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Dose Reconstruction
Project for NIOSH

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1.0 PURPOSE

Technical information bulletins (TIBs) are not official determinations made by the National Institute for Occupational Safety and Health (NIOSH) but are rather general working documents that provide historic 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. TIBs may be used to assist NIOSH staff in the completion of individual dose reconstructions.

In this document the word "facility" is used as a general term for an area, building, or group of buildings that served a specific purpose at a site. It does not necessarily connote an "atomic weapons employer facility" or a "Department of Energy facility" as defined in the Energy Employees Occupational Illness Compensation Program Act of 2000 (42 U.S.C. § 7384l(5) and (12)).

The purpose of this document is to provide guidance for estimating the maximum plausible dose to workers at Atomic Weapons Employers (AWEs). This document may also be used to estimate doses at Department of Energy facilities when the exposures would be adequately estimated by the methods in this document. (The classification of facilities as either DOE or AWE is based on regulatory definitions, not on the type or amount of radiation exposure that might have occurred). This document describes an efficiency process that may be used to expedite the processing of claims requiring dose reconstruction under the Energy Employees Occupational Illness Compensation Program Act (EEOICPA). The exposure matrices in this document are designed to estimate the maximum plausible annual doses for the purpose of expediting certain types of claims.

This document includes general information regarding radiation dose reconstruction, followed by specific facility category information and by the type(s) of radiation exposure that might have occurred at a given facility category. The document was originally created to overestimate doses categorized, generally, as natural uranium handling facilities, and so this revision begins with uranium handling facilities.

2.0 GENERAL INFORMATION

The intake and dose rates in this exposure matrix are to be used in conjunction with the individual worker's covered employment dates, date of birth, and date of cancer diagnosis. Rather than pro-rate intake and dose rates, it is acceptable to use exposure values for a full year, as this will overestimate dose. Because worker bioassay and dosimetry data, and workplace monitoring data cannot be ignored, the worker's data and associated workplace monitoring data, if available, should be reviewed to ensure that a dose reconstruction based on this exposure matrix is an overestimate for a particular worker's circumstances.

Some of the facilities have multiple periods of Atomic Energy Commission (AEC) operation (not including periods designated as remediation). In these cases, the maximum dose estimates in sections designated as X.1 (internal) and X.2 (external) are applied for each period of AEC operations and only for these periods. Direction for estimating exposure during periods of normal (e.g., non-AEC) operation or remediation is provided in sections designated as X.3.

The term "day" in this document refers to a calendar day. The term "workday" is used to describe a day at work. It is assumed that there are 250 workdays per year and that a workday is 8 hours long: this results in 2000 hours of work per year. The distinction between workday and calendar day is especially important when considering internal dose rates for use in calculating organ doses.

NIOSH has concluded that non-DOE sources of radiation exposure are to be reconstructed for the covered employment period. The NIOSH method for adding the non-DOE exposures is to assign the dose distributions in Table A-1, to all AWE employees, unless the doses are lower in the facility-specific data. Because this is an overestimating approach, and the use of analytical x-ray equipment, radioactive sources and materials, and radiography devices would not have been unusual in operational programs, the default is to include the dose from the estimated total industrial exposure distributions from these activities in AWE overestimates. However, the industrial radiography component in Table A-1 can be excluded on a case-by-case basis, when additional information is sufficient to support exclusion (note that job title might not be sufficient to make this determination if there is no information about the location of the radiation source, the area radiation levels and the location of the employees). Note that these estimates of industrial radiation dose represent a reasonable estimate of likely doses for most workers, but any available claim-specific information should be considered when it is available for employees, who were working with or in the direct vicinity of industrial radiation sources and/or devices.

3.0 URANIUM METAL HANDLING FACILITIES

There were approximately 102 AWE facilities that handled natural uranium in support of the atomic weapons program. Of these facilities, 92% operated for 10 years or less and 76% operated for 5 years or less. The maximum time operated was twenty-three years and the minimum time was less than a year.

The uranium metalworking processes at these facilities included reduction and recasting, rolling, machining, extruding, and forging; fuel element fabrication; and uranium recovery from scrap. A large number of the facilities handled only uranium metal. Most of the facilities considered in this section were privately owned and their work was done for the AEC or Manhattan Engineer District (MED) on a short-term basis and/or in addition to their normal commercial operations. Section 3 only addresses uranium, its short-lived progeny and possible contaminants from receipt of recycled uranium. The assumptions regarding recycled uranium contaminants are further discussed in other subsections, but the assumed mass (in parts per billion (ppb)) and activity fractions, based primarily on a review of Fernald and Hanford information, are shown here because of their significance to both internal and external dose reconstruction. The determination of activity fraction is based on depleted uranium, and results in overestimates of the activity fractions for facilities that used normal or enriched uranium with associated larger specific activities.

Table 3-1. Assumed fractions of recycled uranium contaminants at metal handling facilities.

Recycled uranium contaminant	Pu-239	Np-237	Tc-99	Th-232	Th-228
Activity fraction of contaminant in uranium	0.00246	0.00182	0.379	2.73E-06	2.73E-06
Contaminant in ppb of uranium	10	1,040	9,000	10,000	*

*assumes same activity as Th-232

Exposure from processing of thorium, radium, uranium ores, or other radioactive materials (except as recycled uranium contaminants) is not considered in this section. Some uranium scrap recovery operations processed only uranium; however, a number of these operations also processed uranium long-lived progeny, such as thorium-230 and radium-226; some sites were processing small amounts of plutonium or other radioactive contaminants in the waste.

To summarize, Section 3 provides an estimate of the maximum plausible dose from uranium and its short-lived progeny at uranium metal-handling facilities. This section includes consideration of radiation exposure from recycled uranium contaminants within the uranium metal matrix. This section

does not include consideration of exposures from uranium ore processing, long-lived uranium progeny, or other radioactive metal processing, which might also have occurred at some facilities during some periods. If a facility processed other forms of uranium or other radionuclides, additional exposure information must be considered when reconstructing a maximum plausible dose estimate.

