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Dose Reconstruction
Project for NIOSH**

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ACRONYMS AND ABBREVIATIONS

a	alpha particle
AED	Aerodynamic Equivalent Diameter
AMAD	Activity Median Aerodynamic Diameter
CDH	Colorado Department of Public Health and Environment (previously Colorado Department of Health)
DF	dose factor
DOE	U.S. Department of Energy
EPA	U.S. Environmental Protection Agency
ET	Extrathoracic
FEMP	Fernald Environmental Management Project
ICRP	International Commission on Radiological Protection
TBD	technical basis document
TLLa	total long-lived alpha
U.S.C	United States Code
WL	Working Level = 1.3×10^5 MeV of alpha energy in 1 liter of air
WLM	Working Level Month = Exposure from 1 WL or radon daughters for 170 working hours
WSCP	Weldon Spring Chemical Plant
WSP	Weldon Spring Plant
WSQ	Weldon Spring Quarry
WSRP	Weldon Spring Raffinate Pits

4.1 INTRODUCTION

4.1.1 Purpose

Technical basis documents (TBDs) 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 Employees Occupational Illness Compensation Program Act of 2000 (42 U.S.C. § 7384l (5) and (12)).

The preparers of this report used available information about the Weldon Spring Plant (WSP) environment dating back to 1957. They strongly advise the reader to consider remarks in the text on the appropriate use and interpretation of the material.

4.1.2 Content

The term *occupational environmental dose* refers to the radiation dose received in the course of work duties outside plant buildings, but on the WSP site. This TBD considers internal and external exposures to radionuclides in the outdoor environment separately in calculating this dose. Dose reconstructors can use estimated occupational environmental dose to develop a reliable individual dose when a worker was not monitored adequately.

The periods for which this TBD evaluates environmental doses are those during which the U.S. Department of Energy (DOE) or its predecessor agencies (the U.S. Atomic Energy Commission until 1975 and the Energy Research and Development Administration until 1977) had contractors on some or all of the WSP. These include the operational period (1957 to 1967) and the monitoring and remediation periods (1975 to present). The monitoring period, from 1975 through 1984, applies only to the raffinate pits and quarry areas because DOE did not assume control of the chemical plant until 1985.

Workers at the Weldon Spring Chemical Plant (WSCP) received environmental doses during the operational period as a result of stack effluents from buildings, contamination of soil resulting from stack releases, and onsite storage of ore concentrates. Workers at the Weldon Spring Quarry (WSQ) received environmental doses during the same period as a result of suspension and resuspension from contaminated rubble and soil being placed in, or already in, the disposal area. During the remediation period, workers received occupational doses due to the presence of contaminated soils and structures at the WSQ, the WSCP, and the Weldon Spring Raffinate Pits (WSRP) as a result of excavation of contaminated sludges and soils, demolition and removal of contaminated structures, and placement of contaminated media in the onsite disposal facility in the WSRP (see Section 2.2.4 of this Site Profile).

Section 4.2 contains information necessary to estimate internal environmental dose. Section 4.2.1 identifies radionuclides of concern. Section 4.2.2 discusses source terms (release rates) for radionuclides potentially significant to internal environmental dose, available air monitoring data, and estimated annual inhalation intake activities of radionuclides.

Section 4.3 contains information necessary for estimating external environmental dose. It reviews and tabulates ambient doses in a format that is usable for dose reconstruction. These doses have been reported for the WSRP, WSCP, and WSQ areas in remedial investigations supporting the Comprehensive Environmental Response and Liability Act process and in site-wide annual reports, and as a result of other radiological surveys.

4.2 INTERNAL DOSE FROM ONSITE ATMOSPHERIC RADIONUCLIDE CONCENTRATIONS

The focus of this section is internal occupational exposure to radionuclides released from WSP facilities during the period of uranium processing from 1957 through 1967 (processing ended in 1966, and shutdown procedures occurred in 1967) and the period of monitoring and remediation activities from 1975 to the present. Internal exposure could have occurred as a result of inhalation of airborne radioactive particulates or radon during these periods of work activity. Between 1967 and 1984, the site was essentially undisturbed; remedial activities began in 1985 under DOE sponsorship.

Section 4.2.1 addresses the identification of radionuclides of concern, defined as those contributing to 95% of the potential dose. Section 4.2.2 describes specific sources of various radionuclides in the outdoor environment. Section 4.2.3 calculates activity intakes of radioactivity, in the form of air particulates and radon, based on the information reviewed. In addition, Section 4.2.3 discusses appropriate assumptions dose reconstructors should make with regard to intake of radioactivity.

4.2.1 Radionuclides of Concern

4.2.1.1 Airborne Particulate Radionuclides

The WSCP processed uranium ($^{235,238}\text{U}$) and thorium (^{232}Th) ore concentrates and some scrap metal during its operational period (see Section 2.2.4 of this Site Profile). The Plant disposed of waste materials associated with uranium and thorium processing at the chemical plant and other facilities in the quarry (see Section 2.2.3.2 of this Site Profile). The radiological risks associated with the ^{235}U decay series are likely to be much lower than those associated with the ^{238}U series because the abundance of ^{235}U in natural uranium ore concentrates is low (an activity ratio of 0.046:1 for ^{235}U to ^{238}U), and WSP processed only a small amount of slightly enriched uranium (see Section 2.2.4 of this Site Profile).

During the operational period (1957–1967), most particulate radioactive contaminants in air associated with WSP facilities would have been isotopes of uranium and thorium, along with short-lived daughters that grew in to a significant extent in the 10-year period. A Weldon Spring historical offsite dose estimate study by Meshkov et al. (1986) assumed that activities of ^{230}Th , ^{226}Ra , and ^{210}Pb in yellowcake were 5%, 1%, and 1%, respectively, of that of ^{238}U .

During the remediation period (1985 to present), other radioactive daughters will have grown into the contaminated media (soil around the WSCP, WSRP, and WSQ) to a potentially significant extent. *Baseline Assessment for the Chemical Plant Area of the Weldon Spring Site* (DOE 1992) identified the following radionuclides for evaluation: ^{238}U , ^{235}U , ^{232}Th , ^{230}Th , ^{231}Pa , ^{227}Ac , ^{228}Ra , ^{226}Ra , and ^{210}Pb , as well as isotopes of radon for separate evaluation. The rationale for selecting these radionuclides was given as follows. First, all radionuclides in the ^{235}U , ^{238}U , and ^{232}Th decay series with half-lives less than 1 year had established secular equilibrium with longer lived parents in the series because operations had ended more than 20 years before the baseline assessment. This means that all these shorter-lived radionuclides could be assumed to have activities equal to that of the parent. Table 4-1 indicates the relationship between the longer lived parents and shorter lived daughters in each decay

series. Second, the activities of ^{234}U and ^{238}U are equal in natural uranium. Third, the half-life of ^{228}Th is very short in relation to the period between plant closure and the baseline assessment; thus this radionuclide could be assumed to be in transient equilibrium with ^{228}Ra .

Table 4-1. Principal and associated radionuclides in ^{238}U , ^{232}Th , and ^{235}U decay series.

Principal radionuclide ^a		Associated decay chain ^b	
Species	Half-life ^c	Species	Half-life ^c
U-238 decay series			
U-238	4.5E+09 yr	Th-234	24 d
		Pa-234	1.2 min
U-234	2.4E+05 yr		
Th-230	7.5E+04 yr		
Ra-226	1.6E+03 yr	Rn-222	3.8 d
		Po-218	3.1 min
		Pb-214	27 min
		Bi-214	20 min
		Po-214	1.6E+03 μs
Pb-210	22 y	Bi-210	5.0 d
		Po-210	1.4E+02 d
Th-232 decay series			
Th-232	1.4E+10 yr		
Ra-228	5.8 yr	Ac-228	6.2 hr
Th-228	1.9 yr	Ra-224	3.7 d
		Rn-220	55 s
		Po-216	0.15 s
		Pb-212	11 h
		Bi-212	61 min
		Po-212 (64%) ^d	0.30 μs
		Tl-208 (36%) ^d	3.1 min
U-235 decay series			
U-235	7.0E+08 yr	Th-231	26 hr
Pa-231	3.3E+04 yr		
Ac-227	22 yr	Th-227 (98.6%) ^d	19 d
		Fr-223 (1.4%) ^d	22 min
		Ra-223	11 d
		Rn-219	4.0 s
		Po-215	1.8 ms
		Pb-211	36 min
		Bi-211	2.1 min
		Tl-207	4.8 min

- Radionuclides with half-lives greater than 1 year.
- The chain of decay products of a principal radionuclide extending to the next principal radionuclide or stable nuclide.
- Half-lives to 2 significant digits from Tuli (2000).
- Branching ratios in parentheses (branching ratios less than 1% not listed in this table).

A simplified screening analysis was conducted for this TBD to rank WSP radionuclides in the uranium and thorium decay series in order of significance to the total dose from all radionuclides as a result of inhalation during the operational and monitoring/remediation periods. Radionuclides selected as potentially significant are the highest potential dose producers, contributing up to 95% of the total dose. The starting list of radionuclides includes the daughters in the uranium and thorium decay

series, listed in Table 4-1 (with the exception of the short-lived daughters of radon isotopes, which are discussed in Section 4.2.1.2).

During the operational period, the airborne activity of uranium daughters ^{230}Th , ^{226}Ra , and ^{210}Pb are assumed to be 5%, 1%, and 1%, respectively, of the ^{238}U activity for this screening exercise. This is consistent with the assumption in Meshkov et al. (1986). In addition to the fact that no discards of ^{232}Th to the air were measured for this isotope during processing, the activity of ^{232}Th processed in any one year was less than 1% of the natural uranium processed (DOE 1986). Thus, ^{232}Th was probably not a significant contributor to inhalation dose during the operational period. To assess the relative significance of the ^{238}U daughters, this TBD analysis calculated inhalation dose per unit ^{238}U activity of each daughter (except ^{222}Rn and its short-lived daughters). A normalized organ dose was calculated by multiplying the normalized airborne activities (1.0 for ^{238}U and ^{234}U , and 0.05 for ^{230}Th , and so on) by the extrathoracic (ET) airways, bone surface, and lung dose factors from International Commission on Radiation Protection CD1 (ICRP 2001). These organs were identified as those of concern based on the fact that, for the list of radionuclides analyzed, the highest dose factors were associated with these organs, considering all possible absorption types and assuming inhalation particle sizes of 5 μm activity median aerodynamic diameter (AMAD). The analysis evaluated the percent contribution of each radionuclide to the summed normalized doses, the results of which are listed in Table 4-2. For the operational period, ^{234}U and ^{238}U are estimated to contribute 97% of the ET airways and lung dose and 90% of the bone surface dose, while ^{230}Th contributes almost the entire remaining dose for these organs.

During the monitoring and remediation period, normalized soil and sludge activity concentrations of the radionuclides estimated for the WSRP, WSCP, and WSQ in the early 1990s (DOE 1992) were used to assess the potential significance of daughter radionuclides. Most of these estimates are based on measurements. Measurements are a better estimate of the relative concentrations because radiological decay calculations ignore the differential leaching of the chemical species in the soil and sludge. Use of these measured values from the 1990s will overestimate the daughter ingrowth for the pre-1990 period. If measured values for particular radionuclides were not available, the estimated concentrations were biased high (DOE 1992, p. 2-12).

