For inhalation, the daily weighted exposure levels given in Tables 19-22 were converted to annual intakes as given below. The Plant 6 results for 1948 were used for Plants 1 and 2. The results are shown in Tables 29A-29F.

Intake (pCi/yr) = (dust concentration (dpm/m^3) x 1.2 m^3/hr breathed x 2000 hr/yr)/(2.2 dpm/pCi)

For radon, no daily weighted exposure levels are available and the reported levels tabulated in Tables 23-25 apply only to radon and not its daughters. Thus assumptions had to be made to determine conservative exposure levels, occupancy factors, and daughter equilibrium levels. The conversion to Working Level Months per year (WLM/yr) is as given below. The results, including the detailed parameter values used, are shown in Table 30.

Intake (WLM/yr) = (radon level (pCi/L) x equilibrium factor x 12 months/yr)/(100 pCi/L per WL)

(Note that for use in dose reconstruction, radon concentrations reported in units of Ci/L must be converted to units of working level months (WLM) before they can be used in calculations. One working level (WL) is the total amount of energy given off over a long period of time by the short-lived radon-222 daughters in equilibrium with 100 pCi (10^{-10} Ci) of radon, taken to be in one liter of air. Since the daughters will typically not be in equilibrium with the radon if the ventilation is good, this conversion is not simple, but depends on the ventilation conditions. The reported results are reported in this technical basis document in the original units so that the dose reconstructor can readily use an equilibrium factor different from the ones applied in Table 30, if desired.)

For ingestion, the method of NIOSH (2004) is used with the daily weighted exposure levels given in Tables 19-22 to give annual intakes, as given below. This method is expected to yield very conservative results. The Plant 6 results for 1948 were used for Plants 1 and 2. The results are shown in Tables 31A-31F.

Intake (pCi/yr) = (0.2)(dust concentration (dpm/m^3) x 1.2 m^3/hr breathed x 2000 hr/yr)/(2.2 dpm/pCi)

5.4 EXTERNAL DOSE CONSIDERATIONS

External doses for Mallinckrodt workers varied widely depending upon the activity they performed. Operations at the refinery (Plant 6) involved primarily gamma radiation, while operations at the metal plants (i.e., Plant 4 and later 6E) entailed primarily beta radiation (AEC 1949b; AEC 1951b)). ORAU (1980b) stated that there did not appear to be any circumstance that would produce an exposure of 3 rem or greater in less than several hours and they found no incident in which an acute exposure rate of greater than or equal to 3 rem/hour was produced. The tables below are an adaptation of Table I and a summary of information given in Tables II of ORAU (1980b); the first shows the total annual gamma dose by dose level and the second shows the ten highest gamma doses to workers, both from 1947 through June 1957.

Information regarding dose rates for ore and residue drums is given in Table 32 and regarding dose rates for various forms and areas in Table 33. There is little information about conditions in Plants 1 and 2 during the wartime startup; however, some dose rate measurements from 1943 and 1944 appear to have been taken, as shown in Table 33. As noted previously, film badging did not start until late 1945, when Plants 1 and 2 were in the process of largely shutting down. Doses might have been somewhat higher than in Plants 4 and 6 due to possible greater manual involvement and probably somewhat greater bodily proximity to sources, but on the other hand the quantities involved were much lower. The dose rates given in Table 33 suggest that doses were likely to have been higher in later years due probably to the greater volume of uranium processed and to the use of pitchblende

Annual gamma exposures, 1947 – June 1957 (ORAU 1980b).

Year	Workers monitored	Annual gamma exposure, R				
		0 - 1	1 - 5	5 - 10	10 - 15	>/= 15
1947	253	70	131	27	18	7
1948	366	120	171	47	19	9
1949	554	370	141	41	2	0
1950	615	475	133	7	0	0
1951	694	512	171	11	0	0
1952	757	659	88	10	0	0
1953	763	619	142	2	0	0
1954	756	566	188	2	0	0
1955	871	766	105	0	0	0
1956	958	944	14	0	0	0

Ten highest gamma doses to workers, 1947 – June 1957 (ORAU 1980B).

Year	Annual dose, R			Weekly dose, R/week		
	Min	Avg	Max	Min	Avg	Max
1947	14.4	16.1	23.5	0.28	0.31	0.45
1948	14.9	17.0	20.3	0.25	0.33	0.39
1949	7.7	9.0	13.3	0.15	0.17	0.26
1950	4.5	5.4	7.1	0.09	0.11	0.14
1951	5.0	5.9	7.1	0.10	0.11	0.14
1952	5.1	5.9	6.6	0.10	0.12	0.13
1953	4.0	4.6	5.7	0.08	0.09	0.11
1954	3.9	4.4	5.1	0.08	0.09	0.10
1955	3.9	4.4	5.1	0.08	0.09	0.10
1956	1.1	1.4	1.9	0.04	0.06	0.07

ores. It should be noted that the era of pitchblende use (early 1945 on) was mostly covered by film badge monitoring, so that the doses characteristic of this work are known. Thus it is considered to be claimant-favorable to assume that the doses at Plants 1 and 2, for the same type of work, were similar to those at Plant 4 and Plant 6 around 1948.

According to MCW (1955f), at least late in the life of the site, gamma surveys were done bimonthly in most Plant 6 processing areas and monthly at the vent ducts in the digest area. Although these reports do not appear to be available, Table 33 shows some area film badge monitoring results that show what conditions were and of course film badges were consistently worn in the later years. A 1955 list of Plant 6 process cells, dust collectors, and tanks, together with the then-current uses of the tanks, appears in MCW (1955g); maximum emergency nonsurvey stay times (i.e., for urgent access and not inspections) are also given. It appears that the stay times were based on external dose rates. In general, the operator stay times were half the maintenance stay times, presumably because of the need for more frequent access by the operators.

See also Sections 5.1 – 5.2 above for some addition external dose information associated with particular handling activities or operations.

5.4.1 Gamma, Beta (Electron), and Nonspecific Beta-Gamma Exposures

After high-grade pitchblende ores began to be used, refinery workers were exposed to photons from radionuclides in equilibrium with U-238 and U-235. Ra-226, through its Pb-214 and Bi-214 daughters, contributed gammas to workplaces where ore was stored or processed. Upon removal of the uranium daughters, processed material became radiologically innocuous until the passage of time resulted in the ingrowth of Th-234 and Pa-234m and the consequent domination of the dose profile by electrons. Mallinckrodt worker dose records demonstrate this difference, with significant doses for mixed photons and electrons in the refinery operations and high electron doses with little photon dose in the metal plants. Dose reduction measures in plants and equipment resulted in low doses in Plants 6E and 7 compared with the mixed beta-gamma doses in the refinery operations.

In 1946 the principal large-scale source of gamma radiation was said to be the drums of ore as they were stacked in the receiving warehouse (MED 1946c). The gamma dose rate could be as high as 50 mR/hr near stacks of drums of Belgian Congo ore at 25% concentration and with a radium content of about 100 mg/ton (Eisenbud 1975). A 1958 AEC report on uranium mills gave dose rates of 0.8 to 8.0 mR/hr, with an average of 3.0 mR/hr, as the gamma dose rate at three feet from bulk ore concentrates (AEC 1958, Table XI); these dose rates are assumed to be for domestic ores. Dose rates from wastes could run even higher than those for the ore. Dose rates at points adjacent to stacks of drums of radium-bearing residues (precipitates) could run as high as 100 mR/hr adjacent to stack of drums (~ 300 mg Ra/ton) (Eisenbud 1975, Table 2) or up to 275 mR/hr at contact (MCW 1949g). AEC (1948d) gave the gamma contact dose rate with the (Ra-containing) Feinc filtrate residue under equilibrium conditions as over 300 mR/hr; however, they stated, they had no way of knowing how close to equilibrium it was. The dose rate depended strongly on how “aged” the material was: sample pans filled with fresh K-65 read 9 mR/hour while pans held over a day read 20 mR/hr (MCW 1948j).

A study done by AEC of the dose rates from four drums of “aged” K-65 residue grouped on a pallet showed that the dose rate at 6 feet from the grouping was 28 mR/hr, of which the back two contributed 16% and the front two 86% (AEC 1947c). Furthermore, it was found that the dose rate varied approximately as the square of the distance over the range of from three to twelve feet from the center of the grouping. With lead shielding interposed, it was found that the first half-value layer (HVL) was 1/4” and the second was 3/8”.

As noted in Sections 4.7 and 5.2.3, K-65 was brought back in drums from SLAPS and reprocessed starting in early 1948. In January 1949, the gamma dose to the hands of operators opening these drums was studied by placing film badges on their wrists (MCW 1949E). It was found that the gamma dose varied from 3.3 to 5.0 mR per drum opened, with an average of 4.0 mR; for the typical situation of 44 drums opened per 24-hour day, six days per week, by three operators, the total weekly exposure per operator was 352 mR. It was estimated that this could increase by up to a factor of 2 due to drums that might require more time than average to open, variations among operators in opening the drums, and in the number of drums opened per shift. Hence the maximum wrist dose was estimated to be 700 mR per week per operator. At the time, the individual whole-body gamma dose to the operators was 300-500 mR per week; it was thought that 33-67% of this dose was due to the drum opening (MCW 1949e).

Additional specific information regarding gamma doses and dose rates in the ore, refinery, and metal processing areas are shown in Tables 32 and 33: dose rates from drums and railcars are shown in Table 32 and for various plant locations and tasks in Table 33. Additional gamma-beta dose data for ore storage at the Middlesex plant during 1944-1949 is shown below (AEC 1949b, Figure 3); this should be comparable to Mallinckrodt experience because presumably the ore was being routed through Middlesex to be distributed to the processing sites. AEC (1949b) states that the Middlesex doses were about 65% gamma.

Middlesex ore storage worker dose, 1944-1949 (AEC 1949b, Figure 3).

Worker type	Number of workers	Weekly gamma-beta dose, mrep
Guard	11	150
Laborer	20	500-600
Labor Foreman	1	250
Laboratory Technician	2	300
Maintenance	10	150-500, avg 300
Office	---	100
Timekeeper	1	250

It should be noted that operations that were particularly manual were (1) the various dumping, scooping, and scraping operations in which feed, UO_2, UO_3, UF_4, and dust were handled or crucibles and furnaces were cleaned; (2) the "plowing" (scraping) of the centrifuges; and (3) the scraping of cake off the Feinc filter cloths (this was the pitchblende cake during the pitchblende years) and the changing of these cloths. Thus significant external dose reduction usually followed any mechanization or improvement of these processes.

Because the gamma dose arose mainly from the radium and its daughters, the gamma dose was usually significant only in those areas where the source material had not yet had the radium separated; where radium-bearing residues were present; or where uranium products were stored for long enough periods of time that the daughters built up again. This meant that the gamma doses tended to be highest in Plant 2 and later in Plant 6 (AEC 1949b), especially around ore drums and storage areas for the radium-bearing residue, K-65. Thus some shielding had been designed into Plant 6 and more was added in 1948 in some areas (AEC 1949b, MCW 1950e). Still, due to the high doses Mallinckrodt found it necessary in mid-1949 to establish additional restrictions and rotation requirements on warehouse workers, who moved stored pitchblende ore drums from the airport waste storage site (apparently also an overflow ore storage site); brought "sand" (residue) drums back from the airport; unloaded the incoming ore drums from railcars; weighed the ore drums coming in from the railcars and airport site; and loaded K-65 residue drums into the railcars (MCW 1949p). It was estimated, e.g., that the following were typical weekly activities: three trips for retrieving ore drums from the airport pad, three or four drum weighing sessions, one trip for retrieving residue drums, two K-65 railcar loadings, and two ore railcar unloadings (MCW 1949p). It was required that pocket chambers be used when retrieving K-65 drums from the airport pad (MCW 1949p).

The highest principal Feinc cloth operators' dose had been found to be an average of about 465 mrep/week (total beta and gamma) in 1949 and 1950 and the second-highest dose was similar (MCW 1951g). In January 1951, a study of beta and gamma exposures associated with preparing, repairing, cleaning, and changing cloths for the Feinc, C-3, and Recovery filters (MCW 1951a) found that operators received an average of 294 mrep per day and 1470 mrep per week gamma and an average of 82 mrep per day and 408 mrep per week beta. (See Table 33 for some of the measured dose rates.) The "per day" figures were an effective daily dose since the operation was not actually performed daily. These figures were comparable to those found in a previous 1949 study, although some of the tasks the cloth operators were doing in 1949 had been given to the area operators to do (MCW 1951a).

Mallinckrodt (MCW 1950t) summarized the gamma dose data for all high-dose workers as in the table below. Although it would appear from the disparity between the two averages below that a few individuals were getting significantly higher doses than the rest, Mallinckrodt contended that the doses were fairly evenly distributed among workers in the five rotation groups and thus that all that could be accomplished using rotation had been accomplished. For comparison, group data reported by AEC (AEC 1951b) for two periods in 1948-1949 and 1950 are shown.

Comparison of gamma and beta doses, 1946-1950.

MCW 1950t				
	Date	**Avg of high badges, mR or mrep**	**Avg of 90th percentile badges, mR or mrep**	
Gamma only	1946	780	250	
	1947	860	290	
	1948	590	250	
	1949	410	170	
	1950 (1st quarter)	320	---	
Gamma plus beta	1950 (1st quarter)	700	235	
AEC 1951b				
	Period	**Number over 30 mrep/week**	**Number over 150 mrep/week**	**Number over 300 mrep/week**
Gamma only, Plant 6	11/1/48 – 1/24/49 (267)	103	47	17
	1/2/50 – 6/19/50 (314)	124	47	11
Beta only, Plant 4	11/1/48 – 1/24/49 (91)	83	35	32
	1/2/50 – 6/19/50 (89)	65	29	12

The AEC 1951b figures are weekly averages. The number in parentheses after the period is the total number of badges worn per week, on average. The beta figure for “over 300 mrep/week for 11/1/48-1/24/49 included 8 over 700 mrep/week, 5 over 1000 mrep/week, and 2 over 1500 mrep/week; the corresponding inclusions for 1/2/50-6/19/50 were 2, 1, and zero mrep/week.

As noted in Section 5.2 above, there was also up to 2.6 mCi of radium built up in the residue that was processed in 1955-1957 to concentrate thorium, although this was distributed in the 350 tons that was processed into the 3600 gallons of solution sent to Mound (Tables 4 and 6).

Doses registered on film badges worn by people not working directly with the uranium and equipment, such as guards and office workers, was more likely from gamma exposure than from beta exposure. Thisis because they were usually at some distance away from the source (the uranium and its daughters). It is true that the dust was found throughout the plant to varying extents, but that would likely not contribute to the external dose rate much in or near buildings where there was a substantial radium content in any uranium product or residue.

A 1958 AEC report on uranium mills gave 1.5 to 25 mrep/hr, with an average of 15.5, as the beta dose rate at three feet from bulk ore concentrates (AEC 1958, Table XI). AEC estimated the dose to an operator’s hands from removing lids from ore drums at 200-300 mR/day, even after a proposed body shielding window was erected (AEC 1948e). AEC (1948d) gave the beta contact dose rate with the (radium-containing) Feinc filtrate sludge (K-65) under equilibrium conditions as over 500 mrem/hr; however, they stated, they had no way of knowing how close to equilibrium it was.

Regarding experience at the Paducah site, Baker (1958) reported that the Th-234/Pa-234 combination (from U-238 and U-234) produced about 1500 alpha dpm/mg U and 1500 beta dpm/mg U at equilibrium, producing 240 mrad/hr at the surface of U metal, 208 mrad/hr at the surface of UO_3, and 183 mrem/hr at the surface of UF_4. Further, during UO_3 prep by “our suppliers” (e.g., Mallinckrodt), much of the beta-active material was removed, but built back up to 50-100% by the time it got to the UF_6 production facilities (Baker 1958). This suggests that significant buildup could occur before the UO_3 left the Mallinckrodt facilities since the storage time might be weeks and the transport time was likely less than a few days. Eisenbud (1975) pointed out that 90% of equilibrium beta activity was restored by 90 days after vacuum casting. Eisenbud (1975) reported high dose rates, up to 1 rad/week to the body and even more to the hands, from loading of UF_4 into UF_6 reaction vessels. This too implies that if enough time had elapsed, UF_4 loaded at Mallinckrodt into the bombs could also produce relatively high beta dose rates. Metallic uranium in equilibrium with Th-234/Pa-234 could produce up to 235 mrad/hr to the basal epithelium when the metal was in contact with bare skin; heavy gloves would significantly reduce this (Eisenbud 1975).

In addition to the beta dose rate from the uranium as natural uranium, uranium oxide, etc., there were two waste concentrates that produced high beta dose rates. First, when ether was used to extract the uranium from uranyl nitrate, Th-234 and Pa-234m (also called UX1 and UX2 respectively) were left in the aqueous phase (also called the aqueous uranium tails) (Eisenbud 1975). This aqueous solution was filtered, resulting in a residue (cake) containing the beta emitters. MED (1942) stated that 1942 measurements indicated that the intensity was low and that no precautions needed to be taken for disposal; however, MED/AEC appears to have been more concerned about this later on. Another source of the tails was the UO_2-derived shotgun sample, which could have the Th-234 and Pa-234 concentrated to 30-300 times their activity in normal uranium metal in equilibrium, depending on how long the UO_2 had stood between production and sampling (MED 1944m). The fourth and final ether extraction performed in processing a shotgun sample produced a liquor so concentrated in these beta emitters that it was said that the tolerance dose of beta radiation could be reached by keeping the hands above the liquor for 10 minutes per day. Besides that, the chemist handled the sample for 5-10 minutes from removal from the furnace to bottling, wearing no gloves and directly touching the containers. MED (1944m) advised changing from rubber gloves to leather gloves (to increase the dose rate reduction from a factor of 2 to a factor of 3) and using crucible tongs (to increase the distance to about 10") for conveying the evaporation dish to the heating areas.

Second, in the vacuum recasting of the uranium metal, impurities in the metal volatilized and condensed on the cooler portions of the furnace, creating spot deposits (AEC 1949b; Eisenbud 1975). The impurities contained Th-234 and Pa-234, which were concentrated to a significant degree in the deposits (AEC 1949b; Eisenbud 1975); this deposit residue could have "up to 1000 times the beta activity of natural uranium" (AEC 1949b). Manual contact with these deposits during charging, discharging, cleaning, and repair of the furnaces provided "opportunity for hand irradiation of a greater magnitude than whole body" (AEC 1949b), possibly as much as 2-3 rads/week to exposed skin and perhaps to the eyes when the original ore was pitchblende at 25% average enrichment (Eisenbud 1975). Mallinckrodt (MCW 1949a) observed that 25% of Plant 4 workers received over 500 mrep/week beta and 3-6 workers per week received 2000 mrep or more; AEC (1949g) also observed that the beta values (on film badges) from Plant 4 consistently ranged up to 2.7 rep/week.

Because of the high hand doses, the processing of the derbies and billets was studied in 1948 by Mallinckrodt (MCW 1948b). They gave the results to the designer of the new Plant 6E for use in redesigning the process to reduced the doses, first by eliminating exposure to large open surfaces uranium forms and second by prevent exposure to and accumulation of scale and powdered residues (in which the UX1 and UX2 were concentrated), especially in the recasting or remelt step.

Regarding the processing of residues to concentrate thorium, Table 6 shows that with the interruption in the chain occasioned by the removal of the original radon (by venting) and the radium early in the process, the daughters had to build up again to equilibrium from the time the cake was stored through the maximum 15 years of storage. Consequently the strong beta emitters down the chain, such as Pb-214 and Ac-228, are present only in very small quantities.

