



ORAU TEAM Dose Reconstruction Project for NIOSH

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ACRONYMS AND ABBREVIATIONS

| | |
|--------|---|
| AMAD | activity median aerodynamic diameter |
| AEC | U.S. Atomic Energy Commission |
| CF | calibration factor |
| cm | centimeter |
| cpm | counts per minute |
| DIL | derived investigation level |
| DOE | U.S. Department of Energy |
| dpm | disintegrations per minute |
| DU | depleted uranium |
| EDE | effective dose equivalent |
| g | gram |
| HEPA | high-efficiency particulate air |
| hr | hour |
| ICRP | International Commission on Radiological Protection |
| keV | kilovolt-electron, 1,000 electron volts |
| HTO | Tritiated water vapor |
| L | liter |
| LSC | liquid scintillation counting |
| MDC | minimum detectable concentration |
| MDL | minimum detectable level |
| MeV | megavolt-electron, 1 million electron volts |
| min | minute |
| ml | milliliter |
| mR | milliroentgen |
| MT | Metal tritide |
| mrad | millirad |
| mrem | millirem |
| NIST | National Institute of Standards and Technology |
| RBE | relative biological effectiveness |
| ROI | region of interest |
| T | Tritium |
| U.S.C. | United States Code |
| μCi | microcurie |
| μm | micrometer |

5.1 INTRODUCTION

Technical Basis Documents and Site Profile Documents are general working documents that provide guidance concerning the preparation of dose reconstructions at particular sites or categories of sites. They will be revised in the event additional relevant information is obtained about the affected site(s). These documents may be used to assist the National Institute for Occupational Safety and Health 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 of 2000 [42 U.S.C. Sections 7384l(5) and (12)].

The Pinellas Plant internal dosimetry program started with site operations in 1957. Tritium “(T)” and ²³⁸Pu and ²³⁹Pu were the isotopes of primary concern and the only radionuclides monitored routinely. The primary method for monitoring of employees for radionuclide intake was urine bioassay. Pinellas used ¹⁴C and ⁶³Ni, but in much smaller quantities; therefore, Pinellas did not perform bioassays for these radionuclides.

5.2 IN VITRO MINIMUM DETECTABLE CONCENTRATIONS AND COUNTING METHODS

5.2.1 Plutonium Minimum Detectable Concentration Determination and Counting Methods

The minimum detectable concentration (MDC) calculation was based on the formulation in Equation 5-1 for the Pinellas plutonium urinalysis program conducted from 1975 through 1990. The Pinellas facility used MDCs as the decision levels. Pinellas labeled the MDC as *MDL* (minimum detectable level). Because the following formulation was used for the MDL calculation, if the background counts were zero, then the MDL result was zero [this seldom occurred (PDN 240001295)].

$$MDC = \frac{4.65 \times (C_{Background})^{1/2} \times CF}{TREVA} \quad (5-1)$$

where

- MDC* = minimum decision level (μCi/L)
- C_{Background}* = background counts in the region of interest of either ²³⁸Pu or ²³⁹Pu (from 1988 and forward) or peak centroid (from 1975 to 1987)
- T* = count time
- R* = recovery fraction
- E* = average detector efficiency
- V* = sample volume (L), and
- A* = the alpha abundance for the radionuclide in question
- CF* = (Calibration factor) ²⁴²Pu concentration × ²⁴²Pu volume (from at least 1988 to present)

The MDC was the level at which activity was considered present in a sample with a 95% confidence level.

Plutonium MDCs for either ^{238}Pu or ^{239}Pu was calculated with the following formula:

$$A_{Pu} = \frac{\{Net\ ROI_{CTS} (^{238}\text{Pu}\ \text{or}\ ^{239}\text{Pu}) \times CF\}}{\{Net\ ROI_{CTS} (^{242}\text{Pu}) \times sample\ size\}} \quad (5-2)$$

From 1975 through 1982 the determination of a positive count for plutonium was not specifically discussed in the Pinellas Plant documents. In 1983 two bioassay samples equaled the MDLs: One for ^{238}Pu , where the activity equaled the MDL of 9.66×10^{-12} $\mu\text{Ci/ml}$, and one for ^{239}Pu , where the activity equaled the MDL of 1.82×10^{-11} $\mu\text{Ci/ml}$. One other bioassay found a concentration below the MDL for ^{238}Pu and was recounted and found with no positive activity (PDN 240000887).

The two samples from 1983 mentioned above were recounted five times each. In the first case two of the recounts for ^{238}Pu activity were one-half of the MDL (which should have been the decision level), one was equal to the MDL, and two found zero activity. In the second case four of the results found zero activity, and one was equal to the MDL. It is apparent after these recounts that no positive activity was expected (PDN 240000887). (No dose calculations were ever performed at Pinellas for plutonium except under a fictitious worst-case incident scenario.)

In 1988, more specific requirements for the determination of a positive activity sample were developed. Five separate criteria had to be met for the determination of a positive sample. The five criteria were (Weaver 1989):

1. The ratio of ^{238}Pu to ^{239}Pu in a bioassay sample must be about 5:1 ($\pm 20\%$) or ^{238}Pu is detected while ^{239}Pu is not detectable.
2. The result must be greater than the MDL at the 95% confidence level.
3. The precision of the result must be known at 3 sigma (Pinellas used 2 sigma as shown in the bioassay documents) so that the negative range does not include zero.
4. The ^{238}Pu value is above the Derived Investigation Level (DIL) of 3.23×10^{-12} $\mu\text{Ci/ml}$ or the ^{239}Pu value is above the DIL of 2.86×10^{-12} $\mu\text{Ci/ml}$. Calculated DILs are for inhalation pathway, class Y, particle size 1- μm activity median aerodynamic diameter (AMAD), and a yearly sample.
5. The recovery of the tracer must be greater than 50%.

If the first four criteria are met and the fifth is not, a new sample will be processed before further actions are taken.

The bioassay samples were typically counted for about 72 hr, but the actual count times may have varied from 24 to 72 hours. When an unexpected result occurred, Pinellas health physicists requested that a sample be counted at least 6 days to improve statistics.

If ^{239}Pu was detected above the limit in question (MDL, DIL, etc.) in the absence of ^{238}Pu , the source of exposure was investigated but it was not considered a true occupational dose unless otherwise proven.

If the five criteria were met for a bioassay sample, a complete investigation was conducted, which would include additional urine samples, estimation of dose equivalent (the error would be large using only one data point), and a complete survey of Building 400.

No confirmatory evidence of a positive plutonium bioassay or a dose calculation and investigation was found in the Pinellas Plant documentation.

