DOE Review Release 03/21/2013

Document Title:	Document Number:	ORAUT-TKBS-0028-5
Weldon Spring Plant – Occupational Internal Dose	Revision:	01
	Effective Date:	03/15/2013
	Type of Document:	TBD
	Supersedes:	Revision 00

Subject Expert(s): David P. Harrison and Eugene W. Potter

Approval:	Signature on File David P. Harrison, Document Owner	Approval Date:	03/08/2013
Concurrence:	Signature on File John M. Byrne, Objective 1 Manager	Concurrence Date:	03/11/2013
Concurrence:	Signature on File Edward F. Maher, Objective 3 Manager	Concurrence Date:	03/08/2013
Concurrence:	Vickie S. Short Signature on File for Kate Kimpan, Project Director	Concurrence Date:	03/08/2013
Approval:	Signature on File Stuart L. Hinnefeld, Director DCAS	Approval Date:	03/15/2013

☐ New ☒ Total Rewrite ☐ Revision ☐ Page Change

FOR DOCUMENTS MARKED AS A TOTAL REWRITE, REVISION, OR PAGE CHANGE, REPLACE THE PRIOR REVISION AND DISCARD / DESTROY ALL COPIES OF THE PRIOR REVISION.

PUBLICATION RECORD

EFFECTIVE DATE	REVISION NUMBER	DESCRIPTION
06/28/2005	00	First approved issue of new technical basis document for the Weldon Spring Plant – Occupational Internal Dose. Incorporated formal internal and NIOSH review comments. Training is not required. Initiated by Robert Meyer.
03/15/2013	01	Incorporates numerous minor corrections and edits. SRDB numbers and page numbers were added. In Section 5.2.1, the specific activity for slightly enriched (1%) uranium was adjusted to agree with the formula in DOE 2001a. Sections 5.2.2 related to uranium decay products were edited to reflect the fact that the early uranium mills may not have been effective in removing thorium and to increase the amount of Th-230 and daughters to assume in calculations. The maximum concentrations for certain decay products at the WSRP and WSQ were eliminated. A discussion was added in Section 5.2.3 on potential intakes from thoron. Site-specific ratios of Th-230 to other contaminants were developed in Section 5.6.1.1 for use during initial uranium processing. DWA concentrations for thorium dust measurements were added as Attachment A, and these values were used to create a new table of thorium intakes in Section 5.6.1.2. Thoron guidance was also added to this section. A statement was added in Section 5.6.1.1 to use Friday urine sampling data statistics to avoid underestimating intakes. The estimated annual exposure from radon was increased in Section 5.6.1.3, and Section 5.2.4 on recycled uranium was updated in response to issues raised in the Advisory Board Work Group. Thorium-232 intake rates updated to reflect an 8-hour workday normalized to a calendar for each of the years for Th-232 operations, and equations were included in the text regarding calculation of the median and 95th percentiles of the Th-232 intake rates. Incorporates formal internal and NIOSH review comments. Constitutes a total rewrite of the document. Training required: As determined by the Objective Manager. Initiated by David P. Harrison.

TRADEMARK INFORMATION

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

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
CFR	Code of Federal Regulations
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
DOL	U.S. Department of Labor
DWA	daily weighted average
EEOICPA	Energy Employees Occupational Illness Compensation Program Act of 2000
g	gram
GSD	geometric standard deviation
hr	hour
ICRP	International Commission on Radiological Protection
IMBA	Integrated Modules for Bioassay Analysis
kg	kilogram
L	liter
lb	pound
m	meter
MAC	maximum allowable concentration
MCW	Mallinckrodt Chemical Works
MDA	minimum detectable amount
MDC	minimum detectable concentrations
MeV	megaelectron-volt, 1 million electron-volts
min	minute
mL	milliliter
MPC	maximum permissible concentration
mrem	millirem
nCi	nanocurie
NCRP	National Council on Radiation Protection and Measurements
NIOSH	National Institute for Occupational Safety and Health
NLO	National Lead of Ohio
ORAU	Oak Ridge Associated Universities

PAEC	potential alpha energy concentration
pCi	picocurie
POC	probability of causation
ppb	parts per billion
PUREX	plutonium-uranium extraction
SRDB Ref ID	Site Research Database Reference Identification (number)
U.S.C.	United States Code
WL	working level
WLM	working level month
WSCP	Weldon Spring Chemical Plant
WSP	Weldon Spring Plant
WSQ	Weldon Spring Quarry
WSRP	Weldon Spring Raffinate Pits
WSSRAP	Weldon Spring Site Remedial Action Project
yr	year
α	alpha particle
μCi	microcurie
μg	microgram
μL	microliter
μm	micrometer
μs	microsecond
§	section or sections

5.1 INTRODUCTION

Technical basis documents and site profile documents are not official determinations made by the National Institute for Occupational Safety and Health (NIOSH) but are rather general working documents that provide historical background information and guidance to assist in the preparation of dose reconstructions at particular sites or categories of sites. They will be revised in the event additional relevant information is obtained about the affected site(s). These documents may be used to assist NIOSH staff in the completion of the individual work required for each dose reconstruction.

In this document the word “facility” is used as a general term for an area, building, or group of buildings that served a specific purpose at a site. It does not necessarily connote an “atomic weapons employer facility” or a “Department of Energy [DOE] facility” as defined in the Energy Employees Occupational Illness Compensation Program Act [EEOICPA; 42 U.S.C. § 7384l(5) and (12)]. EEOICPA defines a DOE facility as “any building, structure, or premise, including the grounds upon which such building, structure, or premise is located … in which operations are, or have been, conducted by, or on behalf of, the Department of Energy (except for buildings, structures, premises, grounds, or operations … pertaining to the Naval Nuclear Propulsion Program)” [42 U.S.C. § 7384l(12)]. Accordingly, except for the exclusion for the Naval Nuclear Propulsion Program noted above, any facility that performs or performed DOE operations of any nature whatsoever is a DOE facility encompassed by EEOICPA.

For employees of DOE or its contractors with cancer, the DOE facility definition only determines eligibility for a dose reconstruction, which is a prerequisite to a compensation decision (except for members of the Special Exposure Cohort). The compensation decision for cancer claimants is based on a section of the statute entitled “Exposure in the Performance of Duty.” That provision [42 U.S.C. § 7384n(b)] says that an individual with cancer “shall be determined to have sustained that cancer in the performance of duty for purposes of the compensation program if, and only if, the cancer … was at least as likely as not related to employment at the facility [where the employee worked], as determined in accordance with the POC [probability of causation[1]] guidelines established under subsection (c) …” [42 U.S.C. § 7384n(b)]. Neither the statute nor the probability of causation guidelines (nor the dose reconstruction regulation, 42 C.F.R. Pt. 82) restrict the “performance of duty” referred to in 42 U S. C. § 7384n(b) to nuclear weapons work (NIOSH 2010b).

The statute also includes a definition of a DOE facility that excludes “buildings, structures, premises, grounds, or operations covered by Executive Order No. 12344, dated February 1, 1982 (42 U.S.C. 7158 note), pertaining to the Naval Nuclear Propulsion Program” [42 U.S.C. § 7384l(12)]. While this definition excludes Naval Nuclear Propulsion Facilities from being covered under the Act, the section of EEOICPA that deals with the compensation decision for covered employees with cancer [i.e., 42 U.S.C. § 7384n(b), entitled “Exposure in the Performance of Duty”] does not contain such an exclusion. Therefore, the statute requires NIOSH to include all occupationally-derived radiation exposures at covered facilities in its dose reconstructions for employees at DOE facilities, including radiation exposures related to the Naval Nuclear Propulsion Program. As a result, all internal and external occupational radiation exposures are considered valid for inclusion in a dose reconstruction. No efforts are made to determine the eligibility of any fraction of total measured exposure for inclusion in dose reconstruction. NIOSH, however, does not consider the following exposures to be occupationally derived (NIOSH 2010b):

- Background radiation, including radiation from naturally occurring radon present in conventional structures
- Radiation from X-rays received in the diagnosis of injuries or illnesses or for therapeutic reasons

[1] The U.S. Department of Labor (DOL) is ultimately responsible under the EEOICPA for determining the POC.

5.1.1 Purpose

The purpose of this technical basis document (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. Section 5.6 details assessment of radionuclide intakes. Section 5.7 provides a summary of instructions to dose reconstructors (DRs). Attributions and annotations, indicated by bracketed callouts and used to identify the source, justification, or clarification of the associated information, are presented in Section 5.8.

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 ca. 1964, p. 4; Mason 1977, p. 29).

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, p. 3). Worker exposures to airborne uranium were, in some cases, many times the tolerance level.

During the 24 years of operation, the MCW Uranium Division processed uranium ores and concentrates into uranium trioxide (UO_3), uranium hexafluoride (UF_6), and uranium metal (Meshkov et al. 1986, p. 27). Uranium dioxide (UO_2) was an intermediate product in the conversion of UO_3 to UF_4 (ORAUT 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 lower concentrations of decay products. The historical documents report that WSP did not receive the high-quality pitchblende ores (Ingle 1991, pp. 3-7). The WSP processed thorium during the 3 years before closure in 1966 (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 ca. 1964, pp. 5-29; Mason 1977, p. 29), 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 *Weldon Spring Plant – Occupational External Dosimetry* (ORAUT 2013). This section covers internal deposition of the radionuclides of concern.

