms of tritium (i.e., tritium gas, tritiated water (HTO), uranium tritide, and certain organically bound tritium compounds [e.g., methane, acetone, and octane]). Because urinalysis is ineffective for detecting insoluble forms of tritium (DOE 2006), potential intakes of insoluble forms of tritium are considered to be unmonitored at the Pinellas Plant.

Table 5-3 shows that the tritium urinalysis MDCs and reporting levels varied over the years at the Pinellas Plant. When available, the information in an individual worker's dosimetry records is to be used. If claim specific details regarding the urine sample MDCs and reporting levels are not available, the dose reconstructors should use the higher of the two values in Table 5-3 for a given period as the default MDC for the internal dose calculations. For example, a default MDC of 0.90 µCi/L would be used for 1957, and a default MDC of 0.67 µCi/L would be used for 1963. Note that using reporting levels that are below the MDC could result in an underestimate of the missed internal dose.

Table 5-3. Default tritium urinalysis MDCs and reporting levels.[a,b]

Period	MDC µCi/L	Reporting level[c] µCi/L
1957–Jul 1958	0.90[d]	0.3
Aug 1958–Dec 1958	0.90[d]	1.0
Jan 1959	0.80[d]	1.0
Feb 1959–Dec 1959	0.80[d]	N/A
1960–Jun 1961	0.50[d]	N/A
Jul 1961–1968	0.50[d]	"<"[e]
1969–1970	0.50[f]	0.67
1971–Mar 1974	0.20[d]	0.67
Apr 1974–Dec 1974	0.20[d]	0.10
1975–1986	0.10	0.10
1987–1989	0.01	N/A
1990–1997	0.006[g]	N/A

a. Sources: Claims; GE 1968, 1971, undated a, undated b, undated c; Holliday 1983; Burkhart and Richardson 1986; Burkhart 1995a.
b. Urinalysis method was liquid scintillation counting for tritium throughout Pinellas Plant site history.
c. All reporting levels are based on what was observed in the available dosimetry records for the Pinellas Plant, and appear to have been consistently used for all workers that were monitored for tritium intakes; N/A – not applicable, because no reporting level was used during this period.
d. MDC value calculated by ORAUT from data in urine sample counting logs and procedures.
e. If no specific reporting level value can be determined from the dosimetry records assume the reporting level for this period was <0.67 µCi/L.
f. Assumes MDCs were not improved after 1968.
g. MDC value obtained from Burkhart 1995a.

5.4.1.2 Plutonium Urinalysis Detection Levels

Plutonium urinalysis detection levels varied for individual sample results and were reported in terms of MDCs. With the exception of the 1975 and 1976 plutonium bioassay data, the MDCs associated with the specific sample results are always reported with the plutonium bioassay results. However, to be favorable to the claimant, the best available MDC will be used to evaluate the Pinellas Plant's isotopic plutonium urinalysis data. As a result, the isotopic plutonium MDCs for the Los Alamos National Laboratory (LANL) will be used for evaluating the plutonium urinalysis data against the criteria specified in Section 5.7.3.

The LANL was one of the forerunners in developing plutonium bioassay methods and was one of the few sites performing an isotopic plutonium analysis on urine samples when the Pinellas Plant began using this analysis method. As a result, the analysis method that was used at the Pinellas Plant was likely based on the method being used at LANL, which makes the LANL MDCs the best available MDC information available for use with the Pinellas Plant data. The TBD for the Los Alamos National Laboratory – Occupational Internal Dose indicates that the MDC for both ^{238}Pu and ^{239}Pu is 0.03 pCi/24-hr sample for the 1975–1990 timeframe (ORAUT 2009). An MDC of 0.03 pCi/24-hr sample is equivalent to 4.8×10^{-5} dpm/ml and 2.1×10^{-11} µCi/ml.

