| **Division of Compensation Analysis and Support**<br><br>Technical Basis Document for Atomic Energy Operations at Blockson Chemical Company, Joliet, Illinois | Document Number: DCAS-TKBS-0002<br>Effective Date: 12/20/2010<br>Revision No.: 03<br><br>Page 1 of 47 |
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| Subject Experts: Tom Tomes, Sam Glover<br><br>Approval: Signature on file Date: 12/20/2010<br>James W. Neton, Associate Director for Science | Supersedes:<br><br>Revision No.: 02 |

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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 A PER is required for dose reconstructions performed with the superseded document. |
| Draft | 3/30/2007 | 01-A | Rough draft for internal review. Added dose from additional locations on site and from additional radionuclides. Additional descriptions and editorial changes. |
| Draft | 4/11/2007 | 01-B | Draft issued for review. |
| Draft | 6/14/2007 | 01-C | Incorporated review comments. Revised internal and external dose modeling. Added additional site information and radiological data discussion. |
| 6/18/2007 | 6/20/2007 | 01 | Revision 01. Changes results in an increase of dose, and a PER is required. |
| 11/21/2007 | 11/21/2007 | 02 | Changed footnote 2 of Table 4a and footnote 3 of Table 12a to require consideration of Type M and Type S thorium in Building 55. Corrected an error in the liver dose in Table 8. Corrected errors in columns of Table 7 and resulting graph in Figure 6. No other changes were made. |
| Draft | 11/16/2010 | 03-A | Revision to incorporate Special Exposure Cohort determination and changes made to the dates of the AWE covered period. Those changes affected the radon section (3.3) and the dose during the residual contamination period (5.0). Changes were also made to the modeled external doses (4.2) from drums of uranium concentrates; this change resulted in a lower external dose due to correction of an error made in the modeled photon flux in the previous version. |
| 12/20/2010 | 12/20/2010 | 03 | Incorporated review comments. |

## 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. § 7384I(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 to receive compensation if certain requirements are met.

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 Secretary of Health and Human Services has determined that dose from radon cannot be reconstructed for radon exposures received from March 1, 1951, through June 30, 1960. Therefore, a class of employees from Blockson Chemical Company has been added to the Special Exposure Cohort (SEC) (HHS 2010).

The following summary is to help provide consistency in dose reconstructions and to help ensure 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.

This site profile provides specific information on historical practices and radiation exposures at the Blockson Chemical Company facility in Joliet, Illinois.

## 2.0 Site Description and Operational History

The Blockson Chemical Company manufactured wet-process phosphoric acid from Florida phosphate rock, which was subsequently used to manufacture other chemicals at the plant (Barr et al. 1955, Clegg and Foley 1958). 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 from the existing operations (Stoltz, Jr. 1958). Building 55 was constructed to house uranium recovery operations (Blockson 1951, DOE 2007). It operated through June 1960 (AEC c1963, DOE 2010). The entire Blockson plant stopped production of chemicals in June 1991 (EPA 1993a). Building 55 was demolished in 1996 (Olin 2007).

### 2.1 Blockson Facilities Affected by Uranium Operations

The U. S. Department of Energy (DOE) lists the Blockson Chemical Company as an Atomic Weapons Employer from 1951 through June 1960 with a Residual Radiation period ending in October 2009. The DOE covered facility description specifies Blockson operated Building 55 to produce uranium for the AEC (DOE 2010). The listing further states that "This listing is also intended to cover the AEC-funded laboratory, pilot plant and oxidation process, which also occurred at Blockson, and was related to the work in Building 55." The U. S. Department of Labor (DOL) sent NIOSH an interpretation of the areas they consider to be covered under EEOICPA, citing the current designation that was published in the Federal Register (FR) on August 23, 2004 (DOL 2007). The FR listed "Blockson Chemical Company (Building 55 and related activities)" as the covered facility (FR 2004). The DOL determined that the designation applies to "Building 55, the pilot plant and associated processes (including the oxidation process) and all the areas in which these processes where performed." Consequently, NIOSH conducted research to identify changes made to operations on site to accommodate uranium extraction work.

Blockson received phosphate rock primarily on barges. The rock was unloaded and put into silos. A conveyor transferred the rock to a large outdoor calciner (furnace), where it was heated to a high temperature. The purpose of the calcination was to break down organic material and to prepare the feed material for digestion. After calcination, samples were obtained and sent to the laboratory for analysis. A former worker has said the sampling was to test for fineness to determine if the contents of the furnace were acceptable for transfer to Building 40 or if additional calcination was needed (OCAS 2007a, OCAS 2007b, Stoltz, Jr. 1958). Calcination was already being performed at Blockson prior to AEC work; however, changes to calcining procedures were made when Blockson began uranium extraction work. Calcination would leave uranium and some other elements in a reduced state. Oxidizing conditions were desired to improve uranium recovery in subsequent processing steps. In previous years Blockson had done extensive research on calcination. When uranium extraction started, they pulled out the old research and modified the calcination procedure to help maintain uranium in an oxidized state (Stoltz, Jr. 1958). Therefore, calcining operations at Blockson can be considered a related activity and were affected by the AEC contract work, and for the purpose of dose reconstruction are considered in this site profile.

A conveyor transferred the calcined phosphate feed to the acid plant, Building 40 (once known as Building 25) (OCAS 2007b). Blockson modified their existing operations by adding an oxidizer to the phosphoric acid to increase uranium recovery in subsequent steps. Following the flow of raw material into the plant, this is the second operational change known to have occurred as a result of AEC work. Several oxidizers were tested and chlorine was chosen to be added to the phosphoric acid (Stoltz, Jr. 1958). Monosodium phosphate was produced from the acid. For the AEC work, Blockson added a filter, tank and other equipment to increase capacity so all Blockson phosphate could be diverted through the uranium recovery process (Blockson 1951). Former workers have indicated that the feed to Building 55 came from Building 40 (OCAS 2007a). Therefore, since chlorination was performed on the acid and the building equipment was altered somewhat, Building 40 is considered in this profile for the purpose of dose reconstruction.

Laboratory work was performed to develop the Blockson process for uranium extraction. All of the AEC funded laboratory work on process development leading up to construction and operation of Building 55 was documented in a report (Blockson 1953a). Specifics on the laboratory facility are not available, although the process work is described in detail. The work consisted of various procedures and reagents used to precipitate uranium from the existing processes at Blockson. The source (phosphate rock and derived products) contained the same radioactive constituents as the product processed in the areas described above and eventually in Building 55. During the operational period Blockson also analyzed samples collected from the pans used to dry the uranium concentrate in Building 55 (Blockson 1953b).

A pilot plant was constructed concurrent with the laboratory developmental work. Pilot plant work and results are also documented by Blockson (1953a). Details are provided that include parameters used for process steps, dates of operation, and uranium analytical results for the various procedures and steps being studied. The laboratory work and pilot plant operations have been considered in this site profile for the purpose of dose reconstruction.

## 2.2 Process Development

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 2007). Laboratory studies began at Blockson at about that time (Stoltz, Jr. 1958) and, due to the urgency of the program, pilot plant construction was begun simultaneously with the start of laboratory work (Blockson 1953a). Documentation indicates that pilot plant runs lasted about three weeks, ran 24 hours a day at 25% of expected production capacity, and that several runs were performed: April 29, 1951, to June 21, 1951; July 23, 1951, to August 10, 1951; and November 25, 1951, to January 6, 1952 (Blockson 1953a). Various uranium recovery methods were investigated and tested through pilot plant runs, and by July 1951 Blockson had determined the most effective process (Blockson 1951). The building in which Blockson constructed the pilot plant is not known.

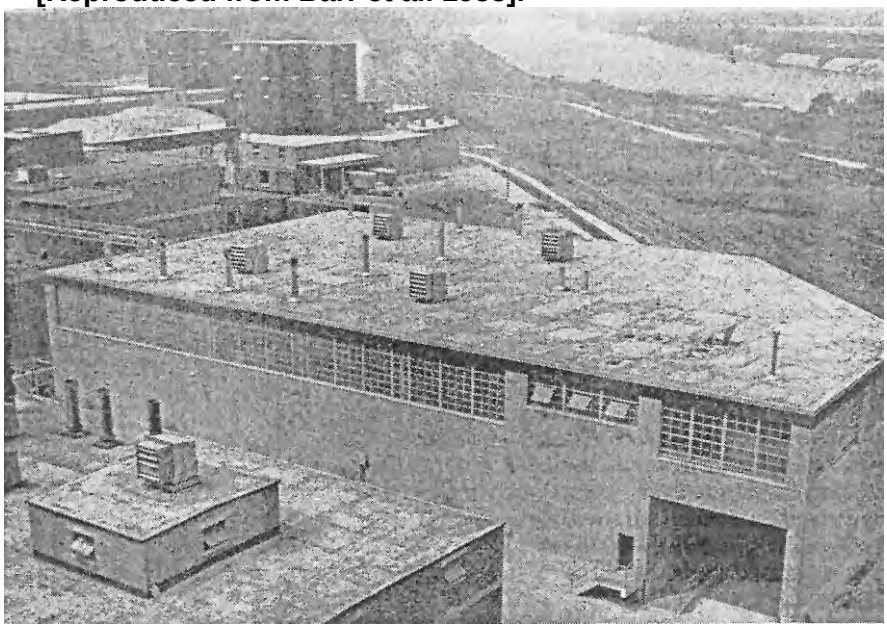
Blockson sent a letter to the AEC on July 31, 1951 (Blockson 1951) in which the uranium recovery process was summarized. The letter included best estimates of production, fixed capital requirements, manufacturing costs, and general contract conditions which would be acceptable to Blockson Chemical Company. This letter states that all process equipment could be housed in the new building (Building 55) with the exception of an 8x12 filter, a storage tank, and pumps. The new filter was necessary to enable all Blockson‘s liquors to be processed through the monosodium phosphate step, which was the liquor to be diverted to Building 55. A monosodium phosphate liquor storage tank was added to provide storage capacity for monosodium liquors from which the uranium had been removed. These two items, with associated pumps and other accessories, were to be located in their existing plant buildings where those operations were already being performed.

