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PUBLICATION RECORD

EFFECTIVE DATE	REVISION NUMBER	DESCRIPTION
10/07/2005	00	New technical basis document for the Pinellas Plant – Occupational Internal Dose. First approved issue. Training required: As determined by the Task Manager. Initiated by Mark D. Notich.
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.
02/21/2012	02	With the exception of Section 5.2.2, all plutonium discussions were eliminated from this TBD, discussed at the October 2011 working group meeting for the Pinellas Plant Site Profile. Because internal exposures to plutonium were unlikely at the Pinellas Plant, the plutonium discussions after Section 5.2.2 were determined to be unnecessary and a potential source of confusion. 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.
07/18/2016	03	Sections 5.7.1.1 and 5.7.1.2 were modified to incorporate changes agreed upon by the Pinellas Plant Working Group. Additionally, Section 5.3.1 was modified to provide more details about the tritium contamination smear collection and analysis procedures. Sections 5.2.1 and 5.3.1 were modified to provide additional details, which included changes to the References Section. Section 5.2.1 was modified to provide additional details regarding organically bound tritium. Section 5.4.2 was modified to provide additional details regarding the analysis of the tritium bioassay samples. Incorporates formal internal and NIOSH review comments. Training required: As determined by the Objective Manager. Initiated by Brian P. Gleckler.

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ACRONYMS AND ABBREVIATIONS

AEC	U.S. Atomic Energy Commission
AWE	atomic weapons employer
Bq	becquerel
Ci	curie
cm	centimeter
d	days
DOE	U.S. Department of Energy
DOL	U.S. Department of Labor
dpm	disintegrations per minute
DU	depleted uranium
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
hr	hour
HT	tritium gas
HTO	tritium oxide (also known as tritiated water)
ICRP	International Commission on Radiological Protection
IMBA	Integrated Modules for Bioassay Analysis
in.	inch
keV	kiloelectron-volt, 1,000 electron-volts
L	liter
LMSC	Lockheed Martin Specialty Components
m	meter
MDC	minimum detectable concentration
mL	milliliter
MMSC	Martin Marietta Specialty Components
MPC	maximum permissible concentration
MT	metal tritide
mrem	millirem
NBS	National Bureau of Standards
NIOSH	National Institute for Occupational Safety and Health
NRC	U.S. Nuclear Regulatory Commission
OBT	organically bound tritium
ORAU	Oak Ridge Associated Universities

pCi	picocurie
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
RTG	radioisotopically-powered thermoelectric generator
SAR	safety analysis report
SEC	Special Exposure Cohort
SRDB Ref ID	Site Research Database Reference Identification (number)
Sv	sievert
T	tritium
T_2	tritium gas
T_2O	tritium oxide
TBD	technical basis document
TRS	Tritium Recovery System
U.S.C.	United States Code
yr	year
µCi	microcurie
§	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 historical background information and guidance to assist in the preparation of dose reconstructions at particular Department of Energy (DOE) or Atomic Weapons Employer (AWE) facilities or categories of DOE or AWE facilities. They will be revised in the event additional relevant information is obtained about the affected DOE or AWE facility(ies). These documents may be used to assist NIOSH staff in the evaluation of Special Exposure Cohort (SEC) petitions and the completion of the individual work required for each dose reconstruction.

In this document the word "facility" is used to refer to an area, building, or group of buildings that served a specific purpose at a DOE or AWE facility. It does not mean nor should it be equated to an "AWE facility" or a "DOE facility." The terms AWE and DOE facility are defined in sections 7384l(5) and (12) of the Energy Employees Occupational Illness Compensation Program Act of 2000 (EEOICPA), respectively. An AWE facility means "a facility, owned by an atomic weapons employer, that is or was used to process or produce, for use by the United States, material that emitted radiation and was used in the production of an atomic weapon, excluding uranium mining or milling." 42 U.S.C. § 7384l(5). On the other hand, a DOE facility is defined 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 [DOE] (except for buildings, structures, premises, grounds, or operations … pertaining to the Naval Nuclear Propulsion Program);" and with regard to which DOE has or had a proprietary interest, or "entered into a contract with an entity to provide management and operation, management and integration, environmental remediation services, construction, or maintenance services." 42 U.S.C. § 7384l(12). The Department of Energy (DOE) determines whether a site meets the statutory definition of an AWE facility and the Department of Labor (DOL) determines if a site is a DOE facility and, if it is, designates it as such.