5.4.2 Analysis Methods for *In Vitro* Urine Samples

5.4.2.1 Analysis Methods for Tritium

Liquid scintillation counting (LSC) was used to analyze the urine samples for tritium. A sample consisted of 0.2 mL of urine combined with 16.5 mL of aqueous scintillation mix (1 L toluene, 0.2 L ethanol, 3.5 g PPO, 0.12 g POPOP) (GE 1964). From 1957 through at least 1964, urine samples were typically counted for 5 minutes (GE 1957, 1958, 1959, 1960, 1964). By 1966, the typical count times for the urine sample had been reduced to 1 minute (GE 1966, 1968, undated a). Counting efficiencies for the years of 1957–1960 ranged from about 5% up to about 13% (GE 1957, 1958, 1959, 1960). By 1964, counting efficiencies ranged from about 13% up to about 18% (GE 1964, 1968, 1971).

Beginning at least as early as 1966, daily urine samples were composited for screening purposes. Aliquots of the daily urine samples collected at the end of each shift from individual workers were composited with aliquots of the daily urine samples from up to 40 workers. These composite samples were then analyzed during the following shift. If a composite sample was above the reporting level, the composite group was subdivided into 4 groups and 4 new composite samples were created and analyzed. For the subdivided composite samples that were above the reporting level, the individual workers' urine samples used to make the subdivided composite sample were analyzed individually. Weekly and monthly urine samples were not composited with urine samples from other workers (GE undated a).

A review of the available bioassay records indicates that very few daily urine sample results were reported, even for workers that were likely on a daily sampling frequency. The data indicates that the daily samples for individual workers may have been composited into a weekly composite sample or that only one of the urine samples collected during a given week was analyzed individually. The available procedures discuss the analysis of the weekly sample for individual workers that were on a daily urine sampling frequency, but no specifics about the weekly sample could be found. A few daily sample results do show up in the records periodically, but they could be the result of the analysis of the individual samples that would have been performed after a composited screen sample was found to be over the reporting level.

The composite samples were prepared by combining a 1 mL aliquot from each worker's daily urine sample in a paper cup. A procedure dating from the 1970s was first to add charcoal to the solution to decolorize the solution and filter the composite through Whatman #1 filter paper. Then 0.2 mL of the composite sample was pipetted into a vial containing 16.5 mL of the aqueous scintillation mix. The composite sample vials were then placed in the freezer for cooling and counted for 10 minutes on a Beckman multichannel liquid scintillation counter (GE undated a).

The 1970's urine samples for individual workers were processed in a similar manner with the following exceptions: 1) the samples were not composited, 2) the charcoal was added directly to the individual sample container, 3) 5 mL of the urine sample was poured into a paper cup, and 4) the samples were initially counted for 1–2 minutes. On the Beckman multichannel liquid scintillation counter, the preset error control for channel 1 was set to 15%, the preset time was set to 1–2 minutes, and the action control switch was set to cycle repeat. When the initial count of an individual's urine sample result was above the reporting level, the result was verified by processing and counting a second sample for 5 minutes (preset error ± 5%) (GE undated a).

5.4.2.2 Analysis Methods for Plutonium

Urine samples being analyzed for plutonium were prepared in a number of steps. The first step was a 1-2 week dissolution process. The second step was a solution purification with the use of an anion exchange resin column. The third step was a 1-day electroplating. The final step was alpha spectroscopy. In addition, Pinellas used National Institute of Standards and Technology (NIST)-traceable sources of ^{238}Pu and ^{239}Pu to set up the ROIs (Christy 1988).

In 1988, two major changes to plutonium sampling and analysis were made. The first change was from a one-channel analysis to an energy region of interest (ROI) technique (Weaver 1989a). The use of polished stainless-steel planchets was implemented (Christy 1988).

By 1989, the Pinellas chemistry counting laboratory used a Series 100 Canberra alpha spectroscopy system [as shown in a sample spectrum comparison to NIST ^{238}Pu and ^{239}Pu source spectrums (GE 1989b)]. This alpha spectroscopy system likely used a silicon barrier detector system. The energy ROI for ^{238}Pu was 5,409.7 to 5,535.5 keV and that for ^{239}Pu was 5,063.4 to 5,186.6 keV. The 1988 and 1989 bioassay result folders (GE 1987–1990) show that the ROIs varied. The 200-keV interval between the ROIs for ^{238}Pu and ^{239}Pu indicates that very good resolution was attained for this counting system. A 24- to 72-hr counting time was usually used, but in 1989, a 24-hr counting time was used. A bioassay recount was made for 6 days or 144 hr to improve counting statistics (Weaver 1989b).

