ORAU Team Dose Reconstruction Project for NIOSH Technical Basis Document: Basis for Development of an Exposure Matrix for Blockson Chemical Company	Document Number: ORAUT-TKBS-0002 Effective Date: 10/10/2003 Revision No.: 00 Controlled Copy No.: _____ Page 1 of 21
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Basis for Development of an Exposure Matrix for
BLOCKSON CHEMICAL COMPANY
Joliet, Illinois
Period of Operation: August 15, 1952 through March 1962

1.0 Site Description, Operational History, and Process

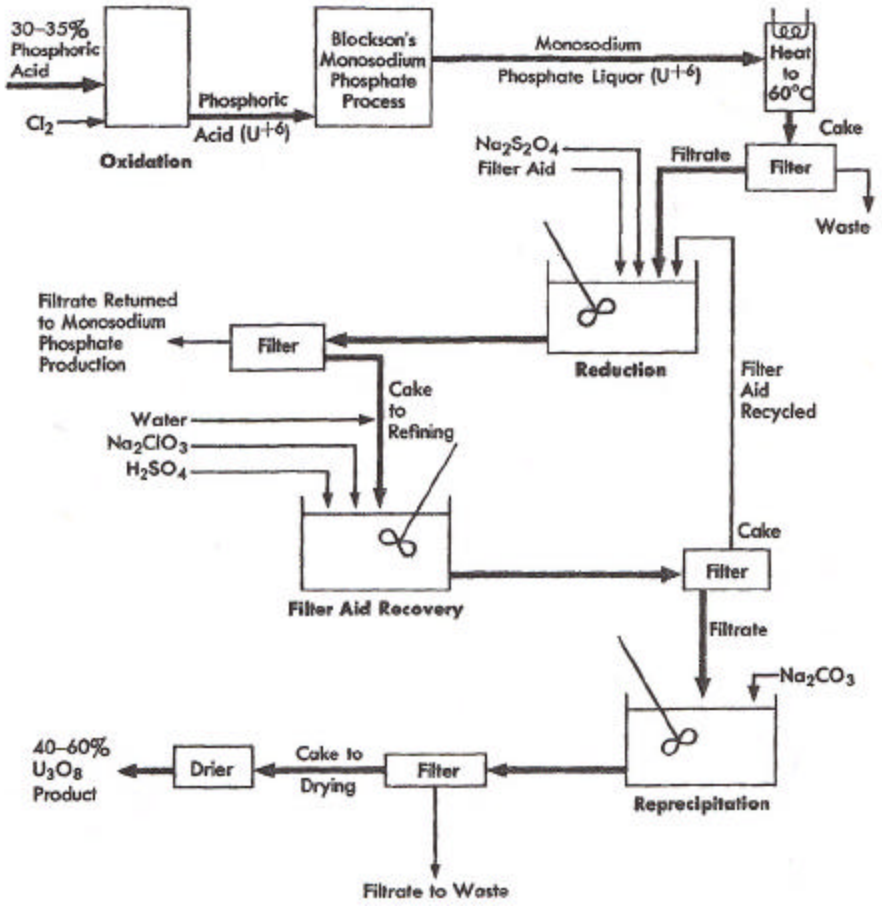
In 1951, the U.S. Atomic Energy Commission (AEC) contracted with the former Blockson Chemical Company (contract number AT(49-1)-606) to develop a process to extract uranium from wet phosphoric acid (US DOE 1985, US DOE 2002). This letter contract was later replaced by another contract (contract number AT(49-1)-611) that was signed October 15, 1951. Under this second contract, Blockson constructed, at its own expense, a facility (Building 55) to house uranium recovery equipment at their plant in Joliet, Illinois. The AEC furnished and installed the uranium recovery equipment (US DOE 1985). On August 15, 1952, Blockson began production and delivery of uranium concentrates to the AEC (Stolz, Jr. 1958). According to the contract, production was limited to not more than 50,000 pounds of uranium per year (US DOE 1983, 1985).

In 1955, Blockson was sold to the Olin Mathieson Chemical Corporation who assumed the liabilities and obligations of Blockson under all contracts, as stated in contract number AT(49-1)-611 Amendment 1. The Olin Corporation continued the uranium recovery program under contract with the AEC. The actual amount of uranium produced for the AEC is not known. In March 1962, the uranium recovery work was discontinued with the expiration of the contract (US DOE 1985).

According to the contract signed in October of 1951, Blockson, and later Olin Mathieson, was responsible for the health and safety of the employees at the site and for conforming to AEC health and safety regulations and requirements. In Amendment 3, effective January 1, 1958, this statement was deleted. Personnel with the Formerly Utilized Sites Remedial Action Program (FUSRAP) conducted records searches for information regarding the uranium recovery activities at Blockson. No records of health and safety inspections by the AEC were found as a result of their search, although there was evidence of periodic visits by AEC personnel to review and audit process operations (US DOE 1985).