Some dose rate information for exposure rates from laundry equipment and clothing appears in Section 5.3.5 and in Table 27 (from the text and Table 1 of Utnage 1958b). This is mainly beta radiation.

MED did some studies to determine the shielding afforded by gloves and clothing. MED (1943c) reported that leather gloves of 1 mm thickness cut the beta dose rate by a factor of 2, while 2-mm gloves cut the beta dose rate by over a factor of 4. MED (1944i) gave the reduction factor as 2 in a 1944 study of rubber gloves, with the source being uranium metal in equilibrium. MED (1944a) reported the results of July-September 1944 measurements of the activity inside 16 pairs of gloves; the results showed that contamination did appear to get into gloves used at the recast furnaces but not in those used for billet sawing. This implied that some of the recast dose to the hand came from contamination inside the gloves. Based on all these results, MED (1944i) gave the time limit for handling uranium metal as 4 hours per day with rubber gloves and recommended that the time be limited to 2 hours per day with other gloves.

AEC did some further glove and clothing shielding and contamination studies using an 18” x 24” sheet of uranium metal in equilibrium with Th-234 and Pa-234 (AEC 1950i), with the following results. Denim coveralls (9-oz weight) “absorbed” an average of 22% of the beta from the source, with a standard deviation of 7.5%, for distances varying from 5 inches to 3 feet. Neoprene-covered cotton gloves shielded an average of 50%. Measurements on the inside surfaces of three cotton gloves used to handle uranium showed contact beta dose rates of 23-47 mrep/hr from contamination; these gloves had been taken at random from workers. (These figures are included in Table 27.) The Mallinckrodt glove program for contact with radioactive material was said to be sketchy and inadequate (MCW 1955d), implying that the use of gloves was not consistent.

AEC (1949g) stated the following regarding pocket meters (pocket chambers). These were typically worn by a worker two at a time for an entire work day before recharging. The readings of the two were averaged and the averaged reading was tabulated with the film badge results. The pocket meters were vibration- and leakage-tested before first use, then calibrated for 100 mR +/- 5 mR full-scale and 50 mR +/- 10 mR half-scale. Mallinckrodt suggested discontinuing the use of the pocket meters in 1949 (AEC 1949g, MCW 1949f). In MCW (1949i), they argued that the highest possible (gamma) dose rate in their plants was 300 mR/hr, that exposures of greater than 1000 mR per week were almost unknown any more, and that operations were such that extended occupancy of high-dose-rate areas was not necessary. They stated that they had not found any dose-heavy sources or operations that could not have been found by the use of film badges or surveys; also, their latest statistical calculations showed a greater than 99.9% probability of a true correlation between film badge results and pocket meter readings, but the error at the 90% limits around the regression line gave an error of +/- 38% for a dose of 300 mR per week. Thus, they concluded, using both film badges and pocket meters constituted an unnecessary double check of dose. AEC at first refused to allow them to discontinue using the pocket meters (AEC 1949g), presumably because it was the only real-time check and the only immediate post-work check of accumulated dose available, but then relented and allowed such use to be discontinued (AEC 1949h; MCW 1950e).

Detector types and measurement methods were not specified in most reports and papers. However, it is known that there were numbered, set survey points (“observation stations”) at which detector measurements were always made (AEC 1949g), MCW (1946g) stated that external dose rates were measured with an “Ion-meter”. AEC (1949g) stated that one Mallinckrodt health physicist preferred the Victoreen 247 gamma survey meter and the Landsverk electrometer, while the other preferred the Zeus meter (detector) for general use. Utnage (1958b) stated that for surface contamination measurements in the laundry (of clothing, equipment, and floor surfaces), a Victoreen 356 alpha survey meter and a Thyac beta-gamma meter with a thin-wall tube were used.

AEC (1949a) gave information regarding the Rauland Zeus detector used by Mallinckrodt. The reader is referred to the reference for the details of, e.g., the window area and the accompanying table of data. However, the conclusion was that the free area of the window was only slightly greater than 50%. This information was supplied to Mallinckrodt in the context of comparing film measurements and detector measurements, with the conclusion that the assumed effective size of a Mallinckrodt source (such as a tank or contaminated area) might be greater than had been assumed.

Counters and meters were maintained and calibrated weekly or monthly on a set schedule by Mallinckrodt technicians (MCW 1955d). Because of the lack of space and the corrosive atmosphere in the plant, there were no instrument monitors (i.e., instantaneous, continuous, or integrating monitors) in any area except for one provided by AEC on an experimental basis (MCW 1950e). Hence film badges were placed at selected locations in the process areas to serve as integrating area monitors (see Section 5.4.3.6 below).

5.4.2 Neutrons

No neutron exposure measurements are available. However, Dupree-Ellis et al. (2000) deemed neutron exposures at Mallinckrodt to be minimal. This conclusion appears to be correct based on what is known about the source material and its handling. Nevertheless, neutron production by means of the alpha-neutron reaction neutron and production by the early RaBe source used in the Shotgun Laboratory (see Section 5.2.4) were analyzed in the preparation of this technical basis document.

In the analysis of neutron production by the alpha-neutron reaction, the forms of uranium and thorium that would produce neutrons at the highest rates were identified as UF_4 and ThF_4, the latter being an intermediate product in the processing of the thorium-containing waste cake to a thorium nitrate solution (see Sections 4.5 and 5.2.3). The uranium oxide forms, soda salt ($Na_2U_2O_7$), and thorium nitrate were also considered, as indicated in Table 34. Note that as long as there is a reasonable amount of target material (fluorine, oxygen, or sodium) available and intimately mixed with the source material (uranium or thorium or their alpha-emitting daughters), neutron production essentially depends on the amount of the source material.

Regarding the thorium, the total amount of cake processed to obtain the thorium nitrate solution was 350 tons, or 700,000 lbs (Table 4). It is not known how much was processed at a time. However, it is conservative to assume that the processing was done in the same type of large batch-processing tank system used for processing ore (this maximizes the quantity at one time). This is also appropriate because the cake was digested and subject to extraction in the same manner as the ore and soda salt feed were. From Table 4, at one time a typical monthly processing would include about 80,000 lbs of Eldorado black oxide, 120,000 lbs of Vitro black oxide, and 60 tons of Vitro soda salt, for a total of 260,000 lbs per month or about 8700 lbs/day. Also from Table 4, K-65 ore was processed at another period at the rate of 8000-12,000 lbs/day. Thus it is appropriate to assume that a reasonable high average processing volume of feed per day was 10,000 lbs. At this rate, it would take 70 working days to process the cake.

It can be assumed, from the way the ore and other feeds were processed, that one batch was moved from the digest tank through etherization, etc., without being mixed with another batch, at least until after the ether extraction. Thus it can be concluded that no more than 1/70th of the total Th amount was present as ThF_4 at any time. This is conservative because it assumes that all of the amount processed in one day was together in one tank or container at a time. It is also conservative because the processing is known to have taken up to 18 months, not just 70 working days, so the batches are likely to have been much smaller and the amount of ThF_4 present at any time in a container would also have been much smaller. This is particularly likely since the ThF_4 was said to have been moved to the Hot Lab (in Plant 6) in liquid form for final processing to thorium nitrate (AEC 1955c) and thus the individual transport container quantity was limited by the amount that could be transported as a liquid. Finally, it is conservative because any self-shielding or container shielding has been ignored.

Regarding uranium forms, as shown in Table 4, a fiber or steel drum container of UO_3, UO_2, or soda salt would weigh 75 lbs and a steel drum container of UO_2O_8 (ore feed) would weigh 100 lb; most of this weight would be uranium, so it may be reasonably and conservatively assumed that the entire weight is uranium. While a larger volume could be found in, e.g., a digest tank, the liquid and the thick tank wall would provide a great deal of shielding. A larger volume could be found in an array of containers, but a great deal of self-shielding would be involved and a person would likely not spend a great deal of time near an array. Thus it is likely that that dose rate from a single container (being temporarily stored, loaded, or transported) will be the typical dose source.

The following conservative assumptions were made regarding the conversion from neutron production rate in the Th container to an annual dose at the receptor point.

1. Point source geometry used to produce a nominal ambient dose rate, with the distance to the receptor point being taken from the center
2. Irradiation geometry of 75% AP and 25% ROT (from Table 18 of this technical basis document for “ionium plant operator”)
3. A U-238/U-235 breakdown of 99.3% to 0.72% by weight (for the natural uranium forms); a Th-230/Th-232 breakdown of 11.6% to 88.4% (for the thorium forms)
4. Neutron energy of 1.5 MeV (1.4-1.6 MeV is the energy range of the neutrons produce by the thorium isotopes and 1.0 - 2.0 MeV is the energy range of neutrons produced by the other isotopes; also, the neutron flux-to-dose conversion factor varies slowly in this range)
5. Neutron flux-to-dose conversion factor for 1.5 Mev of 1.3×10^{-4} rem/hour per neutron/cm^2-sec
6. 1 hour per day spent by the receptor at 1 foot from the container and 3 hours per day were spent at 3 feet, every working day for a year
7. For the purposes of considering the effect of including daughter contributions, for full equilibrium of the daughters of natural uranium or the thorium cake mix down to radon (which would not be in chemical union with the target and would likely be largely vented), the neutron emission rate due to the parent isotope would be multiplied by the following factor: 8.1 for U-238, 13.5 for U-235, 2.2 for Th-230, and 20.9 for Th-232. However, this was applied only to demonstrate the maxima, since the daughters were not likely to have built up significantly from the parent for the uranium forms and were separated before the production of the ThF_4 in the thorium case

The results are shown in Table 34. Conversion was made from ambient dose equivalent (H*(10)) to organ dose equivalent using the factors given in Appendix B of NIOSH (2000a).

In the analysis of the RaBe source production, the source was taken to contain 100 mg of Ra-226 (Mason 1977). There is no indication as to how long the source was used, so the period of use may be taken to have started in about April 1942 and, as indicated in Section 5.2.4, to have ended in September 1944. According to Shleien (1992), Tables 7.5 and 8.4.1, a RaBe source emits neutrons of average energy 4.0 MeV at a rate of up to 1.3×10^7 neutrons per sec per curie and the specific activity of Ra-226 is 0.0366 TBq per gram, or 0.989 curies per gram. This gives a source strength of 1.29×10^6 neutrons per sec for the 100-mg source, resulting in whole-body dose rates of 16 mrem/hr at one foot and 1.77 mrem/hr at three feet.

Assuming that a laboratory worker spent one hour per day at one foot from the source and one hour per day at three feet from the unshielded source, the worker would receive a dose of 10.7 rem over the 29 months, or 4.44 rem per year. Note that these estimates are conservative since by the various descriptions of laboratory work, most of the time in the laboratory would have been spent in preparing the samples (e.g., by grinding or chemical additions) and the source would have been completely shielded when not in use and likely partially or completely shielded when in use. These results are shown in Table 34, along with the organ doses. Conversion was made from ambient dose equivalent (H*(10)) to organ dose equivalent using the factors given in Appendix B of NIOSH (2000a).

5.4.3 Information and Available Data Regarding Film Badges and Extremity Dosimeters

5.4.3.1 Information Regarding the Type and Composition of Film Badges Used at Mallinckrodt

The same film badge was in use throughout Mallinckrodt uranium operations for AEC (MCW 1961a), i.e., from some time in 1945 through the end of operations in 1958. This was a two-element type of dosimeter. It is not clear whether the same type of film badge was used during the decontamination period (see Section 8.0), but it would appear so.

Mallinckrodt (MCW 1956i) described the badge as follows. The badge frame was made by the A. M. Samples Machine Company of Knoxville. The front was shaped so that a 1-mm thick cadmium shield could be inserted to cover the upper two-thirds of the film, with a similar 1-mm cadmium shield covering the back of the film. A large rectangular window covered the badge front to allow the identification number (the “health number”) perforation in the cadmium shield to be unblocked. A small rectangular window in the lower part of the badge front below the cadmium shield was covered only by the identification picture sandwiched between plastic sheets; a similar window in the badge back, below the cadmium shield, was covered only by a sheet of clear celluloid. The film was a DuPont Type 552 film packet containing two dental-size films wrapped together. The more sensitive of the two films was Type 502 and the less sensitive was Type 510. Mallinckrodt (MCW 1956i) remarked that the beta and gamma radiation generally encountered at the Mallinckrodt plants had less effect on photographic film than the same dose of X-rays that were often monitored with such film badges.

Mallinckrodt (1955i) also described the badge as having a metal frame made by the A. M. Samples Company and as using DuPont Type 550 x-ray film provided by the Dick X-Ray Company. The identification picture was said to be laminated between two pieces of thermo-coated acetate, each 0.005 inches thick, with the total thickness not exceeding 0.02 inch. MCW (1950m) stated that the Mallinckrodt badge had 3.3 mm between the face of the badge and the first surface of the film; the average absorber density between the badge face and the film was given as 81 mg/cm^2.

In an unpublished report the badge was described as follows:

> *The A.M. Samples stainless steel badge holder with open-window and cadmium filters permitting beta and gamma differentiation and measurement. DuPont Type 552 dosimeter film was used in the badge. The film was processed by techniques calibrated and standardized with film exposed to standard gamma and beta radiation sources. Gamma standards were obtained by exposing film to a platinum encapsulated radium needle. Beta standards were obtained using an aged, natural uranium block as a source.*

According to MCW (1956f), before the badge was issued, the week number was exposed on the edges of the film with 70 kV, 10 ma X-rays for 0.33 second. Control badges were exposed to a 10-mg radium source in a platinum needle, which was the primary gamma standard and whose intensity was taken to be 8.4 mR/hr at one meter. An earlier standard, No. R-515, sent to Mallinckrodt in 1948 by an AEC consultant on behalf of AEC, had 9.98 mg of radium, with a platinum-iridium wall thickness of 0.5 mm, producing 5.4 mR/hr at one meter from the center of the capsule (AEC 1948l).

The primary beta standards were aged metallic uranium slabs about 1/16" thick. Eight-month-old uranium was regarded as "aged" enough to have the Th-234 and the Pa-234 in secular equilibrium with the U-238; the beta contribution of the Th-230 and its daughters was considered to be negligible. The beta intensity at the surface was taken to be 239 mrep/hr (see also Section 5.4.3.2 below). Other details of the calibration, such as exposure times and distances, do not appear to be available.

5.4.3.2 Film Badge Technical and Processing Information

The University of Rochester processed the Mallinckrodt film badges from at least 1 July 1947 to 1 June 1948 (Rochester 1950, MCW 1950k); Rochester presumably also processed the badges before that under contract to AEC, as suggested by MCW (MCW 1950k). Mallinckrodt processed its own film badges from 1 June 1948 to 1 January 1950 (Rochester 1950, MCW 1948g, MCW 1950k). As is indicated by the Mallinckrodt health group monthly reports (e.g., MCW 1951b) and the 1955 Mallinckrodt health program description (MCW 1955d), Mallinckrodt appears to have continued to process their own badges even after this period. As the health group monthly reports (AEC 1950m; MCW 1951f) indicate, however, Mallinckrodt often had a significant backlog in reading the film badges; Mallinckrodt (MCW 1951e) observed that doses were frequently reported a week or more after the dose was incurred and thus expedited processing was requested for badges from such workers as the burnout and Feinc operators. But it was not unknown for this problem to occur when Rochester was processing the badges: results typically took two weeks to be sent back to Mallinckrodt and on about 19 August 1946 results were almost two months later in coming back to St. Louis from Rochester because of Rochester personnel's being on duty at the Pacific Crossroads tests (MED 1946c). This was one of the reasons that Mallinckrodt took over the processing of its own film badges.

No procedures and little other film badge specification data have been found to date. See Section 5.4.3.1 for information about the film badges themselves. There is also not much information about how the film badges were processed by either Rochester or Mallinckrodt. However, MCW (1950r) stated that an AEC report (NYOO-57) was an attempt to measure the surface dose rate from uranium (presumably metal) received through the stratum corneum of the fingers, at 7 mg/cm^2. According to MCW (1950r), this report gave a dose rate of 239 mrep/hr through a depth of 7 mg/cm^2 and 178 mrep/hr through 44 mg/cm^2. Mallinckrodt had been using a value of 265 mrep/hr in their film badge measurements, but now agreed to use the NYOO report value of 239 mrep /hr. Mallinckrodt did so because the report results were based on a filter closely approximating the dead skin layer; thus this basis was more similar to Mallinckrodt's operating conditions than their old value's basis. However, as MCW (1950r) pointed out, AEC's tolerance level was based on the "field strength" of the radiation, not on the amount of radiation at any particular depth of any particular material; it was thought that any other reference level would be impossible because of the energy distribution of the betas.

There are some documents indicating problems or variations from the norm in badge readings. MED (1945l) stated that recently read films worn from 26 November through 4 December 1945 showed abnormal density combinations and that it was inconceivable that all of the density was due to radiation exposure. It was noted that a Mallinckrodt manager had admitted that the films might have been subjected to high temperatures. It was also noted that the problem could not have occurred in

Rochester (where the films were being read at the time): the Mallinckrodt films of two weeks earlier had also shown high exposures and since that time, Rochester had been very careful in handling films from Mallinckrodt.

Mallinckrodt (MCW 1948d) reported that there was considerable fluctuation in readings on badges worn in the digest and Feinc filter areas; the explanation was given that any interruption in the process that caused a holdup of material allowed the material to “age” (i.e., daughter products to build up) and thus to exhibit more gamma activity. Breakdowns of equipment, which tended to result in higher exposures to operators doing the repairs, were another reason given for the variations.

There was a series of meetings and correspondence between Mallinckrodt and AEC regarding whether certain readings were due to beta or soft gammas and whether the AEC and Mallinckrodt methods of correction for shield absorption (in the badge) were consistent (AEC 1950b; AEC 1950f; MCW 1950k; AEC 1950g; AEC 1950i; MCW 1950x). This issue involved the subtraction of the film density value under the beta “shield” from the value under the window. Section 7.1 of this technical basis document states the assumption to be made about this subtraction; however, the references cited may be consulted in case of any suspect beta readings corresponding to the 1949-1950 period. In connection with this issue, Mallinckrodt undertook to do an experiment to determine whether the beta reading from the open window of the film badge was due to primary (directly emitted) beta radiation from surface contamination in an area or to secondary beta (i.e., produced by scatter of the hard gammas) (MCW 1950x). A radium source was used to expose four sets of film: one pair was exposed with a 4800-gauss magnet in place and the other pair with a set of phantom magnet yokes in place. Additionally, one of each pair had a 7/16” Lucite shield in place to eliminate direct betas from the radium source. In this way, the betas were drawn off by the magnet (including any secondary betas) or were taken out by the shield, allowing for a 98% reduction by the magnet in beta radiation arriving at the appropriate films and thus the assumption of a nearly pure gamma component being measured by those films. Mallinckrodt determined that at most a reduction of 19% in the open window could be attained by aggressive cleaning of surface contamination in an area, i.e., that most beta measured by an area film badge was not direct beta. MCW (MCW 1950x) observed, however, that film badges actually worn by workers show greater window densities for equivalent shield densities than area film badges. This was attributed to workers’ badges’ more frequently actually “seeing” a primary beta source, e.g., during manhole access. Thus during some types of work, workers actually received beta radiation that they would not receive merely by standing in the area.

Mallinckrodt (MCW 1948k) stated that while badge readings were supposed to be accurate within 10% (apparently an AEC requirement), Mallinckrodt’s standards were within 5% as checked by “New York” (presumably Rochester or NYOO). Mallinckrodt regarded the method of reporting by Rochester prior to 1 July 1947 as “inconclusive” (MCW 1950k), presumably because of the rounding off to the nearest percent of the tolerance level and possibly also because of uncertainty in the interpretation of what the tolerance was and of what zero dose might actually translate to.

