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ACRONYMS AND ABBREVIATIONS

AEC	U.S. Atomic Energy Commission
AMAD	activity median aerodynamic diameter
Bq	becquerel
BZA	breathing-zone air
CEDE	committed effective dose equivalent
cm	centimeter
d	day
d/s	disintegrations per second
DAC	derived air concentration
dpm	disintegrations per minute
DR	dose reconstructor
DOE	U.S. Department of Energy
DWA	daily weighted average
EEOICPA	Energy Employees Occupational Illness Compensation Program Act
g	gram
hr	hour
ICRP	International Commission on Radiological Protection
IMBA	Interactive Modules for Bioassay Analysis
kg	kilogram
L	liter
m	meter
MAC	maximum allowable concentration
MCW	Mallinckrodt Chemical Works
MDC	minimum detectable concentrations
min	minute
ml	milliliter
MPC	maximum permissible concentration
mrem	millirem
nCi	nanocurie
NCRP	National Council on Radiation Protection and Measurements
pCi	picocurie
ppb	parts per billion
U.S.C.	United States Code
WL	working level
WLM	working level month

WSP	Weldon Spring Plant
WSCP	Weldon Spring Chemical Plant
WSQ	Weldon Spring Quarry
WSRP	Weldon Spring Raffinate Pits
WSSRAP	Weldon Spring Site Remedial Action Project
yr	year
µCi	microcurie
µg	microgram
µm	micrometer

5.1 INTRODUCTION

Technical Basis Documents (TBDs) 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 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 employer facility" or a "Department of Energy facility" as defined in the Energy Employees Occupational Illness Compensation Program Act of 2000 [EEOICPA; 42 U.S. C. § 7384l (5) and (12)].

5.1.1 Purpose

The purpose of this TBD is to provide a profile of internal dosimetry practices at the Weldon Spring Plant (WSP). This document contains technical information on the history, methods, and interpretation of monitoring data for the evaluation of occupational internal dose to WSP workers.

5.1.2 Scope

This TBD covers the methods used to assess internal radiation dose to workers at WSP. The WSP includes the Weldon Spring Chemical Plant (WSCP), the Weldon Spring Raffinate Pits (WSRP), and the Weldon Spring Quarry (WSQ). Internal radiation dose is the dose to a worker from deposition of radionuclides in the body. Such deposition can occur as a result of inhalation of radionuclides in airborne dust, incidental ingestion of radionuclides, and intake through intact skin or wounds. Because of the nature of the nuclides encountered at WSP, intake through the skin was unlikely.

The radionuclides of concern for WSP are the naturally occurring isotopes of uranium (^{234}U, ^{235}U, and ^{238}U) and their decay products (primarily ^{230}Th and ^{226}Ra). Because the WSP also processed some natural thorium, this TBD includes material on ^{232}Th and its decay products (^{228}Ra and ^{228}Th). However, due to the small amount of thorium processed at WSP, the primary radionuclides of concern for internal radiation dose are the isotopes of uranium.

Section 5.2 provides information on the source term for dosimetrically significant radionuclides at WSP. Sections 5.3 and 5.4 describe *in vitro* and *in vivo* measurements, respectively. Section 5.5 discusses air monitoring and dust studies, and Section 5.6 details assessment of radionuclide intakes.

5.1.3 Background on Mallinckrodt Chemical Works and the Weldon Spring Plant

Mallinckrodt Chemical Works (MCW) operated its Uranium Division from 1942 to 1966, initially at its downtown location (Destrehan Street), then later at the WSP site. Construction of WSP was authorized by the U.S. Atomic Energy Commission (AEC) in 1955, and operations began in 1957. In 1964 the plant employed 600 individuals, 80 of whom were in technical and managerial positions (MCW 1964; Mason c. 1977).

The functions performed at the two MCW facilities were essentially the same. One of the reasons for the construction of the new facility was the level of contamination present at the Destrehan Street site (Dupree 1979a). Worker exposures to airborne uranium were, in some cases, many times the tolerance level.

During the 24 yr of operations, the MCW Uranium Division processed uranium ores and concentrates into uranium trioxide (UO_3), uranium hexafluoride (UF_6), and uranium metal (Meshkov 1986). Uranium dioxide (UO_2) was an intermediate product in the conversion of UO_3 to UF_4 (ORAU 2005b). The feed material at the Destrehan Street plant included high-quality pitchblende (a natural uranium ore that includes the decay products generally in equilibrium). The WSP facility primarily handled uranium concentrates (e.g., yellowcake) received from uranium-milling facilities with very low concentrations of decay products. The historical documents report that WSP did not receive the high-quality pitchblende ores (Ingle 1991). The WSP processed thorium during the 2 yr before closure in 1966 and perhaps during earlier years as well (see Section 5.6).

Production at WSP waned during the later part of its operational period. In 1964, MCW published a document intended to encourage diversification and utilization of the existing facilities (MCW 1964; Mason c. 1977), which reflected that decline. The document described the plant facilities and briefly mentioned the operational safety program.

The WSP handled uranium concentrates. The primary concerns for radiation doses to WSP workers from uranium operations is internal deposition of uranium and beta dose from the short-lived decay products of uranium (^{234}Th and ^{234}Pa). External doses from the short-lived uranium decay products, including beta doses, are covered in *Technical Basis Document for the Weldon Spring Plant – Occupational External Dosimetry* (ORAU 2005a). This section covers internal deposition of uranium.

There were four periods in the history of the WSP:

- Operational: 1957 to 1967
- Transfer to the U.S. Army: 1968 to 1974
- Environmental monitoring: 1975 to 1985
- Remediation: 1985 to 2001

The operational period and the remediation period are of primary concern in relation to internal dosimetry at WSP, with some potential applicability of operations in 1967 before transfer to the U.S. Department of Defense. EEOICPA does not cover the period when the WSP site was under control of the Army, and, therefore, this TBD does not cover that period.

Fuel for the Atomic Age – Completion Report on St. Louis Area Uranium Processing Operations, 1942-1967 (MCW 1967) describes the history and elements of the MCW health and safety programs. Uranium was initially considered primarily a heavy-metal poison, and the radioactivity level was considered low enough so that "small scale, short-term operations would not present a radiation problem." MCW's health and safety program was based on standard industry procedures for handling toxic chemicals. The MCW radiation safety program developed for the Destrehan Street site was applied at the WSP site when the transition between the two facilities was made.

The principal source of internal deposition of radionuclides for both the Destrehan and WSP sites was inhalation of dust generated during the operations, initial cleanup, and maintenance periods. Table 5-1 lists the history of dust-generating operations.

Tables 5-2 and 5-3 summarize the operations at WSP from the more detailed discussion in *Technical Basis Document for Weldon Spring Plant – Site Description* (ORAU 2005b). Ore concentrates containing 60% to 70% yellowcake were sampled on receipt at the facility. Some of the material was repackaged in drums and some sent directly for processing. The concentrates were digested with nitric acid to produce uranium-nitrate solution that was then purified by solvent extraction and denitrated to produce UO_3.

Table 5-1. History of dust-generating activities.[a]

Period	Activity
1957-1966	Operation of the uranium feed materials plant • Uranium concentrates converted to uranium trioxide, uranium tetrafluoride, and uranium metal • Some thorium processing between 1965 and 1966 • Raffinate from processing removed to raffinate pits • 14,500 metric tons of uranium materials received for processing and sampling per year between 1958 and 1964
December 1966	Plant closed • Hopper and process lines emptied • Dust collectors cleaned out
January 1967-?	Site used as interim storage depot for yellowcake later shipped to other plants for refining and processing.
1967	Buildings 103 and 105 transferred from the AEC to the Army for herbicide production
March 1968	Army started decontamination and equipment removal
December 1968	Construction of herbicide facility began; project terminated in early 1969 before renovation was complete.
March 1968-June 1969	Decontamination and equipment removal for Buildings 103 and 105 (see table 5.2 for building operations description) • About 1,000 metric tons rubble removed to the quarry • About 2,000 metric tons scrap moved to Tennessee • About 200 metric tons steel parts moved to Ohio • About 100 metric tons uranium oxide removed from the buildings
1969-1985	Site remained essentially undisturbed
1985	Remediation initiated by DOE – Weldon Spring Site Remedial Action Project (WSSRAP)
October 1986	MK-Ferguson and Jacobs Engineering assumed responsibility for the WSSRAP

a. Adapted from Meshkov (1986) and Lesperance, Siegel, and McKinney (1992).

Table 5-2. Potential internal radionuclide exposure for production buildings.[a]

Bldg. no.	Building description	Building operations	Potential radionuclide exposures
101	Sampling	Sampling of ore concentrates containing 60 – 70% yellowcake Some material repackaged in drums – some sent to Bldg 103 for processing	U-nat dust, Ra-226, Th-230, Po-210 and Pb-210 Rn-222 and its short-lived decay products Th-232 and decay products in 1966
103	Digestion	Materials digested with nitric acid. Uranium bearing solution sent to Bldg 105 for purification. Materials returned for denitration after purification and sent to Bldg 201.	U-nat dust (as yellowcake and UO_3), Ra-226, Th-230, Po-210 and Pb-210 Rn-222 and its short-lived decay products Th-232 and decay products starting November 1963
105	Purification	Materials were purified by solvent extraction and returned to Bldg 103 for denitration.	Wet process but some potential for uranium or thorium (in 1966) dust exposure. Uranium would have been the major internal exposure component but Rn-222 and decay product exposure possible
108	Acid recovery	Recovering and re-concentrating nitric acid	Radon gas and its decay products
201	Green salt (UF_4) plant	Feed from Bldg 103 (after denitration) converted to UF_4.	Potential for uranium exposure as green salt dust or natural thorium (1965-66). No significant Th-230, Ra-226 or decay product exposure.
301	Metals plant	Mg used to convert UF_4 to U metal. Rotary kiln used to convert U metal chips to U_3O_8. U fuel cores produced; acceptable cores shipped to reactor sites.	Potential for uranium exposure as green salt dust and U_3O_8 or natural thorium (1965-66). No significant Th-230, Ra-226 or decay product exposure. Th-232 and decay products starting November 1963
403	Chemical pilot plant	Small-scale chemical processes	U-nat dust, Ra-226, Th-230, Po-210 and Pb-210 Rn-222 and its short-lived decay products Th-232 and decay products starting November 1963
404	Metallurgical pilot plant	Small-scale metallurgical processes	Potential for uranium exposure as green salt dust and U_3O_8 or natural thorium (1965-66) No significant Th-230, Ra-226 or decay product exposure.
407	Analytical and research labs	Small-scale research and analytical work on products and processes.	U-nat dust, Ra-226, Th-230, Po-210 and Pb-210 Rn-222 and its short-lived decay products Th-232 and decay products from 1965 to 1966

a. Adapted from ORAU (2005b).

Table 5-3. Products or intermediates in production buildings.

Building no.	Products or intermediates present
101	Ore concentrates, yellowcake
103	Yellowcake, UO_3, Uranium solution in nitric acid Purified uranium solution Denitrated uranium solution
105	Uranium solution (UO_3) in nitric acid Purified uranium solution
108	Recovered nitric acid
201	Purified, denitrated uranium solution UO_3, UO_2 (intermediate compound) Green salt (UF_4)
301	Green salt (UF_4) Uranium metal U_3O_8

The UO_3 was converted to green salt (UF_4) in Building 201. Green salt was one of the final products of the plant.

Uranium metal was produced using magnesium to convert UF_4 to the metallic form. A rotary kiln was used to convert uranium metal chips to U_3O_8. Uranium fuel cores were produced at WSP and shipped directly to reactor sites.

In contrast to the Destrehan Street facility, WSP did not deal with pitchblende ores. From 1946 to 1955, MCW processed pitchblende ores with concentrations up to 60% uranium (Dupree 1998) along with very high concentrations of ^{226}Ra. There is no indication in the records that WSP ever processed the high-activity concentration pitchblende. Therefore, while inhalation of radon decay products was a potential contributor to dose at the Destrehan facility, it did not add significantly to worker doses at WSP. Dupree (1998) stated the judgment that the uranium doses of the MCW population were considerably larger than doses from exposure to radon.

