Office of Compensation Analysis and Support Technical Basis Document for Atomic Energy Operations at Blockson Chemical Company, Joliet, Illinois	Document Number: OCAS-TKBS-0002 Effective Date: 9/11/2006 Revision No.: 0 Controlled Copy No.:_____ Page 1 of 27
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RECORD OF ISSUE/REVISIONS

ISSUE AUTHORIZATION DATE	EFFECTIVE DATE	REV. NO.	DESCRIPTION
Draft	6/29/2006	00-A	Document initiated to establish the technical basis for radiation dose reconstruction for former workers from the Blockson Chemical Company.
7/31/2006	9/11/2006	00	Incorporates resolution of review comments. This supersedes ORAUT-TKBS-0002. A PER is required for dose reconstructions performed with the superseded document.

1.0 Introduction

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

In this document the word "facility" is used as a general term for an area, building, or group of buildings that served a specific purpose at a site. It does not necessarily connote an "atomic weapons employer facility" (AWE facility) or a "Department of Energy facility" as defined in the Energy Employees Occupational Illness Compensation Program Act of 2000 [EEOICPA; 42 U.S.C. § 7384l(5) and (12)]. EEOICPA, as amended, provides for employees who worked at an AWE facility during the contract period and/or during the residual contamination period.

Employment at an AWE facility is categorized as either (1) during the contract period (i.e., when the AWE was processing or producing material that emitted radiation and was used in the production of an atomic weapon), or (2) during the residual contamination period (i.e., periods that NIOSH has determined there is the potential for significant residual contamination outside of the period in which weapons-related production occurred). For contract period employment, all occupationally-derived radiation exposures at covered facilities must be included in dose reconstructions. NIOSH does not consider the following exposures to be occupationally-derived:

- radiation from naturally occurring radon present in conventional structures; and
- radiation from diagnostic X-rays received in the treatment of work-related injuries.

For residual contamination period employment, only the radiation exposures defined in 42 U.S.C. § 7384n(c)(4) (i.e., radiation doses received from DOE/AEC-related work) must be included in dose reconstructions. Radiation dose received from DOE/AEC-related work includes: (1) radiation from radon consistent with NIOSH's policies for including such radiation in the contract period; and, (2) medical screening X-rays, but not diagnostic X-rays for the treatment of work-related injuries. It should be noted that: (1) under subparagraph A of § 7384n(c)(4), radiation associated with the Naval Propulsion Program is specifically excluded from the employee's radiation dose; and, (2) under subparagraph B of this section, radiation from a source not covered by subparagraph A that cannot be reliably distinguished from radiation that is covered by subparagraph A is considered part of the employee's radiation dose. This site profile covers only exposures resulting from nuclear weapons-related work. Exposures resulting from non-weapons related work, if applicable, will be covered elsewhere.

Section 2.0 of this document describes the Blockson Chemical Company site and its history including some information about the radiological processes and source terms as well as the radiological controls and monitoring practices. Sections 3.0 and 4.0 discuss internal and external dose, respectively.

The following summary is to help provide consistency in dose reconstructions and to help assure that all components of dose are adequately addressed when doses are reconstructed. It also provides some information regarding the radiological processes and source terms, as well as information on the radiological controls and monitoring practices. While not meant to substitute for a complete site profile, it represents the best understanding of the covered site at this time and provides assumptions for estimating doses when specific dose-related information is not available in individual records.

This site profile provides specific information on historical practices and radiation exposures at the Blockson Chemical Company facility in Joliet, Illinois. Blockson Building 55 and related activities are covered under EEOICPA (DOE 2006).

2.0 Site History

In 1950-1951 the U. S. Atomic Energy Commission (AEC) approached several phosphate rock consumers about the possibility of recovering the uranium from the phosphate rock they processed. At the Blockson Chemical Company plant, the AEC was interested in the uranium that could be separated from the phosphoric acid. In early 1951 the research staff at Blockson began an evaluation of the available research data and preliminary experimentation that the AEC made available to them. They determined that the only economically feasible approach applicable to the Blockson process would be to make the uranium recovery a by-product process (Stolz, Jr. 1958).

On March 6, 1951, the AEC entered into letter contract number AT(49-1)-606 with Blockson Chemical Company to develop a process to extract uranium from wet phosphoric acid (DOE 1983, DOE 2006). This letter contract was later replaced by contract number AT(49-1)-611 on October 18, 1951 (AEC 1951). Under the contract, Blockson constructed Building 55, at its own expense, to house uranium recovery equipment at their plant in Joliet, Illinois. The uranium recovery plant was constructed to recover uranium from phosphoric acid being produced by Blockson from normal commercial operations. On August 15, 1952, Blockson began production and delivery of uranium concentrates to the AEC (Stolz, Jr. 1958).

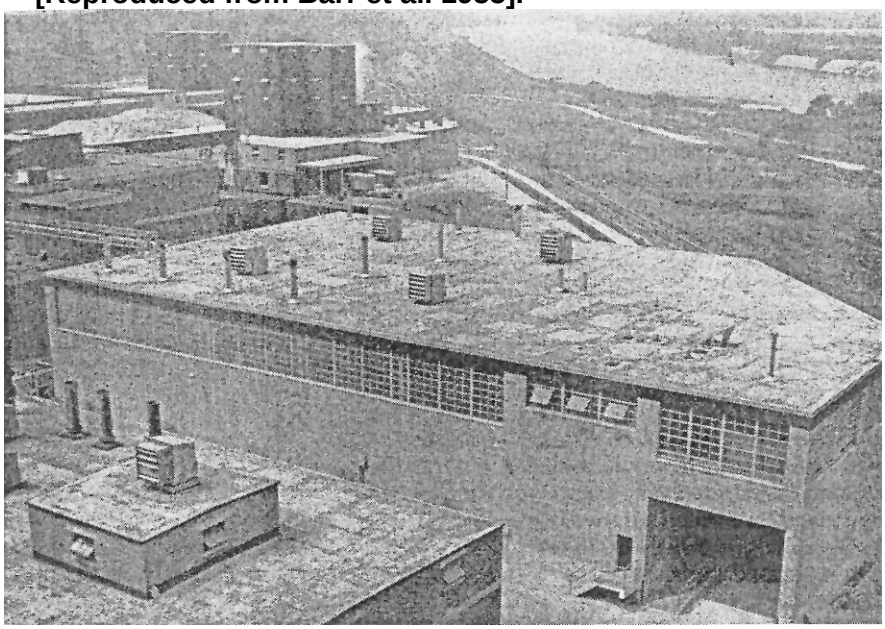
In 1955, Blockson was sold to the Olin Mathieson Chemical Corporation who assumed the liabilities and obligations of Blockson under all contracts, as stated in contract number AT(49-1)-611 Amendment 1. The Olin Corporation continued the uranium recovery program under contract with the AEC. In March 1962, the uranium recovery work was discontinued with the expiration of the contract (DOE 1983).