The recovery plant was put into operation on August 15, 1952. The process was patented and the patent, USP 2743156, was assigned to the AEC (Stolz, Jr. 1958). A one-story, 100-by-175-foot building was built specifically to house the uranium recovery process (US DOE 1983, US DOE 1985). The recovery plant was designed to be capable of recovering uranium from 1500 tons of phosphate daily (Stolz, Jr. 1958). Figure 1 shows the schematic flowchart of the Blockson process for the recovery of uranium from wet phosphoric acid.

Figure 1. The Blockson process for the recovery of uranium from phosphoric acid. [Reproduced from Clegg and Foley 1958].



The Blockson Chemical Company manufactured wet-process phosphoric acid from Florida phosphate rock (Barr et al. 1955, Clegg and Foley 1958). The Blockson plant produced technical phosphates rather than fertilizers from wet phosphoric acid (Wilkinson 1976). In the process, the phosphate rock is calcined and then digested with sulfuric acid resulting in phosphogypsum and phosphoric acid. The phosphogypsum partitions most of the calcium and radium, and the phosphoric acid partitions around 90% of the uranium. Very little uranium is lost to the phosphogypsum. The phosphoric acid is then converted into monosodium phosphate and other phosphorus derivatives. The uranium by-product is precipitated from the monosodium phosphate stream. The monosodium phosphate liquor is heated and clarified. Sodium hydrosulfite ($Na_2S_2O_4$) is added to precipitate the uranium. The liquor is filtered and the filtrate is returned to the phosphate-processing plant. The precipitate, containing about 5% U_3O_8 is slurried in water in an upgrading step in which the uranium is redissolved. The uranium is then reprecipitated as sodium uranous phosphate. The slurry is filtered and the precipitate, known as yellowcake and containing 40 to 60% U_3O_8, is dried for shipping (Clegg and Foley 1958, McGinley 2002, Wimpfen 2002). The uranium content of the phosphate rock consumed in these processes averaged about 0.014% U_3O_8 (Stolz, Jr. 1958).

2.0 Estimation of Internal Exposure

The greatest potential for internal exposure associated with the uranium recovery process arises in the final packing areas. Here the essentially pure uranium compound is dried and barreled for shipping resulting in a potentially dusty operation (Eidson and Damon 1984, US NRC 2002b, Wimpfen 2002). In all other areas of the plant, wet processes are used and the surface contamination and dust exposures are minimal (Clegg and Foley 1958, US NRC 2002b).

A study was done (Eidson and Damon 1984) of uranium aerosols generated during yellowcake packaging operations at four uranium mills. The study described a sequence of steps common to all four uranium mills:

1. No activity. This is when the plant is shut down for maintenance or all available yellowcake was packaged during a previous shift. Workers are generally not present during this step.

2. Drum loading. This occurs when a drum is placed under a hopper containing the dried yellowcake. The yellowcake is allowed to fall into the drum. The amount of time workers spend in this area varies as it depends on the size of the yellowcake inventory in the hopper. (It is not clear whether or not a hopper was used at the Blockson Chemical facility.)

3. Drum uncovering. This step occurs when a filled drum is removed from beneath the hopper. In some cases, the drum may be vibrated to compact the yellowcake before uncovering.

4. Powder sampling. This occurs when a worker takes a sample of yellowcake to analyze for moisture content.

5. Lid sealing. This occurs when a worker places a lid on the drum and seals it.

6. Other activities. This step includes maintenance and hosing area and equipment with water to clean. Hosing the packaging area to clean is a routine operation at uranium mills.

During the study, aerosol samples were taken in yellowcake packaging areas before, during, and after drums of yellowcake were filled and sealed. Median aerosol concentrations during the study ranged from 40 to 340 μg U/m^3. Results from analysis of the air samples showed that appreciable amounts of airborne uranium would be expected to deposit in the nasopharyngeal region of the respiratory tract if inhaled by a worker not wearing respiratory protection (Eidson and Damon 1984).

In order to estimate worker exposure at the Blockson uranium recovery facility, the total amount of U_3O_8 produced from 1952 to 1962 was estimated. A report showed that by the end of 1955, Blockson produced 1,221,470 pounds of uranium concentrate (US DOE 1985), which is roughly 600 pounds of U_3O_8 per day assuming a U_3O_8 concentration of 60%. However, the Blockson process was designed to process only 1500 tons of phosphate daily, which is approximately 400 pounds of U_3O_8 per day. Another document indicates that production was limited by contract to not more than 50,000 pounds of U_3O_8 per year (US DOE 1983). Production was stopped in March 1962 (US DOE 1985). To estimate the source term, it was assumed that production was limited by contract to 50,000 pounds of U_3O_8 per year. Assuming the same rate of production in 1952 and 1962 as in 1953 through 1961, the amount of U_3O_8 produced is estimated at 18,900 and 12,300 pounds in 1952 and 1962, respectively. These annual estimated production values were used to calculate the amount of U_3O_8 produced per day shown in Table 1.

Table 1. Calculation of the quantity of U_3O_8 aerosolized per day from the estimated amount of yellowcake produced each year.