Prior to readout at Mallinckrodt, badges were disassembled in a laboratory hood (MCW 1956f). A Weston photographic analyzer, Model 877, with two stabilizers, was used in reading out the films in 1949; in 1956, a Welch Densichron densitometer was being used (MCW 1956f). Due to the assumed near proportionality of film density and radiation exposure of any given type, Mallinckrodt assumed that the following relationship was valid: Net window density from beta exposure alone = Actual net window density - Net window density from gamma exposure alone (MCW 1956f).

5.4.3.3 Film Badge Monitoring Periods and Wearing Practices

Workers were not individually monitored for external dose prior to June 1945. Film badge records are available for the weeks ending 10 June, 24 June, and 1 July 1945 (by memorandum report, MED 1945o) and for 29 July, 5 August, 12 August, and 26 August of 1945; then there is a gap until the week ending 9 December 1945. Total (accumulated) doses for the period from 9 December 1945 to 25 March 1946 were reported by individual in a memorandum from the University of Rochester School of Medicine and Dentistry (Rochester 1950). Besides the 1945 records, records appear to be available, with some gaps, for most weeks from 1946 through 1948; 1950 through 1951; 1952; and 1954 into 1958. Furthermore, film badge records for the postoperations decontamination period (see Section 8.0) may not be available either. It is not clear why records are missing after 1945, since clearly film badges were used continuously from 1945 on. Hence it is possible that these records may be found at some later time.

ORAU (1980b) stated that all Mallinckrodt employees who were cleared to have access to production areas wore film badges at all times, with the purpose being to provide maximum assurance that all exposures were registered. Contemporary references indicate that film badges were issued as a combination security-exposure badge to all employees, except for “office females” who presumably never entered process areas (MCW 1955d, MCW 1955c, MCW 1956i). MED (1944p) describes the film badge as being numbered and being combined with a photo ID. AEC (AEC 1948c) states that for Plant 6, the badge was a “photo [ID] film badge” for cleared employees; a film badge with a red insert and the employee clock number for employees awaiting Q clearance; plastic film badges for technicians, engineers, and management, with photo film badges in a separate rack at guard shack or changehouse for when they entered the plant from the clean buildings; plastic photo badges for office employees who did not go into plant areas; film badges with a red insert bearing a large letter V and a number, for subcontractors going into Limited and/or Contaminated areas (such badges to be issued by the guards); and a plastic badge (with no film) with bearing a large letter V and a number, for subcontractors going into Controlled and/or uncontaminated areas and for business visitors to the offices (also to be issued by the guards); and ditto last for business visitors to the offices. MCW (MCW 1955c) also states that all visitors and outside contractors entering process areas were issued badges. It is not clear how early the practice of issuing film badges to visitors and subcontractors entering process areas began.

AEC (AEC 1948c) states that film badges were to be used at Plant 4 for a trial period, suggesting that that Plant 4 employees were not badged prior to early 1948; it was recommended that employees be issued film badges but that subcontractors and business visitors be issued a plastic badge (no film) with a green insert and number.

In 1955, toward the end of operations, it was realized that since an “expansion” group had moved to new quarters in the “Rock Island Building” (presumably in St. Louis), film badges had not been worn by this group during normal activities, thus leaving a gap in exposure data for these individuals (MCW 1955a). A directive was therefore issued for badges to be worn by this group, but of course the missed period was not covered in the records. Similarly, the Mallinckrodt Uranium Division employees who transferred to the Weldon Spring site startup group in 1956 were not being monitored any longer for radiation exposure with film badges; those who might still have business at the Mallinckrodt St. Louis site were to have their badges kept for them in St. Louis (MCW 1956b). Thus these individuals were likely to have intermittent badge readings associated with the St. Louis site.

Since there were no potential sources of acute external exposure, the aim was to keep chronic exposures below tolerance levels (MCW 1950e; MCW 1955d). All exposures over 50% tolerance were reported to supervisors (MCW 1955d).

From 22 April 1946 through the end of MED/AEC work in 1957 or 1958, film badges were processed on a weekly basis as part of a routine dose monitoring program. Badges for Plant 4 began to be changed only every two weeks in early September 1954 (MCW 1954d); all badges began to be changed every two weeks as of 30 January 1955, because of a shortage of health personnel to read the badges and the comparatively low doses then being recorded on the badges (MCW 1955b, MCW 1956i). Mallinckrodt (MCW 1955d; MCW 1956e) stated that (all) badges were changed every two weeks or more often when desired for information or because of expected higher dose. Film badge results were summarized quarterly and annually (MCW 1955d).

The guards were directed that if there was no replacement film badge available for a given used badge when the badges were exchanged by the guards at midnight on Sunday, they were to remove the used one anyway (MCW 1948d). The guards were to notify the safety department of the situation on Monday morning and the individual was to wear his “current” badge for the first half of the shift, until a replacement badge was provided by the safety department. It is not clear from the reference if the used badge was for the week that ended on Sunday or for the week before that, although if it was the latter, that would explain why the individual still had a current badge that he could wear.

Workers were directed to wear their badges when they went to the Airport (SLAPS) or “the Range” (MCW 1949q); the Range appeared to be a firing range (near or at the storage area) where the guards would practice shooting and where some materials were stored (e.g., AEC 1949e reports that some UO_2 was drawn from this storage area for use in production). Plant 6 workers were also told to wear their Plant 6 badges when they visited Plant 4, and vice versa for Plant 4 workers, and not to wear a visitors’ badge (MCW 1948k).

As shown in the comparison table in Section 5.4.1, data from AEC (AEC 1951b) shows that a weekly average of 358 badges were issued to Plants 4 and 6 workers in November 1948 through January 1949 and a weekly average of 403 badges were issues in January to June 1950. From the series of Mallinckrodt health group reports (an example of one is MCW 1951c), some 2000-3600 badges were read per month in the early 1950’s, corresponding to about 460-830 badges issued weekly; these included visitors’ badges and probably also some area monitoring badges and some experimental, duplicate, and supplementary (double-badging) badges.

5.4.3.4 Film Badge Record Types, Arrangement, and Availability

Mallinckrodt dose records were of three types: complete records of weekly film badge results, listings of total doses by employee over a specified time period such as the “Mallinckrodt_1946” file (MCW undated), and plant dose summaries. Records found to date show weekly badge processing cycles, with the following exceptions: records that show total dose by specified time period, records from the very later period of operations, and records from the decontamination and decommissioning period. Assignment of individual annual doses was based on deep-dose exposure (Dupree-Ellis et al. 2000).

For most of the period of operations, the complete records are weekly lists of employee names with beta and gamma doses. For the gamma doses some results are shown as “50*” and the asterisk refers to a footnote that reads “indicates less than” (MCW Undated). Values of 60 and 80 are sometimes asterisked in the beta column. Occasional values of “0” are found in the gamma column as well. Some records list employee names with total doses over specified time periods, with a start date and end date.

Listings of total doses by employees over a specified time period other than a week are found in the dose reconstruction project file “Mallinckrodt_1946” (MCW undated). The earliest results are of this form also and are recorded by total dose and number of weeks worked in the dose reconstruction project file "Mallinckrodt Radiation Summary APR 46 to MAR 48." This document also supplies other important information for external dose reconstruction. Many of the 1948-1958 records have annual totals written on them; ORAU (1980a) stated that a sampling of records showed that the summing for the 1950-1958 records was sufficiently accurate that the totals could be accepted (i.e., for epidemiological study purposes), but not the summing for 1948-1949.

Dose summaries (in memorandum form) generally listed doses by plant, number and percentages of badges in dose ranges from 0-50 mrep/week, 51-100 mrep/week, 101-150 mrep/week, etc. based on the total beta and gamma for an employee. Doses are not listed for employees having less than 150 mrep in a week; for the dose categories above 150 mrep/week, individual names are listed with gamma and beta dose results. All employees not working at Plants 4, 6E, or 7 were included in the Plant 6 report (MCW 1956a).

Up to 1950, Mallinckrodt recycled clock numbers, i.e., reused them as employees left and new employees were hired (MCW 1950a). However, this created difficulties with keeping dose records because as Mallinckrodt recognized, dose records needed to have identification numbers that would "remain unique in perpetuity" for tracking purposes. The film badge numbering system was therefore changed as of January 1950 and clock numbers were no longer used on the badges. However, in tabulating the doses (e.g., in memorandum form), the clock numbers were used. The potential for badge confusion due to non-unique numbers prior to 1950 must be recognized.

Some codes used in the records to identify the type of job or work area (e.g., M for maintenance) are found in MCW (MCW 1956g) and similar memorandum reports or in Table 5 of this technical basis document. Some codes used in the records to denote readout problems or other items of interest (e.g., GL for Gone Through the Laundry) are found in ORAU (1980a). These are also been shown in Table 5.

Complete records and/or dose summaries may not have been located for all periods of MED/AEC operation as of this writing. This results in gaps for dose monitoring data when no information is available for workers in a given plant, or in some cases, for any Mallinckrodt worker. Dose summaries, when no complete records of weekly film badge results are also available, give no individual data for personnel receiving less than 150 mrep in a week. Many gaps in data are the result of accidents or damage in the workplace or during badge processing. These incidents are usually documented in the record.

One record of extremity dosimeter results (as ring badge readings) has been found – see Section 5.4.3.5 below.

5.4.3.5 Extremity Dosimeters

Because of the high extremity doses in cleaning the high-beta deposits out of the recasting furnaces, in 1949, film rings began to be used "by selected groups" in the metal plant (AEC 1949b). The results were reported for 25 April through 30 May 1949 in AEC (1949d). Figures were said to be "the Oak Ridge results" based on a radium gamma calibration, so presumably these films were read at one of AEC's Oak Ridge sites. The beta exposure was taken to be a factor of 1.2 times the difference between the open window and the shielded reading. Such data from other sites had been found to be variable and inconclusive, probably, AEC thought, because of improper wearing of the rings and the regular badges worn at the same time; AEC had determined that the ring response depended very much on the direction and angle of the beta sources in the process areas. Individual records of the film ring readings do not appear to be available among the regular film badge records, but AEC (1949d) has a long table giving the results by worker name and identification number.

5.4.3.6 Film Badges Used as Area Dosimeters

Film badges were placed in selected locations in the process areas to serve as integrating area monitors (i.e., to register integrated dose in an area regardless of actual occupancy) (MCW 1955d; MCW 1955h). Mallinckrodt (MCW 1955h) opined that such film badges were sometimes hung in out-of-the-way places and not in the area of highest dose rate during occupancy; however, this did not appear to be demonstrated by the comparison between the dose rates measured by the film badges and those measured by a gamma survey meter (MCW 1955h). There is no information as to how often these area film badges were collected and read.

5.4.4 Information and Available Data Regarding Occupational X-Ray Examinations

MED (1944d) stated that the medical program recommended by MED was being followed by Mallinckrodt. This included a routine chest x-ray prior to employment and annually thereafter and a pelvic x-ray for those employees handling fluorides. MED (1944d) thought that the pelvic x-ray could be dispensed with and apparently this was subsequently done. Later documents also indicate that Mallinckrodt uranium processing workers were given a pre-employment physical that included a chest x-ray (MCW 1955d); Mason 1958a); they were also given an annual physical that included a chest x-ray (MCW 1955d; Mason 1958a). That these x-rays were actually given is indicated in the series of Mallinckrodt Health Office and other reports (e.g., MCW 1951b; MCW 1954e; MCW 1955d). No information is available as to how much dose was received during these examinations or if all workers received annual x-rays. Therefore, to be claimant favorable, it is assumed that all personnel who worked in the plants covered by this technical basis received an annual diagnostic chest x-ray. It should also be assumed that personnel working with hydrogen fluoride or in the UO_2-to-UF_4 conversion process in 1942-1944 received an annual pelvic x-ray.

5.4.5 Skin Contaminations and Other Radiological Incidents

No records appear to be available regarding skin contaminations. It seems likely that due to the relatively low radioactivity level of most of the uranium forms and the pervasiveness of the uranium-bearing dust, skin contaminations would have been regarded by safety officials at the time as not significant and thus would not have been recorded. See Section 5.3.5 regarding surface contamination, including clothing and smoker hand data.

No information has been found as to any incidents that may have resulted in significant overexposures to radiation or intakes of radioactivity. However, the following incidents were documented: a 1942 or 1943 explosion in the denitration process area in which agitators began to bind until the motors driving them finally tore loose from the concrete floor (Fleishman-Hilliard 1967); a 1943 ether fire in Buildings 51 and 52 involving a dryer blown apart by an explosion of ether vapors due to burning ether (MED 1943a); a 3 July 1943 fire in a rubbish truck containing “lime” sweepings and other floor sweepings, possibly including metal slag (MED undated b); a 4 May 1946 explosion at Plant 6 involving an explosion due to seepage of ether into the nitric acid tanks due to a malfunction of a check valve; a 19 August 1946 potential significant exposure to workers, probably in the removal of K-65 in the storage area, following which working space was rearranged and the pileup of residue material at the location was reduced (MED 1946c); and a 1947 or 1948 explosion causing the rupture of a nitric acid holding tank due to mechanical failure of a check valve (MED 1948; Fleishman-Hilliard 1967).

Also, there were indications of frequently occurring incidents, such as the occasional spontaneous catching on fire of uranium metal derbies at the derby cleaning station in Plant 6E, which would have put particulates into the air; this problem was spoken of as being brought under control (AEC 1952b).

This may have been a longstanding problem, however, since the reference in early 1944 to the loss of material in "burnout" in the reduction step apparently was to this spontaneous ignition of derbies (MED 1944o). Another such recurring incident was the plugging of floor drains, with the consequent formation of puddles of contaminated liquids on the floor at Plant 6 (AEC 1950m).

One incident of concern to the health department was the hand loading of raw pitchblende ore on 21 February 1950, without the knowledge of the health department (MCW 1950d). Apparently there was a malfunction of part of the ore drum conveyor. Operational pressures motivated the production supervisor to have the star valve at the bottom of the ground ore hopper feeding into the secondary elevator disconnected. A three-man crew then picked up open drums off the conveyor using a chain hoist and tipped their contents onto the floor. One worker then shoveled the ore across the floor to another worker and he to the third worker, who shoveled it into the chute to the secondary elevator. One crew worked one half-hour session and a second crew worked another half-hour session, together handling 12 drums of ore. The crews wore respirators and the downdraft ventilation into the chute was working, but the workers were not otherwise protected from direct radiation or radon, and a crew of maintenance people working around the rod mill nearby were not wearing respirators. It was noted that the gamma exposure at the surface of such an ore drum was 80-100 mR/hr (MCW 1950d); in addition, Table 24 of this technical basis document shows some radon levels for open pitchblende ore drums. However, it was also noted by MCW (MCW 1950d) that health department clearance would have been granted for such an operation if the workers had worn oxygen or air line masks, if stay times were limited to 1 hour per week per worker, and if workers were taken from low-dose areas such as the Ether House or the furnace room.

5.5 OTHER DATA OF DOSIMETRIC INTEREST

5.5.1 <u>Number of Workers</u>

The initial April-July 1942 uranium pilot plant effort included 24 people working as a single project group under a project manager (Fleishman-Hilliard 1967). In 1944, there were 55 guards; 330 workers (including guards) with a clearance for MED work, and 1500 workers on the entire site (presumably including non-MED workers) (MED 1944p). Regarding the total number of workers with potential for exposure, Fleishman-Hilliard (1967) and Mallinckrodt (1994) listed the total number of workers as 250, the former stating that this was in 1948; AEC (AEC 1948e) listed the total number as 250 at Plant 6, but 400 if Plant 4 was included; AEC (1949b) listed the number of workers at Plant 6 as 272 and the number at Plant 4 as 94; Mallinckrodt (MCW 1950w) gave the total number as 487 and added that 275-300 individuals who worked on the project for more than three months had terminated as of 1 October 1949. Mason (1958a) stated that as of the beginning of 1948, more than 100 of the original employees working during the period 1943-1946 were still on the payroll. AEC (1951b) stated that in early 1951, Plant 4 employed 100 people.

AEC dust study reports in the 1950's gave the number of each classification of workers and the number on each shift (e.g., AEC 1954b); some of these AEC reports even listed the names of process and supervisory workers and their job classifications in an appendix. For Plant 6 alone, Mallinckrodt (MCW 1956a) gave the number of manufacturing personnel as 100 and the number of nonmanufacturing personnel as 550. As noted above in Section 5.4.3.3, over 2000 film badges a month were processed in the 1950's.

5.5.2 <u>Number of Hours Worked per Week</u>

From AEC dust study reports (e.g., AEC 1954b), the following information regarding time spent is provided as follows:

Length of work day, including breaks and locker room time	480-520 minutes (8-8.6 hours)
Lunch break	30 minutes
Smoking breaks	30-40 minutes
Clean locker room	20 minutes
Regulated locker room	15 minutes

The longer work day applied to operators and craftsmen, who presumably had to leave their work areas to smoke. There was a 10-15 minute variation in the work day among plants as well. The total smoking break time was 30 minutes for Plants 6E and 7, but 40 minutes for Plants 4 and 6 through about 1955; after that it was 30 minutes for all plants.

AEC-NYOO took the number of work hours per week to be 48 (or six 8-hour days) in calculating some of their early time-weighted average airborne concentrations (AEC 1949b). Also, AEC (1951b) stated that Plants 6E and 7 operated on a six-day, three-shift schedule. However, this does not imply that individual Mallinckrodt workers worked six days a week. MCW (MCW 1949g) [1949] states that the work week is 8 hours per day, 5 days per week. Lippmann (1958) used 40 hours in reporting data regarding Harshaw workers and Harshaw (1946) described its rotation schedule used for round-the-clock operation, which involved a 40-hour week for most individual workers and a 42-hour week for some workers. Besides these references, AEC, in its dust study reports, used the actual time worked, as given above, implying a five-day week. It can thus be assumed that Mallinckrodt workers typically worked for a full 8 hours a day, 5 days a week, or 40 hours per week. When using daily weighted average dust sampling data (e.g., in Tables 12-16 and 19-22), it is important to understand that break, lunch, and locker time was factored into the weighted averages reported by AEC and Mallinckrodt in their air dust studies.

Fleishman-Hilliard (1967) states that once the Plant 2 operations started (ether extraction), they were carried out 24 hours per day. It is not clear what other early processes ran 24 hours per day, but AEC (1951b) implies that Plants 4 and 6 did not. However, individual dust studies cite three shifts for many process workers positions, indicating that operations were in fact carried out 24 hours per day, although perhaps not on the weekends. Guardhouses were manned around the clock, with three shifts per day (MED 1944p).

5.5.3 Job Types, Job Histories, Work Areas, and Work, Access, and Clothing Practices

After about 1950, film badge reports included a short note or keyword about the job or work done or the work area occupied by the individual during the week. After about 1948, many urinalysis sheets also listed such notes or keywords. Various AEC dust reports also list job titles and functional work types. Job titles and types discovered to date are given in Table 18. Note that in the absence of further information, it is not possible at present to distinguish in these records and documents between ordinary or process decontamination and the decontamination that may have been part of preliminary decontamination prior to the decontamination and decommissioning of buildings and plants; however, it is known that there was a decontamination group that did ordinary or process decontamination and special decontaminations as necessary (e.g., of a lab area), so when "decontamination" appears in a record it is this type of work that should be assumed.

As previously stated, to aid in classifying workers whose job titles do not appear in Table 18 and whose work descriptions do not make it clear which job title is appropriate for use, Table 5, the keyword table, which includes information from these notes and from operational information in other references (particularly MED 1946a; AEC 1949b; and AEC 1967). The two tables should be used to help determine the principal occupational activity for an individual with missing or conflicting monitoring data.

