ORAU Team **NIOSH Dose Reconstruction Project** Technical Basis Document for the Portsmouth Gaseous Diffusion Plant – Occupational Environmental Dose	Document Number: ORAUT-TKBS-0015-4 Effective Date: 03/17/2004 Revision No.: 00 Controlled Copy No.: _____ Page 1 of 26
Subject Expert: Donald Palmrose Document Owner Approval: Signature on File Date:03/16/2004 Mark Notich, TBD Team Leader Approval: Signature on File Date: 03/16/2004 Judson L. Kenoyer, Task 3 Manager Concurrence: Signature on File Date: 03/17/2004 Richard E. Toohey, Project Director Approval: Signature on File Date: 03/17/2004 James W. Neton, OCAS Health Science Administrator	Supersedes: None

TABLE OF CONTENTS

Section — Page

Record of Issue/Revisions 3

Acronyms and Abbreviations 4

4.1 Introduction 5

4.2 Internal Dose From Onsite Atmospheric Radionuclide Concentration 5

4.2.1 Ambient Air Sample Collection Network 5

4.2.2 Methodology 6

4.2.3 Estimation of Ambient Airborne Radionuclide Concentrations Prior to 1964 7

4.2.4 Data for Ambient Airborne Radionuclide Concentrations and Annual Airborne Releases 7

4.2.4.1 Airborne Release Estimation for Neptunium, Plutonium, Uranium Daughters, and ^{99}Tc 8

4.2.5 Estimation of Annual Intake from Airborne Radionuclides 9

4.2.6 Contribution of Airborne Radionuclides to External Dose 12

4.3 External Dose 12

4.3.1 Ambient Radiation 16

4.3.1.1 Background Radiation 16

4.3.1.2 Ambient Radiological Conditions in Cylinder Storage Yards 19

4.3.1.3 Ambient Radiological Conditions Inside Buildings 19

4.4 Uncertainty 20

References 21

Glossary 25

LIST OF TABLES

Table		Page
4.2.1-1	Ambient Air Sampling Station Locations	6
4.2.5-1	Maximum Annual Intakes and Geometric Standard Deviations (Bq per year)	10
4.3-1	Instruments Used for Ambient Environmental Monitoring	13
4.3-2	Environmental Measurement Locations	13
4.3-3	Site Interior Ambient Radiological Conditions	14
4.3.1-1	Perimeter Road Ambient Radiological Conditions	16

LIST OF FIGURES

Figure		Page
4.3-1	Approximate TLD Locations	15
4.3.1.1-1	Cumulative Quantity of Material by Year	18
4.3.1.1-2	Annual MTU vs. Work Year Doses (2000 hrs/yr)	18

RECORD OF ISSUE/REVISIONS

ISSUE AUTHORIZATION DATE	EFFECTIVE DATE	REV. NO.	DESCRIPTION
Draft	10/30/2003	00-A	New Technical Basis Document for the Portsmouth Gaseous Diffusion Plant - Occupational Environmental Dose. Initiated by Mark D. Notich.
Draft	12/31/2003	00-B	Incorporates NIOSH review comments. Initiated by Mark D. Notich.
Draft	02/19/2004	00-C	Incorporates additional NIOSH review comments. Initiated by Mark D. Notich.
03/17/2004	03/17/2004	00	First approved issue. Initiated by Mark D. Notich.

ACRONYMS AND ABBREVIATIONS

AEC	U.S. Atomic Energy Commission
Bq	Becquerel
CAF_2:Dy	calcium fluoride:dysprosium
Ci	curie
cm	centimeter
DAC	Direct Air Contact
DCG	Derived Concentration Guide
DOE	U.S. Department of Energy
DU	depleted uranium (uranium with less than 0.7% ^{235}U)
EEOICPA	*Energy Employees Occupational Illness Compensation Program Act*
EPA	U.S. Environmental Protection Agency
ERP	Extended Range Product
GDP	Gaseous Diffusion Plant
GSD	Geometric Standard Deviation
HEU	highly enriched uranium (over 20% ^{235}U)
HF	hydrogen fluoride
LiF	lithium fluoride
MTU	metric tons of uranium
NIOSH	National Institute for Occupational Safety and Health
NRC	U.S. Nuclear Regulatory Commission
ORAU	Oak Ridge Associated Universities
PORTS	Portsmouth Gaseous Diffusion Plant
REMS	Radiation Exposure Monitoring System
RU	recycled uranium
TBD	technical basis document
TLD	thermoluminescent dosimetry
UF_6	uranium hexafluoride
USEC	United States Enrichment Corporation

4.1 INTRODUCTION

This Portsmouth Gaseous Diffusion Plant (PORTS) technical basis document (TBD) describes the potential for internal dose from the breathing of airborne concentrations of radionuclides released on the PORTS site (Section 4.2), the potential for external dose from sources of radiation outside and inside the process buildings (Section 4.3), and possible sources of uncertainty associated with both these sources of environmental dose (Section 4.4).

The receptors of concern whom this TBD addresses are unmonitored workers; that is, employees at PORTS who did not wear personnel dosimetry, and who were not monitored for internal dose. At PORTS, there were several periods when all workers, even those with administrative duties, wore film badges or thermoluminescent dosimeters (TLDs) at all times. However, not all film badges were analyzed. For these periods, representative samples for various groups of workers were analyzed (ATL 2003a). To provide the basis for estimating the environmental dose for years when monitoring did not occur or was not sufficient to apply to coworkers, this TBD provides actual and estimated annual intakes and ambient environmental doses for every year from 1955 to 2001.

4.2 INTERNAL DOSE FROM ONSITE ATMOSPHERIC RADIONUCLIDE CONCENTRATION

4.2.1 Ambient Air Sample Collection Network

In 1964, PORTS established a network of permanent on- and offsite stations to collect continuous ambient air samples (DOE 1994). Sampling data beginning in 1964 are available for five offsite locations (one several miles from PORTS to provide background measurements) and for four onsite locations beginning in 1974 (Goslow 1986; GAC 1975).

The principal purpose of the ambient air monitoring network is to assess whether air emissions from PORTS affected air quality in the surrounding area. This would demonstrate compliance with either U.S. Department of Energy (DOE) Derived Concentration Guidelines (DCGs), and U.S. Environmental Protection Agency (EPA), State of Ohio Environmental Protection Agency, and (as of 1997) Nuclear Regulatory Commission (NRC) regulations for airborne releases to the general public around PORTS. Thus, most of the monitoring stations are off the site, along the site boundary, or next to the Perimeter Road. There have been a limited number of onsite monitoring stations. Therefore, this study considers only those ambient air monitoring locations at or within the PORTS site boundary for worker internal dose. Table 4.2.1-1 lists these air monitors and their locations.

As would be expected, several changes to the sampling program were associated principally with sampling technique and equipment; there were at least two major changes of monitoring locations. From 1964 to 1993, low-volume samplers were the principal type of sampler used, employing several filter types and flow rates. In general, filters were changed after sampling approximately 200 to 210 cubic meters of air. Depending on the set flow rate (1.2 or 0.3 cubic meters per hour), filter exchanges were performed weekly or monthly. Major upgrades of the monitoring network occurred in 1987 (number of locations and sampling technique) and from 1993 to 1995 [addition of high-volume samplers and segregation of monitoring responsibilities between DOE and United States Enrichment Corporation (USEC)].

The general practice for air samples was to perform gross alpha and beta-gamma counts. Gross alpha activity was associated with the release of uranium, and beta-gamma activity was associated with the release of ^{99}Tc and daughter products of uranium (Goslow 1986). If the gross counts exceeded established limits of 100 dpm alpha or 200 dpm beta-gamma, the filters would be analyzed

for specific radionuclides (DOE 1994). Only in the last several years have filters been analyzed for specific radionuclides. However, no air samples exceeded established limits while they were in effect.

There have been releases of radionuclides to the atmosphere since the beginning of operations, including accidental releases. PORTS release data have been estimated or recorded since 1954 for uranium, since 1967 for uranium daughter products, and since 1976 for ^{99}Tc (Goslow 1986). Dose reconstructors must consider several issues, such as the accuracy of the amounts released and the fact that there was no ambient air monitoring network in operation for the first 10 years of operations and not all radionuclides of concern were originally measured in the air effluent release streams. Therefore, this TBD discusses a claimant-favorable methodology for estimating potential radionuclide air concentrations, not only for the period from when operations began to 1964 but also to account for potential uncertainty in the release data. The assumptions developed to address the missing data will be overly conservative to help ensure that the methodology is claimant-favorable.

Table 4.2.1-1. Ambient Air Sampling Station Locations.