Pinellas conducted a quality assurance check on the plutonium bioassay program by comparing several samples of blind ^{238}Pu/^{239}Pu artificial urine samples with the results from the DOE Mound Laboratory. These comparisons were conducted from at least 1987 though 1990. The results compared favorably (Burkhart 1990b).

Plutonium concentrations in urine for ^{238}Pu and ^{239}Pu were calculated with the following formula (GE 1987–1990 pp. 83–93):

$$C_x = \frac{(N_{nx})(C_{242})(V_{242})(1 \times 10^{-6}\ \mu Ci/pCi)}{(N_{n242})(V_s)(2.22\ dpm/pCi)} \quad (5\text{-}6)$$

where

x - atomic mass number for the applicable Pu isotope (i.e., ^{x}Pu = ^{238}Pu or ^{239}Pu)
C_x - concentration of ^{x}Pu in the urine (µCi/mL)
N_{nx} - net counts accumulated during a count time of T_g for the ^{x}Pu region of interest
C_{242} - concentration of ^{242}Pu in tracer (dpm/mL)
V_{242} - quantity of tracer that sample was spiked with (mL)
N_{n242} - net counts accumulated during a count time of T_g for the ^{242}Pu region of interest
V_s - volume of urine used for sample (mL)
T_g - gross count time for the measurement

The uncertainties associated with the ^{238}Pu and ^{239}Pu in urine results were based on the formulation in Equation 5-7 (GE 1987–1990 pp. 83–93). Based on Equation 5-7, the uncertainty was calculated at the 95.44% confidence level (i.e., at the 2σ confidence interval).

$$U_x = \frac{2(N_{nx})^{½}(C_{242})(V_{242})(1 \times 10^{-6}\ \mu Ci/pCi)}{(N_{n242})(V_s)(2.22\ dpm/pCi)} \quad (5\text{-}7)$$

where

x - atomic mass number for the applicable Pu isotope (i.e., ^{x}Pu = ^{238}Pu or ^{239}Pu)
U_x - uncertainty of the ^{x}Pu in urine result (± µCi/mL)
N_{nx} - net counts accumulated during a count time of T_g for the ^{x}Pu region of interest
C_{242} - concentration of ^{242}Pu in tracer (dpm/mL)
V_{242} - quantity of tracer that sample was spiked with (mL)
N_{n242} - net counts accumulated during a count time of T_g for the ^{242}Pu region of interest
V_s - volume of urine used for sample (mL)
T_g - gross count time for the measurement

5.5 *IN VIVO* MINIMUM DETECTABLE ACTIVITIES AND ANALYSIS METHODS

There was no *in vivo* monitoring at the Pinellas Plant.

5.6 INTERFERENCES AND UNCERTAINTIES

5.6.1 Tritium Bioassay

Uncertainties or errors for tritium bioassay measurements were usually not stated in the personnel records or database, except for 1972-1980 at a level of ±10% error. MDCs were calculated based

upon a 95% confidence level and samples were counted to between a 10% to 15% error as indicated in Pinellas Plant procedures (GE undated).

Cross-contamination of the tritium urine samples was a potential interference that would have resulted in an overestimate of a worker's potential soluble tritium exposure. In about 1985, it was stated that one positive tritium urine sample followed by a negative sample would negate the first positive sample if given in the same day at the Pinellas Plant. If the urine sample for tritium was unattainable the same day, the employee was asked to give the sample within the first hour of being on site the next day. It is likely that most personnel did this, thus minimizing the possible cross-contamination issue for tritium bioassay samples.

5.6.2 Plutonium Bioassay

For plutonium bioassay, uncertainties were typically reported with the plutonium urinalysis results after 1976.

There should have been no issue of cross contamination with plutonium samples because they were taken annually and not likely after recent work with the plutonium sources. However, plutonium from the atmospheric testing of nuclear weapons is an interference that was encountered with low-level plutonium measurements. The interference from weapons related plutonium, which is dominated by $^{239/240}$Pu, is readily discernable from the Pinellas Plant plutonium that was dominated by ^{238}Pu.