Accordingly, a Part B claim for benefits must be based on an energy employee's eligible employment and occupational radiation exposure at a DOE or AWE facility during the facility's designated time period and location (i.e., covered employee). After DOL determines that a claim meets the eligibility requirements under EEOICPA, DOL transmits the claim to NIOSH for a dose reconstruction. EEOICPA provides, among other things, guidance on eligible employment and the types of radiation exposure to be included in an individual dose reconstruction. Under EEOICPA, eligible employment at a DOE facility includes individuals who are or were employed by DOE and its predecessor agencies, as well as their contractors and subcontractors at the facility. Unlike the abovementioned statutory provisions on DOE facility definitions that contain specific descriptions or exclusions on facility designation, the statutory provision governing types of exposure to be included in dose reconstructions for DOE covered employees only requires that such exposures be incurred in the performance of duty. As such, NIOSH broadly construes radiation exposures incurred in the performance of duty to include all radiation exposures received as a condition of employment at covered DOE facilities in its dose reconstructions for covered employees. For covered employees at DOE facilities, individual dose reconstructions may also include radiation exposures related to the Naval Nuclear Propulsion Program at DOE facilities, if applicable. No efforts are made to determine the eligibility of any fraction of total measured exposure for inclusion in dose reconstruction.

NIOSH does not consider the following types of exposure as those incurred in the performance of duty as a condition of employment at a DOE facility. Therefore these exposures are not included in dose reconstructions for covered employees (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

5.2 OVERVIEW

This technical basis document (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. For convenience, this TBD refers to 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 2011).

The Pinellas 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, radioisotopically-powered 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).

5.2.1 PURPOSE

The purpose of this 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.

5.2.2 SCOPE

This TBD provides supporting documentation to assist in the evaluation of occupational internal doses in accordance with OCAS-IG-002, *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.

5.3 RADIOACTIVE SOURCE TERM AND LUNG ABSORPTION TYPES

5.3.1 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 types of tritium compounds were present at the Pinellas Plant: tritiated water (HTO or T_2O), tritium gas (HT or T_2), organically bound tritium (OBT), and metal tritides (MTs). Based on the available records, the predominant tritium exposure hazard to the workers was from HTO and HT. However, in some circumstances workers might have also been exposed to OBT compounds and MTs.

The most common OBT compounds at the Pinellas Plant were contaminated pump oils and organic solvents (e.g., alcohol, toluene, acetone, etc.) that were used in the tritium areas. Turbo and vacuum pumps that were used in systems at the Pinellas Plant periodically required maintenance, which included changing the oil. When those pumps were located in tritium areas, the tritium could contaminate the pump oil "through small liberations during process operations." After the oil was drained from a pump, a sample of the oil was collected and analyzed for tritium. The nonvolatile radioactive liquid wastes not miscible in water, such as the contaminated pump oils, were packaged for shipment to an offsite disposal area. The contaminated pump oils were accumulated and stored in Area 108 prior to disposal. Based on the available records, organic solvents for cleaning and degreasing tritium contaminated components were likely the most common source of tritium-contaminated organic solvents. The volatile contaminated organic solvents were routinely disposed of by evaporation in a hood that was exhausted to the effluent control system and the main stack (Ward 1973; Barfield and Johnson 1976; DOE 1987; MMSC 1995, p. 35; LMSC 1997).