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 (DOE 1985). Records of bioassay monitoring have been found during

data capture efforts for the EEOICPA program, but there is no indication from records searches that external dosimetry was utilized at Blockson.

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). Building 55 was a one-story, 100-by-175-foot building built specifically to house the uranium recovery process (AEC 1951, DOE 1983). A photograph of Building 55 is show in Figure 1.

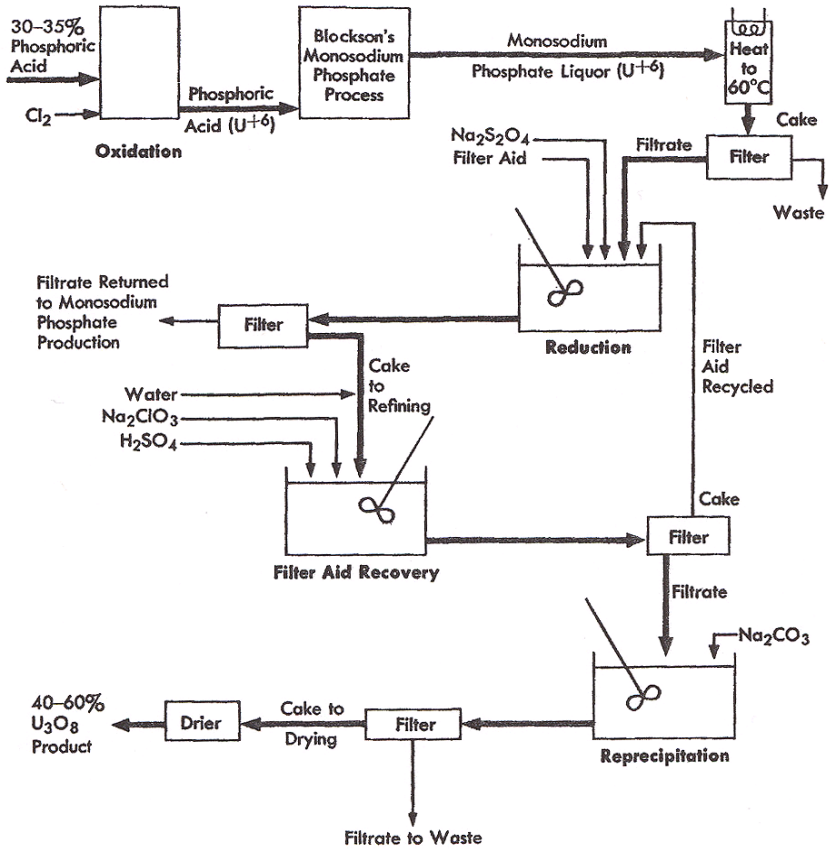
Figure 1: Photograph of the Joliet plant of the Blockson Chemical Company. [Reproduced from Barr et al. 1955].



The Blockson Chemical Company manufactured wet-process phosphoric acid from Florida phosphate rock (Barr et al. 1955, Clegg and Foley 1958). Figure 2 shows a schematic flowchart of the Blockson process for the recovery of uranium from wet phosphoric acid.

The plant produced technical phosphates rather than fertilizers from wet phosphoric acid (Wilkinson 1976). In the Blockson process, the phosphate rock was 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 partitioned about 90% of the uranium. Very little uranium was lost to the phosphogypsum. The phosphoric acid was then converted into monosodium phosphate and other phosphorus derivatives. The uranium by-product was precipitated from the monosodium phosphate stream. The monosodium phosphate liquor was heated and clarified. Sodium hydrosulfite ($Na_2S_2O_4$) was added to precipitate the uranium. The liquor was filtered and the filtrate returned to the phosphate-processing plant. The precipitate, containing about 5% U_3O_8 was slurried in water in an upgrading step in which the uranium was redissolved. The uranium was then reprecipitated as uranous phosphate. The slurry was filtered and the precipitate, known as yellowcake and containing 40 to 60% U_3O_8, was 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).

Figure 2 The Blockson Chemical process for the recovery of uranium from phosphoric acid. [reproduced from Clegg and Foley 1958].



A pre-operational letter from Blockson to AEC in July 1951 provided estimates of construction and operating costs based on Blockson processing 6,000 tons of phosphate rock per week (Blockson 1951). Taking recovery into consideration they estimated producing 50,000 pounds of U_3O_8 per year. AEC reported in 1955 that U_3O_8 production from Blockson was 3,758 pounds in August; 3,407 pounds in September; 5,908 pounds in October; 4,093 pounds in November; and 2,937 pounds in December of 1955 (AEC 1955a, 1955b, 1955c, 1955d, 1955e). This indicates an average production rate of over 4,000 pounds per month in the latter part of 1955. The December report included information that Blockson had produced a total of 121,400 pounds in the 40 months of Blockson operations to that point, an average of about 3,035 pounds of U_3O_8 produced per month through 1955. The total amount of uranium produced from 1955 through 1962 is not precisely known. Per a contract amendment in 1958 production was limited to 50,000 pounds of U_3O_8 per year (US DOE 1985).

3.0 Internal Dose

Workers in Building 55 and related activities were potentially exposed to airborne uranium. Thorium, assumed to follow the uranium in the Blockson process, and radon are also considered to be present in Building 55 and is included in this document.

3.1 Uranium Intakes

The greatest potential for internal uranium exposure in the Blockson uranium recovery process is associated with handling dried uranium compounds in the packaging areas. Here the uranium concentrate (yellowcake) was dried and barreled for shipping, resulting in a potentially dusty operation (Eidson and Damon 1984, US NRC 2002a, Wimpfen 2002). In all other areas of the Blockson plant, wet processes were used, and the surface contamination and potential dust exposures are minimal in comparison with the drying and packaging operations (Clegg and Foley 1958, US NRC 2002b).

There are no air sampling results available from Blockson to characterize airborne concentration of uranium, however, air samples results from mills with some common activities have been published. A study was done (Eidson and Damon 1984) of uranium aerosols generated during yellowcake packaging operations at four uranium mills. The precipitated yellowcake is dewatered in a filter process, and then dried in an oven to produce powdered yellowcake that is placed in 55-gallon drums. Blockson also used a dryer to dewater the yellowcake prior to packaging (Clegg and Foley 1958, Blockson 1951).