Designation	Location	Direction from center of site
Within PORTS boundary (fence line)		
A3	South Access Rd., south of DOE property line	SSW
A12	McCorkle Rd., north of East Access Rd.	ENE
A24	North Access Rd., north of Shyville Rd.	N
A29	U.S. Rt. 23 on near leg of old GAT sign	W
Onsite air sampling locations (near Perimeter Road)		
A10	X-230-J3 west (ditch) environmental monitoring station	W
A35	X-230-J2 south (holding pond) environmental monitoring station	S
A36	X-611 water treatment plant, southeast corner of facility parking lot	ENE
A38	X-230-J environmental storage building	ESE
A39	X-760 chemical engineering facility	Near mid-point
A40	X-100 administration building penthouse	Near mid-point

Source: Table 2.1.8 on page 135 of DOE (1989).

4.2.2 Methodology

Estimating airborne concentrations at locations around the PORTS site using traditional transport modeling approaches is limited by the following factors:

- The numerous release points, which include stacks, vents, and other emission sources
- The characteristics of the release points
- The limited number of air sampling locations
- The relatively short distances between release points and onsite receptor locations
- The density and configurations of buildings at PORTS

Therefore, it is unnecessary to develop a complex dispersal model and perform transport calculations for the PORTS site because such modeling is not likely to yield accurate results nor would it be claimant-favorable.

The methodology for determining internal dose from onsite atmospheric radionuclide concentrations applies directly to the available air sampling data. To be claimant-favorable for each year for which data are available, the maximum site measurement for gross alpha (uranium) and gross beta-gamma (uranium daughter products and ^{99}Tc) will be applied for the entire site. Air sampling measurement error and uncertainty will be accounted for to provide the 95th percentile (+2σ), so the following relationship can determine a geometric standard deviation (GSD) for each radionuclide:

$$GSD = \left(\frac{95th\ percentile}{Maximum\ Annual} \right)^{\left(\frac{1}{1.65585} \right)} \quad \textbf{(4-1)}$$

The annual intake associated with each radionuclide of concern is based on the fraction given in the annual release as appropriately contributing to either gross alpha or gross beta-gamma sampling results. Since the gross alpha is only from uranium radionuclides, the highest air monitoring concentration is directly applied to the fraction of a uranium radionuclide to the total uranium release then multiplied by the assumed individual ventilation rate to yield the annual intake for each uranium radionuclide. The gross beta/gamma is segregated in a similar manner between ^{99}Tc and the uranium daughters where the uranium daughters are further fractioned by 6 percent of the uranium daughter fraction for ^{241}Th and the remainder equally split between ^{234}Th and ^{234m}Pa.

4.2.3 Estimation of Ambient Airborne Radionuclide Concentrations Prior to 1964

Because there are no ambient airborne monitoring data prior to 1964, an alternative empirical approach was developed for estimating airborne radionuclide concentrations prior to 1964. Based on other dose reconstruction developments (ATL 2003b), an empirical relationship of maximum annual air concentration for the entire site with respect to uranium release estimates provides a direction relationship of air concentration to material release. The empirical relationship is described by Equation 4-2. The annual average uranium air concentration measured at the air monitoring station with the maximum sampling measurement is divided by the release estimates for a given year to estimate an annual, site-specific, Chi/Q value.

$$Chi/Q\ (m^{-3}) = \frac{Maximum\ Radionuclide\ Air\ Concentration\ (Ci\ m^{-3})}{Annual\ Radionuclide\ Activity\ Release\ (Ci)} \quad \textbf{(4-2)}$$

The empirical Chi/Q value can be determined for each year for which air sampling data are available. However, due to the significant decrease in annual releases beginning in 1994 (from placing the enrichment cascade in cold shutdown), the empirical Chi/Q values will be determined only for 1964 to 1993 (i.e., the beginning of air monitoring data to the last operational year). To be claimant-favorable, the maximum of all empirical Chi/Q values will be used to estimate radionuclide air concentrations for the entire PORTS site for years without ambient airborne monitoring data based on annual radionuclide release estimates. This approach can estimate annual intake loadings by using the estimated air concentrations resulting from the empirical Chi/Q values in the same process as using the actual maximum annual airborne radionuclide concentration.

4.2.4 Data for Ambient Airborne Radionuclide Concentrations and Annual Airborne Releases

This study relied on several data sources for estimating ambient airborne radionuclide concentrations and for the annual airborne releases by radionuclide. The two principal sources were the annual environmental reports from 1972 and 2001 (the Reference List at the end of this TBD includes all DOE site environmental reports from 1973 through 2002) and a special study of historic radiological releases (Goslow 1986). Measurements from the air monitoring locations listed in Table 4.2.1-1 were collected from these documents and applied to the methodology described above. When data were not provided or additional information was required, these core documents were supplemented by air monitoring information provided by USEC (ATL 2003c) or by estimating the release quantity for specific situations using historical trends or physical processes (such as secular equilibrium).

The data provided by Goslow (1986) include accidental releases in addition to routine emissions. The largest of such releases occurred in March 1978, when a 14-ton feed cylinder fell from its carrier and cracked open, releasing an estimated 4,820 kg of uranium to the atmosphere and an additional 680 kg of uranium through the plant sewers, for a total activity of less than 3 Ci. Other major accidental releases included two valve failures on a tails cylinder in July 1969 and October 1978, and a process malfunction in the Side Purge Cascade in December 1983. In addition, a string of accidental releases of mostly depleted uranium during the first 5 years of plant operation accounted for essentially all the uranium lost to the atmosphere from 1955 through 1958 and 20% of the losses in 1959. Thus, the variations seen in the uranium radionuclide fractions and ultimately in the calculated uranium radionuclide intake values are due to the quantity of depleted or enriched material released during a given year.

The uranium daughters identified as potentially of environmental concern were Thorium-231 (^{231}Th), Thorium-234 (^{234}Th), and Protactinium-234m (^{234m}Pa) (Goslow 1986). PORTS estimated the fraction of ^{231}Th to be 2.5 percent to 6 percent of all uranium daughter radiation measurements; the remaining uranium daughter radiation is evenly split between ^{234}Th and ^{234m}Pa. For the purposes of estimating the intake of specific uranium daughter radionuclides, 6 percent should be from ^{231}Th and 47 percent for ^{234}Th and ^{234m}Pa. Also, since the enrichment cascade was shutdown in 1993, if the quantity of uranium daughters released were not specifically given in the DOE annual environmental reports after 1993, they will be assumed to be in secular equilibrium (i.e., at the same activity as the parent radionuclide).

It is important to note that the release data has values and relationships that one might believe to be atypical. For example, there are years where the quantity of uranium daughters released exceeds the quantity of total uranium released (i.e., 1979, 1981, and 1984). The expected situation is for the uranium and its daughters would exist together and, in accordance with their respective decay half-lives, in secular equilibrium where the activity of uranium daughter would be the same as the uranium parent. The only time that the uranium daughters can be considered in secular equilibrium with the parent uranium is after 1993 when the cascade was shutdown. Another example is isotopic quantity of uranium released. The first two years of releases (1955 and 1956) were zero for ^{234}U while the same years had as the total uranium released as approximately 0.55 and 0.24 Curies. Such atypical data is probably due to the nature of the physics of gaseous diffusion, initial enrichment of feed at Paducah prior to enrichment at PORTS, and how and where in the enrichment cascade the releases were performed.

4.2.4.1 Airborne Release Estimation for Neptunium, Plutonium, Uranium Daughters, and ^{99}Tc

A review of the airborne environmental release data for the PORTS site shows a lack of information during the first twenty years of operation for radionuclides that later became a source of concern with subsequent inclusion in the environmental monitoring program. The principal radionuclides added to the environmental monitoring program were the uranium daughters in 1967 and ^{99}Tc in 1976. However, there could be a concern for releases of transuranics from the processing of recycled uranium and for the releases of uranium daughter prior to 1967 and for ^{99}Tc prior to 1976. This section investigates the potential for these unknown releases and how to include their contribution for the environmental occupational dose.

Based on the information given in the Recycled Uranium Mass Balance Project (BJC 2000), the principal transuranics that were fed into the cascade were Np and Pu radionuclides. Both were determined to have annual inventory levels of less than 100 grams over the entire history of PORTS (see Table 5.1-1 and Figure 5.1-1 of BJC 2000). In addition, the Np and Pu that entered the cascade soon plates out on cascade components (BJC 2000, pg 67). The Np and Pu radonuclides would be

removed with the uranium deposits when the process equipment was removed and decontaminated. The uranium deposits, containing the Np and Pu radionuclides, went into the uranium recovery process so a small fraction of the transuranics stayed in the recovered uranium as impurities or were removed into the liquid or solid wastes stream. Therefore, due to the low levels of Np and Pu radionuclides that entered the cascade, their plating out, and the material staying in liquid and solid product or waste streams, airborne releases of Np and Pu are considered to be in trace amounts only and not contributors to the environmental occupational doses.