Pinellas health physicists analyzed the 16 plutonium bioassay samples for 1988 using the five criteria and the stipulations and determined that there were no positive samples. A summary of the analysis was given in the 1988 report. Ten of the 16 samples were considered not positive because the $^{238}\text{Pu}/^{239}\text{Pu}$ ratio was less than about 5:1, five samples had values less than the MDLs for both ^{238}Pu and ^{239}Pu , and one sample had both the "3 sigma not precise" and a $^{238}\text{Pu}/^{239}\text{Pu}$ ratio of less than 5:1 (Weaver 1989).

If the same criteria were applied to 1989 and 1990 bioassay data, the following would result:

- The actual 16 bioassay samples for 1989 were analyzed using the five criteria and the stipulations and it was determined that there were no positive samples. Six of the 16 samples were considered not positive because of zero activity, seven samples had values of zero or less than the MDLs for ^{238}Pu and ^{239}Pu , two samples had a $^{238}\text{Pu}/^{239}\text{Pu}$ ratio less than 5:1, and one had a zero MDL reported for one radionuclide and zero activity for the second radionuclide (PDN 240001295).
- The actual 12 bioassay samples for 1990 were analyzed using the five criteria and the stipulations and it was determined that there were no positive samples. Four of twelve samples were considered not positive because of zero activity, six of twelve samples had values of zero or less than the MDLs for both or either ^{238}Pu and ^{239}Pu , one sample had a $^{238}\text{Pu}/^{239}\text{Pu}$ ratio less than 5:1, and one samples had a zero MDL reported for one radionuclide and zero activity for the second radionuclide (PDN 240001295).

Table 5-1 lists the average and maximum MDCs and plutonium concentrations in $\mu\text{Ci}/\text{ml}$ by year. Data for 1984 was not found.

It is reasonable to assume that there were no plutonium intakes at the Pinellas plant because:

1. There were no documented plutonium contaminations.
2. There were no documented cases of plutonium air contamination.
3. There were no plutonium-related incidents documented.
4. All potentially maximally exposed personnel (RTG workers) were routinely monitored for plutonium intakes during the time period of RTG manufacture.

5.2.2 Tritium Minimum Detectable Concentrations

Table 5-2 shows that the tritium MDCs varied over the years at Pinellas Plant. The recording levels and dose calculations also changed over time. The MDCs were based on the environmental water sample geometry or the counting methods of earlier years until about the mid-1970s (GE no date). Individual worker results should be used because the stated recording levels as listed in Table 5-2 varied in the actual records.

Table 5-1. *In vitro* measured concentrations or MDC averages for plutonium urinalysis by year.

| Period | Radionuclide | Average MDC $\mu\text{Ci/ml}$ | Maximum MDC $\mu\text{Ci/ml}$ | Average activity concentration $\mu\text{Ci/ml}$ | Maximum activity concentration $\mu\text{Ci/ml}$ | Reference |
|--------|--------------|-------------------------------|-------------------------------|--|--|---------------|
| 1975 | Pu-238 | | | 2.79E-11 | 7.88E-11 | PDN 240000887 |
| | Pu-239 | | | 1.66E-11 | 3.00E-11 | PDN 240000887 |
| 1976 | Pu-238 | | | 3.85E-11 | 1.79E-10 | PDN 240000887 |
| | Pu-239 | | | 1.84E-11 | 6.86E-11 | PDN 240000887 |
| 1977 | Pu-238 | 6.79E-12 | 1.34E-11 | 3.38E-12 | 3.38E-12 | PDN 240000887 |
| | Pu-239 | 6.03E-12 | 1.09E-11 | 1.29E-11 | 4.93E-11 | PDN 240000887 |
| 1978 | Pu-238 | 2.77E-11 | 8.02E-11 | 0.00E+00 | 0.00E+00 | PDN 240000887 |
| | Pu-239 | 2.77E-11 | 8.02E-11 | 1.51E-11 | 4.01E-11 | PDN 240000887 |
| 1979 | Pu-238 | 3.64E-12 | 8.72E-12 | 0.00E+00 | 00E+00 | PDN 240000887 |
| | Pu-239 | 3.09E-12 | 8.89E-12 | 3.79E-12 | 5.51E-12 | PDN 240000887 |
| 1980 | Pu-238 | 6.23E-11 | 3.17E-10 | 3.84E-12 | 3.85E-12 | PDN 240000887 |
| | Pu-239 | 3.41E-11 | 1.90E-10 | 3.34E-12 | 3.34E-12 | PDN 240000887 |
| 1981 | Pu-238 | 5.65E-13 | 2.31E-12 | 0.00E+00 | 0.00E+00 | PDN 240000887 |
| | Pu-239 | 1.70E-13 | 4.72E-13 | 0.00E+00 | 0.00E+00 | PDN 240000887 |
| 1982 | Pu-238 | 9.86E-12 | 1.98E-11 | 9.88E-12 | 9.88E-12 | PDN 240000887 |
| | Pu-239 | 9.81E-12 | 1.98E-11 | 0.00E+00 | 0.00E+00 | PDN 240000887 |
| 1983 | Pu-238 | 1.74E-11 | 3.83E-11 | 8.13E-12 | 1.82E-11 | PDN 240000887 |
| | Pu-239 | 1.70E-11 | 3.83E-11 | 6.88E-12 | 9.09E-12 | PDN 240000887 |
| 1985 | Pu-238 | 4.76E-12 | 8.31E-12 | 4.76E-12 | 8.31E-12 | PDN 240000887 |
| | Pu-239 | 2.42E-11 | 2.21E-10 | 0.00E+00 | 0.00E+00 | PDN 240000887 |
| 1986 | Pu-238 | 1.27E-11 | 3.12E-11 | 4.13E-12 | 4.13E-12 | PDN 240000887 |
| | Pu-239 | 1.51E-11 | 3.23E-11 | 0.00E+00 | 0.00E+00 | PDN 240000887 |
| 1987 | Pu-238 | 7.95E-12 | 1.88E-11 | 6.70E-12 | 1.71E-11 | PDN 240001295 |
| | Pu-239 | 8.00E-12 | 1.88E-11 | 6.25E-12 | 9.11E-12 | PDN 240001295 |
| 1988 | Pu-238 | 1.25E-11 | 3.71E-11 | 1.06E-11 | 8.85E-11 | PDN 240001295 |
| | Pu-239 | 7.69E-12 | 3.17E-11 | 6.41E-12 | 3.47E-11 | PDN 240001295 |
| 1989 | Pu-238 | 3.87E-11 | 1.14E-10 | 1.35E-11 | 4.03E-11 | PDN 240001295 |
| | Pu-239 | 3.40E-11 | 1.27E-10 | 9.94E-12 | 3.25E-11 | PDN 240001295 |
| 1990 | Pu-238 | 2.37E-11 | 3.12E-11 | 6.88E-12 | 1.69E-11 | PDN 240001295 |
| | Pu-239 | 1.91E-11 | 3.94E-11 | 2.25E-12 | 8.28E-12 | PDN 240001295 |

5.2.3 *In Vitro* Methods for Individual Radionuclides

5.2.3.1 *In Vitro* Bioassay for Tritium

Beginning at least as early as 1966, urine samples for liquid scintillation counting (LSC) were submitted on a daily basis at the end of a worker's shifts 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 sample consisted of 0.2 ml of the composite sample combined with 16.5 ml of fluour (1 L toluene, 0.2 L ethanol, 3.5 g PPO, 0.12 g) (GE 1964). The MDC for the samples should have been the MDC for water samples or 0.09 $\mu\text{Ci/L}$, but a value of 0.67 $\mu\text{Ci/L}$ was used as the recording level and listed in some documentation (Author unknown 1973). A chronic infinite dose of 5 mrem whole-body dose was calculated at this 0.67 $\mu\text{Ci/L}$ excretion level (GE no date).