The letter contract was later replaced by contract number AT(49-1)-611 on October 18, 1951 (AEC 1951). Under the contract Blockson began construction of Building 55 at its own expense to house uranium recovery equipment at their plant in Joliet, Illinois. Laboratory work and pilot studies continued during construction of Building 55 in efforts to further improve uranium recovery processes, but no significant changes in the process were made.

## 2.3 Uranium Recovery Operations

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 (Stoltz, 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 shown in Figure 1.

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



Blockson Chemical Company produced technical phosphates rather than fertilizers from wet phosphoric acid (Wilkinson 1976). In the Blockson process, the phosphate rock was calcined, pulverized, and then digested with sulfuric acid resulting in phosphogypsum and phosphoric acid. Blockson modified its calcining procedure to decrease losses of uranium to the phosphogypsum. The phosphoric acid partitioned about 85% of the uranium while the phosphogypsum partitioned most of the calcium and radium (Blockson 1951, Stoltz, Jr. 1958).

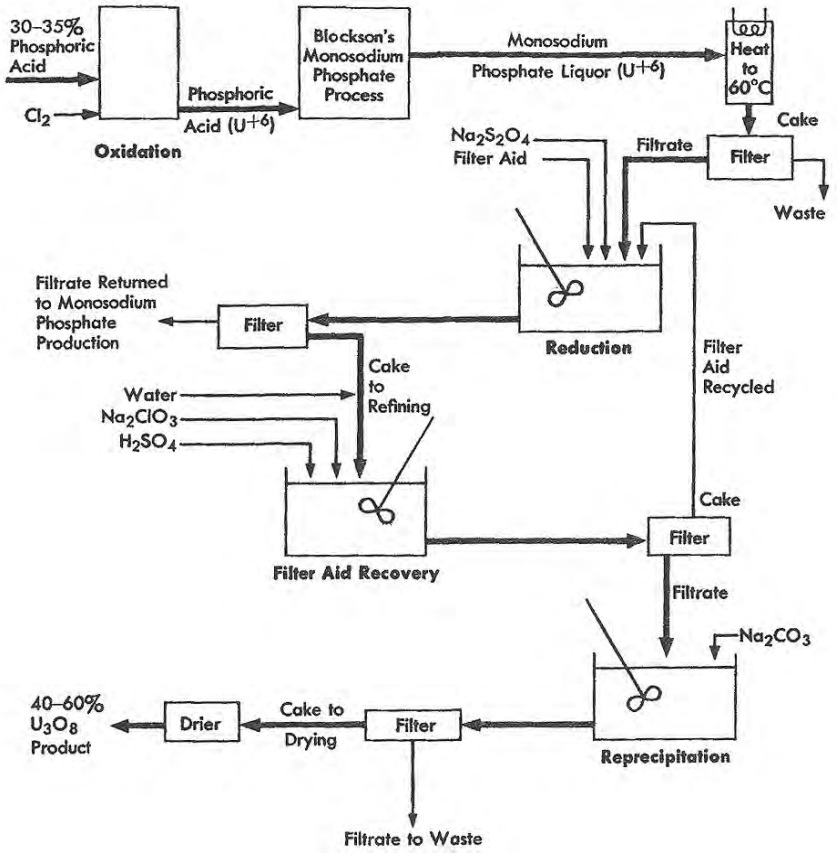
The phosphoric acid was partially neutralized with soda ash to form monosodium phosphate. The acid contained uranium in both the uranous (4+) and uranyl (6+) state. During neutralization of the acid uranous ions tended to partially precipitate when the pH was increased. To increase the recovery of uranium in the monosodium phosphate liquor, Blockson added a chlorination step to the phosphoric acid. Calcium, aluminum and iron salts were formed during conversion, were filtered off, and about 90% of the uranium was recovered in the liquor. The monosodium phosphate liquor was pumped to the uranium recovery building (Building 55) (Blockson 1951, Blockson 1953a, Stoltz, Jr. 1958).

The uranium by-product was precipitated from the monosodium phosphate stream that was diverted to Building 55. In Building 55 monosodium phosphate was ~~h~~heated and clear pressed to remove any fine suspended solids" (Blockson 1951). Sodium hydrosulfite ($Na_2S_2O_4$) was added to precipitate the uranium. A filter aid was also added to help remove the precipitant from solution. The filtrate was returned to the phosphate-processing plant and the precipitate, containing about 5% $U_3O_8$, was further processed to up-grade the product. The precipitate was slurried in a weak sulfuric acid solution with a small amount of chlorine which resulted in a gradual dissolution of the uranium. The filter aid was then filtered off and the filtrate partially neutralized to precipitate the uranium. The uranium was then filtered as uranous phosphate containing 40% to 60% $U_3O_8$. The product was dried and packaged for shipping (Blockson 1951, Stoltz, Jr. 1958, Clegg and Foley 1958). The material precipitated in Building 55 has been

reported to be a fine powdered sodium uranium compound (Wimpfen 2002). Other reports have indicated that the precipitated compound was sodium di-uranate (Cope and Sinclair 1952) and a uranous phosphate (Clegg and Foley 1958). The uranium content of the phosphate rock consumed in these processes was reported to be as high as 0.014% $U_3O_8$ (Stoltz, Jr. 1958).

Figure 2 shows a schematic flowchart of the Blockson uranium recovery process.

**Figure 2 The Blockson Chemical Process for the Recovery of Uranium from Phosphoric Acid [reproduced from Clegg and Foley 1958].**




Accounts from former Building 55 workers (OCAS 2007a) indicate that the precipitated and filtered uranium concentrate in Building 55 was placed in pans; pictures and descriptions indicate the pans nominally measured two and one half feet by two and one half feet by a couple inches high. The pans were placed in a drier overnight and then emptied into a drum. This was a manual operation that was normally performed on the day shift when there were additional workers. The former workers are in general agreement that there were about two to six full time Building 55 workers per shift, with the night shift being minimally staffed. Blockson's pre-operational labor estimates submitted to the AEC allowed for two operators and one chemist per shift and allowed for two additional men on day shift. Additionally, they estimated an average of two full time mechanics and budgeted for a clerk, a developmental chemist, and a foreman (Blockson 1951). The former workers are in agreement that a number of maintenance people were in and out of Building 55 performing various jobs within their area of expertise (OCAS 2007a).

A pre-operational letter from Blockson to the 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. Information from a former AEC official (OCAS 2007c) indicates that from September 1952 through June 1960, a total of 118.3 tons $U_3O_8$ had been produced for the AEC. This equates to a little less than 3,000 pounds of $U_3O_8$ per month through June 1960. This is consistent with a 3,000 pound monthly production estimate from 1958 found in a letter to the Feed Material Production Center, which was receiving the Blockson product at that time and sampling it to determine delivery quantities for payment purposes (AEC 1958a). Additional AEC documentation indicates that the production period at Blockson ended in June 1960 with a total of 118.3 tons of $U_3O_8$ purchased by the AEC (AEC c1963).

There is contradictory information in some literature regarding the total uranium production through 1955. A Department of Energy Formerly Utilized Sites Remedial Action Program (FUSRAP) Report (DOE 1985) indicates that 1.22 million pounds of uranium concentrates had been produced by the end of calendar year 1955. Some newspaper accounts have reported that Blockson produced about 2 million pounds of uranium throughout the AEC contract period. The amount reported in the FUSRAP report is apparently an overestimate by a factor of 10 based on the total production through 1955 as described above from detailed AEC production reports. For example, in order to produce 1.22 million pounds of $U_3O_8$ in that timeframe, Blockson would have had to double the capacity of the entire facility, used phosphate ores with the highest known uranium content (0.03%), and extract 100% of the uranium throughout each of its processes. In reality, uranium content of Florida phosphate ore averages 0.011% and maximum recovery for uranium at Blockson was estimated to be 73%, which results from recoveries of 85% for rock digestion, 90% for monosodium phosphate precipitation, and 95% for $U_3O_8$ precipitation and upgrading steps (Blockson 1951). Those values, along with a throughput of 6,000 tons per week were used to estimate the production at 50,000 pounds of $U_3O_8$ per year. In early 1953 Blockson reported up to that time they had actually achieved a recovery of 60% - 70% (Blockson 1953a). The 1955 AEC reports and the quantities produced through June 1960 (OCAS 2007c) indicate that, on average, Blockson produced less than 35,000 pounds per year, which is less than the pre-production estimate of 50,000 pounds per year. Documentation also indicates that on at least one occasion uranium recovery at Blockson was less than planned. Blockson attributed these lower recoveries to differences in the phosphate rock feed received by the plant.

In 1955, Blockson was sold to the Olin Mathieson Chemical Corporation (later renamed Olin Corporation), who assumed the liabilities and obligations of Blockson. As stated in contract number AT(49-1)-611 Amendment 1, Olin continued the uranium recovery program under contract with the AEC.

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.