5.7 ASSESSMENT OF INTAKES AND DOSES

The Pinellas Plant had an extensive bioassay program from the beginning of operations. Urinalysis started in 1957. Although the earlier techniques had their sensitivity limitations, the detection sensitivity seemed to keep pace with the fast-paced regulatory and safety changes. Seldom did workers achieve or surpass the site action levels of the radionuclides of concern. All Pinellas Plant workers that were potentially exposed to tritium and plutonium were likely monitored for potential internal dose, with the only likely exception being maintenance workers. Information provided by employees at the Pinellas Plant indicates that maintenance workers were often not monitored for internal dose when working in areas with tritium.

5.7.1 Workers Monitored for Soluble Tritium Exposure

5.7.1.1 Assessment of Soluble Tritium Exposures

For the periods that a worker was monitored for soluble tritium exposures, the potential exposures should be assessed in accordance with the recommendations in ORAUT-OTIB-0011 (ORAUT 2004) and ORAUT-OTIB-0060 (ORAUT 2007) using the worker's urine sample data.

5.7.1.2 Assessment of Potential Unmonitored Exposures to Insoluble Tritium

At the Pinellas Plant, insoluble tritium (i.e., certain metal tritides and certain organically bound tritium compounds) were handled only in areas where the more dispersible and more soluble forms of tritium [e.g., elemental tritium (HT), tritiated water (HTO), etc.] were also present. Given that a review of the available dosimetry records indicates that the Pinellas Plant routinely monitored workers with any potential for soluble tritium exposures, any workers with potential metal tritide exposures would have been monitored for soluble tritium exposures. Therefore, insoluble tritium exposures at the Pinellas Plant are only assessed for the periods that workers were monitored for soluble tritium exposures. Given that the least frequent routine monitoring frequency for soluble tritium was monthly, the minimum period for assessing insoluble tritium exposures for intermittently monitored workers should be 1 month.

The 1957 through 1973 monthly health physics reports (GE 1957–1973) were reviewed to determine tritium contamination controls at the Pinellas Plant. These reports indicate that as early as 1959, areas greater than 2×10^{-5} µCi/in^2 (688 dpm/100 cm^2) were recommended for decontamination. In 1969, the control limit was reported as 440 dpm/100 cm^2. This indicates that a routine contamination control program was in place throughout the history of the site and that it would be unlikely to see high contamination levels for extended periods.

The monthly health physics reports also provide information on the maximum tritium surface contamination levels. Between 1957 and 1973, the highest surface contamination level reported was in 1970, 4.4×10^6 dpm/100 cm^2 (10,000 times the control limit). The next highest value was reported in 1959, 1.4×10^6 dpm/100 cm^2 (3.3×10^{-2} µCi/in^2). The majority of the rest of the annual maximum surface contamination levels reported are at least an order of magnitude lower than these two. However, the following assumptions are made to bound the potential unmonitored metal tritide exposure:

- Constant tritium surface contamination level of 4.4×10^6 dpm/100 cm^2 (4.4×10^8 dpm/m^2) (GE 1957–1973);
- Resuspension factor of 1×10^{-6}/m (ORAUT 2008).

Based on these assumptions, a constant tritium air concentration of 440 dpm/m^3 was possible in the areas where tritium was handled. Assuming a breathing rate of 1.2 m^3/hour and the exposure time assumption of 2,600 hours (based on a review of telephone interviews provided by former workers, 50-hour weeks were routine), annual inhalation and ingestion intake rates for insoluble tritium were calculated and are provided in Table 5-4. The values provided in Table 5-4 should be assigned to account for potential insoluble tritium intakes for the periods that a worker was exposed to tritium in addition to any soluble tritium intakes that were assessed based on tritium bioassay data. For partial years of employment or monitoring, the daily intake values in Table 5-4 can be used.

Table 5-4. Insoluble tritium intake rates.

