

<p>ORAU Team Dose Reconstruction Project for NIOSH</p> <p>Technical Basis Document for the K-25 Site – Occupational Environmental Dose</p>	<p>Document Number: ORAUT-TKBS-0009-4 Effective Date: 12/29/2004 Revision No.: 00 Controlled Copy No.: _____ Page 1 of 19</p>
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TABLE OF CONTENTS

<u>Section</u>	<u>Page</u>
Record of Issue/Revisions	3
Acronyms and Abbreviations	4
4.1 Introduction.....	5
4.2 Internal Dose from Onsite Atmospheric Radionuclide Concentrations	5
4.2.1 Onsite Releases to Air	5
4.3 External Dose.....	9
4.3.1 Ambient Radiation	9
4.3.2 Other Potential Sources.....	10
4.3.3 Review of External Dose.....	10
4.4 Uncertainty	10
References	12
Glossary	13
Attachment 4D Occupational Environmental Dose	15
4D.1 Internal Dose from Onsite Concentrations of Radionuclides	16
4D.1.1 Onsite Releases to Air	16
4D.2 External Dose.....	16
4D.2.1 Ambient Radiation	16

LIST OF TABLES

<u>Table</u>	<u>Page</u>
4D-1 Uranium annual intake per year (Bq) and geometric standard deviation.....	17
4D-2 Plutonium annual intake per year (Bq) and geometric standard deviation	18
4D-3 Neptunium and technetium annual intake per year (Bq) and geometric standard deviation.....	19
4D-4 Background gamma radiation levels at two perimeter ORGDP stations	19

LIST OF FIGURES

<u>Figure</u>	<u>Page</u>
4-1 K-25 Site map with environmental monitoring stations noted	6

RECORD OF ISSUE/REVISIONS

ISSUE AUTHORIZATION DATE	EFFECTIVE DATE	REV. NO.	DESCRIPTION
Draft	09/26/2003	00-A	New Technical Basis Document for the K-25 Site – Occupational Environmental Dose. Initiated by Joseph L. Alvarez.
Draft	10/22/2003	00-B	Incorporates comments from NIOSH review. Initiated by Joseph L. Alvarez.
Draft	01/26/2004	00-C	Incorporates additional comments from NIOSH review. Initiated by Jay J. Maisler.
Draft	05/21/2004	00-D	Incorporates additional comments from NIOSH review. Initiated by Jay J. Maisler.
Draft	08/17/2004	00-E	Incorporates additional comments from NIOSH review. Initiated by Jay J. Maisler.
Draft	12/13/2004	00-F	Incorporates additional comments from NIOSH review. Initiated by Jay J. Maisler.
12/29/2004	12/29/2004	00	First approved issue. Initiated by Jay J. Maisler.

ACRONYMS AND ABBREVIATIONS

DOE	U.S. Department of Energy
GSD	Geometric Standard Deviation
ORDR	Oak Ridge Dose Reconstruction
ORHASP	Oak Ridge Health Agreement Steering Panel
ORGDP	Oak Ridge Gaseous Diffusion Plant
ORR	Oak Ridge Reservation
RU	recycled uranium
TLD	thermoluminescent dosimeter
U.S.C	United States Code

4.1 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 facility” as defined in the *Energy Employee Occupational Illness Compensation Program Act of 2000* [42 U.S.C. Sections 7384l (5) and (12)].

The occupational environmental dose refers to the dose received by unmonitored workers outside the facilities. These doses can be internal and external depending on the characteristics of the individual radionuclides. When inhaled, most radionuclides would produce a dose to various organs in the body. These radionuclides are addressed in the following sections.

4.2 INTERNAL DOSE FROM ONSITE ATMOSPHERIC RADIONUCLIDE CONCENTRATIONS

The internal dose for workers outside the facilities is determined from the air concentrations resulting from individual facility releases, and from ground-level releases. The method used to predict air concentrations includes the contribution from soil resuspension indirectly.

Unmonitored workers can be exposed to occupational doses internally from onsite releases to the air and from the resuspension of radioactive materials in soil, and externally from ambient radiation and releases of radioactive noble gases to the air.

4.2.1 Onsite Releases to Air

The internal dose for workers outside the facilities is determined from the air concentrations resulting from releases from stacks, individual building releases, and from the purge cascade and other operations at the Oak Ridge Gaseous Diffusion Plant Site (ORGDP), which this document refers to as the K-25 Site. Unmonitored workers can be exposed to occupational doses internally from onsite releases to the air or from ambient radiation.

A calculation was performed using well-documented source terms developed for the Oak Ridge Dose Reconstruction (ORDR) (Burmeister 1996, 1997, Burns 1997), coupled with some documented environmental monitoring data, to estimate radionuclide-specific airborne concentrations for ²³⁴U, ²³⁵U, ²³⁸U, ²³⁸Pu, ²³⁹Pu, ²⁴⁰Pu, ²⁴¹Pu, ²⁴²Pu, ²³⁷Np, and ⁹⁹Tc (Shonka 2003). These radionuclides were determined to account for the majority of the potential missed dose from inhalation and submersion pathways. The 50th- and 95th-percentile air concentrations for the uranium nuclides were used to compute the geometric standard deviation (GSD) of the uranium source term. This GSD was applied on a year-by-year basis to all other nuclides in the source term. The 50th-percentile air concentration for uranium calculated air concentrations was then used to derive annual intakes based on an assumed individual ventilation rate of 2,400 cubic meters per year. In addition, the GSD of these intakes was estimated based on the 95th-percentile source terms using the following relationship:

$$GSD = \left(\frac{\text{Nuclide activity 95}^{\text{th}} \text{ percentile}}{\text{Nuclide activity 50}^{\text{th}} \text{ percentile}} \right)^{\left(\frac{1}{1.65585} \right)}$$

These annual intakes and GSDs, by year of operation, are listed in Attachment 4D in Tables 4D-1 through 4D-3.

In general, the K-25 Site was a major industrial site with upwards of 2,500 megawatts of heat dissipated in the water and air during the years of operation. Releases from the gaseous diffusion plants had both mechanical momentum and significant thermal energy that lifts the effective height of the release point by 10 to 20 meters. Depending on the location of a specific release point relative to other turbulent thermal sources such as cooling towers, the diffusion can be considerable over that predicted by simple Gaussian plume models. There were multiple release points spread over the site.