The contributions to internal doses due to inhalation of uranium-bearing dust were similar in kind, if not in magnitude, for the MCW Destrehan and WSP workers because the operations at the two facilities were essentially the same. In fact, it appears from the worker records that many of the Destrehan workers were transferred to WSP. In general, because the Destrehan facility had accumulated high levels of surface contamination (Dupree 1979a), the early internal doses (1942 to 1958) for MCW workers were potentially higher than for individuals who worked at WSP during the final years (1958 to 1966) of uranium processing by MCW.

5.1.4 Radiation Protection Practices

The MCW radiation safety program in relation to internal doses evolved during the time when uranium ores and concentrates were being processed at the Destrehan Street site in St. Louis. As noted above, the initial concern was with the chemical toxicity of uranium rather than the potential radiation hazard.

As noted in ORAU (2005a), the WSP employed approximately 600 workers during full production, of whom 300 were likely to have handled uranium. The site processed natural uranium, slightly enriched uranium, depleted uranium and, for a short period, natural thorium. The principal radionuclides to

which workers were exposed were uranium isotopes, thorium isotopes, and ^{234}Pa, the short-lived beta-emitting decay product of ^{238}U. Because the site did not process pitchblende ore, ^{226}Ra and its decay products were not present in significant quantities.

Uranium ore concentrates were converted to UO_3, UF_4, and uranium metal at the WSP by methods that included acid digestion, solvent extraction, and conversion to the metallic form by reaction with magnesium. Table 5-2 lists the specific buildings in which these processes occurred along with the likely contaminants in air in those locations.

MCW's position, even as late as 1965 and as reflected in *Summary of Health Protection Practices* (MCW 1965), was, "There are no noteworthy health risks from radiation at the Weldon Spring Feed Material Plant." The conclusion was based on the fact that the natural uranium feed materials processed at WSP were "essentially free of radioactive daughters which might cause chronic radiation exposure to be higher than AEC limits." However, MCW (1965) also stated:

> *It is the policy of the Weldon Spring Plant to prevent personal injury and to prevent harmful exposure of personnel to chemical irritants, to chemical toxicants, to ionizing radiation, or to any work conditions which may cause illness, impair health, or reduce the effectiveness of employees; to provide suitable monitoring and control programs; to obtain the cooperation of each employee in maintaining effective control programs; to inform individuals if measured radiation exposure exceeds AEC guide levels.*

No descriptions of internal dosimetry programs during the environmental monitoring period (1975 to 1984) have been found.

The remediation program initiated in 1985 involved demolition of the buildings involved in uranium or thorium processing, removal of contaminated materials from the WSQ, and stabilization and solidification of contaminated sludges. The internal dosimetry programs for the WSQ and WSRP were somewhat more rigorous than the program for the buildings that comprised the WSCP because the potential for exposure to thorium and radium was not as high in the WSCP. The high-grade pitchblende ores processed at the Destrehan Street facility were not present at the WSCP. Thorium was processed at WSP for a limited period. The WSP buildings involved in processing thorium were subject to a similar, slightly more rigorous, internal dosimetry program as the WSQ and WSRP.

The internal dosimetry programs during site remediation are described in detail in a series of revisions to the *Internal Dosimetry Program Technical Basis Manual: Weldon Spring Site Remedial Action Project* from 1991 to 2001 (the WSSRAP Technical Basis Manuals; DOE 1991, 1994, 1997, 1998a,b,c, 2000a, 2001). The basic elements of the internal dosimetry program were urine and fecal bioassay, *in vivo* lung counting, and air sampling. Section 5.3 describes the urine and fecal bioassay programs, Section 5.4 discusses *in vivo* lung counting, and Section 5.5 describes air-monitoring practices.

5.2 SOURCE TERM

WSP processed uranium and thorium from feed materials to metal and intermediate products. The primary feed material was natural uranium in the form of yellowcake. WSP also processed depleted uranium and slightly enriched (up to 1%) uranium as well as natural thorium. Table 2-4 in ORAU (2005a) gives the annual (fiscal year) mass receipts of each of these feed materials. Table 5-4 lists the quantity and percent of each type of feed material.

Table 5-4. Mass and percent of feed materials processed.

Material	Mass (kg)	Percent of total mass
Natural U	122,015,944	98.43
Depleted U	167,823	0.14
Slightly enriched U	842,585	0.68
Natural thorium	941,347	0.76
Total	123,967,712	

5.2.1 Isotopic Composition of Uranium

The isotopic composition and the factors to convert uranium mass to activity are necessary for intake and dose assessments. No site-specific isotopic data have been discovered for WSP. It is reasonable to assume that the composition of the natural and depleted uranium feed materials at WSP were the same as the default compositions for the DOE complex. Therefore, the DR should use the default values in the IMBA program for natural and depleted uranium.

For slightly enriched uranium, it is reasonable to assume that the composition of 1% enriched uranium in the *Technical Basis Document for the Fernald Environment Management Project – Occupational Internal Dosimetry* (ORAU 2004a, Table 5-3) is applicable to slightly enriched uranium at WSP. Table 5-5 of this document shows those values.

Table 5-5. Isotopic uranium data for 1% enriched uranium.

Isotopes	Mass %	Relative activity (pCi/µg)	Activity %	Total pCi/µg
U-234	0.01	0.62	63.7	0.973
U-235	1.0	0.02	2.1	
U-236[a]	0	0	0	
U-238	98.99	0.333	34.2	

a. U-236 contributes less than 1% activity in 1% enriched uranium.

Although uranium with enrichments of less than 1% might have been processed at WSP, it is claimant-favorable to assume 1% enrichment for all slightly enriched uranium at WSP.

5.2.2 Uranium Decay Products

The materials handled at WSP were uranium concentrates and, to some extent, natural thorium. The short-lived decay products of ^{238}U, which are ^{234}Th (24-d half-life) and ^{234m}Pa (1.175-min half-life), would have built into equilibrium before the material was handled. Thorium-234 and ^{234m}Pa emit beta particles. The dose from inhaled ^{234}Th is included in the dose from ^{238}U as it builds into equilibrium in the body in a relatively short period of time (less than eight months). The ^{234m}Pa beta is a high-energy beta and contributes to the external dose but, due to its short half-life, does not in itself contribute to internal dose.

The buildup of ^{230}Th from ^{234}U is of little consequence because it has an 80,000-yr half-life. Assuming the age of the material processed to be no greater than 20 yr, the maximum activity concentration of ^{230}Th due to decay of ^{234}U would be 0.02% of the ^{234}U activity concentration. Small amounts of enriched uranium were apparently processed at the WSP site. That would increase the concentration of ^{234}U in natural uranium by a significant amount. Assuming that increase could be as much as a factor of 100, the ^{230}Th concentration due to buildup would still be only a very small fraction of the uranium concentration. The concentration of ^{226}Ra due to buildup would be even smaller because it has a 1,600-yr half-life and would not have had time to build up significantly from ^{230}Th.

The primary source of decay products (^{230}Th and ^{226}Ra) for the materials processed at WSP would, on average, be the residuals in the uranium concentrates. These concentrations were not considered significant in the design of the radiation protection program at WSP. The *Final Generic Environmental Impact Statement on Uranium Milling* (FGEIS) states that the upper range of values for ^{230}Th and ^{226}Ra in yellowcake product, based on published reports from the early 1960s, were 5% of the ^{238}U activity and 0.2% of the ^{238}U activity, respectively (NRC 1980). However, later measurements indicated that these values, especially for ^{230}Th, were overly conservative. The values assumed for the "model mill" that was the subject of the FGEIS, were 0.5% for ^{230}Th and 0.1% for ^{226}Ra. The *Technical Basis Document for the Weldon Spring Plant – Occupational Environmental Dose* (ORAU 2005c) cites values of 5% for ^{230}Th and 1% for ^{226}Ra based on information from Meshkov et al. (1986). The values from ORAU 2005c are claimant-favorable, but very conservative. The shorter lived decay products, ^{210}Pb and ^{210}Po should be assumed to be present at the same activity as the ^{226}Ra in uranium concentrates.

As noted in *Technical Basis Document for the Weldon Spring Plant – Site Description* (ORAU 2005b), four raffinate pits were constructed between 1958 and 1964 to contain process wastes from the WSP. The radionuclide of most concern in the raffinate pits was ^{230}Th due to its high activity concentration and its radiotoxicity. The average and maximum activity concentrations are given in the *Internal Dosimetry Program Technical Basis Manual, Rev. 0* for the WSSRAP (DOE 1991). Use of the maximum activity concentrations provides a claimant-favorable basis for estimating intake. The maximum ^{230}Th activity concentration in Pit 1 was the highest at 34000 pCi/g. The estimated maximum activity concentrations in Pit 1 for ^{238}U, ^{226}Ra, and ^{232}Th were 600 pCi/g, 1700 pCi/g, and 110 pCi/g, respectively. The maximum activity concentration of ^{232}Th in Pit 4 was the highest with a maximum value of 1400 pCi/g (DOE 1991). These concentrations provide a bounding estimate that can be used to calculate claimant-favorable intakes for exposure at the WSRP.

The AEC acquired the WSQ in 1958 for deposition of thorium residues as well as uranium- and radium-contaminated building rubble, equipment and soils from the Destrehan Street site. These materials were deposited to the WSQ between 1959 and 1966. The estimated maximum radionuclide concentrations in the WSQ materials were 2400 pCi/g, 6800 pCi/g, and 2780 pCi/g for ^{238}U, ^{230}Th, and ^{226}Ra respectively (DOE 1991). While the average concentrations were generally lower by at least an order of magnitude, the maximum concentrations provide a bounding estimate that can be used to calculate claimant-favorable intakes from exposure at the WSQ.

5.2.3 Natural Thorium and Decay Products

In contrast to the buildup of decay products of ^{238}U, the conservative (claimant-favorable) assumption for natural thorium (^{232}Th) is that the decay products have built up to equilibrium. Depending on the time since separation, the ^{228}Ra (6.7-yr half-life) and ^{228}Th (1.9-yr half-life) would be significant if not complete. Radium-228 is a beta emitter that decays to ^{228}Th through ^{228}Ac. Thorium-228 decays by alpha emission to ^{224}Ra (3.66-d half-life). Therefore, the claimant-favorable assumption for assessing dose from ^{232}Th is that all decay products should be assumed to be in equilibrium. The *Technical Basis Document for the Weldon Spring Plant – Occupational Environmental Dose* (ORAU 2005c) cites a DOE document (DOE 1986) in support of the assumption that ^{232}Th was present during the operational period at an activity less than 1% of the natural uranium (ORAU 2005c)

As noted in Section 5.2.2, the ^{232}Th concentrations in the WSRP were relatively small compared to the ^{230}Th and ^{226}Ra concentrations, thus would contribute only a very small fraction of the potential dose. In contrast, the estimated maximum ^{232}Th concentration in materials deposited in the WSQ was 4200 pCi/g (DOE 1991). This concentration represents a claimant-favorable bounding estimate for potential exposures to natural thorium at the WSQ.

5.2.4 Recycled Uranium

The extent of the processing of recycled uranium at WSP is not well known (ORAU 2005b, Section 2.2.3). The DR should make the claimant-favorable assumption that all of the uranium processed at WSP after 1961 was recycled uranium. This assumption is consistent with that in *Ohio Field Office Recycled Uranium Recovery Report* (DOE 2000b), which assumed that all uranium receipts at WSP after 1961 were recycled uranium in lieu of better information.

Contaminant radionuclides in recycled uranium that could be dosimetrically significant are plutonium (assume ^{239}Pu), neptunium (^{237}Np), and technetium (^{99}Tc). Site-specific data for the mass fractions of these contaminants are not available. The DR should consider the factors in Table 5-11 of *Technical Basis Document for the Fernald Environmental Management Project – Occupational Internal Dose (ORAU 2004a).* These factors, when multiplied by the assessed uranium gram-value intake, result in the activity per gram of uranium of the contaminants at the levels of 100 ppb ^{239}Pu, 3,500 ppb ^{237}Np, and 9,000 ppb ^{99}Tc.