According to former Blockson workers the AEC maintained a guard and controlled access to Building 55. The workers also indicated that the security was eased in the late 1950s (OCAS 2007a). A former worker provided a copy of an agreement effective January 15, 1957, between Blockson and the International Chemical Worker's Union Local No. 4. Article 33 of that agreement discusses the elimination of security provisions for Department 55 and provisions for pay and movement of union personnel into and out of that department. Relaxing of security requirements was also indicated in a May 1955 AEC memorandum which said a new security classification was being written for uranium phosphate operations that "may completely declassify everything except overall production data" (Robinson 1955). It is not known exactly when AEC security requirements were eased at Blockson, but the statements from the former workers are supported by the AEC's plan to declassify some of the operations.

## 2.4 Radiological Data

The only radiological data from the AEC operational period that is known to exist are bioassay results for uranium from 1954 through 1958. The bioassay results are discussed in Section 3.

Personnel with the FUSRAP Program conducted record 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 FUSRAP program also conducted extensive monitoring of Building 55 in 1978. That data is discussed in other sections of this document.

The Olin Corporation has provided radiological data from surveys conducted in 1983 and 1996 (Olin 2007). The 1983 data consists of total dust measurements, airborne alpha radioactivity measurements, and radon working level (WL) measurements. The 1983 measurements were from several areas but none from Building 55. NIOSH has also obtained copies of 1993 radon flux measurements of the phosphogypsum stacks that were performed for Olin (Olin 1993).

There are two sets of radiological data from 1996. One set of 1996 data is from RSSI who wrote a Radiological Safety Manual for demolition of other buildings unrelated to Building 55. The other 1996 data is from a survey conducted in June by Radiation Safety Associates. The survey was performed on Building 55, which was partially dismantled at the time. Surveys were of various building and equipment components and surrounding soils. Isotopic data from those surveys are discussed in the internal dose section. The contamination levels from that survey are consistent with the values and assumptions used in estimated dose from residual contamination based on the 1978 survey discussed in the section on residual contamination.

## 3.0 Internal Dose

Because phosphate rock contains Naturally Occurring Radioactive Material (NORM), work involving phosphate rock potentially exposes workers to radioactivity. This includes potential exposure to airborne uranium and its progeny, and other radiological constituents that may be present. Handling phosphate rock prior to chemical processing could have produced an airborne concentration of uranium dust that could contain all of its progeny in equilibrium. Internal dose has been evaluated based on two likely exposure scenarios: dose from uranium extraction work, and dose from other work.

The primary uranium isotopes in the phosphate rock are U-238, U-234, U-235 and associated progeny. Dosimetrically significant progeny include Th-230, Ra-226, Rn-222 (radon) and associated progeny. Trace amount of natural thorium (and associated progeny) are also present in phosphate rock (FIPR 1995).

Uranium extraction work potentially exposed workers to uranium that was significantly concentrated. Uranium concentration in the precipitated final dried product produced for the AEC was about four thousand times higher than the concentration in the phosphate rock received by Blockson. Reports of the average uranium concentration in the phosphate rock used by Blockson ranged from 0.011% to 0.014%, while the uranium concentration in the final precipitation in Building 55 was reported to be 40% to 60% (Blockson 1951, Stoltz, Jr. 1958). Other steps in the extraction process produced a much lower concentration of uranium. The uranium extraction work occurred in the laboratory (for research and development, and for analyses), in the pilot plant (for production scale testing of laboratory results), and finally full scale operations in Building 55. Building 55 operated at a much larger capacity and potentially exposed workers to much larger quantity of uncontained radioactive material. Therefore, Building 55 intakes are used to bound intakes from uranium extraction work. Other radionuclides in the uranium extraction work that could have significant impact on internal dose are also considered.

The second internal dose exposure scenario considered in this evaluation is the maximum dose likely to have been received outside of uranium extraction operations. The most significant radioactive elements at Blockson that did not follow uranium through the recovery process are radium and polonium. Most of the radium and polonium was precipitated and filtered off with the calcium during phosphoric acid production in Building 40, which was slurried and pumped to the phosphogypsum piles. Ra-226 is not as significant for internal dose as some of the other uranium progeny present in the rock, such as Th-230. Dose from Th-230 (and other thorium isotopes and progeny that are assumed to report to the phosphoric acid) is accounted for by assuming all of the thorium follows the uranium to Building 55, and is concentrated to produce a more significant internal dose hazard. Handling the radium bearing phosphogypsum was a wet operation and would produce a much less significant airborne concentration than handling dry product, although some dry material would be expected. Significant amounts of dry material would not have accumulated in the plant over time due to Blockson requiring each shift to hose down the work areas (OCAS 2007a). Bounding internal doses from non-uranium extraction work is based on work that was likely to produce the highest airborne radioactivity

concentrations, which was handling dried phosphate rock prior to chemical processing that removed some of the radiological components.

## 3.1 Radionuclide Source Term

Blockson Chemical processed phosphate rock that was mined in Florida. This rock then underwent a number of processing steps, including ore receiving and storage, calcining, crushing, digestion to produce phosphoric acid that was used to produce other compounds of phosphorous, as well as the by-product uranium (Stoltz, Jr. 1958). Hull and Burnett (1996) reported that U-238 and the various progeny are in approximate equilibrium in phosphate rock. Steps that involve only physical processing of the materials, e.g. crushing and storage, do not typically alter the relative concentrations of uranium, thorium, radium, or other radionuclides in the raw materials. However, chemical processing, e.g., phosphoric acid production, may alter this equilibrium and the various radioisotopes will fractionate according to chemical properties (FIPR 1995). Consideration must be given to the effect chemical processing has on the ratios of the various radionuclides present in the phosphate rock.

It can be assumed that the various isotopes of a particular element will react similarly to chemical processing if they are in the same physical and chemical form. For example, if all of the thorium isotopes are present in the phosphate rock in a similar matrix distributed throughout the rock, then it can be assumed that all isotopes of thorium will remain in the same relative concentration as the rock is chemically processed. This may not, however, be a valid assumption if some thorium isotopes are bound in a different matrix in the phosphate rock. FIPR 1995 reported in their study that unreacted minerals in the matrix account for an estimated 5-10% uncertainty in the fraction of the radionuclides reporting to the phosphogypsum.

The distribution of specific uranium and thorium decay chain radionuclides within phosphate source materials and within the various products and waste streams produced by the phosphate ore processing industry has been the subject of numerous studies. While the distribution of radionuclides is in some respects a function of the specific process involved, some generalizations may be made specific to the process employed at Blockson:

- Radiological equilibrium in the uranium chain appears to be maintained in rock that has not been chemically processed (Roessler 1979, FIPR 1995).
- Ra-226 and Po-210 are retained in the phosphogypsum, i.e., do not enter the phosphoric acid stream to any significant degree (Roessler 1984 p7; Guimond 1975 p15; FIPR 1995 p. 1-16).
- Uranium and thorium tends to favor the phosphoric acid phase (Roessler 1984; Guimond 1975, FIPR 1995).
- Since Th-230 is present in the matrix with U-238, it is expected to go into solution along with the uranium when leached in sulfuric acid. Th-232, if occupying a different matrix in the rock, may not be as readily dissolved in sulfuric acid (Coppinger 1959 p. 20).
- Pb-210 is reported by some authors (FIPR 1995 p 1-16) as being retained in the phosphogypsum and by others as reporting to the phosphoric acid (Roessler 1984 p7).

When Olin dismantled Building 55 in 1996 they stopped work and performed radiological surveys and took four media samples of debris in Building 55. They also took several soil samples around Building 55. These samples were obtained when the building had equipment, concrete, roof, and other structures partially demolished. The samples were analyzed by gamma spectroscopy and results were reported for a number of radionuclides in the U-238, U-234, and Th-232 (natural thorium) chain. Soil sample results were typically low. The four samples from inside Building 55 were taken of spots that were identified as contaminated based on scan results. These data are insufficient to form a quantitative conclusion on the ratios of the various radionuclides that were present, but the data indicates elevated uranium with some natural thorium, but at a much lower concentration. Ra-226 was not reported as being detected in those samples, nor was Pb-210, but low concentrations of Pb-214 and Bi-214 were indicated. Ra-226 and Pb-210 were reported in low concentrations in some of the soil samples. One Building 55 sample from the radiological characterization of the Blockson site described as a _yellow powder' had detectable quantities of Th-230 (Olin 2007).

This data indicates that uranium isotopes were present with lesser amounts of natural thorium. Radium and lead were not reported from inside Building 55 (Pb-212 was reported on one sample and is expected due to ingrowth from natural thorium). However, this data is not considered sufficient to quantify the amounts of the various radionuclides to which workers could have been exposed during and after uranium extraction operations.

The Blockson process was specifically engineered to maximize carryover of uranium into the phosphoric acid phase. Other elements would be present at various stages according to their chemical properties. There are uncertainties with chemical recoveries and potential losses of some elements in some of the chemical steps. In lieu of this uncertainty, an exposure model is provided that makes assumptions that result in limiting doses from the various radionuclides that are present in the feed material.

Assumptions for isotopic ratios in the phosphoric acid stream and monosodium phosphate pumped to Building 55:

1. 85% of U reports to phosphoric acid (Blockson 1951, Blockson 1953, Stoltz, Jr. 1958).

2. 4% of Ra-226 reports to acid (Hull and Burnett 1996). Various publications have reported lesser or more percentage goes to the acid. Roessler et al., 1979, reported 1% reports to the acid; other studies have reported virtually no radium reports to the acid, while some have reported more. This document assumes 4% based on Hull and Burnett 1996. No Ra-226 was detected in the surveys done in the later years, although Th-230 and natural thorium was detected as discussed above.