The normalized activity concentrations, defined as the measured activity concentration in sludge or soil divided by the measured ^{238}U activity concentration in the same media, are in Table 2.3 of DOE (1992, p. 2-39). Reasonably assuming that normalized concentrations in air will reflect normalized concentrations in soil or sludge due to resuspension processes, this TBD analysis multiplied the value for each radionuclide (Bq per Bq of ^{238}U) by the corresponding ICRP (2001) organ dose factors (Sv/Bq) to obtain normalized doses. For example, for ^{230}Th in the raffinate pits, the analysis assumed that the normalized airborne activity was 25 Bq per Bq of ^{238}U . The corresponding normalized bone surface dose was calculated by multiplying this activity by the ^{230}Th organ dose factor of 1.5×10^{-3} Sv/Bq for absorption type S and 5 μm AMAD inhaled particles, to obtain a normalized bone surface dose of 3.8×10^{-2} Sv/Bq of ^{238}U inhaled. The percent contribution of each radionuclide to the summed normalized dose for each organ and WSP area was calculated. Table 4-2 lists the results. Separate calculations were done for each area because the measured activity concentrations were significantly different for the WSRP, WSCP, and WSQ. The differences are probably due to the different types of wastes deposited in the waste areas and differential leaching mechanisms for the waste areas versus the general WSCP site.

The radionuclides of concern for the operational and monitoring/remediation periods were ranked according to potential significance, based on the percent contributions calculated above. Table 4-3 lists the final ranking for the three areas. The radionuclides listed in **bold** are those contributing to 95% of the dose per unit activity of ^{238}U and are potentially significant. (Radionuclides alone

Table 4-2. Results of screening calculations for ²³⁸U, ²³²Th, and ²³⁵U decay series radionuclides.

OPERATIONAL PERIOD (1957-1967)													
Radionuclide	Estimated airborne activity normalized to U-238 activity (Bq per Bq U-238)			Percent contribution of each radionuclide to normalized dose									
				ET airway			Bone surface			Lungs			
U-238	1.0			4.5E+01			1.0E+01			4.4E+01			
Th-234	1.0			4.5E-03			3.2E-03			4.8E-02			
Pa-234	1.0			5.3E-03			5.8E-05			1.9E-03			
U-234	1.0			5.2E+01			1.1E+01			5.3E+01			
Th-230	0.05			2.6E+00			7.8E+01			2.6E+00			
Ra-226	0.01			9.0E-02			9.3E-02			2.2E-01			
Pb-210	0.01			9.7E-04			3.7E-01			1.8E-03			
Bi-210	0.01			1.8E-03			1.1E-06			6.3E-03			
Po-210	0.01			5.1E-02			9.7E-03			2.2E-01			
MONITORING/REMEDIAION PERIOD (1975-PRESENT)													
Radionuclide	Estimated airborne activity normalized to U-238 activity (Bq per Bq U-238) ^a			Percent contribution of each radionuclide to normalized dose									
				WSRP			WSCP			WSQ			
				ET airway	Bone surface	Lungs	ET Airway	Bone surface	Lungs	ET airway	Bone surface	Lungs	
U-238 decay series													
U-238	1.0	1.00	1.00	2.7E+00	1.9E-02	2.4E+00	3.3E+01	1.1E+00	2.8E+01	1.6E+01	3.3E-01	1.1E+01	
Th-234	1.0	1.00	1.00	2.7E-04	6.0E-06	2.6E-03	3.3E-03	3.4E-04	3.0E-02	1.6E-03	1.0E-04	1.2E-02	
Pa-234	1.0	1.00	1.00	3.2E-04	1.1E-07	1.0E-04	3.9E-03	6.1E-06	1.2E-03	1.8E-03	1.8E-06	4.9E-04	
U-234	1.0	1.00	1.00	3.2E+00	2.1E-02	2.9E+00	3.9E+01	1.2E+00	3.4E+01	1.8E+01	3.6E-01	1.3E+01	
Th-230	25.0	0.15	1.70	7.8E+01	7.2E+01	7.0E+01	5.7E+00	2.4E+01	4.9E+00	3.0E+01	8.3E+01	2.2E+01	
Ra-226	0.62	0.14	0.55	3.4E-01	1.1E-02	7.4E-01	9.4E-01	1.4E-01	2.0E+00	1.7E+00	1.6E-01	3.0E+00	
Pb-210	0.62	0.14	1.70	3.6E-03	4.3E-02	6.1E-03	1.0E-02	5.5E-01	1.6E-02	5.8E-02	2.0E+00	7.7E-02	
Bi-210	0.62	0.14	1.70	6.8E-03	1.3E-07	2.1E-02	1.9E-02	1.7E-06	5.7E-02	1.1E-01	6.1E-06	2.7E-01	
Po-210	0.62	0.14	1.70	1.9E-01	1.1E-03	7.4E-01	5.3E-01	1.4E-02	2.0E+00	3.0E+00	5.2E-02	9.4E+00	
Th-232 decay series													
Th-232	0.026	0.076	0.13	1.2E-01	7.5E-02	1.4E-01	4.3E+00	1.2E+01	4.8E+00	3.5E+00	6.4E+00	3.2E+00	
Ra-228	0.25	0.076	0.48	9.9E-02	1.7E-02	1.1E-01	3.7E-01	3.0E-01	3.9E-01	1.1E+00	5.6E-01	9.6E-01	
Ac-228	0.25	0.076	0.48	9.0E-04	4.0E-04	1.4E-03	3.4E-03	6.9E-03	5.0E-03	1.0E-02	1.3E-02	1.2E-02	
Th-228	0.25	0.076	0.48	2.2E+00	1.3E-01	3.7E+00	8.2E+00	2.3E+00	1.3E+01	2.4E+01	4.4E+00	3.3E+01	
Ra-224	0.25	0.076	0.48	6.3E-02	4.6E-04	3.5E-01	2.3E-01	7.9E-03	1.3E+00	7.0E-01	1.5E-02	3.1E+00	
U-235 decay series													
U-235	0.046	0.046	0.046	1.3E-01	9.7E-04	1.2E-01	1.6E+00	5.5E-02	1.4E+00	7.7E-01	1.6E-02	5.4E-01	
Th-231	0.046	0.046	0.046	4.6E-06	3.3E-08	5.5E-06	5.7E-05	1.9E-06	6.4E-05	2.7E-05	5.5E-07	2.5E-05	
Pa-231	0.73	0.038	0.0058	3.4E+00	6.2E+00	3.7E+00	2.1E+00	1.8E+01	2.3E+00	1.5E-01	8.3E-01	1.4E-01	
Ac-227	0.58	0.019	0.0029	9.3E+00	2.1E+01	1.1E+01	3.7E+00	3.9E+01	4.2E+00	2.7E-01	1.8E+00	2.5E-01	
Th-227	0.58	0.019	0.0029	3.2E-01	4.0E-03	2.6E+00	1.3E-01	7.4E-03	1.0E+00	9.1E-03	3.4E-04	6.0E-02	
Fr-223	0.58	0.019	0.0029	2.3E-04	1.2E-06	4.9E-05	9.3E-05	2.3E-06	1.9E-05	6.7E-06	1.0E-07	1.1E-06	
Ra-223	0.58	0.019	0.0029	2.9E-01	1.9E-03	1.9E+00	1.2E-01	3.5E-03	7.4E-01	8.4E-03	1.6E-04	4.4E-02	

a. Baseline assessment for chemical plant area – when measured values were not available, estimates are biased high according to DOE (1992, p. 2-12).

contributing less than 5% of the dose to any organ were also eliminated from the list of radionuclides of potential concern) Thus, these results indicate that ²³⁰Th, ²³⁴U, and ²³⁸U are the radionuclides of potential significance for the operational period, and that ²²⁷Ac, ²³¹Pa, ²¹⁰Po, ²²⁸Th, ²³⁰Th, ²³²Th, ²³⁴U, and ²³⁸U are the radionuclides of potential significance for the monitoring and remediation periods.

In addition to the radioactive isotopes of the uranium and thorium decay series, dose reconstructors must consider the potential presence of other radionuclides associated with recycled uranium. Between 1962 and 1966, the WSP might have received some recycled uranium from the Fernald site in southwest Ohio, such that trace amounts of residual transuranic elements (including neptunium and plutonium isotopes), fission products (such as technetium), and reactor-produced uranium isotopes (²³⁶U) might have been present. However, more than 97% of the uranium received by WSP is believed to have been natural uranium (DOE 2000, p. C-19). Upper bound estimates on releases of

Table 4-3. Screening results for ^{238}U , ^{232}Th , and ^{235}U decay series.

Radionuclide ^a	Highest rank in organ dose contribution ^a		
	Raffinate pits ^b	Chemical plant ^c	Quarry
U-238 decay series			
U-238	5 (ET)	2 (2) (ET,L)	4 (ET,L)
Th-234	17 (BS,L)	16 (6) (L)	17 (BS)
Pa-234	18 (ET)	17 (6) (ET)	18 (ET)
U-234	4 ((ET)	1 (1) (ET,L)	3 (ET,L)
Th-230	1 (ET, BS, L)	2 (1) (BS)	1 (BS)
Ra-226	7 (ET)	8 (4) (ET,L)	7 (ET)
Pb-210	6 (BS)	8 (4) (BS)	4 (BS)
Bi-210	15 ((ET,L)	17 (8) (ET,L)	14 (ET,L)
Po-210	9 (L)	8 (5) (L)	5 (L)
Th-232 decay series			
Th-232	5 (BS)	4 (BS)	2 (BS)
Ra-228	9 (BS)	9 (BS)	7 (BS)
Ac-228	16 (BS)	15 (BS)	14 (BS)
Th-228	4 (BS,L)	3 (ET)	1 (L)
Ra-224	11 (L)	11 (L)	7 (L)
U-235 decay series			
U-235	11 (ET)	8 (ET)	9 (ET)
Th-231	21 (ET,BS,L)	20 (BS,L)	20 (ET,BS,L)
Pa-231	3 (ET,BS,L)	3 (BS)	6 (BS)
Ac-227	2 (ET,BS,L)	1 (BS)	5 (BS)
Th-227	6 (L)	12 (L)	15 (BS,L)
Fr-223	18 (BS)	19 (BS)	21 (ET,BS,L)
Ra-223	8 (L)	15 (L)	16 (BS,L)

a. Radionuclides and rank numbers in bold identify those that are the top contributors to 95% of the dose.

b. ET= extrathoracic airways, BS = bone surface, L = lungs.

c. Numbers in parentheses represent ranking for operational period

plutonium isotopes, neptunium, and technetium from WSP are 0.0, 3.0, and 1.2 g per 45 MTU released from stacks, respectively (based on DOE 2000, Table ES-7D, p. ES-25), between 1962 and 1966.

To evaluate the potential impact on dose of contaminants of recycled uranium (^{239}Pu , ^{240}Pu , ^{237}Np , ^{99}Tc), the maximum estimated total discharge of these contaminants to air for all years in which recycled uranium might have been received (1962 to 1967, p. ES-25 in DOE 2000) is assumed to be representative. These discharge values are divided by the estimated total discharge of ^{238}U via stacks, from DOE (1986) and summarized in DOE (2000), to obtain a normalized release of each contaminant to the air (Bq per Bq ^{238}U).