There were four periods in the history of the WSP:

- Operational: 1957 to 1966
- Transfer to the U.S. Army: 1967 to 1974
- Environmental monitoring: 1975 to 1985
- Remediation: 1985 to 2002

The WS Quarry was initially transferred from the Army to the AEC in 1958. The operational period and the remediation period are of primary concern in relation to internal dosimetry at WSP, with some potential applicability of shutdown operations in 1967 before transfer to the U.S. Department of Defense in December 1967. The raffinate pits and the quarry were not transferred to the Army. From 1968 to early 1969 decontamination and dismantling operations commenced to support the herbicide production. However, the defoliant project was canceled in February 1969 whereupon the chemical plant entered a care and custody status by the Army. The AEC didn't have any contractors at the WS Raffinate Pits or the WS Quarry until August 1975 for environmental monitoring. From 1969 to 1981 the status of the site did not change, and from 1981 to 1985 the site was placed in caretaker status. The WS Chemical Plant was transferred from the Army to the DOE in 1985, and remediation efforts began in 1985. 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, pp. 154-166) 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 *Weldon Spring Plant – Site Description* (ORAUT 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.

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

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 1963 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 et al. (1986, pp. 27-33) and Lesperance, Siegel, and McKinney (1992, pp. 30-36).

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% U (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;	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.

Bldg. no.	Building description	Building operations	Potential radionuclide exposures
		acceptable cores shipped to reactor sites.	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 (1965–1966). 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 ORAUT (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

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, p. 5) 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 radium and 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, p. 6) 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 ORAUT (2013), 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, p. 6), 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, p. 6) 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.*

In line with this policy, urine and air sampling programs were conducted during the operational period to monitor the employees' potential exposures.

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 2-7 in ORAUT (2005b) 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	122,015,977	98.43
Depleted U	167,823	0.14
Slightly enriched U	842,585	0.68
Natural thorium	941,347	0.76
Total	123,967,732	100

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 Integrated Modules for Bioassay Analysis (IMBA) program for natural and depleted uranium.

It is reasonable to assume that the slightly enriched uranium processed at Weldon Spring was 1% enriched with a specific activity of 0.783 pCi/μg (DOE 2001a, p. 2-7). Bioassay results in units of mass may be converted to activity using this value. After intake rates are calculated, doses are assessed assuming that the activity is 100% ^{234}U.

Although uranium with enrichments of less than 1% might have been processed at WSP, it is favorable to claimants 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 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 mill 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* 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, p. 18). However, it has been observed that the early mills did not remove thorium as effectively as indicated by this reference. As noted in *Weldon Spring Plant – Site Description* (ORAUT 2005b), four raffinate pits were constructed between 1958 and 1964 to contain process wastes from the WSP. Measurements of the activity concentrations in Raffinate Pits 1, 2, and 3 can be used to determine the relationship between thorium-230 and other impurities during the initial uranium processing (feed preparation and sampling) in Building 101, during the transfer of ore concentrate from Building 101 to Building 103 for nitric acid digestion, and transfer of the uranyl nitrate slurry to the aqueous feed tanks in Building 105 before any separations occurred, as discussed in Section 5.6. The shorter lived decay products, ^{210}Pb and ^{210}Po, for which no raffinate measurements were made should be assumed to be present at the same activity as the ^{226}Ra in raffinate pits.

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, p. 20).

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 at the WSQ between 1959 and 1966.

5.2.3 Natural Thorium and Decay Products

In contrast to the buildup of decay products of ^{238}U, the conservative (favorable to claimant) 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 (5.7-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). For assessing dose from ^{232}Th, it is assumed that ^{228}Th is in equilibrium with ^{232}Th and ^{228}Ra is added at a ratio of 1-to-2 ^{232}Th-to-^{228}Ra (NIOSH 2010a, p. 57). The *Weldon Spring Plant – Occupational Environmental Dose* (ORAUT 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 (ORAUT 2005c).

Thoron (^{220}Rn) is the second daughter product of ^{228}Th, and after a couple of weeks following the processing of thorium ores for thorium purification, can be considered to be in full equilibrium with the parent, ^{228}Th. Thorium-228 is generally in 40-65% equilibrium with ^{232}Th for materials processed at the WSP. The degree of equilibrium is dependent upon both the decay of ^{228}Th (without replenishment from the 5.7 yr half-life of ^{228}Ra) after removal of the thorium daughters and the time it takes the ^{228}Ra to build into equilibrium with ^{232}Th. Figure 5-1 shows the decay and buildup of ^{228}Th and daughters following processing.

Thoron was present and a portion was released during the processing and storage of thorium at the WSP and the associated waste storage locations. The thoron, with its subsequent daughter products, would act as a potential source of internal exposure in the thorium process buildings and at waste storage locations.

There are several aspects of the release and buildup of thorium daughter products that mitigate the exposure to thoron in process and storage configurations. These include the following:

- Because of the very short half-life of thoron (56 seconds), much of the isotope decays within the material matrix; therefore, the diffusion distance in soils is in the range of approximately one inch. Only the quantities of thoron in the first inch of material represent the source term. Table 5-5 provides information for thoron and its daughters.

- In process equipment, the release of thoron gas is largely contained within the containment system and/or the ventilation system. Essentially, the dose from exposure to thoron comes entirely from the daughters ^{212}Pb and ^{212}Bi. A thoron working level (WL) requires 375 pCi/L

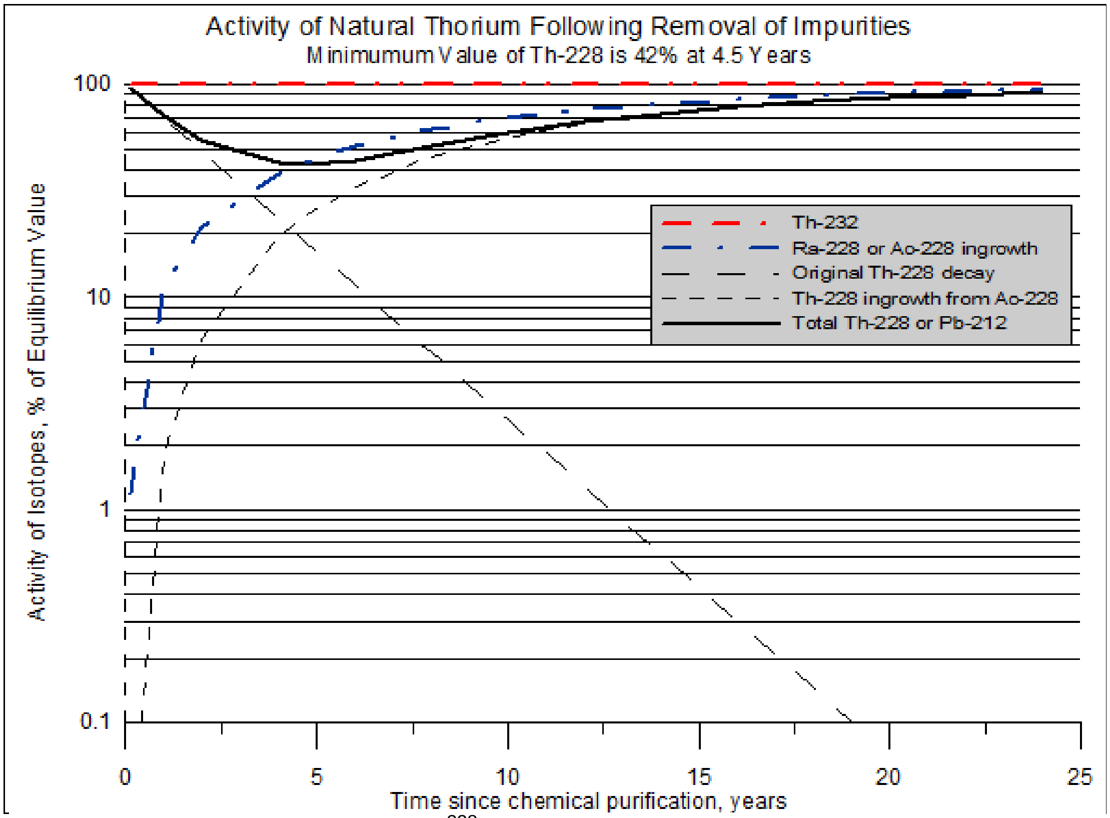

Figure 5-1. Decay and buildup of ^{228}Th and daughters following processing.

Table 5-5. Thoron and isotopic daughters.