According to ORAU (ORAU 1983b), prior to 1 October 1949, there was no "concise" or in-one-place employment history for Mallinckrodt workers in the Uranium Division (i.e., doing MED/AEC work). Then because of the high doses, it was apparently thought that workers who had received more than 90% of their (unspecified) "lifetime" exposure to dust should be removed from uranium work. Consequently, a work history was reconstructed by Mallinckrodt for the period of 1942 to 1 October 1949. Mallinckrodt tried to be as accurate as possible because of the possibility of having to remove workers and apparently the workers examined the reconstructed records, because "the work history had to withstand the scrutiny of each worker" (ORAU 1983b). Four volumes of records were produced: one master list of workers, one volume with histories of those terminating prior to 1 October 1949, one for current Plant 4 workers, and one for current Plant 6 workers.

In the Mallinckrodt Uranium Division, workers in maintenance (shop personnel, skilled craftsmen, and maintenance mechanics) and in plant services (guards, porters, nurses, laundry workers, power plant personnel, hygiene and safety personnel, etc.) were not usually assigned to a particular process, but served the entire division (Hickey and Dupree 1984). An exception may be the "area mechanic" mentioned in some AEC dust sampling reports, but even then it may be that the mechanic was assigned to cover multiple areas and his exposure may not have been characteristic of only one area. However, Hickey and Dupree (1984) note that the uranium monitoring records they examined showed a reasonably uniform exposure for both the field and the shop maintenance personnel and skilled craftsmen. Hence particular area assignments may not be critically important in, e.g., formulating surrogate sets.

ORAU (1977) stated that the highly exposed workers were transferred out of uranium operations in 1950; these were presumably some or all of the 52 workers found to have 100% or more of the lifetime tolerance dose as of October 1949 (AEC 1950a). Mallinckrodt (MCW 1951b) and AEC (AEC 1950e) also stated that some 34 workers were transferred in the summer of 1950: the former states that this was from Plant 6 to Plant 1 "for health reasons" and the latter states that this was from Plants 4 and 6 on the basis of their estimated accumulated exposure to uranium dust. While Building 51 (Plant 2) was said by ORAU to have been closed about 1 January 1947 and by Mason (1977) to have been "sealed" in about 1947, when Plant 6 started up, and information in ORNL (1981) suggests that Plants 1 and 2 might have been in use for some support work until decontamination began in 1948; also, MCW (1949e) indicated that special badges would be issued to workers at Plant 1, although this may have been for intermittent work in Plants 4 and 6. Thus the Plant 1 work, if any, would presumably have been non-uranium work. Whatever the case, Mallinckrodt stated that the transfer had been considered for over a year before it occurred and that it would send a report to AEC (1950d); this report does not appear to be available. Possibly partly as a result of the study reported by AEC (1950a), Mallinckrodt used an employee rotation program from about 1950 on (Fleishman-Hilliard 1967; AEC 1950b; MCW 1955d), the point of which was to keep the weekly dose below the weekly tolerance level or, after about mid-1950, to keep the average weekly dose over a three-month period below the weekly tolerance level.

A 1944 MED security survey report (MED 1944p) gave the following information. In 1944 and presumably all other years as well, access into the MED areas was only through guarded entrances. Hourly rounds of the entire site were made during evening and early morning hours and all day Sundays and holidays; it took a guard 40 minutes to make a complete round of Plant 2. Non-Mallinckrodt truck drivers were allowed to come in, but not truck drivers' helpers unless needed to unload; all trucks were escorted while within the site. MED (1944l) stated that personnel in the "MED plant" were segregated as much as possible from the general (non-MED work) personnel, with the main point of contact being during the lunch hour.

From April 1949 on, the film/ID badges were color-coded to control access (MCW 1949h); for example, Plant 6 was divided into seven zones, each with its own color. "Controlled" areas were areas not expected to become contaminated, such as the offices, the power plant, the pumphouse, and the yard areas. "Limited" areas included the main processing building, the warehouse, the Scale House, the laboratories, and so forth. Mallinckrodt's 1955 formal description of its health program (MCW 1955d) gave the following information, indicating that the area control system was continued throughout the processing years although the definitions might have changed somewhat. Mallinckrodt maintained three levels of controlled areas. These were the regulated areas, which were described as the areas where radioactive materials were processed and handled; the grey areas, which were areas where any radioactive material and contamination was incidental to the function of the area, e.g., labs and production instruments departments; and the clear areas, which were areas where radioactive materials were not required and not permitted, e.g., offices and the cafeteria. Although zero contamination was not possible in the latter areas due to their proximity to the other areas, that was the stated goal of control efforts.

An internal Mallinckrodt memorandum circa April 1948 (AEC 1948c) stated that the Building 101 facilities had been "revised" (implicitly for better control of contamination) and access status in the building and yard areas was to be broken into two classifications. The first was the "clear area", consisting of all areas in which the presence of "plant products" could be completely eliminated or reduced to negligible amounts. The designated clear areas were the guard office, the locker room, the lunchroom, the Health Unit offices and laboratory, the AEC offices, the MCW offices, and the MCW laboratory offices. The second area was the "regulated area", consisting of all areas that could not be kept clear of plant products.

MED (1942) stated that employees were required to change work clothes daily, with the "uniforms" being provided by Mallinckrodt. AEC (AEC 1948c) states that in the clear areas, either street clothing or plant clothing with cover clothing was allowed, except in the lunchroom, where only street clothing was allowed; in the regulated areas, either plant clothing or street clothing with cover clothing was allowed. In 1955, the Mallinckrodt health program was said to include the following requirements regarding clothing (MCW 1955d). Work clothing "from the skin out" was provided for all persons assigned to regulated areas; regulated clothing could not be worn outside regulated areas except under cover clothing; and cover clothing was also provided for brief visits to regulated areas. Because wearing contaminated regulated clothing on public land was undesirable, vehicular transportation was required for workers traveling between regulated areas (even with the cover clothing) (MCW 1955d). Regulated area workers were issued two changes of clothing per day as of 1955 (MCW 1955d) and probably much earlier due to contamination control requirements involving showering (see below).

Visitors were provided with cover clothing (smock, rubbers, etc.) (MCW 1950e). However, as of at least 1950, outside contractor personnel wore their own personal work clothing, on the grounds that visitors' garb would be "too restrictive" and thus unsafe (MCW 1950e). Besides that, suitable change room facilities were not provided for outside contract personnel (MCW 1950e).

Process (manufacturing) workers were required to shower before changing into "clear" (clean) clothing; they typically took two showers a day in 1955 (MCW 1955d), but only one in the 1942-1944 time frame (MED 1942, MED undated a) and apparently only one in April 1950 (MCW 1950e). The procedure was as follows (MCW 1948c). After punching in, workers who would be changing into plant clothing would remove their street clothing in the locker room, don plant clothing in the change room (apparently the same clothing from the second half of the previous day's shift), and enter the work area. At lunchtime, workers would remove the plant clothing in the change room, shower, don street clothing, and go to the lunchroom. After lunch, they would put on a clean set of plant clothing in the changes room and go back to work. At the end of the shift, workers would remove the plant clothing in the change room, shower, and don their street clothing in the locker room. An extra 15 minutes was allowed for the midday showering and changing and clogs were provided for traffic between the locker room, change area, and showers (MCW 1948c). The one-way door allowing traffic to pass only from the locker room to the change area was to be supervised at shift change and lunch hours (MCW 1948c). No one was allowed to leave the plant in plant clothing (without cover clothing, presumably) (MCW 1948c). By February 1949, an AEC consultant, inspecting the change room for Plant 6 process workers, termed the operation of the change room "creditable", i.e., in line with AEC directions (AEC 1949k).

Plant 6 warehouse personnel were to follow the same general procedure in their change room, as were maintenance workers who used the process area locker room. The warehouse personnel leaving Plant 6 on trips to other regulated areas (such as Plant 4, SLAPS, or "the Range"), were allowed to make the trip in plant clothing, but in a "regulated" vehicle only; these vehicles could be jeeps, dumpsters, or large trucks) (MCW 1948c). Also, warehouse personnel leaving on trips to clear areas outside Plant 6 had to shower and change as if they were leaving work and they had to use a "clear" vehicle only (MCW 1948c).

Laboratory personnel who normally wore plant clothing were to follow the general procedure for process workers, except that they used their own locker room and change room; personnel going to the laboratory offices (a clear area) from another clear area could go directly in street clothing but had to use cover clothing to go to the laboratory; and personnel going from working in a laboratory to the lab offices had to put on cover clothing over their plant clothing (MCW 1948c).

The change requirements given above for process workers also applied to maintenance personnel who used the maintenance shop locker room: they were to use the same procedure but in their own locker room (MCW 1948c). Since the maintenance men in the shop might also work in the now non-AEC Plant 1 and presumably other clean areas at the general Mallinckrodt facilities, they had to change into special blue coveralls and either street shoes or clean safety shoes when visiting those areas; similarly, any Plant 1 maintenance men going to the AEC areas to work were to follow the change procedure as for the process workers (MCW 1948c). Instrument Shop personnel going to Plant 4, however, could go in plant clothing but in a regulated vehicle only (MCW 1948c).

Mallinckrodt (MCW 1948c) stated that no one was permitted to eat in plant clothing; that no one who had been working in plant clothing was allowed to eat without previously showering; and that there was to be no eating in regulated areas. Soft drink dispensers and washing facilities were provided in the change room, but hands were supposed to be washed before drinking (MCW 1948c). According to MED (1944p), smoking was not permitted in operational areas except in designated smoking areas or smoking rooms. Smoking was permitted in offices and labs, except where ether or other flammable substances were handled. In 1944, the penalty for smoking in other than permitted areas was loss of employment. Thus it is likely that nearly all smoking was done in designated spaces.

5.5.4 Miscellaneous Product Information

Some quantities and dimensions of potential radiological interest (e.g., for special external dose calculations) are as follows. See also Table 4 and Section 4 for other amounts.

Bomb	10” OD × 40” long	MED 1946a
Bomb liner (lime, etc.)	Depth: 1”	MED 1946a
Uranium billet	4.75” OD × 18” long	MED 1949b
Uranium billet	~ 6” OD	Mason 1977
Ore barrel	3’ high × 18” across	MED 1945a
U metal samples		
Glass tube	2” OD × ¾” long	MED 1945a
Cardboard packing box	5” sq, 1-2 lbs filled	MED 1945a
U eggs (samples from billets)	Packed eight to a box	MED 1945a
Billet packing box, wood	5” × 5” × 13”	MED 1945a

5.5.5 Missouri and St. Louis Area Background Levels

The table below gives some Missouri and St. Louis area background measurements, except that the Applied Nuclear Safety measurements were taken at Building Z of the Mallinckrodt site, used as a control. Note that these measurements were all taken many years after the end of operations and were made for the purpose of characterization for possible future remedial action.

		ORNL 1979		**ORNL 1981**		**Applied Nuclear Safety 1986**	**Bechtel 1987**
Measurement	**Unit**	**Range**	**Average**	**Range**	**Average**	**Average**	**Average**
Ac-227	pCi/g	Below detectable					
Ra-226	pCi/g	0.3 - 1.3	1.05 ± 0.3		1.18		
Th-232	pCi/g	0.3 - 1.3			1.15		
U-238	pCi/g	0.3 - 1.7			1.25		
Gamma background at points distant from site	uR/hr		6.0 ± 1.7	7 - 9	8		11
Beta-gamma (GM)	mrad/hr		0.02	.01 - .04	0.02		
Radon background concentration	pCi/L				1.0	0.33	0.3
Radon daughter background concentration	WL				0.01	0.002	

6.0 DETERMINING RADIOACTIVITY INTAKES DURING THE OPERATING YEARS

Urinalysis and other individual-specific bioassay data should be used to determine the individual's intake of radioactivity, when these data are available. However, some workers will have unmonitored periods or gaps. The intakes over the missing or gap periods will have to be determined either by using comparable (surrogate) worker data or by the use of dust exposure data."For uranium process residues, because the ability is limited to accurately determine the fraction of time a worker spent in any given part of the residue stream areas, the approach to assigning intakes from uranium progeny will be based on the highest dose delivered by either established source term. That is, NIOSH will seperately evaluate the dose from inhalation of either Radium or Thorium bearing residues. Intakes of Thorium bearing residues will be estimated from air monitoring data using the 95th percentile of the distribution of air measurements observed in plant 6. Intakes of Radium bearing residues will be estimated using the worker's individual radon in breath measurements when available or on the 95th percentile of the distribution of breath measurements when no samples for the individual have been located. Thus, the dose will be reconstructed for each source term and the value that provides the highest probably of causation will be used."

{NOTE: APPENDED TO THE END OF SC&A 'S 3RD SUPPLEMENT REVIEW OF THE MCW SITE PROFILE ARE WORK PRODUCT DOCUMENTS DEVELOPED IN RESPONSE TO THE ABRWH'S 6 TASKS CONCERNING "PRIORITY ISSUES FOR DEMONSTRATING FEASIBILITY OF DOSE RECONSTRUCTION FOR MCW DESTREHAN STREET WORKERS." UPON CONCLUSION OF THE ABRWH'S DELIBERATIONS REGARDING THE MALLINCKRODT CHEMICAL WORKERS, THESE DOCUMENTS AND ANY FORTHCOMING RECOMMENDATIONS OF THE BOARD WILL SERVE TO FACILITATE AND EXPEDITE COMPLETION OF THIS SECTION ENTITILED "DETERMINING RADIOACTIVITY INTAKES DURING THE OPERATING YEARS."}

6.1 ASSUMPTIONS AND GENERAL INSTRUCTIONS

The following assumptions should be made in determining individual radioactivity intakes when urinalysis records are used.

1. Except for acute intakes arising from incidents (which should be treated as special cases), the intakes should be considered chronic.

2. The absorption type should be selected using the guidance from the table below (ICRP 1968). To be claimant favorable, the selection of absorption type should be made so as to maximize the dose to the organ of interest. ICRP 66 default parameters related to particles are appropriate. (See Section 5.3.1 for more information.)

Isotope	Form	Absorption Type
U	U_3O_8, UO_2	S
	UO_3, UF_4, U metal	M
	UO_2F_2	F
Th	Oxides, hydroxides	S
	Other	M
Ra	All	M
Po	Oxides, hydroxides	S
	Other	M
Pa	Oxides, hydroxides	S
	Other	M
Ac	Halides, nitrates	M
	Sulfates	F

Note that for residues, the oxide/hydroxide form is the most probable for Th (AM-7) and that the Ra was in the form of the sulfate.

3. The number of work hours per year is 2000, i.e., 40 hours per week, 50 weeks per year. Adjustments should be made in individual cases if more specific information is available, such as prorating for partial years or overtime. If employment information is unclear, the most conservative or claimant-favorable appropriate job title should be used.

   For example, if bioassay data for a worker is missing for an entire year and he spent 5 months of that year as an electrician and 7 months as a mechanic, his intake for the year should be assumed to be 5/12 of the annual intake for an electrician plus 7/12 of the annual intake for a mechanic. The appropriate period from Table 28 should be used for this, depending on how many years of data are missing. However, if he is listed as having worked as an electrician and mechanic for the year, with no time breakdown for these job titles, then it should be assumed that he spent the entire year as a mechanic (the more conservative choice).

4. For Plants 1 and 2, if no plant-specific data are available, the most conservative (usually the earliest) applicable data for Plants 4 and 6 should be used (applied retroactively), as appropriate for the worker activity. (Note that a comparison of Table 10 figures to Tables 12 and 13 figures supports this assumption since the latter are daily weighted averages and the former are more likely spot maxima.)

5. The breathing rate for all workers, if an assumption is needed, should be taken to be 1.2 m^3/hr (as shown in Table 6 of ICRP 66 (ICRP 1994)).

6.2 SPECIAL NOTE REGARDING DETERMINING INTAKES FOR PLANTS NO LONGER UNDER MED/AEC CONTRACTS

There do not appear to be any records of decontamination and decommissioning work and other work at Plants 1 and 2 for the period between the time they ceased operations in 1946 and the time they were released to Mallinckrodt in 1951, i.e., from about 1947 through 1950, although this work was done under an MED/AEC contract. The decontamination itself was conducted by Mallinckrodt (ORNL

1981) and further decontamination was possibly performed in 1954 and possibly 1970 (Mallinckrodt 1994), but all of this work this was likely covered by badging and urinalysis (i.e., the workers were considered to be associated with the still-operating Plants 4, 6, 6E, or 7 and so would have been covered routinely). Thus it is expected that decontamination workers have film badge and probably urinalysis records. If not, then these Plants 1 and 2 decontamination workers, should have their 1949-1950 exposures determined on the basis of decontamination worker data as given in Section 8.0 (i.e., the Section 8.0 data would serve as surrogate data).

For workers that may have been still working in the Plants 1 and 2 buildings on non-decontamination operations, it may be assumed that radioactive operations they formerly worked on continued as previously; this would appear to be a claimant-favorable assumption. Thus the maximum figures (either surrogate data or Tables 29-31 data) for 1942-1946 for these workers can be assumed to apply from 1947-1950 as well.

For the period after decontamination and decommissioning, i.e., from 1951 on, the exposures should be determined as described in Section 8.0, using the data of Table 40 to establish a reasonable and appropriate annual exposure.

NOTE: SECTION 7.0 BELOW NEEDS TO BE REVISED TO INCORPORATE THE SURROGATE FILM BADGE DATA, WHEN AVAILABLE – DON'T USE EXCEPT IN A PROVISIONAL FORM (THE CHEST X-RAY INFORMATION SHOULD BE CORRECT, HOWEVER)

7.0 DETERMINATION OF EXTERNAL DOSES DURING THE OPERATING YEARS

This section applies to determination of external gamma, beta, and neutron doses during the operating years only. For external doses during the decontamination years and the postoperations years (1959-1995), see Section 8.0.

7.1 ASSUMPTIONS AND GENERAL INSTRUCTIONS

The following assumptions should be made in the determination of external dose.

1. Most external dosimetry monitoring records appear to be available, but where such records are missing doses must be estimated. As alternatives for dose estimation information, area monitoring data for external radiation is sparse and although much is known about the characteristics of the process material (source term), the lack of knowledge of such relevant factors as geometry make dose estimation using this data subject to a great deal of uncertainty. Thus the approach incorporating the least uncertainty is likely to be using existing dose monitoring records to create an applicable surrogate(s) set of data.

2. Mallinckrodt dose records demonstrate that a substantial proportion of employees had film results recorded as 0-50 mrep or mr. From this it can be assumed that the dose-monitoring program was sufficiently conservative that even many individuals who did not receive significant occupational exposure were also monitored. It can be further assumed that the converse is true, that individuals who were not monitored were unlikely to receive significant occupational exposure. This assumption is supported by the observation of Hickey and Dupree (1984) that Mallinckrodt health physicists and industrial hygienists presumably monitored those jobs and workers where they thought the problems were and that it can thus be inferred that infrequently monitored workers such as the general administrative staff were not considered to be at risk of significant exposures.

3. Photon doses for all workers should be assumed to be in the 30 – 250 keV energy range. This is consistent with NIOSH (2002a). It is also consistent with the fact that all but a negligible fraction of the gammas emitted by U-235, U-235, and their daughters down to radon are in this range, including the principal gamma of Ra-226 (0.186 MeV) and the principal gammas of Th-230 and Th-232.