Assuming that normalized concentrations of recycled uranium contaminants in air will reflect normalized releases from the stacks due to atmospheric dispersion processes, the TBD analysis multiplied these releases by the corresponding ICRP (2001) organ dose factor (Sv/Bq) for each radionuclide to obtain a normalized dose, with units of Sv per Bq ^{238}U . The organs of most concern were again identified as ET airways (uranium isotopes), bone surface (plutonium isotopes and ^{237}Np), and lungs (^{99}Tc). The percent contribution of each radionuclide to the summed normalized doses for each organ was calculated. These estimates apply to all WSP areas, as information did not allow a more detailed analysis by location. Table 4-4 lists the results of these calculations for contaminants of recycled uranium. The results indicate that ^{238}U contributes approximately 99% of the dose per unit

activity, if the contaminants were present to the extent estimated. Thus, this TBD does not consider the contaminants of recycled uranium to be potentially significant contributors to dose.

Table 4-4. Screening results for recycled uranium contaminants.

Radionuclide	Maximum estimated activity released to air, normalized to U-238 ^a (Bq per Bq of U-238)	Percent contribution to normalized dose			Radionuclides contribute to 95% of total dose?
		ET airways	Bone surface	Lungs	
Pu-239	0.0	0.0E+00	0.0E+00	0.00E+00	No
Pu-240	0.0	0.0E+00	0.0E+00	0.00E+00	No
Np-237	1.40E-04	2.8E-03	9.3E-01	7.0E-03	No
Tc-99	1.36E-03	1.7E-05	4.6E-07	9.6E-05	No
U-238	1.00E+00	1.0E+02	9.9E+01	1.0E+02	Yes

a. Based on DOE (2000).

To summarize, the most significant radionuclides identified as contributing to 95% (but no less than 5% each) of the total inhalation dose are listed in **bold** in Table 4-3. The list includes ²²⁷Ac, ²³¹Pa, ²¹⁰Po, ²²⁸Th, ²³⁰Th, ²³²Th, ²³⁴U, and ²³⁸U. Section 4.2.1.2 discusses radon and its short-lived daughter isotopes.

4.2.1.2 Radon

Isotopes of radon associated with the ²³⁵U, ²³⁸U, and ²³²Th decay series can pose a significant dose to workers depending on the amount of precursor (radium) in the soil and the length of exposure. The dose resulting from inhalation of radon is largely from the short-lived, alpha-particle-emitting radon daughters rather than from the gas itself. Three radon isotopes are generated during decay of ²³⁵U, ²³⁸U, and ²³²Th (see Table 4-1) – ²¹⁹Rn, ²²²Rn, and ²²⁰Rn, respectively. The risks associated with ²¹⁹Rn due to its extremely short half-life (4 sec) and small amount of precursor radium (due to the limited amount of ²³⁵U processed at WSP) were insignificant in comparison to those from ²²²Rn and ²²⁰Rn, and are not relevant. Estimates made in 1990 of potential exposure to ²²²Rn and ²²⁰Rn at the WSQ (DOE 1990) suggested the whole-body effective dose rates resulting from inhalation of these isotopes might be slightly less than 1 mrem/hr (at the time and place the measurement occurred). Thus, this TBD considers the inhalation intakes for these two isotopes to be potentially significant, as discussed in more detail below.

4.2.2 Source Terms for Airborne Radionuclides

Section 4.2.2.1 describes sources of particulate radionuclides and gaseous radon to air in the WSP environment. In addition to emissions of ²³⁸U, ²³⁴U and ²³²Th during ore processing, decay products of these three isotopes were measured during the operational and the monitoring/remediation periods.

4.2.2.1 Pre-1968 Source Terms (Operational Period)

During the operational period (1957-1967), drums of uranium concentrate (yellowcake) were emptied into the hopper at the top of the receiving and sampling building (Meshkov et al. 1986). Dust was collected in three large collectors, each with its own stack, and either repackaged and shipped off the site or reprocessed. In the refinery plant process, uranium-laden dust was generated during material transfers, denitration, reduction and hydrofluorination, and conversion to solid metal. Impurities generated in the purification process prior to denitration were bled off in liquid form in the raffinate, which was pumped to the pits. The denitration process was the dustiest part of the operation. Thus, point sources of uranium emissions from stacks existed during this period.

According to the review of source emissions by Meshkov et al. (1986), only a fraction of radon (about 20%) is released when tied to solid particles, such as those that occur in uranium concentrates.

However, the digestion phase of the refinery process would have released the trapped radon during the operational period. Offgases containing radon isotopes from this process were conveyed to the acid recovery plant at the WSCP. The discharge from the acid recovery plant was the prime source of radon emission and perhaps the only point source for radon isotopes.

Wastes disposed of in the raffinate pits were not a significant source of airborne radionuclides, particulate or gaseous, during the operational period due to the presence of water in the pits. It is unlikely that a significant portion of the sediments became sufficiently dry to be considered a source term during this period when they were being managed, even though Bechtel (1984a) reported that Pits 1 and 2 could become dry during the summer months. NLO (1977) concluded that the inherent consistency of the raffinate material precluded sufficient drying, and the pits did not pose a significant source of airborne contaminants.

Wastes disposed of in the quarry during WSP operations present a diffuse source of particulate emissions to the air as a result of entrainment of contaminated wind-exposed surface soil or rubble. Radon-222 and ^{220}Rn generated from ^{226}Ra and ^{224}Ra , respectively, in the quarry also present a diffuse source of airborne radionuclides. Although most of the material initially dumped in the quarry in 1959 was at an elevation below the average natural water level of 457 feet above mean sea level (MSL) reported in 1967, later disposals maintained a fill elevation of about 480 feet MSL, such that most of the waste was not submerged (*Weldon Spring Raffinate Pits and Quarry Task Force Report 1967*).

Uranium and thorium effluents from the various stacks were not directly measured. The total amount of uranium emitted to the atmosphere has been estimated from monitoring data (Meshkov et al. 1986) and from a materials balance study (DOE 1986). The reported atmospheric discharges from the materials balance study were engineering estimates derived from airflows and other process factors along with stack monitoring data, and are summarized in Table 2-8 in Part 2 of this Site Profile. Based on these two methods, the estimated amount of uranium activity emitted from the plant ranges between approximately 1 and 5 Ci/yr. Meshkov et al. estimated release rates for uranium decay products (^{230}Th , ^{226}Ra , ^{210}Pb) as a fixed percent of the estimated amount of ^{238}U released, assuming the activities of ^{230}Th , ^{226}Ra , and ^{210}Pb in the original uranium concentrate were 5%, 1%, and 1%, respectively, of that of ^{238}U . An estimate of radon release based on the amount of uranium processed during the operational period ranged from 12 to 34 Ci/yr (assuming radium activity was 1% of the uranium activity, radon was in equilibrium with radium, and all radon was released).

4.2.2.2 Post-1974 Source Terms (Monitoring and Remediation Periods)

During the monitoring and remediation periods, from 1975 to the present, diffuse emissions predominate as the source of radionuclides to the air at the WSCP, WSRP, and WSQ. Beginning in 1981, annual environmental monitoring reports have provided estimates of air concentrations of particulate radionuclides and of radon at the WSCP and WSQ. These estimates reflect emissions during the later monitoring period and throughout remediation activities.

4.2.3 Annual Intake of Radionuclides

As noted in Section 4.2.2, releases of radioactive particulates or radon to the atmosphere have not been fully quantified at the WSP for the operational period or the monitoring and remediation periods. Meshkov et al. (1986) attempted to quantify point source emissions for the operational period by considering monitoring data and the results of a materials balance study. However, in considering exposure to radionuclides, material released from stacks provides only a partial picture of how radioactivity becomes airborne. Diffuse sources resulting from wind entrainment or other mechanical

disturbance of radionuclides deposited on the ground or in the waste pits or quarry dominate exposures during the remediation period, and contribute to exposures during the operational period. In light of the importance of diffuse sources and the lack of information available for quantifying source strengths throughout the site, estimated occupational intakes for the WSP rely solely on available air monitoring data for this TBD.

Descriptions of particle size and solubility class are not available for further characterizing the information presented below for air particulates. Dose reconstructors should assume that the activity intake of air particulates is 100% respirable, and the default aerosol size of 5- μm AMAD. In addition, dose reconstructors should base the selection of dose factors on the solubility/clearance class that yields the highest dose to the organ of interest. Calculated annual intakes are based on a 2,400- m^3/yr inhalation rate (ICRP 1994), corresponding to an hourly rate of 1.2 m^3 , for light activity and a 2,000-hr work year. Intakes can be scaled to a different rate or to consider partial-year exposures.

4.2.3.1 Pre-1968 Intakes (Operational Period)

Meshkov et al. (1986) reviewed air monitoring data for airborne particulates in the operational period. This TBD analysis used these same data and an onsite measurement made in 1960 (Holt 1960) to calculate estimated intakes of radioactive airborne particulates according to the methods described below under Airborne Particulate Radionuclides. For radon released during yellowcake processing, Meshkov et al. estimated an annual release rate based on the average amount of uranium concentrate processed annually, which this analysis used to estimate maximum concentrations in the WSCP (see below under Radon). The analysis based estimates of air concentrations of radon released from materials deposited in the WSQ during the operational period on measurements made between 1979 and 1982. The precursors of radon (^{226}Ra and ^{224}Ra) were not likely to have been significantly depleted during the time period between the end of operations and 1979, since this element readily sorbs to soil and is not very mobile (ATSDR 1990), and these radium isotopes are gradually being replenished from the ^{238}U and ^{232}Th decay series. Although temporal fluctuations in radon air concentrations have been noted (Meshkov et al. 1986, MK-Ferguson 1990a, MK-Ferguson 1992a), likely due to climatic variations, there were no obvious downward trends in concentrations at the WSQ until the bulk wastes were removed in the 1990s. (Releases from the WSRP were negligible during the operational period; see Section 4.2.2.1).

Airborne Particulate Radionuclides

Meshkov et al. (1986) provides summarized results of uranium air concentration measurements made at the WSP perimeter and at two offsite locations from 1959 through 1965. The perimeter data used here (Table 4-5) have been converted to units of Bq/m^3 . The original data are reported in units of special uranium microcuries, defined as the sum of 3.7×10^4 dps from ^{238}U , 3.7×10^4 dps from ^{234}U , and 9×10^2 dps from ^{235}U . Converting to units of Bq, the *special uranium* μCi is multiplied by 3.7×10^4 $\text{Bq}/\mu\text{Ci}$ and by a factor of 2.024 to report total uranium activity. Other radionuclides were not measured. As indicated in Section 4.2.1.1, only isotopes of ^{238}U , ^{234}U , and ^{230}Th (and their very short-lived daughters) were significant airborne particulate radionuclides during the operational period. Intakes of ^{230}Th are estimated from measured airborne uranium concentrations in the manner described below.

Although many dust-generating operations occurred inside WSP sampling and refinery buildings, Holt (1960) indicated that decontamination of portable hoppers occurred outside Building 103. According to Holt, general air samples taken downwind during the cleaning of six 5-ton hoppers ranged from 90 to 443 dpm/m^3 . This corresponds to between 1.5 and 7.4 Bq/m^3 of total uranium (assuming the measurement was predominantly of uranium isotopes from the dusty operation), or an average of 4.4 Bq/m^3 total uranium activity concentration.

Table 4-5. Average airborne particulate and radon concentrations at WSCP, WSRP, and WSQ, based on monitoring data.