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 MT uses, only a small portion of the worker population had the potential to be exposed to a dispersible form of an MT, and those scenarios were typically limited to accidents involving MTs. Titanium tritide was also used in the original tritium storage beds at the Pinellas Plant (Burkhart 1990). In the original tritium storage beds, the titanium tritide was sealed inside a glass cylinder (Burkhart 1990). 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, depleted uranium (DU) tritide (Burkhart 1990; Eichman 1979; 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.

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	Tube testing
131	Final tube test	Final tube testing
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	General development
191	Ceramic Product Engineering (CPE) hood room	Ceramic product testing facility
191	Radioanalytical laboratory	Radiological Laboratory, tritium recovery
194	Engineering environmental testing	Engineering environmental testing
200	Test areas	Test areas
800	Accelerator and calibration	Accelerator and calibration
1000	Waste storage	Waste storage

Source: DOE (1995, p. 31); Weaver (1993).

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.

5.3.2 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 undated a). 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 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 source 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 1989).

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. The available information indicates that Pinellas Plant did not receive enough contaminated $^{238}PuO_2$ heat sources for a worker performing the receipt inspections to receive more than a negligible total internal plutonium dose to any organ (i.e. more than 0.001 rem). Because of that and because all other sources of plutonium at the Pinellas Plant were in non-dispersible forms (e.g. encapsulated sources, electro-plated sources, etc…), it is unlikely that any workers at the Pinellas Plant received more than a negligible internal exposure to plutonium, and potential plutonium exposures do not need to be assessed for Pinellas Plant workers. As a result, plutonium is not discussed any further in this TBD.

5.3.3 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 MT 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 DU (mainly ^{238}U) inside the tritium storage beds presented 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.

5.3.4 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.

5.3.5 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.

5.3.6 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.

5.3.7 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.

5.4 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 at the Pinellas Plant to detect potential intakes of radioactive materials.

5.4.1 TRITIUM CONTAMINATION MONITORING

The monthly Health Physics Reports for the Pinellas Plant indicate that contamination monitoring for tritium was performed on a routine basis from the beginning of operations at the Pinellas Plant (GE 1957–1967, GE 1957–1973, GE 1963, GE 1967). 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 monthly reports also indicated that whenever contamination levels were greater than the contamination control limits, decontamination of those areas was initiated. As early as 1959, the contamination control limit for tritium was 2×10^{-5} µCi/in^2 (688 dpm/100 cm^2), which was for radiologically controlled areas. A circa 1966 smear survey procedure indicated that the tritium contamination limit for uncontrolled areas was 220 dpm/100 cm^2, and that the limit for radiologically controlled areas had been reduced to 440 dpm/100 cm^2 (GE ca. 1966, Burkhart 1989).

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 1957–1967, GE 1957–1973, GE 1963, GE 1967). The next highest value was reported in 1959, 1.4×10^6 dpm/100 cm^2 (3.3×10^{-2} µCi/in^2) (GE 1957–1967, GE 1957–1973, GE 1963, GE 1967). The majority of the other reported maximum surface contamination levels are at least an order of magnitude lower than these two.

Early on, it was known that wet smears (a.k.a. swipes) for tritium contamination were more efficient than dry smears. The water geometry for counting was used with the liquid scintillation counter for contamination monitoring. Two undated and slightly different tritium contamination smear collection and analysis procedures have been found for the Pinellas Plant. The only differences between the procedures were the amounts of deionized water being added at the various steps. Both procedures utilized a wetted cotton ball for collecting the contamination smear samples. To collect a contamination smear sample, 2 or 5 mL of distilled water was pipetted into a paper cup containing a cotton ball. The wetted cotton ball was then used to smear the area of interest for contamination (usually an area of 100 to 1,000 cm^2), and placed back into the paper cup. The contamination smear was then taken to the counting laboratory, where 8 or 10 mL of deionized water was added to the cotton ball in the paper cup. The paper cup was then crushed to squeeze out the sample liquid. The sample liquid was filtered through a Whatman #1 filter paper before letting it pass into a new paper cup. Next, a 0.2-mL aliquot of the filtrate was added to 16.5 mL of liquid scintillation cocktail solution, and the sample was counted for 5 minutes (GE 1967–1982). The Whatman #1 filter paper was likely used to filter out the large nonrespirable particles from the sample liquid so that the large particles would not interfere with the liquid scintillation counting step. The Whatman #1 filter paper has a nominal pore size of 11 µm, which would have allowed the respirable tritium particles (i.e., ≤10 µm) to remain in the filtrate. By 1977, the sample liquid from tritium contamination smears was no longer being filtered (ORAUT 2013a).