3.1 URANIUM METAL HANDLING FACILITIES – ESTIMATION OF INTERNAL DOSE

During World War II and the time of the Manhattan Engineer District (MED), the permissible level for natural uranium air dust concentration was set at 500 $\mu g/m^3$ for insoluble uranium compounds and 150 $\mu g/m^3$ for soluble uranium compounds. After the war, the University of Rochester (Rochester, NY) recommended lowering the permissible level to 50 $\mu g/m^3$ for all uranium compounds on the basis of chemical toxicity. This level was also stated as 70 disintegrations per minute per cubic meter (70 dpm/m^3) for natural uranium and was based primarily on animal studies. The Medical Division of the AEC New York Operations Office (NYOO) thought that the "maximum permissible level" should be based on human data and was thus unknown. Therefore, the level of 50 $\mu g/m^3$ was generally referred to as the "preferred level" (AEC 1949). Oftentimes the contractors of the AEC used the term "Maximum Allowable (air) Concentration" (MAC) interchangeably with the "preferred level" (PL) and reported air-sampling results as multiples of the MAC (NLO 1952, AEC 1953). When considering air concentrations reported in multiples of the PL or MAC, the actual assumed value of the PL or MAC should be verified.

In 1949, the Medical Division of the NYOO published a report on the health hazards at seven facilities that produced and/or processed uranium for the AEC. These facilities included Mallinckrodt Chemical Works, Harshaw Chemical Company, Linde Air Products, Electro Metallurgical Company, Simonds Saw and Steel Company, Vulcan Crucible Company (later known as Aliquippa Forge), and Vitro Manufacturing Company. These facilities were the earliest MED- or AEC-constructed or employed uranium processors and, with the exception of Vulcan Crucible, were very large operations in comparison to the other AWE uranium handling facilities. The AEC used the information on work tasks with measured air concentrations in breathing zones, general areas and process areas to determine average air concentrations weighted by exposure times and summed these time-weighted air concentrations to determine daily time-weighted average air concentrations by job categories. Up until the time of the 1949 MED report, surveys by the NYOO indicated that out of 648 exposed workers at these plants, 9% were exposed to uranium air concentrations greater than 125 MAC (greater than 6250 $\mu g/m^3$), 9% were exposed at 25-125 MAC (1250-6250 $\mu g/m^3$), and 82% were exposed to less than 25 MAC (less than 1250 $\mu g/m^3$). As a result of the NYOO report, significant improvements were made in the operational conditions such as re-design of ventilation systems, enclosing some processes, and using remote controls (AEC 1949). By the end of 1949, exposure levels were significantly reduced at these larger plants even though production levels increased (Mason 1958).

In addition, AEC decided to discontinue long-term work at some facilities where resources were not available to improve radiation safety. Vulcan Crucible (Aliquippa Forge) was a steel mill that contracted with the AEC to roll uranium billets into rods on a part-time basis from 1948 to 1949. The contract (AEC Contract No. AT-(30-1)-407) stated that the plant was to arrange to spend "at least two consecutive weeks out of every five consecutive weeks" performing the AEC contract work. Four of the most exposed workers in the 1949 study were from this plant and the report stated that discussions were under way to shut down this operation (AEC 1949). Operations were shut down in late 1949 and the plant was decontaminated to 1950 standards (Adams and Payne 1992).

Early exposure studies were not very selective in terms of particle size (Stannard 1988), and could have under or over accounted for respirable radioactive material. Two early uranium facility particle

size distribution studies measured particle size distributions. Medians ranged from 1.22 to 1.80 µm mass median diameter during rolling (Spiegl et al. no date; probably 1949 or 1950, the year was unspecified), and 3.1 to 4.9 µm [probably mass median diameter, but not specified] in an area of visible dust at an ore processing and scrap recovery facility (AEC 1958). When adjusted for density, these results are considered consistent with International Commission on Radiological Protection (ICRP) Publication 66 default parameters for particle deposition (ICRP 1994a), so dose reconstructions should assume ICRP Publication 66 defaults (including a 5 µm activity median aerodynamic diameter). In addition, the air concentration measurements represented the amount of uranium in the air where the workers were located, but not necessarily, what was taken into the lung or the body. At many of the AWE uranium metal facilities, the daily-weighted average exposures become significant overestimates of annual exposures when multiplied by 250 workdays per year. This is because the uranium operations were intermittent. Usually, AEC measured airborne contamination with alpha counters, so it is reasonable to use the activity rather than the mass designation for the MAC for estimation of the maximum daily intake.

3.1.1 URANIUM INHALATION

Human and animal studies have indicated that oxides of uranium can be very insoluble in the lung (ICRP 1995), indicating absorption type S (0.1% and 99.9% with clearance half-times on the order of 10 minutes and 7000 days, respectively). Other *in vitro* dissolution studies of compounds found at uranium facilities have shown that oxides of uranium exhibit moderate solubility (Eidson 1994; Heffernan et al. 2001) suggesting absorption type M (10% and 90% with clearance half-times on the order of 10 minutes and 140 days, respectively). *In vitro* dissolution tests on oxides produced from uranium metal during depleted uranium armor penetrator tests have indicated multicomponent dissolution rates, with 25% of uranium dissolving with a half-time of less than or equal to 0.14 days and 75% dissolving with a half-time of 180 days. Because there was no specific information on the solubility of aerosols produced at the AWE uranium metal processing facilities, this analysis assumed that both types M and S were available. The selection of absorption type should depend on the organ of interest. Dose reconstructions should assume International Commission on Radiological Protection (ICRP) Publication 66 default parameters for particle deposition (ICRP 1994).

For the purpose of estimating maximum plausible dose, it is assumed that a worker received a constant exposure to a maximum level of 7000 dpm/m^3 (100 MAC) during AEC operation. This document assumes that the intake for a year is derived from the worker being exposed to 100 MAC for eight hours per day, five days per week, 50 weeks per year, whereas time-weighted average studies of even the larger plants like Mallinckrodt show that the majority of time workers were exposed at lower levels than this. No credit is taken for breaks or working at tasks where radioactive material intakes would be much lower or nonexistent.

Although the information presented in Section 3.1, indicates that a small fraction of workers had daily time weighted exposures of greater than 100 MAC, the records also show that the AWE uranium metal handling facilities were not processing uranium on every workday or even every week. A review of some of the busier AWE uranium metal handling facilities, including Simonds Saw and Steel, and Bridgeport Brass (Adrian), support the 100 MAC maximum plausible exposure estimate for each year of AEC operation. In addition, a Y-12 coworker dose study indicates even at the very busy, production-oriented Y-12 plant (which also handled highly enriched uranium) an estimated natural uranium exposure of 100 MAC would overestimate all type M predicted 95th percentile annual intakes and would be a reasonable estimate or overestimate for the 50th percentile type S intakes for most years (ORAUT 2005).