Year	WSRP		WSCP			WSQ	
	Uranium ^a (Bq/m ³)	Radon ^b (Bq/m ³)	Uranium ^a (Bq/m ³)		Radon ^b (Bq/m ³)	Uranium ^a (Bq/m ³)	Radon ^b (Bq/m ³)
			Perimeter	Hopper cleanout			
Operational period							
1957	(c)	(c)	1.7E-02	4.4	9.1E+01	(c)	(c)
1958	(c)	(c)	1.7E-02	4.4	9.1E+01	(c)	(c)
1959	(c)	(c)	1.7E-02	4.4	9.1E+01	(c)	(c)
1960	(c)	(c)	1.2E-02	4.4	9.1E+01	(c)	(c)
1961	(c)	(c)	1.6E-02	4.4	9.1E+01	(c)	(c)
1962	(c)	(c)	5.3E-03	4.4	9.1E+01	(c)	(c)
1963	(c)	(c)	8.4E-03	4.4	9.1E+01	2.6E-04	2.6E+01
1964	(c)	(c)	1.1E-02	4.4	9.1E+01	1.5E-03	2.6E+01
1965	(c)	(c)	3.7E-03	4.4	9.1E+01	1.2E-02	2.6E+01
1966	(c)	(c)	3.7E-03	4.4	9.1E+01	1.2E-02	2.6E+01
1967	(c)	(c)	3.7E-03	4.4	9.1E+01	1.2E-02	2.6E+01
Monitoring/remediation period							
	Gross alpha (Bq/m ³)	Radon ^b (Bq/m ³)	Gross alpha (Bq/m ³)		Radon ^b (Bq/m ³)	Gross alpha (Bq/m ³)	Radon ^b (Bq/m ³)
1975-1979	(c)	4.8E+01 ^e	(f)		(f)	(c)	3.3E+01 ^e
1980	(c)	1.2E+02	(f)		(f)	(c)	2.5E+01
1981	(c)	7.4E+01	(f)		(f)	(c)	2.6E+01
1982	(c)	5.9E+01	(f)		(f)	(c)	2.6E+01
1983	(c)	2.1E+01	(f)		(f)	(c)	2.2E+01
1984	(c)	3.6E+01	(f)		(f)	(c)	2.5E+01
1985	(c)	2.4E+01	(c)		1.5E+01	(c)	2.5E+01
1986	0 ^g	3.4E+01	0 ^g		2.3E+01	(c)	2.4E+01
1987	0 ^g	2.8E+01	0 ^g		3.0E+01	(c)	4.5E+01
1988	0 ^g	4.4E+01	0 ^g		4.1E+01	0 ^g	6.8E+01
1989	2.0E-04	3.3E+01	1.6E-04		3.8E+01	1.9E-05	4.3E+01
1990	1.0E-04	2.0E+01	9.1E-05		2.4E+01	1.7E-05	3.5E+01
1991	5.0E-05	2.2E+01	5.5E-05		2.0E+01	3.3E-06	3.0E+01
1992	3.9E-05	2.2E+01	4.1E-05		1.6E+01	0 ^d	2.7E+01
1993	4.6E-05	1.0E+01	4.6E-05		7.4E+00	3.1E-06	6.1E+01
1994	5.4E-05	2.8E+01	6.3E-05		1.7E+01	2.4E-05	1.7E+02
1995	4.7E-05	3.4E+01	5.1E-05		2.2E+01	8.0E-06	5.8E+01
1996	6.2E-05	5.7E+01	5.9E-05		2.7E+01	3.3E-06	1.3E+01
1997	7.5E-05	1.3E+02	5.5E-05		2.2E+01	0 ^g	1.1E+01
1998	8.7E-05	2.3E+01	7.7E-05		1.3E+01	2.6E-06	7.4E+00
1999	7.7E-05	3.7E+01	7.6E-05		2.5E+01	3.7E-07	1.7E+01
2000	4.1E-05	2.0E+01	5.1E-05		2.6E+01	3.3E-06	1.3E+01
2001	4.1E-05 ^h	2.0E+01 ^h	5.1E-05 ^h		2.6E+01 ^h	3.3E-06 ^h	1.3E+01 ^h
2002	4.1E-05 ^h	2.0E+01 ^h	5.1E-05 ^h		2.6E+01 ^h	3.3E-06 ^h	1.3E+01 ^h
2003	4.1E-05 ^h	2.0E+01 ^h	5.1E-05 ^h		2.6E+01 ^h	3.3E-06 ^h	1.3E+01 ^h
2004	4.1E-05 ^h	2.0E+01 ^h	5.1E-05 ^h		2.6E+01 ^h	3.3E-06 ^h	1.3E+01 ^h

a. The measured concentrations of uranium during the operational period were expressed as a special uranium $\mu\text{Ci}/\text{m}^3$, which is defined as the sum of 3.7×10^4 dps U-238, 3.7×10^4 dps U-234, and 9×10^2 dps U-235 per m^3 . These were converted to Bq/m^3 of total uranium, by multiplying the special uranium $\mu\text{Ci}/\text{m}^3$ by 3.7×10^4 $\text{Bq}/\mu\text{Ci}$ and by the factor 2.024 to account for activity contributions by all isotopes.

b. Radon refers to both ^{222}Rn and ^{220}Rn , and includes natural background contributions.

c. Blank indicates there is no significant air concentration expected for the corresponding years.

d. No data reported in 1966 and 1967; assumed equal to the value in 1965.

e. Assumed equal to average of following 10 years prior to remediation activity disturbances.

f. WSCP was not transferred to DOE until 1985.

g. Value of zero indicates the measured value was less than or equal to the background value.

h. Assumed equal to the value in the year 2000, by which time active remediation was essentially complete

Table 4-5 lists estimated average perimeter air concentrations, and onsite air concentrations during hopper cleanout, of uranium during the operational period for the WSCP and WSQ. These values were obtained from the measured uranium concentrations described above. The average offsite

background value of $7.4E-04 \text{ Bq/m}^3$ (0.02 pCi/m^3 , Meshkov et al. 1986) was subtracted from the values. For 1957 and 1958 at the WSCP, for which measurements are not available, the analysis assumed that perimeter air concentrations were the same as those measured in 1959, the operational year with the highest measured perimeter concentrations. This is probably an overestimate for the first 2 years, when uranium receipts were lower than those during the main production years (1960-1964; Table 2-4 in Part 2 of this Site Profile). Measured concentrations are not available for the WSQ until 1961; until then, only drummed thorium wastes were reportedly stored at the quarry (Section 2.2.3.2 in Part 2 of this Site Profile). Most of the material was dumped at an elevation below the present natural water table of the quarry (*Weldon Spring Raffinate Pits and Quarry Task Force Report* 1967). In 1963 and 1964, an estimated $38,000 \text{ m}^3$ of contaminated rubble, equipment, and soil were placed in the WSQ, and much of this waste was not submerged. Thus, airborne concentrations of uranium and thorium resulting from use of the WSQ were probably negligible before 1963; this observation is supported by the measured concentrations of uranium in air in 1961 and 1962.

The uranium concentrations (Bq/m^3) in Table 4-5 were converted to isotope-specific intakes, in Bq/yr , by assuming the ratio of 1:1:0.05 for ^{238}U : ^{234}U : ^{230}Th activities and multiplying by an appropriate inhalation rate. For perimeter concentrations, intakes were calculated by multiplying by an inhalation rate of $2400 \text{ m}^3/\text{yr}$, representing an assumption of continuous exposure over the work year. To address intakes associated with the localized contaminated dust arising from the reported hopper decontamination operation outside Building 103 (noted above), the analysis assumed that the operation took place for 1 hour per workday (12.5% of the work year) every year during the operational period, and that unmonitored individuals were at a location in which an average concentration of 4.4 Bq/m^3 occurred for 5% of their work year (approximately 24 minutes per day). This corresponds to a yearly inhalation rate of $120 \text{ m}^3/\text{yr}$. The resulting average intakes are presented in Table 4-6.

In Table 4-6, the " ^{238}U and ^{234}U " columns for the operational period represent the sum of ^{238}U and ^{234}U intakes. Since natural uranium accounted for more than 97% of the nuclear materials throughput (Sect. 2.2.2.2 of this SPD), it is reasonable to assume that the activities of these two isotopes are equal in the measured concentrations, as they are approximately equal in natural uranium ore concentrates. Thorium-230 intake is assumed to be 5% of uranium isotope (in this case, either ^{238}U or ^{234}U) intake, consistent with the assumptions by Meshkov et al. (1986), which are based on estimates for yellowcake composition. During the 10-year operational period, the amount of ^{230}Th likely to be present on contaminated surfaces and in air is more reflective of the amount present in the original material being processed than of the daughter ingrowth due to decay of ^{234}U . This assumption is applicable to post-1962 intakes for the WSQ, although the ratio is less certain for waste deposited in the quarry, which has not been well-characterized radiologically.

Although the intake values in Table 4-6 for the operational period can be attributed largely to the one measured value of uranium downwind of the hopper cleanout operation in 1966, it is notable that the average value of 555 Bq/yr for total uranium intake (corresponding to 277 Bq/yr for ^{234}U intake) compares closely with the average estimated intake of 366 Bq/yr for ^{234}U at the Feed Materials Plant in Fernald, Ohio (ORAU 2004), for the same years of operation. The Fernald plant received yellowcake in similar quantities to those received by the WSP between 1957 and 1966, and used similar operations in processing the receipts for ultimate shipment. Table 4-7 lists site-wide maximum intakes, representing the maximum intake each year for the areas listed in Table 4-6.

Radon

Measured air concentrations of radon during the operational period are not reported in the literature available for this TBD. Therefore, the analysis used a very simplistic screening-level model for estimating air concentrations at the WSCP, the location of the acid recovery plant, to estimate air

Table 4-6. Estimated average annual inhalation intake of radioactive air particulates and radon at WSCP, WSRP, and WSQ (assume respirable fraction of sampled particles = 1.0, and based on 2,400 m³/yr inhalation rate).

Year	WSRP			WSCP			WSQ		
	U-238 and U-234 (Bq/yr)	Th-230 (Bq/yr)	Radon (WLM/yr)	U-238 and U-234 ^a (Bq/yr)	Th-230 (Bq/yr)	Radon (WLM/yr)	U-238 and U-234 ^a (Bq/yr)	Th-230 (Bq/yr)	Radon (WLM/yr)
Operational period									
1957	Not significant	Not significant	Not significant	5.7E+02	1.4E+01	8.7E-02	Not significant	Not significant	Not significant
1958	Not significant	Not significant	Not significant	5.7E+02	1.4E+01	8.7E-02	Not significant	Not significant	Not significant
1959	Not significant	Not significant	Not significant	5.7E+02	1.4E+01	8.7E-02	Not significant	Not significant	Not significant
1960	Not significant	Not significant	Not significant	5.6E+02	1.4E+01	8.7E-02	Not significant	Not significant	Not significant
1961	Not significant	Not significant	Not significant	5.7E+02	1.4E+01	8.7E-02	Not significant	Not significant	Not significant
1962	Not significant	Not significant	Not significant	5.4E+02	1.3E+01	8.7E-02	Not significant	Not significant	Not significant
1963	Not significant	Not significant	Not significant	5.5E+02	1.4E+01	8.7E-02	6.2E-01	3.1E-02	2.5E-02
1964	Not significant	Not significant	Not significant	5.5E+02	1.4E+01	8.7E-02	3.6E+00	1.8E-01	2.5E-02
1965	Not significant	Not significant	Not significant	5.4E+02	1.3E+01	8.7E-02	2.9E+01	1.5E+00	2.5E-02
1966	Not significant	Not significant	Not significant	5.4E+02	1.3E+01	8.7E-02	2.9E+01	1.5E+00	2.5E-02
1967	Not significant	Not significant	Not significant	5.4E+02	1.3E+01	8.7E-02	2.9E+01	1.5E+00	2.5E-02