Eidson and Damon's study described a sequence of steps common to all four uranium ore processing mills:

1. No activity. This is when the plant is shut down for maintenance or when all available yellowcake has been barreled. Worker exposure to airborne yellowcake is minimal at this step.
2. Barrel loading. This occurs when a barrel is placed under a hopper containing the dried yellowcake. The yellowcake is allowed to fall into the barrel. The amount of time workers spend in this area depends on the volume of the yellowcake in the hopper. (It is unclear whether a hopper was used at the Blockson Chemical facility.)
3. Barrel uncovering. This step occurs when a filled barrel is removed from beneath the hopper. In some cases, the barrel may be vibrated to compact the yellowcake before removing the barrel from beneath the hopper. (It is not known of the barrels at Blockson were vibrated.)
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 barrel and seals it.
6. Other activities. This step includes maintenance and cleaning of the area with water hoses.

During the study, air samples were taken in yellowcake packaging areas before, during, and after barrels of yellowcake were filled and sealed. Median air concentrations during the study ranged from 0.04 to 0.34 μg U/L (40 to 340 μg U/m^3).

Comparison of the above uranium mill results to Blockson operations cannot readily be made due to differences in operations and in quantities produced. Blockson produced a small quantity of yellowcake from phosphoric acid. Based on the reported masses produced, just a few drums, possibly up to 5 per month, would have been produced, assuming a nominal 1,000 pounds of yellowcake per drum. The mills processed uranium bearing ores that contained larger relative uranium concentrations, and in typically larger quantities.

Although there are no air sampling results available for estimating intakes to Blockson workers, urinalysis results for uranium are available for Blockson workers, which is the preferred method of estimating internal dose received from exposure to uranium. Bioassay results are not available for individual claims submitted to NIOSH for dose reconstruction under EEOICPA, although the names on available results match a few workers whose job at Blockson is known. Default inhalation (or ingestion) intakes are presented below to be used for dose reconstructions when bioassay results are not available for a claim.

The bioassay results were evaluated and assessed to provide intake rates that are favorable to claimants and are considered bounding for Blockson workers. This evaluation assumes two categories of workers. Workers that are considered to have the highest potential for exposure are categorized as production workers in this evaluation. This category includes operators who handle and package dried yellowcake daily. Since there is no definitive data to differentiate exposure rates to production workers who are exposed to the highest concentrations and those exposed only intermittently, e.g., maintenance mechanics, all production workers are assumed to be exposed to the bounding concentration. The other category of exposure in this evaluation are administrative workers. The intake rates for administrative workers are based on the assumption that they would not be exposed to any significant degree in close proximity to the dried uncontained material, but could have been exposed to elevated levels of general area airborne uranium contamination on a continual basis.

Records indicate that Blockson employed approximately 20 people in the uranium recovery operations. In 1951, prior to start up of operations, Blockson sent a document to the AEC that included a best estimate of manufacturing costs, with a breakdown of labor by category. Blockson projected the following personnel needs for the uranium operations: 2 Operators per shift, 1 Chemist per shift, 2 Daymen, 2 Mechanics (on average), 1 Clerk, 1 Development Chemist, and 1 Foreman. From the rate information given and total estimate it could be determined that Blockson assumed 4.2 shifts per day, which would allow for 24 hours operations every day of the week. This results in about 18 full time personnel and two part time personnel according to pre-operational estimates (Blockson 1951).

In September 1953, after start up of operations, the AEC received a request for bioassay analysis services for “about twenty production workers engaged in uranium processing at Blockson Chemical Company” (AEC 1953). The first known sample results from these services were reported in April 1954 by the AEC New York Operations Office Health and Safety Laboratory

(HASL). Subsequently, bioassay results on nine other occasions were reported by HASL through February 1958. Sample results are available for twenty five different workers.

The urinalysis records were found on reports from the AEC New York Operations Office, Health and Safety Division. One hundred twenty two sample results are available with the results ranging from 0 to 17 µg uranium per liter. The analyses were performed by fluorimetry. Of the twenty five workers, nineteen workers had multiple bioassay results, with six workers having a single sample result reported. The nineteen workers with multiple results were used to determine a distribution of inhalation intake rates. One of the nineteen workers had multiple samples over two different time periods with a two year break without urinalysis. For purposes of estimating worker intakes, that workers results were analyzed as two workers, resulting in a distribution of intake rates for twenty workers. The workers with only one sample result were not used, however, those six results ranged from 0 to 6 µg per liter, which fit in the distribution of bioassay results.

Some of the names on the urinalysis reports have been matched with names of some workers whose jobs in Building 55 during uranium recovery work are known. These workers include two process operators, two chemists/analysts, and a supervisor. Some of these workers have been interviewed to confirm the work they performed in Building 55. Their bioassay results support the exposure assumptions in this evaluation, i.e., the highest exposed of those known workers was an operator whose job included drumming dried yellowcake, and the default intake recommendations are favorable in comparison to the data for those workers.

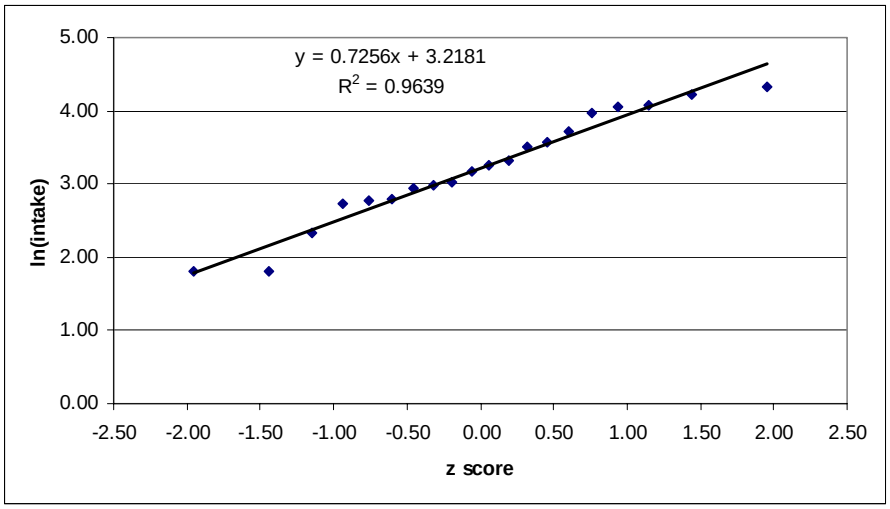
The International Commission on Radiological Protection (ICRP) recommends three material types for solubility of inhaled uranium, Types F, M and S, based on the clearance rate from the lungs (ICRP 1994a). Various studies have shown that U_3O_8 closely corresponds to the clearance rate associated with material Type M. Some studies have also shown that high fired material can produce uranium compounds that clear more slowly from the lungs, i.e., indicative of material Type S (Rucker, et al. 2001). Type M uranium is the most appropriate lung solubility material type based on the process used at Blockson. The U_3O_8 product was produced from wet phosphoric acid by filtering the precipitated uranium and then using a dryer to dewater the solids (Blockson 1953). No calcining or high firing of the uranium material was indicated by the process used at Blockson. Based on these processes and the results of various studies that have been summarized by Rucker, et al., Type M material is used to derive intakes from bioassay results.