Table 5-2. *In vitro* measurement frequencies, measurement types and MDCs for tritium urinalysis.^{a,b}

| Period | Frequency ^d | MDC μCi/L | Recording level μCi/L | Dose record level | Reference |
|-------------|------------------------|---------------|---|---|-------------------------------------|
| 1957 | Weekly | 0.30 | <1 μCi/L | <5 mrem 1 μCi/L = 6 mrem | Claims and PDN 240001043 |
| 1958 | Weekly | 0.30 | <1 μCi/L | <5 mrem 1 μCi/L = 10 mrem | Claims #13872 |
| 1959 | Weekly | 0.30 | 0.01 μCi/L | <1 mrem 1 μCi/L = 10 mrem | Claims #14194 |
| 1960 | Weekly | 0.30 | 0.01 μCi/L | <20 mrem or < | Claims #14194 |
| 1961 | Weekly | 0.30 | 0.01 μCi/L | <20 mrem or < | Claims #14194 |
| 1962 | Weekly | 0.13 | 0.67 or < μCi/L | <20 mrem or < | Claims #14194 |
| 1963 | Weekly | 0.13 | 0.67 or < μCi/L | <20 mrem or < | Claims #14194 |
| 1964 | Weekly | 0.13 | 0.67 or < μCi/L | <5 mrem ? | Claims #14194 |
| 1965-1972 | Weekly | 0.13 | 0.67 or < μCi/L | <5 mrem – 1966 | Claims and GE 1971 |
| 1973-2/1974 | Daily to monthly? | 0.13 | 0.67 μCi/L ^a = 5 mrem infinite dose 1,250 mrem/yr | <5 mrem ? | GE 1973 |
| 3/1974-1986 | Daily to monthly? | 0.10 +/- 0.01 | > or = 0.1 μCi/L (0.42 mrem) | 1978 1 μCi/L = 5 mrem | Claims #14194; GE 1975; GE 1984a |
| 1987-1989 | Daily to monthly | 0.01 | 0.10 μCi/L (5 mrem/μCi/L infinite dose – 12-day eff. T1/2 ^c) | 1987 4.21 mrem/μCi/L infinite dose – 10-day eff. T1/2 ^c | PDN 240000969 |
| 1990-1997 | Daily to monthly | 0.006 | 0.10 μCi/L | 4.21 mrem/μCi/L infinite dose – 10-day eff. T1/2 ^c | PDN 030000352 PDN240001222 |

a. Sources: *Radioanalyses Sensitivities* (GE 1973); Holliday (1983); Burkhart and Richardson (1986).

b. Urinalysis method was LSC for tritium oxide (HTO) throughout Pinellas Plant site history.

c. eff. T1/2 = effective half-life.

d. See section 5.2.4.1 for more detailed frequency information.

A procedure dating from the 1960s was first to add charcoal to the solution to decolorize the solution, filter the composite through Whatman #1 filter paper, and pipette 0.2 ml of sample (with the addition of the 16.5 ml of cocktail solution). The composite sample was then placed in the freezer for cooling and counted for 10 minutes on a Beckman multichannel LSC using the first channel set at 15% preset error. If the composite sample exceeded an integrated infinite dose of 200 mrem for 40 samples or a 5-mrem average infinite dose, then the composite samples were broken into four samples. The positive group of samples was then identified and counted individually. Individual samples at this time were counted for 5 minutes (GE no date, pp. 35–37). This compositing of urine samples was likely used for a screening purpose to reduce the amount of urine samples analyzed and to address possible high intakes more quickly. The composite samples should not affect the actual dose calculations that the dose reconstructor performs.

Beginning at least as early as 1986, urine samples were submitted at least 1.5 to 2 hr after suspected exposure. Participants were instructed to leave at least a 25 ml sample for analysis. The bioassay samples were loaded into the LSC, usually overnight, to count for 30 minutes and or to a ±10% error, whichever occurred first. One ml of urine was added to 9 ml of LSC cocktail. The MDL of 0.006 μCi/L was calculated based on the worst-case background and efficiency values (Burkhart 1995a).

Typical efficiency values for the LSC for tritium in 1957 were about 5% to 8%. The efficiency increased to a maximum of about 40% by the 1990s.

5.2.3.2 *In Vitro* Bioassay for Plutonium

The sample was prepared with a number of steps. The first step was a 1-day 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 (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. Pinellas used National Institute of Standards and Technology (NIST)-traceable sources of ^{238}Pu and ^{239}Pu to set up the ROIs. The use of polished stainless-steel planchets was implemented (Christy 1988).

At an unknown point, the Pinellas chemistry counting laboratory used a Series 100 Canberra alpha spectroscopy system. This alpha spectroscopy system likely used a silicon barrier detector system [as shown in a sample spectrum comparison to a NIST ^{238}Pu and ^{239}Pu source spectrums (GE 1989b, pp. 1–3)]. 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 (PDN 240001295) 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.

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

5.2.4 Frequency of Urine Bioassays

5.2.4.1 Frequency for tritium oxide (HTO) Urine Bioassays

The bioassay program at Pinellas since about 1986 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 technical basis document (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)*

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.

5.2.4.2 Frequency for Plutonium Urine Bioassays

Plutonium bioassays were performed on an annual basis for all radioisotopic thermal generator (RTG) workers from 1975 to 1990. Termination bioassay results that would have been submitted early in 1991 were not found.

5.2.5 Urinalysis Minimum Detectable Concentration Summary

There have not been a variety of methods used historically to analyze for tritium or plutonium at Pinellas. LSC was always used for tritium analysis, and alpha spectroscopy after separation chemistry was always used for plutonium analysis. Table 5-3 summarizes these methods and their associated detection capabilities. Radionuclides that were not included in the bioassay program at Pinellas but were used at unknown quantities are included in Table 5-3.