Figure 4-1 is a drawing of the K-25 Site showing major buildings and the environmental stations selected to estimate dilution.

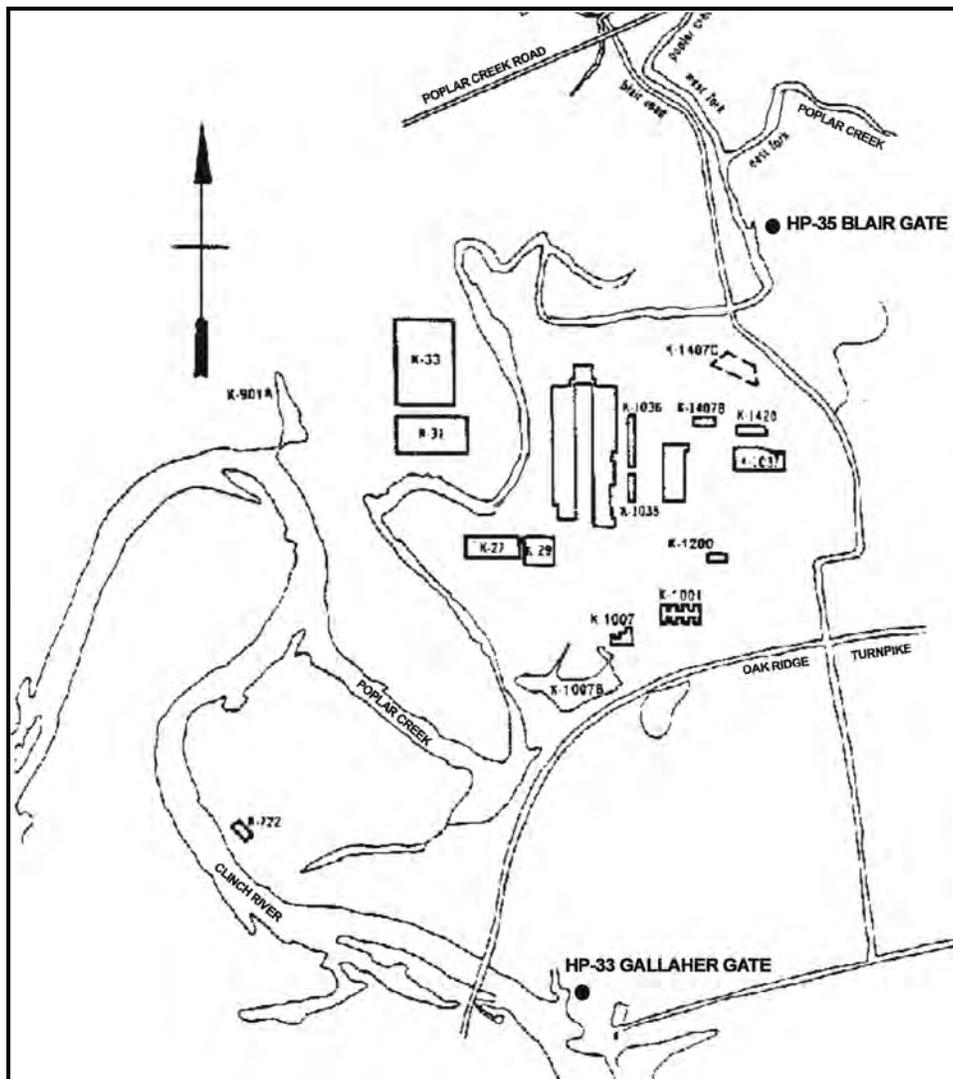


Figure 4-1. K-25 Site map with environmental monitoring stations noted.

To provide a conservative estimate, environmental monitors were chosen that were adjacent to the K-25 Site and generally located in upwind and downwind locations relative to release points. These were where maximum ground-level concentrations from the site would occur. The annual air concentrations for several years in gross alpha and gross beta were divided by the source term for the same years expressed in the same units to provide an equivalent annual dilution volume. This dilution volume was then applied to all years. Use of gross values provides added conservatism in that nearly all of the reported air concentration appears to be background. In addition, the administration area is generally crosswind from the bulk of the industrial site where most of the release points were located.

The values in Tables 4D-1 through 4D-3 in Attachment 4D are estimates of annual intakes derived from the calculated outdoor air concentrations at the K-25 Site and, as such, might not be appropriate for calculating organ doses for workers located inside buildings. However, these values probably represent the upper bound of potential intakes for an individual working in outside areas where individual monitoring was not required. It is assumed that these intakes occurred uniformly in the year listed, so for partial years of employment the intake can be proportionally adjusted by the fraction of the year worked. In addition, these intakes were derived based on an assumed breathing rate of 2,400 cubic meters per year. Therefore, adjustments might be required based on the job classification depending on the individual performing light work or heavy work, for which the respiration rate would be 1.2 m³/h or 1.7 m³/h, respectively. For purposes of dose estimation, radioactive decay was not considered.

Plutonium and uranium releases to the atmosphere represent the greatest potential radiological risk of all the alpha-emitting radionuclides in the K-25 Site environment. For purposes of dose reconstruction, the isotopes of plutonium and uranium contained in the mix that results in the maximum organ dose should be selected and assumed to be present at 100%. This assumption will result in a small, but claimant-favorable, overestimation of the actual dose.

Releases of radionuclides for the K-25 Site were studied as a part of the ORDR. This study was conducted from 1994 to 1999 for the Tennessee Department of Health by scientists and engineers from ChemRisk, a service of McLaren/Hart Environmental Services, assisted by SENES Oak Ridge and Shonka Research Associates. As an element of Task 6 of that study, a source term reconstruction was performed for the primary nuclides of concern, that of the uranium nuclides at various enrichments. In addition, a screening source term was developed for ²³⁷Np and ⁹⁹Tc as an element of Task 7 of the ORDR. The screening source term used plant data and did not independently reconstruct the release. These source terms were used as the starting point for this effort.