The maximum annual uranium inhalation intake due to chronic exposure is estimated by multiplying the maximum air concentration of 7000 dpm/m^3 (5000 μg/m^3) by the ICRP 66 recommended breathing rate of 1.2 m^3/h and by the assumed 2000 work-hours per year. This results in an annual chronic inhalation intake of 1.68E+07 dpm (1.2E+07 μg), which is equal to a daily intake rate of 46,000 dpm/day (3.3E+04 μg/day). For the purpose of calculating annual organ doses, the intake (in dpm) is assumed to be entirely U-234. For intake periods of less than one year, the intake rates shown in Table 3-5 should be adjusted as noted in Section 3.1.6, unless site- and/or claim-specific information are available to justify a total intake equal to less than this document assumed annual intake.

3.1.2 URANIUM INGESTION

In the case where inhalation intakes are calculated from air concentrations, ingestion intakes are also to be considered. NIOSH (2004) indicates that the ingestion rate, in terms of dpm for an 8-hour workday, can be estimated by multiplying the air concentration in dpm per cubic meter by a factor of 0.2, so the uranium ingestion rate based on an air concentration of 7000 dpm/m^3 would be 1400 dpm/workday. To adjust this to ingestion intake per calendar day, 1400 dpm/workday is multiplied by 250 workdays per year and divided by 365 days per year, which equals 959 dpm/d. In accordance with NIOSH 2004, the same f1-value as used for inhalation dose calculations is to be used for ingestion dose calculations.

3.1.3 CONSIDERATION OF URANIUM BIOASSAY DATA

The uranium fusion photofluorimetry urinalyses, which were the typical uranium bioassay method at the AWEs, were frequently performed by the University of Rochester and the AEC's New York Operations Office and were similar to those performed at other AEC facilities. The default detection threshold for uranium urinalysis is assumed to be 10 μg/L based on a reported sensitivity of 5 to 10 μg/L for uranium fluorimetry urinalysis in the early years (Wilson 1958).

Predicted uranium urinalysis results were calculated for the last day of assumed chronic intake periods of 1, 2, 5, 10, 20 and 50 years, assuming the maximum estimated inhalation and ingestion intakes of natural uranium. The results are shown in Table 3-2. Type M uranium comes to 90% equilibrium in urine at about one year resulting in excretion of about 1400 μg/L. Chronic inhalation and ingestion of Type S material results in a uranium urinalysis of about 50 μg/L after 1 year of exposure and about 160 μg/L after 50 years of exposure. A cursory review of uranium urinalysis data from the metal handling facilities (Simonds Saw and Steel, Bridgeport Brass (Havens, Adrian, Extrusion Plant and Seymour Specialty Wire), Chapman and Aliquippa Forge), as well as Y-12 and Paducah indicates that uranium urinalysis results exceeding 100 μg/L are rare and that most results are less than 50 μg/L. In addition, when results did exceed these values, they usually dropped to less than 30 μg/L within a week or so. This gives additional credibility to the assumption that the assumed uranium intake rates are likely to be overestimates for natural uranium. However, it should be noted that assumption of these intake rates for enriched uranium could result in much lower mass-based bioassay results. For 3.5% enrichment, the 1-year and 50-year bioassay results would be about 400 μg/L for type M, and 16 and 49 μg/L, respectively, for type S. For 93.5% enrichment, the 1-year and 50-year bioassay results would be about 14 μg/L for type M, and 0.5 and 1.6 μg/L, respectively, for type S.

3.1.4 INTERNAL DOSE: RECYCLED URANIUM AT URANIUM METAL HANDLING FACILITIES

Recycled uranium might have been processed at uranium metal handling facilities after 1952. An estimate of contaminants that might contribute the most to internal doses, based on a review of recycled uranium contaminants at Hanford and Fernald, is shown below in Table 3-2. (The relative

Table 3-2. Predicted uranium urinalyses from maximum inhalation and ingestion chronic intakes.

Bioassay collection year	Type M		Type S	
	dpm/d	μg/L*	dpm/d	μg/L*
0.5	2,579	1,215	84	39
1	2,909	1,370	108	51
2	3,066	1,444	146	69
5	3,113	1,466	210	99
10	3,138	1,479	255	120
20	3,162	1,490	295	139
50	3,185	1,501	333	157

*Mass results assume natural uranium exposure

internal dose contributions from Tc-99, Th-228 and Th-232 were low enough to be ignored.) It is unlikely that recycled uranium would constitute the entire source term or that processing would result in significant concentration of recycled uranium contaminants. In addition, the activity fractions assume that the uranium specific activity is based on depleted uranium, which increases the proportion of the contaminants by activity. The contaminant levels for depleted uranium overestimate the contaminants in uranium of normal enrichment by about 40%. The contaminants are assumed to be oxides, and it is assumed that a plutonium oxide might behave either as absorption type M or type S material. Neptunium is assumed to behave as absorption type M.

Table 3- 3. Estimated recycled uranium activity fractions for internal dose reconstruction.

Uranium	Pu-239	Np-237
1	0.00246	0.00182

3.1.5 INTERNAL DOSE: DEPLETED OR ENRICHED URANIUM

For a given uranium process, the mass of (long-lived) uranium released to air will not change because of enrichment. In other words, the mass concentration in air would likely be similar for depleted, natural and enriched uranium. However, as the enrichment of U-235 (and therefore U-234) increases, more activity would be in the air, because of an increase in uranium specific activity. (Specific activity refers to the radioactivity per unit mass of material; as the half-life increases, the specific activity decreases.)

Depleting uranium results in reduced activity per mass, because the mass-ratio of the lower specific activity U-238 is increased. Because so few sites handled only depleted uranium, and because adjusting for depletion would reduce the estimated intakes below those estimated for natural uranium, it is assumed that depleted uranium intakes are adequately bounded by the natural or normal uranium intake overestimates.

Uranium enriched by gaseous diffusion or a calutron [an electromagnetic device for separating material by mass] results in a significant increase in the shorter-lived U-235 and U-234 isotopes,

which leads to an increase in specific activity of the uranium. Enriched uranium metal has essentially the same physical characteristics as natural or depleted uranium, but is more radioactive per unit mass of uranium.

One way to estimate intakes from processing enriched uranium, when information is available regarding intakes of natural uranium from the same process, is to consider the different specific activities of the material being processed. The specific activities for various uranium compositions are shown in Table 3-4. Assuming the amount of material that will become airborne is based on the physical and chemical characteristics of uranium, and is not related to its specific activity, it is reasonable to assume that mass-based air concentrations for a given process will be the same for natural and enriched uranium. It follows that an estimate of radioactive air concentrations for enriched uranium based on natural uranium will increase by the ratio of the specific activity of the enriched uranium to the natural uranium. This estimate does not consider changes in handling practices and throughput, which were likely to have occurred during the handling of enriched uranium and which would likely reduce intakes from this overestimating method.