Table 5-3. *In vitro* measurement frequencies, measurement types, and MDCs for urinalysis.^a

| Period | Frequency | Radionuclide | MDC | Record level | Urinalysis method | Restriction | Volume |
|-----------|---|------------------|------------------------------|-----------------------|---|---|--|
| 1957-1997 | Daily to monthly, based on work assignments, 4 hr after work permit, daily for routine workers, weekly or monthly for nonroutine workers. | HTO | 0.30 - 0.006 μ Ci/L | 0.01 - < 1 μ Ci/L | LSC – 2 hr wait for a 25 ml sample. 2 ml maximum combined with 8 to about 15 ml LSC cocktail. | 10 μ Ci/L (1986-1995?) | Spot – 25-ml sample composited, then individual |
| 1975-1990 | Annual, for all RTG workers | Pu-238 Pu-239 | See records See Table 5-1 | NA -See text. | Alpha Spectrometry Procedure changed at least once sometime in 1987, see records. | Never occurred at Pinellas; only if sample considered positive. | 24-hr sample with use of 800 ml maximum or whatever was available. |

a. Sources: *Radioanalyses Sensitivities* (GE 1973); Holliday (1983); Burkhart and Richardson (1986).

5.3 SOURCE TERM AND ISOTOPIC DETERMINATION – SWIPES AND AIR SAMPLING

5.3.1 Contamination Monitoring

5.3.1.1 Tritium Oxide Contamination Monitoring

Contamination monitoring for tritium was performed on a routine basis from the beginning of operations at the Pinellas Plant. 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. It was known early that wet swipes for contamination were more efficient than dry swipes. The water

geometry for counting was used with the LSC for contamination monitoring. This same water geometry was used for HTO until about 1971 for the urine bioassays as well (GE no date).

Contamination monitoring during 1962 indicated levels of about 1.3×10^5 dpm/100 cm² (or up to 1,000 cm²) in the work areas with a maximum of 7.8×10^5 dpm/100 cm² (or up to 1,000 cm²). Personal skin contamination averaged about 5×10^5 dpm/100 cm² (or up to 1,000 cm²) with a maximum of 2.75×10^7 dpm/100 cm² (or up to 1,000 cm²) (Jech 1963a). Contamination monitoring from 1960 indicated levels of up to 0.352 μ Ci in the hood systems with 23 μ Ci in the hoods and up to 0.043 μ Ci on the floors. The maximum internal tritium exposure in 1 month was 34 mrem for 1960 [*Monthly Statistics and Summary 1960* (PDN 240001051)].

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 cm²). 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 MDC was 0.002 μ Ci/area swiped (GE no date). As early as 1966 the uncontrolled area limit for tritium was 220 dpm/100 cm² and the controlled area limit was 440 dpm/100 cm² (GE c. 1966).

In the 1990s 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 (GE 1989b; GE 1995).

5.3.1.2 Plutonium Contamination Monitoring

From the beginning of plutonium source use in the RTG in 1975, the Pinellas Plant routinely monitored the work areas for possible plutonium contamination. 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.

Upon receipt of plutonium sources directly to Building 400, there were leak checks in the receiving area in the containment hood, and the sources were repackaged into 2R storage and transport containers and placed into the storage vault. A 6-dpm/swipe release limit was the control level for the source contamination release limit at the 95% confidence level [an Eberline SAC-4, ESP-1/Ludlum 43-10 or Baird accu-count was used (Burkhart 1989)]. A maximum of 1 of 70 plutonium sources received during the course of one month had a contamination level greater than the detection limit of 6 dpm. All were decontaminated before RTG processing. Plutonium sources greater than 200 dpm contamination levels were sent back to the vendor (Huffman 1979). There was no documentation found indicating that any units had to be sent back.

When received in the glovebox area for processing while in a fume hood, the plutonium sources were checked with a PAC 4S portable alpha monitor, and the local area was monitored with an Eberline alpha-1, -2, -3, or -5 air monitor. This occurred after the user donned a half-face respirator (Type H cartridges), laboratory coat, and gloves. The PAC 4S had a 50 dpm/probe lower limit of detection (Burkhart 1989).

Shipments of $^{238}\text{PuO}_2$ sources were made in the 2H container as a yellow II, Type A, special form material, with a 0.1 mR/hr Transport Index.

5.3.1.3 Nickel Contamination Monitoring

Nickel-63 was used in the production of spark gaps. The radiation levels at the open ends of the gaps measured between 300 and 1,500 cpm. Losses of ^{63}Ni occurred during vacuum firing and ultrasonic cleaning operations. Surveys of the process areas in the 888 Process Area showed no detectable ^{63}Ni (Jech 1963a, p. 7).

Nickel-63 was electroplated on a nickel mesh inside a sealed glass tube (a *krypton*). No internal monitoring was conducted. This is based on the types of operations with these devices, and routine survey data results of the work areas and equipment used. Fictitious worst-case doses were calculated and placed in an "Internal dosimetry technical notes file," but this file has not been found (Weaver 1994a).

5.3.1.4 Carbon-14 Contamination Monitoring

There is no documentation on ^{14}C monitoring even though it likely occurred.

5.3.2 Air Sampling

5.3.2.1 Tritium Air Sampling

Air-monitoring results were indicated in the 1962 (Jech 1963a) report for HTO and T_2 . A portable air monitor (T-290) was capable of an MDC of 1×10^{-4} $\mu\text{Ci/ml}$, and a Kanne ion chamber area monitor was capable of a MDC of below 2×10^{-5} $\mu\text{Ci/ml}$, the U.S. Atomic Energy Commission (AEC) 40-hr level. Typical work area measurements were less than 2×10^{-5} $\mu\text{Ci/ml}$. Incidents approached the 1×10^{-2} $\mu\text{Ci/ml}$ level. Maintenance levels averaged 1×10^{-3} $\mu\text{Ci/ml}$. These levels usually occurred during maintenance operations in gloveboxes or vacuum pumps. The maximum tritium air concentration during normal operations was 6×10^{-4} $\mu\text{Ci/ml}$. (Jech 1963a).

Room monitors were located in all areas where there was a potential for release of tritium. In 1971 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. The alarm was set at about four times the AEC limit of 2×10^{-5} $\mu\text{Ci/ml}$, which is the same as the current derived air concentration agreed to by DOE and the U.S. Nuclear Regulatory Commission. The MDC has been calculated to be as low as 1.3×10^{-5} $\mu\text{Ci/ml}$ (Ward 1971). Areas monitored included Laboratories 108, 109, 157, 158, 182, and 132. Portable samplers using silica gel collection media, or silica gel stations were also set up in some areas. 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. In addition, portable tritium gas (T_2) monitors (the T-290 mentioned above) were used (GE 1987; Ward 1973).

5.3.2.2 Plutonium Air Sampling

Real-time air sampling 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 samplers were set at 10- or 15-cpm alarm points. They were checked with ^{239}Pu check sources on a monthly basis.

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 they were composited quarterly for ^{238}Pu analysis.