To estimate the potential releases of uranium daughters prior to the beginning of release measurements in 1967, a key item to note is the tremendous difference in the half-lives of the parent radionuclides (over millions of years) and of the daughters (minutes to days). Such differences in half-lives results in the daughters being in what is defined as secular equilibrium with the parent radionuclide (i.e., daughter's activity is at the same activity as the parent radionuclide). Therefore, by assuming secular equilibrium, ^{231}Th would have the same activity as its parent, ^{235}U and ^{234m}Pa and ^{234}Th would each have the same activity as its parent, ^{238}U. For most years, this assumption would be claimant favorable since the average uranium daughter released activity was approximately 0.3 of the uranium released activity where this value would be closer to 1.0 if both were in secular equilibrium.

With the discovery of ^{99}Tc in the east drainage ditch (outfall 002) during the first quarter of FY1975, this radionuclide was added to those already being monitored (uranium and uranium daughters). What is known of the physical migration process for ^{99}Tc is that it plates out on surfaces within the cascades and slowly migrates through the enriched cascade, and would result in relatively higher counts for the gross beta monitoring of the release vents once reaching the purge cascade (ATL 2003g). Thus, with the pre-enrichment of the recycled uranium at Paducah first removing the ^{99}Tc from the PORTS feed material and the slow migration time to reach any release point such as the purge cascade, release of ^{99}Tc is considered negligible prior to 1975.

4.2.5 Estimation of Annual Intake from Airborne Radionuclides

Using the maximum airborne radionuclide concentration in a year (either from a direct reading or a calculated value), this study used an assumed individual ventilation rate of 2,400 cubic meters per year to derive claimant-favorable annual intakes for the radionuclides of concern. GSDs associated with these intakes were estimated using Equation 4-1. Table 4.2.5-1 lists annual intakes and GSDs for each radionuclide of concern for each year.

Table 4.2.5-1. Maximum Annual Intakes and Geometric Standard Deviations (Bq per year).

Year	99-Tc	GSD	Uranium daughters	GSD	U-234 (e)	U-235	U-236	U-238	Total U	GSD
1953	—	—	—	—	—	—	—	—	—	—
1954	—	—	—	—	—	—	—	—	—	—
1955[(a)]	0.000E+00	0.000E+00	7.322E+03	1.419E+00	6.011E+01	2.706E+00	1.422E-01	1.174E+02	1.803E+02	1.450E+00
1956[(a)]	0.000E+00	0.000E+00	3.159E+03	1.419E+00	2.598E+01	8.139E-01	6.137E-02	5.108E+01	7.793E+01	1.450E+00
1957[(a)]	0.000E+00	0.000E+00	2.945E+02	1.419E+00	1.100E+00	1.320E-01	5.721E-03	3.563E+00	4.801E+00	1.450E+00
1958[(a)]	0.000E+00	0.000E+00	2.436E+03	1.419E+00	3.497E+01	1.540E+00	4.732E-02	3.629E+00	4.019E+01	1.450E+00
1959[(a)]	0.000E+00	0.000E+00	6.051E+03	1.419E+00	4.289E+01	2.794E+00	1.175E-01	5.349E+01	9.930E+01	1.450E+00
1960[(a)]	0.000E+00	0.000E+00	2.316E+03	1.419E+00	1.540E+01	1.078E+00	4.498E-02	2.171E+01	3.823E+01	1.450E+00
1961[(a)]	0.000E+00	0.000E+00	4.645E+03	1.419E+00	3.277E+01	2.222E+00	9.023E-02	4.111E+01	7.620E+01	1.450E+00
1962[(a)]	0.000E+00	0.000E+00	1.513E+03	1.419E+00	1.628E+01	9.018E-01	2.938E-02	1.214E+01	2.935E+01	1.450E+00
1963[(a)]	0.000E+00	0.000E+00	2.142E+02	1.419E+00	3.079E+00	2.860E-01	4.160E-03	2.200E-02	3.392E+00	1.450E+00
1964	0.000E+00	0.000E+00	1.865E+02	1.466E+00	1.579E+01	1.283E+00	2.099E-02	9.867E-02	1.776E+01	1.434E+00
1965	0.000E+00	0.000E+00	6.216E+01	1.446E+00	5.751E+01	3.890E+00	8.398E-02	8.288E+00	7.104E+01	1.450E+00
1966	0.000E+00	0.000E+00	8.880E+00	1.520E+00	7.804E+00	5.651E-01	1.050E-02	2.153E-01	8.880E+00	1.520E+00
1967	0.000E+00	0.000E+00	2.664E+01	1.445E+00	7.992E+00	3.552E-01	1.050E-02	4.884E-01	8.880E+00	1.520E+00
1968	0.000E+00	0.000E+00	3.552E+01	1.434E+00	7.400E+00	2.960E-01	1.050E-02	1.184E+00	8.880E+00	1.520E+00
1969	0.000E+00	0.000E+00	3.552E+01	1.458E+00	1.830E+00	2.410E-01	1.050E-02	6.809E+00	8.880E+00	1.520E+00
1970	0.000E+00	0.000E+00	4.618E+01	1.101E+00	3.102E+00	1.432E-01	4.514E-03	5.966E-01	3.818E+00	1.082E+00
1971	0.000E+00	0.000E+00	0.000E+00	0.000E+00	2.992E+00	1.834E-01	5.249E-03	1.207E+00	4.440E+00	1.134E+00
1972	0.000E+00	0.000E+00	0.000E+00	0.000E+00	8.626E-01	6.470E-02	1.785E-03	5.823E-01	1.510E+00	1.097E+00
1973	0.000E+00	0.000E+00	7.193E+00	1.268E+00	2.037E+00	9.960E-02	2.729E-03	9.960E-02	2.309E+00	1.138E+00
1974	0.000E+00	0.000E+00	5.044E+01	1.406E+00	1.779E+01	9.884E-01	2.687E-02	4.448E+00	2.273E+01	1.359E+00
1975	0.000E+00	0.000E+00	0.000E+00	0.000E+00	1.009E+01	3.759E-01	1.333E-02	7.170E-01	1.128E+01	1.361E+00
1976	3.609E+01	1.419E+00	1.203E+00	1.419E+00	8.052E+00	3.779E-01	1.039E-02	3.862E-01	8.791E+00	1.351E+00
1977	2.559E+01	1.301E+00	5.214E-01	1.301E+00	2.001E+00	7.773E-02	2.729E-03	2.293E-01	2.309E+00	1.367E+00
1978	2.429E+01	1.308E+00	2.527E+00	1.308E+00	4.778E-01	3.341E-02	1.480E-03	7.396E-01	1.252E+00	1.325E+00
1979	5.479E+00	1.307E+00	4.022E+00	1.307E+00	7.536E-01	2.270E-02	9.553E-04	2.724E-02	8.081E-01	1.344E+00
1980	6.864E+00	1.309E+00	2.638E+00	1.309E+00	7.758E-01	2.442E-02	9.553E-04	5.746E-03	8.081E-01	1.326E+00
1981	6.965E+00	1.252E+00	7.687E+00	1.252E+00	3.472E+00	1.078E-01	4.293E-03	6.386E-02	3.632E+00	1.369E+00
1982	3.181E+01	1.383E+00	2.470E-01	1.383E+00	1.079E+01	5.449E-01	1.365E-02	2.008E-01	1.154E+01	1.386E+00
1983	6.020E+00	1.319E+00	2.672E-01	1.319E+00	6.073E+00	2.943E-01	7.642E-03	9.036E-02	6.465E+00	1.374E+00
1984	5.021E+00	1.221E+00	9.726E-01	1.221E+00	1.666E+00	4.760E-02	2.110E-03	1.190E-01	1.785E+00	1.274E+00
1985	5.445E+00	1.216E+00	6.818E-01	1.216E+00	8.426E-01	4.837E-02	1.291E-03	2.177E-01	1.092E+00	1.180E+00
1986	6.895E+00	1.145E+00	1.594E+00	1.145E+00	1.154E+00	6.145E-02	1.719E-03	7.044E-01	2.098E+00	1.240E+00
1987	1.333E+01	1.300E+00	1.716E-01	1.300E+00	2.575E+00	8.454E-02	1.214E-04	2.226E-02	2.664E+00	1.335E+00
1988	2.113E+01	1.202E+00	2.752E-01	1.202E+00	5.328E+00	1.998E-01	1.998E-03	1.079E-01	5.594E+00	1.299E+00
1989	1.603E+01	1.327E+00	1.374E+00	1.327E+00	1.089E+00	4.156E-02	1.056E-03	2.284E-02	4.085E+00	1.353E+00
1990	2.356E+01	1.391E+00	2.018E+00	1.391E+00	3.113E-01	8.963E-03	4.881E-05	1.248E-02	1.776E+00	1.206E+00

Table 4.2.5-1. (Continued)