Monitoring/remediation period						
Year	WSRP		WSCP		WSQ	
	Gross alpha (Bq/yr)	Radon (WLM/yr)	Gross alpha (Bq/yr)	Radon (WLM/yr)	Gross alpha (Bq/yr)	Radon (WLM/yr)
1975-1979	0	4.5E-02	Not under DOE	Not under DOE	0	3.1E-02
1980	0	1.1E-01	Not under DOE	Not under DOE	0	2.4E-02
1981	0	7.1E-02	Not under DOE	Not under DOE	0	2.5E-02
1982	0	5.6E-02	Not under DOE	Not under DOE	0	2.5E-02
1983	0	2.0E-02	Not under DOE	Not under DOE	0	2.1E-02
1984	0	3.5E-02	Not under DOE	Not under DOE	0	2.4E-02
1985	0	2.3E-02	0	1.5E-02	0	2.4E-02
1986	0	3.3E-02	0	2.2E-02	0	2.3E-02
1987	0	2.6E-02	0	2.8E-02	0	4.3E-02
1988	0	4.2E-02	0	3.9E-02	0	6.5E-02
1989	4.9E-01	3.2E-02	3.9E-01	3.6E-02	4.6E-02	4.1E-02
1990	2.5E-01	1.9E-02	2.2E-01	2.3E-02	4.0E-02	3.3E-02
1991	1.2E-01	2.1E-02	1.3E-01	1.9E-02	8.0E-03	2.9E-02
1992	9.3E-02	2.1E-02	9.9E-02	1.5E-02	0	2.6E-02
1993	1.1E-01	9.6E-03	1.1E-01	7.1E-03	7.5E-03	5.8E-02
1994	1.3E-01	2.7E-02	1.5E-01	1.6E-02	5.8E-02	1.6E-01
1995	1.1E-01	3.3E-02	1.2E-01	2.1E-02	1.9E-02	5.5E-02
1996	1.5E-01	5.4E-02	1.4E-01	2.6E-02	8.0E-03	1.2E-02
1997	1.8E-01	1.2E-01	1.3E-01	2.1E-02	0	1.1E-02
1998	2.1E-01	2.2E-02	1.8E-01	1.3E-02	6.2E-03	7.1E-03
1999	1.9E-01	3.5E-02	1.8E-01	2.4E-02	8.9E-04	1.6E-02
2000	9.9E-02	1.9E-02	1.2E-01	2.5E-02	8.0E-03	1.2E-02
2001	9.9E-02	1.9E-02	1.2E-01	2.5E-02	8.0E-03	1.2E-02
2002	9.9E-02	1.9E-02	1.2E-01	2.5E-02	8.0E-03	1.2E-02
2003	9.9E-02	1.9E-02	1.2E-01	2.5E-02	8.0E-03	1.2E-02
2004	9.9E-02	1.9E-02	1.2E-01	2.5E-02	8.0E-03	1.2E-02

a. Represents the sum of ²³⁸U and ²³⁴U activities. Assume ²³⁸U and ²³⁴U are equal in activity.

concentrations of radon during this period. The model is from Volume 1 of National Council on Radiation Protection and Measurements Report 123 (NCRP 1996).

The NCRP (1996) screening model adopted for these calculations is a simple dispersion model that accounts for potential increases in exposure due to building wake effects for close-in receptors (i.e., those less than 100 m from the source location). Air concentration (C) of radon was calculated from:

$$C = (f Q)/(\pi u h K) \text{ Ci/m}^3 \tag{4-1}$$

where:

C = concentration at receptor (Ci/m³)

Table 4-7. Estimated site-wide maximum inhalation intake of radioactive air particulates and radon at WSCP, WSRP, and WSQ (assume respirable fraction of sampled particles = 1.0, and based on 2,400 m³/yr inhalation rate).

Year	WSP (unspecified location)		
	U-238 and U-234 ^a (Bq/yr)	Th-230 Bq/yr	Radon (WLM/yr)
Operational period			
1957	5.7E+02	1.4E+01	8.7E-02
1958	5.7E+02	1.4E+01	8.7E-02
1959	5.7E+02	1.4E+01	8.7E-02
1960	5.6E+02	1.4E+01	8.7E-02
1961	5.7E+02	1.4E+01	8.7E-02
1962	5.4E+02	1.3E+01	8.7E-02
1963	5.5E+02	1.4E+01	8.7E-02
1964	5.5E+02	1.4E+01	8.7E-02
1965	5.4E+02	1.3E+01	8.7E-02
1966	5.4E+02	1.3E+01	8.7E-02
1967	5.4E+02	1.3E+01	8.7E-02

Year	WSP (unspecified location)	
	Gross alpha (Bq/yr)	Radon (WLM/yr)
Monitoring and remediation period		
1975-1979	0	4.5E-02
1980	0	1.1E-01
1981	0	7.1E-02
1982	0	5.6E-02
1983	0	2.1E-02
1984	0	3.5E-02
1985	0	2.4E-02
1986	0	3.3E-02
1987	0	4.3E-02
1988	0	6.5E-02
1989	4.9E-01	4.1E-02
1990	2.5E-01	3.3E-02
1991	1.2E-01	2.9E-02
1992	9.3E-02	2.6E-02
1993	1.1E-01	5.8E-02
1994	1.5E-01	1.6E-01
1995	1.2E-01	5.5E-02
1996	1.5E-01	5.4E-02
1997	1.8E-01	1.2E-01
1998	2.1E-01	2.2E-02
1999	1.9E-01	3.5E-02
2000	1.2E-01	2.5E-02
2001	1.2E-01	2.5E-02
2002	1.2E-01	2.5E-02
2003	1.2E-01	2.5E-02
2004	1.2E-01	2.5E-02

a. Represents the sum of ²³⁸U and ²³⁴U activities. Assume ²³⁸U and ²³⁴U are equal in activity.

Q = stack or building vent release rate (Ci/s)

f = wind frequency

u = wind speed (m/s)

h = height of effluent release (m)

K = constant (m)

Values used for the independent factors f and K correspond to the defaults in NCRP (1996), such that f is assumed equal to 0.25 (the maximum frequency for any compass point), and K is 1 m. The average wind speed at the WSCP from 1951 to 1970 was 4.2 m/s from the south (Weidner and Boback 1982). The height of effluent release from the acid recovery plant is not known, but is assumed to be 10 m.

The estimate by Meshkov et al. (1986) of an average annual release of radon (^{222}Rn) between 12 and 34 Ci/yr (1.4×10^4 to 4.0×10^4 Bq/s) is based on the following: (1) 5,000 to 14,500 MT of uranium materials were processed per year; (2) 70% of this was uranium; (3) radium activity was 1% of the uranium activity (believed to be an upper end estimate); (4) radon was in equilibrium with radium, and (5) all radon escaped during processing. Using this value in Equation 4-1, the estimated average radon concentration within 100 m of the source is between approximately 30 and 80 Bq/m³ (0.7 and 2 pCi/L). Because the radon concentrations reported here include background radon contributions, the average value of 11 Bq/m³ (0.3 pCi/L) for the Weldon Spring area (MK-Ferguson 2001a) is added to the calculated estimates, such that the average concentration ranges from 41 to 91 Bq/m³. This TBD uses the upper estimate, which is a claimant-favorable assumption.

A similar estimate for thoron release from the WSCP during the operational period was not made, in part because the type of thorium material processed is not sufficiently described in accessible documentation. DOE (1986) states that "natural thorium was typically received in either a nitrate or oxide form," with no mention of it as a concentrated feed material. In addition, DOE (1986) indicates that the amount of thorium material processed ranged from a low of 0.05% of the natural uranium material processed in 1964 to a high of 7% in 1965. According to Wallo (1981), the ThO₂ mass content, and thus the ^{232}Th content, of natural thorium materials (like monazite sand) is approximately 5%. Assuming this and a secular equilibrium between ^{232}Th and ^{224}Ra and ^{220}Rn , it follows that the estimated release of thoron according to the procedure above for radon is between 0.04% and 5% of the radon release. If, however, the thorium material is assumed to be concentrated similarly to uranium (i.e., 70% ^{232}Th), the thoron release could range from 0.5% and 70% of the radon release. Either way, the estimated dose attributable to thoron and its progeny is insignificant under these assumptions with respect to radon and its progeny because radon progeny have a higher estimated equilibrium factor and higher associated dose factors than thoron progeny (MK-Ferguson 2001b).

This TBD analysis estimated radon concentrations for the WSQ during the operational period from measurements made in the vicinity from 1979 through 1982 (Meshkov et al. 1986; MK-Ferguson 1989a). Because the activities of ^{226}Ra and ^{224}Ra are not significantly depleted in the quarry over time, due to limited leaching and continuous production of these isotopes from precursors present in the waste (Section 4.2.3.1), these years reasonably represent radon emanations for the WSQ during the years of operation of the plant. These measurements are applied to the 1963-1967 period only. Prior to that, drummed thorium waste was probably submerged and no significant source of radon probably existed. Table 4-5 lists the average measured concentrations for this area.

Measurements of radon in 1979 through 1982 did not distinguish between ^{222}Rn and ^{220}Rn . Therefore, the analysis made a dose-maximizing assumption that all the radon measured was ^{222}Rn . Because most of the dose from inhaled radon is due to alpha-emitting, short-lived daughters, the working level month (WLM) unit is often a preferred method of reporting inhalation intake exposure to

radon to specify the degree of equilibrium of radon daughters assumed in calculating exposure. For ^{222}Rn , 1 working level (WL) = 3,700 Bq/m³ (100 pCi/L). The calculation of WLM for ^{222}Rn assumes a 2,000-hr/yr occupational exposure period and an outdoor environmental radon daughter equilibrium factor of 30%, which is the upper limit of measured factors for the WSP (MK-Ferguson 2001b). The WLM/yr intake per Bq/m³ for ^{222}Rn is calculated from

$$^{222}\text{Rn} \frac{\text{WLM/yr}}{\text{Bq/m}^3} = \frac{0.3 \times 2000}{3700 \times 170} = 9.54 \times 10^{-4} \quad (4-2)$$

The radon concentrations in Table 4-5 were converted to WLM/yr by multiplying activity concentrations by the factor derived in equation 4-2; the results are in Table 4-6, which lists inhalation intakes for relevant WSP areas. Table 4-7 lists site-wide maximum intakes.

4.2.3.2 Post-1974 Intakes (Monitoring and Remediation Periods)

After 1974, air monitoring occurred in the WSRP and WSQ areas. Measurements of radon in air were reported in annual reports beginning with a monitoring report for 1979 and 1980 (Weidner and Boback 1982), which was specific to the raffinate pits and quarry. In October 1985, the monitoring program was expanded to include the WSCP (Bechtel 1986), which transferred back to DOE at that time. Beginning in 1987, air particulate monitoring was added (MK-Ferguson 1988) at the WSCP and WSRP, and at the WSQ in 1989 (MK-Ferguson 1990b). These data, reported in the annual monitoring reports for 1979 to 2000 (Weider and Boback 1982; Bechtel 1983a,b, 1984b, 1985a, 1986; MK-Ferguson 1987, 1988, 1989b, 1990b, 1991, 1992a, 1993, 1994, 1995, 1996, 1997, 1998a, 1999, 2000b, 2001a), when remediation was complete, are the basis for the following estimates of air concentrations and intakes at the WSRP, WSCP, and WSQ for the post-1974 periods.