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 (Eberline 1970). The Pinellas Plant continued to upgrade the system over the years (Dixon 1988a,b). 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)

5.4 URINALYSIS VOLUME

Beginning in the 1960s HTO urine samples were submitted on a daily basis, at the end of the worked shifts, or at the beginning of the next shift. 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.

For plutonium bioassays the sample size was what was received up to 800 ml. If the sample given was less than 800 ml, then the entire sample was analyzed. If more than 800 ml was submitted, only 800 ml was used. Sample volumes were found to be documented for only several years. Because no dose assessments were made, the concentration values given in the analysis were likely based on the volume used in the analysis. Because volume information data was only found for several years, volume correction is not possible in most cases (Weaver 1989).

5.5 ABSORPTION TYPES FOR RADIONUCLIDES INCLUDED IN THE BIOASSAY PROGRAM

The three radionuclides (T, ^{238}Pu and ^{239}Pu) were handled in quantities or forms sufficient to present a concern for internal exposure according to an extensive review (Burkhart 1995a, p. 1). Three chemical forms of T were present: Water (HTO), metal hydride (primarily ScT_2 , ErT_2 and TiT_2), and

gas (T_2). Plutonium oxide (either ^{238}Pu or ^{239}Pu) was present between 1976 and 1991 only in triply encapsulated heat sources (Burkhart 1995a, p. 1).

ICRP Publication 68 lists plutonium oxides as absorption type S (ICRP 1995, p. 83). A discussion of absorption type for plutonium oxides in ICRP 71 indicates that bioassay data from accidentally exposed workers to $^{238}\text{PuO}_2$ could have been closer to type M (ICRP 1996, p. 329).

5.6 RADIONUCLIDES NOT INCLUDED IN THE BIOASSAY PROGRAM AND METAL TRITIDES

Some Pinellas Plant health physicists investigated whether T metals, primarily titanium alloys, needed to be analyzed separately from HTO.

The metal hydrides (primarily ScT_2 , ErT_2 , and TiT_2) 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 remain intact (Burkhart 1995a p. 2). (Table 5-10 later in this document lists where incidents involving some of these materials occurred.)

Intake prevention occurred by the elimination of the possibility of inhalation by means of ventilation and personnel protective equipment. There was little known of the biological behavior of these materials throughout most of the lifetime of the Pinellas Plant. The inability to detect the deposition of potentially small amounts of the low-energy beta-emitting tritium by whole-body counters or other external means would have made it very difficult to analyze for these soluble forms ICRP Publication 71 indicates that one study in relation to TiT_2 of an *in vitro* study using 1- μm particles (count median diameter) as a serum ultrafiltrate produced an assignment of absorption Type M (ICRP 1996, p 32). An *in vivo* study also indicated a Type M dissolution rate. ICRP 71 assumes that the biokinetics for metal hydride particles follows that of HTO. The Pinellas Plant provided the referenced researchers with the TiT_2 to conduct the research (Weaver 1992a). A recent Health Physics journal article (Zhou and Cheng, 2004) categorizes TiT_2 as ranking between type M and type F. The dose reconstructor should consider TiT_2 as type M based upon ICRP 71 and the recent article in the Health Physics journal. Besides incidents, metal tritide intakes could have occurred from any of the process areas listed in Table 5-8 for HTO processing as presented in section 5.12.

Organic ^{14}C and as CO_2 are considered Class SR-2 (ICRP 1995, p. 22). The chemical form of ^{14}C used at Pinellas is not known. The period of use was likely between 1979 and 1983, as evidenced in the environmental stack monitoring reports of that period (DOE 1979 – 1983).

The form of ^{63}Ni is also not known, though it was plated on the vacuum switch tubes.

5.7 *IN VIVO* MINIMUM DETECTABLE ACTIVITIES, COUNTING METHODS, AND REPORTING PRACTICES

There was no *in vivo* monitoring at Pinellas.

5.8 INTERFERENCE AND UNCERTAINTY

5.8.1 Uncertainty and the Contamination of Samples

“Uncertainties or errors for HTO 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 procedures (GE PDN 240001174, p48) "There was a possibility of cross contamination of urine samples.

In about 1985 it was stated that one positive sample followed by a negative sample would negate the first positive sample if given in the same day 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. 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

Uncertainties or errors for Pu bioassay measurements or the actual bioassay results themselves were usually not stated in the personnel records or database. MDCs were calculated based upon a 95% confidence level and samples were counted to a 2-sigma error as indicated in Pinellas procedures (Weaver 1989) . Note, there were no verified positive bioassay results or known Pu contamination incidents.

5.9 ASSESSMENT OF INTAKE FOR MONITORED AND UNMONITORED WORKERS

5.9.1 Monitored Workers with Measurable Intakes (HTO and Plutonium)

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 HTO and Plutonium potentially exposed workers have likely been monitored. Some conservative assumptions that a dose reconstructor should make are as follows:

- Radionuclide, particle size, and solubility type – These are listed below for bioassay radionuclides. If the work location of the employee is not known, default assumptions for radionuclide, particle size, and solubility type are equal to those in the most closely related facility. If the dose reconstructor decides to calculate plutonium doses, a 5- μ m AMAD particle size should be assumed. Both HTO and T₂ are absorption type V and have no characteristic particle size (ICRP 1995).
- There is no correlation between tritium and plutonium exposures.

5.9.2 Metal Tritide Exposures

A list of job titles that were not likely to receive a MT exposure is listed in Table 5-4 below.

The remainder of this section on metal tritide exposure is reserved until additional exposure data can be researched and developed.

Table 5-4. *Job Titles NOT likely to receive MT Exposures.*^{a,b}

| | | | |
|-------------------------------|-----------------------------|--------------------------------|----------------------------|
| Job Title | Job Title | Job Title | Job Title |
| secretary | key operator | office services coordinator | buyer-equip-plant services |
| resources planning manger | vendor material coordinator | data process computer operator | insurance pension clerk |
| inventory control coordinator | programmer analyst | buyer | financial analyst capital |

| Job Title | Job Title | Job Title | Job Title |
|------------------------------|---|-----------------------------|---------------------------------|
| cashier-accounting clerk | payroll clerk | salary admin clerk | mgr personnel relations |
| supervisor-computer operator | stenographer | secretary –temporary | accounting clerk |
| records retention control | general accountant | area dispatcher | account payable clerk |
| drafter | purchasing expediter | receptionist | general accounts clerk |
| graphic services aid | special mfg program plan and scheduling | typist/clerk | purchasing clerk |
| manager cost accounting | spec graphic design | financial analyst-contracts | supervisor personnel accounting |
| group leader drafting | supervisor-disbursements | spec-telephone services | personnel relations clerk |
| employee benefits clerk | spec-employee store | purchasing clerk | media support coordinator |
| spec general accounting | groundskeeper | personnel account clerk | |

a. Sources: GE no date, PDN 240001303.