4. For the purposes of dose reconstruction, the "beta" readings in the Mallinckrodt dose records are assumed to be equal to the shallow dose, $H_p(0.07)$. Beta dose should thus be assigned as shallow dose, with a claimant-favorable dose conversion factor of 1.0 for application of this shallow dose to the skin, testes, and female breast. It should be assumed that electrons are of energy greater than 15 keV.

   Also, as noted earlier, AEC studied attenuation of betas by workplace clothing (AEC 1950i). Attenuation was determined for coverall cloth to be an average of 21.8% (standard deviation of 7.5%) for distances of 5-36 inches from a sheet of uranium metal in equilibrium with its two major beta-emitting progeny, Th-234 and Pa-234m. Since this replicated work source conditions and clothing type, the results may be applied directly in dose reconstruction. Although variability in the mean attenuation appeared to be related to distance from the source, no pattern could be identified. Hence in dose reconstruction a constant value of 20% attenuation in shallow dose at cancer sites concealed under clothing should be used, i.e., a factor of 0.8 should be applied in the testes and breast cases to allow for shielding by work clothing.

5. When job assignments are known in sufficient detail from the individual work history, geometries may be applied by specific job title from Table 18, which was derived from engineering judgment and from the time-and-motion measurements made as part of the AEC dust monitoring studies. Should the job title(s) be unclear, a reasonable assignment from Table 18 should be made on the basis of applicable job or work area notations in the film badge and/or urinalysis records.

   When individual information is not present in the employee information in sufficient detail to allow an assignment in Table 18, more general categories should be applied to these cases. Although the exposure geometries for the three job categories in uranium facilities listed in NIOSH (2002a) (general laborer, machinist, and supervisor) are useful when little is known about a given uranium facility, details of the Mallinckrodt facilities are understood well enough that the general categories given below should be used when Table 18 cannot be used.

   <u>50% AP/50% ROT</u>

   This category corresponds to the “supervisory” category in NIOSH (2002a). Job titles to which this geometry should be applied are supervisor, foreman, mechanic, and maintenance personnel who had no specific equipment or area assignments or were assigned to multiple buildings.

   <u>25% AP/75% ROT</u>

   This category corresponds to the “general laborer” category in NIOSH (2002a). Job titles to which this geometry should be applied are warehouse worker, general laborer (personnel not assigned to specific process equipment) in process plants, engineer, forktruck driver, instrument technician, and others who had frequent process area access but were not assigned to specific process equipment.

   <u>75% AP/25% ROT</u>

   This category corresponds to the “machinist” category in NIOSH (2002a). This geometry should generally be applied to many process workers assigned to a particular area.

   <u>90% AP/10% ROT</u>

   This category should be used for Mallinckrodt workers who performed process work and who likely received a majority of their dose in the frontal geometry in a short period of time. This geometry should generally be applied to process workers in the higher-dose categories.

   <u>100% ROT</u>

   This category should be used to assign dose to non-process workers, i.e., workers expected to have entered process areas only incidentally if at all and who rarely if ever approached process equipment as part of their assigned duties. Job titles to which this geometry should be applied include clerk, other office workers, and dispensary personnel. (However, some clerks involved in materials accounting had frequent access to process areas and should not be included in this category.)

Note that a survey completed at the end of operations (MCW 1959) stated that in general most of the radioactive contamination was located in floors and in walls below the six-foot level, with walls, steelwork, and metal platforming above the six-foot level usually showing only slight contamination; this suggests except for tall tank contributions, there was not much dose contribution from above. Also, the nature of the process and available floor plans indicate that there was little overhead process piping. Further, the early process involved much direct manual handling of the radioactive material. Finally, for the workers receiving the highest doses, higher-dose activities often imparted much of the dose in a short period of time while they were directly handling the material or equipment. Thus isometric-geometry exposures are included in the detailed geometries in Table 18 for activities likely to result in overhead dose. Error in these assumptions occurs on the side of claimant-favorability by adoption of the ROT as opposed to the ISO geometry in most cases.

6. The descriptions of the film badge in Section 5.4.3.2 above do not suggest calibration using a phantom, so it is likely that open-air calibrations were performed. Thus the recommendation is that Mallinckrodt recorded doses be converted using dose conversion factors for Roentgen-to-H_T dose for photons from Appendix B of NIOSH (2002a). While film badges are known to overestimate doses from low-energy photons, the low-energy component does not seem to be a significant characteristic of the Mallinckrodt spectrum. Thus no modification is proposed to recorded deep doses, once converted to organ doses using the Roentgen-to-H_T dose conversion factor.

7. Examination of entries for “gamma” and “beta” in the Mallinckrodt dose records shows that the beta doses were obtained by subtracting the optical density for the shielded portion of the film from that of the unshielded part. This is also shown by a series of memoranda between the AEC and Mallinckrodt (AEC 1950f, AEC 1950h, MCW 1950o) in which the method of subtracting the two quantities was discussed. For Mallinckrodt, it is assumed that the beta readings were obtained by subtracting the density under the shield from the density under the window and assuming the difference in density was due to beta radiation (MCW 1950o).

8. For the purposes of dose reconstruction, the minimum level of detection should be assumed to be 50 mrep/week, based on the fact that Mallinckrodt individual dose monitoring records list many entries in the gamma column as “50*” and the asterisk refers to a footnote that reads “indicates less than.” Individuals without monitoring data should thus be assigned “incidental dose” for each weekly cycle worked. This should be assigned as photon dose equal to the limit of detection divided by 2 (LOD/2), or 25 mrep, for each weekly cycle worked, as given in NIOSH (2002a). This 25 mrep should assumed to be in the form of photons of energy between 30 and 250 keV and the exposure geometry should be taken to be 100% ROT.

   Note that the use of the LOD/2 method results in “slight positive bias” for monitored workers, as stated in NIOSH (2002a) and that additional claimant-favorability arises from the fact unmonitored workers were less likely to receive significant occupational dose (as per Item 2 above). True doses are thus closer to zero than are “less-than-detectable” or “missed doses” for monitored workers. This overestimating assumption is a claimant-favorable way to account for individuals who may have been incidentally exposed.

9. The lognormal distribution should generally be used to determine missed dose, as stated in NIOSH (2002a).

7.2 INFORMATION REGARDING THE 1950 AEC CUMULATIVE DOSE RECONSTRUCTION STUDY

The lack of early external monitoring data for Mallinckrodt reflects the novelty of the uranium processing industry, the provisional nature of early uranium activities at Mallinckrodt, and the assumption that airborne exposure was the primary hazard. The implementation of a more comprehensive health and safety program at the end of the war and in the early post-war period led to questions about external doses that previously had gone unmeasured. This resulted in the publication of the AEC report "An Estimate of Cumulative Multiple Exposures to Radioactive Materials, Mallinckrodt Chemical Works Plants 4 and 6, July 1942 to October 1949" (AEC 1950a). AEC's results are not quoted here for two reasons. First, in dose reconstruction different assumptions are made than AEC made. Second, the study develops cumulative dose data, but unfortunately does not apply it to individual workers or distinct work groups, while doses applied to individual workers for dose reconstruction should rely on recorded doses for actual worker or worker groups. However, dose reconstructors should be aware of this report in the case of workers who began MED/AEC work before film badges were routinely worn and for whom AEC's "back-calculation" estimate may be found listed in dose records as simply an accumulated dose for the pre-monitoring period of operation.

The cumulative exposure estimates covered workers then employed at Plants 4 and 6 who had been employed in MED/AEC work at Mallinckrodt between July 1942 and October 1949 and who had more than six months of exposure to radioactive materials. AEC's estimates for the dose to the skin were based on film badge data and for the dose to the bone on breath radon analysis (to determine the fixed radium burden) and film badge data. Calculations of lung dose did not include a gamma contribution because AEC deemed it negligible compared to the dose from airborne particle inhalation. It should also be noted that AEC thought that the exposures in the unmonitored years were "at least as severe as they were found to be at the time of our initial studies" (in early 1947); conditions probably were not more favorable and may have been "moderately" more severe. Thus they thought that their extrapolations could possibly, but not necessarily, be somewhat nonconservative.

7.3 GAMMA AND BETA DOSE

NIOSH (2002a), the external dose reconstruction guide, states that the hierarchy of dose determination sources are personal dosimeter (the film badge worn by the individual); pocket ionization chamber; and group or co-worker dosimeter. Pocket chamber data does not appear to be available for Mallinckrodt and in fact, use of the pocket chamber (pocket meter) was discontinued in 1949, as explained in Section 5.4.1. However, such data may come to light at a later date and in that case, some calibration and other information can be found in Section 5.4.1 and the references it cites. NIOSH (2002a) does not list the area monitoring dosimeter, such as Mallinckrodt used in many areas as a secondary means of monitoring. However, such data, if available, could be used in a comparative way if none of the other data sources can be found.

Generally, for a given claim, dates of employment should be compared to the available dose monitoring information. If dose monitoring records are likely to be available, the dose reconstructor should request project personnel to search the available records and list recorded doses for the case (or request that appropriate project personnel do so). Then reconstructed dose should be assigned for each weekly cycle worked during the unmonitored period. Dose reconstructions should then be performed in accordance with the dose reconstruction project external dose procedure (ORAUT 2003a), the NIOSH external dose reconstruction guidance (NIOSH 2002a), and any other applicable project guidance.

7.3.1 Selecting a Surrogate Cohort and Formulating a Surrogate Data Set

Surrogate (analogous co-worker) data, in Mallinckrodt cases, is a matter of identifying an appropriate set of workers with a similar work history. That is, in general, to form a surrogate set the dose reconstructor should identify at least several other workers with appropriate monitoring data who did the same work at the same time as the subject worker; the monitoring results for these surrogate workers should then be pooled to arrive at a statistical representation of the likely dose to the worker. A lognormal distribution should be assumed.

For the unmonitored years, identification of a surrogate set must make use of records from monitored years and of records such as contemporaneous employment rosters showing names of employees and their work groups, to identify potential candidate surrogate co-workers.

For the monitored years, many, perhaps most Mallinckrodt film badge records and urinalysis records had some indication of job title, task, or work area. Thus the formation of a surrogate set of data should be possible for all monitored years by inspection of the weekly dose reports, supplemented as necessary by inspection of the quarterly urinalysis reports. (Note that the urinalysis reports are a useful and available additional source of co-worker and job assignment information for external dose reconstruction even though the urinalysis results themselves are not relevant to external dosimetry.) Dose reconstructors must compare information available from the DOE record and the computer-aided telephone interview (CATI) to the records in order to identify workers with a similar work history.

One complication of selecting a surrogate set for the monitored years is that from some point on in the postwar years (1949?), a rotation system was used in work assignments in order to reduce individual process worker external exposure (MCW 1950t; AEC 1951b). The work done by an individual would therefore vary over the course of a year as he was rotated in and out of high-exposure areas. Thus for best results, where data for an individual known to be working at Mallinckrodt in AEC operations is lacking or is unclear, some care must be taken to identify co-workers whose assignments are most analogous to the workers. The rotation system was done principally on a crew basis, i.e., the same small subgroup of individuals were assigned to the same work at any given time, so the most appropriate individuals are those whose names appear on the known records for the individual as doing the same work at the same time and not, say, individuals doing the same work in a later period. However, if the number of same-time co-workers is deemed by the dose reconstructor to be too small to provide adequate statistics in view of the variation in the doses, the similarly assigned co-workers doing the same work in the period immediately before or after the period of interest should be included. The rotation system apparently ended when the doses went down sufficiently, probably at some time in the early 1950's. Hence from about 1952 on, the records will likely show more sustained assignment of the same work to an individual and identification of same-time coworkers would be less important.

7.3.2 Determining Dose During the 1945 15-Week Monitoring Period and Any Earlier Monitored Period

This section applies to workers who are listed in the Mallinckrodt Radiation Summary (MCW undated). Data for the initial known 15-week monitoring period are given in the Mallinckrodt Radiation Summary (MCW undated). Doses during this period should be determined by extrapolation or interpolation from the total dose listed and from any other dose records from 1945 that may become available. In MCW (undated), only total doses for all weeks worked are listed, so the average weekly dose for each individual should be computed by dividing the listed totals for gamma and beta by the number of weeks worked. This average weekly gamma and beta dose should then be assumed to apply for each week worked during the period if the worker was employed in the AEC-sponsored operations whether or not all the weeks of work were represented in the listing for that worker in MCW (undated)

For example, if MCW (undated) shows dose for a worker covering 12 of the 15 weeks but employment records show that he was employed in the uranium work for all of the 15 weeks (and was not absent from work), then the average gamma and beta doses calculated on the basis of the 12 weeks should be applied to the other 3 weeks as well.

7.3.3 Estimating Dose, Unmonitored Period 1942-1945

7.3.3.1 Workers With Subsequent Dose Monitoring Records

For workers whose covered employment includes some part of the unmonitored period in 1942-1945 and who have dose results from the early monitored period (i.e., from some time in late 1945 on), the Mallinckrodt Radiation Summary (MCW undated), any other film badge records that may be available from 1945, and the film badge records from 1946 should be used to estimate the dose during the unmonitored period.

For workers listed on the radiation summary (MCW undated) or other 1945 dose tabulations who appear to have done the same or similar work during the unmonitored period as during the monitored period, the average weekly gamma and beta doses should be determined as stated in Section 7.3.2. They should then be applied to cover each week worked during the unmonitored period, if the worker appears to have done the same or similar work during the unmonitored period as during the monitored period.

For workers who are not listed on the radiation summary or other 1945 dose tabulations, the dose records for 1946 should be checked. If the work done during the unmonitored period appears to be substantially the same in function and area as during the monitored period, then the 1946 dose should be applied on a weekly average basis to estimate the dose during the unmonitored period. In situations where data for 1945-1946 is incomplete or unavailable, the dose reconstructor should move on to subsequent periods for monitoring data.

For workers who are not listed on the radiation summary or other 1945 and 1946 dose tabulations and for workers who are listed on the summary but whose work during the unmonitored period appears to differ significantly from that during the monitored period, a surrogate worker set should be identified from the radiation summary and other 1945 and 1946 dose tabulations. Appropriate gamma and beta doses should then be calculated and applied for the unmonitored period.

For any worker for whom a surrogate set cannot be identified because of lack of clarity in the records about the nature or area of his work, doses should be assigned from Table 36 (see also Section 7.3.3.2 below). When Table 36 is used, the median dose should be applied to each cycle for which dose is reconstructed during the unmonitored period.

7.3.3.2 Workers Without Subsequent Dose Monitoring Records

This section applies to workers whose work histories indicate that they worked at the AEC-sponsored uranium processing facilities during the unmonitored period (i.e., prior to the start of film badging in about mid-1945) but who had no film badge records for the monitored period. It is assumed that such workers (1) terminated prior to the beginning of dose monitoring, (2) moved to some non-AEC work at Mallinckrodt prior to the beginning of dose monitoring, or (3) performed work at the uranium facilities during the monitored period that did not meet Mallinckrodt-AEC criteria for monitoring.

Workers assumed to have terminated prior to the start of external radiation monitoring

A surrogate dose history should be formulated based on workers with similar job titles or functions and work areas during the monitored period, i.e., from 1945, 1946, or the closest subsequent period for which sufficient data exists.

Workers who were outside the AEC-sponsored uranium operations during the monitored period

A surrogate dose history should be formulated based on workers with similar job titles or functions and work areas during the monitored period, i.e., from 1945, 1946, or the closest subsequent period for which sufficient data exists.

Workers who were apparently in the AEC-sponsored operations during the monitored period

For individuals who have no identified film badge records and who had work assignments subsequent to the unmonitored period that would probably not result in significant exposure (and who were presumably not monitored for that reason), doses should be assigned as median dose from the early monitored period, as given in Table 36. This provides a claimant-favorable estimate that likely addresses any incidental dose that may have been received from 1942-1945.

For individuals who have no identified film badge records and who had work assignments subsequent to the unmonitored period that would appear to have resulted in significant exposure (e.g., they have urinalysis or breath radon records with notations typically indicating access to process areas or other employment or health records clearly indicate some process area access), doses should be assigned as per Section 7.3.2 above.

7.3.3.3 Notes Regarding Dose Summaries

Monitored workers may have received up to 150 mrep in a week without the dose being recorded in a dose summary. There is some probability that the dose received was actually zero: dose summaries consistently show a significant number of badges in the 0-50 mrep dose range. For the purpose of dose reconstruction, the most claimant-favorable assumption is that any monitored worker assigned to a given plant, but not specifically listed in the dose records for the plant for that week, received a dose of 150 mrep that week. Division of the total between gamma and beta components should be based on an average ratio derived from the weeks with specific dose monitoring entries for that individual.

7.3.3.4 Notes Regarding the Application of Later Monitoring Data to the Unmonitored Period

The approach of applying later monitoring data to establish doses in the unmonitored period appears to be generally consistent with the approach of Dupree-Ellis et al. (2000), who stated that in an Oak Ridge Associated Universities study of Mallinckrodt workers, “for the 20.8% of working years in which doses were not monitored [i.e., 1942-1944], an algorithm was used to assign doses”. Also, the use of 1946 data (e.g., MCW (undated)) “should provide valid exposure estimates for this early period” (Dupree-Ellis et al. 1998).

Also, to what extent exposures during early “benchtop” operations differed from production-level exposures is not known. However, processes were developed on the bench and pilot plant levels and then scaled up to the production level; the production level then increased repeatedly throughout the wartime and early postwar years. Thus it is logical to conclude that the process and conditions for benchtop and pilot plant operations were similar to the production-level operations, but on a much smaller scale (including generally much smaller source quantities at a given time).

7.3.3.5 Notes Regarding Table 36

The values in Table 36 were generated from the average doses received by Mallinckrodt "pilot plant" workers during the earliest known period of film badge monitoring, the 15-week program described above (Rochester 1950). It incorporates significant uncertainty for the reasons listed below. However, dose estimates calculated as indicated in this section represent the best information at hand and are still expected to be claimant-favorable.

1. Only total doses were listed for each worker for the number of weeks monitored (n = 1-15); there was no monthly or weekly breakdown.

2. No detail is supplied in the reference as to what activities the workers were engaged in, other than the fact that the location was listed as "pilot plant." The memo, however, states that the doses are from "prior to the operation of the current plant", and is dated around the time that Plant 6 became operational; also, it is known that Plant 4 was called "the pilot plant" after Plant 6 was built and before a pilot plant was established at Plant 6. Thus, though it is unknown whether the listed doses were received at Plant 1, Plant 2, Plant 4, or a combination of plants, the results likely reflect doses received in the early operations prior to the improved control measures presumably implemented in the construction of Plant 6.

3. As noted in Section 7.3.3.1 above, how early "benchtop" operations may have resulted in doses that differ from the production level and thus the estimates in Table 36 is not known. However, the potential source would appear to have been much higher at the production level than at the benchtop or even pilot plant level and the exposure would have been more sustained over time (constant production). Pitchblende was processed only in limited quantities until Plant 6 was built to process it in bulk, shortly after the start of monitoring. So the potential gamma-emitting source would have increased markedly over what it had been in the unmonitored period.

4. As it is not possible to determine accurate production levels for the early period and for the period covered by the 15-week monitoring, no attempt has been made to scale the exposures to reflect the quantity of material processed.

From total doses in Rochester (1950), the dose distribution of the average weekly dose for the 32 workers considered was evaluated and the values in Table 36 were prepared. Distribution of the data and values for the median and geometric standard deviation were calculated using LOGNORM™ and CrystalBall©.

The methodology used to create Table 36 was compared with the techniques discussed in Watson et al. (1994). Although there are some differences (e.g., the equipment changed each several years, unlike the plant studied by Watson et al.), the approach described below resembles the use of a department mean or median dose as closely as possible with the available data.