3. Thorium reports to the acid in same proportion as uranium. Several references indicate the thorium is likely to be somewhat lower than uranium. The assumption of equal recovery of thorium to uranium in the acid results in a higher source term for internal and external dose modeling. If there were more thorium losses to the phosphogypsum stream the doses would be lower.

4. U-238:Th-232 radioactivity ratio in Blockson's rock was 30:1. This is considered a bounding ratio to allow for natural thorium and progeny based on reported U-238 and Th-232 concentrations in phosphate rock (ORAU 2006). Th-232 progeny are assumed to be in equilibrium. Although most of the Ra-228 would have been separated and removed with the phosphogypsum, it is assumed to be in equilibrium with Th-232 for dose modeling to allow for ingrowth over the operational and residual contamination period.

5. Pb-210 is assumed to report to the acid the same as U-238 (85%). The various references cite data indicating lead reports to the phosphogypsum, while others report high percentages reporting to the phosphoric acid. For modeling purposes, Bi-210 and Po-210 are assumed to be equal to Pb-210 for the purpose of bounding exposures and intakes from ingrowth over the operational and residual period.

6. All isotopes reporting to the acid are carried through to the drum of dried uranium concentrate (highest potential source for internal dose and source for external dose model) in the same relative concentration as U-238.

These assumptions result in the following ratios for the liquor sent to Building 55:

**Table 1: Building 55 Relative Radionuclide Concentrations.**

| Radionuclide | Relative Ratio[1] | Notes | Normalized to U-238[1] |
|---|---|---|---|
| U-238 | 85 | Progeny in equilibrium through Th-230 | 1 |
| U-235 | 3.87 | Progeny in equilibrium | 0.0455 |
| Ra-226 | 4 | Progeny in equilibrium | 0.047 |
| Pb-210 | 85 | Equal to U-238 | 1 |
| Bi-210 | 85 | Equal to U-238 | 1 |
| Po-210 | 85 | Equal to U-238 | 1 |
| Th-232 | 2.8 | Progeny in equilibrium | 0.033 |

1. Ratios given are for progeny without consideration of branching ratios, where applicable.

### 3.2 Uranium Intakes

Intakes of radioactive material have been evaluated for uranium extraction operations and for other work at the Blockson Plant, some of which is considered to be associated with uranium recovery work, although not directly involved with uranium extraction. The calciner has been chosen to provide bounding intakes for non uranium extraction work as discussed below; it can be considered an associated activity because Blockson made a change to the procedure to improve uranium recovery in other processes.

### 3.2.1 Calcining

The 1983 survey (Olin 2007, p.7) provided results of 12 general area and personnel breathing zone total dust measurements at several locations in the Blockson plant, none of which are from Building 55. The maximum reported total dust result from the 1983 study was 6.37 mg/m$^3$ and was associated with TSP operations. Gross alpha measurements of the air samples were all less than the sensitivity of the equipment except for a filter platform in Building 40. All of these results are less than the intakes provided below based on a more extensive study done at another wet process phosphate plant. Therefore, the maximum result from the study described below is used to bound intakes at Blockson from work outside uranium extraction operations.

The US EPA performed a thorough study of dust loading and radionuclide concentrations in air throughout phosphate ore processing at an Idaho Phosphate facility (EPA 1978). The facility utilized the wet-process method to process phosphate rock, the same method used by Blockson. Various air samples were taken at locations throughout the plant. The samples were analyzed for total dust loading and airborne radioactivity concentrations. The atmospheric dust concentrations were not provided, but have been calculated from data. Specifically, the total reported mass of dust on the filter paper in mg was divided by the volume of air sampled to derive the atmospheric dust loading in units of mg/m$^3$, which are listed in Table 2. The highest dust concentration in that study was from Calciner #3, which was 50.4 mg/m$^3$, indicative of an operation with likely visible dust. Information in the report indicates that it was located in the Calciner Building, although no other information on the calciner was provided in the report. Samples were also taken from several locations in the phosphoric acid building.

**Table 2: Dust Concentrations in Air during Phosphate Industry Processes**

| Material | Dust Loading (mg/m$^3$) |
|---|---|
| **Grinder mill** | 6.29 |
| Ore unloading, storage | 5.43 |
| Calciner control room | 1.73 |
| Calciner | 50.4 |
| Phos. Acid digester | 15.3 |
| Outside control room | 6.5 |
| Continuous filter | 2.72 |
| Control room | 1.5 |
| TSP storage | 1.37 |
| **TSP dryer** | 33.9 |
| **Acidulation TSP disch.** | 2.63 |
| 200 Ammophos plant | 18.6 |
| 100 Ammophos plant | 8.68 |
| Library | 0.92 |
| **100 plant storage** | 8.61 |
| **200 plant storage** | 4.12 |

Blockson operated a calciner to break down organic matter and to roast the rock into a form that optimized recovery of uranium and for other processing purposes unrelated to uranium extraction work. Their calciner was a large outdoor furnace that utilized a dust collector (OCAS 2007b). The dust concentrations at Blockson‘s calciner would be expected to be lower than the one listed

in Table 1 because of dilution with open air. The use of a dust collector at Blockson also indicates that some effort was made to control dust. Although it could be expected that some localized areas may have had high concentrations for short durations, such as when sampling the calcined rock, average dust concentrations would be expected to be lower than that reported for the operations at the phosphate plant discussed above. Therefore, these dust concentrations are being used to determine bounding dust concentrations for operations at Blockson outside of Building 55.

Intake quantities for uranium and thorium nuclides (and associated progeny) were calculated based on continuous exposure to an airborne dust loading of 50.4 mg/$m^3$. Bounding intakes from uranium and thorium (and associated progeny) were calculated based on a breathing rate of 1.2 $m^3$/hr and being exposed to that high level of dust for 2000 hours per year. For these calculations the uranium mass was assumed to be all U-238, as it represents over 99% of the uranium mass in natural uranium. U-234, Th-230, Ra-226, and other nuclides in the U-238 chain are assumed to be in equilibrium with U-238 during phosphate rock crushing and calcining operations. U-235 was not included because the method of calculating uranium activity allowed for the small intake contribution from U-235. Additional parameters used include an assumed 0.014% uranium content in phosphate rock (Stoltz, Jr. 1958). Intake values have been normalized to calendar days. Ingestion intakes were also calculated using the inhalation intake quantities and applying the methodology in OCAS-TIB-0009 (*Estimation of Ingestion Intakes).*

Polonium has a boiling point of 962$^o$C, or 1764$^o$F (Webelements 2007). Calcining temperatures in other phosphate facilities are known to have increased airborne particulate concentrations of Po-210 due to the high temperature used in the furnace. Radiological data from the Idaho phosphate facility calciner mentioned above indicated that the Po-210 concentration in air at the calciner was 10 times higher than the U-238 concentration (EPA 1978). Elevated Po-210 concentrations from phosphate facility calcination have also been reported by Boothe (1977). Therefore, the intakes of Po-210 from calcining are estimated to be 10 times higher than intakes of U-238. Intakes of concern for internal dose are listed in Table 3.

**Table 3: Inhalation and Ingestion Rate for Calcining.[1,2]**

| | **Inhalation** | **Ingestion** |
|---|---|---|
| **Radionuclides** | **pCi/day** | **pCi/day** |
| U-238, Th-230, U-234, Ra-226, Pb-210 | 16 | 0.47 |
| Po-210 | 160 | 4.7 |
| Th-231, Pa-231, Ac-227[3] | 0.73 | 0.021 |
| Th-232, Ra-228, Th-228 | 0.52 | 0.016 |

1. Intake rates have been normalized to calendar days.
2. Both inhalation and ingestion should be assigned for calcining operations.
3. U-235 is allowed for in the U-238 and U-234 values. Values given are for radionuclides in the U-235 chain.

Except for Po, Pb, and Ra, the solubility class in the calcined phosphate rock may be Type M or S and should be selected to be claimant-favorable. Po is either Type F or M, Pb is assumed to be Type F, and Ra is assumed to be Type M (ICRP 1994b). The inhalation solubility class and

ingestion absorption values should be selected based on values that result in the highest organ dose.

### 3.2.2 Building 55

There are no air sampling results available from Blockson to characterize airborne radioactivity concentrations, however, air sampling results from mills with some common activities have been published. The greatest potential for internal uranium exposure in the Blockson uranium recovery process is most likely 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).

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 in drums (Clegg and Foley 1958, Blockson 1951, OCAS 2007a).

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.
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 if the barrels at Blockson were vibrated.)
4. Powder sampling. This occurs when a worker takes a sample of yellowcake for laboratory analysis. At Blockson this was done prior to the pans of yellowcake being dumped into the barrels.
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 mg 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 descriptions at Blockson are 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 handled and packaged dried yellowcake routinely. 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 is 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 hour 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 ―aboutwenty 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 worker's 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 intake recommendations in this evaluation are favorable in comparison to the data for those workers.