5.2.5 Solubility Classification and Absorption Type

The WSP handled uranium in several different forms. These forms are listed in Table 5-2 along with the facility location (building) in which they were most likely to have contributed significantly to a worker's uranium intake.

Feed materials likely to have been received at the WSP site came primarily from conventional uranium mills and as vanadium-milling wastes in the form of yellowcake (Meshkov 1986). The specific uranium compounds received at the WSP have not been identified. However, according to the Final Generic Environmental Impact Statement on Uranium Mills, "the yellowcake product from an acid leaching plant is a mixture of chemical complexes: diuranates, hydrated oxides, basic uranyl sulfate and other ions" depending on the drying temperature (NRC 1980).

The WSP remediation contractor collected samples of materials from the areas in WSCP buildings with the highest levels of contamination (DOE 2001). The samples were tested for lung solubility class using simulated lung fluid leachate tests. Based on those test results, the contractor assumed specific fractions of solubility classes for contaminants in the building sampled. Table 5-6 lists those fractions for most of the WSCP buildings. The solubility classes in that study were denoted using the ICRP Publication 30 designations of D, W, and Y (ICRP 1979). These classes can be assumed to the current ICRP solubility classes F, M, and S, respectively (ICRP 1995).

Table 5-6. Solubility class summary (DOE 2001).

	U-234			U-235			U-238			Th-232		Th-230		Th-228	
Area	D	W	Y	D	W	Y	D	W	Y	W	Y	W	Y	W	Y
Bldg 101	0.41	-	0.59	0.28	-	0.72	0.41	-	0.59	0.20	0.80	-	1.0	0.29	0.71
Bldg 103	0.20	0.50	0.30	0.20	0.44	0.36	0.75	0.25	-	-	1.0	-	1.0	-	1.0
Bldg 105	0.20	0.50	0.30	0.20	0.44	0.36	0.75	0.25	-	-	1.0	-	1.0	-	1.0
Bldg 108	0.19	0.20	0.61	0.14	0.47	0.39	0.19	0.20	0.61	-	1.0	-	1.0	-	1.0
Bldg 201	0.44	-	0.56	0.51	-	0.49	0.42	-	0.58	-	1.0	-	1.0	-	1.0
Bldg 301	0.12	-	0.88	0.09	-	0.91	0.12	-	0.88	0.03	0.97	0.09	0.91	0.03	0.97
Bldg 403	0.19	0.20	0.61	0.14	0.47	0.39	0.19	0.20	0.61	-	1.0	-	1.0	-	1.0
Bldg 406	1.0	-	-	1.0	-	-	1.0	-	-	0.35	0.65	0.68	0.32	0.65	0.35
Bldg 408	1.0	-	-	1.0	-	-	1.0	-	-	0.35	0.65	0.68	0.32	0.65	0.35
Bldg 410	1.0	-	-	1.0	-	-	1.0	-	-	0.35	0.65	0.68	0.32	0.65	0.35
Bldg 417	1.0	-	-	1.0	-	-	1.0	-	-	0.35	0.65	0.68	0.32	0.65	0.35
Pit 3	1.0			1.0	-	-	1.0	-	-	-	1.0	-	1.0	-	1.0
Q-Bench	0.54	0.12		0.46	0.07	0.47	0.49	0.13	0.38	-	1.0	-	1.0	-	1.0
Q-Sump	0.61			0.52	0.47	0.48	0.58	-	0.42	0.10	0.90	-	1.0	0.10	0.90

In general, uranium metal dust in Building 404 (Metals Pilot Plant) would most likely have been insoluble or type S. While the residues from this building were not analyzed in the solubility study, the solubility class for uranium in residues from Building 301 (Metals Plant) was found to be approximately 90% class Y and 10% class D. The uranium solubility classes for Buildings 406, 408, 410, and 417 were listed as 100% Class D. These buildings were used for administrative and maintenance purposes, and a single composite sample was analyzed to represent the solubility class for all non-production buildings. Given the uncertainty and lack of specific data for such areas of the WSCP, the most claimant-favorable solubility type should be used.

The urine bioassay data distribution shows that Friday afternoon samples averaged nearly two times the uranium concentration as Monday morning samples. This indicates relatively rapid clearance of the previous week's intake. This lends some support to the assumption that at least a portion of the uranium in air at WSP was soluble.

In general, the solubility type to be selected in dose reconstruction for UF_4 exposures should be based on the cancer site.

5.2.6 Particle Size

The deposition in the lung of dust particles containing radioactive materials depends on particle size and breathing rate. The dose to the lung and other organs from radionuclides in the body depends on the solubility of the deposited particles. The excretion rate through the kidneys depends on both solubility and particle size.

DOE assumed a particle size of 1 µm aerodynamic median activity diameter (AMAD) in developing derived air concentrations (DACs) for the WSSRAP. ICRP Publication 66 recommends a default value of 5 µm AMAD for workplace exposures (ICRP 1994b, paragraph 181). Eisenbud (1975) suggests a likely particle size of 5 to 6 µm AMAD based on a mass median diameter of 2 µm and a particle density of 9 to 10 g/cm^3. Sanders (1975) studied particle sizes at the Oak Ridge Y-12 plant and found that the particle sizes in various locations in the facility ranged from 1 to 3 µm mass median diameter, with the bulk of the samples between 2 and 3 µm. The MILDOS program developed in 1981 to estimate doses to members of the public from particulate emissions from uranium mills uses a particle diameter of 1 µm and a density of 8.9 g/cm^3 for emissions from yellowcake processing (NRC 1981).

Lacking specific information on the particle size for airborne uranium at WSP, use of 5 µm AMAD appears to be reasonable. That assumption is supported by the literature.

5.3 *IN VITRO* MEASUREMENTS

5.3.1 Operational Period (1957 to 1967)

5.3.1.1 Uranium

Urine bioassay was the primary method of determining uranium intakes during the production phase. The bioassay program was set up in accordance with the general health physics practices of the period. Grab (single void) urine samples were collected and analyzed for uranium by photofluorimetric analysis. Results of the photofluorimetric analysis were reported as the mass of uranium in milligrams, or sometimes micrograms, per liter of urine.

5.3.1.1.1 Routine Sampling Program

MCW (1965) described the uranium urine bioassay program as follows:

> *The routine sampling program seeks to have one or more persons from each operational group in the plant sample[d] each week. When a person represents his group in the sample, he is asked to give samples on (1) Monday a.m., (2) Friday p.m., and (3) Monday a.m. The Monday sample tends to show the amount semi-fixed in the body, the Friday sample reflects the daily uptake. The sample from each person is analyzed separately and entered in his summary.*
>
> *Each exposed person is scheduled three or more times per year, more frequently if there is reason to suspect increased exposure. The rotation of group representation tends to show the average level and variation within each plant area. Unexposed persons are scheduled less frequently to provide a control base.*

A review of urine data in worker files available up to November 2004 indicates that this program was implemented in about October 1959. From 1957 to mid-May 1959 urine sampling was apparently conducted at random times during the week. From mid-May 1959 through September 1959 sampling occurred primarily on Fridays. Starting in October 1959 the routine sampling program described above appears to have been implemented and continued through 1966.

Workers also submitted urine samples as part of the hiring and termination processes.

5.3.1.1.2 Special Sampling Program

A repeat sample was required if the result of the Monday morning sample was greater than 100 µg/L or if the result of the Friday afternoon sample was greater than 200 µg/L.

Special urine samples were required for known or suspected significant intakes. The DR can identify the results of these special samples in worker files either by the code S or by handwritten notes on the original urine data cards.

5.3.1.1.3 Data Reporting Levels and Minimum Detectable Amounts

No information has been found about the details and quality assurance of the photofluorimetric system and data analysis.

There were apparently no uranium urine results censored or reported as less than the detection level because the recorded data reflect continuous increments of 0.001 mg/L starting at 0.000 mg/L and no less-than values have been observed in the files. It is not known whether a blank was subtracted. If so, negative results were reported as 0.000 mg/L.

Because a site-specific value of the minimum detectable amount (MDA) is not available for WSP, a surrogate value based on reported MDAs for photofluorimetric systems at other facilities in the 1960s was considered. A range of values of 0.001 to 0.014 mg/L has been cited for detection levels of unknown pedigree. An MDA value of 0.008 mg/L was determined from original urinalysis data logs at Rocky Flats (ORAU 2004b) and was based on modern MDA concepts (HPS 1996). The Rocky Flats MDA value included the contribution of a blank subtraction and was based on a 100λ (100 microliter) aliquot from a 24-hr urine sample. This MDA value has a known pedigree and is recommended as the surrogate MDA for WSP, although there may be some differences in methods (e.g., grab sample

versus 24-hr sample, unknown blank subtraction versus blank subtraction, and unknown volume of the aliquot versus 100λ (100 microliters)).

An estimate of the uranium MDA can be derived from Dupree (1979b), who cites an average value of 0.002 ± 0.002 mg/L for "people off the street." Based on the consideration that the cited standard deviation represents the process standard deviation divided by the calibration factor, the estimated MDA is 3.3 times 0.002 mg/L, which is 0.007 mg/L. This value supports the use of 0.008 mg/L as a reasonable surrogate MDA for WSP uranium urine data.

5.3.1.1.4 Interferences and Uncertainties

It is not known whether the WSP uranium urine data were adjusted for excretions of environmental sources of uranium. It is likely that the data were not adjusted. Measurements more sensitive than the Dupree (1979b) value of 0.002 ± 0.002 mg/L made during the remediation period on persons in the WSP vicinity not occupationally exposed to uranium indicated a geometric mean of 0.05205 µg/L (0.00005205 mg/L), "with variation as high as 0.3016 µg uranium per liter of urine at two standard deviations (e.g., 94.5% of the population) (DOE 2001)." Because these values are very low, DRs should disregard this source of interference and use the WSP uranium urine data as recorded.

Possible contamination of a urine sample from uranium on the hands or clothing cannot be ruled out, especially for grab samples after work. Resampling for results over the action levels (Section 5.3.1.1.2) would catch cases of excessive contamination. There are numerous uncertainties generic to fluorimetric uranium measurements. Sample-specific uncertainties were not recorded. There is no reason to believe that the quality of WSP measurements was significantly different from the quality at other facilities of that era.

5.3.1.1.5 Reporting Formats and Codes

The DR can expect to find one to three reports of uranium urine data in a worker's files:

- A photocopy of the original, handwritten urine data cards
- A computer printout, Uranium Urine (MCWURWS)
- A computer printout, Uranium Urine (MCWURDES)

Many WSP workers were assigned to the St. Louis facility before being assigned to WSP. All of the urine data reports contain data from both facilities because WSP was a division of MCW. If no other information is available, DRs should assume that uranium urine data starting in 1957 is associated with WSP. All three reports seem to report the same data, but sometimes in different manners, and all units are milligrams per liter.

Clock No. on the original data cards and Clock Badge on the computer printouts is the worker's employee number. DRs should examine the Clock No. on the photocopies of the original data cards to ensure that the numbers are consistent with the worker because investigators have found discrepancies in the record. The sample date is recorded and should be interpreted as the date of excretion of the grab sample. On some original data cards, this date is listed in the DUE column.

Dept. (on the original data cards) or DEPTJOB (on the computer printouts) is either a cost center number, a job or department title, or sometimes a mixture of the two. Table 5-7 lists the cost center codes. The DR should be aware that the cost center codes for the work groups in the production facilities and services groups (except for Maintenance) changed in early 1963.

Table 5-7. Cost center codes for workers.