There were two uranium source terms developed for K-25 by ORDR. The final source term reported in the Task 6 Final Report did not have 50th- and 95th-percentile values, and several limitations of the work were noted (ChemRisk 1999). These included limited release information for the S-50 plant for 1944, the sole year of operation for that facility. In addition, the K-25 Site releases did not resolve releases from K-1131 against the large material unaccounted for (mass balance) for that building. Both of those source terms are primarily natural enrichment material of low specific activity. In addition, undocumented releases from the cascade were noted from frequent incident reports that had no quantity release data associated with the incident. Finally, corrections for losses from stack sampling were not rigorously treated. One of the primary stack monitors whose results would be impacted is the K-1131 stack, which is primarily a source of near-natural enrichment uranium.

The alternative uranium release estimate was an attempt to address some of the issues mentioned above, and provided 50th- and 95th-percentile estimates. It estimated uncertainty by fitting release

information for periods when it was available, and sampling from the fitted data to fill in periods where monitoring data was unavailable. The Oak Ridge Health Agreement Steering Panel (ORHASP) felt that the conclusions reached were unchanged for either of these treatments, and preferred to use the data without fits, as it reflected the reported plant releases. The uranium release from the two estimates was 16.7 curies (Burns 1997) and 23.2 curies (Burmeister 1996) (50th percentile), respectively.

The screening estimate made for ORDR Task 7 took the inventory of ^{99}Tc and ^{237}Np estimated by the K-25 Site staff and the Site's release fraction and distributed that material throughout the years of recycled uranium (RU) use by establishing the ratio of uranium releases for that period. After 1977, a nominal continuing annual release was used. No estimate was made for plutonium in that Task 7 effort.

To provide a conservative value that was claimant-friendly, the 50th-percentile alternative release estimate from the ORDR source term was used for uranium, and the 95th-percentile was used to calculate the geometric standard deviation along with the following modifications. Plutonium was added in a manner consistent with the previous estimates for ^{99}Tc and ^{237}Np , and with an assumption that the material consisted of five plutonium nuclides in a ratio consistent with weapons-grade plutonium.

Estimates of site releases for ^{99}Tc were used for 1981 and 1982 in lieu of the ORDR estimates because the reported value was larger than the estimate (Jordan 1983). On average, this serves to overstate the ^{99}Tc releases, since the added curies (beyond those asserted by ORDR Task7) had been distributed by the ORDR calculation among all years of RU use.

The K-25 Site is relatively small compared to larger U.S. Department of Energy sites, and radioactive material is released from a large number of release points of varying flow rates and elevations (Shonka 1996). Large quantities of electric power were used (depending on production), with heat rejected, in part, to the atmosphere using a combination of ventilation and cooling towers. Ventilation rates for removal of process heat varied to a considerable degree with the seasons. This, along with building wake effects, result in a highly turbulent environment that tends to mix and dilute the released materials far beyond the rates predicted by atmospheric transport codes. In light of the multiple release points and complexities of transport, a simple conservative method for estimating site concentrations was chosen.

The gross alpha and gross beta boundary air concentrations, reported by the site in environmental monitoring reports from the 1970s and 1980s, were ratioed to the source term (expressed in equivalent units) to establish a dilution volume that was used for all years. The reported boundary air concentrations were nearly identical to those recorded by offsite monitors, indicating that the concentrations came primarily from sources that were not associated with releases from K-25 operations. Thus, this method provides an estimate of annual concentrations that is overstated and conservative. This excess conservatism is used to account for the limitations in the source term noted above as well as other airborne sources including soil resuspension. No separate analysis for soil resuspension was performed.

Once the site boundary intakes are established, the intakes were scaled back to within the fence line using Fig 11.3.4 in the Radiological Health Handbook (Shleien 1992). From this figure based on ground level releases the concentration factor ($\bar{u} \chi/Q$) to distance (x) ratio is -10^2 to 10^1 is evident. Ground level releases provides for the highest intake in the close distances.

In the concentration factor, the average wind speed (\bar{u}) between the boundary location and the fence line would be the same, and therefore isn't a contributing factor in relating the two distances. Also the release (Q) is the same in both cases, leaving the only change in concentration (χ).

The boundary concentrations, as reported in referenced calculations and reports, were taken at approximately 2500 meters from the center of the plant, the fence line is generally less than 100 meters from release points. Therefore, reducing the distance by 1.5 magnitudes to 80 m increases the intake within the fence line by a magnitude of three over the site boundary intake. Intakes from the reference calculations have been increased by a factor of 10^3 and tabulated in the attachments and should be used for unmonitored environmental intake.

4.3 EXTERNAL DOSE

Workers are subjected to external doses from the ambient radiation levels at the K-25 Site. Environmental data was available only from 1973 through 1985. These data was collected from CaF thermoluminescent dosimeters (TLDs) mounted 1 meter above the ground. The TLDs were read monthly for two K-25 Site perimeter stations: HP-33 (Gallagher Gate) and HP-35 (Blair Gate).

Perimeter monitoring stations are not the best source of data for onsite exposure for unmonitored workers. Unlike PORTS and PGDP, records have not revealed a network of environmental dosimeters along the fence line or within the K-25 site. This may be attributed to the practice of including K-25 in environmental reports for the entire Oak Ridge Reservation (ORR) which also includes the Y-12 and ORNL facilities. K-25 was perceived to have little external exposure impact in comparison with other ORR facilities and the monitoring was focused on the area most affected around ORNL. The lesser impact is indeed the case at K-25, but there is little real world data to base such a conclusion. What remains is the manual surveys and environmental placements at sister facilities to draw from.

4.3.1 Ambient Radiation

Site annual environmental reports were reviewed for data that would be useful in reconstructing ambient radiation levels and airborne radionuclide concentrations. Data in these documents included ambient TLD radiation measurements at perimeter stations. The historical TLD results with uncertainty estimates, where available, were tabulated and are presented in Table 4D-4. An asterisk indicates years with data, other years are estimated from these data.

The ambient radiation levels listed in Table 4D-4 were taken directly from the K-25 Site annual and semiannual environmental reports and include contributions from K-25 Site releases and background radiation. The values in Table 4D-4 are average readings in $\mu\text{R/hr}$ from a month-long exposure period. All of these values include the contributions from natural background radiation and nuclear weapons fallout. Therefore, use of these data to account for potential missed or unmonitored dose should in all cases represent an overestimation of the worker's actual dose.