Table 3-4. Uranium intake multiplication factor based on enrichment.

Uranium	Specific activity pCi/μg	Uranium activity multiplier
Depleted	0.4021	1
Natural (normal)	0.6829	1
2%	1.616	2.37
3.5%	2.201	3.22
20%	9	13.2
93.5%	68.11	99.7

Intake activity multipliers are presented for various enrichments in Table 3-4. For sites where it is known that the fraction of enriched uranium processed is only a fraction of the total uranium processed, a weighted multiplier can be derived. If 80% of the uranium source term was natural, and 20% was 93.5% enriched, the multiplication factor would be 0.8 times 1 plus 0.2 times 99.7, which equals 20.74. It is very unlikely that most of the AWEs handled large quantities of highly enriched (93.5%) uranium, because this would have required well-developed nuclear criticality safety programs. Even at low enrichments, the need to address nuclear safety issues would have deterred (but not eliminated) outsourcing of enriched uranium metalwork involving large quantities of material.

For some sites, a more reasonable upper bound can be developed based on a review of the air monitoring data. If the air data appear to be sufficient to adequately bound the air concentrations during AWE operations, the results can be used to determine if the maximum natural uranium intake estimates in this document are bounding. If the natural uranium intake estimate bounds the air data and if bioassays do not exceed the activity-based predictions, no adjustment for enrichment is required for estimating maximum intakes.

3.1.6 OCCUPATIONAL INTERNAL DOSE RECONSTRUCTION ASSUMPTIONS AND SUMMARY

Uranium intakes for overestimating internal dose at uranium metal handling facilities are shown in the table below. Multiple uranium intake possibilities are shown to allow the dose reconstructor to choose the uranium intake for the appropriate enrichment assumption. Note that only one uranium intake rate is to be assigned, in other words, the listed daily intakes of both natural uranium and enriched uranium should not be independently assigned. The type of uranium to assign depends on the

assumptions regarding uranium isotopic composition at the facility. However, it is reasonable to weight multiple intake rates as noted in section 3.1.5. Appendix B provides facility-specific information for use in assigning intakes. Because this document provides an overestimating approach, and the choice of maximum plausible air concentration is based on time-weighting exposure for a year, caution should be used in applying this method to organ dose reconstruction for intake periods less than a year. For intake periods less than one year, a simple overestimating approach is to adjust the daily intake upward by the number of days in an average year, 365, divided by the number of days exposed in that year. Reducing total intake to less than one year of intake requires justification of the chosen intake rate for the particular site and employee circumstances.

For dose calculations, uranium intakes are assumed to be U-234. Uranium oxides can be either absorption type M or S.

Recycled uranium contaminants should be accounted for beginning in 1953 and every year thereafter. Neptunium oxides are type M. Plutonium oxides are assumed to be type M or S.

Table 3-5 lists intake rate assumptions for overestimates of uranium metal handling facilities. The intake mode is chronic. The dose distribution is assumed to be constant.

Table 3-5. Internal exposure summary for operational period.

Radionuclide(s)	Note	Type	Intake mode	Intake dpm/day
U-natural	Only one of the listed uranium intakes is applied. Assume uranium is U-234 for dose calculations.	M, S	Inhalation	4.60E+04
		(a)	Ingestion	9.59E+02
U (2%)		M, S	Inhalation	1.09E+05
		(a)	Ingestion	2.27E+03
U (3.5%)		M, S	Inhalation	1.48E+05
		(a)	Ingestion	3.09E+03
U (20%)		M, S	Inhalation	6.07E+05
		(a)	Ingestion	1.26E+04
U (93.5%)		M, S	Inhalation	4.59E+06
		(a)	Ingestion	9.56E+04
Pu-239	Do not include prior to 1953.	M, S	Inhalation	1.13E+02
		(a)	Ingestion	2.36E+00
Np-237	Do not include prior to 1953.	M	Inhalation	8.38E+01
		(a)	Ingestion	1.75E+00

a. Choose same f_1-value as used for inhalation per NIOSH (2004).

3.2 URANIUM METAL HANDLING FACILITIES – ESTIMATION OF EXTERNAL DOSE

The majority of photons from natural uranium metals are in the 30 to 250 keV energy range. Solid uranium objects provide considerable shielding of the lower energy photons and harden the spectrum, causing the majority of photons emitted from a solid uranium object, such as a thick plate, to have energies greater than 250 keV. While it is recognized that solid uranium sources will have a hardened photon spectrum, exposure to a thin layer of uranium on a surface will result in a larger fraction of exposure to lower energy photons. This analysis assumed workers were exposed to photon energies in the 30 to 250 keV range, which is favorable to claimants when considering both organ dose conversion factors and radiation effectiveness factors. Nonpenetrating dose from natural uranium consists primarily of electrons with energies >15 keV. For consistent presentation, exposure or dose is reported as:

- penetrating, assumed to be associated with photons of energies 30 keV or greater,
- and nonpenetrating, assumed to be associated with photons of energies less than 30 keV or with electrons.

After 1952, small quantities of primarily alpha (Pu-239, Np-237, and Th-232/228) and beta (Tc-99) emitting radionuclides found their way into the uranium metal via recycling. Because of their primarily nonpenetrating radiation types, relatively low activities, and relatively low external radiation hazard when compared to that of their uranium matrix, their contribution to dose is considered adequately addressed by the uranium external dose estimates. A quick check of the relative penetrating dose from an overestimate of recycled uranium contaminants in ground surface contamination shows that they contribute less than 1% to penetrating dose. The assumed nonpenetrating doses from uranium in this document are sufficiently large to bound any small contribution from Tc-99.

Depleted uranium might have been used in small quantities at some facilities, but it is believed that the external dose assumptions are sufficiently large in this document to bound any slight differences in dose contributions from a small, depleted uranium source term. Some uranium metal handling facilities processed enriched uranium. Because of the nuclear safety concerns (criticality events), materials were usually stored away from the work area and throughput was reduced. In addition, enrichment of uranium reduces the activity ratio of U-238, as well as its progeny, which reduces both the photon and the beta exposure from the uranium metal. For enriched uranium, the additional photon dose contribution from U-235 and its progeny does not counteract this decrease. The assumptions for natural uranium are considered bounding for facilities that handled low enriched uranium metal.

3.2.1 Uranium Metal Handling Facilities – Penetrating External Dose

The maximum size of the uranium ingots used in the atomic weapons program was used to estimate external exposure during processing. The dimensions of the two larger ingots are shown below in Table 3-6.