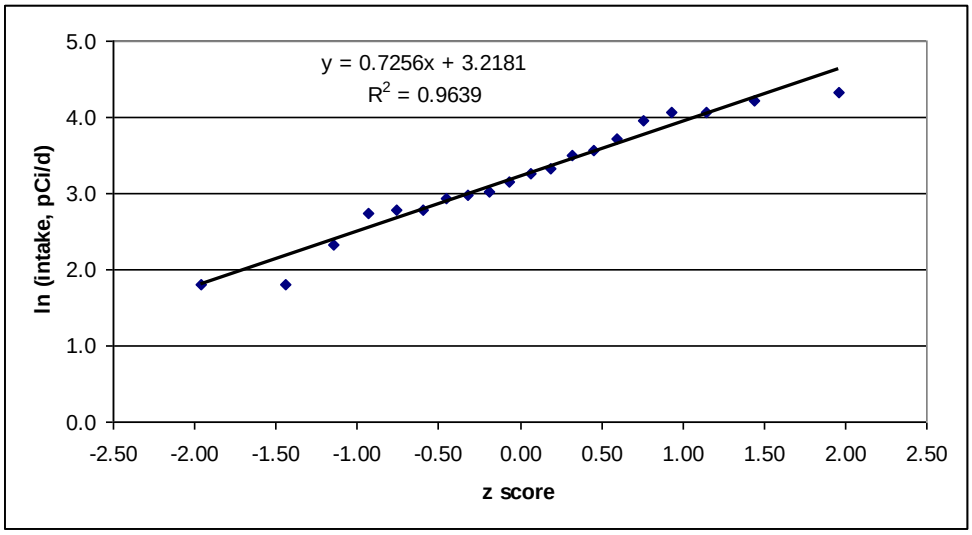
The International Commission on Radiological Protection (ICRP) recommends three material types for solubility of inhaled uranium, Types F, M, and S, based on the clearance rate from the lungs (ICRP 1994a). Various studies have shown that $U_3O_8$ closely corresponds to the clearance rate associated with 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 for uranium extraction 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 1953a). Calcination was done on phosphate rock prior to chemical processing and the intakes of highly insoluble Type S material from calciner could have occurred, as described in Section 3.2.1. After the calcined rock was digested in sulfuric acid and oxidized, no additional calcining or high firing of the uranium material was performed in the process used in uranium recovery in Building 55. 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 is discussed below.

**Figure 3: Distribution of Uranium Intakes.**




Production workers are assumed to have been 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. The total uranium intakes are applied as 50% U-238 and 50% U-234, which allow for intakes of U-235. Intakes of long-lived radionuclides that are significant for internal dose are listed in Table 4a. These bounding intakes should be entered into IREP as constants.

**Table 4a: Inhalation Rate for Building 55.[1,2,3]**

| Radionuclides | Intake (pCi/d) Production workers | Intake (pCi/d) Administrative workers |
|---|---|---|
| U-238, Th-230, U-234, Pb-210, Po-210 | 41 | 13 |
| Th-231, Pa-231, Ac-227[4] | 1.9 | 0.59 |
| Ra-226 | 1.9 | 0.59 |
| Th-232, Ra-228, Th-228 | 1.4 | 0.41 |

1. Intake rates have been normalized to calendar days.
2. Intakes are based on Type M lung solubility for materials likely to have been present in Building 55 operations except for thorium, lead and polonium. Pb-210 is Type F, and Po-210 is Type F or M per ICRP 1994b. Thorium could have been Type M or Type S. Thorium and polonium solubility types must be selected based on the types that provide the largest dose to the organ or tissue of concern.
3. See Table 4b for dose to tissues of the gastrointestinal tract.
4. U-235 is allowed for in the U-238 and U-234 values. Values given are for radionuclides in the U-235 chain.

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 results 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 4b 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. Since bioassay results are from intakes by all pathways, a worker should be assigned Building 55 intakes from inhalation or ingestion, not both.

**Table 4b: Ingestion rate for Building 55.[1,2]**

| Radionuclide | f1 Values[4] | Intake (pCi/d) Production workers | Intake (pCi/d) Administrative workers |
|---|---|---|---|
| U-238, U-234,<br>Th-230<br>Pb-210<br>Po-210 | 0.02<br>0.0005<br>0.2<br>0.1 | 139 | 41 |
| Th-231, Pa-231, Ac-227[3] | 0.0005 | 6.4 | 1.9 |
| Ra-226 | 0.2 | 6.4 | 1.9 |
| Th-232, Th-228<br>Ra-228 | 0.0004<br>0.2 | 4.5 | 1.4 |

1. Intake rates are normalized to units of calendar days.
2. Ingestion intakes provide bounding dose to the stomach, small intestine, upper large intestine, lower large intestine, and colon. See Table 4a for estimating dose to all other tissues.
3. U-235 is allowed for in the U-238 and U-234 values. Values given are for radionuclides in the U-235 chain.
4. f1 values are from ICRP 1994b.

### 3.2.3 Pilot Plant and Laboratory

The location of the pilot plant operation that was built and used in 1951 and early 1952 has not been determined. As discussed in Section 2.2, the pilot plant operated to collect data on a production scale and operated at a much lower capacity than Building 55 and ran intermittently during developmental work. The plant tested methods determined from laboratory scale testing. The operations performed in the pilot plant were documented by Blockson (1953). Since this work involved the same source of radioactive materials as those in Building 55, except on a smaller scale, and only operated intermittently for short periods, the doses modeled for Building 55 workers are used to bound doses received by workers in the pilot plant operation and work in the laboratory.

## 3.3 Radon Exposures

Per the SEC determination, dose cannot be reconstructed from radon exposures occurring from March 1, 1951, through June 30, 1960. See Section 5 for radon exposures during the residual contamination period.

## 4.0 External Dose

Because phosphate rock contains Naturally Occurring Radioactive Material (NORM), work involving phosphates potentially exposes workers to radioactivity. 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 for the Blockson facility during the uranium extraction operational period. Exposure from operations outside of Building 55 has been considered based on studies of doses received by workers at similar plants, which are discussed in Section 4.1. Those studies did not include dose received from uranium extraction operations. Therefore, source term information has been used to estimate external doses to workers involved in uranium extraction operations as described in Section 4.2.

## 4.1 External Dose from Existing Operations

TLD data, evaluated by the Florida Institute of Phosphate Research, (FIPR 1998 tables 14-18) indicate that workers involved in various phases of phosphate rock handling, crushing and phosphoric acid liquor production received between 10 and 210 mrem (maximum) per year (at other wet process phosphoric acid operation sites). Eighty percent of the workers monitored received <10 mrem per year, 15% received >10 but <99 mrem/year and less than 2% received greater than 100 mrem/yr with 210 mrem (or 0.210 rem) being the maximum recorded dose. Radiological survey data available in FIPR 1998, Idaho 1978, and 1996 post site operations surveys of the Blockson plant in Joliet all support the TLD results (OLIN 1996). ORAU 2006 estimates an upper bound external dose to workers at wet process phosphate plants at 0.220 rem/yr. The doses would include external exposures to the radium bearing phosphogypsum that is filtered out during phosphoric acid production, but would not have included doses that have been received due to concentration of uranium, such as Blockson did in Building 55. The 0.220 rem/yr bounding external dose from other phosphate facilities is lower than the modeled external doses from Building 55 discussed in Section 4.2. Therefore, Building 55 modeled doses are considered to be an upper bound of the external dose from the entire Blockson facility.

## 4.2 External Dose from Uranium Extraction Operations

For the purpose of dose reconstruction from uranium extraction work in Building 55, it is assumed that there was a potential for external dose due to exposure from submersion in air, from exposure to contaminated surfaces and from exposure to barrels of the final concentrated uranium product.

Based on radioactivity air concentrations derived from the daily intake rates discussed in Section 3.2.2, and the dose coefficients from Federal Guidance Report No. 12 (EPA 1993), the external dose from submersion in air contaminated with radioactive dust is insignificant in comparison to the favorable evaluation of other dose components and, therefore, is not included in dose reconstructions.

### 4.2.1 Source Term

Although uranium and the short lived progeny of U-238 are the most abundant radionuclides in Building 55, the presence of other radionuclides that may have followed the uranium to Building 55 can produce significant external doses at much lower concentrations. Section 3.1 discussed the various radionuclides that are of significance at Blockson. For modeling external dose in Building 55, assumptions were made to allow for radionuclides that may have been present as contaminants in Building 55. Allowances were also made to consider ingrowth of some

radionuclides due to radioactive decay of parent radionuclides. Those assumptions are included in the Table 1 source term assumptions used to model bounding external doses in Building 55.

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]**

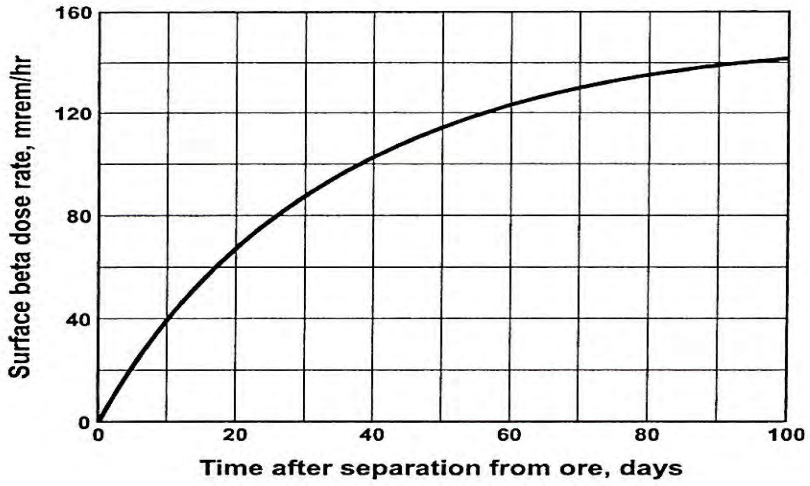
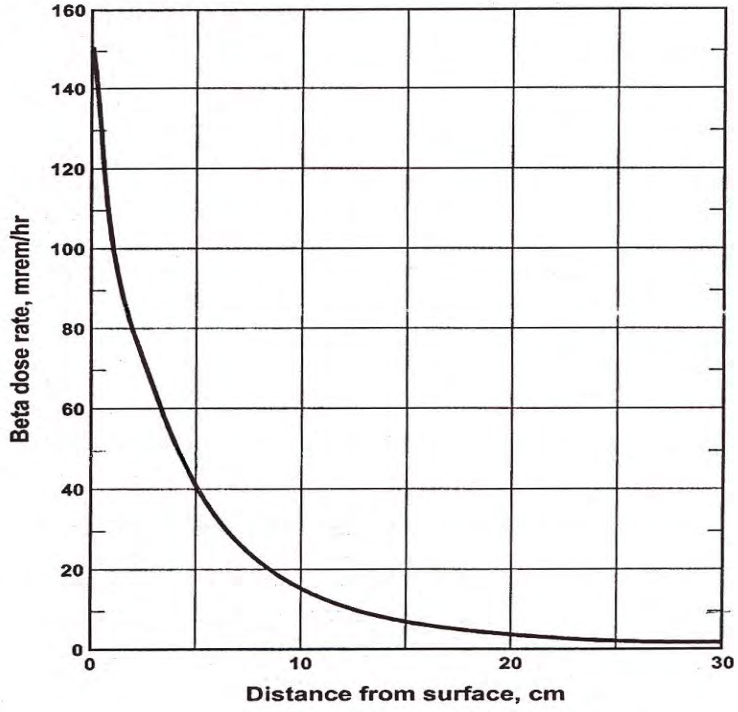