For years that have no reported data, the average of the two stations used is reported for the 50th percentile. The GSD was computed via the equation provided in Section 4.1.1 of this chapter. One data point (1982, HP35: Blair Gate) was not used for computing the GSD for years with no data, because those data are a significant outlier compared to the other data.

Uranium cylinder storage yards remain the only significant source of external exposure at K-25. Surveys in cylinder yards at the sister plants show dose rates up to 200 mrem/2000 h. For simplification, it can be assumed that the cylinder yard inventory grew linearly over time during the production years.

The ORR Annual Environmental Report for 2003 (DOE 2004) reports a dose from the K-25 cylinder yards to a nearby parking area at 4.75 mrem/125h (75 mrem/2000h). The 2003 report was the first to include neutron exposure, which represents 63% of the exposure. The dose at the parking lot was only reported in two other years, but these did not include the neutron component.

Though this dose rate could not have existed in the early years when DU inventory was just beginning, it should be applied to unmonitored workers for each full year of employment at K-25. This would conservatively assign more dose than can be established from the record.

4.3.2 Other Potential Sources

Other sources of photon exposure have been investigated in recent years (DOE 2000a,b). Both of these documents investigated doses from fission products and found such doses to be insignificant. These reports also indicated that K-25 did investigate processing of ¹³⁷Cs waste but neither report found indications of significant dose to individuals from the investigation.

Current levels of fission products as surface and other contamination were also investigated (DOE 1984). Only ¹³⁷Cs was found in trace amounts in 3 of 96 samples taken in 19 buildings. Such evidence shows that ¹³⁷Cs and other photon emitters need not be considered as sources of exposure.

4.3.3 Review of External Dose

The mean dose rate for the region of K-25 is 7.8 μ R/hr (15.6 mR/2000h) (DOE 2001). The mean dose rate used in the evaluation is 8.6 μ R/h (17.2 mR/2000h). The assumption that the entire recorded environmental dose is attributed to K-25 is very conservative because the difference is 0.8 μ R/hr. The airborne releases in Tables 4D-1 through 4D-3 show that releases were comparable in 1944-1973 to those of 1973-1985. There is no indication based on releases that dose estimates should be different. ORAU (2004) found upper bound dose rates for unreported doses for personnel in buildings was 25 μ R/hr (50 mR/2000h) for the years in question. It is unlikely that environmental doses approached this level outside the buildings, except near the cylinder yards. The use of the mean dose rate for K-25 during 1973-1985 as an estimate for 1944-1973 is conservative because the majority of the estimated dose rate is from natural background. The further extension of the estimate to the 95th percentile increases that conservatism.

Because 12.58 μ R/hr (25.2 mR/2000h) is the 99th percentile of the mean dose rate, which is primarily natural background, the true dose from K-25 operations is more than confidently contained in the estimate.

The recommended environmental external dose is 90 mRem/y. This is based on the sum of the mean dose rate for the region (15.6 mRem) and the dose from DU Cylinder Yards (75 mRem). Other sources of environmental external dose need not be considered.

4.4 UNCERTAINTY

As discussed in the previous sections, estimates of annual intakes employed conservative (i.e., claimant-favorable) assumptions. The intake values in Attachment D should represent a reasonable

upper bound of the actual intakes that could have occurred. In instances where more detailed information is known about a particular individual or job classification, other modifying factors could be employed. For example, if, for a particular job classification, there is reason to believe that the actual breathing rate for the claimant might vary markedly from the average of 2,400 cubic meters per 2,000 hours of exposure, the dose reconstructor should use professional judgment to adjust the estimated intakes as necessary whether the individual is engaged in light or heavy work. The respiration rate in these cases for light work is 1.2 m³/hr and 1.7 m³/hr for heavy work.

To estimate the annual intake in these cases, the product of the fractional annual period for each job-dependent level of work and the corresponding ventilation rate are summed to determine the total ventilation volume for the year in cubic meters. The annual intake is then determined by multiplying the annual ventilation volume by the annual average concentration in Tables 4D-1 through 4D-3 for the year and location of interest.

In general, the annual intake values presented in Tables 4D-1 through 4D-3 reflect adjusted source terms and uncertainty analyses provided in the ORDR estimate (ChemRisk 1999). For purposes of this document, no other attempt has been made to quantify other uncertainties.

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GLOSSARY

background radiation

The radiation received by man from natural and environmental sources including cosmic rays, radiation from the naturally radioactive elements in the environment, and natural concentrations of radionuclides in the body (carbon-14, potassium-40). The usually quoted average individual exposure from background radiation for a person living in the continental United States is 250 to 300 millirem per year.

depleted uranium

On the ORR, depleted uranium consisted mostly of U-238 and usually contains between 0.14 and 0.20% uranium-235 by weight. Natural uranium contains 0.72% uranium-235, while enriched uranium contains greater than 0.72% uranium-235 by weight. For example, depleted uranium is generated as a result of the K-25 gaseous diffusion uranium enrichment and is found in the tailings portion of the process outputs.

enriched uranium

On the ORR, enriched uranium typically contained between 0.95% and 99% uranium-235. Natural uranium contains 0.72% uranium-235, while depleted uranium contains less than 0.72% uranium-235.

enrichment of uranium

A process in which the relative abundance of one of the isotopes of uranium is increased with respect to the others. These processes in the past used the difference in the mass of the isotopes to increase the relative fraction of one isotope over the others. The resultant material is enriched in one particular isotope (usually uranium-235) and depleted in its other isotope (uranium-238).

gaseous diffusion enrichment

A process by which uranium hexafluoride is passed through a series of semipermeable molecular barriers for the purpose of separating the lighter uranium-235 isotope from the heavier, more naturally abundant uranium-238 isotope (see enrichment of uranium).

natural uranium

Natural or "normal" uranium contains 0.72% uranium-235. Contrast with enriched uranium, which contains more than the natural concentration of uranium-235, and depleted uranium, which contains less than 0.72% uranium-235.