Table 3-6. Maximum size of the uranium ingots used in processing (Bonfer 2003).

Shape	Dimensions (inches)
Rectangular	24 L × 16 W × 4 H
Cylindrical	20 L × 13 Dia.

It was assumed that the exposure geometry was 100% anterior-posterior (AP). The workers were assumed to spend 8 hours per day performing their duties and no credit was given for breaks or other periods when they might have been away from the work area. Although during performance of the job, the worker could have been any distance from the source (the ingots), to be favorable to claimants, it was assumed the worker spent the entire 8 hours at 1 foot from the source.

For the calculations of exposure rate, it was assumed that the ingots were composed entirely of natural uranium metal that had aged for 100 days, which allows for the ingrowth of the short-lived progeny and adds significant dose contribution from the bremsstrahlung radiations due to beta emissions. More than half of the calculated penetrating dose is due to bremsstrahlung photons. The exposure rate at various distances from the ingots was calculated using MicroShield™ (Grove Engineering 2003) and MCNP (LANL 2003). The calculated values are shown in Table 3-7.

The MCNP calculated dose rates were higher than the MicroShield™ dose rates for both the rectangular and the cylindrical ingot. The calculated dose rate at 1 foot was higher from the rectangular ingot, so that value was used in the dose estimation. The MCNP-calculated dose rate was multiplied by 2000 hours per year, resulting in an annual dose of 4.16 rem. The organ doses are calculated by multiplying the calculated annual dose rate by the mode of the "Ambient Dose Equivalent (H*(10)) to Organ Dose (H_T)", AP, 30 -250 keV photon dose conversion factors in Appendix B of the NIOSH External Dose Reconstruction Implementation Guideline (NIOSH 2002).

Table 3-7. Calculated photon exposure rates from uranium metal ingots.

Shape of ingot	Distance from surface	MicroShield™ exposure rate (mR/hr)	MCNP dose rate (mrem/hr)
Rectangular	1 cm*	7.90	8.26
	1 ft	1.91	2.08
	1 m	0.29	0.373
Cylindrical	1 cm*	8.27	8.44
	1 ft	1.13	1.15
	1 m	0.17	0.185

*The MCNP dose rate is calculated at the surface of the ingot.

The chronic external exposure scenario is only a surrogate to estimate maximum doses, and is really a replacement for individual dosimetry measurements. Annual photon dose based on the sum of individual dosimetry measurements is assumed for probability of causation determination to occur acutely rather than chronically for a given year. This same favorable to claimant assumption is applied here. The annual organ doses from this scenario are entered into the NIOSH IREP program assuming acute dose from photons with energies from 30 to 250 keV and a constant dose distribution.

It was also assumed that workers were exposed to contaminated surfaces during their entire work period. The level of surface contamination was determined by first calculating a terminal settling velocity for 5-μm activity mean aerodynamic diameter (AMAD) particles. The calculated terminal settling velocity of 0.00075 meters per second is multiplied by the assumed maximum air concentration of 7000 dpm/m^3. It was assumed that the surface contamination level was due to 365 days of constant deposition from the constant air concentration. This is considered favorable to claimants as facility housekeeping practices most likely kept contamination to a lower level. At the assumed air concentration level with the calculated terminal settling velocity, a 365-day buildup will result in a contamination level of 1.66E+08 dpm/m^2. The annual organ dose due to exposure to contaminated surfaces was determined by multiplying the surface contamination level by the dose coefficients for contaminated ground surfaces from Federal Guidance Report No. 12 (EPA 1993). These EPA dose conversion factors were modeled using an anthropomorphic phantom standing on an isotropic plane source in air. Dose conversion factors, for organs other than skin, are based on photon exposure (including bremsstrahlung). Skin dose conversion factors include electron plus photon dose. The source term was assumed to be natural uranium contamination at the calculated level, plus its short-lived progeny in equilibrium. The organ doses from this scenario are entered into the NIOSH IREP program assuming an acute dose from photons with energies from 30 to 250 keV and a constant dose distribution.

Submersion penetrating doses were not considered, because the relative organ doses were more than a factor of 1000 smaller than the contaminated surface penetrating doses.

Neutron doses can be considered negligible for natural and depleted uranium. For enriched uranium metal, neutron doses are likely to be much lower than photon doses. Intimately mixed uranium and

low atomic number materials are unlikely to occur at the metal processing facilities, which further reduces the likelihood of significant neutron doses.

3.2.2 Uranium Metal Handling Facilities – Nonpenetrating External Dose

To estimate maximum nonpenetrating forearm and hand exposures, the maximum estimates of shallow dose in the Bethlehem Steel site profile (ORAUT 2004a) were used. Nonpenetrating exposures at Bethlehem Steel were based on the assumption that 60% of the time, a worker was exposed to 0.15 rad/hour (1 foot from semi-infinite uranium source) and 40% of the time, a worker was exposed to 0.09 rad/hour (3 feet from semi-infinite uranium source), which results in an average dose rate of 0.126 rad/hr. For a 2000-hour work-year, this would result in a nonpenetrating dose of 252 rad/year. The estimated annual dose of 252 rad is only included for dose reconstruction of skin on the hands and forearms.

To put this into perspective, the annual skin dose at a depth of 7 mg/cm^2 would be 230 rad for a worker who kept a portion of their skin in direct contact with a uranium slab (nonpenetrating dose rate at 7mg/cm^2 is 0.230 rad/hour (ORAUT 2004a)) for 1000 hours (half the hours in a typical a work year). The contact shallow dose rate might be used as a starting value to determine other skin dose when claim-specific data indicates other areas of the skin were in close proximity to the uranium. For instance, when information indicates that workers sat on the rods and there is a possibly related skin cancer, it would be reasonable to estimate some fraction of time sitting on the rod, e.g., 30 minutes per workday, which would result in a “bare skin” exposure of 28.8 rad in a year. When clothing is taken into account, the garments would reduce the dose to the skin. To estimate the effect of clothing, it was assumed a 65% Dacron/35% cotton lab coat plus a white cotton glove were representative of the fabric thickness: their uranium reduction factors would be 0.91 and 0.89, respectively (DOE 2004), for a total reduction factor of 0.81, which would reduce the annual skin dose to 23.3 rad in a year.

The estimated dose from nonpenetrating radiation impinging on areas of the body other than the hands and forearms is estimated as 10 times the 1-foot photon dose. This estimate is based on a preliminary review of film badge data that indicates the nonpenetrating dose is usually no more than a factor of 10 larger than the penetrating dose. This assumption results in a maximum nonpenetrating dose of 41.6 rem in a year to areas of the body other than hands and forearms. Nonpenetrating doses are reconstructed for tissues and organs near the body surface, including breast, penis, testes, skin and eye, by applying an organ dose conversion factor.