**Figure 5. Beta Dose Rate from Yellowcake Separated from Ore for More Than 100 Days as a Function of Distance from the Surface. [Reproduced from US NRC 2002a]**




### 4.2.2 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 radionuclide concentrations were calculated based on the ratio to $^{238}U$ for determination of the number of photons and electrons per decay of $^{238}U$. For the purposes of this evaluation, branching ratio adjusted equilibrium was assumed. A significant amount of progeny was included, including portions of the $^{232}Th$ chain, based on the discussion in Sections 3.1 and 4.1.2. The details are provided in Table 6.

**Table 6: Radioactivity Ratios for External Dose Modeling.**

| | Activity Concentration in aged Uranium Metal (Bq/g) | Relative activity concentration (normalized to $^{238}U$) |
|---|---|---|
| $^{238}U$ | 12200 | 1 |
| $^{234}Th$ | | 1 |
| $^{234m}Pa$ | | 1 |
| $^{234}Pa$ | | 0.0016 |
| $^{234}U$ | | 1 |
| $^{230}Th$ | | 1 |
| $^{226}Ra$ | | 0.047 |
| $^{222}Rn$ | | 0.047 |
| $^{218}Po$ | | 0.047 |
| $^{214}Pb$ | | 0.047 |
| $^{214}Bi$ | | 0.047 |
| $^{214}Po$ | | 0.047 |
| $^{210}Pb$ | | 1 |
| $^{210}Bi$ | | 1 |
| $^{210}Po$ | | - |
| $^{235}U$ | | 0.045492 |
| $^{231}Th$ | | 0.045492 |
| $^{231}Pa$ | | 0.045492 |
| $^{227}Ac$ | | 0.045492 |
| $^{227Th}$ | | 0.044855 |
| $^{223}Fr$ | | 0.000628 |
| $^{223}Ra$ | | 0.045492 |
| $^{219}Rn$ | | 0.045492 |
| $^{215}Pa$ | | 0.045492 |
| $^{211}Pb$ | | 0.045492 |
| $^{211}Bi$ | | 0.045492 |
| $^{211}Po$ | | 0.000126 |
| $^{207}Tl$ | | 0.045366 |
| $^{232}Th$ and progeny | | |
| $^{232}Th$ | | 0.033 |
| $^{228}Ra$ | | 0.033 |
| $^{228}Ac$ | | 0.033 |
| $^{228}Th$ | | 0.033 |
| $^{224}Ra$ | | 0.033 |
| $^{220}Rn$ | | 0.033 |
| $^{216}Po$ | | 0.033 |
| $^{212}Pb$ | | 0.033 |
| $^{212}Bi$ | | 0.033 |
| $^{208}Tl$ | | 0.01188 |
| $^{212}Po$ | | 0.02112 |

## Dose 30 cm from drum

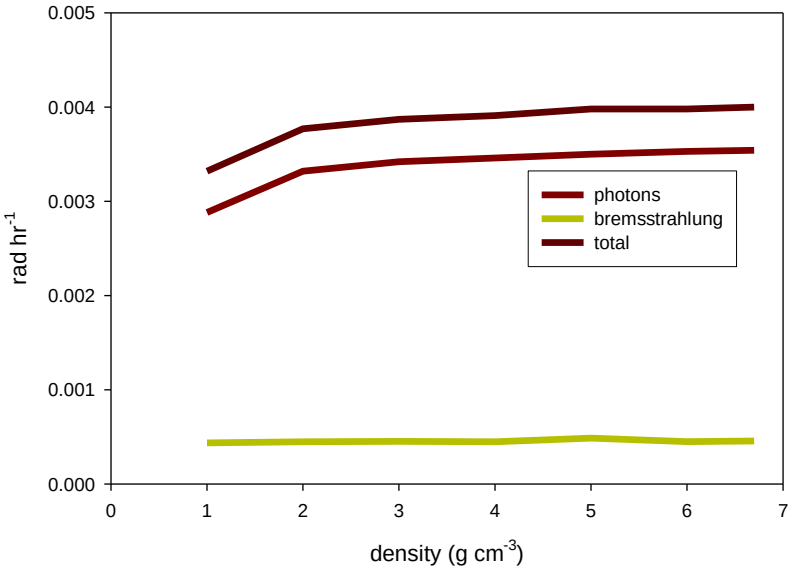
The dose rate was determined at 77.9 cm above the ground, 30 cm from the edge of the drum for both the photon and beta emissions of natural uranium and its progeny. ICRP Publication 74 Table A.1 was used to convert the photon flux to units of air kerma.

The effect 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 $cm^{-3}$. 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 $cm^{-3}$. Results are provided in Table 7 and Figure 6.

**Table 7: Dose Rates from Drums of Yellowcake with Thorium Contamination.**

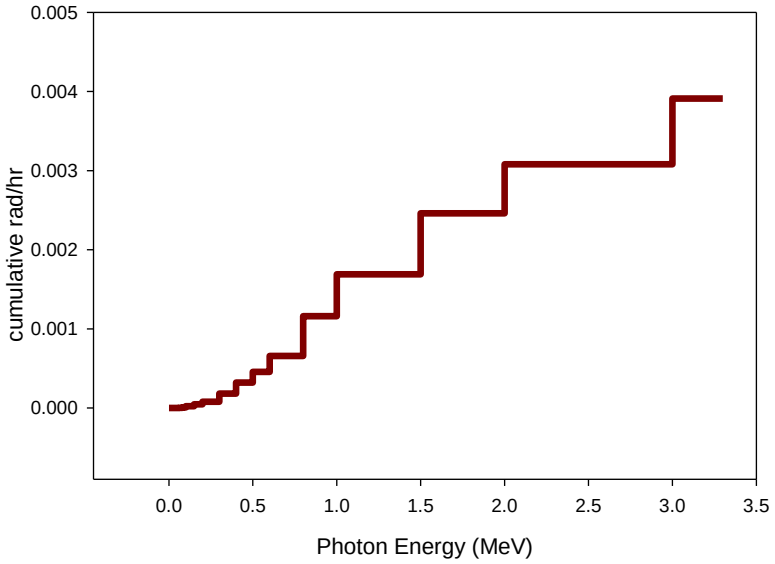
| Density of U3O8 (g $cm^{-3}$) | Activity of $^{nat}U$ in drum (Ci) | Photon emission dose(rad/hr) | Bremsstrahlung dose (rad/hr) | Total dose rate at 30 cm (rad/hr) |
|---|---|---|---|---|
| 1 | 6.242E-02 | 2.88E-03 | 4.37E-04 | 3.32E-03 |
| 2 | 1.248E-01 | 3.32E-03 | 4.48E-04 | 3.77E-03 |
| 3 | 1.873E-01 | 3.42E-03 | 4.52E-04 | 3.87E-03 |
| 4 | 2.497E-01 | 3.46E-03 | 4.48E-04 | 3.91E-03 |
| 5 | 3.121E-01 | 3.50E-03 | 4.86E-04 | 3.98E-03 |
| 6 | 3.745E-01 | 3.53E-03 | 4.49E-04 | 3.98E-03 |
| 6.7 | 4.182E-01 | 3.54E-03 | 4.57E-04 | 4.00E-03 |

**Figure 6: Effect of Density on Dose Rate at 30 cm from Drum of $U_3O_8$ Contaminated with Thorium.**



Cumulative energy distribution results are shown in Figure 7. Assignment of 10% dose from 30-250 keV photons and 90% dose from >250 keV photons was verified to be favorable to claimants as only approximately 1% 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$^3$.

**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 values 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 (30 cm) from the drum. This was normalized to 400 hours per work year. The 95th percentile exposure time was determined by assuming the worker spent 2000 hours per year at a distance of 1 foot from the drum. This results in a whole body dose distribution with a median value of 1.592 rad per year with a geometric standard deviation of 2.7.

To calculate organ doses for use in the NIOSH Interactive RadioEpidemiological Program (NIOSH-IREP) Monte Carlo methods were used to multiply the whole body dose and energy split times the triangular organ dose conversion factors for kerma to organ dose found in NIOSH External Dose Reconstruction Implementation Guideline (NIOSH 2006a). The results are annual doses that are lognormally distributed. The results are in Table 8. For skin, air kerma values were multiplied by an organ dose conversion factor of 1.0.