percentiles

If a large set of data is arranged from its smallest value to its largest, and this list is divided into 100 classes containing nearly equal numbers of data, then each percentile represents the highest value within that class. Thus 5% of the data are less than or equal to the 5th percentile, and approximately 95% of the data are greater than or equal to the 5th percentile. The median is defined as the 50th percentile, which divides the data (approximately) into halves.

source term

The quantity, chemical and physical form, and time history of contaminants released to the environment from a facility.

uncertainty

The level of confidence in a given estimate based on the quality and quantity of the available data. Inherent uncertainties are generated by a number of sources including uncertainties in measurements, absence of data due to the lack of environmental monitoring, lack of knowledge about some physical processes and operational procedures, and the approximate nature of mathematical models used to predict the transport of released materials.

uranium

A naturally occurring radioactive metal that, in natural ores, has an atomic weight of approximately 238. The two principal natural isotopes are uranium-235 (0.7%) and uranium-238 (99.3% of natural uranium). Natural uranium also includes a very small amount of the daughter uranium-234 by weight. The activity associated with this U-234 is significant as U-235 enrichment increases. Uranium has been used chiefly in nuclear reactors and nuclear explosives.

**ATTACHMENT 4D
OCCUPATIONAL ENVIRONMENTAL DOSE**

TABLE OF CONTENTS

4D.1 Internal Dose from Onsite Concentrations of Radionuclides	16
4D.1.1 Onsite Releases to Air	16
4D.2 External Dose.....	16
4D.2.1 Ambient Radiation	16

LIST OF TABLES

4D-1 Uranium annual intake per year (Bq) and geometric standard deviation.....	17
4D-2 Plutonium annual intake per year (Bq) and geometric standard deviation	18
4D-3 Neptunium and technetium annual intake per year (Bq) and geometric standard deviation.....	19
4D-4 Background gamma radiation levels at two perimeter ORGDP stations	19

ATTACHMENT 4D OCCUPATIONAL ENVIRONMENTAL DOSE

Unmonitored site workers could be exposed to occupational doses internally from the onsite radiation releases to the air and externally from ambient radiation.

4D.1 INTERNAL DOSE FROM ONSITE CONCENTRATIONS OF RADIONUCLIDES

4D.1.1 Onsite Releases to Air

If a worker spent significant time in areas for which radionuclide concentrations were calculated and the worker was not included in the internal dosimetry program, the data in Tables 4D-1 through 4D-3 can be used to estimate unmonitored intakes. These tables list average derived intake (assuming a 2,400-cubic-meter-per-year ventilation rate) at the 50th percentile and the GSD based on the 95th percentile. For plutonium, neptunium, and technetium the GSD was computed from the total uranium. The decision to use these air concentrations to estimate unmonitored intake should be made cautiously because the worker's intake could be significantly overestimated.

4D.2 EXTERNAL DOSE

4D.2.1 Ambient Radiation

The ambient radiation levels in Table 4D-4 are listed in $\mu\text{R/hr}$ for various perimeter locations at the ORGDP. The values denoted with a "*" are derived from empirical Site data. The other values are computed from the empirical Site data and propagated to the other years of Site operations. The empirical values in Table 4D-4 include contributions from natural background radiation and nuclear weapons fallout. Therefore, use of these data to account for potential missed or unmonitored dose should, in all cases, represent an overestimation of the worker's actual dose.

Table 4D-1. Uranium annual intake per year (Bq) and geometric standard deviation.