It should be noted that the beta (0.07 mg/cm^2) dose rate at the surface of an aged uranium slab is approximately 0.23 rad per hour. Melting and casting can result in increases of beta dose rates to a range of 1 to 20 rad per hour, because these operations cause the short-lived uranium progeny to concentrate at the surface of the metal (DOE 2004). The dose rates decrease as the short-lived progeny decay or are physically removed. Because this phenomena has been recognized for some time, it is likely that extra care was taken to minimize dose after uranium was just melted or cast, but dose reconstructors should be aware of these higher dose operations as they review individual claims.

3.2.3 Occupationally Required Medical X-Ray at Uranium or Thorium Metal Handling Facilities

Information regarding whether or not occupationally required medical x-ray examinations were performed at AWEs is not always available. AEC usually, but not always, required “preemployment”, periodic (annual), and termination medical examinations of workers involved in the larger uranium and

thorium programs. The term "preemployment" as used here, means prior to performing AEC-contracted radiological work. Although a typical examination most likely involved a PA radiography chest examination using 14 x 17 inch films, photofluorography examinations might also have been employed. ORAUT-PROC-0061, *Occupational X-Ray Dose Reconstruction for DOE Sites* (ORAUT 2004b), Table 1 shows that photofluorography examinations might have continued through 1961 at some DOE facilities. For the maximizing approach, occupationally required medical x-ray examinations are assumed to consist of photofluorography chest examinations through 1961 at the metal handling facilities. Beginning in 1962, required x-ray examinations are assumed to consist of PA chest x-rays. Organ doses can be obtained or assigned in accordance with information from the current revision of ORAUT-OTIB-0006, *Technical Information Bulletin: Dose Reconstruction from Occupationally Related Diagnostic X-Ray Procedures* (ORAUT 2003).

3.2.4 Occupational External Dose Reconstruction Assumptions and Summary

Table 3-8 summarized the components of external dose reconstruction for both operational and residual exposure periods. The basis for the residual exposure assumptions are presented in the next section 3.2.5. To expedite and simplify inclusion of organ doses due to industrial radiation sources, it is reasonable and favorable to claimants to use the 1939 total dose distribution for all years adjusted by the mode of the "Ambient Dose Equivalent (H*(10)) to Organ Dose (HT)", AP, 30 -250 keV photon dose conversion factors in Appendix B of the NIOSH External Dose Reconstruction Implementation Guideline (NIOSH 2002), until a more efficient method is available for inclusion of these doses.

Table 3-8. External exposure summary.

Source	Exposure Information	Notes	Period of applicability	Annual exposure	IREP distribution
Uranium metal	Penetrating (Photons, 30-250 keV, AP, acute)		Operational covered period	4.16 rem	Constant
Uranium contamination	Penetrating (Photons, 30-250 keV, AP, acute)	Accounts for skin dose and penetrating dose from surface contamination. Does not account for nonpenetrating dose to other shallow organs.	Operational covered period and residual period	Radiation from 1.66E+08 dpm/m^{2}of natural uranium surface contamination, plus equilibrium contamination amounts of short-lived progeny	Constant
Uranium metal	Nonpenetrating (Electrons >15 keV, acute)	Skin on hands and forearms	Operational covered period	252 rem	Constant
Uranium metal	Nonpenetrating (Electrons >15 keV, acute)	Skin (other than on hands and forearms) and shallow organs	Operational covered period	41.6 rem	Constant
Uranium contamination	Nonpenetrating (Electrons >15 keV, acute)	Shallow organs other than skin	Residual period	10 times the penetrating dose derived from uranium contamination	Constant
Industrial Radiography	Penetrating (Photons, 30-250 keV, AP, acute)		Operational covered period	See Appendix A	Lognormal
Other industrial sources	Penetrating (Photons, 30-250 keV, AP, acute)		Operational covered period	See Appendix A	Lognormal
Medical x-ray	Photofluorographic chest examination	Initial plus one examination per year plus termination	Before 1962	See ORAU 2003	
Medical x-ray	PA radiographic chest examination	Initial plus one examination per year plus termination	After 1961	See ORAU 2003	

3.3 URANIUM METAL HANDLING FACILITIES – ESTIMATION OF EXPOSURE TO RESIDUAL RADIOACTIVITY

During operations with uranium, airborne contamination can be caused directly from operations with uranium bearing materials as well as by the resuspension of surface contamination. This airborne contamination can be removed by settling on floors or other surfaces or by ventilation (both engineered and natural building ventilation). The surface contamination levels can be increased by

the airborne contamination settling or directly by operations with uranium. This level is decreased by various incidental means such as general housekeeping or tracking of contamination out of the area on equipment or personnel.

For the purpose of estimating the airborne contamination from residual contamination, it was assumed that during uranium operations, these factors reached an equilibrium that caused the airborne concentration and surface contamination to remain at a constant level. Once the generation source from uranium operation was removed, the other factors then caused the over mass of uranium to decrease at some rate due to the remaining removal factors.

This analysis assumes that 1) uranium operations directly accounted for half the airborne concentration in the facility during operations and 2) all the remaining removal mechanisms eliminated only 1% of the uranium mass per day. With these assumptions in place, an equation can be written for the airborne concentration as a function of time.

$$C(t) = 50MAC * e^{-0.01*t}$$

Where:

C(t) = Airborne activity at time t (pCi/m^3)
t = time since uranium operations ended (days)
MAC = Maximum Allowed Concentration (70 dpm/m^3)
0.01 = fraction of material removed per workday

When this is multiplied by the breathing rate of 9.6 m^3 per workday, the daily intake is calculated. This equation can then be integrated from the end of uranium operations to infinity to determine the total amount of uranium that could be inhaled from residual contamination. This integration results in:

$$I = 9.6 * 50MAC / 0.01$$

Where:

I = total intake from inhalation of residual activity (dpm)
9.6 = daily breathing rate (m^3/workday)

A total uranium inhalation intake of 3.36E+6 dpm was assumed to occur during the residual contamination period. This equates to 20% of the 1.68E+7 pCi annual inhalation intake during operations. Therefore, the total uranium inhalation intake from residual contamination can be overestimated by assuming one additional year of inhalation exposure at the operational level of 100 MAC. In addition, one additional year of intake of the recycled uranium contaminants would also be assumed. The intake rates for airborne exposures at 100 MAC are listed in Table 3-5.