Work Year	Number Of Days Operated per Year	Pounds of U_3O_8 Produced Annually (lbs)	Pounds of U_3O_8 Produced per Day (lbs/day)	Quantity of U_3O_8 Aerosolized per Day (lbs/day)
8/15/1952-12/31/1952	138	18,900	137	1.32E-04
1953-1961	365	50,000	137	1.32E-04
1/1/1962-3/31/1962	90	12,300	137	1.32E-04

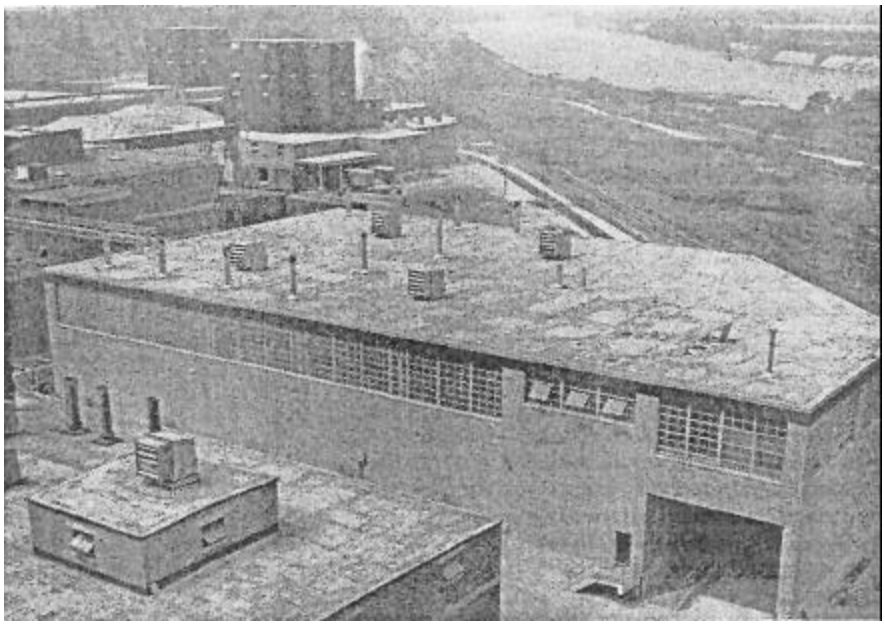
Thus, the estimated total amount of U_3O_8 produced at Blockson from 1952 to 1962 was 480,000 pounds, which is approximately 800,000-1,200,000 pounds of uranium concentrate (for a U_3O_8 concentration of 60 to 40%).

Table 2 shows the quantities used to calculate the daily intake by inhalation of natural uranium based on the estimated source term. The daily concentration of uranium in the air at the Blockson plant was calculated by assuming that 9.6E-07 (US NRC 2002c) of the U_3O_8 produced per day was aerosolized. This value was divided by the estimated interior volume of Building 55. The volume of the building was estimated by using the reported dimensions of 100-by-175 feet (US DOE 1983). Although Building 55 was reported to be a one-story building (US DOE 1985), drawings of the building that were used to show radiological survey locations indicate that there were four levels. The first level contained a loading dock and storage room, the second level contained change rooms and a lab, the third where there was a soundproof booth, and the fourth level which consisted of catwalks that allowed access to various parts of the process equipment (US DOE 1983). The interior height of the building was assumed to be 30 feet: 8 feet for each of the first two levels and 7 feet each for the third and fourth levels. Based on the radiological survey report (US DOE 1983) and the photograph shown below in Figure 2, this is a reasonable assumption. The estimated volume of the building was reduced by 10% to account for equipment displacement. This gives an estimate of 13,400 m^3 for the interior volume of the Building 55.

Table 2. Estimation of the daily intake of natural uranium based on source term.

Quantity of Natural U in Air per day (μg)	5.1E+04
Natural U Air Concentration Per day ($\mu g/m^3$)	3.8
Activity of Air Concentration (pCi/m^3)	2. 6
Breathing Rate (m^3/h)	1.2
Daily Intake (pCi/d)	25

Figure 2. Photograph of Building 55 of the Blockson Chemical Company. [Reproduced from Barr et al. 1955].



The daily air concentration of uranium activity was multiplied by the breathing rate for adult light workers to obtain an estimated intake of 25 pCi per day of natural uranium due to inhalation. The breathing rate was calculated from the volume of air breathed by an adult light worker shown in Table 6 on pg. 23 of ICRP Publication 66 (ICRP 1994). The light worker category assumes an activity of 1/3 sitting and 2/3 light exercise.

Although no air monitoring data were found for the Blockson facility, urinalysis data for 25 workers were found for the period between 4/20/1954 and 2/20/1958. Urine samples were received by the AEC on 10 different dates. The number of samples that were analyzed for each worker varied from 1 to 10 with values ranging between 0.000 to 0.017 mg/L. The Health and Safety Division of the AEC New York Operation Office performed the analyses. The method of analysis was the fluorometric method, which had a detection limit during that time period of 0.0038 mg/L (Wilson 1958). These data were used to fit intakes for each of the workers using the IMBA-Expert program (Birchall 2002). The data were fitted using an assumed relative error of 30%. A chronic inhalation intake from 08/15/1952 to 03/31/1962 was assumed for each worker. The material was assumed to be Absorption Type M (ICRP 1995) and ICRP 66 default parameters were used in the IMBA-EXPERT program (Birchall 2002) to calculate intakes. The resulting calculated chronic intake rates were lognormally distributed with a median of 24 pCi/d and a geometric standard deviation of 1.6.