Work group	Cost center code	
	To early 1963	Starting 1963
General Operations		
General Engineering	110	(a)
Chemical Technology	120	(a)
Metal Technology	121	(a)
Plant Services	140	(a)
WSP		
Administration	300	(a)
Production Facilities		
Sampling Plant		
Sampling	310	110
Repackaging	321	110
Refinery		
Digest – Raffinate	320	120
Pot Room	320	120
Extraction	320	120
Green Salt Plant		
Green Salt	330	150
Metal Plant		
Dingot Reduction	360, 340	180
Extrusion, General	359	200
Extrusion, Gamma (Dingot Extrusion)	361, 351	210
Extrusion, Alpha (Dingot to Rods)	364, 352	220
Scrap Activity	358	280
Vacuum Outgassing	363	(a)
Core Fabrication	362, 365 to 369	250
Services Groups or Facilities		
Maintenance	370	370
Storeroom, Receiving, & Shipping	372	530
Disposal of Construction Inventory	373	(a)
Instrument Shop	374	(a)
Boiler House	376	(a)
Water Plant	378	(a)
Warehouse	380	550
Engineering	390	560
Plant Protection	392	510
Laundry	394	(a)
Custodians	396	520
Occupancy, Labor	398	(a)
Technical Division		
Administrative	400	(b)
Metallurgical Development (Building 404)	410	(b)
Special Projects	460	460
Process Development (Building 403)	420, 450	(b)
Scrap Plant	No Code	290
Research Laboratory	430	(b)
Analytical Laboratory	440	(b)
Administration Division		
Uranium Division Management	600	600
Salaried Personnel Administration	605	(b)
Accounting	610	610
Purchasing	620	(b)
Motor Pool	621	(b)
Administrative Services	630	(b)
Production Control	640	640
Health and Safety	650	(b)
General Division Expenses	660	(b)
Data Processing	670	670
Industrial Relations	680	(b)
Cafeteria	685	(b)

a. Unknown – likely discontinued or merged with another cost center
b. Unknown – likely the same as the code to early 1963

The sample result (in milligrams per liter) is reported under columns headed by SCHEDULED, RESULTS, or no heading on the original data cards, and under the heading MGUPERL on the computer printouts. On the original data cards, the sample result is frequently followed by two dashes and a number, which has been assumed to be the pH of the urine sample.

Following the sample result on the original data cards or in the column headed by SAMTYPE on the computer printouts, an asterisk indicates a Friday afternoon sample. However, not all Friday samples are flagged by the asterisk. If the sample day is important, the DR could wish to determine the actual day using the Microsoft® Excel WEEKDAY function, which converts a date to a day-of-the-week code where 1 is Sunday, 2 is Monday, and so forth. Written on the original data cards after the result or under SAMTYPE on the computer printouts are notations that indicate pre-employment samples (code P), termination samples (code T), or special samples (code S).

5.3.1.1.6 Work Group Data

Urine bioassays were performed routinely as described in Section 5.3.1.1.1. Urine samples were obtained weekly from representative individuals in areas of WSP where uranium was handled. The data from the representative individuals were intended to be used to assess the intake by coworkers so that the work group was continuously monitored. Individual urine bioassay results supplemented by contemporaneous data from coworkers could provide the best measure of that person's uranium intake because the sampling for an individual worker could have occurred during quiescent operational periods.

Because most of the work group urine data summaries have not been discovered, the data have been recreated. Approximately 28,000 urine bioassay results were recorded during the operational period (1958 to 1966). Tables 5-8 to 5-17 provide median, 95th-percentile, and maximum concentrations for routine urine bioassay samples by year.

The data were analyzed by major work location, cost center or job description, and sample day (Monday or Friday). In some cases cost centers were combined to increase the number of individual analyses. The data set for 1958 includes a mixture of WSP and Destrehan Street workers and was coded in the original records by job description or work location rather than cost center. In cases where there were five or fewer records, the tables contain only the maximum urine bioassay result.

Table 5-8. Composite uranium urine data summary.

Year	Type of analysis	Number of records	Urine uranium concentration (mg/L)		
			Median	95th percentile	Maximum
1958	All	1,872	0.005	0.025	0.203[a]
	Routine	1,714	0.005	0.025	0.203
	Routine - Monday	321	0.004	0.021	0.047
	Routine - Friday	158	0.006	0.023	0.069
1959	All	2,285	0.005	0.024	0.339
	Routine	2,090	0.006	0.026	0.339
	Routine - Monday	1,124	0.004	0.017	0.041
	Routine - Friday	522	0.009	0.048	0.339
1960	All	4,396	0.012	0.038	0.759
	Routine	4,246	0.012	0.040	0.759
	Routine - Monday	2,602	0.010	0.026	0.088
	Routine - Friday	1,347	0.018	0.068	0.759
1961	All	4,184	0.011	0.036	0.344
	Routine	4,077	0.011	0.036	0.344
	Routine - Monday	2,608	0.010	0.025	0.062
	Routine - Friday	1,279	0.018	0.050	0.344
1962	All	3,083	0.010	0.032	0.700
	Routine	2,954	0.007	0.024	0.700
	Routine - Monday	1,847	0.008	0.020	0.064
	Routine - Friday	946	0.013	0.044	0.700
1963	All	3,481	0.014	0.042	0.340
	Routine	3,358	0.014	0.041	0.336
	Routine - Monday	1,922	0.014	0.029	0.336
	Routine - Friday	1,074	0.018	0.056	0.258
1964	All	3,476	0.012	0.052	0.626
	Routine	3,264	0.012	0.048	0.626
	Routine - Monday	1,767	0.012	0.029	0.282
	Routine – Friday	1,049	0.014	0.060	0.626
1965	All	2,980	0.010	0.036	0.812
	Routine	2,804	0.009	0.032	0.812
	Routine - Monday	1,460	0.009	0.027	0.347
	Routine – Friday	869	0.010	0.045	0.318
1966	All	2,145	0.006	0.029	0.459
	Routine	1,680	0.007	0.028	0.459
	Routine - Monday	882	0.008	0.025	0.066
	Routine – Friday	526	0.007	0.039	0.459

a. A recorded value of 7.8 mg/L was deleted from the data set as an outlier. This does not affect the median or 95th-percentile value.

Table 5-9. Uranium urine data summary by cost center for 1958.

Cost center	All samples (mg/L)			No. of records
	Median	95th percentile	Maximum	
300-370	0.002	0.010	0.012	16
400-440	0.000	0.011	0.011	7
600-670	0.003	0.010	0.012	29
Acct	0.003	0.006	0.010	17
Adm	0.004	0.015	0.017	16
Aec	0.003	0.010	0.010	9
Anal Lab	0.004	0.012	0.039	105
Boiler	0.002	0.008	0.008	14
Chem Op	0.004	0.030	0.061	24
Decon	0.003	0.006	0.006	9
Dx	0.012	0.029	0.040	25
Engineering	0.003	0.014	0.020	33
Furnace	0.008	0.019	0.022	24
Gr Salt	0.005	0.018	0.025	30
Guard	0.004	0.013	0.014	41
H&S	0.004	0.007	0.016	28
Instruments	0.002	0.018	0.022	37
Lab	0.004	0.018	0.018	27
Laundry	0.005	0.012	0.024	15
Maint	0.006	0.020	0.098	79
Metals	0.006	0.015	0.101	94
Mtns	0.006	0.025	0.064	137
Office	0.002	0.008	0.016	97
Pilot Plant	0.005	0.014	0.028	16
Pl4	0.010	0.026	0.078	24
Pl6	0.012	0.045	0.048	58[a]
Pl7	0.016	0.053	0.062	54
Porter	0.005	0.016	0.020	25
Prod Dev.	0.004	0.010	0.011	25
Refinery	0.009	0.041	0.203	127
Research Lab	0.004	0.012	0.022	49
Sampling	0.011	0.040	0.049	31
Stores	0.004	0.009	0.029	16
Warehouse	0.004	0.017	0.051	48

One outlier at 7.8 mg/L was not included.

Table 5-10. Uranium urine data summary by cost center for 1959.

Cost center	Monday samples (mg/L)			Number of records	Friday samples (mg/L)			Number of records
	Median	95th percentile	Maximum		Median	95th percentile	Maximum	
110-140	0.004	0.010	0.010	14	N/A[a]	N/A	0.012	2
300-310	0.002	0.015	0.022	35	0.020	0.048	0.048	12
320	0.006	0.016	0.023	24	0.014	0.088	0.104	40
321-340	0.008	0.018	0.020	25	0.017	0.043	0.067	18
359-369	0.005	0.016	0.034	60	0.012	0.036	0.049	47
370	0.009	0.016	0.019	40	0.014	0.058	0.206	25
372	0.003	0.012	0.017	17	0.004	0.023	0.023	8
374	0.006	0.015	0.018	27	0.006	0.024	0.024	13
376-379	0.002	0.027	0.039	26	0.003	0.035	0.035	13
380	0.004	0.019	0.041	53	0.004	0.071	0.094	32
390	0.003	0.007	0.012	40	N/A	N/A	0.003	3
392	0.005	0.016	0.028	38	0.003	0.023	0.026	18
394-396	0.006	0.019	0.019	30	0.006	0.023	0.023	14
400-430	0.002	0.013	0.020	24	N/A	N/A	0.041	5
440	0.003	0.014	0.020	73	0.004	0.012	0.015	16
450	0.004	0.012	0.016	20	0.018	0.079	0.080	15
460	0.005	0.017	0.020	21	0.038	0.156	0.156	14
500-640	0.002	0.014	0.024	79	N/A	N/A	0.008	5
650	0.003	0.009	0.011	28	N/A	N/A	0.007	3
670-688	0.006	0.013	0.023	27	N/A	N/A	0.011	1
320 Ref	0.005	0.018	0.028	26	0.020	0.057	0.059	22
330 Gr Salt	0.012	0.020	0.020	20	0.017	0.023	0.023	11
370 Misc	0.006	0.015	0.019	16	0.014	0.051	0.051	12
370 Mtns	0.006	0.019	0.035	164	0.011	0.048	0.083	94
370 Mtns Misc	0.006	0.012	0.014	42	0.011	0.025	0.044	24
Office	0.002	0.009	0.018	17	0.006	0.009	0.009	8
Other	0.003	0.012	0.015	66	0.004	0.022	0.022	28

a. N/A = not applicable (fewer than five records)

Table 5-11. Uranium urine data summary by cost center for 1960.

Cost center	Monday samples (mg/L)			Number of records	Friday samples (mg/L)			Number of records
	Median	95th percentile	Maximum		Median	95th percentile	Maximum	
110	0.001	0.019	0.020	18	N/A[a]	N/A	0.013	1
300	0.007	0.018	0.019	32	N/A	N/A	N/A	0
310	0.015	0.026	0.038	39	0.033	0.071	0.085	29
320	0.012	0.026	0.060	204	0.019	0.062	0.110	129
321-340	0.014	0.025	0.032	76	0.026	0.055	0.080	57
359	0.013	0.022	0.029	41	0.025	0.069	0.161	22
360-369	0.013	0.028	0.061	105	0.016	0.058	0.092	68
370	0.012	0.028	0.064	106	0.015	0.053	0.162	58
378-379	0.012	0.027	0.027	14	0.015	0.082	0.082	7
380-398	0.008	0.020	0.028	120	0.012	0.041	0.088	34
400-430	0.007	0.030	0.071	38	0.018	0.050	0.050	9
450	0.010	0.027	0.058	91	0.027	0.092	0.161	39
460	0.013	0.029	0.060	64	0.026	0.060	0.078	37
600-680	0.007	0.018	0.028	114	N/A	N/A	0.017	1
320 Ref	0.012	0.029	0.079	155	0.032	0.126	0.216	107
330 Gr Salt	0.013	0.029	0.030	43	0.022	0.070	0.092	29
360 Metals	0.013	0.026	0.029	29	0.017	0.036	0.058	24
361 Conv Ext	0.021	0.058	0.058	14	0.036	0.075	0.075	7
362 Cons Fab	0.010	0.021	0.029	18	0.014	0.023	0.030	16
370 Misc	0.010	0.025	0.037	63	0.014	0.064	0.084	36
370 Mtns	0.010	0.026	0.038	375	0.017	0.064	0.228	256
370 Mtns Elec	0.010	0.024	0.028	52	0.013	0.063	0.076	34
370 Mtns Mech	0.010	0.014	0.014	11	0.008	0.012	0.012	6
370 Mtns Misc	0.012	0.028	0.036	42	0.017	0.036	0.038	31
370 Mtns Pipe	0.012	0.030	0.060	31	0.018	0.064	0.069	20
370 Painter	0.015	0.027	0.032	16	0.013	0.162	0.162	11
372 Stores	0.010	0.024	0.032	38	0.012	0.027	0.029	20
374 Inst	0.010	0.024	0.032	61	0.012	0.036	0.060	38
376 Boiler	0.005	0.018	0.020	26	0.006	0.024	0.024	7
380 Warehouse	0.010	0.026	0.034	115	0.012	0.048	0.067	73
392 Plant Pr	0.006	0.017	0.021	41	0.011	0.030	0.030	13
396 Custodian	0.014	0.027	0.088	48	0.014	0.081	0.759	33
440 Anal Lab	0.010	0.021	0.036	92	N/A	N/A	N/A	0
650 H&S	0.007	0.017	0.019	31	NA	NA	0.003	1
Other	0.010	0.027	0.079	124	0.018	0.066	0.266	45

a. N/A = not applicable (five or fewer records)

Table 5-12. Uranium urine data summary by cost center for 1961.