The Blockson bioassay results were reported in µg per liter, and were converted to µg per day by multiplying by a daily excretion rate of 1.4 liters, then converted to pCi/day by multiplying by 0.677 pCi per µg of natural uranium. Individual worker intakes were determined using IMBA-Expert™ by assuming a chronic inhalation intake of Type M uranium with parameters recommended by the ICRP (ICRP 1994b).

For daily intake rate calculation purposes, intakes were assumed to have occurred beginning in the year sampled and ending with the last sample date. The results were given an absolute error of 1.0, which equally weights the sample results for fitting purposes in IMBA-Expert. Daily intake rates ranged from 6 to 76 pCi/day. The intake rate results fit well to a lognormal

distribution having a median value of 25 pCi/day with a geometric standard deviation of 2.1, as shown in Figure 3. The analysis of bioassay results with the assumption of 100% of the intake via inhalation is a claimant-favorable assessment of intakes from all pathways except when calculating dose to certain tissues of the gastrointestinal tract. Dose from ingestion of uranium is discussed below.

Figure 3: Distribution of uranium intake rates.



Production workers are assumed to be continually exposed at the 95th percentile intake rate of 82 pCi/day, and administrative personnel are assumed to be exposed continually at the median intake rate of 25 pCi/day. These bounding intakes should be entered into IREP as constants. The presence of associated radionuclides that could be present and contribute to significant internal dose is assumed (ORAU 2006). The results are summarized in Table 1a below.

Table 1a: Inhalation rate for operations.

Worker category	Intake rate for Type M material[1]	Distribution
Administrative	25 pCi/day total U	Constant value
Administrative	0.35 pCi/day Th-228[2]	Constant value
Administrative	0.35 pCi/day Th-232[2]	Constant value
Production workers	82 pCi/day M total U	Constant value
Production workers	1.1 pCi/day Th-228[2]	Constant value
Production workers	1.1 pCi/day M Th-232[2]	Constant value

1. Intake rates are normalized to units of calendar days. The intake period for operations is March 1, 1951, through March 31, 1962.
2. Thorium intake rates are derived from ratios in ORAU 2006. Solubility types for thorium are based on recommendations in IRCP Report 68 (ICRP 1994b).

Workers also had the potential to ingest uranium from contact with contaminated surfaces or from eating or drinking in the area. When deriving intakes from the bioassay results, a chronic ingestion of uranium resulted in a higher dose to certain tissues of the gastrointestinal tract when compared to the dose from the inhalation intakes described above. Therefore, intakes are presented in Table 1b based on the presumption that all the uranium in the workers urine was due to ingestion. Although inhalation is the most common mode of intake in a production facility the

presumption of the ingestion pathway provides an upper bounding value for dose from ingestion. A worker should be assigned intakes from inhalation or ingestion, not both.

Table 1b: Ingestion rate for operations.

Worker category	Intake rate Type M Material[1,3]	Distribution
Administrative	83 pCi/day total U	Constant value
Administrative	1.2 pCi/day Th-228[2]	Constant value
Administrative	1.2 pCi/day Th-232[2]	Constant value
Production workers	270 pCi/day total U	Constant value
Production workers	3.6 pCi/day Th-228[2]	Constant value
Production workers	3.6 pCi/day Th-232[2]	Constant value

1. Intake rates are normalized to units of calendar days. The intake period for operations is March 1, 1951, through March 31, 1962.
2. Thorium intake rates are derived from ratios in ORAU 2006.
3. Ingestion intakes provide bounding dose to the stomach, small intestine, upper large intestine, lower large intestine, and colon. The f1 values are 0.02 for uranium ingestions and 0.0005 for thorium ingestions.

3.2 Radon Exposures

Radon exposures to workers from uranium extraction work at phosphate plants have been evaluated for the NIOSH EEOCIPA dose reconstruction project (ORAUT 2006). For reconstructing lung doses, all workers at Blockson are to be assigned an exposure of 0.036 WLM (working level month) per year due to radon progeny.

Table 2: Radon exposures.

Dose component	Annual dose/exposure[1]	Distribution
Radon progeny	0.036 WLM (lungs only)	Lognormal, GSD=2.0
Radon progeny	75 rem alpha (ET1 only)[2]	Lognormal, GSD=2.0
Radon progeny	0.30 rem alpha (ET2 only)[2]	Lognormal, GSD=2.0
Radon gas	0.002 rem alpha (non-respiratory tract tissues only)	Constant value

1. Exposure and dose values from ORAU 2006.
2. ET1 and ET2 dose conversion factors from NIOSH 2006.

4.0 External Dose

External dosimetry data is not known to exist for Blockson workers, and data capture efforts for the EEOICPA dose reconstruction project have not found any direct radiation survey results from the Blockson facility. Therefore, source term information has been used to estimate external doses to workers. Blockson's uranium recovery process was a by-product process designed to fit into the existing phosphate process (Stolz, Jr. 1958). The primary radionuclides of interest for potential external exposure in Building 55 are U-238 and daughter radionuclides Th-234 and Pa-234m.

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 (DOE 1983). In the

manufacture of this phosphoric acid, phosphate rock was 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).

For the purpose of dose reconstruction, it is assumed that there was a potential for external exposure from the following sources:
- Submersion in air contaminated with yellowcake dust,
- Barrels of yellowcake,
- Contaminated surfaces,
- Medical x-rays.

Based on an assumed limiting air concentration of 8.5 pCi/m^3 derived from the 95th percentile daily intake rates discussed above, the external dose from submersion in air contaminated with uranium dust was calculated based on dose coefficients for U-238 and daughter radionuclides Th-234 and Pa-234m from Federal Guidance Report No. 12 (EPA 1993). All organ doses are less than 0.001 rem per year. These are insignificant in comparison to the favorable evaluation of other dose components and, therefore, are not included in dose reconstructions.

Dose from occupationally required medical X-rays have also been considered and assumed to have occurred, although no information has been found to indicate that Blockson or the AEC required X-rays of the workers.

4.1 Source Term

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 (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 (NRC 2002a). Figure 4 shows the rise in beta dose rate during 100 days after separation from ore.

Figure 5 shows that the beta dose rate from the surface of yellowcake decreases rapidly as a function of distance from the surface. Rapid decrease in the beta dose rate with distance, and the shielding afforded by shoes and clothing, reduces dose from beta radiation, particularly from yellowcake deposited on floors.