Isotope	Half life	Atoms/7.5 pCi	α MeV/atom	PAEC[a] MeV per 7.5 pCi
Radon-220	56 sec	23	14.6	335
Polonium-216	0.15 sec	<1	b	-
Lead-212	10.64 hr	15,476	7.8	1.21E+5
Bismuth-212	60.6 min	1,469	7.8	0.12E+5
Polonium-212	0.3 µs	-	b	-
Thallium-208	3 min	73	no α	-
			Total	**1.33E+5**

a. PAEC = potential alpha energy concentration.

b. The alpha energy is emitted only by Po-216 and essentially Bi-212. In reality Bi-212 emits an alpha only 36% of the time but the other 64% of the time it emits a beta then an alpha almost immediately through the decay of Po-212 (304 nsec half-life). The average alpha energy released by a Bi-212 decay is then 7.8 MeV (6.07 Mev * 0.36 + 8.785 MeV * 0.64). Since Po-216 ultimately decays by two alphas (Po-216 and Bi-212) the alpha energy released per Po-216 atom is 14.6 MeV. Pb-212 and Bi-212 both decay with an effective alpha energy of 7.8 MeV.

thoron and ^{216}Po to produce measured daughter products ^{212}Pb and ^{212}Bi at 7.5 pCi/L assuming a conservative equilibrium factor of 0.02 [1].

The primary focus of the radiological safety programs for thorium was to define the air concentrations in the work place, the results of which provided the means of controlling worker exposures to levels below the permissible levels. Weldon Spring Plant records do not indicate specific analyses to define the concentrations of thoron daughter activities; however, it was standard practice to provide delayed counts on the air samples for up to 96 hours to allow the short-lived radon and thoron daughter products to decay.

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, and thus would contribute only a very small fraction of the potential dose. In contrast, the estimated maximum ^{232}Th concentration in materials deposited at the WSQ was 4200 pCi/g (DOE 1991, p. 18).

5.2.4 Recycled Uranium

The extent of the processing of recycled uranium at WSP is not well known (ORAUT 2005b, Section 2.2.3). The DR should make the favorable to claimant 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, p. 118), 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). For the periods that included recycled uranium, the DR should consider the ppb conversion factors in ORAUT-TKBS-0017-5, *Technical Basis Document for the Fernald Environmental Management Project – Occupational Internal Dose* (ORAUT 2004). These factors, when multiplied by the calculated 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 ^{99}Tc. For example, 1 ppb-Pu/U equals 1×10^{-9} gm-Pu/gm-U. For ^{239}Pu the ppb conversion factor is 62.89 pCi-Pu/gm-U per ppb, for ^{237}Np the factor is 0.714 pCi-Np/gm-U per ppb, and for ^{99}Tc the factor is 17.15 pCi-Tc/gm-U per ppb.

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 et al. 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, p. 18).

The WSP remediation contractor collected samples of materials from the areas in WSCP buildings with the highest levels of contamination (DOE 2001b, p. 14). 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 International Commission on Radiological Protection (ICRP) Publication 30 designations of D, W, and Y (ICRP 1979). These classes were generally assigned to the more recent ICRP absorption types of F, M and S, respectively (ICRP 1995).

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 information in Table 5-6 is provided for general information only and guidance that says the most favorable material solubility type (F, M, or S) should be selected in the absence of definitive information for a particular dose reconstruction.

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

Area	U-234			U-235			U-238			Th-232		Th-230		Th-228	
	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

5.2.6 Particle Size

Lacking specific information on the particle size for airborne uranium at WSP since there were no particle size distribution studies conducted, the ICRP (1994b) value of 5 µm aerodynamic median activity diameter (AMAD) should be used (ORAUT 2007b).

5.3 *IN VITRO* MEASUREMENTS

5.3.1 Operational Period (1957 to 1966)

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 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 (ORAUT 2007) and was based on modern MDA concepts (HPS 1996, pp. 44-49). The Rocky Flats MDA value included the contribution of a blank subtraction and was based on a 100λ (100 µL) aliquot from a 24-hour urine sample. This MDA value has a known pedigree and is recommended as the MDA for WSP, although there may be some differences in methods (e.g., grab sample versus 24-hour sample, unknown blank subtraction versus blank subtraction, and unknown volume of the aliquot versus 100λ (100 µL)).

An estimate of the uranium MDA can be derived from Dupree (1979b, p. 26), 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 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, p. 26) 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 2001b, p. 23)." 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 fluorometric 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

Work group	Cost center code	
	To early 1963	Starting 1963
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 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.

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.

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. A recorded value of 7.8 mg/L was deleted from the 1958 data set as an outlier. This does not affect the median or 95th-percentile value. In cases where there were five or fewer records, the tables contain only the maximum urine bioassay result.

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

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 2002)

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, 2001b). 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

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

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

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.

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	0.018	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.

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	0.015	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

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 Weldon Spring Plant – Occupational Environmental Dose (ORAUT, 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, p. 29) using laser fluorimetry and as 0.1 µg/L in 1994 to 1998 (DOE 1994, p. 31; 1997, p. 22; 1998a, p. 21) and 0.0524 µg/L in September 1998 to 2001 (DOE 1998b, p. 20; 1998c, p. 20; 2000a, p. 20; 2001b, p. 22) using kinetic phosphorescence analysis. Uranium results of 0.2 µg/L or greater were considered positive for occupational uranium intakes in 1997 (DOE 1997, p. 23), and results of 0.3 µg/L or greater (DOE 1998c, p. 21; 2000a, p.21, 2001b, p. 23) were considered positive for occupational uranium intakes from 1998 through 2001.

5.4 *IN VIVO* MEASUREMENTS

5.4.1 Operational Period (1957 to 1966)

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 (natural thorium or ^{232}Th) 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, p. 6)

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 2002)

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, p. 28).

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 2001b, p. 20). Revision 7 (DOE 2001b, p. 20) 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 2001b, p. 30) states that such measurements were normally reserved for “those incidents where the intake was suspected to exceed 500 mrem CEDE ….” Revision 7 (DOE 2001b, p. 30) 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 1966)

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, p. 16).

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, p. 16). 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 8-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, p. 114). 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 (HPS 1960, Volume 3, p. 112). However, a provisional level of 3×10^{-11} µCi/cm^3 was recommended. This is the value that was adopted in Annex 1 of AEC Manual Chapter 0524, “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, p. 3):

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

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 8-hour workday 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. The results of the dust studies are summarized in the following section.

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 ca. 1966, pp. 5-46) 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-hour delay to allow for the decay of radon progeny. Samples from thorium areas were counted after a minimum delay of 100 hours.

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 (natural thorium, ^{232}Th) airborne dust concentrations were routinely measured and evaluated throughout the operating period. These data, in the form of DWA concentrations, were summarized for the thorium processing period 1963 through 1966 (MCW ca. 1966, pp 15-47) and are reproduced in Attachment A. 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 5-19 summarizes these calculations.

Table 5-19. Time-weighted average ^{232}Th 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

Operation	Bldg.	Year	Fraction of MAC	Total effort
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 = thorium nitrate tetrahydrate.

5.5.2 Transfer to the U.S. Army (1967 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 (NLO) 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 personnel dosimetry programs for this period.

5.5.4 Remediation Period (1985 to 2002)

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 ORAUT-TKBS-0028-2, *Weldon Spring Plant – Site Description* (ORAUT 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 for each nuclide. The calculated minimum detectable concentrations (MDCs) for a 6-hour 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 hour a day 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, pp. 45, 49) required BZA samples for one in five individuals spending more than 1 hour a day 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, p. 32), 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 2001b, p. 32), 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 Weldon Spring Plant – Occupational Environmental Dose (ORAUT, 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 1966)

5.6.1.1 Uranium and Related 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 5-8 to 5-17 (Section 5.3.1.1.6) 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). The statistical data associated with Friday urine sampling should be used in dose reconstructions to avoid underestimating uranium intakes.

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 1966

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. Measurements of the activity concentrations in Raffinate Pits 1, 2, and 3 can be used to determine the relationship between ^{230}Th and other impurities during the initial uranium processing (feed preparation and sampling) in Building 101, during the transfer of ore concentrate from Building 101 to Building 103 for nitric acid digestion, and transfer of the uranyl nitrate slurry to the aqueous feed tanks in Building 105 before any separations occurred. The shorter-lived decay products for which no raffinate measurements were made (e.g., ^{210}Pb and ^{210}Po) can be assumed to be present at the same activity as their ^{226}Ra parent in the mill concentrate feeds. The other uranium streams (e.g., recycled uranium) had been previously processed and contained essentially no thorium. Table 5-20 gives the results of a statistical analysis of raffinate pit measurements. The data were taken from the 1989 Waste Assessment Radiological Characterization of the Weldon Spring Site Raffinate Pits (MK-Ferguson 1989, pp. 72-75).

Table 5-20. Activity ratios of uranium decay products and other impurities to ^{230}Th activity in Weldon Spring raffinate pits.

	Pits 1		**Pit 2**		**Pit 3**		**Pit 4**	
Isotope	**Mean**	**95th %**	**Mean**	**95th %**	**Mean**	**95th %**	**Mean**	**95th %**
Ra-226	0.090	0.367	0.021	0.032	0.021	0.034	0.197	0.585
Ra-228	0.004	0.013	0.005	0.006	0.004	0.011	0.251	1.003
Th-228	0.004	0.011	0.004	0.006	0.006	0.010	0.326	1.508
Th-232[a]	0.006	0.016	0.006	0.009	0.009	0.016	0.326	1.545

a. Measurements of thorium-232 in Raffinate Pits 1, 2, and 3 were not used due to alpha peak interference from thorium-230 (MK-Ferguson 1989, p. 75). Thorium-232 was calculated assuming that thorium-228 was in 65% equilibrium, except for Pit 4 where reliable measurements were available.