Year	99-Tc	GSD	Uranium daughters	GSD	U-234	U-235	U-236	U-238	Total U	GSD
1991	6.022E+01	1.302E+00	2.737E+00	1.302E+00	4.724E+00	1.384E-01	4.724E-04	2.092E-01	5.062E+00	1.299E+00
1992	1.332E+01	1.298E+00	8.880E-02	1.298E+00	1.228E+00	3.589E-01	1.133E-03	2.078E-01	1.322E+01	1.326E+00
1993	1.434E+02	1.316E+00	1.355E-01	1.316E+00	2.123E+00	5.586E-02	2.383E-04	1.080E-01	2.309E+00	1.266E+00
1994	5.329E+00	1.190E+00	8.737E-02	1.190E+00	5.231E-01	1.655E-02	1.606E-04	9.989E-03	7.104E-01	1.214E+00
1995	1.411E+01	1.000E+00	5.664E-01	1.000E+00	1.313E+00	3.847E-02	3.889E-04	3.459E-02	1.386E+00	1.198E+00
1996[b]	1.045E+01	1.000E+00	1.161E+00	1.000E+00	2.999E+00	1.077E-01	2.154E-03	2.468E-01	3.356E+00	1.208E+00
1997[b]	1.607E+01	1.316E+00	1.785E+00	1.316E+00	2.121E+00	9.193E-02	1.051E-03	6.963E-01	2.910E+00	1.219E+00
1998[b]	1.576E+02	1.000E+00	1.751E+01	1.000E+00	8.890E+00	3.186E-01	1.058E-02	6.138E-01	9.833E+00	1.361E+00
1999[c]	3.692E+01	1.421E+00	4.103E+00	1.421E+00	1.966E+01	7.011E-01	1.336E-02	2.266E+00	2.264E+01	1.304E+00
2000[c]	1.120E+01	1.000E+00	1.245E+00	1.000E+00	4.167E+00	1.486E-01	2.831E-03	4.801E-01	4.798E+00	1.287E+00
2001[d]	3.552E-01	1.318E+00	3.552E-02	1.318E+00	2.042E-01	8.702E-03	1.954E-03	7.637E-02	2.913E-01	—

a. Average empirical Chi/Q value from 1964 through 1993 used to estimate the annual intake.
b. Applied release amounts as given in annual NESHAP reports 1995-1998 (ATL 2003h).
c. Applied average fractions for isotopic releases as given in annual NESHAP reports 1995-1996 (USEC 1996, 1997, 1998, 1999)
d. Air sample measurements by radionuclide, not gross alpha or gross beta-gamma.
e. For the years of 1955 and 1956, the potential intake of U-234 is based on an average fraction of 50% for U-234 activity toward the total uranium activity during the years 1957 through 1962.

4.2.6 Contribution of Airborne Radionuclides to External Dose

From the data shown in Table 4.2.5-1, the levels of annual intakes during the early years of operations at PORTS are two to three orders of magnitude higher than the years after 1966. High airborne concentrations can have a significant contribution to the direct radiation exposure through the pathway of submersion within a cloud of radioactive material. With such variations in the air concentrations used to determine the estimated annual intakes, a demonstration of the potential occupational impacts would show the potential contribution to the external site dose from submersion. Since the highest airborne intake is due to uranium daughters based on the 1955 intake, these radionuclides and concentration levels would demonstrate the most claimant-favorable conditions. The submersion dose was calculated using the air submergence dose factors in Federal Guidance Report No. 12 for the uranium daughter radionuclides of ^{241}Th, ^{234}Th, and ^{234m}Pa. As previously stated, the contribution of each radionuclide is 6 percent of the uranium daughter fraction for 241Th and the remainder equally split between ^{234}Th and ^{234m}Pa. The potential contribution to the annual external exposure from these uranium daughters at the highest airborne concentrations only yields 1.2 µrem. Therefore, airborne radioactivity at the PORTS site over all years of concern would have a negligible impact to the annual occupational external exposure.

4.3 EXTERNAL DOSE

Workers are subjected to external doses from ambient radiation levels. Until September 1981, external gamma radiation levels were measured with a calibrated open-shield Geiger-Muller (GM) tube and solid-state scalar tubes 3 feet above ground level. Beginning in September 1981, TLD readings of gamma radiation were taken and compared to GM measurements from earlier in the year. TLD readings were considered more accurate than GM measurements because the GM device takes more discrete readings and could miss fluctuations in gamma radiation levels (DOE 1982). The minimum detection limit for the GM tube assembly is not known; however, in the period of 1972 through 1981, the minimum dose rate detected was 0.0048 mrem per hour (DOE 1974). Lacking the technical specifications for the GM tube, this value can be used as a claimant-favorable minimum detection limit. Information is not available on the TLD detection limit.

Between 1981 and 1992, TLDs used to measure onsite environmental gamma radiation levels contained calcium fluoride:dysprosium (CaF_2:Dy) chips. Commercially available CaF_2-based environmental dosimeters have a detection limit of 5 mR per month that the dose reconstructor can use for the detection limit of the TLDs used in the early 1990s (ICN 2003). These TLDs were used because of their sensitivity.

Beginning in late 1992, PORTS began using TLDs containing lithium fluoride (LiF) chips to monitor ambient conditions. These TLDs had the same technology used to monitor worker doses (DOE 1996). TLDs used from 1981 through the mid-to-late 1990s were not calibrated to measure neutrons.

TLDs were upgraded in the mid-to-late 1990s to be able to measure beta, gamma, and neutron radiation. These TLDs can differentiate dose between the different forms of radiation and between shallow and deep dose. Neutron dose is included in the deep dose (DOE 1999a). The minimum limit of detection is 10 mrem for beta-gamma and 10 mrem for neutrons (ICN 2003).

Table 4.3-1 summarizes the history of the type of instruments used for ambient environmental monitoring.

Throughout the history of the site, environmental radiological measurements were taken at outdoor locations around Perimeter Road and at offsite locations to monitor public dose (locations 3, 12, 24,

29, A36, A39, A40, and 874). Since 1998, ambient radiological conditions at six major facilities (X-7725, X-326, X-345, X-744G, X-745C, and X-745E) at the interior of the site have been recorded, in addition to those at the Perimeter Road locations, to assist in monitoring worker dose. Figure 4.3-1 shows the locations of the public and worker monitoring points. Table 4.3-2 includes a description of each location. The TLDs used to monitor facilities at the interior of the site are outside the buildings.

Data collected by TLDs used to monitor public dose were used to establish ambient environmental doses for most areas and most years because these data are all that is available to establish ambient environmental conditions inside the site boundary. Ambient environmental doses for workers were not monitored consistently until 1998 (see Table 4.3-3) because of the emphasis on monitoring public dose and because throughout much of PORTS history most workers were monitored with personal dosimetry. The ambient environmental data for workers, listed in Table 4.3-3, do not provide enough information and are too specific to the buildings listed in the table for application over all years and all outdoor areas.

The Perimeter Road TLD locations changed over the years. For some of these locations (described in the previous paragraph), TLD readings might have been taken off the site during certain years, but are attributed to the particular location as the nearest point to the industrial area. For example, in the early years location 29 on the west side of the site was at the U.S. Highway 23 exit ramp of the main access road to the site. This location was later moved to the Perimeter Road/main access road intersection. This TBD assumes that all public dose readings were at the Perimeter Road.

Table 4.3-1. Instruments Used for Ambient Environmental Monitoring.

Instrument	Period of use	Minimum detection level
GM tube	1955-1981	1.0×10^{-4} mR per hour
TLD (CAF_2:Dy chip)	1981-1992	5 mR (based on current TLDs with CAF_2:Dy chip)
TLD (LiF chip)	1992 – mid to late 1990s	Unknown
ICN 760 TLD (LiF chip)	Mid to late 1990s – present	10 mrem for beta-gamma 10 mrem for neutrons

Table 4.3-2. Environmental Measurement Locations.

Designation (Other designations used over the years for measurements taken in this vicinity)	Location
3 (39, X-230-J2, A35)	Generally, south side of site, near south holding pond.
12 (PP933)	Power pole 933 opposite East Access Road.
24 (PP906)	Power pole 906 opposite North Access Road.
29 (10, PP862)	Power pole 862 opposite main access road.
A36 (X-611)	Vicinity of water treatment plant.
A39 (PP722)	Power pole 722 at south end of Pike Ave. between X-231 oil biodegradation plots.
A40 (35, X-100)	Vicinity of Guard headquarters.
PP518	Power pole 518 near X-104.
PP1406 (PP1404A)	Near X-7725 and warehouses on west side of site.
874	On power pole 874 at northwest corner of X-745C Depleted Uranium Storage Yard.