Figure 4. Beta dose rate on the surface of yellowcake. [Reproduced from NRC 2002a]

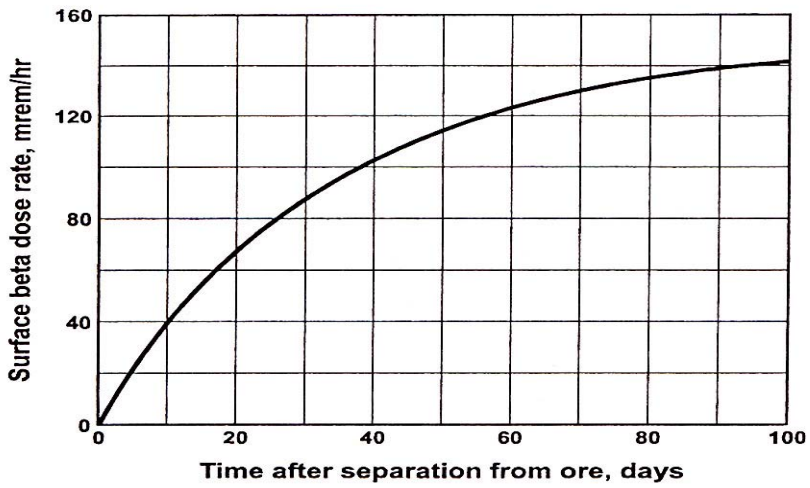
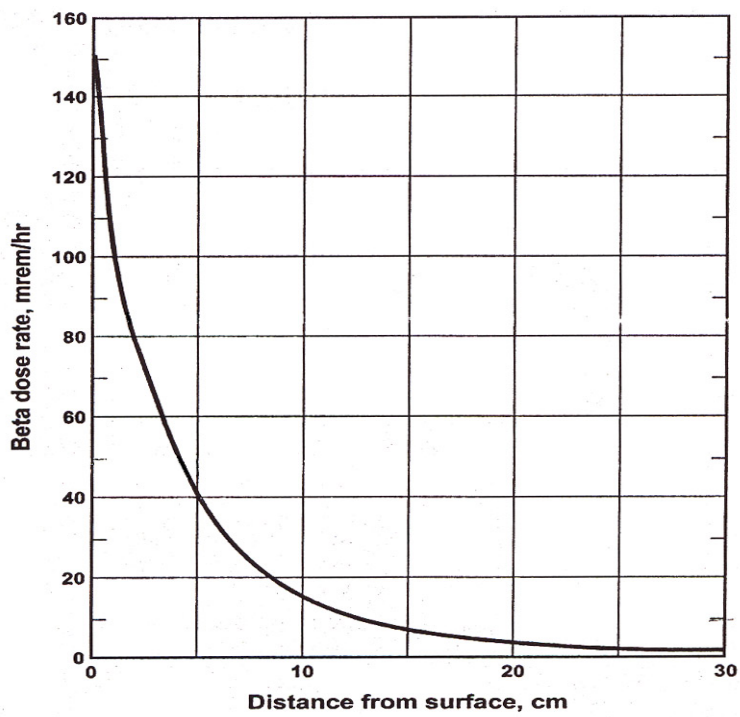


Figure 5. Beta dose rate from yellowcake separated from ore for more than 100 days as a function of distance from the surface. [Reproduced from US NRC 2002a]



4.2 Exposure from Drums of Uranium

MCNPX (version 2.5.0) was used to determine the dose rate per curie of ^{238}U regardless of the actual activity in the drum. This was later adjusted for actual source activity to compare actual

dose rates. All radionuclides were ratioed with respect to ^{238}U to determine the number of photons and electrons per decay of ^{238}U. Anderson and Hertel (Anderson and Hertel 2005) showed that the short lived nuclides (^{234}Th, ^{234m}Pa, ^{234}Pa, and ^{231}Th) are very close to equilibrium (adjusted for branching ratios) at 100 days. For the purposes of this evaluation, branching ratio adjusted equilibrium was assumed. ICRP Publication 74 Table A.1 was used to convert the photon flux to units of air kerma using the conversion factors in Table 3 below (ICRP 1996; Stabin and da Luz 2002).

Table 3: Activity of aged uranium following separation and 100 day ingrowth.

	Activity Concentration in aged Uranium Metal (Bq/g)	Relative activity concentration (normalized to ^{238}U)
^{238}U	12200	1
^{234}Th	12200	1
^{234m}Pa	12200	1
^{234}Pa	19.52	0.0016
^{234}U	12200	1
^{235}U	555	0.045492
^{231}Th	555	0.045492

Dose 30 cm from drum

The dose rate was determined at 77.9 cm above the ground, 30 cm from the edge of the drum for both the photon and beta emissions of natural uranium and its progeny. Results of these are provided in Table 4 and Figure 6.

Table 4: Uranium dose rates from drums of yellowcake.

Density of U_3O_8 (g cm^{-3})	Activity of U in drum (Ci)	Photon emission dose(rad/hr)	Bremsstrahlung dose (rad/hr)	Total dose rate at 30 cm (rad/hr)
0.5*	3.121E-02	3.96E-04	3.20E-4	7.16E-4
1	6.242E-02	5.00E-04	3.60E-04	8.60E-04
2	1.248E-01	5.54E-04	3.76E-04	9.30E-04
4	2.497E-01	5.84E-04	3.84E-04	9.69E-04
6	3.745E-01	5.84E-04	3.64E-04	9.48E-04
6.7	4.182E-01	5.81E-04	3.74E-4	9.56E-4

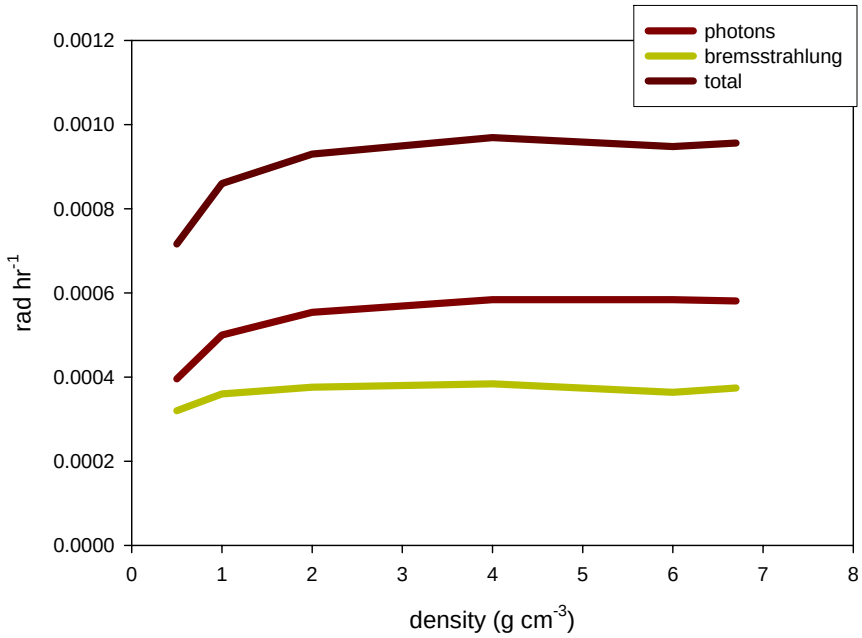
*The drum begins to noticeably impact the dose rates at low material concentration.