The calculated chronic intake rate of 24 pCi/d is used to estimate internal organ dose for workers with no monitoring records. The annual dose for the organ of interest should be calculated in the IMBA NIOSH program assuming exposure to a natural uranium mixture. The start and end dates for the chronic intake should encompass the period of time the employee worked during the potential exposure period of 8/15/1952 to 03/31/1962. The annual organ doses can then be entered into the NIOSH IREP program as the annual dose due to chronic exposure to alpha radiation using a lognormal distribution with a geometric standard deviation (GSD) of 1.6.

Considerable variation in the behavior of U_3O_8 has been observed with some studies indicating Absorption Type M and other studies indicating Absorption Type S. The ICRP in Publication 71 recommends the use of Absorption Type M in the absence of specific information (ICRP 1995). The application of Absorption Type M in intake calculations using urinalysis data resulted in a daily intake value consistent with the value estimated from the source term. The ICRP Publication 66 (ICRP 1994) default values should be used for the deposition parameters in the IMBA NIOSH program.

It is interesting to compare the measured uranium air concentrations obtained in the Eidson and Damon (1984) study with the air concentrations calculated for Building 55 from urinalysis data and source term estimates. Note that the U_3O_8 production rate at Blockson was significantly lower than the production rates at the uranium mills in the study. Production rates at conventional uranium mills average around 4000 pounds of U_3O_8 per day (Eisenbud 1987, US DOE 1997). The Blockson process was designed to produce only about 400 pounds of U_3O_8 per day (Stolz, Jr. 1958), which is only about 10% of the average production capacity of conventional mills. In the uranium mill study, aerosol samples were taken in yellowcake packaging areas before, during, and after drums of yellowcake were filled and sealed. Four drums (containing approximately 1000 pounds of yellowcake) were loaded in succession and sealed. Powder samples were taken from each before they were sealed and aerosols generated by yellowcake sampling were sampled. An additional set of aerosol samples were taken during the drum-sealing step. To obtain aerosol samples during the drum-loading step, two of the mills loaded more than one drum simultaneously but at different rates. Thus, assuming there is a direct relationship between U_3O_8 production rate and uranium air concentration, it is possible to make a rough comparison. Table 3 compares the uranium air concentration in Building 55 to uranium air concentrations measured during the uranium mills study.

Table 3. Comparison of Blockson Building 55 uranium air concentrations calculated from urinalysis results and source term estimates to air concentrations measured in uranium mills.

	Estimated Intake Rate (pCi/day)	**Uranium Air Concentration (pCi/m^3)**	**Uranium Air Concentration (mg U/m^3)**
From urinalysis results	24	2.5	3.6
From source term estimates	25	2.6	3.8
Uranium mills study (Eidson and Damon 1984)			40-340
Uranium mills study (reduced by a factor of 10 to compare to Blockson design rates)			4-34
Uranium mills study (reduced by a factor of 25 to compare to estimated Blockson production rates)			1.6-14

3.0 Estimation of Radon Exposures

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4.0 Estimation of External Exposure

The primary radionuclides of interest for potential external exposure in Building 55 are U-238 and daughter radionuclides Th-234 and Pa-234m. The uranium recovery process at Blockson was a by-product process that was designed to fit into the already existent phosphate process (Stolz, Jr. 1958). At the Blockson facility, a side-stream of the phosphoric acid was diverted to Building 55 where the uranium was separated (Wimpfen 2002). This phosphoric acid was an intermediate product in Blockson's normal commercial production of technical phosphates (US DOE 1983). In the manufacture of this phosphoric acid, phosphate rock is digested with sulfuric acid resulting in phosphoric acid and phosphogypsum. The uranium remains with the phosphoric acid and the radium preferentially follows the phosphogypsum (Roessler et al. 1979, Laiche and Scott 1991). Therefore, the potential radium exposure was due to the commercial operation already in progress at Blockson and not due to the AEC-related work. A radiological survey of Building 55 that was done in 1978 showed that contamination within the building was primarily uranium (US DOE 1983).

For the purpose of dose reconstruction, it is assumed that there was a potential for external exposure from four sources: submersion in air contaminated with yellowcake dust, exposure from contaminated surfaces, exposure from contaminated skin, and exposure to drums of yellowcake.

For estimating external exposure due to submersion in air contaminated with yellowcake dust, the estimated air concentration values (Table 3, from urinalysis results) and an assumed 2000-hr work year were combined with dose coefficients for U-238 and daughter radionuclides Th-234 and Pa-234m from Federal Guidance Report No. 12 (US EPA 1993). Table 4 shows external annual organ dose estimates due to submersion of workers in air contaminated with yellowcake dust. The cumulative dose from 1952 through 1962 is less than 1 mrem and is therefore not included in the dose estimation.

Table 4. Annual organ doses due to submersion in air contaminated with yellowcake dust.