Hours worked per year[a]	Annual intake rates (pCi/yr)		Daily intake rates[b] (pCi/d)	
	Inhalation	Ingestion[c]	Inhalation	Ingestion[c]
2,600	6.18E+05	1.29E+04	1.69E+03	3.53E+01

a. Based on a 10-hour workday, 5 days a week, for 52 weeks a year.
b. These daily intake rates are calculated for a calendar year (i.e., 365 d/yr) and can be entered directly in the IMBA computer program without any conversions.
c. Based on the recommendation in OCAS-TIB-0009 (NIOSH 2004) for best estimates.

The available information indicates that insoluble tritium compounds representing lung absorption Types M and S were present at the Pinellas Plant. Because there is insufficient information to indicate which insoluble tritium compounds a worker might have been exposed to, potential exposures should be assessed for both types of insoluble tritium compounds.

5.7.2 Workers Not Monitored for Soluble Tritium Exposure

Workers at the Pinellas Plant were likely only exposed to on-site levels of environmental radioactivity during the unmonitored periods of their employment.

5.7.3 Workers Monitored for Plutonium Exposure

Because only encapsulated plutonium sources were used at the Pinellas Plant and because there is no indication of any plutonium releases at the site, plutonium intakes were very unlikely at the Pinellas

Plant, and were most likely only limited to the personnel performing the receipt inspections on the plutonium sources. Therefore, no missed plutonium doses need to be assigned for the monitored Pinellas Plant workers when all of their plutonium bioassay results were negative. In the event that a case is encountered where the Pinellas Plant worker had any positive bioassay results, as defined by the criteria below, potential plutonium intakes and doses will need to be assessed and assigned in accordance with the recommendations of ORAUT-OTIB-0060 (ORAUT 2007).

The following are criteria to confirm that a Pinellas Plant plutonium bioassay result is positive:

1. The plutonium bioassay result is greater than the MDC;

2. The plutonium result is not from a preoperational/baseline sample, because preoperational/baseline bioassay measurements are not an indication of exposure;

3. The ^{238}Pu result is greater than the ^{239}Pu result;

4. The sample result is greater than its reported 2σ uncertainty value, such that the lower part of the range does not include zero.

Because uncertainty values are not available for the 1975 and 1976 plutonium bioassay results, the 1975–1976 plutonium bioassay results are considered to be positive if they meet the first three criteria above. Even though some of the documentation on the Pinellas Plant's internal dosimetry practices indicates that the plutonium bioassay results were being calculated at the 95% confidence level (i.e., at a 1.96σ uncertainty), the reported uncertainties were actually calculated at the 95.44% confidence level (i.e., at a 2σ uncertainty) (GE 1975–1986, 1987–1990, Holliday 1983). The equation used for reported MDLs (Equation 5-5 above) indicates that the MDLs were calculated to the 95% confidence level.

If the plutonium bioassay results provided by the DOE are not in units of concentration and do not include the sample volumes analyzed, the dose reconstructor will need to obtain the concentration data from captured Pinellas Plant records. For the years of 1975–1990, compilations of all of the Pinellas Plant's plutonium bioassay results are available in two captured documents (GE 1975–1986, 1987–1990) with the exception of the 1984 data. At the time of this revision, no similar compilation of plutonium bioassay data has been found for 1984. Note that an automated search of the Site Research Database records may have already located the records for a specific claim and provided them as Personnel Exposure files under the DOE files in the NIOSH OCAS Claims Tracking System (NOCTS).

5.7.4 Workers Not Monitored for Plutonium Exposure

As indicated in Sections 5.2.2, 5.3.3.2, and 5.7.3, plutonium intakes were very unlikely at the Pinellas Plant. Therefore, no potential plutonium exposures need to be assessed for Pinellas Plant workers that were not monitored for plutonium exposures.

5.8 INCIDENT HISTORY

Table 5-5 lists a chronology of some of the unusual events that occurred and the resultant radiological releases (if known) from startup to 1989. Most of the releases were through the environmental stack release system, but some elevated tritium levels occurred in other areas.

Table 5-5. Chronology of unusual events and significant activities in relation to internal dosimetry.