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 1989; LMSC 1995).

5.4.2 TRITIUM AIR SAMPLING

Monitoring for airborne tritium 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 (MPC) 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-hour 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 (NRC) (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.

5.4.3 BIOASSAY MONITORING PRACTICES

Tritium and plutonium were the only radioactive materials that bioassay monitoring was routinely performed for. However, plutonium bioassay monitoring is not discussed in this TBD for the reasons indicated in Section 5.2.2. Nonroutine bioassay monitoring might have been performed for other radioactive materials; however, no documentation regarding other potential types of bioassay monitoring has been found.

5.4.3.1 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.

5.4.3.2 Tritium Bioassay Monitoring

Since about 1986, the bioassay program at the Pinellas Plant 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 1984b, 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).

The Pinellas Plant 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.

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), International Commission on Radiological Protection, 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).

$$DCF_{\infty} = \frac{V_{BW}\left(2.22\times10^{6}\,\frac{\text{dpm}}{\mu\text{Ci}}\right)\left(1{,}440\,\frac{\text{min}}{\text{day}}\right)\left(5.7\times10^{-3}\,\frac{\text{MeV}}{\text{disintegration}}\right)\left(1.6\times10^{-6}\,\frac{\text{erg}}{\text{MeV}}\right)\left(1{,}000\,\frac{\text{mrem}}{\text{rem}}\right)}{M_{BW}\,\lambda_{E}\,QF\left(100\,\frac{\text{erg/g}}{\text{rad}}\right)} \quad (5\text{-}1)$$

$$V_{BW} = \frac{M_{BW}\left(1\times10^{-3}\,\frac{\text{L}}{\text{mL}}\right)}{d_{W}} = 42L \quad (5\text{-}2)$$

$$\lambda_{E} = \frac{\ln(2)}{T_{½E}} \quad (5\text{-}3)$$

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).

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

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).

5.5 IN VITRO BIOASSAY DETECTION LEVELS AND ANALYSIS METHODS

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

5.5.1 TRITIUM URINALYSIS DETECTION LEVELS

Tritium urinalysis is capable of detecting intakes only of soluble forms of tritium (i.e., tritium gas, tritiated water, uranium tritide, and certain organically bound tritium compounds). 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.

5.5.2 TRITIUM ANALYSIS METHODS FOR IN VITRO URINE SAMPLES

Liquid scintillation counting was used to analyze the urine samples for tritium. Urine samples were initially analyzed by the Health Physics Laboratory using a Tri-Carb liquid scintillation counter (GE 1964; GE 1967–1982, pp. 52–98; Johnson 1977). Based on a sample analysis form from 1964, a urine sample prepared for liquid scintillation counting 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 2,5-diphenyloxazole (PPO), and 0.12 g 1,4-bis[5-phenyloxazol-2-yl]benzene (POPOP) (GE 1964 pp. 89–97). 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, 1967–1982). 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).

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; Author unknown (1973); GE (1968, 1971, undated a,b,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 the Oak Ridge Associated Universities (ORAU) Team 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).