Organ	Annual Organ Dose (rem)
Adrenal	5.17E-08
U Bladder	5.32E-08
Bone Surface	1.55E-07
Brain	6.73E-08
Breast	7.97E-08
Esophagus	5.06E-08
Stomach Wall	5.71E-08
Small Intestine	4.98E-08
Upper Large Intestine Wall	5.21E-08
Lower Large Intestine Wall	5.07E-08
Kidney	5.83E-08
Liver	5.80E-08
Lung	6.53E-08
Muscle	6.46E-08
Ovaries	4.95E-08
Pancreas	4.88E-08
Red Bone Marrow	6.03E-08
Skin	3.64E-06
Spleen	5.81E-08
Testes	6.86E-08
Thymus	6.15E-08
Thyroid	6.85E-08
Uterus	4.84E-08

Clegg and Foley (1958) state that freshly separated yellowcake has a very low gamma emission rate; therefore, external radiation is of no particular concern at this stage of the process. However, due to ingrowth of daughter radionuclides in the yellowcake, the radiation levels increase for several months following production (US NRC 2002b).

For accumulations of processed yellowcake dust, the surface beta dose rate from U-238 daughters is negligible just after separation, but rises steadily until Pa-234m and Th-234 reach equilibrium concentrations. After a few months, the beta surface dose rate is about 150 mrem/hr (US NRC 2002b). Figure 3 shows the rise in beta dose rate during 100 days after separation from ore.

Figure 3. Beta dose rate on the surface of yellowcake. [Reproduced from US NRC 2002b]

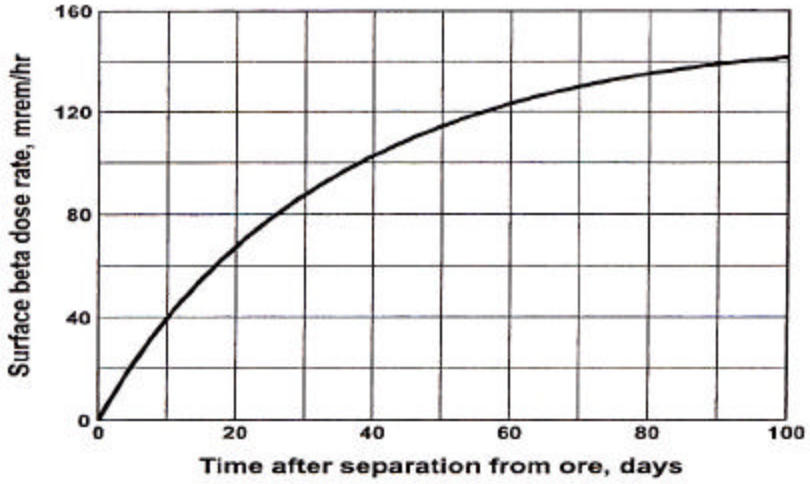

Figure 4. Beta dose rate from yellowcake separated from ore for more than 100 days as a function of distance from the surface. [Reproduced from US NRC 2002b]

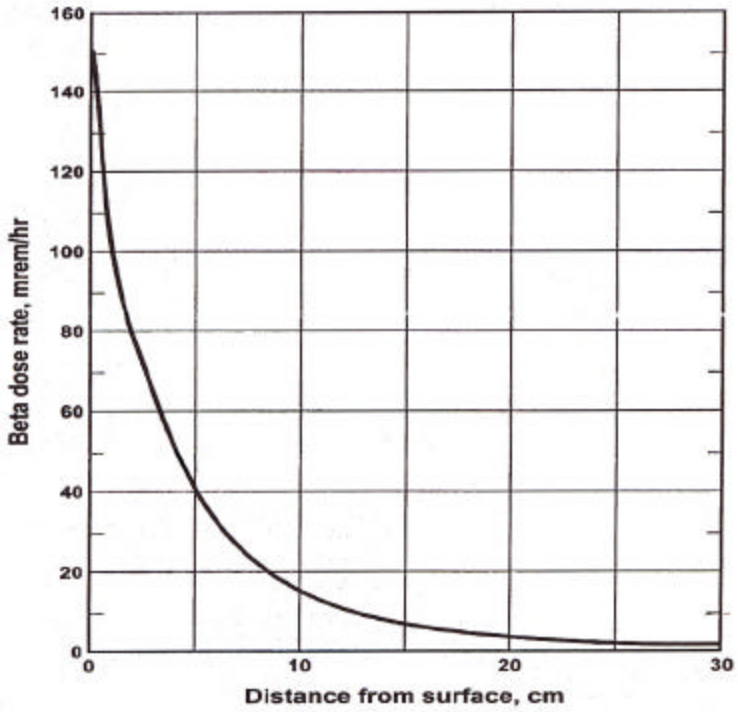

Figure 4 above shows that the beta dose rate from the surface of yellowcake decreases rapidly as a function of distance from the surface. The rapid decrease in the beta dose rate with distance, and the shielding afforded by shoes and clothing, reduces dose from electron exposure, particularly from yellowcake deposited on floors.

The most likely possibility of external exposure from surface contamination was assumed to occur in the yellowcake packaging area. Because the AEC had strict material accountability procedures, the accumulation of process material was likely controlled. However, to be claimant favorable, it was assumed that a certain amount of yellowcake was allowed to build up between cleanings.