Date	Description	Curies of tritium released
12/57	Foot monitor installed in Area 108	
12/10/57	Operator error in reading manometer in Room 18	458
02/11/58	Error estimating amount of tritium remaining in charging system in Room 18	1,253
07/08/58	Glass system breakage in Room 22	280
03/07/58	Glass system breakage in Room 18	567
08/16/58	Operator error with tritium loader valve position – Room 21	780
08/18/58	Glass manifold breakage – Room 21	1,180
02/10/59	Operator error in valve positioning – Room 8	286
02/20/59	Hand contamination – operator not wearing gloves	
02/21/59	Area contamination – operator broke glass system	
03/12/59	Operator contaminated during system cleaning by another worker (Room 14)	
06/04/59	Personnel error working on SECS test system – Room 21	753
06/05/59	Area contamination-diffusion pump exploded in hood 14	
06/18/59	Excess air released from tritium loading system	423
01/1960	Operator error – stopcock left open on tritium loading system	40
02/05/60	Glass manifold broke from strain	72
02/11/60	Operator error – stopcock left open on tritium loading system	308
03/25/60	Operator error caused exposure to 3 employees	
05/14/60	Broken flask caused area contamination – Room 10	
06/21/60	Ion gauge exploded – room 16	
07/08/60	Sample bulb dropped – room 23	6.8
07/13/60	Manifold shattered, exposing worker – room 23	
08/12/60	Contamination spread TiH_2 in area 108 from broken flask	
09/10/60	Requirement for full anti-contamination clothing in Area 108 reduced to lab coats for normal production operations	
11/14/60	Began using NBS Handbook 69 (NBS 1959) for maximum permissible concentrations (MPCs)	
04/61	Area contamination from system breakage	
10/62	A modified personal monitor was installed in Area 108	
12/62	Breathing air supply line connected to Area 108 exhaust duct	
09/63	Employees found falsely identifying urine samples	
03/30/65	Broken flask – room 9	
03/20/65	Flask explosion – room 12	
05/66	SECS cold water removal problems	252
01/27/67	Glove box vacuum pump oil degassed	32
10/12/67	Personnel contamination – O-ring mishandled – room 18	
06/18/68	Acid cleaning explosion – Area 181	
02/69	Leaking flange at absorption pump in Area 108	8
02/69	Area contamination when pump exhaust lines were cut during hood removal – room 2	8
11/05/69	Area contamination in building 400 associated with D-bed	?
11/69	Building 400 cell #3 contaminated with TiH_2 from used flask storage; all flasks moved to burning pad west of 400.	
01/70	Area contamination/personnel exposure from flaking tube part in gas lab	
02/70	Area contamination from pressurized sorb pump (air expansion) – room 2	

Date	Description	Curies of tritium released
11/20/70	Area contaminated when operator used vacuum cleaner on ScH_2 dust in Area 182D	
12/28/70	SECS column saturated due to air leak in area 108	117
03/12/71	Copper gasket uncovered in Room 18 hood – high internal dose	7.3
10/21/71	Tritium release from improperly baked evaporator system in area 182D	129
11/10/71	Area contamination from T loaded disc –Auger spectrometer sample – Area 154	0
12/01/71	High internal exposure – room 18 hood work	1
04/72	Area contaminated from liquid discharge in area 182D	1.5
08/03/72	Leaking absorption pump	12
05/73	Area contaminated with ErH_2 film in Areas 156, 157, and 158	
11/73	Fire in boom box- building 200	0
01/31/75	Improper valve closure on uranium bed	150
02/10/75	Absorption pump leak – area 182D	42
01/30/76	Contaminated 6 inch valve	0
04/13/76	Oven fan blade broke tubes	0
02/77	Packaging fixtures in area 182D glove box	28
09/11/79	Work in Room 18 hood – internal dose	5.7
04/80	Area contaminated from film flaking – Area 158B	0
08/80	Contaminated electron microscope	0
04/81	Three waste drums found out gassing during truck loading were removed to Area 108; a procedure to check all drums for out gassing prior to transfer to the storage building was set up.	
02/25/82	Tritium recovery system (TRS) valve left in wrong position after maintenance	8.6
04/20/82	Operator left TRS valve in wrong position after maintenance – area 108	48
05/24/82	TRS valve left in wrong position after maintenance – area 108, exhaust unit 513	9.5
09/01/82	Sample bulb leak	3
01/05/83	Tritium storage bed oxidation problems	130
01/19/83	Absorption pump leak – area 208	9
04/05/83	Bed heater control failure – Area 108	0
05/84	Tritium air monitor system in Area 182 was rearranged	
07/25/84	Absorption pump sieve dumped into drum in area 108	67
12/09/85	Sorb pump overheat – area contaminated	0
06/24/86	Mass spectrometer oil change – workers exposed to tritium gas	1.5
11/04/87	Test of oxygen regeneration unit	12
02/11/88	Leaking sample bulb in area 108	8
05/05/88	Purge left on over third shift in Area 108 – SECS overpressure	2.7
09/88	Lab area release over 2-week period	16.2
01/06/89	Water in SECS line vented in area 182D	1
09/07/89	Loss of control of radioactive material	0