Airborne Particulate Radionuclides

Table 4-5 lists air monitoring data for airborne particulate radionuclides during the monitoring and remediation periods according to year and WSP location. Figures 4-1 and 4-2 show the locations of monitoring stations for the WSRP and WSCP and for the WSQ, respectively. Radioactive air particulates were not monitored prior to the beginning of remediation activities at the site. In 1987 and 1988, monitoring of particulates at a few perimeter locations for the WSRP and WSCP, and in 1989 at the WSQ, indicated that measured gross alpha concentrations were statistically indistinguishable from background (MK-Ferguson 1988, 1989b, 1990b). Therefore, the concentrations listed in Table 4-5 for 1987 and 1989 at the WSRP and WSCP and in 1989 at the WSQ are zero. Because all areas of the WSP were essentially undisturbed between 1975 and the beginning of remediation activities (which began in 1990 at the WSQ), this analysis assumed that radioactive air particulate concentrations prior to 1987 are the same as those measured in 1987 through 1989; therefore, Table 4-5 lists particulate concentrations from 1975 through 1986 as insignificant. There is no reason to believe that the late 1980s data do not adequately represent differences in vegetative cover or wind events during the unmonitored period.

From 1989 to 2000, concentrations of radioactive air particulates were measured and reported at locations identified in Figures 4-1 and 4-2, even though many of the concentrations were statistically indistinguishable from background. The measurements occurred at perimeter locations for the WSRP, WSCP, and WSQ. To consider the possibility of higher onsite concentrations, the analysis used the ratio of onsite radon concentration to perimeter radon concentration to scale the perimeter particulate data. As explained in Section 4.2.3.2.2, the maximum ratio of average onsite to average perimeter radon concentrations is 2.0. It is reasonable to assume that this ratio represents the ratio of average onsite to average perimeter air particulate concentrations because the source of the airborne radioactivity is the same. Therefore, the measured perimeter concentrations of airborne

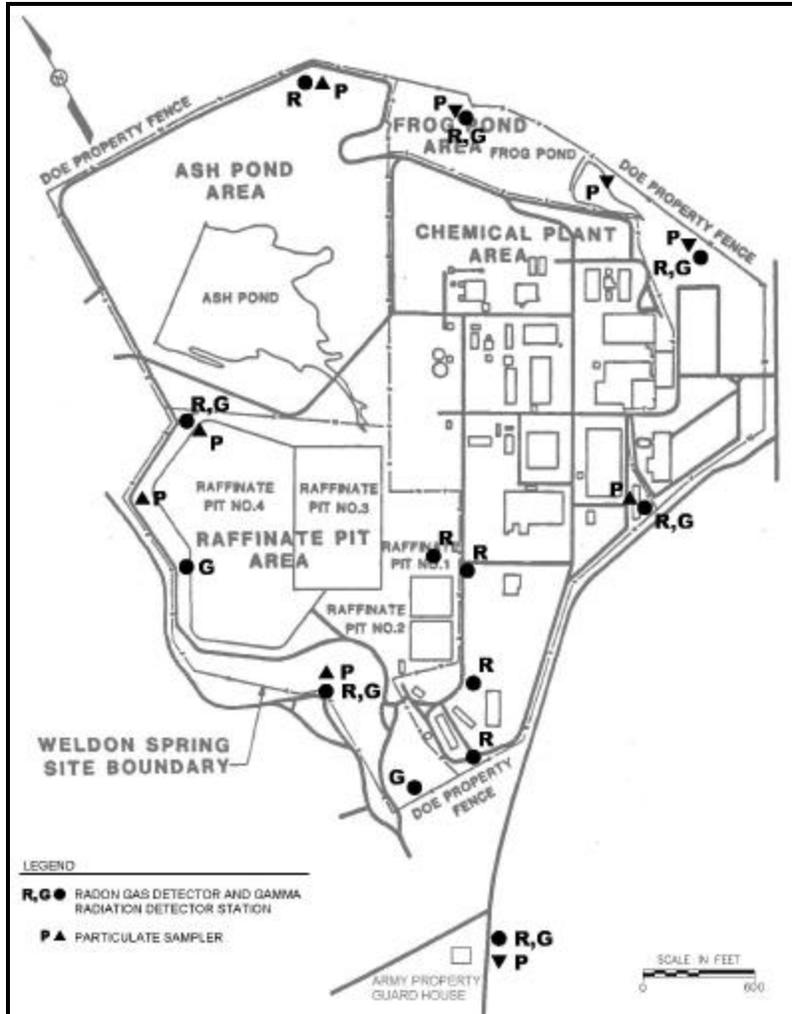


Figure 4-1. Air particulate (P), radon (R), and gamma (G) monitoring stations in WSCP and WSRP.

particulate radionuclides were multiplied by a factor of 2 to represent the average onsite airborne concentrations in Table 4-5.

Section 4.2.1.1 identifies ^{227}Ac , ^{231}Pa , ^{210}Po , ^{228}Th , ^{230}Th , ^{232}Th , ^{234}U , and ^{238}U as potentially significant isotopes of concern during the remediation period. However, air concentrations of these isotopes were measured in terms of gross alpha concentrations at the WSP for most years of interest. The lack of refinement is probably because (1) the radionuclides of interest are alpha-emitters, and (2) the perimeter data for the WSRP, WSCP, and WSQ, representing all that are available for air particulates, seldom showed air concentration measurements higher than those at offsite background monitoring stations. Some isotope-specific data appear in annual reports in the late 1990s but, because the ratio of isotopes probably would be highly variable by location due to the variability in isotopic composition of the contaminated soil, the utility of these data is limited.

Air particulate monitoring ended at the end of 2000 for all WSP areas because radioactive waste handling activities were essentially complete and no critical receptor air monitoring data had demonstrated a dose to the public of greater than 1 mrem (MK-Ferguson 2000a). Therefore, this TBD assumes that radioactive air particulate concentrations measured in 2000 are a reasonable estimate of the air concentrations in subsequent years to the present.

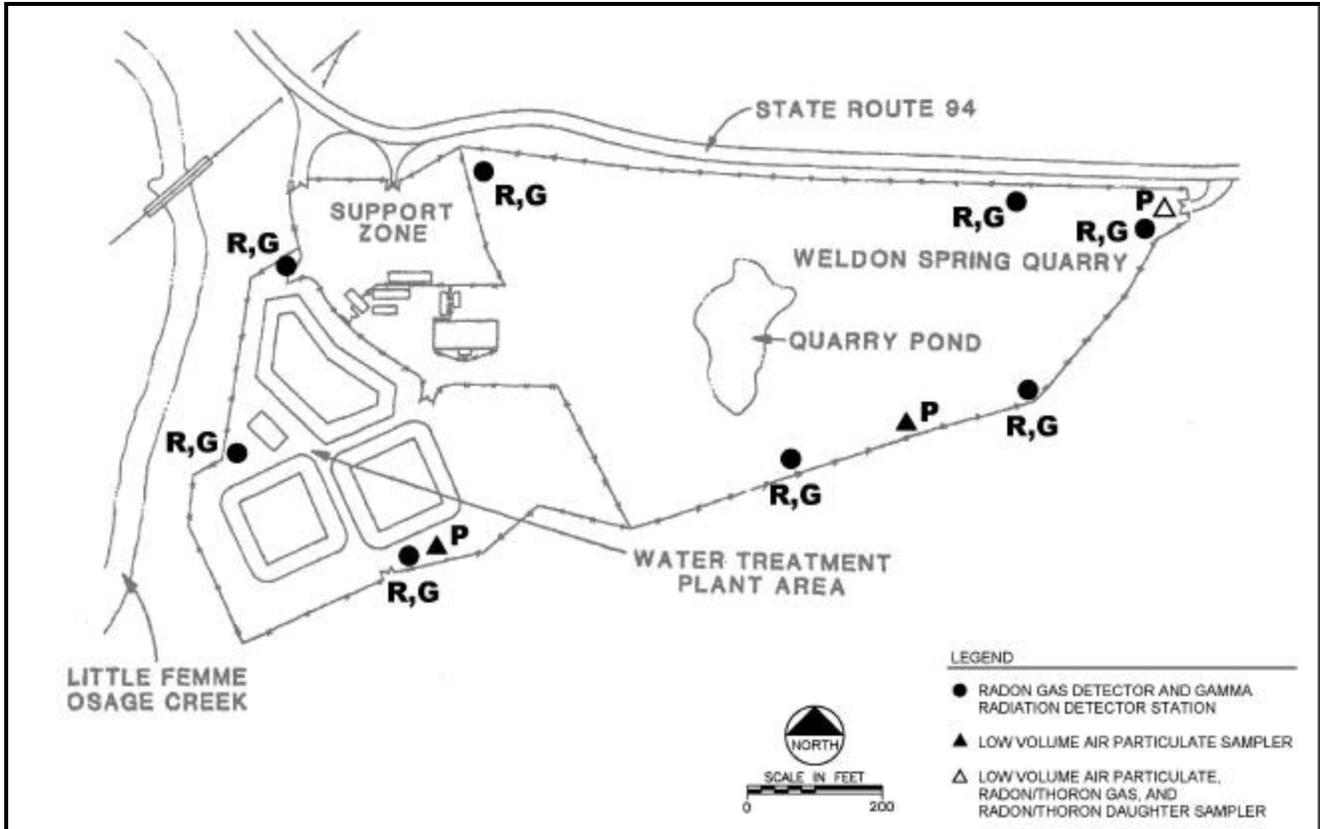


Figure 4-2. Air particulate (P), radon (R), and gamma (G) monitoring stations in WSQ.

Table 4-6 lists estimated annual gross alpha activity intakes of airborne particulate radionuclides in terms of gross alpha (Bq/yr) between 1975 and 2004. The table does not include estimates for the WSCP from 1975 to 1984 because this facility did not transfer to DOE until 1985. The intakes were obtained by multiplying the respective gross alpha concentrations (Bq/m^3) in Table 4-5 by the assumed inhalation rate of $2,400 \text{ m}^3/\text{yr}$.

Table 4-7 lists site-wide maximum gross alpha intakes for the remediation period. These intakes correspond to the highest value, by year, for any location listed in Table 4-6.

Dose reconstructors must assign the gross alpha activity intakes to a radionuclide or radionuclides likely to produce most of the organ dose. In lieu of this information, it is reasonable to consider the radionuclides capable of maximizing the calculated dose. Table 4-8 lists the ICRP (2001) organ dose factors for ^{238}U , ^{234}U , ^{230}Th , ^{210}Po , ^{232}Th , ^{228}Th , ^{231}Pa , and ^{227}Ac . Only organs for which one or more isotopic dose factors exceed $2 \times 10^{-5} \text{ Sv}/\text{Bq}$ are listed. This lower limit was selected based on consideration of the highest site-wide maximum gross alpha intake reported in Table 4-6, which is $0.49 \text{ Bq}/\text{yr}$. This intake would correspond to an organ dose of at least $10^{-5} \text{ Sv}/\text{yr}$ ($1 \text{ mrem}/\text{yr}$). The highest organ dose factor for each organ or tissue is shown in **bold** type in Table 4-8. Except for dose to kidneys, the highest dose factor is associated with ^{227}Ac . However, because airborne particulates in the postoperational period probably arise from contaminated soils, it is reasonable to also consider the relative prevalence of these radionuclides in soil. Table 4-9 lists the activity concentration ratios estimated for these radionuclides in raffinate sludge (which became a potential source of air contamination during remediation), WSCP soil, and quarry soil (DOE 1992). Based on the information in Tables 4-8 and 4-9, Table 4-10 was constructed to identify the most important radionuclide contributing to the gross alpha based on consideration of organ dose factors and relative

concentrations in soil for the three areas (WSRP, WSCP, and WSQ). The percentages in parentheses show the estimated percent contribution to total dose for the radionuclide listed, assuming the relative concentrations in Table 4-9 are representative of the air concentrations encountered. From Table 4-10, it is evident that ^{230}Th is generally the most important contributor to dose for the WSRP and WSQ, which is a reasonable result considering the elevated concentration of this radionuclide relative to other alpha-emitters present (Table 4-9). Ac-227 and uranium isotopes are more important at the WSCP, where the ^{230}Th concentrations are not found to be as relatively significant. Table 4-10 indicates that small contributors to activity may be relatively large contributors to dose.