To estimate the quantity of yellowcake contamination on surfaces, the air concentration determined from urinalysis results was multiplied by the indoor deposition velocity and the assumed deposition time. The indoor deposition velocity is dependent on the physical properties of the room (air viscosity and density, turbulence, thermal gradients, surface geometry) and the particles (diameter, shape, density). Because these characteristics are unknown, the terminal settling velocity was calculated for an aerosol with the ICRP 66 default particle size of 5 μm activity mean aerodynamic diameter (AMAD) (ICRP 1994). The calculated terminal settling velocity of 0.00075 m/s was used as an estimate of the velocity of deposition to surfaces in the building. This value is within the range of deposition velocities (2.7E-06 to 2.7E-03 m/s) measured in various studies (US NRC 2002a) and is considered claimant-favorable. Also, room air exchange rates, ventilation, and plant housekeeping practices are unknown so it was assumed that there was a steady state air concentration and that surface contamination was the result of 365 days (1 year) of settling.

The estimated surface contamination is multiplied by the dose coefficients for U-238 and daughter radionuclides Th-234 and Pa-234m for contaminated ground surface from Federal Guidance Report No. 12 (US EPA 1993). Table 7 shows maximum external dose estimates due to exposure to ground surface contamination.

Table 5. Annual organ doses due to exposure to ground surface contamination. Bold italics indicate annual dose greater than 1 mrem.

Organ	**Annual Dose (rem)**
Adrenal	1.2E-05
U Bladder	1.3E-05
Bone Surface	3.3E-05
Brain	1.2E-05
Breast	1.6E-05
Esophagus	1.1E-05
Stomach Wall	1.3E-05
Small Intestine	1.2E-05
Upper Large Intestine Wall	1.2E-05
Lower Large Intestine Wall	1.2E-05
Kidney	1.3E-05
Liver	1.3E-05
Lung	1.3E-05
Muscle	1.6E-05
Ovaries	1.2E-05
Pancreas	1.1E-05
Red Bone Marrow	1.3E-05
Skin	***5.9E-03***
Spleen	1.3E-05
Testes	1.6E-05
Thymus	1.3E-05
Thyroid	1.4E-05
Uterus	1.2E-05

With the exception of dose to the skin, the annual organ dose for each of the organs is less than 1 mrem. These values are significantly lower than the some of the exposure rates measured in the 1978 survey (US DOE 1983). According to this survey, the median external exposure rate at 1 meter was 0.03 mR/h with a maximum of 0.3 mR/h. Therefore, to estimate potential external exposure to contaminated surfaces in the plant, the median exposure rate was multiplied by the Exposure (R) to Organ Dose (rem) photon dose conversion factors from Appendix B of the NIOSH External Dose Reconstruction Implementation Guideline (DHHS 2002). The exposure geometry was assumed to be

isotropic and the exposure rate was divided evenly between the conversion factors for photons with energy between 30 and 250 keV and photons with energy greater than 250 keV. Table 6 shows the calculated annual organ doses from exposure to contaminated surfaces during plant operations.

Table 6. Annual organ doses due to exposure to contaminated surfaces.

	Annual Organ Dose (rem)		
Organ	**Photons E=30-250 keV**	**Photons E>250 keV**	**Total**
Bladder	1.61E-02	1.94E-02	3.55E-02
Red Bone Marrow	1.67E-02	2.00E-02	3.67E-02
Bone Surface	2.81E-02	2.04E-02	4.86E-02
Breast	2.12E-02	2.22E-02	4.35E-02
Colon	1.55E-02	1.90E-02	3.45E-02
Esophagus	1.50E-02	1.96E-02	3.46E-02
Eye	2.23E-02	2.28E-02	4.50E-02
Ovaries	1.48E-02	1.88E-02	3.36E-02
Testes	1.90E-02	2.08E-02	3.98E-02
Liver	1.70E-02	2.00E-02	3.70E-02
Lung	1.88E-02	2.12E-02	3.99E-02
Remainder organs	1.67E-02	1.99E-02	3.65E-02
Skin	2.19E-02	2.28E-02	4.47E-02
Stomach	1.70E-02	1.99E-02	3.69E-02
Thymus	1.84E-02	2.05E-02	3.89E-02
Thyroid	1.92E-02	2.14E-02	4.06E-02
Uterus	1.46E-02	1.81E-02	3.27E-02

The organ doses in the second and third columns of Table 6 are entered into the NIOSH IREP program assuming a chronic exposure and a lognormal distribution with a GSD of 4.0. The organ doses in the second column are attributed to photons with E=30-250 keV and the organ doses in the third column are attributed to photons with E>250 keV.

It was also assumed that there was a potential to receive a shallow dose from electrons due to skin contaminated with yellowcake. The amount of skin contamination was calculated by using a measured deposition velocity for 4-μm particles to skin of 0.012 m/s (Andersson et al. 2002, Fogh et al. 1999). For simplification, it was assumed that the material deposited on the skin during an 8-hour period was deposited at the beginning of the shift and the worker took a shower at the end of the shift. The estimated amount of skin contamination was combined with electron dose-rate conversion factors for U-238 and daughter radionuclides Th-234 and Pa-234m for skin in contact with radionuclides (Kocher and Eckerman 1987). The worker was assumed to receive exposure from skin contamination only during the hours worked. Based on these assumptions, the annual dose to the skin due to electron exposure from skin contaminated with yellowcake is estimated to be 0.0018 rem. However, this skin dose is negligible compared to the shallow dose estimated from exposure to a drum of aged yellowcake. This scenario is described next.