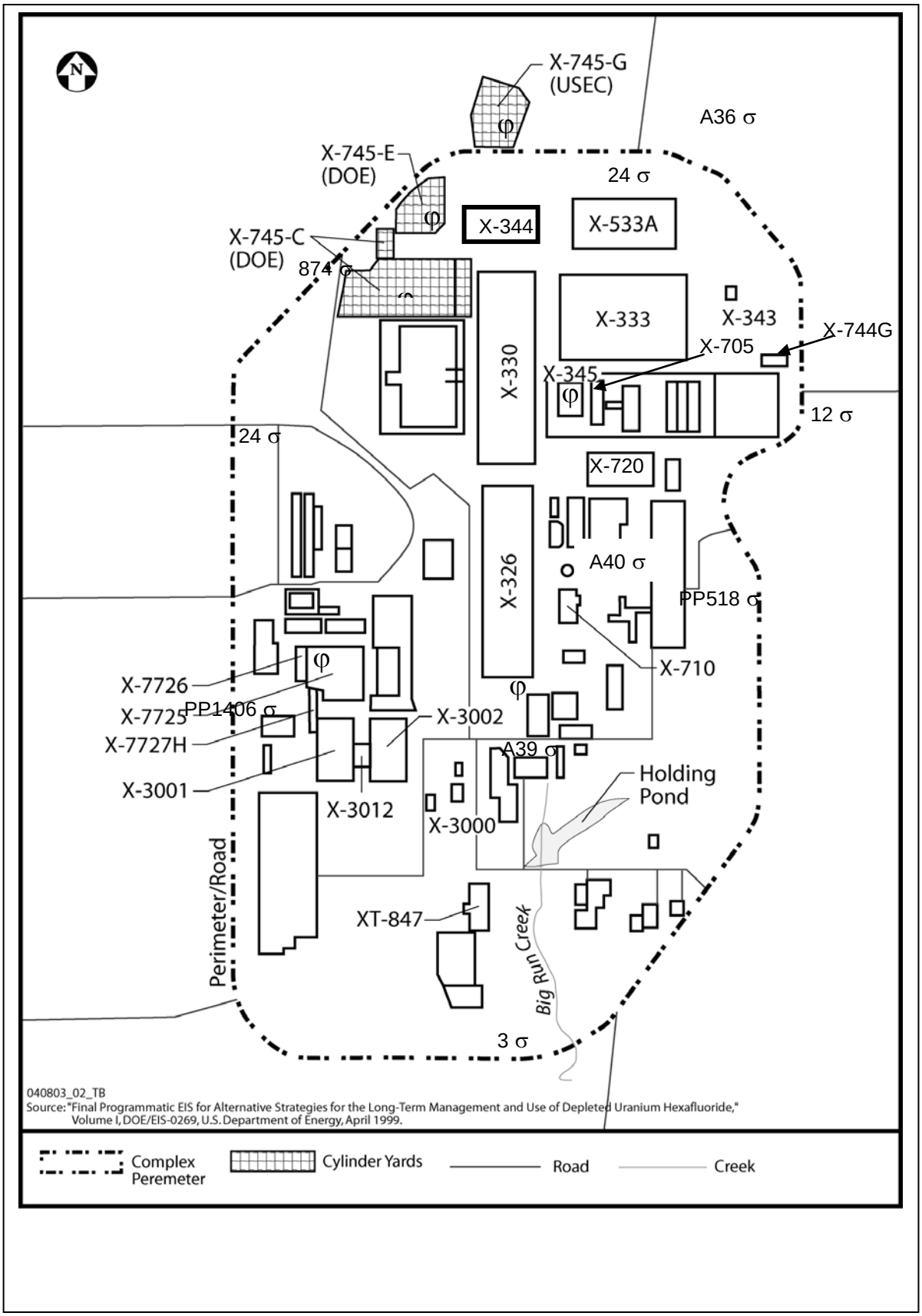
Designation (Other designations used over the years for measurements taken in this vicinity)	Location
X-7725	Waste Storage Facility.
X-326	Process Building.
X-330	Process Building.
X-333	Process Building.
X-344	Containment Building
X-345	Special Nuclear Material (SNM) Storage Building.
X-705	Decontamination Building.
X-720	Maintenance and Stores Building.
X-744G	Bulk Storage Building.
X-745C, X-745E	Depleted Uranium Cylinder Storage Yards.

Table 4.3-3. Site Interior Ambient Radiological Conditions.

Mean annual dose by location[a], [b] (mrem/year)												
	X-7725		X-326		X-345		X-744G		X-745C		X-745E	
Year	Avg. deep dose	Avg. shallow dose	Avg. deep dose	Avg. shallow dose	Avg. deep dose	Avg. shallow dose	Avg. deep dose	Avg. shallow dose	Avg. deep dose	Avg. shallow dose	Avg. deep dose	Avg. shallow dose
1998	21	36	2	4	19	26	—	—	35	39	20	21
1999	5	10	1	4	1	3	23	19	37	30	56	47
2000	14	—	0	—	0	—	285	—	122	—	178	—
2001	23	—	0	—	0	—	1,056	—	142	—	175	—

a. 2,000-hour work-year, prorated from an 8,736-hour year.
b. Includes beta, gamma, and neutron. Neutrons included in deep dose.
— No data.

Figure 4.3-1. Approximate TLD Locations.



Ambient Radiation

The environmental radiological profile has been developed for PORTS for use by dose reconstructors when personal dosimetry or bioassay program participation is not required. Annual site environmental reports, health physics surveys, and other reports were reviewed for data that would be useful in reconstructing ambient radiation levels. Data in these documents (see reference section for citations) include TLD radiation measurements. Tables 4.3-3 and 4.3.1-1 list the TLD results. The following sections discuss these tables further.

Table 4.3.1-1. Perimeter Road Ambient Radiological Conditions.

Mean annual dose by location[a, b] (mrem/year)										
Year	874	PP518	29 (10, PP862)	24 (PP906)	12 (PP933)	PP1406	3 (39, X-230-J2, A35)	A40 (35, (X-104)	A36 (X-611)	A39 (PP722)
1954	23.8	23.8	23.8	23.8	23.8	23.8	23.8	23.8	23.8	23.8
1972	—	—	23.6	23.8	22.4	—	24.0	—	—	—
1973	—	—	27.8	26.4	25.6	—	27.2	—	—	—
1974	—	—	28.2	27.0	26.0	—	26.8	—	—	—
1975	—	—	19.2	18.8	18.4	—	18.8	—	—	—
1976	—	—	20.6	20.4	20.0	—	20.0	—	—	—
1977	—	—	23.8	24.0	24.2	—	24.4	—	—	—
1978	—	—	21.8	21.4	20.6	—	21.2	—	—	—
1979	—	—	21.8	21.2	22.0	—	20.2	—	—	—
1980	—	—	21.4	20.8	20.8	—	21.6	—	—	—
1981	—	—	21.0	22.6	20.2	—	19.8	—	—	—
1982	—	—	14.4	15.6	14.6	—	16.2	17.4	—	—
1983	—	—	15.6	12.8	13.4	—	15.2	15.8	—	—
1984	—	—	12.4	6.4	12.8	—	20.4	11.4	—	—
1985	—	—	13.8	13.8	13.8	—	13.8	13.8	—	—
1986	239.3	—	20.3	16.4	19.1	—	16.9	16.4	20.6	19.1
1987	267.5	—	21.6	20.0	21.4	—	18.9	19.4	24.5	22.8
1988	260.6	—	19.6	17.4	19.4	—	20.8	16.6	23.2	21.6
1989	257.2	—	22.7	20.8	20.7	—	20.7	17.4	22.5	22.7
1990	250.7	—	23.7	17.9	24.7	—	20.0	19.6	21.0	—
1991	261.0	16.7	21.5	17.0	21.7	21.7	20.1	NR	20.9	—
1992	246.5	29.1	36.6	17.4	19.9	44.8	33.7	NR	25.6	—
1993	150.4	24.8	25.4	24.0	23.6	12.2	35.9	26.0	26.9	—
1994	27.8	22.9	24.6	30.3	24.6	22.1	29.8	19.9	32.3	—
1995	90.0	20.0	16.3	16.8	17.4	17.3	18.1	15.5	19.4	—
1996	112.0	14.0	22.1	26.9	28.4	13.4	15.7	10.5	29.7	—
1997	131.4	13.8	21.5	25.9	27.5	13.9	15.2	10.4	28.5	—
1998	135.0	22.0	23.3	23.3	24.3	23.1	24.7	20.4	24.3	—
1999	131.4	18.5	19.2	16.9	20.6	20.8	20.4	15.8	20.1	—
2000	149.2	24.5	22.0	20.6	29.5	22.0	24.0	19.7	23.8	—
2001	150.4	19.1	23.2	16.8	27.2	19.2	20.8	15.7	19.5	—
2002	142.4	18.4	23.3	15.7	27.1	18.8	20.4	14.8	18.4	—

a. 2,000-hour work-year, prorated from an 8,736-hour year.
b. Data for all years are from PORTS annual environmental reports with the exception of 1993 – 2002 for locations PP518, PP862, PP906, PP933, PP1406, A35, A40, A36 and 874, which were obtained from USEC by separate communication (ATL 2003f).
— No data.

4.3.1.1 Background Radiation

The ambient radiation measured by TLDs near Perimeter Road included natural background radiation, nuclear weapons testing fallout, and cosmic radiation (DOE 1994). These TLDs provided an indication of worker dose levels in the general proximity of Perimeter Road (with the exception of the cylinder storage yards), but not inside buildings. Table 4.3.1-1 lists annual dose levels for

locations near Perimeter Road. These data, available for 1954 and 1972-2002, are representative of dose for a 2000-hour work-year.