By the late-1960s, an analysis procedure indicates that 1 teaspoon of charcoal was being added to each urine sample, to decolorize the solution, and the solution was filtered through a Whatman #1 filter paper before adding the 0.2-mL urine aliquot to the scintillation mix. The individual urine samples were counted in the Tri-Carb liquid scintillation counter for 1 minute. If the 1-minute count was above the reporting level, the sample was recounted for 5 minutes. The net count rate for the sample was determined by subtracting a background count rate before calculating the tritium concentration of the urine sample (GE 1967–1982, pp. 52–98).

The background count rates were obtained from uncontaminated urine samples, which were usually obtained from new employees or individuals who did not come into contact with radioactivity. Except for the count time, the background urine samples appear to have been prepared and analyzed like the other urine samples. The count time for the background urine samples was 100 minutes to reduce the statistical error. The results from the background urine samples were recorded on a Counter Check Log (GE 1967–1982, pp. 72–74).

By 1973, the Health Physics Laboratory was using a Beckman Model LS-100 liquid scintillation spectrometer to analyze the urine samples (Johnson 1977; Ward 1973). On the Beckman counter, the preset error control for channel 1 was set to 15%, the preset time was set to 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 0.67-µCi/L reporting level, the result was verified by processing and counting a second sample for 5 minutes with a preset error ±5% (GE 1967–1982, pp. 220–221).

Sometime between 1977 and 1986, the analysis of urine samples transitioned from the Health Physics Laboratory to the Environmental Chemistry Laboratory (a.k.a., ECL and Environmental Health Section of the Chemistry Laboratory) (Johnson 1977; Molloy 1987). The ECL used two Beckman liquid scintillation counting systems, one for high-level radioactivity samples (20 to 1,000,000 cpm) and one for low-level radioactivity samples (1 to 30 cpm). Urine samples were evaluated as high-level samples. In 1986, the ECL modified and evaluated their two Beckman liquid scintillation counting systems to make them more automated and help with the increasing sample analysis loads (Molloy 1987).

By 1977, the ECL was using Handifluor, a premixed scintillation cocktail, for some samples (GE 1976–1987). In 1982, Ready-Solv EP, another premixed scintillation cocktail, was evaluated and approved for use for in-plant tritium samples; it had a triton/toluene base (Barfield 1982). The use of premixed scintillation cocktails eliminated the need for the laboratory to spend time creating those mixes. However, as with the earlier cocktails, the premixed cocktails still used flammable organically based solvents, which created mixed hazardous waste (i.e., both radioactive and flammable wastes for spent tritium samples). In 1986, Opti-Fluor, a water soluble premixed cocktail that uses a higher flash point solvent, was evaluated and approved for use as a viable substitute for the more flammable cocktails. Tritium contaminated liquid scintillation samples with Opti-Fluor were not considered to be mixed hazardous waste when the samples were no longer needed (Klee 1986).

An undated procedure that appears to be from the late 1960s indicates that daily urine samples from multiple workers were composited for screening purposes (GE 1967–1982, pp. 52–98). However, the exact use of those screening sample results is uncertain. Based on a review of dosimetry records in the Site Research Database (GE 1957 to 1960, 1964, 1966, 1968, 1971, undated b, GE undated c) and Pinellas Plant claims, there is no indication of composited sample results being provided in the workers' dosimetry records.

5.6 IN VIVO MINIMUM DETECTABLE ACTIVITIES AND ANALYSIS METHODS

There was no in vivo monitoring at the Pinellas Plant.

5.7 INTERFERENCES AND UNCERTAINTIES

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 a).

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 (GE 1986). 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.

5.8 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.

5.8.1 WORKERS MONITORED FOR SOLUBLE TRITIUM EXPOSURE

5.8.1.1 Assessment of Soluble Tritium Exposures

For the periods that a worker was monitored for soluble tritium exposures, the potential exposures to HT, HTO, and OBT should be assessed using the worker's urine sample data. Because there is insufficient information to determine what fraction of a worker's tritium exposure was attributable to OBT compounds, both 100% HT and HTO doses and 100% OBT doses should be assessed, and then only the most favorable to the claimant dose should be assigned.