There was also the potential for exposure to drums of yellowcake during drum loading, sealing and sampling, and moving the drums to storage. It was assumed that 50 drums of yellowcake were loaded and packed each year (1000 pounds per drum, 50,000 pounds per year). MicroShield© and MCNP calculations were done to estimate the exposure to a drum of yellowcake at the surface of the drum, at 30 cm (1 ft), and at 1 m. Also, NIOSH/OCAS provided results of survey measurements of partially filled drums of UF_4 at the DOE facility at Fernald. Measurements were taken at the sides of the drum at the center and bottom. The mean measurements for the center and the bottom of the

drum were averaged together to get a dose rate of 1.3 mrem/h at the surface. To get an estimate of the dose rate at 1 foot from the UF_4 drums, the surface dose rate was divided by the average ratio of the surface to 1 foot calculated dose rates obtained with MicroShield and MCNP. Table 7 shows the results of the calculations for the yellowcake drums and the UF_4 drums.

Table 7. Results of calculations of the exposure rate from a drum of yellowcake and a drum of UF_4.

		Exposure Rate	
	Surface (side)	**30 cm (1 ft)**	**1 m**
MicroShield (mR/h)	5.5E-01	8.4E-02	2.2E-02
MCNP (mrem/h)	5.6E-01	1.3E-01	3.6E-02
UF_4 (mrem/h)	1.3E+00	2.4E-01	

The UF_4 values were larger and, to be conservative, were used to estimate the annual dose. During an interview, a claimant stated that he spent 8 hours per day, 1 or 2 days per week loading drums onto trucks and boxcars. Thus, to be claimant favorable in estimating the most likely annual dose, it was assumed the worker was 1 foot from the drum of UF_4 for 8 hours per day, 1 day per week, and 50 weeks per year. It was assumed that the amount of time spent loading barrels was lognormally distributed, and the assumption that the worker was exposed 40 hours per week for 50 weeks per year was considered to be the upper 95th percentile. Thus, the annual dose due to exposure to drums of UF_4 (analog for yellowcake) was calculated to be 0.096 rem.

The organ doses were calculated by multiplying the estimated annual dose of 0.096 rem by the "Ambient Dose Equivalent (H*(10)) to Organ Dose (H_T)" photon dose conversion factors found in Appendix B of the NIOSH External Dose Reconstruction Implementation Guideline (DHHS 2002). The exposure geometry was assumed to be anterior-posterior (AP) and the dose rate was divided equally between photons with E=30-250 keV and photons with E>250 keV. Table 8 below shows the annual organ doses due to the potential exposure to drums of yellowcake.

Table 8. Annual organ doses due to exposure to drums of yellowcake.

	Annual Organ Dose (rem)		
Organ	**Photons E=30-250 keV**	**Photons E>250 keV**	**Total**
Bladder	4.51E-02	4.37E-02	8.88E-02
Red Bone Marrow	2.30E-02	3.58E-02	5.88E-02
Bone Surface	4.39E-02	3.80E-02	8.19E-02
Breast	4.61E-02	4.64E-02	9.24E-02
Colon	3.85E-02	4.19E-02	8.04E-02
Esophagus	2.51E-02	3.70E-02	6.21E-02
Eye	4.54E-02	4.36E-02	8.89E-02
Ovaries	3.48E-02	4.07E-02	7.56E-02
Testes	5.23E-02	4.68E-02	9.91E-02
Liver	3.86E-02	4.24E-02	8.11E-02
Lung	3.60E-02	4.16E-02	7.75E-02
Remainder organs	3.21E-02	3.91E-02	7.11E-02
Skin	3.25E-02	4.14E-02	7.39E-02
Stomach	4.56E-02	4.40E-02	8.96E-02
Thymus	5.11E-02	4.43E-02	9.54E-02
Thyroid	5.24E-02	4.82E-02	1.01E-01
Uterus	3.67E-02	3.89E-02	7.56E-02

The organ doses in the second and third columns of Table 8 are entered into the NIOSH IREP program assuming a chronic exposure and a lognormal distribution with a GSD of 2.7. The organ doses in the second column are attributed to photons with E=30-250 keV and the organ doses in the third column are attributed to photons with E>250 keV.

It was assumed that there was a potential to receive a shallow dose from exposure to open drums during drum loading and sealing. According to Figure 4, the dose rate at 1 foot from the surface of aged yellowcake is between 1 and 2 mrem/h. Therefore, to be claimant-favorable, it was assumed that the claimant spent 8 hours per week, 50 weeks per year at 1 foot from the surface of aged yellowcake at a dose rate of 2 mrem/h. Again, the time of exposure was assumed to be lognormally distributed and an exposure time of 40 hours per week, 50 weeks per year was assumed to be the 95th percentile. Thus, the annual shallow dose from exposure to open drums of yellowcake is assumed to be 0.8 rem per year with a GSD of 2.7.