5.10 TRITIUM DOSE DISTRIBUTION INFORMATION

The dose calculation was initially estimated by assuming an acute dose integrated to infinity for a standard man, which yielded a calculation of about 5 mrem/ μ Ci/L using a 12-day biological half-life. This result was used until about 1987 (Holliday 1983) when it changed to about 4.21 mrem/ μ Ci/L (see Section 5.10.1 for the calculation).

It appears that most of the time the recording level was the basis for the calculation and recording of the infinite dose calculation. For example, if "<20" was indicated in the dose column listed as "mrem(B)" (even if 4 months equaled <20, four times the recorded dose was still zero), or if "<" or "<0.67" was indicated in the concentration column (μ Ci/L), no dose calculation was performed or added into the personal dose record. With the advent of the bioassay dose calculating computer program in about 1986, all doses were calculated but only results above 0.1 μ Ci/L were counted and cumulative doses above 1 mrem for the year were recorded in personnel annual.

A health physics report for 1962 compiled a summary of urine sample data from 1959-1962 (Jech 1963a, p. 6); Table 5-5 lists this information. The excretion rates are categorized into five 1- μ Ci/L bands. The maximum excretion levels are given along with the maximum personal dose due to tritium uptake.

The maximum yearly tritium deposition of 6.7 μ Ci/L was found in a technician who worked on a glovebox line vacuum ion pump in 1962. The air concentration was 6.5×10^{-4} μ Ci/cm² with the technician wearing a fresh air mask and protective clothing. The dose was calculated to be 70 mrem. The yearly dose of 0.2 rem was incurred by a technician engaged in normal hood work. A concentration of 5.3 μ Ci/L and 15 positive samples of 1 to 3 μ Ci/L occurred. The cause of the concentrations was traced to the handling of aging tank oil with compromised containment (Jech 1963a, p. 6).

Table 5-6 lists data in dose bands for 1986 to 1991. The highest dose is recorded in parentheses. As shown, most of the monitored tritium dose was at values below 100 mrem/year during this period.

Table 5-7 lists maximum and average tritium dose for the years that the information is available. This table will be updated as more information becomes available. The results discussed in Tables 5-5 and 5-7 (up to 1975) utilized a 1.7 RBE for HTO. The results discussed in Table 5-6 utilized a RBE of 1.0.

Table 5-5. Summary of tritium urinalysis from 1959 to 1962.^{a,b}

| Description | 1959 | 1960 | 1961 | 1962 |
|--------------------------------------|--------------|--------------|--------------|------|
| Number of samples | 506 | 657 | 712 | 836 |
| Percent of samples <1 µCi/L | 14.8 | 19.0 | 9.6 | 4.9 |
| Number of samples 1 – 2 µCi/L | 54 | 97 | 61 | 24 |
| Number of samples 2 – 3 µCi/L | 12 | 18 | 5 | 8 |
| Number of samples 3 – 5 µCi/L | 7 | 5 | 3 | 5 |
| Number of samples >5 µCi/L | 2 | 5 | 1 | 4 |
| Maximum annual concentration (µCi/L) | 7 | 7.2 | 7.4 | 6.7 |
| Maximum yearly exposure (rem) | 0.15 | 0.17 | 0.16 | 0.20 |
| Number of persons 0.1 to 0.2 rem | Not recorded | Not recorded | Not recorded | 10 |
| Number of persons 0 to 0.1 rem | Not recorded | Not recorded | Not recorded | 39 |
| Number of persons sampled | Not recorded | Not recorded | Not recorded | 49 |

a. Jech (1963a, p. 6).

b. The dose calculations above included a 1.7 RBE for tritium.

Table 5-6. Summary of tritium urinalysis from 1986 to 1991.^{a,b}

| Calendar year | Number of workers monitored | Number <10 mrem | Number <10–100 mrem | Number <101 – 500 mrem | Number >500 mrem |
|---------------|-----------------------------|-----------------|---------------------|------------------------|------------------|
| 1986 | 194 | 39 | 21 (86) | 0 | 0 |
| 1987 | 139 | 26 | 8 | 1 (105) | 0 |
| 1988 | 129 | 28 | 13 | 1 (130) | 0 |
| 1989 | 201 | 33 | 12 (97) | 0 | 0 |
| 1990 | 177 | 17 | 6 (31) | 0 | 0 |
| 1991 | 202 | 14 | 8 | 1 (101) | 0 |

a. Source: Weaver (1992b).

b. A RBE of 1.0 was used.

Table 5-7. Summary of average and maximum tritium doses derived from tritium urinalysis by year.^b

| Calendar year | Number of workers monitored | Average tritium dose ^a (mrem) | Maximum tritium dose (mrem) | References |
|---------------|-----------------------------|--|-----------------------------|------------------------------|
| 1956 | | | | |
| 1957 | | | | |
| 1958 | | | 367 | GE 1958 |
| 1959 | | | 150 | Jech 1963a |
| 1960 | | | 170 | Jech 1963a |
| 1961 | | | 160 | Jech 1963a |
| 1962 | 49 | | 200 | Jech 1963a |
| 1963-1975 | | | | |
| 1976 | 181 | 20.00 (7.29) | 117 | Holliday 1977 |
| 1977-1978 | | | | |
| 1979 | | | 236 | Mauer 1979 |
| 1980-1982 | | | | |
| 1983 | | | 91 | PDN 240001310 |
| 1984 | | | 236 | PDN 240001310 |
| 1985 | | | 120 | PDN 240001310 &1205 |
| 1986 | 194 | | 86 | PDN 240001205 |
| 1987 | 139 | | 105 | PDN 240001205 |
| 1988 | 129 | 16.85 (4.38) | 130 | PDN 240001205; Majestic 1989 |
| 1989 | 201 | (2.77) | 97 | PDN 240001205 |
| 1990 | 177 | 7.21 (1.04) | 31 | PDN 240001205; Harder 1991 |
| 1991 | 202 | 12.76 (1.93) | 101 | PDN 240001205 |
| 1992 | 164 | (0.91) | 35 | PDN 240001205 |

| Calendar year | Number of workers monitored | Average tritium dose ^a (mrem) | Maximum tritium dose (mrem) | References |
|---------------|-----------------------------|--|-----------------------------|---------------|
| 1993 | 134 | (0.77) | 21 | PDN 240001205 |
| 1994 | 226 | 1.6 (0.80) | 6 | Weaver 1995 |
| 1995 | | | | |
| 1996 | | | | |
| 1997 | | | | |

- a. () Values indicate averages of positive and non-positive bioassays results averages. Non () values indicate only positive bioassay result averages.
- b. A RBE of 1.7 was used from 1954-1975. A RBE of 1.0 was used from 1976 –1994.