Ingestion of material is most closely related to removable surface contamination rather than fixed surface contamination. Airborne concentrations are also related to removable surface contamination. Therefore, the removable contamination levels are assumed to decrease at the same rate as the airborne concentrations. For this reason, the same approach will be taken for ingestion estimates following periods of operations. That is, the ingestion intakes from residual contamination will be estimated by assuming one additional year of ingestion at the same rate it occurred during operation.

The external radiation caused by residual contamination can be overestimated by assuming the contamination levels that existed during operations continued to exist after operations ceased. The

method for estimating the annual dose from this source of radiation is specified in section 3.2.1 for penetrating dose and for nonpenetrating skin dose. Nonpenetrating annual organ doses for the breast and testes can be estimated by multiplying their annual penetrating doses determined by this method by a factor of ten. These external doses should be added to an individual's estimate for each year the individual worked at the facility following the end of uranium operations, during periods of DOE remediation, and during breaks between AEC operations in the case of multiple operational periods.

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ATTACHMENT A
Industrial Radiation Doses to Be Included in Maximum Estimates

Table A-1 is a summary of annual external dose distributions for industrial users of radiation sources based on EPA (1984) summary dose data for four radiography periods and three other industrial exposure periods during 1960 through 1980, which were further analyzed as cumulative probability distributions (Eheman and Tolbert 1999). The dose data for the reported periods were assumed to occur in a year in the middle of the reported range. These dates and doses were fit to a line to estimate earlier doses. It was assumed that the doses did not change after 1979. The geometric standard deviations (GSDs) prior to 1960 were calculated by fitting the dates and product of the geometric means (GMs) and their GSDs, i.e., the 84th percentile doses, to a line; predicting the pre-1960 years and than dividing those 84th percentile values by the geometric mean (assumed to be the median) to estimate GSDs. For each year of a worker's covered employment period, the industrial radiography plus the other industrial exposure is to be included in the dose reconstruction, except as noted in Section 2.0.

Table A-1. Annual industrial dose distributions

Year	Industrial radiography exposure		Other industrial exposure		Total industrial exposure	
	GM rem	GSD	GM rem	GSD	GM rem	GSD
1939	0.432	3.51	0.126	2.69	0.558	3.32
1940	0.424	3.5	0.125	2.67	0.548	3.31
1941	0.415	3.49	0.123	2.65	0.539	3.30
1942	0.407	3.48	0.122	2.64	0.529	3.29
1943	0.399	3.47	0.121	2.62	0.519	3.27
1944	0.390	3.46	0.119	2.6	0.509	3.26
1945	0.382	3.45	0.118	2.58	0.500	3.25
1946	0.374	3.44	0.116	2.56	0.490	3.23
1947	0.365	3.43	0.115	2.54	0.480	3.22
1948	0.357	3.41	0.113	2.52	0.470	3.20
1949	0.349	3.4	0.112	2.5	0.461	3.18
1950	0.340	3.39	0.110	2.48	0.451	3.17
1951	0.332	3.37	0.109	2.46	0.441	3.15
1952	0.324	3.36	0.108	2.44	0.431	3.13
1953	0.316	3.34	0.106	2.42	0.422	3.11
1954	0.307	3.33	0.105	2.39	0.412	3.09
1955	0.299	3.31	0.103	2.37	0.402	3.07
1956	0.291	3.29	0.102	2.34	0.393	3.04
1957	0.282	3.27	0.100	2.32	0.383	3.02
1958	0.274	3.25	0.099	2.29	0.373	3.00
1959	0.266	3.23	0.098	2.27	0.363	2.97
1960	0.250	3.13	0.091	2.12	0.341	2.86
1961	0.250	3.13	0.091	2.12	0.341	2.86
1962	0.250	3.13	0.091	2.12	0.341	2.86
1963	0.250	3.13	0.091	2.12	0.341	2.86
1964	0.250	3.13	0.091	2.12	0.341	2.86
1965	0.250	3.13	0.091	2.12	0.341	2.86
1966	0.171	2.94	0.091	2.12	0.262	2.66
1967	0.171	2.94	0.091	2.12	0.262	2.66
1968	0.171	2.94	0.091	2.12	0.262	2.66
1969	0.171	2.94	0.091	2.12	0.262	2.66
1970	0.171	2.94	0.069	1.53	0.240	2.54
1971	0.119	2.3	0.069	1.53	0.188	2.02
1972	0.119	2.3	0.069	1.53	0.188	2.02
1973	0.119	2.3	0.069	1.53	0.188	2.02
1974	0.119	2.3	0.069	1.53	0.188	2.02
1975	0.119	2.3	0.069	1.53	0.188	2.02
1976	0.119	2.3	0.069	1.53	0.188	2.02
1977	0.119	2.3	0.069	1.53	0.188	2.02
1978	0.119	2.3	0.069	1.53	0.188	2.02
1979	0.119	2.3	0.069	1.53	0.188	2.02
1980+	0.107	2.1	0.071	1.56	0.178	1.88

ATTACHMENT B
Uranium Metal Handling

Table B-1. List of applicable facilities.