Because of a lack of environmental dosimetry data prior to the year 1972, an extrapolation will be assumed based on the available data. To support the assumptions, it is necessary to determine if there is a correlation between either the production levels (in metric tons uranium (MTU)) and/or the accumulation of material (i.e., tails in the cylinder yards) at the PORTS site, and the available environmental dosimetry. The annual production levels are available in Appendix IV of the "Recycled Uranium Mass Balance Project, Portsmouth, Ohio Site Report" (BJC 2000). However, annual data on the accumulation of tails material in the cylinder yards does not appear to be publicly available. Instead, an assumption is made that a significant majority of the feed material processed at PORTS would eventually leave the enrichment cascade as tails material and placed into long-term storage in one of the site's cylinder yards. Therefore, the cumulative quantity of feed material by year based on the data given in Appendix IV of the Recycled Uranium Mass Balance Project Report would be proportional to the accumulation of the tails material.

Figure 4.3.1.1-1 shows the cumulative quantity of feed material by year at PORTS versus the available environmental dosimetry data previously presented in Tables 4.3-3 and 4.3.1-1. This figure demonstrates that no direct relationship exists between the quantity of material accumulated at the PORTS site and the environmental dosimetry readings for the years available (after 1972). Figure 4.3.1.1-2 compares the same environmental dosimetry data to the annual MTU processed by PORTS from 1955 to 1997. This result would show if there is a potential for "building shine" as a contributor to external environmental radiation levels within the central area of PORTS. It is also important to compare the site readings to distant reading locations (i.e., background radiation levels). If there is an effect caused by the quantity of material processed then it should be an additive value to the site's background radiation levels. As shown in Figure 4.3.1.1-2, the environmental dosimetry data fluctuates more with the background radiation levels than the quantity of material processed. Therefore, the amount of feed stock processed per year (i.e., building shine or general radiation levels) does not significantly contribute to the environmental dose that all site workers would be expected to receive. It can therefore be assumed that the data presented in Tables 4.3-3 and 4.3.1-1 are representative of ambient radiological conditions for the entire period of 1954 to 2002.

In the past, personal dosimetry devices worn by site workers were stored in racks at the guard gates. Based on a 1989 study to review the practice of using storage racks, some employees were allowed to leave PORTS with their personal dosimetry. Beginning in 1999, all workers were allowed to leave PORTS with their personal dosimetry. Procedures require an exchange of dosimetry if the worker believes the dosimetry might have been inadvertently exposed away from PORTS due to dose from a medical exam, airport security system, or other sources (ATL 2003d).

Exposure to neutrons is not a factor in PORTS outdoor areas except in the uranium hexafluoride (UF_6) cylinder storage yards (see Section 4.3.1.2). Other areas where neutron dose might be of concern are indoors (Cardarelli 1997).

If background is not to be subtracted, the maximum value of 35.9 mrem from Table 4.3.1-1 can be used to assign annual ambient environmental dose to workers in areas near Perimeter Road such as the general employee parking lots; the guard gates on the outer perimeter of the security area; the switchyards; warehouses X-744S, T, and U; process buildings, and wastewater facility X-611. (The values in Table 4.3.1-1 for location 874 are much higher, but are specific to the UF_6 cylinder storage yards and not to the rest of the facility.) If background is to be subtracted, the ambient radiation for these areas should be assigned a value of 0 mrem because of the argument presented above.

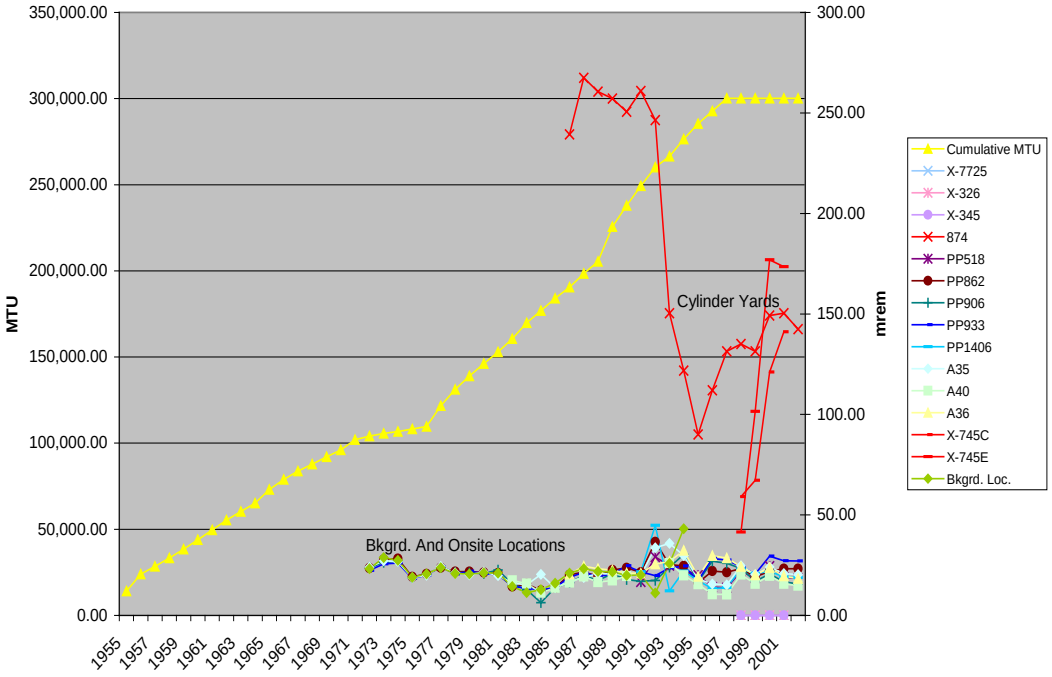


Figure 4.3.1.1-1. Cumulative Quantity of Material by Year

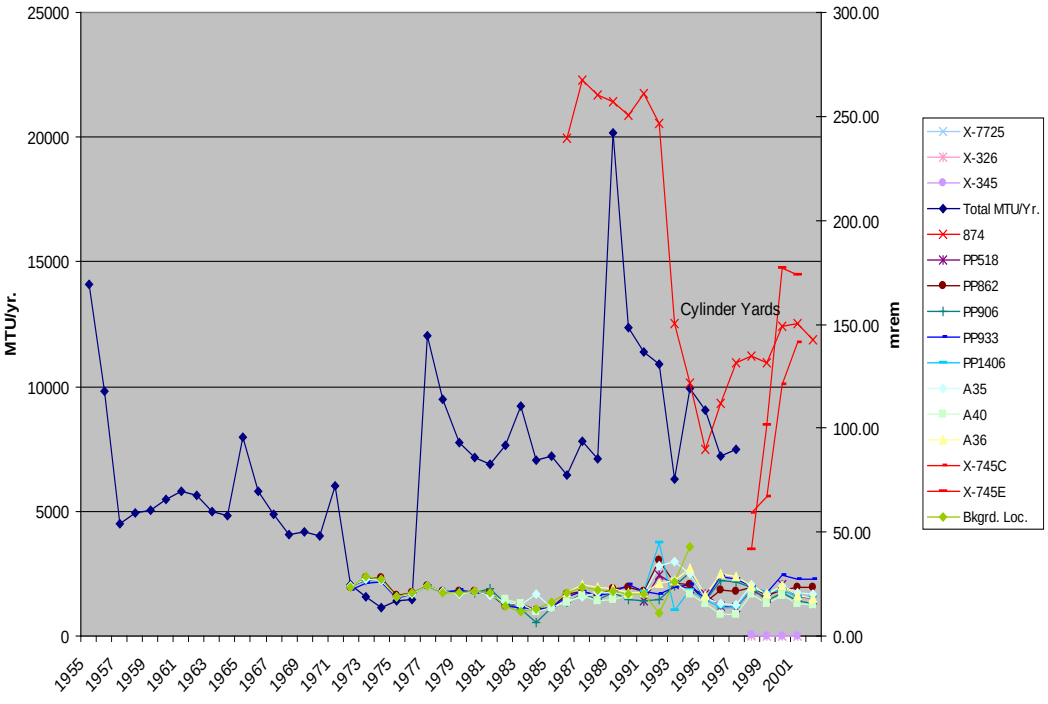


Figure 4.3.1.1-2. Annual MTU vs. Work Year Doses (2000 hrs/yr)

The data in Table 4.3-3 are representative of ambient radiological conditions for the areas immediately surrounding X-7725, X-326, X-345, and 744G. These data are applicable only to 1998 through 2001, as specified in the table. It is difficult to apply the data in Table 4.3-3 for other years because of its variability and limited amount. The value of 35.9 mrem per year should be used for 1954 to 1997 outside these four buildings. For 744G, an outdoor ambient dose of more than 1,000 mrem for 2001 (2,000-hour work-year) is expected because of the increasing inventory of nuclear material stored in this facility (ATL 2003e). The data in Table 4.3-3 associated with X-745C and X-745E cylinder storage yards are discussed in Section 4.3.1.2.