Table 4-8. Organ dose factors for alpha-emitting radionuclides of concern at WSP.

Radionuclide	Critical organ 50-year committed dose factor ^{a,b,c} (Sv/Bq)						
	Bone surface	Kidneys	Liver	Red marrow	Ovary/testes ^d	ET airway	Lung
U-238	1.0E-05	3.6E-06	1.3E-06	1.1E-06	3.4E-07	6.5E-05	3.4E-05
U-234	1.1E-05	4.0E-06	1.5E-06	1.1E-06	3.9E-07	7.5E-05	4.1E-05
Th-230	1.5E-03	2.2E-05	1.9E-05	5.2E-05	1.2E-05	7.4E-05	4.0E-05
Po-210	9.4E-07	7.5E-06	3.9E-06	1.6E-06	1.6E-07	7.3E-06	1.7E-05
Th-232	1.5E-03	2.3E-05	2.3E-05	5.7E-05	1.3E-05	1.1E-04	7.7E-05
Th-228	2.8E-04	1.2E-05	7.7E-07	2.2E-05	2.4E-06	2.1E-04	2.1E-04
Pa-231	4.4E-03	4.3E-07	2.8E-07	3.6E-04	8.2E-09	1.1E-04	7.2E-05
Ac-227	1.9E-02	9.5E-08	4.3E-03	1.5E-03	2.3E-04	3.8E-04	2.2E-04

- Critical organ* is defined here as the organ associated with the highest committed dose over all absorption classes, for particles of 5- μm AMAD.
- Dose factors from ICRP (2001).
- For ET Airways and Lung, the least soluble form (absorption type S) is associated with the highest dose factor; for all other organs, the most soluble form listed in ICRP (2001) (absorption type M or F) is associated with the highest dose factor.
- The highest dose factor for ovary and testes is listed (in most cases, the value is the same for both organs).

Table 4-9. Normalized concentrations of radionuclides in soil at WSP.

Radionuclide	Raffinate sludge activity normalized to U-238 concentration ^a	Chemical plant soil activity normalized to U-238 concentration ^a	Quarry soil activity normalized to U-238 concentration ^a
U-238	1.00E+00	1.00E+00	1.00E+00
U-234	1.00E+00	1.00E+00	1.00E+00
Th-230	2.50E+01	1.50E-01	1.70E+00
Po-210	6.20E-01	1.40E-01	1.70E+00
Th-232	2.60E-02	7.60E-02	1.30E-01
Th-228	2.50E-01	7.60E-02	4.80E-01
Pa-231	7.30E-01	3.80E-02	5.80E-03
Ac-227	5.80E-01	1.90E-02	2.90E-03

- Most measurements made during remediation period in 1989 and 1990 (DOE 1992).

Based on the information in these tables, the following recommendations are made for selecting representative radionuclides for the gross alpha intakes. For the raffinate pits area (WSRP), it is recommended that it be assumed that the combined $^{238,234}\text{U}$ activity is 10% of the total, ^{230}Th is 85% of the total, ^{228}Th is 1% of the total, and ^{227}Ac is 4% of the total activity. For the chemical plant area (WSCP), it is recommended that it be assumed that the combined $^{238,234}\text{U}$ activity is 85% of the total, ^{230}Th is 10% of the total, ^{228}Th is 4% of the total, and ^{227}Ac is 1% of the total activity. For the quarry area (WSQ), it is recommended that it be assumed that the combined $^{238,234}\text{U}$ activity is 35% of the

Table 4-10. Largest contributors to dose for the WSRP, WSCP, and WSQ.^a

Organ of interest	WSRP	WSCP	WSQ
Bone surface	Th-230 (72%)	Ac-227 (40%)	Th-230 (85%)
Kidneys	Th-230 (97%)	U-234/238 (52%)	Th-230 (56%)
Liver	Ac-227 (84%)	Ac-227 (91%)	Th-230 (56%)
Red marrow	Th-230 (53%)	Ac-227 (49%)	Th-230 (75%)
Ovary/testes	Th-230 (69%)	Ac-227 (54%)	Th-230 (82%)
ET airway	Th-230 (79%)	U-234/238 (75%)	Th-230 (32%)
Lung	Th-230 (76%)	U-234/238 (67%)	Th-228 (36%)

a. Based on organ dose factors in Table 4-8, and relative contributions to soil concentration listed in Table 4-9.

total, ²³⁰Th is 54% of the total, ²²⁸Th is 10% of the total, and ²²⁷Ac is 1% of the total activity. These recommendations are summarized as:

Percent activity to assume for gross alpha activity			
	WSRP	WSCP	WSQ
^{238/234} U (total)	10	85	35
²³⁰ Th	85	10	54
²²⁸ Th	1	4	10
²²⁷ Ac	4	1	1

Radon

During the monitoring and remediation periods, outdoor radon was measured at the WSCP, WSRP, and WSQ (Bechtel 1986, MK-Ferguson 1987, 1988, 1989a,b, 1990b, 1991, 1992a,b, 1993, 1994, 1995, 1996, 1997, 1998a,b, 1999, 2000b, 2001a). At the WSCP, only perimeter measurements occurred except in 1999 and 2000, when some measurements occurred inside the perimeter where the disposal cell received much of the remediation wastes. The ratio of radon concentrations in the disposal cell area to the WSCP perimeter ranged from 1.4 to 5.0 for 1999 and 2000.

At the WSRP, radon measurements occurred at the perimeter monitoring stations between 1985 and 2000, at one location inside the perimeter during 1985 and 1986, and at several interior locations during the years of active remediation (1998-2000). The maximum ratio of average radon concentrations at the interior locations to the average at the perimeter was 2.0.

This TBD assumes that the average radon concentration measured at the perimeter stations might underestimate the average radon concentration over the WSCP and WSRP onsite areas during the remediation period by a factor of 2. Therefore, the measured concentrations were multiplied by 2 to provide an average net radon concentration for each of the two areas (WSCP and WSRP). Table 4-5 lists the derived concentrations, which include contributions of radon from naturally occurring ²²⁶Ra and ²²⁸Ra. Again, these concentrations reflect total radon (²²²Rn and ²²⁰Rn) as measured by F-type alpha track detectors.

At the WSQ, radon measurements occurred only at the perimeter monitoring stations except in 1989, when monitors were placed inside the perimeter down to the quarry floor. The ratio of the average perimeter radon concentration to the average interior concentration was 3.6. This is a significant increase; however, the interior of the quarry is fairly inaccessible, with a quick drop in elevation and an abundance of vegetation. Thus, this TBD assumes that the perimeter concentrations are reasonably representative of exposure concentrations except during the active removal of the quarry bulk waste in 1993, 1994, and 1995. The TBD assumes the radon concentration during this period of active

remediation to be 3.6 times the concentration measured at the perimeter, as reflected in Table 4-5 values.

The TBD analysis used equation 4-2 to convert the radon concentrations in Table 4-5, and Table 4-6 lists the results, including inhalation intakes for relevant areas of the WSP. Table 4-7 lists site-wide maximum intakes.

4.3 EXTERNAL EXPOSURE TO ONSITE ATMOSPHERIC RADIONUCLIDE CONCENTRATIONS

The external dose-producing radionuclides of concern for the WSP are those associated with the ^{235}U , ^{238}U , and ^{232}Th decay series, and include short-lived daughter isotopes of radium. External ambient exposure at the WSP is a result of gamma and X-ray radiation emitted from radionuclides in the ore concentrate stored on the site during the operational period, and from radionuclides in the raffinate pits and quarry. An aerial radiological survey of the WSP (Jobst 1976) indicated that for most of the site the normal terrestrial gamma exposure rate is 3 to 6 $\mu\text{R/hr}$ at 1 meter above ground level, and the average cosmic exposure rate is approximately 4 $\mu\text{R/hr}$. For continuous exposure over a year, this represents an annual ambient dose of 61 to 88 mrem/yr. However, the aerial survey found elevated terrestrial exposures over the raffinate pits and the quarry, indicating the presence of manmade changes from the natural radioisotopes.

4.3.1 Ambient Exposure during Operational Period (1957-1966)

Available documents do not contain monitoring data describing the ambient exposure rate at the WSP during the operational period. During this period, ambient dose rates in excess of natural background radiation would have been primarily due to gamma radiation from short-lived decay products of ^{238}U in the vicinity of the storage pad for the drums containing the ore concentrate (Meshkov et al. 1986, p. 46). Exposure to radionuclides accumulated in the raffinate pits and quarry would be mitigated somewhat by the shielding and physical barrier provided by water covering the pits entirely and some of the waste in the quarry for much of the year.

In lieu of site-specific data for the WSCP during the operational period, the TBD analysis used ambient dose rates for the Feed Materials Plant in Fernald, Ohio, as reported in the Occupational Environmental Dose TBD for the Fernald Environmental Management Project (FEMP; ORAU 2004). FEMP received yellowcake for processing similar to that received at the WSP, in similar quantities. A comparison of the natural uranium processed (shipped) between 1957 and 1966 by the WSP and FEMP indicated a nearly identical yearly average of approximately 11.5×10^6 kg uranium (DOE 1986; RAC 1995). From the Fernald TBD, ambient dose rates reported for the early operational period (prior to 1976) are not based on measurements made during those years, but were scaled from measured rates in later production years, according to production level. The reported ambient rate is averaged over all FEMP areas except the K-65 Silo area, which is infrequently occupied, and which does not have a physical counterpart at WSP. The net (background subtracted) average dose rate for 1956 to 1970 was 0.18 mrem/hr, or 1,576 mrem/yr for continuous exposure. Assuming this is a reasonable dose rate estimate for the WSP during the operational period, the analysis added the WSP site background rate of 99 mrem/yr (Bechtel 1986) to derive an ambient dose rate for continuous exposure of 1,675 mrem/yr. This corresponds to a 2,000-hr/yr occupational exposure of 382 mrem.

4.3.2 Ambient Exposure during Monitoring and Remediation Periods (1975-2004)

Between 1982 and 2000, ambient exposure was monitored using thermoluminescent dosimeters at many perimeter locations around the WSCP, the WSRP, and the WSQ (Bechtel 1983b, 1984a, 1985a, 1986; MK-Ferguson 1987, 1988, 1989a,b, 1990b, 1991, 1992a,b, 1993, 1994, 1995, 1996, 1997, 1998a,b, 1999, 2000b, 2001a). In 1986, locations in the WSCP and the WSRP were monitored, although the results did not indicate a significant difference in exposure rate from the perimeter stations. Figures 4-1 and 4-2 show the locations of monitoring stations.