Site name	Years of operation	U enrichment	Account for other radiation sources?*
Ajax Magnethermic Corp.	1958–1962	Normal	No
Alba Craft	1952–1957	Normal	No
Aliquippa Forge	1947–1950	Normal	No
Allegheny-Ludlum Steel	1950–1952	Normal	No
Aluminum Co. of America (NJ)	1944	Normal	No
Aluminum Co. of America (PA)	1944–1945	Normal	No
AMCOT	1961–1962	Normal	No
American Bearing Corp.	1954–1959	Normal	No
American Chain and Cable Co.	1944	Normal	No
American Machine and Foundry	1951-1954	Normal	Thorium
American Peddinghaus Corp.	1978	Normal	No
Anaconda Co. (Connecticut) [a]	1942, 1956–1959	Normal	No
Armco-Rustless Iron & Steel	1948	Normal	No
Associated Aircraft Tool and Manufacturing	1956	Normal	No
B&T Metals	1943	Normal	No
Baker Brothers	1943–1944	Normal	No
Baker-Perkins Co.	1956	Normal	No
Bell Telephone Laboratories[b]	1943–1944	Normal	No (small check sources not considered a health hazard)
Bendix Aviation (Pioneer Division)	1960	Normal	No
Besley-Wells	1953	Normal	No
Bethlehem Steel	1949–1952	Normal	No
Birdsboro Steel & Foundry[c]	1951–1952, 1962	Normal	No
Bliss & Laughlin Steel	1948–1952	Normal	No
Bloomfield Tool Co.	1947, 1951	Normal	No
Bridgeport Brass Co., Adrian	1954-1962	2%	Thorium
Bridgeport Brass Co., Havens Lab		Normal	Thorium
Brush Beryllium Co. (Cleveland)	1942-1943, 1949-1953	Normal[g]	Thorium
Brush Beryllium Co. (Detroit)	1940s-1950s	Normal	No
BWX Technologies (Virginia)	1959, 1968-1972, 1985-2001	93.5%	Thorium, Plutonium, LINAC
C.H. Schnoor	1943–1951	Normal	No
C.I. Hayes, Inc.	1964	Normal	No
Callite Tungsten Co.	1944	Normal	No
Carboloy Co.[d]	1956	3.5% [k]	No
Carborundum	1944 (termination prior to 1960)	Normal	Plutonium 1960-1962
Carpenter Steel Co.	1943–1944	Normal	No
Chambersburg Engineering Co.	1957	Normal	No
Chapman Valve	1948–1949	Normal	No
Cincinnati Milling Machine Co.	1963	Normal	No
Combustion Engineering	1965–1972	93.5%	No
Copperweld Steel	1943–1946	Normal	No
Crane Company	1947-1949	Normal	Thorium
Dow Chemical Co. (Madison Site)	1957–1960	Normal	Thorium
Electro Circuits, Inc.	1952–1953	Normal	No
Extruded Metals Company	1944	Normal	No
Extrusion Plant	1962-present	3.5% [k]	No
Fenn Machinery Co.	1950	Normal	No
Fenwal, Inc.	1967–1968	0.95%	No

Table B-1 (Continued). List of applicable facilities

Site name	Years of operation	U enrichment	Account for other radiation sources?*
Frankford Arsenal	1952–1954	Normal[h]	No
Granite City Steel	1953–1966	Normal	Radiography (betatron)
Great Lakes Carbon Works[e]	1952-1958	93.5%	Thorium
Gruen Watch	1956	Normal	No
Heald Machine Co.	1960	Normal	No
Heppenstall Co.	1955	Normal	No
Herring-Hall Marvin Safe Co.	1943–1951	Normal	No
Hunter Douglas Aluminum Corp.	1959–1963	Normal	Thorium
International Nickel Co., Bayonne Laboratories	1951–1952	Normal	No
International Register	1943	Normal	No
Ithaca Gun Co.	1961–1962	Normal	No
J.T. Baker Chemical Co.	1948, 1957–1958	Normal	
Jessop Steel Co.	1950–1954	Normal	No
Joslyn Manufacturing and Supply Co.	1944–1952	Normal	No
Kaiser Aluminum Corp.	1959	Normal	No
La Pointe Machine and Tool Co.	1956	Normal	No
Landis Machine Tool Co.	1952	Normal	No
Magnus Brass Co.	1954–1957	Normal	No
McKinney Tool and Manufacturing	1944	Normal	No
Medart Co.	1951–1952	Normal	No
Midwest Machining	1944	Normal	No
Mitchell Steel Co.	1954	Normal	No
New York University	1946–1952	Normal	No
Norton Company	1945-1957	Normal	Thorium
Oliver Corp.	1956–1957, 1961–1962	3.5% [k]	No
Podbeliniac Corp.	1957	Normal	No
Precision Extrusion Co.	1949–1950, 1956–1959	Normal	No
Quality Hardware and Machine Co.	1944–1945	Normal	No
R.W. Leblond Machine Tool Co.	1961	Normal	No
Reed Rolled Thread	1955	Normal	No
Revere Copper and Brass	1943-1950s	Normal	No
Roger Iron Co.	1956	Normal	No
Sciaky Brothers, Inc.	1953	Normal	No
Seymour Specialty Wire	1962–1964	Normal	No
Simonds Saw and Steel	1948-1956	Normal[i]	Thorium
Southern Research Institute	1955–1958, 1962, 1976	Normal	No
Sperry Products, Inc.	1952–1953	Normal	No
Star Cutter Corp.	1956	Normal	No
Stauffer Metals, Inc.	1961	Normal	No
Superior Steel Co.	1952–1957	Normal[i]	No
Sylvania Corning Nuclear Corp. – Bayside	1947-1962	Normal[j]	Thorium
Sylvania Corning Nuclear Corp. – Hicksville Plant	1947-1962	3.5%[j, k]	Thorium
Titus Metals[f]	1956	93.5%	No
Tocco Induction Heating Div.	1968–1969	Normal	No
Torrington Co.	1951–1953	Normal	No
Tube Reducing Co.	1952, 1957	Normal	No
U.S. Steel Co., National Tube Division	1959–1960	Normal	No
University of Michigan	1944	Normal	No
Vulcan Tool Co.	1959	Normal	No
W.E. Pratt and Company	1943-1946	Normal	No

Table B-1 (Continued). List of applicable facilities.

Site name	Years of operation	U enrichment	Account for other radiation sources?*
Wah Chang	1956-1979, 1971-1972	Normal	Thorium
Westinghouse Nuclear Fuels Division	1971-1972	3.5%[k]	Plutonium
Wolverine Tube	1943-1946	Normal	Thorium
Wyckoff Drawn Steel Co.	1943	Normal	No
Wyckoff Steel Co.	1950	Normal	No

(*Additional sources of radiation exposure have been identified for the listed site. An estimate of maximum plausible dose might be needed to account for radiation exposure from these other radiation sources. Updates will be made to this list as additional information becomes available.)

a. Anaconda was involved in gaseous diffusion barrier production in 1942. Based on available documentation, it appears that the later uranium work might have occurred in 1956 and in 1959, not 1956 through 1959, as noted by the DOE Office of Advocacy. Additional information is being sought to address this discrepancy.
b. Bell Telephone Laboratories was involved in gaseous diffusion barrier production.
c. Concerns about radiography exposure were noted in at least one claim for this site. The described exposure might have occurred outside the covered period.
d. Their license indicated that enrichment up to 3% was allowed, 3.5% is used here because it is listed in Table 5.
e. TREAT reactor fuel, assumed HEU was 93.5%.
f. Argonaut reactor fuel, probably in the range of 20 to 90%, assumed 93.5%.
g. UF_4 to metal production
h. UCl_4
i. Although some EU was processed, normal uranium intakes will still provide an overestimate.
j. Uranium hydrides
k. Enrichment assumptions are subject to change based on review of additional data; it is likely that the combination of enrichment and air concentration assumptions will overestimate intakes.