The mean values in Table 5-20 represent the amount of each isotope that as a fraction of thorium-230 an employee would have been exposed to over the long term. It is unlikely that any employee would have been continuously exposed to fractions greater than the 95th percentile; therefore, the 95th percentile of the ratios to thorium-230 in Table 5-20 can be used to bound the intakes from the other potentially important radionuclides. The data for Pit 4 were not used in this analysis since natural thorium wastes were deposited there. The activity concentrations for the uranium decay products and impurities are determined by multiplying the 95th-percentile ratios in Table 5-20 by their corresponding ^{230}Th concentration value for Pits 1, 2, and 3 stated in Table 5-21.

Table 5-21. Intakes of uranium decay products and other impurities (nCi/mg-U) based on raffinate pit measurements.

Isotope	**Intakes based on Pit 1**	**Intakes based on Pit 2**	**Intakes based on Pit 3**	**Bounding intakes[a]**
Pb-210[b]	0.125	0.007	0.011	0.125
Po-210[b]	0.125	0.007	0.011	0.125
Ra-226	0.125	0.007	0.011	0.125
Ra-228	0.005	0.002	0.002	0.005
Th-228	0.004	0.001	0.002	0.004
Th-230[c]	0.340	0.340	0.340	0.340
Th-232	0.006	0.002	0.003	0.006

a. Indicates the higher value of intakes based on Pits 1, 2, and 3.
b. Measurements assumed to be equal to the intake of Ra-226.
c. Th-230 is assumed to be equal to the U-234 activity, or one-half the total natural uranium activity.

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 favorable to claimants.

5.6.1.2 Thorium-232 Intakes

No quantitative *in vitro* bioassay results have been observed for thorium (Sections 5.3.1.2 and 5.4.1). Late in the thorium processing period a limited amount of *in vivo* counting was performed. The *in vivo* data should be considered if available. However, dust studies for thorium operations (Section 5.5.1.4 and Appendix A) were routine (MCW 1966). The dust studies are presented in terms of daily weighted activity (DWA) or daily weighted exposure (DWE). The DWA data provide a clear picture of when (on a monthly timeframe) and where thorium processing and packaging operations occurred. The original air sampling data used to generate the DWA value for each job have not been located, so an alternate approach to characterizing the variability of the exposure in the workplace is necessary. Using Atomic Weapon Employer data, Davis and Strom (2008) demonstrated that the point estimate DWA value will favorably characterize the exposure potential by assuming that the DWA value is the arithmetic mean of a lognormal distribution with a geometric standard deviation (GSD) of 5. Air concentrations for each thorium operation are obtained from DWA summary tables (Attachment A). Thus, for example, in March 1965 the DWA value assumed to represent the arithmetic mean (AM) of exposure to thorium workers performing the task "Oven drying ThO_2 sol-pan transfer" was 3 alpha dpm per m^3. It is assumed that the reported DWA value is the AM from a lognormal distribution with a geometric standard deviation (GSD) of 5. Equation 2 from Strom and Stansbury (2000) defines the relationship between GSD and σ

$$\sigma = ln(GSD),$$

where σ is the standard deviation of the natural logarithm of the observations in a lognormal distribution. Equation 13 from Strom and Stansbury (2000) is used to compute the median which is equal to the geometric mean (GM) in a lognormal distribution:

$$GM = AM * e^{-\sigma^2/2}$$

Table 2-2 in Strom (2007) includes the following formula for μ, the natural logarithm of the GM:

$$\mu = ln(GM)$$

Equation 5 in Strom (2007) defines the 95th-percentile value of the lognormal distribution:

$$95th\ percentile = e^{(\mu+1.645\sigma)}$$

Thus, a lognormal distribution with AM = 3 and GSD = 5 has a GM of 0.82 and a 95^{th} percentile value of 11.6 alpha dpm/m^3.

Inhalation intake rates based on the median and 95th percentile DWA ^{232}Th air concentration (calculated using the equations stated above and the DWA values from Attachment A) are calculated assuming a breathing rate of 1.2 m^3 per hour and an 8-hour workday. This 8-hour workday is normalized to a calendar day based on the number of workdays in a year. The number of workdays in 1963 was 43 (21 for November and 22 for December), there were 250 workdays per each year in 1964 and 1965, and there were 195 workdays in 1966 since thorium operations ceased in September 1966. The ingestion intake rate is based on guidance in NIOSH (2004) and ORAUT (2012). The ingestion intake rates, which are stated in Table 5-22, were derived from the inhalation intake rates in Table 5-22 by multiplying the inhalation rate by 0.02.

The original version of this site profile cautioned that Work Unit data at Weldon Spring may not be a reliable indicator of work assignment. This caution was based on conditions that existed at the Mallinckrodt Destrehan Street facility, the predecessor to the Mallinckrodt-operated Weldon Spring Plant. An interview with a former worker established that Work Unit designation at Weldon Spring

Table 5-22. Thorium-232 inhalation and ingestion intake rates (Bq per year).

Plant name and building number	Work group title and cost center	Inhalation (Bq/year)				Ingestion (Bq/year)			
		1963	**1964**	**1965**	**1966**	**1963**	**1964**	**1965**	**1966**
Sampling - 101	Sampling - 0402	0	0	125	304	0	0	2	6
	Other occupants of Sampling Plant	0	0	9	22	0	0	0	0
Refinery - 103	Digest-Raffinate - 0501	196	4810	5100	2320	4	96	102	46
	Other occupants of Refinery	14	341	360	165	0	7	7	3
Extraction - 105	Extraction	0	52	0	0	0	1	0	0
	Other occupants of Extraction Plant	0	4	0	0	0	0	0	0
Metals - 301	Metals - other than Reduction 0702	47	1010	21550	1595	1	20	431	32
	Other occupants of Metals Plant	3	72	1525	113	0	1	31	2
Scrap Plant - 403	Scrap Plant - 0601	645	14700	2535	0	13	294	51	0
	Other occupants of Scrap Plant	46	1040	180	0	1	21	4	0
Undetermined location	Any worker not otherwise specified above whose work location is unknown	46	1040	1525	165	1	21	31	3

was reliable (ORAUT 2009). Consequently, the Work Unit (also called Work Group or Job Code) designation can be used as an indicator of potential thorium exposure.

The ^{232}Th inhalation intake rates were calculated on a month-by-month basis for the work tasks associated with each location where thorium was processed or handled. The maximum monthly intake values were determined by facility, which were then summed for each year of thorium operations and summarized in Table 5-22. When more than one task occurred in a location during a month, the task yielding the highest intake rate was used in the calculation. Operators in Work Groups identified as being directly involved in the process are assigned intakes based on the 95th-percentile value of the lognormal DWA distribution (the intake value in the top line for each group in Table 5-22) for the most highly exposed task in each plant. Other occupants of the same plant are assigned intakes based on the 50th-percentile value, which is the intake value in the bottom line for each group in Table 5-22. For workers in job categories, for example maintenance or janitorial services, who may have been exposed but whose location cannot be reliably established, an intake rate based on the 50th-percentile inhalation value for the plant yielding the highest intake value during that year is recommended. For example, the highest median value (50th percentile) for 1963 is 46 Bq per year from the Scrap Plant. Workers with undetermined work locations would be assigned an inhalation intake of 46 Bq per year ^{232}Th and an ingestion intake of 1 Bq per year ^{232}Th for 1963. Thorium-228 should be assumed to be in equilibrium with ^{232}Th and thus the activities of ^{228}Th are equal to the ^{232}Th activities stated in Table 5-22. Radium-228, which is a beta emitter and would not have been detected on the alpha air sample, is added at a ratio of 1-to-2 ^{232}Th-to-^{228}Ra (NIOSH 2010a, p. 57). For operational periods before or after the time frame of Table 5-22, no occupational internal dose from ^{232}Th is recommended. It is appropriate to recognize that the DWA data often are annotated with statements indicating that airline or half-mask respiratory protection equipment was used or prescribed in some of the thorium operations.

For purposes of calculating the ^{232}Th intake rates, the use of respiratory protection is ignored because the quality of the respiratory protection equipment and diligence in the use of the equipment cannot be established. Nevertheless, the routine use of respiratory protection equipment during thorium operations suggests that the thorium intake rates are quite likely to be biased in favor of the claimant.

For workers not in these buildings or professions during this period, thorium intakes should be assessed as environmental intakes in accordance with *Weldon Spring Plant – Occupational Environmental Dose* (ORAUT 2005c) unless there is information in the worker's file that indicates involvement in thorium operations.

5.6.1.3 Intakes of Other Radionuclides

The annual average environmental radon concentrations for the WSCP, WSRPs, and WSQ are provided in Table 4-5 of ORAUT (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 ORAUT (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. There are no personal dosimetry data or air concentration measurements available for radon in the WSCP during the operational period.