Potential exposures to HT and HTO should be assessed assuming that 100% of the tritium intake was attributable to HTO because there is insufficient information to determine what fractions of a worker's tritium exposure were attributable to HT and HTO. Between HT and HTO, HTO is the more dosimetrically significant form of tritium. The calculations should be performed using the Excel® workbook *Tritium Doses from Urine Data Workbook* in accordance with the recommendations in ORAUT-OTIB-0011, *Technical Information Bulletin: Tritium Calculated and Missed Dose Estimates* (ORAUT 2004b).

Potential exposures to OBT should be assessed using the Integrated Modules for Bioassay Analysis (IMBA) computer code, and assuming 100% of the tritium intake is attributable to OBT. The calculations should be performed in accordance with the recommendations in ORAUT-OTIB-0060, *Internal Dose Reconstruction* (ORAUT 2014). The following is additional guidance for using the IMBA computer code to assess OBT intakes and doses. When the available information indicates that the intake mode is inhalation, "Hydrogen-3 (organic)" should be selected for the radionuclide, "Vapor" should be selected as the route, then the "SR-2" vapor class option (the only option) should be selected in "ICRP Defaults." When the available information indicates that the intake mode is ingestion, "Hydrogen-3 (organic)" should be selected for the radionuclide, "Ingestion" should be selected as the route, then the "Ing" absorption option (the only option) should be selected in "ICRP Defaults." When the intake mode is unknown, inhalation should be the assumed intake mode, in accordance with ORAUT-OTIB-0060.

5.8.1.2 Assessment of Potential Unmonitored Exposures to Insoluble Tritium

At the Pinellas Plant, insoluble tritium compounds (i.e., certain MTs) were handled only in areas where the more dispersible and more soluble forms of tritium (e.g., HT, 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 insoluble tritium 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.

For the Pinellas Plant, little individual contamination smear data is available. However, the monthly Health Physics Reports summarized the status of the radioactive contamination control program and reported the highest contamination levels measured for a given month. Therefore, all of the available Health Physics Reports (i.e. the various reports for 1957–1973) were reviewed to determine the tritium contamination controls at the Pinellas Plant (GE 1957–1967, GE 1957–1973, GE 1963, GE 1967). These reports indicate that whenever contamination levels were greater than the contamination control limits, decontamination was initiated in those areas. The reports also confirmed that a routine contamination control program was in place throughout the history of the site and that it would have been unlikely for high contamination levels to exist for extended periods. As early as 1959, the contamination control limit for tritium was 2×10^{-5} µCi/in^2 (688 dpm/100 cm^2), which was for radiologically controlled areas. A circa 1966 smear survey procedure indicated that the tritium contamination limit for uncontrolled areas was 220 dpm/100 cm^2, and that the limit for radiologically controlled areas had been reduced to 440 dpm/100 cm^2 (GE ca. 1966, Burkhart 1989). An interview of a former Pinellas Plant Health Physicist also confirms those radioactive contamination control practices. According to him, radiological control personnel would take contamination smears in the morning and, if contamination was identified, they would then mop up the area and re-survey. The likelihood of routine surface contamination level in the millions of dpm/100 cm^2 should be considered unusual and short in duration. It was also stated that “metal tritide contamination was cleaned up fairly quickly” (ORAUT 2013c).

The monthly Health Physics Reports also provide information on the maximum tritium surface contamination levels. Between 1957 and 1973, the highest reported surface contamination level was 4.4×10^6 dpm/100 cm^2 (10,000 times the control limit), which occurred in 1970 (ORAUT 2016). The next highest value reported was 1.4×10^6 dpm/100 cm^2 (3.3×10^{-2} µCi/in^2), which occurred in 1959 (ORAUT 2016). The majority of the rest of the reported annual maximum surface contamination levels are at least an order of magnitude lower than these two. However, the following assumptions are made to bound the potential unmonitored insoluble tritium exposures:

- Constant tritium surface contamination level of 4.4×10^6 dpm/100 cm^2 (4.4×10^8 dpm/m^2) (ORAUT 2016);
- Resuspension factor of 5×10^{-5}/m (ORAUT 2013b).