4.3.1.2 Ambient Radiological Conditions in Cylinder Storage Yards

Several TLDs were near the UF_6 cylinder storage yards, so elevated ambient radiation levels are likely near the storage yards in comparison to other outdoor areas of the site. TLD 874, on a power pole on Perimeter Road at the X-745C cylinder storage yard, monitors public dose. The average dose from 1986 to 2002 was 1,116 mrem per year with a maximum of 1,172 mrem over 24 hours and 365 days per year. This is equivalent to an average of 255 mrem and a maximum of 267 mrem for a 2,000-hour work-year, as indicated in Table 4.3.1-1 for TLD 874. An approximately 40% drop in dose readings at this location in 1993 was probably due to a change in the type of TLD monitor in late 1992 (this cannot be confirmed but it corresponds to a period in which TLDs were upgraded). The TLDs used to monitor this dose were not calibrated to detect neutrons until 1998. Because the TLD on the power pole was probably not within several feet of a UF_6 cylinder but was used to monitor public dose, neutrons are not an exposure factor; the 10-ton cylinders produced radiation levels of 0.5 mrem/hr neutron dose equivalent (DE) rate at the surface and would quickly decrease to background radiation levels due to distance (i.e., $1/r^2$ rule) (Cardarelli 1997).

In comparison, Table 4.3-3 lists results for a different set of TLDs in cylinder yards X-745C and X-745E from 1998 to 2001. These TLDs are in the cylinder yards, not on the perimeter of the yards, to focus on worker dose. They were calibrated to measure neutrons, which are included in the values for deep dose equivalent. The average dose for the two cylinder yards over 4 years was 130 mrem (deep plus shallow dose), with a maximum of 178 mrem. These values are less than those measured on the power pole.

In another study, the average radiation dose to cylinder yard workers from 1990 to 1995 ranged from 55 to 196 mrem/year, based on personal dosimetry (DOE 1999b). These workers conducted activities in other site facilities wearing the same TLDs and probably did not work in the cylinder yards 2,000 hours each year.

Given the above information, an ambient radiation dose of 267 mrem (2,000 hour work-year) should be applied to cylinder yard workers as claimant-favorable. The portion of this dose attributable to neutrons can be assumed to be 178 mrem per year. This value for the annual neutron dose is obtained from Table 4.3-3 as the maximum annual deep dose equivalent reported in the period 1998 to 2001 for the cylinder storage yards X-745C and X-745E. This assumes that the neutron dose is the only contributor to the deep dose equivalent. All of these assumptions are claimant-favorable.

4.3.1.3 Ambient Radiological Conditions Inside Buildings

Because all workers were badged and monitored throughout much of the PORTS operating history, coworker dose data can be used to assign dose to unmonitored workers for the periods during which all workers were not badged, or were badged but not analyzed. Other sources of information that describe potential radiation doses can be used as backup to coworker data.

In 1983, a survey by the Centers for Disease Control and Prevention (CDC) recorded area beta and gamma measurements for buildings X-326, X-330, X-333, X-344, X-705 and X-720. The highest gamma measurement was 4,000 mrem per year. The highest median gamma measurement was 160 mrem per year. Both of these measurements occurred in X-344 (CDC 1987). The 160-mrem median gamma measurement can be used to indicate conservative ambient conditions inside all buildings that contain radiological materials or contamination. The highest gamma measurement of 4,000 mrem is not an external dose reading that could be considered reflective of overall operations and should not be assumed over an exposure period because personal dosimetry results do not indicate this type of exposure level at PORTS, according to the information from REMS (Section 4.3.1.2).

4.4 UNCERTAINTY

As discussed in the previous sections, estimates of annual intakes employed conservative (i.e., claimant-favorable) assumptions. For example, applying maximum annual dose to all years is a claimant-favorable assumption.

The locations of the monitoring points from which data are summarized in this TBD add uncertainty to the results. These points, as stated above, were around the site perimeter and off the site to monitor public dose. Until 1998, external environmental exposures were not monitored in relation to workers; that is, monitoring locations were not normally at the interior of the site among the process buildings. Because of data availability, however, public dose information had to be used for worker environmental doses. The maximum value of environmental dose is recommended for years for which data are unavailable to compensate for lack of worker-specific environmental dose information.

All external environmental dose data are adjusted to reflect a 2,000-hour work-year. The data were originally reported in site environmental reports as representative of an employee who works at the site 24 hours per day, 365 days per year. Using a permanently located employee at the site, however, is an unrealistic assumption that might overstate onsite environmental doses.

REFERENCES

ATL (Advanced Technologies and Laboratories International, Inc.), 2003a, "Film Badge Program Procedures," Memorandum to File, Germantown, Maryland, October 22.

ATL (Advanced Technologies and Laboratories International, Inc.), 2003b, "Application of Empirical Chi/Q When Ambient Air Samples Are Not Available," Memorandum to File, Germantown, Maryland, September 2.

ATL (Advanced Technologies and Laboratories International, Inc.), 2003c, "Portsmouth High Volume Ambient Air Sampling Data for 1993 through 2000," Memorandum to File, Germantown, Maryland, October 20.

ATL (Advanced Technologies and Laboratories International, Inc.), 2003d, "Dosimetry Badging During Non-Work Hours," Memorandum to File, Germantown, Maryland, October 20.

ATL (Advanced Technologies and Laboratories International, Inc.), 2003e, "Environmental Dose at 744-G and Treatment of Background," Memorandum to File, Germantown, Maryland, October 20.

ATL (Advanced Technologies and Laboratories International, Inc.), 2003f, "Quarterly External Gamma Radiation Levels for 1993 to 2002," Memorandum to File, Germantown, Maryland, October 21.

ATL (Advanced Technologies and Laboratories International, Inc.), 2003g, "Historical Information Concerning the Release of ^{99}Tc from PORTS Prior to 1975," Memorandum to File, Germantown, Maryland, December 1.

ATL (Advanced Technologies and Laboratories International, Inc.), 2003h, "USEC NESHAP Reports for the Years 1995 through 2002," Memorandum to File, Germantown, Maryland, December 1.

BJC (Bechtel Jacobs Company LLC) 2000, Recycled Uranium Mass Balance Project, Portsmouth, Ohio Site Report, BJC/PORTS 139/R1, U. S. Department of Energy, Office of Environmental Management, Washington, DC

Cardarelli, J, 1997, *NIOSH Health Hazard Evaluation Report*, HETA 96-0198-2651, Portsmouth, Ohio.

CDC (Centers for Disease Control), 1987, *Health Physics Survey Report of Portsmouth Gaseous Diffusion Plant*, Division of Surveillance, Hazard Evaluations and Field Studies, Cincinnati, Ohio, September.

DOE (U.S. Department of Energy), 1973, *Portsmouth Gaseous Diffusion Plant Environmental Monitoring Report – 1972*, GAT-740, Goodyear Atomic Corporation, Piketon, Ohio, February 28.

DOE (U.S. Department of Energy), 1974, *Portsmouth Gaseous Diffusion Plant Environmental Monitoring Report* – 1973, GAT-781, Goodyear Atomic Corporation, Piketon, Ohio, May 3.

DOE (U.S. Department of Energy), 1975, *Portsmouth Gaseous Diffusion Plant Environmental Monitoring Report – 1974*, GAT-824, Goodyear Atomic Corporation, Piketon, Ohio, May 1.

DOE (U.S. Department of Energy), 1976, *Portsmouth Gaseous Diffusion Plant Environmental Monitoring Report for Calendar Year 1975*, GAT-866, Goodyear Atomic Corporation, Piketon, Ohio, April 30.

DOE (U.S. Department of Energy), 1977, *Portsmouth Gaseous Diffusion Plant Environmental Monitoring Report for Calendar Year 1976*, GAT-914, Goodyear Atomic Corporation, Piketon, Ohio, August 8.

DOE (U.S. Department of Energy), 1978, *Portsmouth Gaseous Diffusion Plant Environmental Monitoring Report for Calendar Year 1977*, GAT-955, Goodyear Atomic Corporation, Piketon, Ohio, May 31.

DOE (U.S. Department of Energy), 1978, *Portsmouth Gaseous Diffusion Plant Environmental Monitoring Report for Calendar Year 1978*, GAT-2001, Revision 1, Goodyear Atomic Corporation, Piketon, Ohio, July 5.

DOE (U.S. Department of Energy), 1980, *Portsmouth Gaseous Diffusion Plant Environmental Monitoring Report for Calendar Year 1979*, GAT-2005, Goodyear Atomic Corporation, Piketon, Ohio, June 11.

DOE (U.S. Department of Energy), 1981, *Portsmouth Gaseous Diffusion Plant Environmental Monitoring Report for Calendar Year 1980*, GAT-2009, Goodyear Atomic Corporation, Piketon, Ohio, August 8.