Date	U-234		U-235		U-238	
	50 th percentile	GSD	50 th percentile	GSD	50 th percentile	GSD
1944	5.57E-02 Bq	1.99	2.65E-03 Bq	1.99	5.76E-02 Bq	1.99
1945	1.69E-01 Bq	1.98	8.03E-03 Bq	1.98	1.74E-01 Bq	1.98
1946	6.34E-02 Bq	1.6	3.05E-03 Bq	1.6	1.58E-02 Bq	1.6
1947	6.99E-02 Bq	1.67	3.16E-03 Bq	1.6	1.57E-02 Bq	1.6
1948	6.75E-02 Bq	1.69	3.06E-03 Bq	1.6	1.64E-02 Bq	1.6
1949	7.44E-02 Bq	1.64	3.36E-03 Bq	1.6	2.66E-02 Bq	1.6
1950	8.50E-02 Bq	1.6	3.89E-03 Bq	1.6	3.46E-02 Bq	1.6
1951	8.49E-02 Bq	1.6	6.64E-03 Bq	1.6	9.84E-02 Bq	1.6
1952	1.74E-01 Bq	1.6	7.76E-03 Bq	1.6	1.69E-01 Bq	1.6
1953	3.09E-01 Bq	1.6	1.56E-02 Bq	1.6	1.81E-01 Bq	1.6
1954	3.36E-01 Bq	1.6	1.39E-02 Bq	1.6	6.24E-02 Bq	1.6
1955	9.83E-02 Bq	1.83	5.49E-03 Bq	1.6	6.70E-02 Bq	1.6
1956	1.30E-01 Bq	1.96	5.84E-03 Bq	1.6	5.49E-02 Bq	1.6
1957	1.43E-01 Bq	1.89	6.44E-03 Bq	1.6	6.13E-02 Bq	1.6
1958	4.82E-01 Bq	1.6	2.31E-02 Bq	1.6	3.78E-01 Bq	1.6
1959	2.64E-01 Bq	1.6	1.14E-02 Bq	1.6	9.31E-02 Bq	1.6
1960	2.01E-01 Bq	1.6	9.65E-03 Bq	1.6	1.66E-01 Bq	1.6
1961	1.82E-01 Bq	1.6	8.08E-03 Bq	1.6	1.25E-01 Bq	1.6
1962	1.16E-01 Bq	1.6	5.39E-03 Bq	1.6	2.22E-02 Bq	1.6
1963	1.93E+00 Bq	1.6	1.02E-01 Bq	1.6	1.25E-01 Bq	1.6
1964	3.58E-02 Bq	2.89	1.59E-03 Bq	1.91	3.07E-03 Bq	1.78
1965	3.00E-01 Bq	1.6	1.59E-02 Bq	1.6	6.17E-02 Bq	1.6
1966	3.40E-02 Bq	2.97	1.51E-03 Bq	1.95	2.49E-03 Bq	1.93
1967	3.38E-02 Bq	2.98	1.50E-03 Bq	1.96	2.41E-03 Bq	1.96
1968	3.39E-02 Bq	2.97	1.50E-03 Bq	1.95	2.66E-03 Bq	1.88
1969	3.53E-02 Bq	2.91	1.57E-03 Bq	1.92	3.81E-03 Bq	1.65
1970	3.56E-02 Bq	2.9	1.60E-03 Bq	1.91	3.24E-03 Bq	1.75
1971	6.40E-02 Bq	2.2	3.15E-03 Bq	1.6	9.34E-03 Bq	1.6
1972	4.43E-02 Bq	2.58	2.06E-03 Bq	1.69	6.57E-03 Bq	1.6
1973	1.37E-01 Bq	1.6	8.56E-03 Bq	1.6	3.87E-02 Bq	1.6
1974	5.24E-01 Bq	1.6	2.98E-02 Bq	1.6	8.28E-02 Bq	1.6
1975	2.64E-01 Bq	1.6	1.55E-02 Bq	1.6	4.97E-02 Bq	1.6
1976	8.37E-02 Bq	1.86	5.33E-03 Bq	1.6	1.56E-02 Bq	1.6
1977	4.61E-02 Bq	2.55	2.18E-03 Bq	1.7	4.79E-03 Bq	1.6
1978	3.84E-02 Bq	2.79	1.75E-03 Bq	1.84	3.68E-03 Bq	1.67
1979	3.72E-02 Bq	2.83	2.73E-03 Bq	1.6	6.06E-03 Bq	1.6
1980	6.29E-02 Bq	2	4.34E-03 Bq	1.6	1.64E-02 Bq	1.6
1981	4.20E-02 Bq	2.62	3.19E-03 Bq	1.6	9.16E-03 Bq	1.6
1982	3.32E-02 Bq	2.98	2.89E-03 Bq	1.6	9.93E-03 Bq	1.6
1983	3.41E-02 Bq	2.96	1.52E-03 Bq	1.95	2.50E-03 Bq	1.93
1984	3.40E-02 Bq	2.97	1.51E-03 Bq	1.95	2.46E-03 Bq	1.94
1985	3.39E-02 Bq	2.97	1.51E-03 Bq	1.95	2.45E-03 Bq	1.94
1986	3.74E-04 Bq	1.6	1.99E-05 Bq	1.6	2.45E-05 Bq	1.6
1987	6.65E-05 Bq	1.6	3.30E-06 Bq	1.6	5.57E-05 Bq	1.6
1988	2.11E-04 Bq	1.6	9.84E-06 Bq	1.6	2.39E-04 Bq	1.6
1989	1.12E-05 Bq	1.6	7.14E-08 Bq	1.6	1.56E-04 Bq	1.6
1990	1.72E-04 Bq	1.6	7.02E-06 Bq	1.6	2.82E-04 Bq	1.6
1991	4.23E-03 Bq	1.6	1.87E-04 Bq	1.6	5.63E-03 Bq	1.6
1992	1.06E-02 Bq	1.6	4.51E-04 Bq	1.6	1.57E-02 Bq	1.6
1993	2.39E-03 Bq	1.6	1.22E-04 Bq	1.6	1.67E-03 Bq	1.6
1994	1.86E-03 Bq	1.6	9.43E-05 Bq	1.6	1.39E-03 Bq	1.6
1995	5.17E-04 Bq	1.6	4.95E-06 Bq	1.6	2.28E-03 Bq	1.6

Table 4D-2. Plutonium annual intake per year (Bq) and geometric standard deviation.