**Table 8: Annual Organ Dose from Exposure to Drums of Yellowcake.**

| | 30-250 keV photons | | >250 keV photons | |
|---|---|---|---|---|
| **Organ** | **dose, rem** | **GSD** | **dose, rem** | **GSD** |
| Bladder | 0.190 | 2.7 | 1.589 | 2.7 |
| RBM | 0.082 | 2.8 | 1.221 | 2.7 |
| Bone Surface | 0.216 | 2.7 | 1.338 | 2.7 |
| Breast (female) | 0.218 | 2.7 | 1.675 | 2.6 |
| Colon | 0.150 | 2.8 | 1.505 | 2.6 |
| Esophagus | 0.090 | 2.9 | 1.274 | 2.6 |
| Eye | 0.224 | 2.7 | 1.583 | 2.7 |
| Ovaries | 0.131 | 2.8 | 1.425 | 2.7 |
| Testes | 0.251 | 2.7 | 1.715 | 2.7 |
| Liver | 0.154 | 2.8 | 1.512 | 2.7 |
| Lung | 0.143 | 2.8 | 1.461 | 2.7 |
| Remainder | 0.123 | 2.9 | 1.357 | 2.7 |
| Stomach | 0.191 | 2.7 | 1.614 | 2.6 |
| Thymus | 0.221 | 2.8 | 1.637 | 2.7 |
| Thyroid | 0.241 | 2.7 | 1.744 | 2.7 |
| Uterus | 0.142 | 2.8 | 1.390 | 2.7 |
| Skin1 | 0.159 | 2.7 | 1.433 | 2.7 |

### 4.2.3 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 a U-238 airborne concentration of 4.3 $pCi/m^3$. 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 radionuclides 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 organs is insignificant in comparison to the favorable assignment of dose from drums of yellowcake. Beta dose to the skin is discussed in Section 4.4.

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. Direct reading radiation results are available 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/hr. 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 by assuming a worker was exposed to the 0.03 mR/hr rate for 2,000 hours per year, which results in an annual exposure of 60 mR, or 0.06 R. To allow for uncertainty this value is applied as the median of a lognormal distribution. 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.

The 1978 survey suggests that either the facility was only contaminated in localized spots, or it indicates that some areas of surface contamination were 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 1960. The 7 spots were on small areas of floor (reported to be about 0.5 $m^2$ each), 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 1960. 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 bounded by the assumptions made in deriving doses from drums of yellowcake. Therefore, dose from contaminated surfaces is only assigned during the residual contamination period.

### 4.2.4 Beta Dose

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. 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 an upper shallow dose of 0.8 rem per year with a geometric standard deviation of 2.7. The 0.8 rem per year was adjusted to allow for beta dose from other radionuclides that are assumed to be present in the uranium per the ratios in Table 6. The relative activity of each radionuclide was applied to Federal Guidance Report No. 12 dose conversion factors for skin for exposure to contaminated surfaces. Those factors indicate that U-238, Th-234, Pa-234m, Pa-234, and U-234 account for about 66% of the skin dose for the Table 6 ratios. Adjusted beta dose is provided in Table 9. The calculated beta doses

have not been reduced to allow for doses to areas of the skin that are typically covered by clothing that reduces beta dose to the skin.

It is also assumed that there was a potential for workers to receive a shallow beta dose to the skin contaminated with yellowcake. Skin contamination from contaminated air was estimated by using the measured skin deposition velocity of 4-µm particles to skin of 0.012 m/s (Andersson et al. 2002; Fogh 1999), assuming that the material was deposited on the skin for an entire 8-hour shift. The dose is negligible when compared to the shallow dose estimate from exposure to a drum of aged yellowcake discussed above and the estimated dose from contact with contaminated surfaces described below.

Skin dose has also been considered from contact with contaminated work clothing. Average dose data from contaminated clothing at Mallinckrodt indicate levels of 1.5 mrem/hour (AEC 1958b). The Mallinckrodt dose rate is used as a bounding condition for Blockson because Mallinckrodt handled materials of similar radiological constituents, but in larger quantities and with a higher radioactive material content. It is assumed that the workers were exposed to that level for 1000 hours per year, which is considered an upper bound condition. This results in a dose to the skin of 1.5 rem per year. Doses are applied as electrons > 15keV.

Former workers have said that the filtering operation in Building 55 exposed their hands directly to the filter cake that contained the uranium. They said they wore gloves for this work, but sometimes would have to take the gloves off and use their bare hands to remove the product from the filters (OCAS 2007a). Dose for this activity has been estimated for the hands and forearms. Yellowcake concentrations in the product delivered to the AEC was estimated to be 40% - 60%. To determine an upper bound dose, an estimate was made of shallow dose to the hands based on direct contact with pure yellowcake. Surface dose rates on yellowcake have been reported to be about 203 mrad per hour (DOE 2000). The time of direct contact has been assumed to be 2 hours per week, 50 weeks per year during the operational period. The 66% adjustment factor discussed above was applied to allow for non-uranium contaminants. This results in an annual dose of 30 rem to the hands and forearms. The dose applies to filter operators only and are applied as electrons > 15keV.

Table 9 contains a summary of shallow dose from electrons.

**Table 9: Beta Dose to Skin.**

| **Dose component Beta dose, E>15 keV** | **Annual dose[1]** | **Distribution** |
|---|---|---|
| Dose from drums of yellowcake | 1.2 rem per year | Lognormal, GSD=2.7 |
| Dose from contaminated clothing | 1.5 rem per year | Constant |
| Dose to hands and forearm from contact with yellowcake | 30 rem per year (filter operators only) | Constant |

1. Beta dose is applicable for the operational period only.

## 4.3 Occupational Medical Dose

Dose from occupationally required medical X-rays has 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 10 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 10 as the mean of a normal distribution with a 30% standard deviation.

**Table 10: 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 where the beam enters the body.

## 5.0 Dose from Residual Contamination

The whole body median dose rate of 0.060 R/year derived from the discussion in Section 4.2.3 above was used to calculate various organ doses. Photon dose is split 10% 30-250 keV and 90% >250 keV ranges. The Roentgen to organ dose conversion factors for isotropic geometry in the External Dose Reconstruction Implementation Guideline (NIOSH 2006a) were used to estimate organ doses. Skin doses were calculated using an organ dose conversion factor of 1.00. Results are in Table 11.

**Table 11: Annual Dose from Residual Contamination.[1]**

| | **Photons 30-250 keV** | **Photons >250 keV** |
|---|---|---|
| Organ | dose, rem | dose, rem |
| Bladder | 3.2E-03 | 3.5E-02 |
| RBM | 3.3E-03 | 3.6E-02 |
| Bone Surface | 5.6E-03 | 3.7E-02 |
| Breast (female) | 4.2E-03 | 4.0E-02 |
| Colon | 3.1E-03 | 3.4E-02 |
| Esophagus | 3.0E-03 | 3.5E-02 |
| Eye | 4.5E-03 | 4.1E-02 |
| Ovaries | 3.0E-03 | 3.4E-02 |
| Testes | 3.8E-03 | 3.7E-02 |
| Liver | 3.4E-03 | 3.6E-02 |
| Lung | 3.8E-03 | 3.8E-02 |
| Remainder | 3.3E-03 | 3.6E-02 |
| Stomach | 3.4E-03 | 3.6E-02 |
| Thymus | 3.7E-03 | 3.7E-02 |
| Thyroid | 3.8E-03 | 3.9E-02 |
| Uterus | 2.9E-03 | 3.3E-02 |
| Skin1 | 6.0E-03 | 5.4E-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 (OCAS 2007a). After cessation of uranium recovery work the main source of contamination (precipitated and dried yellowcake) was no longer present. Therefore, the derived median U-238 inhalation rate of 13 pCi/day is used as the inhalation intake rate of U-238 at the start of the residual contamination period on July 1, 1960. 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 in March 1978 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 alpha air concentration of 0.029 $pCi/m^3$. Application of a breathing rate of 9.6 $m^3$/day results in a potential alpha particle inhalation of about 0.28 pCi/day at the time of the March 1978 survey. The alpha activity is assumed to be equal amounts of U-238, U-234, and Th-230 and that they account for all alpha

particles, resulting in a U-238 intake of about 0.092 pCi/day. The 1960 and 1978 daily intake rates were used to estimate the annual intakes from the following equation.

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

where:

| | | |
|---|---|---|
| $I_t$ | = | daily intake rate at time t |
| t | = | time (days) since July 1, 1960 |
| $I_0$ | = | daily intake on July 1, 1960 |
| e | = | base of the natural logarithms |
| λ | = | exponential constant |

The derived U-238 median intake of 13 pCi/day on July 1, 1960, was substituted for $I_0$. The 0.092 pCi/day derived U-238 intake from the 1978 survey was substituted for $I_t$ on April 1, 1978. The time between July 1, 1960, and April 1, 1978 is 6483 days. This resulted in the following equation to calculate the exponential constant λ.

$$\frac{0.092\ pCi}{d} = \frac{13\ pCi}{d} * e^{-\lambda * 6483 d}$$

The constant λ was determined to be 0.000764 $day^{-1}$.