7.3.4 Determining Dose – Monitored Period 1946 - 1958

This section applies for workers who have monitoring data after the initial 15-week monitoring period and for whom doses must be estimated for the operating years, i.e., 1946-1958. Plants 1 and 2 were not used for AEC work after about 1946 and underwent decontamination and decommissioning (D&D) in 1948-1950, after which they were released to Mallinckrodt in 1951 for non-AEC work. Similarly, the other plants stopped being used at varying points in 1957-1958 and underwent D&D thereafter. This section does not apply to any worker or work area after the end of operations; see Section 8.0 for dose estimation for the D&D years and the later postoperations years.

Because there is not a formal date or dates that can serve as a cutoff for the dose reconstructor to use, some judgment will have to be applied as to how much of the dose record should be counted as contributing to dose from AEC operations. This is particularly true since after the operations and D&D years, Mallinckrodt received a license from the AEC for the use of certain radioactive materials and so some workers may have been continued to be badged. Also, there may be continuing badge records for some individuals for years after operations ended at the St. Louis (Destrehan) site because these individuals transferred to or made visits to the Weldon Spring site, also run by Mallinckrodt, and had badges issued for work there.

7.3.4.1 Monitored Workers

For workers with monitoring data following the initial 15-week monitoring period (i.e., MCW undated), i.e., during the monitored period of early 1946 on, either the workers' own data or the data provided by a surrogate worker set(s) should provide sufficient information for dose estimation. Interpretation of the dose records should be made according to the applicable previous sections of this technical basis document.

When there is missing data in the monitoring records, they should be filled in accordance with the assumptions of Section 7.1, Items 7 and 8; the guidance of NIOSH (NIOSH 2002a); and the dose reconstruction project external dose reconstruction procedure (ORAUT 2003a). This is expected to consist largely of bridging small gaps by interpolation and bridging larger gaps using statistical surrogate data and/or standard missed dose estimates.

7.3.4.2 Unmonitored Workers – Plants Still Operating under MED/AEC Contracts

As stated above, most workers who received significant occupational exposure were likely to have been monitored and thus if a worker was not monitored, especially for a lengthy period of employment, this indicates that he was deemed to have little or no exposure potential. To account for the possibility of an unmonitored individual receiving incidental exposure to photons, dose should be assigned to unmonitored Mallinckrodt employees as follows.

1. An attempt should be made to find a surrogate category. The average or median value for the surrogate category should then be assigned to the worker as per Section 7.3.1.

2. If no surrogate category can be identified – i.e., if there are no monitored workers with the same job title or the job title listed in the subject's records cannot be associated with any surrogate job title – then either the Mallinckrodt subject matter expert for the dose reconstruction project should be consulted for guidance (and possibly further research) or the subject should be assigned a nominal category. Normally, the appropriate nominal category will be a low-exposure job title such as office worker.

7.3.5 Unmonitored Workers – Plants No Longer Operating under MED/AEC Contracts

As noted above, Plants 1 and 2 for the period between the time they ceased operations in early 1946 and the time they were released to Mallinckrodt in 1951, i.e., from 1946 through 1950, it can be claimant-favorably assumed for any employees still working on non-decontamination operations that operations still continued as previously. The decontamination itself was conducted by Mallinckrodt (ORNL 1981) and further decontamination was performed in 1954 and possibly in 1970 (Mallinckrodt

1994), but this was likely covered by badging (i.e., the workers would have been considered to be associated with the still-operating Plants 4, 6, 6E, or 7 and so would have been covered routinely).

Thus non-badged (unmonitored) workers not performing decontamination may have their former exposures (or those of surrogate workers doing non-decontamination work in previous years) projected through 1946-1950 as if the work continued. Unbadged decontamination workers, if any, should have their 1946-1950 exposures determined on the basis of decontamination worker surrogate data (see Section 8.0).

For the period after decontamination and decommissioning, i.e., from 1951 on, the exposures should be determined as described in Section 8.0.

7.3.6 Conversion to Organ Dose

For deep dose, the conversion to organ dose for the relevant worker category should be performed using assumptions of Section 7.1 and the energy-, geometry-, and radiation-specific dose conversion factors given in NIOSH (2002a).

7.4 X-RAY DOSE

7.4.1 Mallinckrodt-Specific Information

Employees of Mallinckrodt (cf. Section 5.4.5) received an annual occupationally related diagnostic x-ray (MCW 1955d; Mason 1958a). Up to some time in 1944, a pelvic x-ray was also performed for employees working with fluorides (see Section 5.4.4). The annual chest x-rays and presumably the pelvic x-rays were taken at the Barnes Hospital (the Washington University School of Medicine) (AEC 1950m; MCW 1955d) and the x-ray records remained the property of the hospital (AEC 1950m). There is no evidence so far in the Mallinckrodt documents to indicate when the annual chest x-rays began, although Fleishman-Hilliard suggests that this was done from the start since Mallinckrodt insisted at the outset that the Washington University School of Medicine be engaged to do the physical examinations. A claimant-favorable assumption would be that chest x-rays were performed annually from 1942-1958. Since the radiographs were made at a hospital, and hospital procedures routinely used both PA and lateral chest views much more commonly than non-hospital facilities, the annual dose from chest x-rays for Mallinckrodt workers should consist of the dose from both views. No evidence so far indicates that photofluorographic chest films were performed. An additional note regarding x-rays during the decontamination and decommissioning period is given in Section 8.1.

Since no actual x-ray output measurements or x-ray technique factors are available for the Barnes Hospital in Mallinckrodt records, default values for entrance kerma appropriate for this time period will be used in the calculation of organ dose conversion factors for use in dose reconstruction.

7.4.2 Chest X-Rays

Information to be used in dose reconstruction for the early years for which no specific information is available is provided in ORAUT-OTIB-0006, the dose reconstruction project technical information bulletin covering diagnostic x-ray procedures (ORAUT 2003c). ORAUT (2003c) should be referred to regarding the underlying bases, interpretation details, and a sample of a summary data table where actual beam data are known.

Doses for organs not listed in ICRP Publication 34 (ICRP 1982) but specified in the IREP code should be determined by analogy with anatomical location as indicated below.

Analogues for IREP organs not included in ICRP 34.

Anatomical location	ICRP 34 reference organ	IREP organ analogues
Thorax	Lung	Thymus Esophagus Stomach Bone surface Liver/gall bladder/spleen Remainder organs
Abdomen	Ovaries	Urinary/bladder Colon/rectum Uterus
Head and neck	Thyroid	Eye/brain

As ORAUT (2003c) notes, for any individual entrance skin exposure (ESE) or derived organ dose, an uncertainty of ± 30% at the one sigma confidence level may be assumed; for further conservatism it may be appropriate to assume that errors are all positive and thus that only the + 30% should be used.

7.4.3 Pelvic X-Rays

Guidance regarding pelvic x-ray examinations has not yet been issued by NIOSH or the dose reconstruction project. Thus consideration of this dose contribution will have to be deferred. Note that for Mallinckrodt, such contributions would be applicable to only some workers, i.e. those working with fluorides; presumably this class would consist of workers involved with the UO_2-to-UF_4 and the UF_4-to-derby processes, including workers who handled the hydrogen fluoride supply and storage functions. Also, such contributions would be applicable only from perhaps 1942 to 1944 at most, as indicated in Section 7.4.1.

7.5 NEUTRON DOSE

As indicated in Section 5.4.2, there was no neutron monitoring done at Mallinckrodt. Therefore it is not expected that there will be any neutron monitoring data or neutron dose rate measurement data found for Mallinckrodt. Thus any estimate of neutron dose will have to be calculated on the basis of source term data, as recognized by NIOSH (2002a). As explained in Section 5.4.2, the only potentially significant source of neutron exposure would have been neutrons produced by the alpha-neutron reaction in materials in which uranium and thorium were mixed with elements of low atomic number such as fluorine and oxygen. Table 34 shows the results of the dose rate calculations and also some calculated doses, together with the occupancy assumptions on which they were based. Assumptions not given in the table may be found in Section 5.4.2.

The annual doses from the uranium forms are negligible compared to the probable doses received by process and lab workers, who were the only workers likely to have received any neutron exposure at all and theirs was clearly negligible. Hence the neutron dose contribution from the uranium forms does not need to be investigated further. While this is probably the case with the thorium forms also, the thorium neutron doses should be considered for appropriate worker cases (claimants). Note that this dose is applicable only for those years or parts of years in which the thorium was processed, i.e., from after July 1955 through approximately March 1957.

Thus in assigning neutron dose, only the thorium workers need be assigned dose and the doses calculated as given in Table 34 are claimant-favorable values.

7.6 OTHER DOSE CONTRIBUTIONS

7.6.1 Extremity Dose

Given the relatively small number of claims and the unknown proportion requiring calculation of extremity dose, this subject is not treated in this TBD. Extremity dose estimates, when necessary, will need to be formulated on a case-by-case basis. See Section 5.4.3.5 for information about individual ring dose data and interpretation.

7.6.2 Submersion Dose

Submersion dose is likely to be significant only for the skin, testes, and breast, and is not used when testes dose is used to estimate dose to the prostate. As dose reconstructions are based upon the partial film badge dose monitoring records, submersion doses are not separately calculated.

7.6.3 Shallow Dose

As stated in Section 7.1, Item 6, Mallinckrodt dose records contain “beta” values obtained by subtracting the optical density of the film behind the cadmium shield from that behind the open window. These recorded values are assumed to be equivalent to $H_p(0.07)$. For the purposes of dose reconstruction, shallow dose should be assigned from the film badge data using a claimant-favorable dose conversion factor of 1.0.

8.0 DETERMINATION OF EXPOSURES DUE TO RESIDUAL CONTAMINATION REMAINING FROM MED/AEC OPERATIONS

As part of the process of dose reconstruction, the doses from postoperational decontamination and decommissioning activities and from postoperational work activities must be determined, insofar as they are due to residual contamination remaining after MED-AEC operations ceased. For this purpose, operations at the Mallinckrodt St. Louis main site are considered to have ended in 1957-1958 and decontamination and decommissioning (D&D) of the various buildings used for MED/AEC work to have begun in 1959 and continued through 1961; and the postoperational period to have extended from 1962-1995. Regular operations at SLAPS are considered to have ceased in 1958 and the caretaker/D&D operations period to have extended from 1959-1962.

8.1 DESCRIPTION OF THE DECONTAMINATION AND DEMOLITION WORK AT THE ST. LOUIS MAIN SITE, INCLUDING HEALTH AND SAFETY PRECAUTIONS

As noted in Section 3.0, for buildings in Plants 1 and 2, decontamination had been done some years earlier. Mason 1977 states that this started in 1949; Mallinckrodt (1994) states that there were programs to do so in 1950 and 1954 and even in 1970; and ORNL (1981) mentions only the decontamination of 1948-1950. Little is known of the Plants 1 and 2 decontamination campaign because although Mason (1977) states that detailed reports of the work, including the final surveys, were prepared, these records are not available.

What is presently known is as follows. In 1946, Mallinckrodt took core samples of the subsoil under Buildings 51 and 52 in Plant 2 (MED 1946f), boring through the concrete floor, presumably at AEC's behest. This indicates that even the earliest characterization effort likely took place after the start of film badging and urinalysis; decontamination workers may have been given badges and urinalysis due to the dust and dose rate potential of the work. The decontamination that began in 1949 was performed by "Main Plant" (Plants 1 and 2) crews under directions from Mallinckrodt Uranium Division "monitors", as the health and safety specialists were called (Mason 1977). The contaminated waste material was collected and delivered to AEC, presumably in drums or boxes, while contaminated equipment was either recycled to Plant 6 or was transferred to AEC for disposal (Mason 1977). Some of the material apparently ended up buried at SLAPS. When the work was done to the satisfaction of the AEC St. Louis Area office, monitoring personnel from AEC-NYOO did a final survey of Plants 1 and 2 (Mason 1977). When AEC-NYOO was satisfied, AEC released the plants to Mallinckrodt for unrestricted use in 1951 (FUSRAP 2002; DOE 1981).

With regard to Plants 4 and 6, some preliminary work, such as some initial cleanup, was done in 1955 and in 1958-1959 (AEC 1955b; MCW 1958; MCW 1959; MCW 1961b). For example, the process equipment was thoroughly cleaned by Mallinckrodt at the time the plants went into standby, before the formal decontamination and decommissioning work began (MCW 1961b). Some initial surveys to evaluate Plants 4, 6, 6E, and 7 were then performed (AEC 1955b; MCW 1958; MCW 1959; MCW 1961b). It was determined that many buildings at the site were so contaminated that it would have been too expensive to clean them to usable levels, and they were torn down. All of the Plant 4 buildings and most of the Plant 6 buildings were torn down, but much of Plant 6E and 7 survived (Mallinckrodt 1994), amounting to about 20% of the MED/AEC plants at the site (MCW 1961b).

MCW (1961b) describes the health and safety (H&S) aspects of the decontamination and demolition (D&D) work at Plants 4, 6, 6E, and 7; this is the source of the information that follows in the rest of this section (8.1), except where another source is explicitly indicated. This work was performed by several construction and demolition companies apparently contracting directly with AEC. Mason (1977) states that one of them, or the principal one, was Arch Wrecking Company. H&S coverage was provided by Mallinckrodt, again under contract with AEC, because AEC considered that while the D&D companies were qualified to deal with normal hazards of the work, they were not experienced in the sort of chemical and radiological hazards they would encounter at the Mallinckrodt site (MCW 1961b; Mason 1977). The Mallinckrodt H&S coverage was by experienced full-time personnel from the Destrehan (St. Louis) site and part-time personnel from the Weldon Spring site, which was also run by Mallinckrodt (MCW 1961b; Mason 1977). Oversight was provided by AEC (MCW 1961b; Mason 1977); Mason (1977) states that only AEC made the decisions regarding items that could be released for restricted or unrestricted use. The Mallinckrodt Uranium Division performed a final decontamination survey of the four remaining plants for AEC in 1960-1961 and gave a report to AEC (Mason 1997). Only part of the report of this closeout survey is available.

A physical checkup of all prospective workers on the project was provided. This included chest x-rays (PA and lateral, using full-size 14 x 17 films) and blood tests. Approximately 125 contract workers were examined. Work clothing was provided and postwork showers were required.

Film badges were issued to all persons working in or visiting the Destrehan site during all phases of equipment removal, decontamination, and demolition. These were of the standard badge type always used at Mallinckrodt: an A. M. Samples stainless steel film badge holder with open-window and cadmium filters, using DuPont Type 552 dosimeter film; the gamma standard was a platinum-encapsulated radium needle and the beta standard was an aged natural uranium block. Film badges were processed monthly, with 80 to 124 contractor badges processed each month. Film badges worn by Mallinckrodt and AEC personnel were those provided them through the Weldon Spring facility for their work at the Destrehan site in St. Louis.

Workers were also required to submit urine samples before the start of the project, at approximately six-week intervals during the project, and at the end of the project. The regular samples were taken on Friday after work and on Monday before work. Special samples were taken from those working in high dust exposure areas or in special operations such as sandblasting contaminated tanks. Urinalysis was done at Weldon Spring as a uranium-in-urine fluorometric analysis using a Jarrell-Ash fluorimeter, sodium fluoride as the fluxing agent, and a platinum dish as the sample holder. Each urine sample was analyzed in triplicate, with a 0.1 ml aliquot used each time.

AEC, the Mallinckrodt H&S people, and the D&D companies held pre-work planning discussions to determine how to do the work on the basis of the surveys. As work began in each area, further meetings and work area inspections were held to re-evaluate the approach for cleaning the particular area, based on how much contamination was revealed by equipment removal, how tenacious the contamination was, etc. Thus any of various decontamination and survey methods were used, depending on the characteristics of the area. Methods used for decontamination included mechanical “scratching” (to remove embedded particles on tar and gravel roofs); dry sandblasting; water blasting; shotblasting; steam cleaning with caustic-detergent additives; paint stripping; acid-detergent wash; immersion in solvents; burning; and sweeping, shoveling, and vacuum-cleaning.

Some particular operations performed and precautions taken were as follows.

1. In scrap removal and decontamination, areas were first inspected to verify that there was no gross radioactive material content and to plan the means of containment, such as removal to containers or sealing. Approximately 80% of the equipment was handled during the D&D work.

   To steam-clean scrap, surfaces were wetted using high-pressure fog or spray, to which a caustic-detergent agent had usually been added; they were removed with hoists; and washing with water hoses was done, with runoff retained in the localized area. Grossly contaminated items were steam-cleaned on a grating placed over a vat, with the overflow going into the sewer through a weir. Half-face respirators were used for steam cleaning.

Paint stripping was also done where necessary prior to steam cleaning, with the stripper being applied by brushing or dipping, often on the grating over the vat. Nickel-bearing steel pieces were cleaned in the open air by lowering the pieces into a vat containing acid-detergents and allowing them to soak for several hours; then a wire brush was used to loosen the scale and steam-cleaning was used to wash it away. Solvents (chlorinate hydrocarbons) were used on some items of low contamination level; this cleaning was done in the open air using one vat for cleaning and one for rinsing, with the solvent recirculating through a 5-micron Cuno filter and with the workers wearing face shields and gloves. After draining, the items were dried with forced air.

For sandblasting of relatively portable items, a large walk-in hood in Building 116 of Plant 6E was reactivated. Tarpaulins were used to extend the hood area and to enclose the hood face partially. The hood exhaust was through a high-efficiency bag filter. For any item that could not be taken to this hood, a portable canvas hood was set around the item, with ventilation provided. About 60 vessels were cleaned by dry sandblasting; the remaining several vessels (5%) that could not be cleaned to required levels had been used in the pitchblende digestion and radium extraction operations.

Insulation was burned off cable with added fuel, followed by a water rinse.

2. In building and structure decontamination, the most effective and intensive surface cleaning was done by dry or wet sandblasting. Dry sandblasting was the preferred method for removal in depth of masonry materials, paint, and encrusted surface contamination. A method called Vacu-blast, using steel shot, was also used; it provided vacuum pickup of the resulting particulates but recovery of the shot was time consuming. Full-face air-line supply masks and hoods were required, so care was taken to locate the air pump intakes upwind of the sandblasting operation or in a dust-free area. Generated dust was recovered or washed to sumps. Air changes were minimized by controlling outside openings; it was recognized that this increased interior concentrations in the short term, but it decreased exterior concentrations and, by decreasing air velocities in the area, accelerated the settling of the dust.

   Water blasting was used on many roof structural parts and beams and was also used on other parts to remove loose particles before sandblasting. Scrubbing with a detergent and brush was also used. Sweeping, shoveling, and vacuum cleaning were generally done with respiratory protection, due to the dust.

3. In decontamination of concrete pads in the yard, dry sandblasting was done in a canvas-covered booth on wheels, which was moved by two workers to sit over the spot to be blasted. Ventilation in the booth was provided by a fan discharging through a large furnace cleaning bag; the materials collected in the bag were removed every time the bag was moved. The operator in the booth wore supplied-air respiratory protection and his helper outside wore a half-face mask. It was noted that the radiation levels of the particles collected by the fan and caught in the filter bag were higher than the levels of the coarser particles that remained on the surface being cleaned, so that capture of radioactivity appeared to be efficient.

Dust from process areas was considered to be contaminated even before sampling and thus masks at a minimum were required for all dust-producing and fume-producing operations, including torch cutting. Respirators used for radiological hazards included high-efficiency mechanical dust respirators; bottled-air-supplied masks (for short-duration entries into high-concentration atmospheres and emergency use); and airline-supplied masks and hoods. Airline-supplied masks and hoods were required for sandblasting operations. The half-face dust masks were typically the MSA Dustfoe 66.