DOE (U.S. Department of Energy), 1982, *Portsmouth Gaseous Diffusion Plant Environmental Monitoring Report for Calendar Year 1981*, GAT-2013, Goodyear Atomic Corporation, Piketon, Ohio, April.

DOE (U.S. Department of Energy), 1983, *Portsmouth Gaseous Diffusion Plant Environmental Monitoring Report for Calendar Year 1982*, GAT/GDP-1111, Goodyear Atomic Corporation, Piketon, Ohio, April 27.

DOE (U.S. Department of Energy), 1984, *Portsmouth Uranium Enrichment Complex Environmental Monitoring Report for Calendar Year 1983*, GAT-4012, Goodyear Atomic Corporation, Piketon, Ohio, April.

DOE (U.S. Department of Energy), 1985, *Portsmouth Uranium Enrichment Complex Environmental Monitoring Report for Calendar Year 1984*, GAT-4016, Goodyear Atomic Corporation, Piketon, Ohio, May.

DOE (U.S. Department of Energy), 1985, *Portsmouth Uranium Enrichment Complex Environmental Monitoring Report for Calendar Year 1984*, GAT-4017, Goodyear Atomic Corporation, Piketon, Ohio, May.

DOE (U.S. Department of Energy), 1986, *Portsmouth Uranium Enrichment Complex Environmental Monitoring Report for Calendar Year 1985*, GAT-1138, Goodyear Atomic Corporation, Piketon, Ohio, May.

DOE (U.S. Department of Energy), 1987, *Environmental Surveillance of the U.S. Department of Energy Portsmouth Gaseous Diffusion Plant and Surrounding Environs During 1986*, ES/ESH-1/V4, POEF-1158, Martin Marietta Energy Systems, Oak Ridge, Tennessee, April.

DOE (U.S. Department of Energy), 1989, *Portsmouth Gaseous Diffusion Plant Site Environmental Report for 1988*, ES/ESH-8/V4, POEF-2010, Martin Marietta Energy Systems, Oak Ridge, Tennessee, May.

DOE (U.S. Department of Energy), 1990, *Portsmouth Gaseous Diffusion Plant Environmental Report for 1989*, ES/ESH-13/V4, POEF-2025, Martin Marietta Energy Systems, Oak Ridge, Tennessee, October.

DOE (U.S. Department of Energy), 1991, *Portsmouth Gaseous Diffusion Plant Environmental Report for 1990*, ES/ESH-18/V4, POEF-2070, Martin Marietta Energy Systems, Oak Ridge, Tennessee, September.

DOE (U.S. Department of Energy), 1992, *Portsmouth Gaseous Diffusion Plant Environmental Report for 1991*, ES/ESH-22/V4, POEF-2090, Martin Marietta Energy Systems, Oak Ridge, Tennessee, October.

DOE (U.S. Department of Energy), 1993, *Portsmouth Gaseous Diffusion Plant Environmental Report for 1992*, ES/ESH-37, POEF-3030, Martin Marietta Energy Systems, Oak Ridge, Tennessee, September.

DOE (U.S. Department of Energy), 1994, *Portsmouth Gaseous Diffusion Plant Annual Site Environmental Report for 1993*, ES/ESH-50, POEF-3050, Martin Marietta Energy Systems, Oak Ridge, Tennessee, November.

DOE (U.S. Department of Energy), 1996, *U.S. Department of Energy Portsmouth Site Annual Environmental Report for 1994*, ES/ESH-63, POEF-3055, Lockheed Martin Energy Systems, Oak Ridge, Tennessee, March.

DOE (U.S. Department of Energy), 1997, *U.S. Department of Energy Portsmouth Annual Environmental Report for 1995*, POEF-LMES-85, Lockheed Martin Energy Systems, Piketon, Ohio, April.

DOE (U.S. Department of Energy), 1997, *U.S. Department of Energy Portsmouth Annual Environmental Data for 1996*, DOE/OR/11-1655 & D1, POEF-LMES-157 & D1, Lockheed Martin Energy Systems, Piketon, Ohio, September.

DOE (U.S. Department of Energy), 1998, *U.S. Department of Energy Portsmouth Annual Environmental Data for 1997*, DOE/OR/01-1752 & D1, BIC/PORTS-39, Bechtel Jacobs Company, Piketon, Ohio, September.

DOE (U.S. Department of Energy), 1999a, *U.S. Department of Energy Portsmouth Annual Environmental Data for 1998*, DOE/OR/11-3031, Bechtel Jacobs Company, Piketon, Ohio, December.

DOE (U.S. Department of Energy), 1999b, *Final Programmatic Environmental Impact Statement for Alternative Strategies for the Long-Term Management and Use of Depleted Uranium Hexafluoride*, Volume I, DOE/EIS-0269, Office of Nuclear Energy, Science and Technology, Washington, D.C., April.

DOE (U.S. Department of Energy), 2001, *U.S. Department of Energy Portsmouth Annual Environmental Data for 1999*, DOE/OR/11-3052 & D1, Bechtel Jacobs Company, Piketon, Ohio, January.

DOE (U.S. Department of Energy), 2001, *U.S. Department of Energy Portsmouth Annual Environmental Data for 2000*, DOE/OR/11-3077 & D1, Bechtel Jacobs Company, Piketon, Ohio, October.

DOE (U.S. Department of Energy), 2002, U.S. Department of Energy Portsmouth Annual Environmental Data for 2001, DOE/OR/11-3106 & D1, Bechtel Jacobs Company, Piketon, Ohio, November.

Goslow, G. A., 1986. *Historical Radionuclide Release Report*, GAT-1124, Goodyear Atomic Corporation, Piketon, Ohio, January.

Goslow, G. A., 2003, "Requested Data for NIOSH Dose Reconstruction, Part 1," August 26.

ICN (ICN Worldwide Dosimetry Service), 2003, "Product Data Sheets – TLDs," https://www.dosimetry.com/dx/external/default.html, (accessed on September 23, 2003).

ICRP (International Commission on Radiological Protection), 2001, *ICRPDOSE2: ICRP Database of Dose Coefficients: Workers and Members of the Public*, Version 2.0.1, copyright 1998-2001.

NIOSH (National Institute of Occupational Safety and Health), 2002, *Internal Dose Reconstruction Implementation Guidelines*, Rev. 0, OCAS-IG-002, Cincinnati, Ohio.

GLOSSARY

alpha radiation
Radiation consisting of charged particles of the isotope helium-4, consisting of two protons and two neutrons. Alpha radiation is common for heavier nuclei.

background radiation
Background radiation is the radiation received that is not associated with a worker's occupation. This includes cosmic, terrestrial, and man-made sources.

Becquerel
The derived SI unit of radioactivity equal to one disintegration per second.

beta radiation
Radiation consisting of charged particles of very small mass (i.e., the electron) emitted spontaneously from the nuclei of certain radioactive elements. Most (if not all) of the direct fission products emit beta radiation. Physically, the beta particle is identical to an electron moving at high velocity.

deep dose equivalent
The dose equivalent at the respective depth of 1.0 cm in tissue.

derived concentration guidelines (DCG)
A calculated concentration that would result in an annual dose of 100 millirem.

dosimetry
The science of assessing absorbed dose, dose equivalent, effective dose equivalent, etc., from external and/or internal sources of radiation.

exposure
As used in the technical sense, exposure refers to a measure expressed in roentgens (R) of the ionization produced by photon radiation (i.e., gamma and X-rays) in air.

film badge
In general, a "film packet" that contains one or more pieces of film in a light-tight wrapping. The film when developed has an image caused by radiation that can be measured using an optical densitometer.

gamma rays
Electromagnetic radiation (photons) originating in atomic nuclei and accompanying many nuclear reactions (e.g., fission, radioactive decay, and neutron capture). Physically, gamma rays are identical to X-rays of high energy, the only essential difference being that X-rays do not originate in the nucleus.

neutron
A basic particle that is electrically neutral, having nearly the same mass as the hydrogen atom.

radiation
Alpha, beta, neutron, and photon radiation.

radiation exposure monitoring system (REMS)
An online DOE database that contains records of personal dosimetry information.

radioactivity
The spontaneous emission of radiation, generally alpha or beta particles, gamma rays, and neutrons from unstable nuclei.

rem
A unit of dose equivalent, which is equal to the product of the number of rad absorbed and the "quality factor."

shallow dose equivalent
The dose equivalent at the respective depth of 0.07 mm of the skin or an extremity.

thermoluminescent dosimeter (TLD)
A holder containing solid chips of material that when heated release the stored energy as light. The measurement of this light provides a measurement of absorbed dose.

uranium hexafluoride (UF_6) cylinder storage yard
Storage yards at the Paducah and Portsmouth Gaseous Diffusion Plants and Oak Ridge K-25 site. These yards maintain cylinders containing depleted UF_6. The cylinders, which typically weigh 10 and 14 tons, contain depleted UF_6 primarily in a solid form.