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 8.7×10^{-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, p. 47) states that the prime source of radon emissions was the acid recovery plant. The ^{222}Rn activity released annually was estimated to be 34 Ci per year (Meshkov et al. 1986, pp. 47–48), and this was assumed to be released from Building 103 ventilation stacks. 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. The ^{222}Rn equilibrium concentration is calculated by dividing the radon influx (^{222}Rn activity/pCi/hour) by the product of the number of air changes per hour (ach) and the volume of the ventilated area. Using the Building 103 volume of 2.6×10^4 m^3 (2.6×10^7 L) and 1 ach, the ^{222}Rn equilibrium concentration is calculated to be 150 pCi/L. A working level (WL) is defined as 100 pCi/L ^{222}Rn in full equilibrium with its short-lived alpha-emitting progeny. However, because full equilibrium is never achieved in occupied spaces, an indoor equilibrium factor of 0.5 is assumed. Using the number of work hours in a month of 170 in defining the WLM and a 2,000-hour working year, the estimated annual exposure is calculated to be 8.8 WLM for 100% occupancy for 2,000 hours per year.

As described in Section 5.2.3, there are several aspects of the release and buildup of the daughter products that mitigate the exposure to thoron in process and storage configurations.

From November 1963 through September 1966, natural thorium was processed on an intermittent batch basis in the refinery and oxide production/firing systems at the Weldon Spring Plant; therefore, NIOSH's evaluation of operational exposures to thoron at Weldon Spring is confined to those years. During 1966 (the maximum production year), the Weldon Spring Plant processed approximately 100,000 kg of thorium per month for 6 months, or up to 5,000 kg of ^{232}Th per day. Typical thorium processing at other facilities averaged <1,000 kg of thorium per day. NIOSH recognizes that the thorium processing was not continuous, but rather was operated more on a batch basis. Given the specific activity of thoron in thorium feed materials, and assuming the feed materials were received with at least a 1-year delay since processing, it is possible to calculate the amount of thoron in process per day (approximately 0.35 Ci thoron [2]) during the period of the maximum production rate. By assuming a conservative equilibrium factor for a plant configuration with large buildings and engineered ventilation (0.02) [3], it is possible to determine the concentration of thoron and its daughters to achieve 1 WL. The release fraction of the thoron from thorium can be calculated by comparing the particulate thorium in the working environment of the process equipment to the inventory amounts in process. The thoron releases are expected to be less than the particulate materials, since the gaseous state will be more easily captured by the ventilation systems.

Potential exposure to thoron is mitigated by several factors, as discussed in Section 5.2.1.6, including the very short half-life of thoron and the resulting limitations on the diffusion distance within the material matrix, and the effective shielding provided by the thorium processing apparatus. Thus, thoron is not a potential source of exposure to be broadly accounted for across the evaluated class of Weldon Spring workers. Using the distribution of values in Table 5-19, the 50th percentiles and 95th percentiles of DWA were estimated as 21 and 250 alpha dpm/m^3, respectively, over the entire thorium processing period. Assuming that this activity consisted solely of ^{232}Th, ^{228}Th, and ^{224}Ra, release fractions for particulates of approximately 7.6×10^{-7} and 9.0×10^{-6} were calculated for the 50th and 95th percentiles, respectively. As discussed previously, these would be conservative estimates for the thoron gas, which is more easily captured by the ventilation. A volume of 100,000 ft^3 (100 by 100 by 10 ft) was assumed for the processing area [4]. Using the equilibrium factor of 0.02, estimates of 2.9×10^{-3} and 3.5×10^{-2} WLM/yr for the thoron daughters were obtained for the 50th and 95th percentiles, respectively. These are conservative since they are based on the peak production rate of 5,000 kg ^{232}Th per day. DRs can apply these values to the 3-year period of thorium processing, depending on whether it appears that a worker was occasionally or routinely exposed to thorium operations.

5.6.2 Environmental Monitoring Period (1975 to 1984)

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

5.6.3 Remediation Period (1985 to 2002)

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 2001b, p. 36), 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 ORAUT (2005c).

5.7 SUMMARY OF INSTRUCTIONS TO DOSE RECONSTRUCTORS

Table 5-23 contains a summary of the instructions to DRs.

Table 5-23. Instructions to DRs.

Intake	Data/Information to be used
Operational Period (1957–1966)	
Uranium (U-234, U-235, U-238)	Primary: individual worker’s uranium bioassay data. Supplemental: workgroup data (Tables 5-8 through 5-17). Default isotopic assumptions: • Natural uranium, before 1961 • Natural uranium, recycled, 1961 to 1962 • Enriched (1%) uranium, recycled, 1963 to 1966, specific activity of 0.783 pCi/µg • Use Section 5.2.4 for recycled uranium contaminants
Uranium contaminants and decay products (less radon)	Bldg. 101, Bldg. 103, Bldg. 105 prior to separations: assigned based on uranium bioassay (Table 5-21). Others: NA, assume pure uranium after separations.
Thorium (Th-232 and Th-228) and Ra-228 (1963–1966)	Use calculated intake rates based on DWA estimates for workers involved in thorium processing (Table 5-22). Th-228 activity equal to Th-232 activity. Ra-228 activity equal to two times Th-232 activity.
Radon	8.8 WLM/yr for 100 percent occupancy for 2000 hours.
Thoron (1963–1966)	2.9E-03 and 3.5E-02 WLM/yr, for 50th and 95th percentiles, respectively.
Environmental Monitoring Period (1975–1984)	
Any	All intakes in this period should be assessed as environmental intakes using ORAUT-TKBS-0028-4.
Remediation Period (1985–2002)	
Any	The worker’s file will contain a detailed report of the pertinent data to be used for the assessment.

5.8 ATTRIBUTIONS AND ANNOTATIONS

Where appropriate in this document, bracketed callouts have been inserted to indicate information, conclusions, and recommendations provided to assist in the process of worker dose reconstruction. These callouts are listed here in the Attributions and Annotations section, with information to identify the source and justification for each associated item. Conventional References, which are provided in the next section of this document, link data, quotations, and other information to documents available for review on the Project's Site Research Database (SRDB).

[1] Rich, Bryce. Oak Ridge Associated Universities (ORAU) Team. Health Physicist. August 2010.
This statement is based on experience.

[2] Potter, Eugene. ORAU Team. Health Physicist. August 2010.
This statement assumes equilibrium with ^{228}Th, which is in 65% equilibrium with ^{232}Th.

[3] Rich, Bryce. ORAU Team. Health Physicist. August 2010.
This statement is based on experience.

[4] Potter, Eugene. ORAU Team. Health Physicist. August 2010.
This is a reasonable estimate for the size of the processing area.

REFERENCES

AEC (U.S. Atomic Energy Commission), 1963, "Standards for Radiation Protection," Chapter 0524, *AEC Manual*, Washington, D. C., August 12. [SRDB Ref ID: 13037]

Battelle Memorial Institute, Battelle-TIB-5000/PNWD-3741, 2007, *Technical Information Bulletin: Default Assumptions and Methods for Atomic Weapons Employer Dose Reconstructions, Rev 00*, April 2. [SRDB Ref ID: 32016]

Davis, A.J. and D.J. Strom, 2008, "Uncertainty and Variability in Historical Time-Weighted Average Exposure Data", pp. 145-160 of Health Physics Journal Vol 94(2), February 1. [SRDB Ref ID: 41616]

DOE (U.S. Department of Energy), 1986, *Historical Nuclear Materials Balance Report for the Former AEC-Owned Weldon Spring Chemical Plant*, DOE/OR-872, Weldon Spring, Missouri, July. [SRDB Ref ID: 8252]

DOE (U.S. Department of Energy), 1991, *Internal Dosimetry Program Technical Basis Manual: Weldon Spring Site Remedial Action Project*, Rev. 0, DOE/OR/21548-241, Oak Ridge Operations Office, Oak Ridge, Tennessee, November. [SRDB Ref ID: 79457]

DOE (U.S. Department of Energy), 1994, *Internal Dosimetry Program Technical Basis Manual: Weldon Spring Site Remedial Action Project*, Rev. 1, DOE/OR/21548-241, Oak Ridge Operations Office, Oak Ridge, Tennessee, September. [SRDB Ref ID: 79459]

DOE (U.S. Department of Energy), 1997, *Internal Dosimetry Program Technical Basis Manual: Weldon Spring Site Remedial Action Project*, Rev. 2, DOE/OR/21548-241, Oak Ridge Operations Office, Oak Ridge, Tennessee, January. [SRDB Ref ID: 79461]

DOE (U.S. Department of Energy), 1998a, *Internal Dosimetry Program Technical Basis Manual: Weldon Spring Site Remedial Action Project*, Rev. 3, DOE/OR/21548-241, Oak Ridge Operations Office, Oak Ridge, Tennessee, January. [SRDB Ref ID: 79465]

DOE (U.S. Department of Energy), 1998b, *Internal Dosimetry Program Technical Basis Manual: Weldon Spring Site Remedial Action Project*, Rev. 4, DOE/OR/21548-241, Oak Ridge Operations Office, Oak Ridge, Tennessee, September, Oak Ridge Operations Office, Oak Ridge, Tennessee. [SRDB Ref ID: 79467]