Regarding interior air sampling, samples were taken on a routine basis during the work shift, with filters being changed at the end of each daily sampling period; also, additional samples were drawn, either in the general work area or downwind, during the particular operations listed above and during other scrap cleaning. Air sampling was done with high-volume air samplers by Gelman Hurricane, Staplex Type TF1A, or the equivalent. The collection medium was Whatman #41 or HV-70 filter paper or Gelman AM-4 membrane filters. Sampling rates averaged 5 liters of air per square centimeter of filter media. Analysis for uranium was done by acid-leaching the filter, then counting an aliquot by photofluorometric analysis.

It was estimated that about 1% of the work during equipment removal, decontamination, and demolition was in areas where the airborne uranium concentration was on the order of 10-100 MPCa; for this category, full-flow air-supply masks were used. 10% of the effort was estimated to have been spent in areas where the concentration was 1 to 10 MPCa; for this category, personal half-face, dust-type masks were used. The rest of the work was done in areas of less than 1 MPCa. General area ("zone") concentrations during sandblasting were found to be higher around the local tarpaulin enclosures than around the fixed hood, but still below the (unspecified) AEC guide levels.

Regarding exterior air sampling, the generation of dust during demolition was at or near ground level; there were no discharge points to the upper atmosphere since the stacks were not being used. Dust sampling stations were established around the property line so as to have at least one always downwind of the work. The average of the uranium-in-air concentrations measured outside the buildings during demolition was 4×10^{-13} uCi/cc, said to be about one-tenth of the MPCa. It was observed that the gross alpha concentrations in the outside air were not much different from the uranium measurements, leading to the conclusion that radionuclides other than uranium were not present in significant quantities. The gross beta activities measured were judged to reflect the background seen at the time due to nuclear weapons testing.

The cleanup goals were surface alpha activity, 1000 dpm/100 cm^2 average and 5000 dpm per 100 cm^2 maximum spot; beta-gamma activity, 0.1 mrep/hr average and 1.0 mrep/hr maximum spot. This was achieved for the site and buildings, although the average beta-gamma level in some areas was somewhat above 0.1 mrep/hr. For alpha radiation surveys, the survey meters were calibrated in dpm per 100 cm^2 against normal uranium; they could be of the air ionization, proportional, or scintillation type and could detect 500-50,000 dpm per 100 cm^2, depending on type. For beta-gamma activity surveys, the survey meters were calibrated in mrad/hr or mrep/hr against normal uranium; they could be of the Geiger tube, air ionization, or scintillation type, could measure 0.3 to >1.0 mrad/hr, depending on type, and had to have a minimum unshielded probe face area of 2 square inches.

For surveys of removable contamination, for both alpha and beta-gamma, two wipe passes with half of a 4" disk of Whatman paper over 100 cm^2 were performed. All material items exceeding 50 lbs were monitored separately. Material items weighing less than 50 lbs, except for pipes, were monitored as a lot using a 20% random sample, providing the lot was from the same plant sources; if destined for uses other than smelting, they were monitored individually. Items were monitored both inside and out, for both fixed and removable contamination. If an item could not be monitored internally, it was considered to be contaminated. If it could be, then for beta-gamma, an item was considered to be contaminated if any surface scan reading was greater than 1.0 mrep/hr or if any smear exceeded 0.3 mrep/hr; no wipe test was done if all scan readings were less than 0.3 mrep/hr.

For alpha, an item was considered to be contaminated if any surface scan reading was greater than 25000 dpm per 100 cm^2, if the average of the surface scan readings was greater than 5000 dpm per 100 cm^2 or if any smear exceeded 2000 dpm per 100 cm^2; no wipe test was done if no scan reading exceeded 2000 dpm per 100 cm^2.

Because alpha probes were too large to be put inside pipes but Geiger tubes were not, the beta-gamma measurement was used as a surrogate for the alpha measurement; this was said to be because Th-234 and Pa-234, both beta emitters, reached equilibrium with uranium in approximately 250 days. However, it was recognized that nonequilibrium conditions or the presence of other beta and gamma-emitting isotopes could significantly affect the accuracy of the estimate of inferred uranium content. Pipes were not sampled as a lot. They were cleaned and unbent or unkinked, when possible, before monitoring. The presence of visible uranium deposits automatically caused a pipe to be classified as contaminated, as did the presence of bends, elbows, valves, fittings, or crushed sections. A Thyac meter on a ten-foot pole (“ten-foot probe”) was used for long sections.

Final contamination measurements are given in detail in MCW (1961a) and in summary in MCW (1961b). During the final survey, the following instruments were used: for alpha, the Eberline PAC-3G gas proportional counter and the Victoreen Model 356 ionization chamber survey meter; for beta-gamma, the Victoreen Model 389C Thyac (G-M) survey meter, background about 0.1 mrep/hr. Calibrations were as given above. A spot sampling method was used for wall, ceilings, and beams, while a path scanning method was used for floors, yards, pads, and roofs. The alpha measurements for the roofs, however, were spot readings. During demolition and the final survey, it could be seen that some amount of residual contamination was buried at the site. This contamination was generally located adjacent to foundations, between joints and concrete floors, under storage pads, and between joints and structural steel members. This was judged to be fixed and not to present a problem.

Table 37 shows the experience reported on a groupwise basis in MCW (1961b), including film badge, urinalysis, and air sampling results.

8.2 POST-DECONTAMINATION CONDITIONS AT THE ST. LOUIS MAIN SITE

After the decontamination of Plants 1 and 2, the plants were released to Mallinckrodt in 1951 (FUSRAP 2002; DOE 1981). No information about post-decontamination conditions there appears to have been produced until the ORNL survey of 1977 (ORNL 1981).

Following the decontamination or demolition of the buildings composing Plants 4, 6, 6E, and 7, post-decontamination use of the site with residual MED/AEC contamination began. This is considered to have started in 1962 and continued through 1995, the year when the Formerly Utilized Sites Remedial Action Project (FUSRAP) took over the site for final decontamination to modern standards. But in fact, release of the site to Mallinckrodt may have occurred plant by plant over 1961-1962, as suggested by Mallinckrodt (1994). A survey was done at the end of the decontamination period to demonstrate that the site was ready for release by AEC. This survey is documented in MCW (1961a).

No further survey of any of the plants to gauge the residual contamination appears to have been done in the post-D&D period until 1977, when a survey team from Oak Ridge National Laboratory did a detailed contamination, radiation, and radon survey of the site (ORNL 1981). A summary of the ORNL contamination findings is shown in Table 38. Also shown in Table 38 are radon concentrations reported in 1981 by ORNL (ORNL 1981) and in 1986 and 1990 by Applied Nuclear Safety (1986; 1991). These show that the post-decontamination radon levels were low compared to the levels of the operational years (Table 25).

8.3 GENERAL CONSIDERATIONS AND APPROACH FOR ST. LOUIS MAIN SITE DECONTAMINATION AND POST-DECONTAMINATION DOSE RECONSTRUCTION

As noted in Section 8.0, exposures during the D&D work appear to have been completely monitored, with film badge, urinalysis, and supplementary air sampling data being available. For individual D&D workers for which personal monitoring data are unavailable, surrogate exposure data should be assembled by the dose reconstructor based on work group identity and the nature of the work performed.

An applicable note in MCW (1961b) should be mentioned here with respect to surrogate data for the D&D period. In MCW (1961b) it was stated that Mallinckrodt experience at the Destrehan plant during the operating years was that for individuals whose daily integrated particulate air exposure was 50 mg of uranium per cubic meter of air, the after-work urine samples collected on Friday typically were in the range of 0.050-0.060 mg per liter and the before-work samples collected on Monday were in the range of 0.025-0.030 mg per liter. The D&D urine samples typically were found to be about half this level. (Note that the reference gave 50 “mg” per cubic meter of air as the daily integrated exposure, but it is likely that 50 micrograms per cubic meter was meant, since that was the standard tolerance level and would have been the logical quantity to cite.)

It should be noted regarding the post-D&D years that although some employees (remaining) at the St. Louis main site may be found to have urinalysis records extending into the late 1960’s, this monitoring was apparently done for Mallinckrodt’s post-AEC work with radioactive materials (e.g., its columbium-tantalum processing operations). These urinalysis records may be helpful in establishing the elimination curve of radioactivity from the body for anyone who (1) worked in the uranium processing operations up to 1958 and then worked in Mallinckrodt’s non-AEC operations or (2) was involved in the D&D work and then continued to work at the site in Mallinckrodt’s non-AEC work. However, these records may be misleading because they may indicate new uptakes from non-AEC sources. Also, people who worked in the decontaminated and released buildings from the release date on would likely not have been covered by film badge and urinalysis programs; in any case, persons monitored after 1961 were not monitored under the aegis of AEC. Thus bioassay or external dose information for them, if any, may not be available in dose reconstruction project files.

8.4 GENERAL CONSIDERATIONS AND APPROACH FOR SLAPS POST-OPERATIONS DOSE RECONSTRUCTION

Because SLAPS did not receive wastes on a regular basis after the end of Mallinckrodt St. Louis main site operations in 1958, the “postoperations” period for SLAPS may be considered to have started at that point, i.e., in 1959. The end of this period is set at the end of 1962 because in that year control of the site was passed to a private entity by the granting of a license to the entity by AEC.

There was no decontamination of the site per se during the period 1959-1962. However, some material was removed and some new material was added, as follows. In April 1959, a railroad siding and loading facilities were installed, apparently in preparation for moving out the residues and other material (AEC 1959); the tailings from the magnesium fluoride slag processing were then sent to the Fernald site (AEC 1959). This appears to have been the only removal of residues from the site during this period (the rest were removed after the transfer of control to the private entity.) Additionally, some of the scrap and rubble from the demolition and decontamination of the main St. Louis site facilities was buried at the west end of the site (AEC 1972; ORNL 1979) in about 1959-1960; this was the same area where contaminated metal and debris had been buried previously (presumably in 1948-1950, following the decontamination of Plants 1 and 2), and where in 1952 a large area was filled in deeply with earth (AEC 1964).

It is not clear who did the work, but it seems likely – and is assumed here -- that the burial, at least, was done by the same contractor(s) as were doing the demolition and removal work at the St. Louis main site. Thus it appears likely – and is assumed here -- that this work was at least overseen by AEC and Mallinckrodt health and safety people. Because they were done in the same time period, it is similarly assumed that the railroad construction and the tailings removal were performed with AEC and Mallinckrodt health and safety oversight and support. These assumptions are strengthened by the fact that at some point in about 1959, a building consisting of a changeroom, a shower, and office space was moved to the site (AEC 1959; AEC 1964).

No information is available as to the conditions under which the railroad facility installation and the tailings removal were done, but presumably these were typical construction and earthmoving processes such as had taken place previously at SLAPS. No information is available as to the conditions under which the 1959-1960 burial was done, but presumably it was a typical process of digging a hole(s) in the ground, depositing the debris, and pushing the removed soil back into the hole. As noted above, there was a building installed that included a changeroom, shower, and office. Thus it appears that normal contamination control procedures of the time were followed.

The workers might not have been badged, however, because the potential for exposure may have been viewed as too low; for example, during 1949, the crane and bulldozer operators at SLAPS (who were apparently working only with the residue piles and not with any aspect of the K-65 storage or removal operation) were not being badged because they averaged less than 8 hours per week on the raffinate heap work and the radiation level from the raffinate/residue heaps was believed to be low (MCW 1949g). If the workers were badged, however, this was undoubtedly done through the Weldon Spring site, as was done for the decontamination and decommissioning workers at the main St. Louis site. Thus it is not clear if the workers were badged but in any case the Weldon Spring film badge records are presently unavailable.

Before and after the burial and tailings removal work, it can be assumed that only caretaker functions were performed, that is, that periodically a guard would visit the site to check the locks and verify that entry into or disturbance of the site had been made. In addition, there were likely some inspection visits by AEC in the company of prospective purchasers of the residues. The AEC people may have been wearing their own badges, but there is no indication that the visitors did.

This technical basis document covers Mallinckrodt employees only, but an effort has been made to provide information about other workers on the Mallinckrodt St. Louis main site and SLAPS, for information and for efficiency of later evaluation of the other workers' exposure, if required. Thus the categories of exposed workers are assumed to include all those below, but they should not be assumed to be Mallinckrodt employees. Visitors are not included.

Note that in 1976 and 1978, Oak Ridge National Laboratory surveyed SLAPS for AEC years after the piles had been removed (ORNL 1979). They found that although much of the soil inside the SLAPS fence had terrestrial levels of U-238 and Ra-226, there were several places where the levels were elevated. The highest U-238 level was 260 pCi/g (normal range 0.3 – 1.7) and the range of elevated Ra-226 values was 1.4 – 78 pCi/g (normal range 0.3 – 1.3). Ac-227 was also found in the range of 0.5 – 77 pCi/g; the source was most likely the Sperry cake because of its concentration of Pa-231, a precursor of Ac-227, but the highest reading was found where the AM-7 was stored and some elevated readings were also said to be associated with a location near where the barium sulfate cake was stored.

8.5 CALCULATION OF DECONTAMINATION AND POST-DECONTAMINATION INTERNAL AND EXTERNAL EXPOSURES FOR THE ST. LOUIS MAIN SITE WHEN INDIVIDUAL AND SURROGATE DATA ARE LACKING

To estimate claimant-favorable doses to workers from MED/AEC contamination alone when individual data and surrogate data from records are lacking, use was made of the results of the initial and release Mallinckrodt contamination and dose rate surveys, i.e., those performed before and after the 1959-1961 decontamination (reported in MCW 1958 and MCW 1959 for before decontamination; in MCW 1961a for after). The data taken from these references and used in the calculation of exposure is shown in summary form in Table 39; it must be stressed that this represents only a small and select subset of data from larger sets of hundreds of data points.

The RESRAD-BUILD computer code (ANL 2003) was used to calculate annual exposures from inhalation of airborne particulates and radon (and its daughters). Separate hand calculations were performed to estimate annual ingestion and external exposures.

Source terms for both types of calculations were derived from the measured data shown in Table 39. Conservative maximum averages of surface and bulk (volume) contamination were used to produce the Inhalation and radon source terms for RESRAD-BUILD and the inhalation RESRAD-BUILD results were used to produce the source term for the ingestion calculations, while the maximum (or average maximum) and typical measured dose rates from gamma and beta radiation were used in the hand calculations to estimate annual external exposures.

For RESRAD-BUILD parameters other than the source term, conservative but Mallinckrodt-suitable values were used, when they could be determined; when no specific or suitable values could be determined, conservative default values given in the RESRAD-BUILD manual (ANL 2003) or other guidance documents were used. These values should be claimant-favorable.

8.5.1 <u>Assumptions Made in RESRAD-BUILD and Related Internal Exposure Calculations</u>

The principal assumptions made for the RESRAD-BUILD calculations are given below.

1. The inhalation and radon source terms were derived on the basis of the highest average surface and bulk contamination levels, respectively, in each plant, regardless of building or room location. Without regard to the actual correspondence of gamma dose rate, surface contamination, and bulk contamination levels in particular rooms, the room model (work area) was assumed to contain the highest measured average surface contamination concentration, the highest bulk contamination concentration, and the highest spot or average gamma and beta dose rates found anywhere in the respective plant. This is a claimant-favorable assumption since in fact there was no such location where all these indicators were simultaneously at the maximum for the plant. These sources are given in Table 39.

2. The room model was assumed to have the measured surface concentration over all wall and floor surfaces and to have the measured bulk concentration extend 6 inches into the walls and floors, which were taken to be concrete. This is reasonable because in MCW (1958) contamination in concrete was measured typically from 4 to 6 inches into the concrete (but much less in steel) and in MCW (1958; 1959; 1961a) the overheads were generally found to be far less contaminated than the walls and floor and so would contribute negligibly to the total.

3. Source contamination was measured as gross alpha and as either total beta and total gamma separately (MCW 1958; MCW 1959) or total beta-gamma together (MCW 1961a). Because of the difficulty of determining the degree of equilibrium of the uranium with its daughters, it was assumed for the inhalation case that there was 100% equilibrium; this appears to be reasonable because of the length of time between the cessation of operations and the beginning of intrusive decontamination operations. The source terms were then determined by assuming that the alphas were being emitted by U-238, U-234, Th-230, and Ra-226 (together yielding 97.8% of the total alpha emissions -- see Section 6.1, Item 5) and by U-235, Pa-231, and Ac-227 (together yielding 2.2% of the total alpha emissions).

4. The worker was assumed to spend his entire work time (8 hours per work day) in the room, i.e., in the most contaminated area of the given plant.

5. The worker was assumed to spend 2000 hours per year in the one location. The takedown of a building may have been on the order of weeks and decontamination on the order of months; however, continuous decontamination and demolition work over the course of a 50-week year is assumed for such workers.

6. The room size was taken to be about 10' x 20' x 10 ' high (3 m x 6 m x 3 m). There were many process areas that were larger, but they were often partitioned and they were undoubtedly decontaminated in sections. Thus assuming a smaller room would be conservative in terms of concentrating or confining the contamination in the ventilated space.

7. One air change per hour was assumed. While only limited information is available regarding the ventilation systems at Mallinckrodt, it was clear that the process areas had forced ventilation. These were apparently not always used during significantly dusty work, as suggested by MCW (1961b), but in those cases respirators were worn by the workers and enclosures were typically used as well. The enclosures had forced ventilation (MCW 1961b). Thus it is reasonable to assume that either the normal forced ventilation was used in the general area, in which case one air change per hour is a conservative rate, or vented enclosures were used, in which case the air change rate would have been far higher, the worker would have been wearing a respirator, and the calculated exposure would represent a marked overestimate of the likely actual exposure.

8. For the decontamination years case, the resuspension factor for the transferable contamination was assumed to be 1×10^{-4}; for the post-decontamination years case, it was conservatively assumed to be 1×10^{-6}. The latter value is based on NRC (2002) and the former is taken to be a conservative and thus claimant-favorable value for non-respirator work, as is consistent with the discussion and tables in the RESRAD-BUILD manual (ANL 2003).

9. The deposition (settling) velocity was taken to be 0.00075 m/sec, a reasonable value for particles of 5 μm, as shown in Figure J.3 of ANL (2003).

10. The removable fraction for the decontamination years case was assumed to be 30%, based on the fact that some early decontamination was done at the end of operations (e.g., rinsing out the process vessels and vacuuming the floors). The removable fraction for the inhalation dose calculation in the post-decontamination years case was assumed to be 10%. This should be reasonably conservative since the post-decontamination period followed an extensive decontamination. Default erosion, radon emanation, and associated values were used because they are conservative and thus claimant favorable.

11. Three types of exposure levels were used to represent the different exposure potentials of different types of workers. “High” exposure potential represents those working in the most contaminated areas, i.e., the former process areas; these would be those performing decontamination or later working substantial periods of time in the former process areas. “Moderate” exposure potential represents those accessing the less contaminated areas or infrequently accessing the former process areas; these would be those supervising the decontamination on an intermittent basis or those having only occasional need to enter the former process areas. It would also include those working in the former laboratories. “Low” exposure potential represents those who accessing the slightly contaminated or uncontaminated areas, such as the former office areas.

12. Since the inhalation and radon calculations did not depend on the position of the receptor in the room model and since the isotope proportions were taken to be the same at the beginning of the calculation, one wall of the maximally contaminated plant was modeled with the D&D sources for the surface case and similarly for the volume case; the same was done with the post-D&D sources for the surface and volume cases. The results were then ratioed to produce results for the entire wall and floor area for the various plant and exposure potential cases.