Date	Pu-238		Pu-239		Pu-240		Pu-241		Pu-242	
	50 th percentile	GSD								
1944										
1945										
1946										
1947										
1948										
1949										
1950										
1951										
1952										
1953	3.12E-06 Bq	1.16	2.13E-05 Bq	1.16	5.07E-06 Bq	1.16	1.47E-04 Bq	1.16	7.17E-10 Bq	1.16
1954	1.37E-06 Bq	1.16	9.33E-06 Bq	1.16	2.22E-06 Bq	1.16	6.43E-05 Bq	1.16	3.14E-10 Bq	1.16
1955	1.42E-06 Bq	1.16	9.70E-06 Bq	1.16	2.31E-06 Bq	1.16	6.68E-05 Bq	1.16	3.26E-10 Bq	1.16
1956	6.67E-07 Bq	1.16	4.55E-06 Bq	1.16	1.08E-06 Bq	1.16	3.14E-05 Bq	1.16	1.53E-10 Bq	1.16
1957	6.81E-07 Bq	1.16	4.65E-06 Bq	1.16	1.11E-06 Bq	1.16	3.21E-05 Bq	1.16	1.56E-10 Bq	1.16
1958	3.81E-06 Bq	1.16	2.60E-05 Bq	1.16	6.18E-06 Bq	1.16	1.79E-04 Bq	1.16	8.73E-10 Bq	1.16
1959	1.09E-06 Bq	1.16	7.45E-06 Bq	1.16	1.77E-06 Bq	1.16	5.13E-05 Bq	1.16	2.50E-10 Bq	1.16
1960	2.04E-06 Bq	1.16	1.39E-05 Bq	1.16	3.31E-06 Bq	1.16	9.60E-05 Bq	1.16	4.68E-10 Bq	1.16
1961	1.51E-06 Bq	1.16	1.03E-05 Bq	1.16	2.46E-06 Bq	1.16	7.12E-05 Bq	1.16	3.47E-10 Bq	1.16
1962	3.81E-07 Bq	1.16	2.60E-06 Bq	1.16	6.18E-07 Bq	1.16	1.79E-05 Bq	1.16	8.74E-11 Bq	1.16
1963	2.00E-06 Bq	1.16	1.37E-05 Bq	1.16	3.25E-06 Bq	1.16	9.43E-05 Bq	1.16	4.60E-10 Bq	1.16
1964	9.37E-08 Bq	1.16	6.40E-07 Bq	1.16	1.52E-07 Bq	1.16	4.41E-06 Bq	1.16	2.15E-11 Bq	1.16
1965	5.26E-07 Bq	1.16	3.59E-06 Bq	1.16	8.54E-07 Bq	1.16	2.47E-05 Bq	1.16	1.21E-10 Bq	1.16
1966	6.74E-08 Bq	1.16	4.60E-07 Bq	1.16	1.09E-07 Bq	1.16	3.17E-06 Bq	1.16	1.55E-11 Bq	1.16
1967	6.30E-08 Bq	1.16	4.30E-07 Bq	1.16	1.02E-07 Bq	1.16	2.97E-06 Bq	1.16	1.45E-11 Bq	1.16
1968	8.93E-08 Bq	1.16	6.10E-07 Bq	1.16	1.45E-07 Bq	1.16	4.20E-06 Bq	1.16	2.05E-11 Bq	1.16
1969	1.18E-07 Bq	1.16	8.08E-07 Bq	1.16	1.92E-07 Bq	1.16	5.57E-06 Bq	1.16	2.71E-11 Bq	1.16
1970	9.27E-08 Bq	1.16	6.33E-07 Bq	1.16	1.51E-07 Bq	1.16	4.36E-06 Bq	1.16	2.13E-11 Bq	1.16
1971	1.38E-07 Bq	1.16	9.44E-07 Bq	1.16	2.24E-07 Bq	1.16	6.50E-06 Bq	1.16	3.17E-11 Bq	1.16
1972	1.48E-07 Bq	1.16	1.01E-06 Bq	1.16	2.41E-07 Bq	1.16	6.98E-06 Bq	1.16	3.40E-11 Bq	1.16
1973	2.73E-07 Bq	1.16	1.87E-06 Bq	1.16	4.44E-07 Bq	1.16	1.29E-05 Bq	1.16	6.27E-11 Bq	1.16
1974	5.61E-07 Bq	1.16	3.83E-06 Bq	1.16	9.11E-07 Bq	1.16	2.64E-05 Bq	1.16	1.29E-10 Bq	1.16
1975	1.01E-07 Bq	1.16	6.93E-07 Bq	1.16	1.65E-07 Bq	1.16	4.77E-06 Bq	1.16	2.33E-11 Bq	1.16
1976	9.67E-08 Bq	1.16	6.61E-07 Bq	1.16	1.57E-07 Bq	1.16	4.55E-06 Bq	1.16	2.22E-11 Bq	1.16
1977	5.90E-08 Bq	1.16	4.03E-07 Bq	1.16	9.58E-08 Bq	1.16	2.78E-06 Bq	1.16	1.35E-11 Bq	1.16
1978	5.90E-08 Bq	1.16	4.03E-07 Bq	1.16	9.58E-08 Bq	1.16	2.78E-06 Bq	1.16	1.35E-11 Bq	1.16
1979	5.90E-08 Bq	1.16	4.03E-07 Bq	1.16	9.58E-08 Bq	1.16	2.78E-06 Bq	1.16	1.35E-11 Bq	1.16
1980	5.90E-08 Bq	1.16	4.03E-07 Bq	1.16	9.58E-08 Bq	1.16	2.78E-06 Bq	1.16	1.35E-11 Bq	1.16
1981	5.90E-08 Bq	1.16	4.03E-07 Bq	1.16	9.58E-08 Bq	1.16	2.78E-06 Bq	1.16	1.35E-11 Bq	1.16
1982	5.90E-08 Bq	1.16	4.03E-07 Bq	1.16	9.58E-08 Bq	1.16	2.78E-06 Bq	1.16	1.35E-11 Bq	1.16
1983	5.90E-08 Bq	1.16	4.03E-07 Bq	1.16	9.58E-08 Bq	1.16	2.78E-06 Bq	1.16	1.35E-11 Bq	1.16
1984	5.90E-08 Bq	1.16	4.03E-07 Bq	1.16	9.58E-08 Bq	1.16	2.78E-06 Bq	1.16	1.35E-11 Bq	1.16
1985	5.90E-08 Bq	1.16	4.03E-07 Bq	1.16	9.58E-08 Bq	1.16	2.78E-06 Bq	1.16	1.35E-11 Bq	1.16
1986	5.90E-08 Bq	1.16	4.03E-07 Bq	1.16	9.58E-08 Bq	1.16	2.78E-06 Bq	1.16	1.35E-11 Bq	1.16
1987	5.90E-08 Bq	1.16	4.03E-07 Bq	1.16	9.58E-08 Bq	1.16	2.78E-06 Bq	1.16	1.35E-11 Bq	1.16
1988	5.90E-08 Bq	1.16	4.03E-07 Bq	1.16	9.58E-08 Bq	1.16	2.78E-06 Bq	1.16	1.35E-11 Bq	1.16
1989	5.90E-08 Bq	1.16	4.03E-07 Bq	1.16	9.58E-08 Bq	1.16	2.78E-06 Bq	1.16	1.35E-11 Bq	1.16
1990	5.90E-08 Bq	1.16	4.03E-07 Bq	1.16	9.58E-08 Bq	1.16	2.78E-06 Bq	1.16	1.35E-11 Bq	1.16
1991	5.90E-08 Bq	1.16	4.03E-07 Bq	1.16	9.58E-08 Bq	1.16	2.78E-06 Bq	1.16	1.35E-11 Bq	1.16
1992	5.90E-08 Bq	1.16	4.03E-07 Bq	1.16	9.58E-08 Bq	1.16	2.78E-06 Bq	1.16	1.35E-11 Bq	1.16
1993	5.90E-08 Bq	1.16	4.03E-07 Bq	1.16	9.58E-08 Bq	1.16	2.78E-06 Bq	1.16	1.35E-11 Bq	1.16
1994	5.90E-08 Bq	1.16	4.03E-07 Bq	1.16	9.58E-08 Bq	1.16	2.78E-06 Bq	1.16	1.35E-11 Bq	1.16
1995	5.90E-08 Bq	1.16	4.03E-07 Bq	1.16	9.58E-08 Bq	1.16	2.78E-06 Bq	1.16	1.35E-11 Bq	1.16

Table 4D-3. Neptunium and technetium annual intake per year (Bq) and geometric standard deviation.