Cost center	Monday samples (mg/L)			Number of records	Friday Samples (mg/L)			Number of records
	Median	95th percentile	Maximum		Median	95th percentile	Maximum	
110-300	0.007	0.015	0.024	61	0.019	0.036	0.036	8
310	0.016	0.028	0.038	35	0.031	0.062	0.064	25
320	0.011	0.027	0.035	205	0.021	0.055	0.140	127
321-350	0.014	0.034	0.052	86	0.025	0.054	0.066	58
359	0.012	0.034	0.039	44	0.022	0.050	0.052	28
360-369	0.011	0.028	0.049	154	0.021	0.040	0.058	83
370	0.011	0.026	0.062	128	0.018	0.048	0.344	78
380	0.009	0.021	0.029	30	0.012	0.030	0.032	25
390	0.005	0.013	0.022	61	N/A[a]	N/A	N/A	0
392-398	0.008	0.022	0.039	107	0.011	0.037	0.063	79
400-460	0.011	0.026	0.050	200	0.020	0.050	0.118	66
600-680	0.007	0.019	0.027	183	N/A	N/A	0.013	3
320 Ref	0.012	0.026	0.040	154	0.022	0.081	0.127	93
330 Gr Salt	0.010	0.024	0.030	49	0.024	0.054	0.055	27
370 Mill	0.018	0.025	0.025	10	0.019	0.031	0.031	6
370 Misc	0.009	0.022	0.028	57	0.015	0.064	0.080	39
370 Mtns	0.011	0.025	0.042	341	0.018	0.042	0.058	223
370 Mtns Elec	0.009	0.022	0.034	42	0.019	0.032	0.052	24
370 Mtns Misc	0.010	0.025	0.029	86	0.016	0.044	0.050	54
372 Stores	0.010	0.023	0.027	33	0.010	0.026	0.027	22
374 Inst. Shop	0.008	0.021	0.024	45	0.014	0.040	0.050	28
376 Boiler	0.010	0.020	0.021	26	0.008	0.010	0.010	7
380 Warehouse	0.009	0.022	0.026	105	0.011	0.032	0.036	66
392 Plant Pr	0.009	0.018	0.021	48	0.008	0.014	0.015	15
440 Anal Lab	0.009	0.019	0.029	96	N/A	N/A	N/A	0
Other	0.008	0.024	0.035	123	0.021	0.061	0.094	43

a. N/A = not applicable (five or fewer records).

Table 5-13. Uranium urine data summary by cost center for 1962.

Cost center	Monday samples (mg/L)			Number of records	Friday samples (mg/L)			Number of records
	Median	95th percentile	Maximum		Median	95th percentile	Maximum	
120-290	0.009	0.020	0.035	17	0.028	0.071	0.071	11
300-310	0.009	0.021	0.028	63	0.032	0.064	0.090	22
320	0.009	0.022	0.040	138	0.017	0.048	0.700	85
321	0.014	0.034	0.034	11	0.026	0.141	0.141	14
340	0.010	0.027	0.038	34	0.018	0.046	0.046	20
359	0.008	0.020	0.020	29	0.018	0.035	0.036	20
361-369	0.010	0.020	0.024	43	0.014	0.040	0.110	31
360-369	0.009	0.018	0.018	32	0.015	0.039	0.039	24
370	0.010	0.024	0.028	117	0.014	0.051	0.116	92
380	0.006	0.015	0.026	104	0.010	0.024	0.031	74
390	0.006	0.019	0.022	54	N/A[a]	N/A	N/A	0
392	0.005	0.014	0.014	13	N/A	N/A	0.010	3
394-398	0.010	0.016	0.026	34	0.011	0.020	0.020	30
396	0.007	0.018	0.018	24	0.014	0.026	0.026	17
400-460	0.008	0.022	0.047	213	0.016	0.040	0.044	44
600-680	0.006	0.015	0.020	152	N/A	N/A	0.002	1
320 Ref	0.009	0.022	0.064	77	0.015	0.043	0.056	37
330 Gr Salt	0.009	0.027	0.028	29	0.013	0.028	0.041	17
370 Misc	0.010	0.026	0.046	33	0.013	0.037	0.044	26
370 Mtns	0.009	0.024	0.047	244	0.013	0.038	0.652	190
370 Mtns Elec	0.008	0.015	0.018	26	0.010	0.022	0.068	21
370 Mtns Misc	0.009	0.016	0.027	30	0.015	0.030	0.220	21
372 Stores	0.008	0.015	0.020	37	0.007	0.019	0.027	24
374 Inst. Shop	0.009	0.015	0.029	42	0.014	0.026	0.028	21
376 Boiler	0.007	0.016	0.026	23	0.007	0.012	0.012	6
392 Plant Pr	0.007	0.013	0.017	43	0.008	0.012	0.012	14
Other	0.008	0.024	0.028	92	0.011	0.055	0.181	33

a. NA = not applicable (five or fewer records)

Table 5-14. Uranium urine data summary by cost center for 1963.

Cost center	Monday samples (mg/L)			Number of records	Friday samples (mg/L)			Number of records
	Median	95th percentile	Maximum		Median	95th percentile	Maximum	
110	0.021	0.035	0.259	34	0.027	0.078	0.178	24
120	0.016	0.035	0.085	135	0.024	0.078	0.143	66
150	0.012	0.023	0.023	14	0.014	0.035	0.035	9
180	0.014	0.027	0.034	33	0.020	0.034	0.036	22
200	0.012	0.028	0.033	18	0.030	0.120	0.120	14
250-290	0.019	0.043	0.052	50	0.027	0.078	0.100	26
300-310	0.014	0.027	0.048	34	0.037	0.054	0.083	16
320	0.014	0.026	0.052	148	0.027	0.112	0.185	72
321-340	0.014	0.037	0.046	42	0.025	0.068	0.137	28
350-369	0.014	0.032	0.336	75	0.016	0.078	0.205	47
370	0.016	0.040	0.059	115	0.021	0.054	0.122	75
372-378	0.010	0.026	0.035	97	0.013	0.036	0.042	43
380-396	0.012	0.023	0.029	87	0.016	0.035	0.039	39
400-460	0.012	0.029	0.048	115	0.014	0.056	0.081	47
500-510	0.008	0.021	0.025	43	0.012	0.024	0.045	22
520-530	0.013	0.027	0.028	51	0.012	0.035	0.041	25
550	0.013	0.025	0.027	46	0.012	0.027	0.041	36
600-690	0.008	0.021	0.065	91	0.008	0.023	0.027	33
330 Gr Salt	0.013	0.027	0.029	46	0.020	0.042	0.044	21
370 Elec	0.012	0.023	0.025	28	0.018	0.059	0.059	13
370 Mill	0.016	0.029	0.029	31	0.027	0.058	0.258	25
370 Misc	0.015	0.027	0.091	91	0.016	0.043	0.111	59
370 Mtns	0.014	0.028	0.041	178	0.019	0.052	0.095	117
370 Pipe	0.014	0.029	0.045	22	0.029	0.058	0.058	14
380 Warehouse	0.012	0.027	0.029	69	0.012	0.025	0.035	50
392 Plant Pr	0.010	0.026	0.027	50	0.014	0.027	0.034	31
Other	0.012	0.029	0.070	95	0.015	0.036	0.087	55

Table 5-15. Uranium urine data summary by cost center for 1964.

Cost center	Monday samples (mg/L)			Number of records	Friday samples (mg/L)			Number of records
	Median	95th percentile	Maximum		Median	95th percentile	Maximum	
110	0.020	0.066	0.282	84	0.032	0.103	0.232	75
120	0.012	0.029	0.090	197	0.016	0.071	0.182	115
150	0.012	0.026	0.028	57	0.014	0.050	0.067	36
180	0.017	0.043	0.054	32	0.018	0.056	0.226	29
200	0.019	0.053	0.078	32	0.018	0.088	0.114	25
290	0.012	0.038	0.054	32	0.013	0.070	0.106	16
310-321	0.014	0.036	0.106	72	0.018	0.063	0.118	53
350-360	0.011	0.073	0.104	48	0.008	0.085	0.358	26
370	0.012	0.025	0.062	138	0.017	0.066	0.626	86
372-380	0.012	0.022	0.026	44	0.010	0.025	0.027	25
380-390	0.010	0.026	0.029	66	0.012	0.040	0.084	34
392	0.008	0.023	0.026	41	0.008	0.018	0.040	25
400-460	0.012	0.027	0.068	94	0.013	0.048	0.165	34
500-510	0.008	0.020	0.024	72	0.006	0.022	0.024	44
520	0.010	0.021	0.029	64	0.012	0.033	0.044	35
530	0.010	0.026	0.064	28	0.008	0.017	0.052	16
550-569	0.010	0.022	0.029	118	0.008	0.026	0.031	75
600-690	0.008	0.017	0.027	73	0.010	0.016	0.024	14
370 Elec	0.012	0.024	0.029	40	0.011	0.050	0.052	28
370 Mill	0.014	0.026	0.028	33	0.024	0.052	0.081	22
370 Misc	0.012	0.025	0.028	99	0.009	0.046	0.058	63
370 Mtns	0.017	0.062	0.125	102	0.013	0.029	0.064	57
370 Pipe	0.013	0.029	0.053	37	0.013	0.052	0.090	27
Other	0.012	0.027	0.042	95	0.011	0.050	0.059	52

Table 5-16. Uranium urine data summary by cost center for 1965.

	Monday samples (mg/L)			Number of	Friday samples (mg/L)			Number of
Cost center	Median	95th percentile	Maximum	records	Median	95th percentile	Maximum	records
110	0.018	0.034	0.042	41	0.018	0.094	0.152	19
120	0.011	0.025	0.047	156	0.014	0.072	0.318	102
150	0.011	0.035	0.347	61	0.011	0.036	0.049	35
180	0.020	0.045	0.052	45	0.024	0.102	0.146	26
200	0.015	0.024	0.028	19	0.016	0.069	0.088	16
290	0.010	0.015	0.021	26	0.010	0.025	0.039	19
310-321	0.011	0.042	0.065	29	0.012	0.024	0.024	16
350	0.007	0.018	0.027	36	0.005	0.012	0.024	19
370-379	0.008	0.022	0.045	165	0.010	0.055	0.193	120
380-390	0.010	0.026	0.050	46	0.011	0.026	0.033	27
392	0.004	0.010	0.015	28	0.006	0.015	0.016	18
400-460	0.009	0.039	0.078	85	0.017	0.045	0.058	25
500-510	0.006	0.015	0.016	60	0.005	0.012	0.016	34
520	0.007	0.018	0.020	57	0.008	0.022	0.030	33
530	0.006	0.018	0.022	19	0.012	0.022	0.022	7
550-569	0.007	0.018	0.027	109	0.007	0.014	0.026	73
600-690	0.006	0.014	0.016	65	0.005	0.011	0.011	5
370 Elec	0.008	0.024	0.027	28	0.008	0.027	0.031	22
370 Mill	0.012	0.030	0.048	35	0.012	0.048	0.060	22
370 Misc	0.010	0.024	0.057	88	0.008	0.025	0.039	63
370 Mtns	0.011	0.025	0.069	69	0.012	0.029	0.061	48
370 Pipe	0.012	0.020	0.024	37	0.011	0.027	0.042	26
Other	0.010	0.020	0.075	85	0.009	0.033	0.055	43

Table 5-17. Uranium urine data summary by cost center for 1966.