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 with 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 intake rates provided in Table 5-4 should be used to account for potential insoluble tritium intakes for the periods when 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 (ORAUT 2016).

Hours worked per year[a]	Annual inhalation (pCi/yr)	Annual ingestion[c] (pCi/yr)	Daily[b] inhalation (pCi/day)	Daily ingestion[c] (pCi/day)
2,600	3.09E+07	5.80E+05	8.47E+04	1.59E+03

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 day/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, and then only the most favorable to the claimant dose should be assigned.

As indicated above, unmonitored intakes of insoluble tritium will be assessed for all of the periods that workers were monitored for soluble tritium exposures. However, in addition to the two bounding assumptions mentioned above, there are other reasons that the assessment of those chronic unmonitored insoluble tritium intakes will likely result in overestimates of the workers' exposures to insoluble tritium. Those reasons include the following:

- the available information on the Pinellas Plant indicates that only a relatively small portion of the workforce had the potential to be exposed to dispersible forms of insoluble tritium;
- there is no indication of routine insoluble tritium exposure scenarios (chronic exposures), and the known exposure scenarios were typically limited to accidents (acute exposures);
- most of the tritium contamination data that was used for the unmonitored insoluble tritium intakes were based on results for soluble tritium, which were also the highest contamination survey results;
- as indicated above, the highest reported surface contamination level was 4.4×10^6 dpm/100 cm^2, which was actually for soluble tritium; whereas, the highest known insoluble tritium contamination level was only 1.89×10^5 dpm/100 cm^2 (i.e. only about 4% of the surface contamination level used to estimate the unmonitored insoluble tritium intakes) (GE 1957–1973).

Even though the unmonitored intakes above likely result overestimates of the workers' exposures to insoluble tritium, those unmonitored intakes are still considered to be reasonable estimates.

5.8.2 WORKERS NOT MONITORED FOR SOLUBLE TRITIUM EXPOSURE

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

5.9 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.

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 leaking uranium storage 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).

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	Not applicable
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	Not applicable
02/21/59	Area contamination – operator broke glass system	Not reported
03/12/59	Operator contaminated during system cleaning by another worker (Room 14)	Not reported
06/04/59	Personnel error working on Stack Effluent Control System (SECS) test system – Room 21	753
06/05/59	Area contamination-diffusion pump exploded in Hood 14	Not reported
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	Not reported
05/14/60	Broken flask caused area contamination – Room 10	Not reported
06/21/60	Ion gauge exploded – Room 16	Not reported
07/08/60	Sample bulb dropped – Room 23	6.8
07/13/60	Manifold shattered, exposing worker – Room 23	Not reported
08/12/60	Contamination spread TiH_2 in Area 108 from broken flask	Not reported
09/10/60	Requirement for full anticontamination clothing in Area 108 reduced to laboratory coats for normal production operations	Not reported
11/14/60	Began using NBS Handbook 69 (NBS 1959) for MPCs	Not applicable
04/61	Area contamination from system breakage	Not reported
10/62	A modified personal monitor was installed in Area 108	Not applicable
12/62	Breathing air supply line connected to Area 108 exhaust duct	Not applicable
09/63	Employees found falsely identifying urine samples	Not applicable
03/30/65	Broken flask – Room 9	Not reported
03/20/65	Flask explosion – Room 12	Not reported
05/66	SECS cold water removal problems	252
01/27/67	Glovebox vacuum pump oil degassed	32
10/12/67	Personnel contamination – O-ring mishandled – Room 18	Not reported
06/18/68	Acid cleaning explosion – Area 181	Not reported
02/69	Leaking flange at absorption pump in Area 108	8