The assumption was also made that workers received an annual occupationally related diagnostic x-ray. The exposure geometry was assumed to be posterior-anterior (PA) (DHHS 2002). The air kerma at skin entrance for the diagnostic chest x-ray was estimated to be 0.108 R (Scalsky 2003). The organ doses were calculated by multiplying the air kerma by the "Kerma (K_a) to Organ Dose (H_T)" photon dose conversion factors for 30-250 keV photons found in Appendix B of the NIOSH External Dose Reconstruction Implementation Guideline (DHHS 2002). Table 9 below shows the annual organ doses due to the assumed annual diagnostic chest x-ray. The values in Table 6 are entered into the NIOSH-IREP program as the annual dose due to an acute exposure to photons (E=30-250 keV). The distribution is assumed to be normal with a standard deviation of 30%.

Table 9. Annual organ doses due to the assumed annual diagnostic chest x-ray.

Organ	**Annual Dose (rem)**
Bladder	0.074
Red Bone Marrow	0.124
Bone Surface	0.178
Breast	0.059
Colon	0.095
Esophagus	0.105
Eye	0.022
Ovaries	0.110
Testes	0.061
Liver	0.101
Lung	0.133
Remainder organs	0.110
Skin	0.110
Stomach	0.077
Thymus	0.048
Thyroid	0.052
Uterus	0.096

5.0. Estimation of Exposure to Residual Activity

After conclusion of the AEC activities in Building 55 in March of 1962, the building continued to be used for chemical processing and production of phosphate products from phosphate rock (US DOE 1983). Prior to the 1978 survey by Argonne National Laboratory, there were no records of any radiological surveys or decontamination activities at the site. The results of the 1978 survey showed

that thirty-three localized areas and three larger general areas exceeded allowable limits for uranium and radium-226. In 15 of those locations, contamination was determined to be removable and available for transfer to other areas. Thus, dose due to exposure to residual activity is estimated for the purpose of dose reconstruction.

According to this survey, the median external exposure rate at 1 meter was 0.03 mR/h with a maximum of 0.3 mR/h. Therefore, to estimate potential external exposure to contaminated surfaces in the plant, the median exposure rate was multiplied by the Exposure (R) to Organ Dose (rem) photon dose conversion factors from Appendix B of the NIOSH External Dose Reconstruction Implementation Guideline (DHHS 2002). The exposure geometry was assumed to be isotropic and the exposure rate was divided evenly between the conversion factors for photons with energy between 30 and 250 keV and photons with energy greater than 250 keV. Table 10 shows the calculated annual organ doses from external exposure to residual radioactivity after the end of AEC operations at the site.

Table 10. Annual organ doses due to external exposure to residual radioactivity.

	Annual Organ Dose (rem)		
Organ	**Photons E=30-250 keV**	**Photons E>250 keV**	**Total**
Bladder	1.61E-02	1.94E-02	3.55E-02
Red Bone Marrow	1.67E-02	2.00E-02	3.67E-02
Bone Surface	2.81E-02	2.04E-02	4.86E-02
Breast	2.12E-02	2.22E-02	4.35E-02
Colon	1.55E-02	1.90E-02	3.45E-02
Esophagus	1.50E-02	1.96E-02	3.46E-02
Eye	2.23E-02	2.28E-02	4.50E-02
Ovaries	1.48E-02	1.88E-02	3.36E-02
Testes	1.90E-02	2.08E-02	3.98E-02
Liver	1.70E-02	2.00E-02	3.70E-02
Lung	1.88E-02	2.12E-02	3.99E-02
Remainder organs	1.67E-02	1.99E-02	3.65E-02
Skin	2.19E-02	2.28E-02	4.47E-02
Stomach	1.70E-02	1.99E-02	3.69E-02
Thymus	1.84E-02	2.05E-02	3.89E-02
Thyroid	1.92E-02	2.14E-02	4.06E-02
Uterus	1.46E-02	1.81E-02	3.27E-02

The organ doses in the second and third columns of Table 10 are entered into the NIOSH IREP program assuming a chronic exposure and a lognormal distribution with a GSD of 4.0. The organ doses in the second column are attributed to photons with E=30-250 keV and the organ doses in the third column are attributed to photons with E>250 keV. The residual contamination exposure period is assumed to begin on April 1, 1962 and end on the employee's last day of work in Building 55.

The maximum internal exposure from residual radioactivity was estimated by assuming that the facility was uniformly contaminated at the level of maximum smear result (considered removable contamination) of 640 dpm/100 cm^2. This value was multiplied by a resuspension factor of 1E-06 m^{-1} (US NRC 2002c). This resulted in an estimated maximum residual air concentration of 0.03 pCi/m^3. Assuming a breathing rate of 1.2 m^3/h and a 2000-h work year results in a possible annual inhalation intake of 71 pCi. This value is considered negligible as it results in an annual dose of less than 1 mrem to the maximally exposed organ and is not included in the dose reconstruction.

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