Date	Np-237		Tc-99	
	50 th percentile	GSD	50 th percentile	GSD
1944				
1945				
1946				
1947				
1948				
1949				
1950				
1951				
1952				
1953	3.88E-05 Bq	1.16	4.98E+00 Bq	1.16
1954	1.70E-05 Bq	1.16	4.98E+00 Bq	1.16
1955	1.76E-05 Bq	1.16	4.98E+00 Bq	1.16
1956	8.27E-06 Bq	1.16	4.98E+00 Bq	1.16
1957	8.45E-06 Bq	1.16	4.98E+00 Bq	1.16
1958	4.72E-05 Bq	1.16	4.98E+00 Bq	1.16
1959	1.35E-05 Bq	1.16	4.98E+00 Bq	1.16
1960	2.53E-05 Bq	1.16	4.98E+00 Bq	1.16
1961	1.88E-05 Bq	1.16	4.98E+00 Bq	1.16
1962	4.73E-06 Bq	1.16	4.98E+00 Bq	1.16
1963	2.49E-05 Bq	1.16	2.86E+00 Bq	1.16
1964	1.16E-06 Bq	1.16	2.86E+00 Bq	1.16
1965	6.53E-06 Bq	1.16	2.86E+00 Bq	1.16
1966	8.37E-07 Bq	1.16	2.86E+00 Bq	1.16
1967	7.82E-07 Bq	1.16	2.86E+00 Bq	1.16
1968	1.11E-06 Bq	1.16	2.86E+00 Bq	1.16
1969	1.47E-06 Bq	1.16	2.86E+00 Bq	1.16
1970	1.15E-06 Bq	1.16	2.86E+00 Bq	1.16

Date	Np-237		Tc-99	
	50 th percentile	GSD	50 th percentile	GSD
1971	1.72E-06 Bq	1.16	2.86E+00 Bq	1.16
1972	1.84E-06 Bq	1.16	3.11E+00 Bq	1.16
1973	3.39E-06 Bq	1.16	5.16E+00 Bq	1.16
1974	6.96E-06 Bq	1.16	7.14E+00 Bq	1.16
1975	1.26E-06 Bq	1.16	2.36E+00 Bq	1.16
1976	1.20E-06 Bq	1.16	2.84E+00 Bq	1.16
1977	7.32E-07 Bq	1.16	4.18E-07 Bq	1.16
1978	7.32E-07 Bq	1.16	4.18E-07 Bq	1.16
1979	7.32E-07 Bq	1.16	4.18E-07 Bq	1.16
1980	7.32E-07 Bq	1.16	4.18E-07 Bq	1.16
1981	7.32E-07 Bq	1.16	1.67E-02 Bq	1.16
1982	7.32E-07 Bq	1.16	1.05E-02 Bq	1.16
1983	7.32E-07 Bq	1.16	4.18E-07 Bq	1.16
1984	7.32E-07 Bq	1.16	4.18E-07 Bq	1.16
1985	7.32E-07 Bq	1.16	4.18E-07 Bq	1.16
1986	7.32E-07 Bq	1.16	4.18E-07 Bq	1.16
1987	7.32E-07 Bq	1.16	4.18E-07 Bq	1.16
1988	7.32E-07 Bq	1.16	4.18E-07 Bq	1.16
1989	7.32E-07 Bq	1.16	4.18E-07 Bq	1.16
1990	7.32E-07 Bq	1.16	4.18E-07 Bq	1.16
1991	7.32E-07 Bq	1.16	4.18E-07 Bq	1.16
1992	7.32E-07 Bq	1.16	4.18E-07 Bq	1.16
1993	7.32E-07 Bq	1.16	4.18E-07 Bq	1.16
1994	7.32E-07 Bq	1.16	4.18E-07 Bq	1.16
1995	7.32E-07 Bq	1.16	4.18E-07 Bq	1.16

Table 4D-4. Background gamma radiation levels at two perimeter ORGDP stations.

Date	Average	GSD
1944	9.8 uR/hr	1.08
1945	9.8 uR/hr	1.08
1946	9.8 uR/hr	1.08
1947	9.8 uR/hr	1.08
1948	9.8 uR/hr	1.08
1949	9.8 uR/hr	1.08
1950	9.8 uR/hr	1.08
1951	9.8 uR/hr	1.08
1952	9.8 uR/hr	1.08
1953	9.8 uR/hr	1.08
1954	9.8 uR/hr	1.08
1955	9.8 uR/hr	1.08
1956	9.8 uR/hr	1.08
1957	9.8 uR/hr	1.08
1958	9.8 uR/hr	1.08
1959	9.8 uR/hr	1.08
1960	9.8 uR/hr	1.08
1961	9.8 uR/hr	1.08

Date	Average	GSD
1962	9.8 uR/hr	1.08
1963	9.8 uR/hr	1.08
1964	9.8 uR/hr	1.08
1965	9.8 uR/hr	1.08
1966	9.8 uR/hr	1.08
1967	9.8 uR/hr	1.08
1968	9.8 uR/hr	1.08
1969	9.8 uR/hr	1.08
1970	9.8 uR/hr	1.08
1971	9.8 uR/hr	1.08
1972	9.8 uR/hr	1.08
1973	7.4 uR/hr	1.09
1974	7.6 uR/hr	1.09
1975	8.2 uR/hr	1.08
1976	10.5 uR/hr	1.07
1977	7.6 uR/hr	1.09
1978	8.8 uR/hr	1.08
1979	9.6 uR/hr	1.07

Date	Average	GSD
1980	7.7 uR/hr	1.09
1981	7.8 uR/hr	1.09
1982	10.2 uR/hr	1.07
1983	9.9 uR/hr	1.07
1984	8.2 uR/hr	1.08
1985	8.8 uR/hr	1.08
1986	9.8 uR/hr	1.08
1987	9.8 uR/hr	1.08
1988	9.8 uR/hr	1.08
1989	9.8 uR/hr	1.08
1990	9.8 uR/hr	1.08
1991	9.8 uR/hr	1.08
1992	9.8 uR/hr	1.08
1993	9.8 uR/hr	1.08
1994	9.8 uR/hr	1.08
1995	9.8 uR/hr	1.08

* Denotes data derived from empirical site measurements.