Date	Description	Curies of tritium released
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	Not reported
11/69	Building 400 cell #3 contaminated with TiH_2 from used flask storage; all flasks moved to burning pad west of 400.	Not reported
01/70	Area contamination and personnel exposure from flaking tube part in gas lab	Not reported
02/70	Area contamination from pressurized sorb pump (air expansion) – Room 2	Not reported
11/20/70	Area contaminated when operator used vacuum cleaner on ScH_2 dust in Area 182D	Not reported
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 tritium-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
05/24/73	Areas 156, 157, and 158b contaminated with tritium due to an MT particle spill in Area 158b. The MT particles were from a zirconium/aluminum getter.	Not reported
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-in. valve	0
04/13/76	Oven fan blade broke tubes	0
02/77	Packaging fixtures in Area 182D glovebox	28
09/11/79	Work in Room 18 hood – internal dose	5.6
04/80	Area contaminated from film flaking – Area 158B	0
08/80	Contaminated electron microscope	0
04/81	Three waste drums found outgassing during truck loading were removed to Area 108; a procedure was put in place to check all drums for outgassing before transfer to the storage building.	Not reported
02/25/82	Tritium recovery system (TRS) valve left in wrong position after maintenance	8.6
04/20/82	TRS valve left 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	Not applicable
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	Laboratory 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

On [Redacted], a release of 5.6 Ci of HTO from the main exhaust stack occurred at approximately [Redacted]. The release resulted from nonroutine modifications being made to the [Redacted] system in [Redacted] of [Redacted]. Analyses of urine samples from the technician performing the work

indicated that the worker received a body burden of 62.2 µCi/L, which the Pinellas Plant equated to a dose of 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 problem was identified due to the stack monitor alarm and subsequent security notification to health physicists, who traced the release to [Redacted] of [Redacted] (Maurer 1979).

In 1995, several leak standards (cylinders containing hydrogen, Freon, and argon), 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 the Pinellas Plant conducted a contamination survey at the vendor's site and provided urinalyses for all requested personnel (Burkhart 1995b).

Each of the above incidents indicates that the Pinellas Plant 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 plant operations in 1957.

5.10 ATTRIBUTIONS AND ANNOTATIONS

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

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GLOSSARY

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

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 radionuclide material in the body (in vivo measurement) or in biological material excreted or removed from the body (in vitro measurement) and analyzed for purposes of estimating the quantity of radioactive material in the body. Also called radiobioassay.

bioassay procedure
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
Amount of radioactive material in an individual's body at a particular point in time.

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

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

dose equivalent
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.

intake

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

internal dose or exposure

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

internal dose assessment

Estimation of an intake of radioactive material and the consequent radiation dose based on bioassay or other measurements in the work environment.

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

Measurements 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.

kiloelectron-volt (keV)

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

limit of detection

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
See *absorption* type.

metal tritide (MT)
Metal 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 activity or amount
Smallest amount (activity or mass) of an analyte in a sample that can be detected with a probability β of nondetection (Type II error) while accepting a probability α of erroneously deciding that a positive (nonzero) quantity of analyte is present in an appropriate blank sample (Type I error).

minimum detectable concentration (MDC)
Minimum detectable activity (or amount) in units of concentration. See *minimum detectable activity*.

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

organically bound tritium (OBT)
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

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

soluble tritium

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

special monitoring

Monitoring in addition to the routine monitoring program carried out for special purposes such as estimating the amount of radionuclide deposited in a person after a known or suspected accidental intake or after a known or suspected environmental release. See *routine monitoring*.

spot sample

In relation to bioassay, a single void of urine.

tritium

Radioactive isotope of hydrogen that contains one proton and two neutrons in its nucleus. It decays by beta emission and has a radioactive half-life of about 12.5 years.

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.