5.10.1 Relative Biological Effectiveness for Tritium

An RBE of 1.7 was used to calculate tritium exposures for 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. Exposures through 1975 were corrected to a 1.0 RBE. Calculations for 1976 and subsequent years were made with an RBE of 1.0 (Holliday c. 1976). In the exposure records, most of the tritium dose records were corrected from 1957 to 1975 by dividing by 1.7, as indicated in handwritten calculations in the claimants' DOE dose files. All of the results discussed in Section 5.10 were not corrected for the 1.7 RBE. (Tables 5-5 and 5-7 used an RBE of 1.7, Table 5-6 used an RBE of 1.0). The infinite tritium dose calculations were used in the Pinellas computer program beginning about 1976.

The infinite tritium dose calculations were as follows (Weaver 1994b):

$$D^{\#} \text{ rem}/\mu\text{Ci}/\text{day} = \frac{(2.22 \times 10^6 \text{ dpm}/\mu\text{Ci}) \times (60 \text{ min}/\text{hr}) \times (24 \text{ hr}/\text{day}) \times (5.7 \times 10^{-3} \text{ MeV}/\text{disintegration}) \times (1.6 \times 10^{-6} \text{ erg}/\text{MeV})}{(100 \text{ erg}/\text{g}/\text{rad}) \times (\mathbf{A} \text{ or } \mathbf{B})} \quad (5-3)$$

The assumptions for these calculations were:

- Total body water = 42,000 g* **A**
- Total body soft tissue = 63,000 g* **B**
- Quality factor (rem/rad) = 1.0*
- Tritium effective half-life = 10 days*
- Tritium removal constant $\lambda = 0.069/\text{days}$*

Using Assumption **A**, the value of $D^{\#}$ rem/ μ Ci/day is 6.94×10^{-6} . Using Assumption **B**, that value is 4.63×10^{-6} . Therefore:

$$D(\text{infinity}) = D^{\#}/\lambda \times 42 \text{ L} \times 1,000 \text{ mrem}/\text{rem} = 4.21 \text{ mrem}/\mu\text{Ci}/\text{L} \quad (\mathbf{A}) \quad (5-4a)$$

$$D(\text{infinity}) = D^{\#}/\lambda \times 63 \text{ L} \times 1,000 \text{ mrem}/\text{rem} = 4.21 \text{ mrem}/\mu\text{Ci}/\text{L} \quad (\mathbf{B}) \quad (5-4b)$$

The tritium dose program used from 1987 calculated effective dose equivalent (EDE) using the default 10-day effective half-life, or the program corrected the EDE up or down depending on the individual bioassay history (Weaver 1994b). A 5-mrem/ μ Ci/L dose factor was used in the tritium program before 1987, and 4.21-mrem/ μ Ci/L was used from 1987 to 1997 (GE 1986).

The trigger for the GEDOSE computer program used since 1987 was 0.1 μ Ci/L, which for any single sample (if exceeded) led to calculation of dose from all sample concentrations above zero (Burkhart 1995a).

The default effective half-life was 10 days, but if a different half-life was observed, that half-life was used in the calculations.

5.11 *IN VITRO* URINALYSIS COUNTING FORM CODE INFORMATION

Because the claimant records sent by DOE contain handwritten cards that show all of the recorded internal dose information for a claimant, discussion of the various computer printouts generated by either Pinellas or a vendor (Landauer) is not necessary.

One records issue that could be confusing is the determination of the difference between the recording level for the MDC and the recording level for what was considered a positive dose. As illustrated in Table 5-2, there were a number of variations over the 40-year history of tritium monitoring. From 1957 through 1959 no tritium bioassay result was considered positive below 1 $\mu\text{Ci/L}$, even though the recording level for the bioassay result varied from $<1 \mu\text{Ci/L}$ down to $0.01 \mu\text{Ci/L}$. In addition, nothing under a 5-mrem dose was considered positive. There are, however, secondary entries in the 1957 and 1958 records written next to the doses that were calculated as above 5 mrem which were counted. These values were used in the handwritten summary reports written in about 1985. (Some of these summaries failed to add some internal doses.) As of 1960 the recording level for tritium MDL was considered a less-than value, which was probably $0.67 \mu\text{Ci/L}$ but the dose that was considered positive needed to be greater than 20 mrem.

In comparing calculated doses to the actual recorded doses in the dosimetry records (1990 and 1991), if the dose was calculated to 10.8 mrem, 10 mrem was entered. The result was rounded down.

5.12 TRITIUM AND PLUTONIUM POTENTIAL RADIOACTIVE MATERIAL USE AREAS

There were a number of areas at the Pinellas Plant that had the potential for tritium or plutonium radioactive material area designations (see Tables 5-8 and 5-9). There have been a number of incidents with tritium contamination, as discussed in Section 5.13. There were no records found of plutonium incidents involving contamination.

Table 5-8. Potential tritium radioactive material use areas.

| Location | Name | Activity |
|----------|-----------------------------------|--|
| 107 | Tube Assembly | Vacuum tube manufacturing & 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 & 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 (no date).

Table 5-9. Plutonium radioactive material management areas.

| Location | Name |
|----------|---------------|
| 400 | RTG Area |
| 200 | Test areas |
| 800 | Accelerator |
| 1000 | Waste Storage |

5.13 INCIDENT HISTORY

Table 5-10 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 HTO levels occurred in other areas.

Several examples of incidents are described to provide perspective on the operational health physics safety responses. On April 5, 1963, elevated amounts of HTO gas (up to about a level of 1×10^{-4} $\mu\text{Ci/ml}$) were noted on the room and stack monitors. Upon investigation the sniffer tube from the Kanne ion chamber was used to trace the contamination to a leaking shaft seal on the glovebox line recirculation pump. Corrective actions included increased hood ventilation and closure of the uranium tritide bed in Glovebox #1. All involved personnel, including those in health physics, were sampled for HTO. All results were below the $1\text{-}\mu\text{Ci/L}$ MDC of that time (Jech 1963b).

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 $\mu\text{Ci/m}^3$. Bioassays of personnel in Building 400 revealed a maximum level of 5.3 $\mu\text{Ci/L}$ 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 $\mu\text{Ci/L}$. 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).