13. Because 1959-1961 data for Plants 1 and 2 was not available and because the post-D&D surface and volume data of Plants 1 and 2 were bounded by the Plants 4, 6, 6E, and/or 7 data, as shown by comparisons of the data for all the plants in ORNL (1981), it was assumed that they could be assigned the maximum post-D&D inhalation exposures for the “High” category from calculations for the other plants. Thus it was assumed that it was not necessary to create separate source terms and models for inhalation exposures for Plants 1 and 2.

14. Because 1959-1961 data for Plants 1 and 2 was not available, it was assumed to be appropriate to use the post-D&D calculated radon exposure data for the other plants. While the typical radon measurement data of Plants 1 and 2 given in ORNL (1981) were in the same range as the radon measurement data for Plants 4, 6, 6E, and/or 7 given in ORNL (1981), there was one high area reading in one building in Plant 1 and one in Plant 2 that far exceeded the other readings. Although the high reading may not be representative of the true exposure, it was assumed that the maximum calculated exposure for the year 1975 (the evaluated year closest to 1977, the year the ORNL survey was done) could be ratioed by the factor or difference seen in the ORNL (1981) data for the respective plants and the calculation. Also, due to the lack of variation in the calculated doses over time, it was assumed that the calculated 1962 post-D&D results, after ratioing, could be extrapolated back to 1959 for Plants 1 and 2 (since they had already been cleaned up and were not in D&D). Thus it was assumed that it was not necessary to create separate source terms and models for radon exposures for Plants 1 and 2.

8.5.2 Results of RESRAD-BUILD Calculations and Conversion to Dosimetrically Useful Quantities

For use in dose reconstruction, the inhalation and radon doses that were the results of computations in RESRAD-BUILD had to be converted back to activity units, in this case to pCi and WLM of intake, respectively. The RESRAD family of codes uses the dose conversion factors for inhalation given in Eckerman et al. (1988), as also listed in the RESRAD-BUILD manual (ANL 2003). The radon conversion is also from the RESRAD-BUILD manual (ANL 2003). Since the conversion factors are applied at the end of the RESRAD-BUILD calculation, it is appropriate to reverse the conversion using the factors. The converted results are given in Table 40.

Because of RESRAD-BUILD's limitation on how many yearly printouts can be made, inhalation and radon exposures were calculated for each year for the D&D years case but only for the first few years and every five years thereafter for the post-D&D years case. This is clearly appropriate since as the output data show, the values change little from year to year. So although multiple years may be indicated in the column headings in Table 40, the figures below them are for each year and are not the sum for the indicated years.

8.5.3 Assumptions for and Results of Ingestion Calculations

It was assumed that it was appropriate to use the annual inhalation intakes derived from the RESRAD-BUILD inhalation dose results in order to produce source terms for the calculation of bounding ingestion doses according to the methodology of NIOSH (2004).

1. The appropriate inhalation exposures in pCi that were calculated as in Section 8.3.2 from the RESRAD-BUILD results were converted from pCi per year to an effective air concentration in pCi/m^3, assuming a breathing rate of 1.2 m^3/hr and a 2000-hour work year.

2. The air concentration in pCi/m^3 was multiplied by 0.2 to obtain the pCi ingested per day.

3. The pCi ingested per day was multiplied by 250 work days per year to obtain the pCi ingested per year.

The results are shown in Table 40.

8.5.4 Assumptions for and Results of External Exposure Calculations

For the external exposure calculations, separate calculations were performed for gamma and beta radiation. For Plants 4, 6, 6E, and 7, the high maximum average and typical average measured dose rate values in accessible areas were used for the "High" exposure potential case; moderate maximum average and typical average values for the "Moderate" case; and low maximum average and typical average values for the "Low" exposure case. These are defined as in Item 11 of Section 8.4.1 above.

1. Because exposure (dose) rates were used, the source terms did not have to be translated into activity units. However, while the pre-decontamination survey measured values were given separately as gamma and beta dose rates (MCW 1958; MCW 1959), the final release survey gave values as combined beta-gamma dose rates (MCW 1961a). These were ratioed using the pre-decontamination data in order to produce separate gamma and beta dose rates. The resulting assumed source terms are shown in Table 39.

2. In the surveys, the measurement point for betas or mixed beta-gamma radiation was usually a contact or near-contact dose rate for both walls and floors, while for gammas, it was most often a contact or near-distance dose rate for walls and a three-foot measurement for floors. However, the gamma-only measurements with a microrem-reading detector was usually taken in the middle of each survey grid block. It was thus assumed that the measured beta dose rate and the measured mixed beta-gamma dose rate represented all-beta radiation emanating from a wall surface to a receptor point at 1 cm from a wall surface; similarly, the measured gamma dose rate was assumed to represent gamma radiation emanating from a wall surface to a receptor point at 1 meter from a wall surface.

3. A series of calculations were done to see what size of source (e.g., point, small-radium, large-radius) was most appropriate for the measured data for each type of radiation. It was found

for both beta and gamma that an large-radius source was most appropriate. For the gamma case, it was assumed that the source was of infinite radius (because it was not a very large increase from, e.g., a 4-m radius while the room could be assumed to be on the order of the room used for the RESRAD-BUILD calculation, or about 3.3 x 6.7 meters for the wall lengths). For the beta case, it was assumed that the source was of essentially infinite radius, i.e., 8.5 m, the range of the most energetic beta emitted from the uranium-daughter source mix.

4. The dose rates as per Item 2 above were used to determine the areal source strength for beta and gamma separately and then the source strengths were used to calculate the dose rates at 1 foot and 1 meter for beta and at 1 cm and 1 foot for gamma. As per NIOSH direction, the respective dose rates at 1cm were then considered to be the maximum dose rates, the dose rates at 1 foot the most likely dose rates, and the dose rates at 1 meter the minimum dose rates.

5. The receptor was assumed to stay in the maximum average location for his exposure potential for two hours per work day and in the typical average location for his exposure potential for six hours per work day, for a total time of 2000 hours per year. This ignored break time, which was usually spent in areas of very low or no contamination, such as a lunchroom.

6. For Plants 1 and 2, there was not sufficient data to create the three dose potential categories and there was data for only the one time (i.e., the 1977 ORNL survey data point (ORNL 1981)). Thus there is only a single category of dose potential and the same measured numbers are used for all the post-D&D years. Because the figures were generally mixed beta-gamma and not separate gamma and beta exposure rates, except for one specific gamma measurement, the maximum value was assumed to represent gamma and beta in turn, except for that one gamma measurement. It should be recognized that the exposures calculated in this way for Plants 1 and 2 are likely to be extremely conservative.

Once the annual gamma doses had been calculated for each case from ambient measurement data, the doses had to be converted to photon doses to individual organs. Conversion factors from Appendix B of NIOSH (2002a) were used to do this. It was assumed that for gammas, half of the dose was from photons of energy between 30 and 250 keV and half from photons of energy greater than 250 keV. A geometry of 50% AP and 50% ROT was assumed, as seems reasonable from Section 5.4.3.1 above. The results are shown in Table 41.

The beta dose rate data given in Table 39 above was used to calculate electron doses to skin, breast, and testes. To account for attenuation by clothing (see Section 7.6.3), a factor of .8 was applied in the calculation of the breast and testes doses. The results are shown in Table 42.

8.6 CALCULATION OF DECONTAMINATION AND POST-DECONTAMINATION INTERNAL AND EXTERNAL EXPOSURES FOR SLAPS WHEN INDIVIDUAL AND SURROGATE DATA ARE LACKING

There does not appear to be any work time or work practice information available for SLAPS for the period 1959-1962. There also does not appear to be available any external dose rate, radon, or dust concentration data available for SLAPS for this period. Thus work conditions must be inferred that are judged to be conservative based on indications in the reference; source terms or dose rates must be inferred from the data from other periods, chosen because the known sources of those other periods are comparable to or higher than those for 1959-1962.

The exposure modes during the removal work at SLAPS are considered to be external gamma and beta radiation, radioactive dust; and radon. The radioactive sources would have been any residues dumped on the ground in or near the work area. It is not clear which residues were dumped where on the site, thus it must be assumed that the digging for burial and removal was near the worst source. (It is assumed that burial would not have been done under the piles, as that would have required moving more material and would also have involved more health hazards.)

The worst source appears to be the barium sulfate (AJ-4) wastes. (Note that the K-65 had been removed years earlier.) First, although as Table 4 indicates there was more total uranium in the AM-7 and AM-10 raffinates -- 113 and 48 tons respectively versus the 29 tons total in the leached and unleached barium sulfate residues – the barium sulfate contained the residual radium that had not been captured in the K-65 precipitation. As Table 4 shows, this was about 1 mg per ton. The radium would contribute to both the external and radon exposures. Second, even after the removal of the wastes in the late 1960's, a beta-gamma survey showed that the area that had contained the AJ-4 residues read higher than any other area (AEC 1972), also suggesting a high radium content compared to the other two residues. Third, as Table 33 shows, the gamma dose rates were highest from the "aged" barium cake material, which would correspond to the piles found after 1958 at SLAPS. Thus the assumption is made that the calculated doses can be based on barium sulfate as the maximal residue.

8.6.1 Assumptions for Work Time and Worker Types

It is not clear how long the work burial and removal work took, but it does not appear to have been a continuous process since the generation of material during the St. Louis main site D&D would not have been continuous. Thus it is assumed that the work took place over a period of six months, evenly spaced out over 1959-1960. The excavation/removal workers were assumed to be present for 8 hours per day, while the Health and Safety workers and AEC representatives were assumed to be present for 4 hours per day. The guard is assumed to have been present for two hours a day (to open up the site and lock up, if necessary) during the work and for one hour per week during the remainder of the time from 1959-1962, i.e., during three and a half years. AEC representatives are assumed to have spent two hours per month (i.e., 12 two-hour visits) at the site during 1961-1962. The fractions of the work year (2000 hours) spent at the site are thus as shown below.

Fraction of the work year spent at SLAPS.

	1959	**1960**	**1961-1962**
Bulldozer/truck driver, crane operator	0.25	0.25	---
Other material workers (e.g., riggers)	0.25	0.25	---
Health & Safety worker	0.125	0.125	---
AEC	0.125	0.125	0.012
Guards	0.081	0.081	0.025

8.6.2 Assumptions for Radioactive Source Terms

AEC (AEC 1949m) states that the barium sulfate residue contained 0.1% uranium (by weight) and 4×10^{-9} grams of Ra-226 per gram of residue. Assuming that the other daughter isotopes of U-238 were in equilibrium with their parents and assuming a natural uranium mix, this gives:

Concentration of radionuclides in barium sulfate residue.

Isotope	Quantity, pCi/g
U-238	3.35E+02
Ra-226	3.96E+03
Th-230	3.35E+02
Ac-227	2.47E-02

For each pCi of inhalation or ingestion intake, assume:

Isotope	Quantity, pCi
U-238	7.24E-02
Ra-226	8.55E-01
Th-230	7.24E-02
Ac-227	5.34E-06

8.6.3 Assumptions for Inhalation Exposure Calculations

It is conservative to assume that the residue – which was deposited loose on the ground in piles – can be treated in calculations as though it were soil. ORNL (ORNL 1979) performed a calculation of the concentration in air due to wind alone and due to mechanical resuspension (i.e., excavation or other soil disturbances), based on measurements of the radioactive content of SLAPS soil and on measured wind speeds and directions at SLAPS. Although these results are not directly applicable since the residues had been removed by the time the measurements and the calculation were done, it appears that the assumptions made in the calculation were such that it can be scaled to the actual situation, that is, by treating the barium sulfate residue as though it resuspended in the same way as the soil when disturbed by wind or mechanical action.

Inferred values of airborne dust concentrations for barium sulfate residue piles.

Radionuclide	Reference point (ORNL ORNL 1979)			Ratios		Barium sulfate		
	Wind resuspension, pCi/m^3	Mechanical resuspension, pCi/m^3	Mean soil conc, pCi/g	Ratio, wind resupension/ mean soil conc	Ratio, mech resupens/ mean soil conc	Mean residue conc, pCi/g	Wind resuspension, pCi/m^3	Mech resuspension, pCi/m^3
Ra-226	1.00E-08	0.03	12.6	7.94E-10	2.38E-03	3.96E+03	3.14E-06	9.42E+00
Th-230	1.00E-08	0.03	8.3	1.21E-09	3.63E-03	3.35E+02	4.05E-07	1.22E+00
Ac-227	6.00E-09	0.01	8.8	6.80E-10	1.13E-03	2.47E-02	1.68E-11	2.80E-05
U-238	4.00E-08	0.1	8.3	4.83E-09	1.21E-02	3.35E+02	1.62E-06	4.05E+00

The wind suspension values in the next-to-last column are to be applied when no work is going on (e.g., when the guard or AEC is making an inspection visit) and the mechanical resuspension values in the last column are to be applied when work is going on.

As for the St. Louis main site case, the assumption is made that the worker breathing rate is 1.2 m^3/hr and that the standard work year is 2000 hours.

The results are shown in Table 43.

8.6.4 Assumptions for Ingestion Exposure Calculations

Although the method of NIOSH (2004) appears to be inappropriate for the SLAPS work situation because of the outdoor location, it is clear that the method will give a very conservative result and it is an approved method. Thus it will be applied as follows in this technical basis document until further guidance is provided.

1. The air concentration in pCi/m^3 from Section 8.6.3 was multiplied by 0.2 to obtain the pCi ingested per day.

2. The pCi ingested per day was multiplied by 250 work days per year and by the fraction of the year from Section 8.6.1 to obtain the pCi ingested per year.

The results are shown in Table 43.

8.6.5 Assumptions for and Results of Radon Exposure Calculations

No radon measurements have been found for SLAPS after 1949, i.e., after the high-radium K-65 residue was removed. The content of the barium sulfate residue, 4 x 10^{-9} g per gram of residue, is equivalent to about 4000 pCi/g of residue, as given in Section 8.6.2. This can be compared to the 1.0-1.2 pCi/g of soil that ORNL (ORNL 1979; ORNL 1981) found to be the typical Missouri soil content. Clearly the concentration of Ra-226 was elevated over background levels and thus radon emanation would be greater than in typical Missouri soils.

ORNL (1979) provides some measured data by which the radon emanation rate can be related to surface soil content of radium. The maximum onsite radon concentration measured was 1.3 x 10^{-1} pCi/l of air. The average surface soil concentration of Ra-226 for the three highest radon-emanating points was found from data in Tables 3 and 13 of ORNL (1979) to be 1.3 pCi/g. This gives a ratio of 0.1 pCi/l of radon per pCi/g of Ra-226. Thus for the assumed 3960 pCi/g in the residue (see Section 8.6.2 above), there would be a radon concentration in the air of 396 pCi/l.

Although in the outdoors environment the equilibrium factor of the radon daughters is likely to be very low, the conservative assumption will be made that the ratio is .4. The occupancy factors can be taken to be as given above. Thus we can calculate the SLAPS intakes as shown below.

Annual intake, WLM = occupancy factor x equilibrium factor x .12 WLM/yr per pCi/l
x radon concentration, pCi/l

The results are shown in Table 43.

8.6.6 Assumptions for and Results of External Exposure Calculations

The gamma dose rate of .5 mR/hr reported in MCW (MCW 1949g) for a bulldozer cab was used to calculate the exposures from the piles of barium sulfate residue. Since the piles themselves are assumed not to be disturbed, the beta dose rates are assumed to be negligible at the worker positions. Thus only the gamma radiation is assumed to contribute. Also, the bulldozer cab appeared to be appropriate because the work appeared to consist entirely of processes such as excavation, dumping from trucks, and inspection from trucks.

Once the annual gamma doses had been calculated for each worker case from the gamma dose rate, considered as ambient measurement data, the doses had to be converted to photon doses to individual organs. Conversion factors from Appendix B of NIOSH (2002a) were used to do this. It was assumed that the dose was from photons of energy between 30 and 250 keV. A geometry of 100% ROT was assumed because of the large-pile geometry.

The results are shown in Table 44.

8.7 USE OF DECONTAMINATION AND POST-DECONTAMINATION EXPOSURE DATA FOR ST. LOUIS MAIN SITE DOSE RECONSTRUCTION

As stated above, where urinalysis and film badge data are available for an individual, they should be used to determine the internal and external exposure to the individual; the methodology of Sections 6.0 and 7.0 respectively should be used as guidance for this, as appropriate. Also, for the D&D years it should be assumed that a worker had an annual x-ray (see Sections 8.1 and 7.4 above for the rationale and for guidance, respectively). It should be remembered that as noted in Section 8.1 above, some AEC and Mallinckrodt personnel film badge records for this period may be found in the Weldon Springs site set of records (and not in the Destrehan-St. Louis site set of records) because their badges were issued from Weldon Springs.

Where there is missing dose data in records, the dose reconstructor should identify surrogate worker data from records available in the dose reconstruction project, based on similarity of job title and location. Surrogate data are not provided in this TBD for this time period because of the low number of likely claimants requiring it and because records have not been located. Caution must be taken in the case of workers whose work histories show them apparently still doing work with radioactive materials after the decontamination period, since (1) such work during the post-decontamination years was not AEC-contracted work, (2) it is uncertain what work may have been done and what materials may have been used, and (3) there may be no available records covering these years.

Where appropriate surrogate data cannot be found for the decontamination years case, the data of Table 37 may be used to provide surrogate data on a group basis. A conservative and claimant-favorable value should be assumed. In particular, the default assumption should be “High” exposure, where that choice is available. Where gaps still exist, reference can then be made to the data given in Tables 40-42.

For the post-decontamination years case, where individual surrogate data cannot be found for the post-decontamination years case, as is likely, the data of Tables 40-42 should be used. The default assumption should be “High” exposure, where that choice is available.

In both cases, in dose reconstruction using Tables 40-42, the applicable years of employment for each indicated period should be determined and then the number of years should be multiplied by the annual value.

With regard to determining the exposure potential category in Tables 40-42, it should most often be possible to determine the degree of exposure of a worker by his job title, e.g., if a worker was a secretary, then the worker should be assigned to the “low” exposure potential category. In cases where it is not possible to determine the category, then the “moderate” or “high” category should be used.

8.8 USE OF DECONTAMINATION AND POST-DECONTAMINATION EXPOSURE DATA FOR SLAPS DOSE RECONSTRUCTION

As stated earlier, where urinalysis and film badge data are available for an individual, they should be used to determine the internal and external exposure to the individual; the methodology of Sections 6.0 and 7.0 respectively should be used as guidance for this, as appropriate. If a worker was badged or given urinalysis during the years 1959-1961, it should be assumed that a worker had an annual x-ray (since if the SLAPS workers were involved in the decontamination and demolition operations, whether as Mallinckrodt workers or as contractor workers) this would have been the case. It should be remembered that as noted in Section 8.1 above, some AEC and Mallinckrodt personnel film badge records for this period may be found in the Weldon Springs site set of records (and not in the Destrehan-St. Louis site set of records) because their badges were issued from Weldon Springs; this may be the case as well for the D&D contractor workers.

Where there is missing dose data in records, the dose reconstructor should identify surrogate worker data from records available in the dose reconstruction project, based on similarity of job title and location. Surrogate data are not provided in this TBD for this time period because of the low number of likely claimants requiring it and because records have not yet been located.

Where appropriate surrogate data cannot be found for the years 1959-1962, as is likely, the data of Tables 43-44 should be used to determine conservative annual intakes and external doses. In determining both intakes and external doses, the applicable years of employment for each indicated period should be determined and then the number of years should be multiplied by the annual value. For the intakes, the isotopic breakdown should be assumed to be as given in Section 8.6.2. The default assumption should be "High" exposure, where that choice is available.

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