	Monday samples (mg/L)			Number of	Friday samples (mg/L)			Number of
Cost center	Median	95th percentile	Maximum	records	Median	95th percentile	Maximum	records
110	0.015	0.030	0.039	19	0.025	0.053	0.053	12
120	0.009	0.030	0.052	130	0.015	0.048	0.100	83
150	0.006	0.018	0.019	25	0.011	0.054	0.054	13
180	0.012	0.036	0.036	20	0.025	0.048	0.048	9
200	0.006	0.027	0.027	17	0.011	0.024	0.024	9
290	0.006	0.024	0.024	22	0.006	0.036	0.036	12
310-321	0.009	0.021	0.022	20	0.016	0.459	0.459	11
350	0.002	0.008	0.008	13	0.002	0.009	0.009	15
370-379	0.006	0.027	0.046	107	0.008	0.050	0.052	61
380-390	0.007	0.021	0.025	26	0.008	0.029	0.029	13
400-460	0.006	0.025	0.035	53	0.008	0.032	0.045	24
500-510	0.002	0.008	0.011	26	0.004	0.008	0.010	35
520	0.006	0.025	0.018	34	0.004	0.024	0.024	25
530	0.005	0.019	0.021	24	0.004	0.022	0.022	12
550-569	0.005	0.012	0.020	72	0.004	0.012	0.028	38
610-690	0.006	0.011	0.012	18	0.004	0.008	0.008	4
370 Elec	0.012	0.027	0.027	13	0.009	0.036	0.036	10
370 Mill	0.008	0.024	0.027	32	0.008	0.034	0.071	16
370 Misc	0.008	0.021	0.028	53	0.010	0.033	0.036	29
370 Mtns	0.009	0.022	0.066	46	0.009	0.050	0.055	25
370 Pipe	0.006	0.015	0.024	22	0.015	0.088	0.088	13
Other	0.006	0.019	0.027	58	0.004	0.025	0.213	37

5.3.1.2 Thorium

There is no indication discovered so far that a routine urine-sampling program was implemented for thorium. No urine bioassay data for thorium have been found in the worker files.

5.3.2 Environmental Monitoring Period (1975 to 1984)

No personnel bioassay monitoring appears to have been conducted during this period.

5.3.3 Remediation Period (1985 to 2001)

An extensive, state-of-the-art bioassay monitoring program was conducted during the 1991 to 2001 period to detect intakes greater than 100 mrem committed effective dose equivalent (CEDE). This program is well defined in the WSSRAP Technical Basis Manual revisions (DOE 1991, 1994, 1997, 1998a,b,c, 2000a, 2001). The focus of the program was to conduct bioassay based on workplace action levels for air sampling, nasal wipe analysis, and wipe analysis of the inside of respirators at the end of each day they were used. These action levels triggered fecal sampling, urine sampling, and *in vivo* measurements, as appropriate.

There are no Technical Basis Manuals available to document the bioassay monitoring program during the early part of the remediation period (1985 to 1990). If no individual bioassay data and no applicable co-worker bioassay data are available for a claimant for the 1985 to 1990 time period, the environmental data for 1985 to 1990, described in the Technical Basis Document for the Weldon Spring Plant – Occupational Environmental Dose (ORAU, 2005c) can be used to estimate intake. Alternatively, individual or co-worker bioassay data for the subsequent year (1991) may be used to estimate worker intakes.

Routine uranium urine sampling also occurred monthly for at-risk workers. Uranium MDAs were reported as 1 µg/L in 1991 (DOE 1991) using laser fluorimetry and as 0.1 µg/L in 1994 to 1998 (DOE 1994, 1997, 1998a) and 0.0524 µg/L in September 1998 to 2001 (DOE 1998b,c, 2000a, 2001) using kinetic phosphorescence analysis. Uranium results of 0.2 µg/L or greater were considered positive for occupational uranium intakes in 1997 (DOE 1997), and results of 0.3 µg/L or greater (DOE 1998c, 2000a, 2001) were considered positive for occupational uranium intakes from 1998 through 2001.

5.4 *IN VIVO* MEASUREMENTS

5.4.1 Operational Period (1957 to 1967)

There is no indication that WSP had an *in vivo* measurement program or performed any *in vivo* measurements for uranium, but there is an indication that *in vivo* measurements were performed on some WSP workers for thorium in 1966:

> *From July 11 through July 27, 1966, Y-12 personnel visited the Weldon Spring plant and set up the portable Whole Body Counter for in vivo thorium counting to quantify body burden deposition and the risk inherent with using the current Atomic Energy occupational air concentration limits (3.7E-11 µCi/ml). During this period of testing, 200 measurements were made in the monitoring of 148 persons. The determination of workers to be monitored was done on a strictly voluntary basis. A good cross representation of workers volunteered. The interpretation of the result is as follows:*
>
> 1. *Workers who showed net counts less than 60 counts per 20 minutes had less than detectable amounts of thorium in their lungs and were therefore given a 'negative' result.*
>
> 2. *Workers showing net counts in excess of 60 counts per 20 minutes but less than 204 were interpreted as a 'trace' of thorium.*

3. *Net counts in excess of 204 counts for 20 minutes were considered as 'positive' evidence of thorium lung burdens. A person who showed 204 counts for 20 minutes was considered to have at least one lung burden.*

The overall results showed workers involved in areas 101, 103, 301, 403, Maintenance, and Health and Safety, which were principal exposure positions, had a more frequent occurrence of 'trace' detections. No workers monitored showed a 'positive' designation. (Ingle 1991)

The reports of these measurements observed in the worker files are titled "Thallium 208 in Vivo Results." This indicates that ^{208}Tl was measured as a marker for thorium. Thallium-208 is in approximate equilibrium (with a branching ratio of 0.337) with ^{228}Th, which might not have been in equilibrium with the thorium series parent ^{232}Th following chemical purification of the natural thorium feed materials. The quantification of thorium depositions from these *in vivo* ^{208}Tl measurements is, therefore, uncertain without knowledge of the degree of equilibrium of the thallium with the ^{232}Th parent. The record only gives a qualitative indication, background or trace, of the detection of ^{208}Tl as a marker for thorium.

5.4.2 Environmental Monitoring Period (1975 to 1984)

No personnel *in vivo* monitoring appears to have been conducted during this period.

5.4.3 Remediation Period (1985 to 2001)

An *in vivo* measurement program was included in the design of the WSSRAP internal dosimetry program to evaluate intakes of ^{238}U and ^{232}Th. Because the number of radiological workers exposed to airborne radioactivity at the WSP site was expected to be small, WSSRAP could not justify the expense of having its own *in vivo* measurement system. Instead, the program was initially based on detection sensitivities provided by the Helgeson Scientific Services mobile counting laboratory. The lower limit of detection cited for that system was 74 Bq (2 nCi) for natural uranium and 37 Bq (1 nCi) for ^{232}Th in the lung (DOE 1991).

Later, the program was based on detection sensitivities of the *in vivo* measurement system at the Fernald Environmental Management Project. The sensitivities (of unstated pedigree) cited for that system were 2.0 nCi for ^{238}U and 1.2 nCi for ^{228}Ac, assumed to be in secular equilibrium with ^{232}Th (DOE 2001). Revision 7 (DOE 2001) of the WSSRAP Technical Basis Manual states:

> *This assumption* [of secular equilibrium] *will not necessarily be true in an actual worker intake.*

and

> *It is important to note that these 'typical' detection limits are highly dependent upon the individual worker's physical features such as height and chest size. The Weldon Spring site has sent individuals to the Fernald site for lung counts, and detection limits were 2.5 times higher than the typical values due to the individual's physical features.*

In vivo lung measurements could have been performed as a special bioassay measurement following a suspected or actual intake. Revision 7 (DOE 2001) states that such measurements were normally reserved for "those incidents where the intake was suspected to exceed 500 mrem CEDE …."

Revision 7 (DOE 2001) also states that "requiring lung counts for affected workers" was one of the actions "never used."

5.5 AIR CONCENTRATION AND DUST MEASUREMENTS

5.5.1 Operational Period (1957 to 1967)

Two types of air sampling were conducted by MCW at the WSP. A daily weighted-average (DWA) concentration index was calculated based on a combination of breathing-zone air (BZA) samples and general air samples (MCW 1965).

Semi-fixed location general air samplers were located in each process building. The results of the samples were used to assess changes in plant air concentrations due to equipment malfunction or incorrect operation. However, these data were considered to have no direct value in assessing individual intakes or doses.

Samples were analyzed for gross alpha activity only. The action level for uranium dust exposure was 700 alpha dpm/d (MCW 1965). As noted above, this was a time-weighted average exposure. This was derived from the guideline concentration of 70 alpha dpm/m^3 and the assumption that the average breathing rate for workers is 10 m^3/d for an eight-hour workday.

BZA samples were collected at WSP during specific times, and the data was applied to specific jobs.

5.5.1.1 Maximum Acceptable Concentrations

The time-weighted average maximum allowable concentration (MAC) for uranium was 70 alpha dpm/m^3 (3.2×10^{-11} µCi/cm^3) or 50 µg/m^3 (74 alpha dpm/m^3). The 70 alpha dpm/m^3 MAC was apparently derived by rounding down the radioactivity equivalent of the mass concentration MAC.

The AEC standards for radiation protection applicable in 1965 listed the allowable concentrations for uranium in air for occupational exposure as 7×10^{-11} µCi/cm^3 for soluble forms and 6×10^{-11} µCi/cm^3 for insoluble forms. The WSP guideline of 70 alpha dpm/m^3 (3.2×10^{-11} µCi/cm^3) was more restrictive than the AEC standard.

There are no indications in the radiation protection program documents that the WSP MAC was adjusted in areas where thorium was processed. The values for soluble and insoluble natural uranium from the 1959 ICRP II convention for maximum permissible concentrations (MPCs) were 7×10^{-11} µCi/cm^3 and 6×10^{-11} µCi/cm^3, respectively (HPS 1960, Volume 3). As noted above, these values were adopted by the AEC. The ICRP II MPCs for soluble and insoluble natural thorium were much more restrictive at 2×10^{-12} µCi/cm^3 and 4×10^{-12} µCi/cm^3, respectively. However a provisional level of 3×10^{-11} µCi/cm^3 was recommended. This is the value that was adopted in AEC Appendix 0524 Annex 1, Standards for Radiation Protection (AEC 1963). This provisional level for the MPC for natural thorium is similar to the time-weighted average MAC for uranium used at WSP.

Based on the information available, it is reasonable to assume that the gross alpha MAC of 70 dpm alpha/m^3 was applied across the WSP.

5.5.1.2 Special Curie for Uranium

The air-sampling data were reported in alpha particle disintegrations per minute rather than curies because the former was an unambiguous designation. Until 1973 when the National Council on

Radiation Protection and Measurements (NCRP) discouraged its use, the *special curie* was commonly employed for natural uranium. The special curie was defined in 1959 as follows (NCRP 1973):

$$\text{Special curie} = 3.7 \times 10^{10} \text{ d/s } ^{238}\text{U} + 3.7 \times 10^{10} \text{ d/s } ^{234}\text{U} + 9 \times 10^{8} \text{ d/s } ^{235}\text{U}$$
$$= 7.49 \times 10^{10} \text{ d/s}$$

The definition was altered slightly in 1963 to use 1.7×10^{9} d/s for ^{235}U. It is important to understand the use of the term special curie when the data are reviewed. In addition, MCW used the ratio between the measured DWA and the guideline or standard as an index of exposure.