The affect of density of the drummed uranium concentrate on the modeled dose rate was evaluated. The effective density of the drummed material was assumed to be variable up to 6.7 g cm^{-3}. The results are shown in Figure 6.

Based on these results, the effective density of drummed U_3O_8 concentrate makes little difference in the calculated dose rates from 1 to 6.7 g cm^{-3}. This would not be true if significant contaminants were introduced. Assuming the material is pure U_3O_8 with a density of 4 g cm^{-3} is favorable to claimants without excessive bias because of the self shielding created by additional materials. The introduction of significant quantities of additional materials essentially will

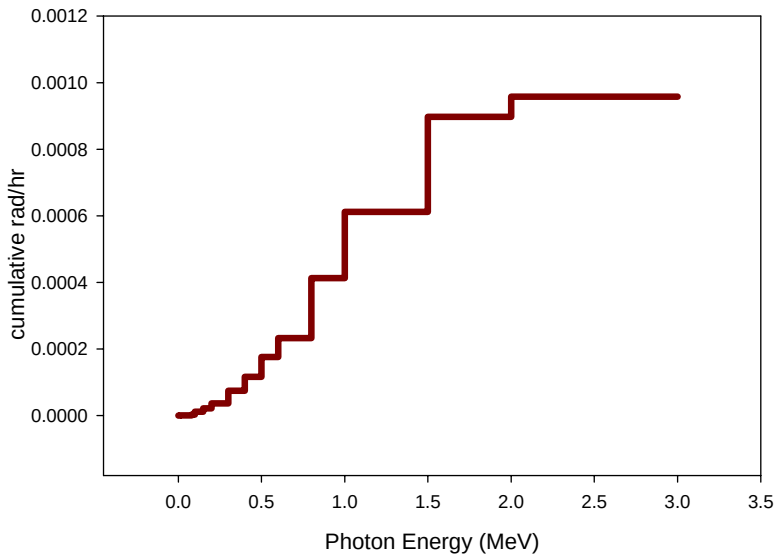
provide additional shielding and reduce the effective Z of the material, both of which will reduce the dose rate. Furthermore, as is evidenced by the drop at 0.5 g cm^{-3}, the steel shell of the drum makes a bigger impact on the dose results for low density uranium. This dose rate compares well with the result obtained by Anderson et al. (2005) from a bare, cylindrical uranium ingot at 30.48 cm (16.51 cm radius, 50.8 cm height) for a total of 1.15 mrem/hour. Differences arise from changes to geometry and shielding from the drum walls.

Figure 6: Effect of density on dose rate at 30 cm from drum of U_3O_8.



Assignment of a 50%/50% distribution of dose from 30-250 keV photons and >250 keV photons was verified to be favorable to claimants as only approximately 8% of the total dose comes from photons less than 300 keV (originating from both bremsstrahlung and photon emissions) in a drum with density of 4 g/cm^3. Cumulative energy distribution results are shown in Figure 7.

Figure 7: Cumulative dose rate from photons and bremsstrahlung at 30 cm from a drum containing U_3O_8 (density 4 g/cm^3).



The air kerma dose rates were converted to annual organ doses by assuming a worker's exposure time was lognormally distributed. The median exposure time was determined by assuming all workers were working eight hours per day, one day per week at a distance of 1 foot from the drum. This was normalized to 400 hours per work year. The 95th percentile exposure time was determined by assuming the worker spent a standard 2000 hour work year at a distance of 1 foot from the drum. This results in a kerma dose distribution with a median value of 0.387 rad per year with a geometric standard deviation of 2.7. The dose distribution was assumed to be 50% from photons 30-250 keV and 50 % from photons > 250 keV.

For the purposes of calculating organ doses for use in the NIOSH Interactive RadioEpidemiological Program (NIOSH-IREP) to the whole body doses were input into CRYSTAL BALL® using Monte Carlo methods to multiply times the triangular organ dose conversion factors for kerma to organ dose found in NIOSH External Dose Reconstruction Implementation Guideline (NIOSH 2002). The results are annual doses that are lognormally distributed. The results are in Table 5. For skin, the air kerma values were multiplied by 1.0.

Table 5: Annual organ dose from exposure to drums of yellowcake.

	30-250 keV photons		>250 keV photons	
Organ	**dose, rem**	**GSD**	**dose, rem**	**GSD**
Bladder	0.223	2.8	0.213	2.7
RBM	0.100	2.9	0.163	2.7
Bone Surface	0.266	2.8	0.179	2.8
Breast (female)	0.264	2.6	0.223	2.7
Colon	0.179	2.8	0.198	2.7
Esophagus	0.107	2.9	0.167	2.7
Eye	0.265	2.7	0.215	2.7
Ovaries	0.158	2.8	0.193	2.7
Testes	0.300	2.7	0.225	2.7
Liver	0.123	2.7	0.166	2.7
Lung	0.171	2.8	0.200	2.6
Remainder	0.146	2.8	0.185	2.6
Stomach	0.232	2.7	0.213	2.7
Thymus	0.263	2.7	0.224	2.7
Thyroid	0.285	2.7	0.228	2.7
Uterus	0.174	2.8	0.188	2.7
Skin	0.194	2.7	0.194	2.7

4.3 Exposure from Contaminated Surfaces

Estimates of external dose from surfaces contaminated with uranium have been performed. The 95th percentile intake rates from inhalation were used to derive an average airborne concentration of 8.54 pCi/m^3. A terminal settling velocity of 0.00075 m/s was used as an estimate of the velocity of deposition to surfaces in the building. The value is within the range of deposition velocities measured in various studies (NRC 2002b). It was assumed that uranium settled on plant surfaces at a steady state 24 hours per day for 365 consecutive days with no cleanup or removal of contamination.

The estimated surface contamination results for U-238 and associated daughter radionuclides Th-234 and Pa-234m were multiplied by the Dose Coefficients for Exposure to Contaminated Ground Surface found in Table III.3 of Federal Guidance Report No. 12 (EPA 1993). With the exception of dose to the skin, the annual dose for all organ is less than 0.001 rem and is insignificant in comparison to the favorable assignment of dose from drums of yellowcake. Skin dose is discussed in section 4.4 below.