Several examples of incidents are described to provide perspective on the operational health physics safety responses. On November 5, 1969, Cell #3 in Building 400 was found to be contaminated with HTO from stored leaking uranium beds. Some floors were found to have 4,000 dpm/100 cm^2 of HTO contamination, but the air concentration of the exhaust measured about 600 µCi/m^3. Bioassays of personnel in Building 400 revealed a maximum level of 5.3 µCi/L and a maximum infinite personnel internal dose of 50 mrem (GE 1969).

On September 11, 1979, a release of 5.6 Ci of HTO from the main exhaust stack occurred at approximately 11:43 a.m. The release resulted from nonroutine modifications being made to the uranium bed process system in Hoodroom 18, Area 108. The technician performing the work received a forearm skin exposure that resulted in a body burden of 62.2 µCi/L. This was calculated to 236 mrem. The reasons for the incident included inadequate system modification procedures, a nonfunctional hood monitor, inadequate testing of the hood monitor, and improper use of personal protective equipment. An operator alert system for the HTO monitor was added to the monitoring equipment at about this time in response to this incident. The identification of the problem occurred due to the stack monitor alarm and subsequent security notification to health physicists, who traced the release to Room 108 (Mauer 1979).

On about March 22, 1995, leak cylinders containing hydrogen, Freon, and argon cylinders (which had been sent to an outside vendor) were found contaminated with HTO to a level of about 600,000 dpm/100 cm^2. The vendor was notified, and Pinellas conducted a contamination survey at the vendor's site and provided urinalysis for all requested personnel (Burkhart 1995b).

Each of the above incidents identifies that Pinellas took some routine operational actions to maintain exposures below the applicable limits of the time. Routine air monitoring and contamination monitoring usually led to identification of problems. It is evident that if personnel were suspected of internal exposure to tritium, they were promptly asked to submit a urine sample. Work orders quite often required bioassays after completion of the work (GE 1992, 1978–1989). In addition, outside contractors were given tritium bioassays from the start of Pinellas operations in 1957.

5.9 ATTRIBUTIONS AND ANNOTATIONS

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

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GLOSSARY

activity median aerodynamic diameter (AMAD)
The diameter of a unit density sphere with the same terminal settling velocity in air as that of the aerosol particle whose activity is the median for the entire aerosol.

acute exposure
Radiation exposure to the body delivered in a short period. See *chronic exposure*.

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

beta particle
See *beta radiation*.

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

bioassay
Measurement of amount or concentration of radioactive material either in the body or in biological material excreted or removed from the body. Another word for *radiobioassay*.

bioassay procedure
A procedure used to determine the kind, quantity, location, and retention of radionuclides in the body by direct (*in vivo*) measurements or by *in vitro* analysis of material excreted or removed from the body.

body burden
The quantity of radioactive material contained in the individual's body at a particular point in time.

chronic
Pertaining to low-level intakes received on a continuous basis.

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

derived investigation level (DIL)
A value based on the regulatory control level and the minimum detectable concentration of the assay method.