Reported ambient exposure values included background except for 1985, when an average background rate of 99 mrem/yr was subtracted from the total measured values at the monitoring stations. Many of the values reported for 1985 were zero, representing a net value (exposure rate minus background rate) of either zero or less than zero. To use these numbers for 1985, the TBD analysis added the background value of 99 to the reported net value. Therefore, the average ambient exposures for 1985 might be a slight overestimate due to the addition of background to a zero value that actually might have been less than zero.

Ambient exposure rates in excess of natural background radiation exposure rates existed prior to remediation in areas of the WSP that are close to waste in the raffinate pits or quarry, as recognized in Jobst (1976). A comprehensive radiological survey of the WSRP in 1982 and 1983 (Bechtel 1984b) indicated an average onsite gamma exposure rate of 23 μ R/hr for that site, which gives a dose rate of approximately 200 mrem/yr for continuous exposure (8,760 hr/yr). This is higher than the perimeter monitors reported for the WSRP prior to remediation. A comparison of the average survey-based value (200 mrem/yr) to the average perimeter value in 1983 of 88 mrem/yr (Bechtel 1984a) suggests the area inside the WSRP perimeter is more appropriately characterized by an exposure rate approximately twice the average perimeter values. The modified ambient dose rate in Table 4-11 represents twice the average perimeter values for 1984 through 2005, and is recommended for environmental dose reconstruction. For 1975 through 1983, the value of 200 mrem/yr in Table 4-11 is the average survey value for 1982-1983, from Bechtel (1984b), as is considered the best value to use for those years.

A similar radiological survey of the WSQ in 1984 and 1985, before remediation occurred, indicated gamma exposure rates ranging from 8 μ R/hr (similar to the overall area background rate) to 286 μ R/hr over the quarry floor, where contamination was greatest (Bechtel 1985b). These exposure rates correspond to annual doses between 70 and 2,500 mrem/yr for continuous exposure (8,760 hr/yr). An average value was not provided, nor were the original data available, so the utility of this information is limited. The extent of the high exposure rate area is not defined. The survey report did state, however, that characterization of the WSQ was "extremely difficult because of the rough terrain", and that "the area was densely vegetated." Figure 4-3 includes a topographic map of the WSQ area, along with locations of gamma exposure rate measurements. The survey included many of the relatively inaccessible areas of the quarry, where significant exposure was unlikely except during excavation of quarry bulk wastes. Thus, the perimeter exposure rates are appropriate for estimating exposures except during the excavation period, which occurred during most of 1993, 1994, and 1995 (MK-Ferguson 1994, 1995, 1996). This TBD assumes that the exposure rate during the years the quarry bulk wastes were being excavated to be approximately 2,500 mrem/yr for continuous (8,760-hr/yr) exposure, in lieu of other relevant data. It is not reasonable to assume that an individual spent 24 hours a day at this location, and the dose reconstructor should make a special effort to define accurately the potential time of exposure inside the WSQ perimeter when calculating the best estimate dose. Table 4-11 lists the appropriate value to assume for external environmental exposure at the WSQ.

Table 4-11. Estimated ambient onsite dose for WSRP, WSCP, and WSQ at Weldon Spring.

Year	WSRP average ^a (mrem/yr) ^b	WSCP average ^c (mrem/yr) ^b	WSQ average ^d (mrem/yr) ^b
1957-1967	Not significant	1,675	Not significant
1975-1979	200	(e)	97
1980	200	(e)	97
1981	200	(e)	97
1982	200	(e)	85
1983	200	(e)	95
1984	236	(e)	125
1985	223	198	123
1986	184	155	96
1987	139	145	95
1988	126	120	73
1989	140	131	81
1990	133	132	77
1991	150	140	84
1992	137	135	73
1993	125	118	2,500
1994	120	118	2,500
1995	140	144	2,500
1996	135	165	62
1997	135	185	61
1998	126	139	57
1999	124	124	55
2000	61	113	48
2001	61 ^f	113 ^f	48 ^f
2002	61 ^f	113 ^f	48 ^f
2003	61 ^f	113 ^f	48 ^f
2004	61 ^f	113 ^f	48 ^f

- Prior to 2000, average value is twice the average measured at perimeter locations for the WSRP.
- Based on 8,760 hr/yr exposure; includes contributions from natural background.
- Prior to 2001, average value is twice the average measured at perimeter locations for the WSCP.
- Except for 1993-1995, values are the average measured at perimeter locations for the WSQ.
- WSCP did not transfer to DOE until 1985.
- Not monitored after 2000; assumed equal to average for 2000 for respective location.

A 1987 radiologic characterization of the WSCP and WSRP (Marutzky et al. 1988) used portable scintillation counters to record external radiation, in counts per minute, at 880 locations across the WSRP and WSCP; cross-correlation with a pressurized ionization chamber allowed estimates of exposure rate, in $\mu\text{R/hr}$. Exposure rates ranged from 9 to 287 $\mu\text{R/hr}$, which corresponds to a dose rate between 79 and 2,500 mrem/yr for continuous exposure (8,760 hr/yr). One location had an exposure rate of 287 $\mu\text{R/hr}$; the next highest value was 85 $\mu\text{R/hr}$. Fewer than 10% of the 880 locations surveyed had exposure rates measuring 15 $\mu\text{R/hr}$ or above. Assuming 10% of the locations were at 85 $\mu\text{R/hr}$ and the remaining 90% were at approximately 11 $\mu\text{R/hr}$ produces an average exposure rate of approximately 18 $\mu\text{R/hr}$. This is a reasonable assumption for the WSCP area because few of the exposure rates over 15 $\mu\text{R/hr}$ exceed 20 $\mu\text{R/hr}$, and many of the contaminated areas are probably not those in which occupational exposures would normally occur. The elevated

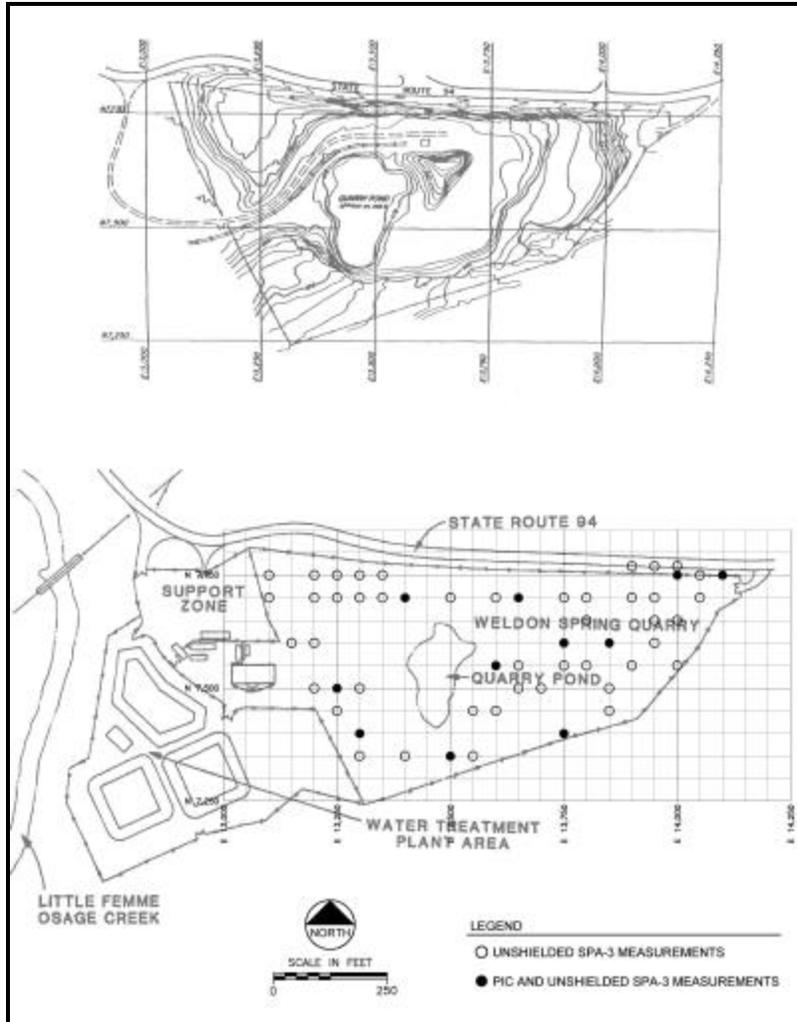


Figure 4-3. Topographic map and locations of gamma exposure rate measurements for WSQ (from Bechtel 1985b).

exposure rates were in the areas surrounding the buildings, in onsite dump areas, in Ash Pond, and in drainages of Frog Pond, but it is not possible to sort out exposure rates as a function of location in the reports. An exposure rate of $18 \mu\text{R/hr}$ represents an annual dose of 158 mrem/yr for continuous exposure. In 1987, the reported WSCP perimeter exposure rate was 72 mrem/yr. Thus, for the WSCP, the TBD analysis assumed that until remediation was essentially completed in 2000, the average onsite exposure rate was approximately twice the measured perimeter rate. Table 4-11 lists values reflecting this assumption.

Measured exposure rates are not available for 1975 through 1981 for the WSQ or for 1975 through 1982 for the WSRP. It is reasonably assumed that the averaged exposure rates from 1982 through 1989 for the WSQ and from 1983 through 1989 for the WSRP adequately represent these earlier years, when the entire site was essentially undisturbed. Table 4-11 lists these averaged rates.

Table 4-12 lists site-wide maximum external exposure rates for WSP during the remediation period.

Table 4-12. Estimated maximum site-wide ambient dose at WSP.

Year	Maximum for WSRP, WSCP, and WSQ ^a (mrem/yr) ^b
1957-1967	1,675
1975-1979	200
1980	200
1981	200
1982	200
1983	200
1984	236
1985	223
1986	184
1987	145
1988	126
1989	140
1990	133
1991	150
1992	137
1993	2,500
1994	2,500
1995	2,500
1996	165
1997	185
1998	139
1999	124
2000	113
2001	113 ^c
2002	113 ^c
2003	113 ^c
2004	113 ^c

- a. Maximum of annual recommended average value for each area (from Table 4-11).
- b. Based on 8,760-hr/yr exposure.
- c. Not monitored after 2000; assumed to be equal to the site-wide maximum for the WSP for 2000.

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GLOSSARY

alpha particles

Positively charged particles of discrete energies emitted by certain radioactive materials; alpha particles usually expend their energy in short distances and will not usually penetrate the outer layer of skin; they are a significant hazard only when taken into the body where their energy is absorbed by tissues.

becquerel

A special unit of activity. One Becquerel equals 1 nuclear transition 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.

curie

A special unit of activity. One curie equals 3.7×10^{10} nuclear transitions per second.

exposure

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

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.

isotope

Elements having the same atomic number but different atomic weights; identical chemically but having different physical and nuclear properties.

radiation

Alpha, beta, neutron, and photon radiation.

radioactivity

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

radionuclide

A radioactive isotope of an element, distinguished by atomic number, atomic weight, and energy state.

raffinate

A term loosely applied to waste from the uranium or thorium extraction step. Solids that result from the neutralization of this are also called "raffinates".

rem

A unit of dose equivalent equal to the product of the number of rad absorbed and the quality factor.

thermoluminescent dosimeter

A holder containing solid chips of material that when heated will release stored energy as light. The measurement of this light provides a measurement of absorbed dose.