DOE (U.S. Department of Energy), 1998c, *Internal Dosimetry Program Technical Basis Manual: Weldon Spring Site Remedial Action Project*, Rev. 5, DOE/OR/21548-241, Oak Ridge Operations Office, Oak Ridge, Tennessee, November. [SRDB Ref ID: 79469]

DOE (U.S. Department of Energy), 2000a, *Internal Dosimetry Program Technical Basis Manual: Weldon Spring Site Remedial Action Project*, Rev. 6, DOE/OR/21548-241, Oak Ridge Operations Office, Oak Ridge, Tennessee, January. [SRDB Ref ID: 79471]

DOE (U.S. Department of Energy), 2000b, *Ohio Field Office Recycled Uranium Recovery Report*, Ohio Field Office. [SRDB Ref ID: 3644]

DOE (U.S. Department of Energy), 2001a, *Guide of Good Practices for Occupational Radiological Protection in Uranium Facilities*, DOE-STD-1136-2000, Change Notice No. 3, U.S. Department of Energy, Washington D.C., December. [SRDB Ref ID: 4617]

DOE (U.S. Department of Energy), 2001b, *Internal Dosimetry Program Technical Basis Manual: Weldon Spring Site Remedial Action Project*, Rev. 7, DOE/OR/21548-241, Oak Ridge Operations Office, Oak Ridge, Tennessee, December. [SRDB Ref ID: 32724]

Dupree, E. A., 1979a, *Preliminary Report on Mortality Study of MCW Workers,* Review Committee, Uranium Division, Oak Ridge Associated Universities, Oak Ridge, Tennessee, July 25. [SRDB Ref ID: 13182]

Dupree, E. E., 1998, "Assessment of Internal Radiation Exposure for Workers in the Mallinckrodt Cohort Mortality Study," letter to D. Utterback (National Institute for Occupational Safety and Health), Oak Ridge Associated Universities, Oak Ridge, Tennessee, July. [SRDB Ref ID: 16228]

Dupree, E. E., 1998, *Mallinckrodt Expanded Cohort Study, Proposed Work Plan for Internal Radiation Dose Assessment*, attachment to Letter to D. Utterback (National Institute for Occupational Safety and Health (NIOSH)), July 1998, Oak Ridge Associated Universities (ORAU). Oak Ridge, Tennessee. [SRDB Ref ID: 16228]

Eisenbud, M., 1975, "Early Occupational Exposure Experience with Uranium Processing," pp. 9–24 of *Conference on Occupational Health Experience with Uranium, Arlington, Virginia, 28-30 April 1975*, U.S. Energy Research and Development Administration. [SRDB Ref ID: 4613]

HPS (Health Physics Society), 1960, *Report of the ICRP Committee II on Permissible Dose for Internal Radiation (1959)*, Universities Press, Belfast, Ireland. [SRDB Ref ID: 11106]

HPS (Health Physics Society), 1996, *An American National Standard - Performance Criteria for Radiobioassay*, HPS N13.30-1996, McLean, VA. [SRDB Ref ID: 23441]

Ingle, J. N., 1991, "Mallinckrodt Chemical Works: Four Plant Study Classifications of Radium, Radon and Thorium Exposure," memorandum, NIPF AU 005101, Oak Ridge Associated Universities, Oak Ridge, Tennessee, September 16. [SRDB Ref ID: 12400]

ICRP (International Commission on Radiological Protection), 1979, *Limits for Intakes of Radionuclides by Workers*, ICRP Publication 30, Part 1, Pergamon Press, Oxford, England. [SRDB Ref ID: 29220]

ICRP (International Commission on Radiological Protection), 1994a, *Dose Coefficients for Intakes of Radionuclides by Workers*, Publication 68, Pergamon Press, Oxford, England.

ICRP (International Commission on Radiological Protection), 1994b, *Human Respiratory Tract Model for Radiological Protection*, Publication 66, Pergamon Press, Oxford, England.

ICRP (International Commission on Radiological Protection), 1995, *Age-Dependent Doses to Members of the Public from Intake of Radionuclides: Part 4 Inhalation Dose Coefficients*, Publication 71, Pergamon Press, Oxford, England. [SRDB Ref ID: 22729]

Lesperance, A. M., M. R. Siegel, and M. D. McKinney, 1992, *A Review of Occupational Safety and Health Issues Relevant to the Environmental Restoration Program: Selected Case Histories and Associated Issues*, PNL-8184, DE93 003287, Pacific Northwest Laboratory, Richland, Washington, October. [SRDB Ref ID: 13336]

MCW (Mallinckrodt Chemical Works), 1961, *Uranium Process Dust Studies from 1960 and 1961*, Weldon Spring, Missouri. [SRDB Ref ID: 12363]

MCW (Mallinckrodt Chemical Works), ca. 1964, *Weldon Spring: An Introduction*, Mallinckrodt Chemical Works, Weldon Spring, Missouri. [SRDB Ref ID: 13179]

MCW (Mallinckrodt Chemical Works), 1965, *Summary of Health Protection Practices*, Health/Safety Department, Weldon Spring, Missouri. [SRDB Ref ID: 14998]

MCW (Mallinckrodt Chemical Works), ca. 1966, *Summaries of Dust Concentrations at Production Jobs, Life of Operation — from Year 1957 through 1966*, Weldon Spring, Missouri. [SRDB Ref ID: 15000]

MCW (Mallinckrodt Chemical Works), 1967, *Fuel for the Atomic Age – Completion Report on St. Louis Area Uranium Processing Operations, 1942-1967*, St. Louis, Missouri. September. [SRDB Ref ID: 6538]

Mason, M. G., 1977, handwritten note, May 22. [SRDB Ref ID: 13179, p. 29]

Meshkov, N., P. Benioff, J. Wang, and Y. Yuan, 1986, *Weldon Spring Historical Dose Estimate*, ANL/EES-TM-308, DE86 014085, Argonne National Laboratory, Energy and Environmental Services Division, Argonne, Illinois, July. [SRDB Ref ID: 13117]

MK-Ferguson, 1989, *Waste Assessment Radiological Characterization of the Weldon Spring Site Raffinate Pits*, Rev. 0, DOE/OR/21548-062; MK-Ferguson Company and Jacobs Engineering Group; August. [SRDB Ref ID: 12541]

NCRP (National Council on Radiation Protection and Measurements), 1973, *Specification of Units for Natural Uranium and Natural Thorium*, NCRP Statement No. 4, Bethesda, Maryland, December 28. [SRDB Ref ID: 14851]

NIOSH (National Institute for Occupational Safety and Health), 2004, *Estimation of Ingestion Intakes*, OCAS-TIB-009, Rev. 0, Office of Compensation Analysis and Support, Cincinnati, Ohio, April 13.

NIOSH (National Institute for Occupational Safety and Health), 2010a, Special Exposure Cohort, Petition Evaluation Report Petition SEC-00143, Division of Compensation Analysis and Support, Cincinnati, Ohio, April 16.

NIOSH (National Institute for Occupational Safety and Health), 2010b, *Radiation Exposures Covered for Dose Reconstructions Under Part B of the Energy Employees Occupational Illness Compensation Program Act*, DCAS-IG-003, Rev. 0, Division of Compensation Analysis and Support, Cincinnati, Ohio, November 5.

NRC (Nuclear Regulatory Commission), 1980, "Appendix G. Calculations for Radiological Assessments," pp. 13–93, *Final Generic Environmental Impact Statement on Uranium Milling*, NUREG-0706, Vol. III, Office of Nuclear Material, Washington, D.C., September. [SRDB Ref ID: 33684]

NRC (Nuclear Regulatory Commission), 1981, *MILDOS – A Computer Program for Calculating Environmental Radiation Doses from Uranium Recovery Operations*, NUREG/CR-2011, Washington, D.C., April. [SRDB Ref ID: 83173]

ORAUT (Oak Ridge Associated Universities Team), 2004, *Technical Basis Document for the Fernald Environmental Management Project – Occupational Internal Dose*, ORAUT-TKBS-0017-5, Rev. 00, Oak Ridge, Tennessee, May 28.

ORAUT (Oak Ridge Associated Universities Team), 2005b, *Weldon Spring Plant – Site Description*, ORAUT-TKBS-0028-2, Rev. 00 Oak Ridge, Tennessee, June 24.

ORAUT (Oak Ridge Associated Universities Team), 2005c, *Weldon Spring Plant – Occupational Environmental Dose*, ORAUT-TKBS-0028-4, Rev. 00, Oak Ridge, Tennessee June 28.

ORAUT (Oak Ridge Associated Universities Team), 2007a, *Rocky Flats Plant – Occupational Internal Dose*, ORAUT-TKBS-0011-5, Rev. 02, Oak Ridge, Tennessee, August 17.

ORAUT (Oak Ridge Associated Universities Team), 2007b, *Internal Dose Reconstruction*, ORAUT-OTIB-0060, *Rev 00*, February 6.

ORAUT (Oak Ridge Associated Universities Team), 2009, *Personal Communication with Former Weldon Spring Plant Design Engineer,* October 29. [SRDB Ref ID: 76970]

ORAUT (Oak Ridge Associated Universities Team), 2012, *Dose Reconstruction during Residual Radioactivity Periods at Atomic Weapons Employer Facilities,* ORAUT-OTIB-0070, *Rev. 01*, Oak Ridge, Tennessee, March 5.