5.5.1.3 Dust Exposure Calculation

The total dust exposure worksheets used to record data for the MCW St. Louis site have a provision for entering the dust concentration, but the applicable worksheet column does not list the units. The values are most likely the index (i.e., the DWA divided by the guideline), and there is no indication of the source of the data. The intake calculated from urine bioassay and the intake calculated from measured dust concentrations were averaged with equal weight given to each source. There is no evidence to show when use of these forms was discontinued and no indication that they were used for WSP employees.

The Annual Personnel Internal-External Radiation Exposure Report form, apparently in use by 1959 for WSP, includes a section for average dust concentration in disintegrations per minute per cubic meter by calendar quarter. None of the exposure reports reviewed had any data in that section. This indicates that the dust concentration was not routinely recorded. However, because the average dust concentration, when recorded, is in units of disintegrations per minute per cubic meter, the average daily intake can be calculated by assuming a breathing rate of 10 m^3/d for typical light work for an eight-hour work day or by using a job-specific value.

No specific in-plant air monitoring analysis sheets were found, but samples of the forms for reporting perimeter air sample data were available. These forms could also have been used for in-plant measurements. The forms include information on the sampling rate, time, and the gross alpha activity. The samples were analyzed for alpha and beta activity.

5.5.1.4 Dust Studies

A study of specific areas and jobs in Building 301 was conducted in 1961 (MCW 1961). Time-weighted average concentrations were calculated based on the number of work hours at various positions. The measured concentrations were reported in microcuries per cubic centimeter using the special curie unit and in micrograms per cubic meter. The data were used primarily as a basis for recommending actions to reduce concentrations. There is no indication that the data were used to assess intake.

An undated document titled *Summaries of Dust Concentrations at Production Jobs* (MCW c. 1966) provides data on time-weighted average dust concentrations for various work areas for the period from 1958 to 1966. The data were summarized for historic use in evaluating worker dust exposures. The dust samples were collected on open-face Whatman No. 41 or membrane filters with areas ranging from 3 to 5 cm^2. The membrane filters had a pore size of 0.8 µm. The flow rate ranged from 10 to 20 L/min. The report notes that the samples were taken either as fixed general air samples or as “hand held breathing zone type.”

The filters were analyzed for gross alpha by scintillation counters. Samples from uranium areas were counted after a minimum 24-hr delay to allow for the decay of radon progeny. Samples from thorium areas were counted after a minimum delay of 100 hr.

For uranium work areas, a work group index was calculated based on an average time-weighted exposure for various job titles. The jobs were rotated among the workers in each work group so the work group index could provide a reasonable time-weighted average concentration in milligrams uranium per cubic meter. The activity on the filter was converted to mass concentration using a factor of 1.5 alpha dpm/µg U (0.68 pCi/µg U). Table 5-18 summarizes the work group concentrations.

Table 5-18. Work group time-weighted average exposure for uranium operations.

Work group title	Daily weighted average concentration (µg U/m^3)								
	1958	**1959**	**1960**	**1961**	**1962**	**1963**	**1964**	**1965**	**1966**
Sampling and Repackaging Cost center 110	65	55	45	55	32	31	20	60	
Digest-Raffinate, Cost center 120	17		7			11	8	7	
Extraction, Cost center 120	4		3		5		2	2	
Denitration (pot room), CC 120	117		70		22		45	16	
Green Salt, Cost center 150	40	37	37	53	53[a]	16[a]	16	22	
Dingot Reduction, Cost Center 180			115	65		150[a]	150[a]	78	50
Metal – other than reduction Cost centers 200, 210, 220, 280				44	52	63	40[b]	55[e]	
Core area, Cost center 250			20						
Pilot/Scrap Plants, Cost center 290			50[c]		50[c]	50[c]	50[c]	50[c]	
Special Projects[d]									

a. Estimate by extension to other years or averaged over other years.
b. 50% of effort on uranium jobs; also worked with thorium.
c. Overall average - not weighted.
d. No averages given.
e. 40% of effort on uranium jobs; also worked with thorium

Thorium was used intermittently in various operations at WSP from November 1963 to 1966. Table 5-19 lists the thorium dust concentration data by year.

Table 5-19. Work group time-weighted average exposure for thorium operations.

Work group title	Daily weighted average concentration (dpm/m^3)		
	1963	**1964**	**1965**
Pot Room		78[a]	50[b]
Metals other than reduction		17[b]	113[c]
Scrap plant	180[d]	117[c]	28[e]

a. 40% of the effort on thorium at this index.
b. 50% of the effort on thorium was at this index.
c. 60% of the effort on thorium jobs.
d. 17% of the effort on thorium.
e. 80% of the effort on thorium.

The dust concentration data can be used to indicate general levels of airborne radionuclides. The percentage of effort stated in the Table 5-19 indicates the fraction of the time the work involved thorium. As noted previously, there is no indication that the dust concentration data were used in calculating doses to individual workers at WSP.

Dust concentrations were also measured for individual operations for specific years. These operations were intermittent and did not represent the total work year. The time-weighted average

concentrations as a fraction of the MACs were calculated. The MAC for thorium at the time was 70 dpm/m^3. Table 5-20 summarizes these calculations.

Table 5-20. Time-weighted average thorium dust concentrations.

Operation	Bldg.	Year	Fraction of MAC	Total effort
Repackaging ThO_2 feeds	101	1966	0.1	10 shifts
Hopper packaging	101	1966	0.2	10 shifts
Conveyors at repackaging	101	1966	0.1	10 shifts
Oven drying ThO_2 solution	101	1965-66	<0.1	2 person d
Oven drying ThO_2 solution	101	1965	0.2	Intermittent
Hopper feed and digestion	103	1966	0.1	No information
Raffinate	103	1966	<0.1	No information
Misc. operations	103	1966	<0.1	Non-routine job
ThO_2 repackaging	103	1964	3.7	Short term – Airline masks used in hood
Extraction	105	1966	<0.1	No information
Crystal denitration	103	1963-64	1.25	Four months – Airline masks used
Crystal denitration	103	1964-65	0.9	Airline masks prescribed
TNT[a] liquor denitration	103	1965	0.7	Eight months – Airline masks prescribed
Solution drying – vacuum unload	103	1965-66	0.6	10 months – "Comfo" respirators prescribed
Pot denitration	103	1965	2	One month – Airline mask prescribed for some tasks
High firing at recast furnace	301	1963-64	0.3	Eight months – Airline masks for some tasks
High firing in billet heaters	301	1964-66	0.15	No information
TNT[a] repackaging	301	1964-65	0.04	No information
ThO_2 repackaging	301	1965	30	One month – Airline masks and cover clothing used
ThO_2 repackaging	301	1965	0.8	Airline masks used for some tasks
Kiln calcining sump cake	301	1966	0.3	"Comfo" respirators used for some tasks
TNT repackaging	403	1963-64	0.2	Eight months
ThO_2 repackaging	403	1963-64	3.3	Eight months – Airline masks used on some operations
ThO_2 repackaging	403	1964-65	0.3	13 months
Fluid bed denitration	403	1965	0.6	Three months
Fluid bed denitration	403	1965-66	0.4	No information

a. TNT = trinitrotoluene.

5.5.2 Transfer to the U.S. Army (1968 to 1974)

The site was transferred to the Army in 1967 for use in herbicide production. Some areas of Buildings 101 and 103 were partially cleaned up in preparation for the new mission. However, the project was terminated in early 1969, and no further cleanup or construction occurred. No dust measurement data were found for that period.

5.5.3 Environmental Monitoring Period (1975 to 1984)

With the exception of the WSQ and WSRP, the remainder of the site including the WSCP was turned over to the U.S. Army Corp of Engineers in 1969. No major activities took place from 1969 to 1985. There were no AEC contractors on site from 1969 to 1975, after which the AEC contracted with National Lead of Ohio to perform environmental monitoring around the WSQ and WSRP. Bechtel National took over management of the WSRP and WSQ in 1981, and the WSCP site was transferred from the Army to DOE in 1985.

No monitoring data were available in relation to personal dosimetry programs for this period.

5.5.4 Remediation Period (1985 to 2001)

DOE contracted with MK-Ferguson to manage the remediation project in 1986. The WSRP and the WSQ were put on the U.S. Environmental Protection Agency's National Priorities List in 1987 and 1989, respectively. Site cleanup began in earnest in 1990 as described in ORAU (2005b).

No specific air-sampling records were available for this period. However, the internal dosimetry programs were well documented in the WSSRAP Technical Basis Manuals (DOE 1991, 1994, 1997, 1998a,b,c, 2000a, 2001). The air-sampling program consisted of area samples and BZA samples. Area sample data were not used routinely for dose assessment, but were used (only) if no BZA sample data were available and bioassay measurements could not confirm potential intakes.

At first, area samples were taken with both high-volume (1,100 L/min) and low-volume (40 L/min) samplers. The filters were analyzed by an offsite laboratory for isotopic content. The MDA on the filters was 1 pCi. The calculated minimum detectable concentrations (MDCs) for a 6-hr sampling period were 2.7×10^{-15} µCi/ml and 6.9×10^{-14} µCi/ml for the high- and low-volume samplers, respectively. The gross alpha MDC for the high-volume sampler was 1.5×10^{-14} µCi/ml.

By 2001, area air sampling was normally performed with low-volume samplers. The samples were analyzed for long-lived gross alpha activity. The results were compared to an area-specific effective DAC developed based on isotopic concentrations in the dust. The MDC for the system as it was routinely used (sampling and counting times, etc.) was normally less than 1.2×10^{-14} µCi/ml.

BZA samples were routinely used with bioassay results to assess worker intakes. The BZA samples had higher MDCs, generally below 2.1×10^{-13} µCi/ml (1991) and 1.9×10^{-13} µCi/ml (2001). To assess worker intakes, the gross alpha concentrations were apportioned to specific radionuclides based on isotopic analysis of area samples.

Area air sampling was required in all areas where a worker could have an annual intake greater than 2% of the annual limit of intake. Based on the various revisions to the WSSRAP Internal Dosimetry Technical Basis Manual, the requirements for breathing-zone samples varied slightly over time. At first, every worker spending at least 1 hr/d in the WSQ or WSRP airborne radioactivity areas was required to wear a BZA. Later requirements were somewhat less strict. Revision 1 of the Technical Basis Manual (DOE 1994) required BZA samples for one in five individuals spending more than 1 hr/d in areas with concentrations greater than 10% of the DAC and one in three individuals in the WSQ or WSRP areas. As of Revision 2 (DOE 1997), one in four workers in areas with concentrations greater than 2% of the DAC were required to wear BZA samplers.

The internal dosimetry program required that routine monitoring of environmental ^{222}Rn (radon) and ^{220}Rn (thoron) and their decay products be instituted when an individual was likely to receive an annual intake of 10% or more of the annual limit of intake. According to Revision 7 of the Technical Basis Manual (DOE 2001), that threshold was never exceeded. Therefore routine monitoring data for radon gas or short-lived decay products are not available. Environmental radon measurements that were taken periodically are provided in the Technical Basis Document for the Weldon Spring Plant – Occupational Environmental Dose (ORAU, 2005c). These measurements provide a basis for estimating worker exposure to short-lived radon decay products.

5.6 ASSESSMENT OF INTAKES

5.6.1 Operational Period (1957 to 1967)

5.6.1.1 Uranium Intakes

The worker's urine bioassay data are the primary data available to the DR to quantify the uranium intake for the worker who is the subject of the claim. These data can be supplemented by work group monitoring data because essentially continuous bioassay monitoring for a worker was simulated by at least one worker in the group being sampled each week with Monday – Friday – Monday sampling for

potentially exposed workers (Section 5.3.1.1.1). The work group data have been reconstructed from urine data for all WSP workers by cost center code. Tables 5-7 to 5-16 (Section 5.3.1.1.5), list the median, the 95th percentile, and the maximum values of the uranium urine data per year for Monday samples and Friday samples by cost center and calendar year. The worker's urine data reports provide the cost center (Section 5.3.1.1.5).