Although the modeled external doses are assumed to be insignificant based on airborne uranium concentrations there could have been localized spots with greater potential for exposure. The only direct reading radiation results available are those from surveys done from March to November 1978 by Argonne National Laboratory (ANL) (DOE 1983).

The ANL survey was performed throughout Building 55, including plant surfaces, tanks, pipes, and other process equipment. The report estimated that some areas were not accessible, but that they had surveyed an estimated 95% of the floors and 90% of the walls. The surveys included contamination surveys. A dose rate was taken at contact and at 1 meter on all 63 spots in which contamination was detected. The dose rates at 1 meter on 7 of the 63 "hot" spots ranged from 0.04 mR/hr to 0.2 mR/hour. The other 56 spots had 1 meter dose rates indistinguishable from background. The reported background dose rate on the instrument used was between 0.02 mR/hr and 0.03 mR/hr. The results of the 7 spots with measurable 1 meter dose rates included the background dose rates. From a review of the survey map and results it seems improbable that a worker could be significantly exposed above the background rate of 0.03 mR/hr for significant time. However, in the absence of individual dosimeter data, whole body dose rates are modeled by a lognormal distribution with the 0.03 mR/hr rate assumed to be the median value for 2,000 hours per year. The geometric standard deviation is 3.2, which was determined by assuming that the 95th percentile dose rate is equal to the maximum observed result of 0.2 mR/hr.

There are no survey results to use to assign external doses until the 1978 survey by ANL. The 1978 survey suggests that either the facility was only contaminated in localized spots, or it indicates that some areas of surface contamination was not subject to the same contamination depletion rate that would be expected from natural processes and from general cleaning and weathering that would have occurred since 1962. The 7 spots were on small areas of floor (reported to be about 0.5 m^2 each) and on a pipe inlet and on a spot on a machine. Additionally, since the normal non-AEC related operations at Blockson produced byproduct that contained small amounts of uranium and daughter products, it is unknown if that would have contributed to the contamination of the facility after AEC operations ended in 1962. Based on the above considerations and the absence of more data, an assumption that is favorable to claimants is made for deriving dose rates prior to 1978. It is assumed that the few spots in the facility with measurable dose rates were representative of the entire facility from the operational period to the present or until the end of the workers' EEOICPA covered employment. The deep dose rates during the operational period are adequately modeled by the assumptions made in deriving doses from drums of yellowcake. Therefore, dose from contaminated surfaces is only assigned during the residual contamination period.

4.4 Beta Dose

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 dose rate is about 150 mrem/hr (NRC 2002a). Figure 4 shows the rise in beta dose rate during 100 days after separation from ore.

It is assumed that there was a potential to receive a shallow dose from exposure to open drums of yellowcake during drum loading and sealing. According to Figure 5 the dose rate at 1 foot from the surface of aged yellowcake is between 1 and 2 mrem/hour. It is assumed that the production workers 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/hour. This results in a shallow beta dose of 0.8 rem/year. To allow for uncertainty, the time of exposure was assumed to be lognormally distributed with the 95th percentile exposure time assumed to 40 hours per week, 50 weeks per year. This results

in a shallow dose of 0.8 rem per year. Results for dose reconstructions are in Table 6. The calculated beta doses have not been reduced to allow for doses to areas of the skin that are typically covered by clothing resulting in a reduction of the beta dose to the skin.

Table 6: Shallow dose.

Dose component	Annual dose/exposure[1]	Distribution
Beta dose, E>15 keV	0.8 rem per year	Lognormal, GSD=2.7

4.5 Occupational Medical Dose

Dose from occupationally required medical X-rays have also been considered and assumed to have occurred, although no information has been found to indicate that Blockson or the AEC required X-rays of the workers. For the AEC operational period at Blockson, employees are assumed to have received an annual chest X-ray. Organ doses are listed in Table 7 below and are based on an assumed Posterior-Anterior (PA) exposure with minimal collimation. Dose values are reproduced from Table 6-5 of "Dose Reconstruction from Occupationally Related Diagnostic X-ray Procedures" (ORAU 2005.). The annual doses are applied as dose from 30-250 keV photons using the values in Table 7 as the mean of a normal distribution with a 30% standard deviation.

Table 7: Annual organ dose from medical X-rays.

Organ	Annual dose, rem photon 30-250 keV
Thyroid	3.48E-02
Eye/brain	6.40E-03
Ovaries	2.5E-02
Liver/gall bladder/spleen	9.02E-02
Urinary bladder	2.5E-02
Colon/rectum	2.5E-02
Testes	5.0E-03
Lungs (male)	8.38E-02
Lungs (female)	9.02E-02
Thymus	9.02E-02
Esophagus	9.02E-02
Stomach	9.02E-02
Bone surfaces	9.02E-02
Remainder	9.02E-02
Breast	9.80E-03
Uterus	2.5E-02
Bone marrow (male)	1.84E-02
Bone marrow (female)	1.72E-02
Skin	2.70E-01[1]

1. Skin dose is for skin in the primary beam.

5.0 Dose from Residual Contamination

The whole body median dose rate of 0.060 R/year derived from the discussion in section 4.3 above was used to calculate various organ doses. Photon dose is divided equally between 30-250 keV and the >250 keV ranges. The Roentgen to organ dose conversion factors for isotropic geometry in the External Dose Reconstruction Implementation Guideline (NIOSH 2002) were used to estimate organ doses. Skin doses were calculated assuming a conversion factor of 1.00. Results are in Table 8.

Table 8: Annual dose from residual contamination.[1]

	Photons E=30-250 keV	Photons E>30-250 keV
Organ	**dose, rem**	**dose, rem**
Bladder	1.61E-02	1.94E-02
RBM	1.67E-02	2.00E-02
Bone Surface	2.81E-02	2.04E-02
Breast	2.12E-02	2.22E-02
Colon	1.55E-02	1.90E-02
Esophagus	1.50E-02	1.96E-02
Eye	2.23E-02	2.28E-02
Ovaries	1.48E-02	1.88E-02
Testes	1.90E-02	2.08E-02
Liver	1.70E-02	2.00E-02
Lung	1.88E-02	2.12E-02
Remainder	1.67E-02	1.99E-02
Stomach	1.70E-02	1.99E-02
Thymus	1.84E-02	2.05E-02
Thyroid	1.92E-02	2.14E-02
Uterus	1.46E-02	1.81E-02
Skin	3.00-02	3.00E-02

1. For dose reconstructions, annual doses are applied as lognormal distributions with a GSD of 3.2.

Uranium inhalation intakes during the residual contamination period have been derived from the operational period intakes and from estimated airborne radioactivity derived from the 1978 FUSRAP survey. While the uranium recovery operations could result in high localized air concentrations, air concentrations from resuspension of residual contamination would be more consistent throughout the area. Interviews with former workers indicate that housekeeping was performed regularly to reduce build up of material on the floors. After cessation of uranium recovery work the main source of contamination (precipitated and dried yellowcake) was no longer present. Therefore, the derived median inhalation rate of 25 pCi/day is used as the inhalation intake rate at the start of the residual contamination period on April 1, 1962. Thereafter, airborne radioactivity from resuspension of contamination in the facility and corresponding intakes are assumed to decrease according to an exponential model described below.