ORAUT (Oak Ridge Associated Universities Team), 2013, *Weldon Spring Plant – Occupational External Dosimetry*, ORAUT-TKBS-0028-6, Rev. 01, Oak Ridge, Tennessee, February 6.

Sanders, M., 1975, "Analysis of Long-Term Data on Uranium in Air at Y-12," pp. 364–386, *Conference on Occupational Health Experience with Uranium, Held at Arlington, Virginia, April 28-30, 1975*, ERDA-93, U.S. Energy Research and Development Administration, Washington, D.C. [SRDB Ref ID: 4613]

Strom, D.J and P.S. Stansbury, "Determining Parameters of Lognormal Distributions from Minimal Information", 2000, pp. 877-880 of American Industrial Hygiene Association Journal Vol 61, PNNL-SA-32215, November/December. [SRDB Ref ID: 13722]

GLOSSARY

absorption type

Categories for materials according to their rate of absorption from the respiratory tract to the blood, which replaced the earlier inhalation clearance classes. Defined by the International Commission on Radiological Protection, the absorption types are F: deposited materials that are readily absorbed into blood from the respiratory tract (fast solubilization), M: deposited materials that have intermediate rates of absorption into blood from the respiratory tract (moderate rate of solubilization), and S: deposited materials that are relatively insoluble in the respiratory tract (slow solubilization). Also called solubility type. See *inhalation class*.

activity median aerodynamic diameter (AMAD)

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)

International System unit of radioactivity equal to 1 disintegration per second; 1 curie equals 37 billion (3.7×10^{10}) Bq.

bioassay

Measurement of amount or concentration of radionuclide material in the body (*in vivo* measurement) or in biological material excreted or removed from the body (*in vitro* measurement) and analyzed for purposes of estimating the quantity of radioactive material in the body. Also called radiobioassay.

bioassay procedure

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

Amount of radioactive material in an individual's body at a particular point in time.

chronic exposure

Radiation dose to the body delivered in small amounts over a long period (e.g., days or years).

curie (Ci)

Traditional unit of radioactivity equal to 37 billion (3.7×10^{10}) becquerels, which is approximately equal to the activity of 1 gram of pure ^{226}Ra.

depleted uranium (DU)

Uranium with a percentage of ^{235}U lower than the 0.7% found in natural uranium.

dose

In general, the specific amount of energy from ionizing radiation that is absorbed per unit of mass. Effective and equivalent doses are in units of rem or sievert; other types of dose are in units of roentgens, rads, reps, or grays.

dose equivalent (*H*, DE)

In units of rem or sievert, product of absorbed dose in tissue multiplied by a weighting factor and sometimes by other modifying factors to account for the potential for a biological effect from the absorbed dose. See *dose*.

dosimetry
Measurement and calculation of internal and external radiation doses.

enriched uranium (EU)
Uranium in which processing has increased the proportion of ^{235}U to ^{238}U to above the natural level of 0.7% by mass. Reactor-grade uranium is usually about 3.5% ^{235}U; weapons-grade uranium contains greater than 90% ^{235}U.

exposure
(1) In general, the act of being exposed to ionizing radiation. See *acute exposure* and *chronic exposure*. (2) Measure of the ionization produced by X- and gamma-ray photons in air in units of roentgens.

inhalation class
Former respiratory tract inhalation classification scheme developed by the International Council on Radiological Protection for inhaled material according to its rate of clearance from the pulmonary region of the lung. Materials were classified as D (days, half-life less than 10 days), W (weeks, 10 to 100 days), or Y (years, more than 100 days). See *absorption type*, which superseded this concept.

insoluble
Having very low solubility. No material is absolutely insoluble. See *absorption type*.

intake
Radioactive material taken into the body by inhalation, absorption through the skin, injection, ingestion, or through wounds.

internal dose or exposure
Dose received from radioactive material in the body.

internal dose assessment
Estimation of an intake of radioactive material and the consequent radiation dose based on bioassay or other measurements in the work environment.

***in vitro* bioassay**
Measurements to determine the presence of or to estimate the amount of radioactive material in the excreta or in other biological materials removed from the body.

***in vivo* bioassay**
Measurements of radioactive material in the human body utilizing instrumentation that detects radiation emitted from the radioactive material in the body.

ionizing radiation
Radiation of high enough energy to remove an electron from a struck atom and leave behind a positively charged ion. High enough doses of ionizing radiation can cause cellular damage. Ionizing particles include alpha particles, beta particles, gamma rays, X-rays, neutrons, high-speed electrons, high-speed protons, photoelectrons, Compton electrons, positron/negatron pairs from photon radiation, and scattered nuclei from fast neutrons.

isotope
One of two or more atoms of a particular element that have the same number of protons (atomic number) but different numbers of neutrons in their nuclei (e.g., ^{234}U, ^{235}U, and ^{238}U). Isotopes have very nearly the same chemical properties. See *element*.

lung solubility type

See *absorption type*.

minimum detectable activity or amount (MDA)

Smallest amount (activity or mass) of an analyte in a sample that can be detected with a probability β of nondetection (Type II error) while accepting a probability α of erroneously deciding that a positive (nonzero) quantity of analyte is present in an appropriate blank sample (Type I error).

minimum detectable concentration (MDC)

Minimum detectable activity (or amount) in units of concentration. See *minimum detectable activity*.

monitoring

Periodic or continuous determination of the presence or amount of ionizing radiation or radioactive contamination in air, surface water, groundwater, soil, sediment, equipment surfaces, or personnel (for example, bioassay or alpha scans). In relation to personnel, monitoring includes internal and external dosimetry including interpretation of the measurements.

natural uranium (U, U-nat, NU)

Uranium as found in nature, approximately 99.27% ^{238}U, 0.72% ^{235}U, and 0.0054% ^{234}U by mass. The specific activity of this mixture is 2.6×10^7 becquerel per kilogram (0.7 microcuries per gram).

occupational dose

Internal and external ionizing radiation dose from exposure during employment. Occupational dose does not include that from background radiation or medical diagnostics, research, or treatment, but does include dose from occupationally required radiographic examinations that were part of medical screening.

rad

Traditional unit for expressing absorbed radiation dose, which is the amount of energy from any type of ionizing radiation deposited in any medium. A dose of 1 rad is equivalent to the absorption of 100 ergs per gram (0.01 joules per kilogram) of absorbing tissue. The rad has been replaced by the gray in the International System of Units (100 rads = 1 gray). The word derives from <u>r</u>adiation <u>a</u>bsorbed <u>d</u>ose.

radionuclide

Radioactive nuclide. See *nuclide*.

rem

Traditional unit of radiation dose equivalent that indicates the biological damage caused by radiation equivalent to that caused by 1 rad of high-penetration X-rays multiplied by a quality factor. The sievert is the International System unit; 1 rem equals 0.01 sievert. The word derives from <u>r</u>oentgen <u>e</u>quivalent in <u>m</u>an; rem is also the plural.

routine monitoring

Monitoring carried out at regular intervals during normal operations.

working level (WL)

Unit of concentration in air of the short-lived decay products of ^{222}Rn (^{218}Po, ^{214}Pb, ^{214}Bi, and ^{214}Po) and ^{220}Rn (^{216}Po, ^{212}Pb, ^{212}Bi, ^{212}Po) defined as any combination of the short-lived radioactive progeny of radon or thoron in 1 liter of air, without regard to the degree of equilibrium, that results in the ultimate emission of 130,000 MeV of alpha energy; 1 WL equals 2.083×10^{-5} joules per cubic meter.

working level month (WLM)

Unit of exposure to radon progeny defined as exposure for 1 working month (170 working hours) to a potential alpha energy concentration from of 1 WL; 1 WLM equals 1 WL times 170 hours, which is 0.00354 joule-hours per cubic meter.

ATTACHMENT A
DUST CONCENTRATIONS FOR THORIUM PRODUCTION JOBS

Table A-1. Dust concentrations for thorium production jobs.

Plant name & no.	Materials	Work group title & cost center[a]	Job title[b]	Period[c]	Workers/ day	Daily weighted average (DWA) concentrations (alpha dpm/m^3)				Notes
						1963	1964	1965	1966	
Sampling 101	Extracted thorium-natural	Sampling 0402	Oven drying ThO_2 sol-pan transfer	Mar 1965	2	-	-	3	-	Test date: Mar 30, 1965. Five ovens operated on a 24-hour cycle at 400 lbs/oven. Experimental, intermittent operation.
			Oven drying ThO_2 sol-vacuum unload	Aug 1965	2	-	-	16	-	Test date: Aug 9, 1965. Experimental, intermittent operation.
			Repackaging ThO_2 feeds for digestion-drum dumping	Apr–Aug 1966	1	-	-	-	9	Total project operating time estimated at 1 week or 10 shifts intermittent operation.
			Repackaging ThO_2 feeds for digestion-hopper packaging	Apr–Aug 1966	1	-	-	-	12	Total project operating time estimated at 1 week or 10 shifts.
			Repackaging ThO_2 feeds for digestion-outgoing drum conveyor	Apr–Aug 1966	1	-	-	-	7	Total project operating time estimated at 1 week or 10 shifts (empty drums).
Refinery 103	Extracted & re-extracted thorium-232 & daughters	Digest-Raffinate 0501	Hopper feed & digestion	Apr–May 1966	3	-	-	-	6	Test date: Apr 22, 1966. One hopper (15,000 pounds) made 1 tank batch, 20,000 lbs/day product on first cycle; recycle is liquid feed.
			Raffinate treatment & disposal	May–Sep 1966	3	-	-	-	2	-
			Misc. digestion of drummed scrap-wet feed	May–Sep 1966	2	-	-	-	4	Test date: May 3, 1966. Material was damp to wet; floor sweepings were used as feed. Non-routine job.