If specific information is not available in the worker's file, the DR should consider the following default uranium source terms:

- Natural uranium, before 1961
- Natural uranium, recycled, 1961 to 1962
- Enriched (1%) uranium, recycled, 1963 to 1967

Because the feed uranium and uranium during processing were purified to some degree, it is reasonable to assume that the contributions of the long-lived uranium progeny, i.e., ^{230}Th and ^{226}Ra, were small in most areas of the WSP. However, for workers in Building 101 where the uranium concentrates were initially processed, the concentrations of ^{230}Th and ^{226}Ra should be assumed to be 5% of the ^{238}U activity and 1% of the ^{238}U activity, respectively as described in Section 5.2.2.

Because site-specific particle size data are not available, the default value of 5 µm AMAD should be used.

If the absorption type of the uranium to which the worker was exposed cannot be discerned from the data in the worker's file, the DR should use the absorption type that is the most claimant-favorable.

5.6.1.2 Thorium Intakes

No quantitative *in vitro* or *in vivo* bioassay results have been observed for thorium (Sections 5.3.1.2 and 5.4.1). However, dust studies for thorium operations (Section 5.5.1.4) indicate that airborne thorium dust concentrations exceeded the MAC for several of the operations studied.

The suggested approach to assess natural thorium intakes is to use the approach in Section 5.3 of the Fernald TBD for internal dosimetry (ORAU 2004a). Application of the Fernald approach to WSP is based on the assumption that the thorium operations at Fernald are reasonably applicable to thorium operations at WSP. Assumptions and information used for the Fernald approach that also appear to be valid for WSP are:

1. Few, if any, *in vitro* analyses exist in worker files. The *in vivo* results that exist are not quantified for thorium, and there is not enough information available to quantify the thorium deposition from the *in vivo* results.

2. Although respiratory protection equipment was available and its use required, it cannot be assumed that the respirators were always used or that there was a tight seal to the face when they were used. No respiratory protection factor is assumed.

3. The MAC of 100 dpm/m^3 (4.5×10^{-11} µCi/cm^3) was used at Fernald for control purposes. The MAC of 70 dpm/m^3 was used at WSP for control purposes, or a factor of 0.7 less than the control level at Fernald.

4. Workers could have been exposed to airborne thorium above the MAC level. It is assumed that 100 hr/yr at 10 MAC accounts for intermittent exposures to high levels of airborne thorium.

5. Normal operations would reasonably be controlled to levels well under the MAC. An intake of 0.1 MAC for 500 hr/yr was assumed for normal operations.

Under these assumptions, the claimant-favorable assumption results in an annual exposure of 735 MAC-hr (in comparison to 1,050 MAC-hr for Fernald) or an intake of approximately 40 nCi Th/yr (in comparison to 60 nCi Th/yr for Fernald).

The Fernald approach further considers the isotopic composition and daughter contributions:

> *Therefore, in the absence of monitoring data a claimant-favorable default intake is 30 nCi/y (164 pCi/d) each of Th-232 and Th-228 (the alpha emitting isotopes detected on the air samples) plus a 60 nCi/y (164 pCi/d) intake of Ra-228. The Ac-228 beta emitter adds about 1% to the effective dose and therefore can be ignored in the dose calculations. Although an equal intake of Ra-224 might have occurred, because of its alpha emissions, it is adequately accounted for in the thorium intake assumption, which is based on detection of all alpha emitters on the air samples* (ORAU 2004a).

The corresponding default intake values for WSP are 20 nCi/yr each of ^{232}Th and ^{228}Th (the alpha-emitting isotopes detected on the air samples) plus a 40-nCi/yr intake of ^{228}Ra. This approach should be applied to those who worked directly with thorium or in the immediate vicinity of thorium processes (e.g., chemical or metal operators and maintenance workers as well the Health and Safety staff who monitored the operations).

Thorium operations were mainly in the pilot plants (Building 403) and the refinery (Building 103) (DOE 1986). The MCW Four-Plant Study considered workers in any of the following locations to have been exposed to thorium dust:

> *Bldg. 101 – sampling and repackaging*
> *Bldg. 103 – digestion, denitration, packaging*
> *Bldg. 105 – organic extraction, purification*
> *Bldg. 201 – interim storage*
> *Bldg. 301– high temperature firing, packaging*
> *Bldg. 403 – packaging, fluid bed denitration*
> *Bldg. 404 – No thorium operations but some exchange of personnel with 403.*
> *Maintenance – All crafts involved in repair of thorium handling machinery.*
> *Janitors – Clean up of contaminated equipment and laundry.*
> *Warehousing – storage, shipping* (Ingle 1991).

It is claimant-favorable to apply the default approach to any worker in these buildings or professions in the period of thorium processing (November 1963 to 1966). For workers not in these buildings or professions during this period, thorium intakes should be assessed as environmental intakes in accordance with *Technical Basis Document for the Weldon Spring Plant – Occupational Environmental Dose* (ORAU 2005c) unless there is information in the worker's file that indicates involvement in thorium operations.

5.6.1.3 Intakes of Other Radionuclides

Intakes of other radionuclides, such as radon, should be assessed as environmental intakes in accordance with the *Technical Basis Document for the Weldon Spring Plant – Occupational Environmental Dose* (ORAU, 2005c). There are no personal dosimetry data or air concentration measurements available for radon in the WSCP during the operational period.

The annual average environmental radon concentrations for the Weldon Spring Chemical Plant (WSCP), Weldon Spring Raffinate Ponds (WSRP), and Weldon Spring Quarry (WSQ) are provided in Table 4.5 of ORAU (2005c). The site-wide maximum annual exposures to radon decay products, by year, are provided in Table 4-7 of that same document. These estimates are based on environmental radon gas releases. The calculation in ORAU (2005c) was based on the assumption that all radon gas generated by decay of Ra-226 was released during processing. The environmental concentrations were calculated for the areas within 100 meters of the assumed release point, the acid recovery plant stack.

Occupational exposures to radon decay products in Working Level Months (WLM) were calculated assuming an equilibrium factor of 0.3, a value that is typical for outdoor environments. The estimated annual radon decay product exposure for the years 1957 to 1967 was 8.7E-2 WLM. The equilibrium factor for indoor environments, thus the exposure for a given ^{222}Rn concentration, may be higher than for outdoor environments, depending on the general building ventilation.

The Weldon Spring Historic Dose Estimate (Meshkov et al., 1986) states that the prime source of radon emissions was the acid recovery plant. The denitration process that Meshkov (1986) describes as the "dustiest part of the whole operation" was carried out inside large hoods. Therefore, it is reasonable to assume that most of the radon released during processing went out the stack and only a small fraction was released into the room. However, the radon released to the environment and present in the area would have been drawn back into the building through the ventilation systems. Under those circumstances, the radon concentration inside the building would have been approximately equal to the radon concentration outdoors. If a typical indoor equilibrium factor of 0.5 is assumed rather than the outdoor factor of 0.3, the estimated annual exposure would have been 1.5E-1 WLM for 100 percent occupancy for 2000 hours per year.

5.6.2 Environmental Monitoring Period (1975 to 1984)

All intakes in this period should be assessed as environmental intakes in accordance with ORAU (2005c).

5.6.3 Remediation Period (1985 to 2001)

Because WSSRAP conducted an extensive, state-of-the-art internal dosimetry program during remediation that was designed to detect and evaluate intakes of 100 mrem CEDE or more for "all occupational radionuclide intakes in a year (other than radon, thoron, and their progeny)" (DOE 2001), it is reasonable to consider that the worker's file will contain a detailed report of the pertinent data used for the assessment. It is also reasonable to expect that the data will be self-explanatory and can be used by the DR as found. If needed, a detailed description of the internal dosimetry program is available in the series of WSSRAP Technical Basis Manuals (DOE 1991, 1994, 1997, 1998a,b,c, 2000a, 2001).

In addition to assessing specific intakes, the DR should assess the environmental intakes in accordance with ORAU (2005c).

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GLOSSARY

activity median aerodynamic diameter (AMAD)
The diameter of a unit density sphere with the same terminal settling velocity in air as that of the aerosol particle whose activity is the median for the entire aerosol.

becquerel (Bq)
A unit of radioactivity equal to one nuclear disintegration per second.

bioassay
Measurement of amount or concentration of radioactive material either in the body or in biological material excreted or removed from the body. Another word for *radiobioassay*.

bioassay procedure
A procedure used to determine the kind, quantity, location, and retention of radionuclides in the body by direct (*in vivo*) measurements or by *in vitro* analysis of material excreted or removed from the body.

body burden
The quantity of radioactive material in an individual's body at a particular point in time.

chronic
Low-level intakes received on a continuous basis.

class
The respiratory tract classification scheme in ICRP Publication 30 for inhaled material according to its rate of clearance from the pulmonary region of the lung. Materials are classified as D (days), W (weeks), or Y (years), according to how fast they clear the lungs; the half-times are class D in less than 10 days, class W in 10 to 100 days, class Y in more than 100 days. Recent recommendations in ICRP Publication 66 modified the lung model; instead of class D, W, and Y, the standard nomenclature is now lung absorption types F (fast), M (moderate), and S (slow).

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

depleted uranium
Uranium having a percentage of ^{235}U smaller than 0.7% found in natural uranium.

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

dose equivalent (*H*)
The product of the absorbed dose *D*, the quality factor *Q*, and any other modifying factors *N*. Dose equivalent is expressed in units of rem (or sievert), where 1 rem equals 0.01 sievert.

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

enriched uranium
Uranium that has been processed to contain a higher concentration of the isotope ^{235}U than in natural uranium.

exposure
The general condition of being subjected to ionizing radiation, such as by exposure from external sources or to sources inside the body.

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

intake
The amount of radionuclide taken into the body by inhalation, absorption through intact skin, injection, or ingestion or through wounds. Depending on the radionuclide involved, intakes can be reported in units of mass, activity, or potential alpha energy.

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

internal dose assessment
An assessment of the intake and associated internal radiation dose to workers based on measurements taken in the work environment or from individual bioassay measurements.

in vitro
Outside the living body and in an artificial environment. Typically used for bioassay of a contaminant in excreta, such as in fecal or urine samples. The word derives from Latin meaning *in glass*.

in vivo
In the living body of a plant or animal. Bioassay counting analysis of radionuclides in the human body. The word derives from Latin meaning *in [something] alive*.

isotope
Nuclides having the same number of protons in the nuclei (same atomic number) but having differing numbers of neutrons (different mass number).

lung solubility type (F, M, or S)
A classification scheme for inhaled material according to its rate of clearance from the pulmonary region of the lung.

minimum detectable amount (MDA)
The smallest amount (activity or mass) of an analyte in a sample that will be detected with a probability of nondetection (Type II error) while accepting a probability of erroneously deciding that a positive (nonzero) quantity of analyte is present in an appropriate blank sample (Type I error).

minimum detectable concentration (MDC)
The minimum detectable amount expressed in units of concentration.

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

natural uranium
Uranium occurring in the natural state that has not been through a ^{235}U enrichment process.

occupational dose
An individual's ionizing radiation dose (external and internal), which results from that individual's work assignment. Occupational dose does not include doses received as a medical patient or doses from background radiation or participation as a subject in medical research programs.

rad
A unit of absorbed dose equal to 100 ergs per gram of irradiated material.

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

radionuclide
A radioactive species of an atom characterized by the constitution of its nucleus specified by the total of the number of protons and neutrons.

rem
A unit of dose equivalent, equal to the product of the absorbed dose in rad and the quality factor, where 1 rem equals 0.01 sievert. The word derives from *roentgen equivalent in man*.

routine monitoring
Monitoring carried out at regular intervals during normal operations.

type
The rate of material absorption from the lung to the blood; includes types F (fast), M (moderate), and S (slow).

working level (WL)
That concentration of short-lived ^{222}Rn decay products (^{218}Po, ^{214}Pb, ^{214}Bi, and ^{214}Po) or short-lived ^{220}Rn decay products (^{216}Po, ^{212}Pb, ^{212}Bi, ^{212}Po) in one liter of air that will result in the eventual emission of 1.3×10^5 MeV of alpha energy.

working level month (WLM)
Exposure to a concentration of radon decay products in air equal to 1.0 working level (WL) for 170 hours.