Extensive radiological surveys were performed starting in March 1978. Airborne radioactivity concentration at 16 years post operation was estimated by assuming that the facility was uniformly contaminated at the level of the maximum alpha smear result of 640 dpm/100 cm^2 as reported in the 1978 survey (DOE 1983). This value was multiplied by a resuspension factor of 1E-06 m^{-1} (NRC 2002c), which results in an estimated maximum residual air concentration of 0.03 pCi/m^3. Applying a breathing rate of 1.2 m^3/hr for 2000 hours per year results in a potential inhalation of about 0.2 pCi/day at the time of the March 1978 survey. The 1962 and 1978 daily intake rates were used to estimate the rate of reduction of intakes according to the following equation.

$$I_t = I_0 * e^{-\lambda t}$$

where:

I_t	=	daily intake rate at time t
t	=	time (days) since April 1, 1962
I_0	=	daily intake on April 1, 1962
e	=	base of the natural logarithms
λ	=	exponential constant

The derived intake of 25 pCi/day on April 1, 1962, was substituted for I_0. The 0.2 pCi/day derived intake from the 1978 survey was substituted for I_t on April 1, 1962. The time between April 1, 1952, and April 1, 1962 is 5844 days. This resulted in the following equation to calculate the exponential constant λ.

$$\frac{0.2\ pCi}{d} = \frac{25\ pCi}{d} * e^{-\lambda * 5844 d}$$

The constant λ was determined to be 0.000826 day^{-1}.

Average daily inhalation intake rates for each year between 1962 and 1978 were then calculated and are given in Table 9a below. The 1978 intakes are relatively low and are applied for all subsequent years. The methods used for derivations of these intakes are considered bounding, and the corresponding annual doses are considered constants for purposes of dose reconstruction.

Table 9a: Inhalation intake rate from residual contamination[1]

Year	Uranium Type M pCi/day	Th-228 Type M pCi/day	Th-232 Type M pCi/day
1962 (Apr. 1 – Dec. 31)	22	0.31	0.31
1963	17	0.24	0.24
1964	13	0.17	0.17
1965	9.4	0.13	0.13
1966	7.0	0.10	0.10
1967	5.1	0.070	0.070
1968	3.8	0.052	0.052
1969	2.8	0.039	0.039
1970	2.1	0.028	0.028
1971	1.5	0.021	0.021
1972	1.1	0.016	0.016
1973	0.84	0.012	0.012
1974	0.62	0.009	0.009
1975	0.46	0.006	0.006
1976	0.34	0.005	0.005
1977	0.25	0.003	0.003
1978 – end	0.19	0.003	0.003

1. Inhalation intakes are not assigned for calculating dose to the stomach, small intestine, upper large intestine, and lower large intestine. See Table 9b for ingestion intakes for those tissues.

Inhalation of uranium is considered to be proportional to the airborne concentration. For the residual period this is a function of the amount of loose contamination present on plant surfaces. Likewise, the potential for ingestion of uranium is a function of the amount of loose contamination present on plant surfaces. This indicates a relationship between airborne concentration and ingestion of material. Bounding ingestion intakes were evaluated and discussed above for the AEC operational period. Based on the median intake rate, that evaluation indicates that doses to the gastrointestinal tract are bounded by an ingestion of 83 pCi/day, which is based on the evaluation of bioassay results under the assumption that no inhalation occurred.

To estimate bounding intakes for ingestion during the residual contamination period the daily ingestion rate is reduced at the same rate as the inhalation intakes described above. This results in the following expression to determine ingestion intakes during the residual contamination period.

$$I_t = \frac{83\ pCi}{d} * e^{-0.000826 d^{-1} * t}$$

where:

I_t	=	daily ingestion intake rate at time t
83 pCi/d	=	bounding ingestion intake on April 1, 1962
t	=	days since April 1, 1962

The above equation was used to derive average daily ingestion intake rates for each year between 1962 and 1978. Results are presented in Table 9b. The ingestion intake rate is based on bounding dose to the stomach, small intestine, upper large intestine, and lower large intestine. The modeled ingestion intake rate assumes that all material assimilated from contaminated surfaces is via the ingestion pathway. Therefore, inhalation intakes are not assigned for these tissues. The doses are considered constants for dose reconstruction purposes.

Table 9b: Ingestion intake rate from residual contamination[1]

Year	Uranium Type M pCi/day	Th-228 Type M pCi/day	Th-232 Type M pCi/day
1962 (Apr. 1 – Dec. 31)	74	1.0	1.0
1963	57	0.78	0.78
1964	42	0.58	0.58
1965	31	0.43	0.43
1966	23	0.32	0.32
1967	17	0.23	0.23
1968	13	0.17	0.17
1969	9.3	0.13	0.13
1970	6.9	0.095	0.095
1971	5.1	0.070	0.070
1972	3.8	0.052	0.052
1973	2.8	0.038	0.038
1974	2.1	0.028	0.028
1975	1.5	0.021	0.021
1976	1.1	0.016	0.016
1977	0.83	0.011	0.011
1978 – end	0.62	0.008	0.008

1. Ingestion intakes are assigned only when calculating dose to the stomach, small intestine, upper large intestine, and lower large intestine. No inhalation intake is assigned for these tissues. The f1 values are 0.02 for uranium ingestions and 0.0005 for thorium ingestions.

6.0 Dose Reconstruction Summary

For EEOICPA dose reconstruction purposes, exposure starts March 1, 1951, or the first date the employee has covered employment at Blockson, whichever is later. The end of the operational period is March 31, 1962. Residual contamination doses start on April 1, 1962.

External doses are assigned according to the values specified in Tables 5, 6, 7, and 8, as applicable to the tissue of concern. The external dose values in Tables 5 and 6 are normalized annual doses and need to be fractioned for partial years worked. For external dose from residual contamination, the values in Table 8 are also normalized annual doses and need to be fractioned for partial years worked. The occupational medical x-ray doses in Table 7 are to be assigned by assuming one exposure per year of covered employment during the operational period.

In the absence of worker-specific bioassay data, the default intakes are assigned. For dose to the stomach, small intestine, upper large intestine, and lower large intestine, intakes are assigned from Tables 1b, 2, and 9b. For all other tissues assign intakes from Tables 1a, 2, and 9a. Natural uranium doses can be calculated by assuming 50% of the intake is from U-234 and 50% is from U-238.

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