Table 5-10. Chronology of unusual events and significant activities in relation to internal dosimetry.

| Date | Description | Curies released of Tritium |
|----------|---|----------------------------|
| 12-57 | Foot monitor installed in Area 108 | |
| 12-10-57 | Operator error in reading manometer in Room 18 | 458 |
| 2-11-58 | Error estimating amount of tritium remaining in charging system in Room 18 | 1,253 |
| 7-8-58 | Glass system breakage in Room 22 | 280 |
| 3-7-58 | Glass system breakage in Room 18 | 567 |
| 8-16-58 | Operator error with tritium loader valve position – Room 21 | 780 |
| 8-18-58 | Glass manifold breakage – Room 21 | 1,180 |
| 2-10-59 | Operator error in valve positioning – Room 8 | 286 |
| 2-20-59 | Hand contamination – operator not wearing gloves | |
| 2-21-59 | Area contamination – operator broke glass system | |
| 3-12-59 | Operator contaminated during system cleaning by another worker (Room 14) | |
| 6-4-59 | Personnel error working on SECS test system – Room 21 | 753 |
| 6-5-59 | Area contamination-diffusion pump exploded in hood 14 | |
| 6-18-59 | Excess air released from tritium loading system | 423 |
| 1-1960 | Operator error – stopcock left open on tritium loading system | 40 |
| 2-5-60 | Glass manifold broke from strain | 72 |
| 2-11-60 | Operator error – stopcock left open on tritium loading system | 308 |
| 3-25-60 | Operator error caused exposure to 3 employees | |
| 5-14-60 | Broken flask caused area contamination – Room 10 | |
| 6-21-60 | Ion gage exploded – room 16 | |
| 7-8-60 | Sample bulb dropped – room 23 | 6.8 |
| 7-13-60 | Manifold shattered, exposing worker – room 23 | |
| 8-12-60 | Contamination spread TiH ₂ in area 108 from broken flask | |
| 9-10-60 | 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) | |
| 4-61 | 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 | |
| 9-1963 | Employees found falsely identifying urine samples | |
| 4-5-63 | Uncontrolled spread of tritium air contamination | |
| 3-30-65 | Broken flask – room 9 | |
| 3-20-65 | Flask explosion – room 12 | |
| 5-1966 | SECS cold water removal problems | 252 |
| 1-27-67 | Glove box vacuum pump oil degassed | 32 |
| 10-12-67 | Personnel contamination – O-ring mishandled – room 18 | |
| 6-18-68 | Acid cleaning explosion – Area 181 | |
| 2-69 | Leaking flange at absorption pump in Area 108 | 8 |
| 2-69 | Area contamination when pump exhaust lines were cut during hood removal – room 2 | 8 |
| 11-5-69 | 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. | |
| 1-70 | Area contamination/personnel exposure from flaking tube part in gas lab | |
| 2-70 | Area contamination from pressurized sorb pump (air expansion) – room 2 | |
| 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 |
| 3-12-71 | 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 |
| 12-1-71 | High internal exposure – room 18 hood work | 1 |
| 4-1972 | Area contaminated from liquid discharge in area 182D | 1.5 |
| 8-3-72 | Leaking absorption pump | 12 |
| 5-73 | Area contaminated with ErH ₂ film in Areas 156, 157 and 158 | |

| Date | Description | Curies released of Tritium |
|---------|---|----------------------------|
| -11-73 | Fire in boom box- building 200 | 0 |
| 1-31-75 | Improper valve closure on uranium bed | 150 |
| 2-10-75 | Absorption pump leak – area 182D | 42 |
| 1-30-76 | Contaminated 6 inch valve | 0 |
| 4-13-76 | Oven fan blade broke tubes | 0 |
| 2-1977 | Packaging fixtures in area 182D glove box | 28 |
| 9-11-79 | Work in Room 18 hood – internal dose | 5.7 |
| 4-80 | Area contaminated from film flaking – Area 158B | 0 |
| 8-80 | Contaminated electron microscope | 0 |
| 4-81 | 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. | |
| 2-25-82 | Tritium recovery system (TRS) valve left in wrong position after maintenance | 8.6 |
| 4-20-82 | Operator left TRS valve in wrong position after maintenance – area 108 | 48 |
| 5-24-82 | TRS valve left in wrong position after maintenance – area 108, exhaust unit 513 | 9.5 |
| 9-1-82 | Sample bulb leak | 3 |
| 1-5-83 | Tritium storage bed oxidation problems | 130 |
| 1-19-83 | Absorption pump leak – area 208 | 9 |
| 4-5-83 | Bed heater control failure – Area 108 | 0 |
| 5-84 | Tritium air monitor system in Area 182 was rearranged | |
| 7-25-84 | Absorption pump sieve dumped into drum in area 108 | 67 |
| 12-9-85 | Sorb pump overheat – area contaminated | 0 |
| 6-24-86 | Mass spectrometer oil change – workers exposed to tritium gas | 1.5 |
| 11-4-87 | Test of oxygen regeneration unit | 12 |
| 2-11-88 | Leaking sample bulb in area 108 | 8 |
| 5-5-88 | Purge left on over third shift in Area 108 – SECS overpressure | 2.7 |
| 9-88 | Lab area release over 2 week period | 16.2 |
| 1-6-89 | Water in SECS line vented in area 182D | 1 |
| 9-7-89 | Loss of control of radioactive material | 0 |

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 HTO, they were promptly asked to submit a urine sample. Work orders quite often required bioassays after completion of the work (PDN 240001443; PDN 240001441). In addition, outside contractors were given HTO bioassays from the start of Pinellas operations in 1957.

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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

Pertaining to intakes received “acutely”, i.e., within a short period of time.

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.

derived investigation level (DIL)

A value based on the regulatory control level and the minimum detectable concentration of the assay method.

dose

A general term for absorbed dose, dose equivalent, effective dose equivalent, committed dose equivalent, committed effective dose equivalent, or total effective dose equivalent.

dose equivalent (H)

The product of absorbed dose (D) in rad (or gray) in tissue, a quality factor (Q), and other modifying factors (N). Dose equivalent is expressed in units of rem (or sievert) (1 rem = 0.01 sievert).

exposure

The general condition of being subjected to ionizing radiation, such as by exposure to ionizing radiation from external sources or to ionizing radiation sources inside the body. In this document, exposure does not refer to the radiological physics concept of charge liberated per unit mass of air.

insoluble material

A term loosely used to describe the relative degree of solubility of a material in body fluids. Recognizing that no material is absolutely insoluble, the terms low solubility or poorly soluble are preferable.

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.

lung solubility type (F, M or S)

A classification scheme for inhaled material according to its rate of clearance from the pulmonary region of the lung.

minimum detectable amount

The smallest amount (activity or mass) of an analyte in a sample that will be detected with a probability, \exists , of non-detection (Type II error) while accepting a probability, \forall , 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 (personnel)

The measurement of radioactivity in the whole body, in a region of the body, in material eliminated from the body or in the air for reasons related to the estimation of intake of radioactive material. The term *monitoring* includes interpretation of the measurements.

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.

radiation

Ionizing radiation: alpha particles, beta particles, gamma rays, X-rays, neutrons, high-speed electrons, high-speed protons, and other particles capable of producing ions. Radiation, as used in this part, does not include nonionizing radiation, such as radio- or microwaves, or visible, infrared, or ultraviolet light.

recording level

A value below which data or results were considered to be too low to record and thus may not have been maintained.

rem

A special unit for dose equivalent. One rem is equal to 0.01 sievert.

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.

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.