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

dose equivalent (H)

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

exposure

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

gamma radiation

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

insoluble tritium

Less soluble forms of tritium, which have type M or S lung absorption properties (DOE 2006).

intake

The amount of radionuclide taken into the body by inhalation, absorption through intact skin, injection, ingestion, or through wounds. Depending on the radionuclide involved, intakes may be reported in units of mass, activity, or potential alpha energy.

internal dose or exposure

The dose equivalent received from radioactive material taken into the body (i.e., internal sources).

internal dose assessment

An assessment of the intake and associated internal radiation dose to workers based on measurements taken in the work environment or from individual bioassay measurements.

***In vitro* measurement**

Measurements to determine the presence of or to estimate the amount of radioactive material in the excreta or in other biological materials removed from the body.

***In vivo* measurement**

The measurement of radioactive material in the human body utilizing instrumentation that detects radiation emitted from the radioactive material in the body.

ionizing radiation

Radiation of high enough energy to remove an electron from a struck atom and leave behind a positively charged ion. High enough doses of ionizing radiation can cause cellular damage. Ionizing particles include alpha particles, beta particles, gamma rays, X-rays, neutrons, high-speed electrons, high-speed protons, photoelectrons, Compton electrons, positron/negatron pairs from photon radiation, and scattered nuclei from fast neutrons.

isotope

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

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

limit of detection (LOD)
Minimum level at which a particular device can detect and quantify exposure or radiation. Also called lower limit of detection and detection limit or level.

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

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

metal tritide
Metals that absorb tritium atoms in the crystalline structure of the metal. Metal hydrides and tritides are the most compact way to store hydrogen or tritium. Because of that, metal tritides are most often used as a method of retaining or storing tritium.

minimum detectable amount
The smallest amount (activity or mass) of an analyte in a sample that will be detected with a probability, β, of non-detection (Type II error) while accepting a probability, α, of erroneously deciding that a positive (non-zero) quantity of analyte is present in an appropriate blank sample (Type I error).

minimum detectable concentration (MDC)
The minimum detectable amount expressed in units of concentration.

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

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

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

occupational dose
An individual's ionizing radiation dose (external and internal) resulting from that individual's work assignment. Occupational dose does not include doses received as a medical patient or

doses resulting from background radiation or participation as a subject in medical research programs.

organically bound tritium (OBT)
A type of tritiated material in which the tritium has formed a chemical bond with an organic material, typically via a carbon-tritium bond.

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

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

rad
Traditional unit for expressing absorbed radiation dose, which is the amount of energy from any type of ionizing radiation deposited in any medium. A dose of 1 rad is equivalent to the absorption of 100 ergs per gram (0.01 joules per kilogram) of absorbing tissue. The rad has been replaced by the gray in the International System of Units (100 rads = 1 gray). The word derives from radiation absorbed dose.

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

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

radioisotopically-powered thermoelectric generator (RTG)
Generator that obtains its power from passive (natural) radioactive decay using thermocouples to convert the heat of decay into electricity.

reporting level
A value below which data or results were considered to be too low to record and thus may not have been maintained. For example, when the reporting level was "<0.67 µCi/L" the sample result was only reported as "<0.67", or when the reporting level was "0.01 µCi/L" any sample result below that value was reported as "0.00" (zero).

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

routine monitoring
Monitoring carried out at regular intervals during normal operations.

sievert

The special name for the International System unit of dose equivalent. One sievert equals 1 joule per kilogram, which equals 100 rem.

soluble tritium

All forms of tritium, except for those that have type M or S lung absorption properties.

special monitoring

Monitoring carried out in actual or suspected abnormal conditions (i.e., measurements performed to estimate the amount of radionuclide deposited in a person when an intake is known or is suspected to have occurred).

spot sample

A single void of urine.

tritium

A radioactive isotope of hydrogen, which has two neutrons. It is also known as or denoted as hydrogen-3, ^{3}H, H-3, and T.

U.S. Atomic Energy Commission (AEC)

Federal agency created in 1946 to assume the responsibilities of the Manhattan Engineer District (nuclear weapons) and to manage the development, use, and control of nuclear energy for military and civilian applications. The U.S. Energy Research and Development Administration and the U.S. Nuclear Regulatory Commission assumed separate duties from the AEC in 1974. The U.S. Department of Energy succeeded the U.S. Energy Research and Development Administration in 1979.

X-ray

See *X-ray radiation*.

X-ray radiation

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