ATTACHMENT A
DUST CONCENTRATIONS FOR THORIUM PRODUCTION JOBS

Plant name & no.	Materials	Work group title & cost center[a]	Job title[b]	Period[c]	Workers/ day	Daily weighted average (DWA) concentrations (alpha dpm/m^3)				Notes
						1963	1964	1965	1966	
Refinery Pot Room 103	Extracted thorium-natural	Pot Room 0502[d]	Pot denitration & sol drying-hand transfers	Nov–Dec 1963; May–Aug 1964	6	88	88	-	-	Test dates: Nov 1963 and May 1964. Airline masks used to enter pot sections.
			Pot denitration (crystals) & sol drying-vacuum unload	Aug–Dec 1964	6	-	60	-	-	Test samplings for vacuuming and packaging collected from Sep 1964 to Oct 1965. Airline masks prescribed for entrance into pot or packaging sections. Small crucibles packaged.
			Pot denitration (liquor) & sol drying-vacuum unload	Jan–Aug 1965	6	-	-	48	-	Airline masks prescribed for packaging and unloading; half-face dust respirators (Comfo) for misc. attending duties in pot sections. Small crucibles packaged.
			Sol drying (fluid bed sol)-vacuum unload	Aug 1965–Jan 1966; Jun–Sep 1966	6	-	-	38	38	Packaging in small crucibles (500 lb). Comfo respirators prescribed in pot sections.
			Pot denitration-hand unload & bucket dump to sol tank	Oct–Nov 1965	6	-	-	144	-	Pot denitration used for special project and to maintain production at scheduled rate when fluid bed was down. Airline masks prescribed for dumping and scooping. Comfo respirators were used for vacuuming and packaging activities.

ATTACHMENT A
DUST CONCENTRATIONS FOR THORIUM PRODUCTION JOBS
Page 3 of 7

Plant name & no.	Materials	Work group title & cost center[a]	Job title[b]	Period[c]	Workers/day	Daily weighted average (DWA) concentrations (alpha dpm/m^3)				Notes
						1963	1964	1965	1966	
			ThO_2 repackaging-multi-purpose hood	Aug 1964	2	-	260	-	-	Short-term operation. Airline masks used in hood.
			DWA index for the Pot room work group	Apr 1964–Dec 1965	-	-	78	50	-	For Apr-Dec 1964, approximately 45% of the effort (manpower) was on thorium jobs. For 1965, approximately 50% of the effort was on thorium jobs. Results are yearly indices and represent DWAs for 3 job titles.
Extraction 105	Extracted thorium and thorium-232 daughters	Extraction	Pumper decanter, pulse column, strippers, and NOK (product liquor)	Jun–Sep 1966	9	-	-	-	2	-
Metals 301	Extracted thorium-natural	Metals-other than Reduction 0702	High-firing 1,000 lb crucibles at recast (furnace)	Nov–Dec 1963; May–Sep 1964	6	21	21	-	-	Test dates included Dec 1963, May 1964, and Oct 1964. Airline masks used for entrance into enclosure for thermocouple change and inspection; operation performed 7 days/week.

ATTACHMENT A
DUST CONCENTRATIONS FOR THORIUM PRODUCTION JOBS
Page 4 of 7

Plant name & no.	Materials	Work group title & cost center[a]	Job title[b]	Period[c]	Workers/ day	Daily weighted average (DWA) concentrations (alpha dpm/m^3)				Notes
						1963	1964	1965	1966	
			TNT repackaging	Oct–Dec 1964	2	-	3	-	-	Test date: Oct 5, 1964. For part of this period, crushing was done externally by electric grabs and transferred to the refinery without repacking TNT.
			High-firing 500 lb crucibles in billet heaters	Sep 1964–Jan 1966; Jun–Sep 1966	3	-	11	11	11	Test date Nov 4, 1964. Four heaters were installed, each handling a 500-lb crucible on a 24-hourcycle.
			Repackaging ThO_2 in recast enclosure	Oct–Dec 1965; Jan 1966; Jun–Sep 1966	2	-	-	2060 (55)	55	The DWA result of 2,060 dpm/m^3 is for Oct 1965; airline masks and cover clothing were used inside the transfer enclosure, which exhausted to the dust collector. Test dates: Sep 16, 1965 (experimental batch) and Sep 27, 1965- improvements were recommended. The DWA result of 55 dpm/m^3 is for Nov 1965 and later. Test dates: Feb 24, 1966 and Jul 14, 1966. Airline masks were used for all scooping operations.

ATTACHMENT A
DUST CONCENTRATIONS FOR THORIUM PRODUCTION JOBS
Page 5 of 7

Plant name & no.	Materials	Work group title & cost center[a]	Job title[b]	Period[c]	Workers/ day	Daily weighted average (DWA) concentrations (alpha dpm/m^3)				Notes
						1963	1964	1965	1966	
			Kiln calcining of NLO sump cake	Apr–Jul 1966	3	-	-	-	20	Test dates: Apr 21, 1966; Jun 8-9, 1966. Comfo (MSA) dust respirators used on dumping platform and at packaging stations. Feed was normally wet, ranging from damp to a surface water layer.
			DWA index for the Metals (other than Reduction) work group	May 1964–Dec 1965	-	-	17	113	-	For 1964, approximately 50% of the (manpower) effort was on thorium jobs, but none (on thorium) during the first 4 months. For 1965, approximately 60% of effort was on thorium jobs. Results are yearly indices and represent DWAs for 2 to 3 job titles.

ATTACHMENT A
DUST CONCENTRATIONS FOR THORIUM PRODUCTION JOBS

Plant name & no.	Materials	Work group title & cost center[a]	Job title[b]	Period[c]	Workers/ day	Daily weighted average (DWA) concentrations (alpha dpm/m^3)				Notes
						1963	1964	1965	1966	
Scrap Plant 403	Extracted thorium-natural	Scrap Plant 0601 [e]	TNT crystals-repackaging	Nov–Dec 1963; May–Sep 1964	6	15	15	-	-	Equivalent production rate of 1 ton of ThO_2/day.
			ThO_2 repackaging-from recast crucibles	Nov–Dec 1963; May–Sep 1964	6-8	290	290	-	-	After Sep 1964, smaller crucibles were used. Airline masks were used on all scooping operations; operations conducted inside multi-purpose walk-in hoods. Exhaust was filtered by auto-air-mat (paper rolls). Manpower for 1964 was variable (5 in May to 15 in Jul); normal manpower was 6 to 8. ThO_2 repackaging included screening, blending, and product packaging.
			ThO_2 repackaging-from billet heater crucibles	Sep 1964–Oct 1965	4	-	20	20	-	Test date: May 24, 1965. Airline masks used on all scooping operations. Shallow crucibles could be scooped with breathing zone more distant from the opening than with recast crucibles (lower concentrations observed). Long-handles scoops were used in the blender drum.

ATTACHMENT A
DUST CONCENTRATIONS FOR THORIUM PRODUCTION JOBS

Plant name & no.	Materials	Work group title & cost center[a]	Job title[b]	Period[c]	Workers/ day	Daily weighted average (DWA) concentrations (alpha dpm/m^3)				Notes
						1963	1964	1965	1966	
			Fluid bed denitration-product to drums	May–Jul 1965	6	-	-	44	-	Test date Jun 1965. Experimental use of NLO liquor during May.
			Fluid bed denitration-wet feed & product	Aug 1965–Jan 1966; Jun–Sep 1966	6	-	-	29	29	Liquor feed and sol product at 1 ton/day production rate.
			DWA index for Scrap Plant work group	Nov 1963–Dec 1965	-	180	117	28	-	The 1963 index was for a 6-man group (Nov-Dec) with 17% of manpower on thorium. The 1964 index was for a 6-man group with 60% of manpower on thorium. The 1965 index was for a 12-man group (Jan-Jun) and a 6-man group (Jul-Dec) with 80% of manpower on thorium. Results are yearly indices and represent DWAs for 2 to 3 job titles.

a. “Cost Center” refers to an employer charge code. Cost center codes are published in some TBDs and can be used for identifying employee work areas.
b. “Job Title” is the term used in the referenced report; this term more accurately represents a task description rather than an employee job title.
c. “Time Period” was recorded from the air sampling data sheets and was estimated if date ranges were incomplete.
d. Manpower in this group ranged from 18 to 21 men during the years 1963 through 1965; base production rate was 1 ton ThO_2/day.
e. Average production rate was 1 ton ThO_2/day.