Average daily inhalation intake rates for each year from 1960 through 1996 were then calculated and are given in Table 12a below. 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 12a: Inhalation Intake Rate from Residual Contamination.[1,2,3,4]**

| | **U-238, Th-230, U-234, Pb-210, Po-210** | **Th-231,[5] Pa-231, [5] Ac-227, [5] Ra-226** | **Th-232, Ra-228, Th-228** |
|---|---|---|---|
| Year | pCi/day | pCi/day | pCi/day |
| 1960 | 12 | 0.56 | 0.40 |
| 1961 | 9.9 | 0.45 | 0.33 |
| 1962 | 7.5 | 0.34 | 0.25 |
| 1963 | 5.6 | 0.26 | 0.19 |
| 1964 | 4.3 | 0.20 | 0.14 |
| 1965 | 3.2 | 0.15 | 0.11 |
| 1966 | 2.4 | 0.11 | 0.081 |
| 1967 | 1.8 | 0.085 | 0.061 |
| 1968 | 1.4 | 0.064 | 0.046 |
| 1969 | 1.06 | 0.049 | 0.035 |
| 1970 | 0.80 | 0.037 | 0.026 |
| 1971 | 0.61 | 0.028 | 0.020 |
| 1972 | 0.46 | 0.021 | 0.015 |
| 1973 | 0.35 | 0.016 | 0.011 |
| 1974 | 0.26 | 0.012 | 0.009 |
| 1975 | 0.20 | 0.009 | 0.0066 |
| 1976 | 0.15 | 0.0069 | 0.0050 |
| 1977 | 0.11 | 0.0052 | 0.0038 |
| 1978 | 0.086 | 0.0040 | 0.0028 |
| 1979 | 0.065 | 0.0030 | 0.0021 |
| 1980 | 0.049 | 0.0023 | 0.0016 |
| 1981 | 0.037 | 0.0017 | 0.0012 |
| 1982 | 0.028 | 0.0013 | 0.0009 |
| 1983 | 0.021 | 0.0010 | 0.0007 |
| 1984 | 0.016 | 0.0007 | 0.0005 |
| 1985 | 0.012 | 0.0006 | 0.0004 |
| 1986 | 0.0092 | 0.0004 | 0.0003 |
| 1987 | 0.0070 | 0.0003 | 0.0002 |
| 1988 | 0.0053 | 0.0002 | 0.0002 |
| 1989 | 0.0040 | 0.0002 | 0.0001 |
| 1990 | 0.0030 | 0.0001 | 0.0001 |
| 1991 | 0.0023 | 0.0001 | 0.0001 |
| 1992 | 0.0017 | 0.0001 | 0.0001 |
| 1993 | 0.0013 | 0.0001 | 0.00004 |
| 1994 | 0.0010 | 0.0000 | 0.00003 |
| 1995 | 0.0008 | 0.0000 | 0.00002 |
| 1996 | 0.0006 | 0.0000 | 0.00002 |

Footnotes for Table 12a
1. Inhalation intakes are not assigned for calculating dose to the stomach, small intestine, upper large intestine, lower large intestine, or colon. See Table 12b for ingestion intakes for those tissues.
2. 1960 intakes from residual contamination start on July 1. Building 55 was demolished in 1996.
3. Intakes are based on Type M lung solubility for materials likely to have been present in Building 55 operations except for thorium, lead and polonium. Pb-210 is Type F, and Po-210 is Type F or M per ICRP 1994b. Thorium could have been Type M or Type S. Thorium and polonium solubility types must be selected based on the types that provide the largest dose to the organ or tissue of concern.
4. See Table 12b for dose to tissues of the gastrointestinal tract.
5. U-235 is allowed for in the U-238 and U-234 values. Values given are for radionuclides in the U-235 chain.

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 shown in Section 3.2.2 (administrative scenario), 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.

$$I_t = \frac{I_0 \ pCi}{d} * e^{-0.000764 d^{-1} * t}$$

where:

$I_t$ = daily ingestion intake rate at time t
$I_0$ = bounding ingestion intake on July 1, 1960
t = days since July 1, 1960

The above equation was used to derive average daily ingestion intake rates for each year between 1960 and 1996. Results are presented in Table 12b. 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 12b: Ingestion Intake Rate from Residual Contamination[1]**

| | U-238, Th-230, U-234, Pb-210, Po-210 | Th-231,[5] Pa-231,[5] Ac-227,[5] Ra-226 | Th-232, Ra-228, Th-228 |
|---|---|---|---|
| **Year** | **pCi/day** | **pCi/day** | **pCi/day** |
| 1960 | 38 | 1.8 | 1.3 |
| 1961 | 31 | 1.4 | 1.0 |
| 1962 | 24 | 1.1 | 0.78 |
| 1963 | 18 | 0.82 | 0.59 |
| 1964 | 14 | 0.62 | 0.45 |
| 1965 | 10 | 0.47 | 0.34 |
| 1966 | 7.7 | 0.35 | 0.25 |
| 1967 | 5.8 | 0.27 | 0.19 |
| 1968 | 4.4 | 0.20 | 0.15 |
| 1969 | 3.3 | 0.15 | 0.11 |
| 1970 | 2.5 | 0.12 | 0.083 |
| 1971 | 1.9 | 0.088 | 0.063 |
| 1972 | 1.5 | 0.067 | 0.048 |
| 1973 | 1.1 | 0.050 | 0.036 |
| 1974 | 0.83 | 0.038 | 0.027 |
| 1975 | 0.63 | 0.029 | 0.021 |
| 1976 | 0.48 | 0.022 | 0.016 |
| 1977 | 0.36 | 0.016 | 0.012 |
| 1978 | 0.27 | 0.012 | 0.0090 |
| 1979 | 0.21 | 0.0094 | 0.0068 |
| 1980 | 0.16 | 0.0072 | 0.0051 |
| 1981 | 0.12 | 0.0054 | 0.0039 |
| 1982 | 0.089 | 0.0041 | 0.0029 |
| 1983 | 0.067 | 0.0031 | 0.0022 |
| 1984 | 0.051 | 0.0023 | 0.0017 |
| 1985 | 0.038 | 0.0018 | 0.0013 |
| 1986 | 0.029 | 0.0013 | 0.0010 |
| 1987 | 0.022 | 0.0010 | 0.0007 |
| 1988 | 0.017 | 0.0008 | 0.0006 |
| 1989 | 0.013 | 0.0006 | 0.0004 |
| 1990 | 0.010 | 0.0004 | 0.0003 |
| 1991 | 0.0072 | 0.0003 | 0.0002 |
| 1992 | 0.0055 | 0.0003 | 0.0002 |
| 1993 | 0.0041 | 0.0002 | 0.0001 |
| 1994 | 0.0031 | 0.0001 | 0.0001 |
| 1995 | 0.0024 | 0.0001 | 0.0001 |
| 1996 | 0.0018 | 0.0001 | 0.0001 |

Footnotes for Table 12b
1. Ingestion intakes are assigned only when calculating dose to the stomach, small intestine, upper large intestine, lower large intestine, and colon. No inhalation intake is assigned for these tissues.
2. 1960 intakes from residual contamination start on July 1.
3. The f1 values are the same as those in Table 4b.

The DOE performed radon measurements in Building 55 at Blockson in 1978 and reported maximum radon gas and radon progeny values of 0.61 pCi/L and 0.0061 WL (working levels), respectively.

Olin had radon progeny measurement taken at Blockson in 1983 during plant operations; the highest value during that survey was reported to be 0.0042 WL from an active work area (STPP). Only one measurement was taken from the phosphogypsum area, and it was lower.

Bounding radon exposures during the residual contamination period at Blockson is presumed to be from the phosphogypsum piles (or stacks) containing the many tons of waste generated in producing phosphoric acid from phosphate rock. Although process related radon exposures from work with phosphates would have continued after June 1960, when the uranium recovery work ended, doses from those exposures are not considered during the residual period because they are not covered under EEOICPA.

Based on the 118.3 tons of $U_3O_8$ that had been produced for the AEC during the entire production period (August 1952 through June 1960), assuming a 50% uranium recovery and a 0.01% uranium content in the phosphate rock, about 2.4 million tons of phosphate rock was processed during the uranium recovery years; this nominal quantity is consistent with the 6,000 ton weekly throughput estimated by Blockson prior to uranium operations in 1951. Based on information in a FIPR report (1995), the amount of phosphogypsum waste generated would have been about equal to the rock processed, in which most of the Ra-226 would reside in an insoluble form. The waste was deposited on-site and stored indefinitely.

Olin stopped operations at the former Blockson plant in 1991 and the plant was permanently closed and the buildings were demolished by 1997. Olin had radon flux measurement taken to demonstrate compliance with the EPA's (40 CFR Part 61) 20 pCi/$m^2$-s radon flux limits for inactive phosphogypsum stacks. Three hundred flux measurements were taken from the phosphogypsum stacks at Blockson in September 1993 during varying weather conditions and at different locations. The weighted mean radon flux for the total stack area was reported to be 4.4 pCi/$m^2$-s, with the sides of the stack having the highest mean value of 10.1 pCi/$m^2$-s. The values were compared to flux and radon gas measurements reported from phosphogypsum stacks at phosphate plants in Texas and in Florida.

The inactive phosphogypsum stack at Texas City Chemicals was measured in February 1983 through September 1984 (inactive since 1970). The results of those tests were summarized by the United States District Court for the Southern District of Texas in records from a lawsuit (Court records, 1989). The combined average flux from those measurements was 10.5 pCi/$m^2$-s. The court reported that flux measurements from similar inactive stacks in Florida and another

plant (location not identified) were 4.5 and 4.4 pCi/$m^2$-s, respectively. The range of those values is nearly identical to the range of the mean total stack and highest mean stack area at Blockson.

In addition to radon flux measurements, radon gas concentrations at Texas City Chemicals were also measured on top of the phosphogypsum stack and near the Administration Building some 200 to 300 yards from the stack; radon concentrations above background (0.14 pCi/L), were reported to be 0.42 pCi/L and 0.32 pCi/L, respectively. Radon concentrations at other locations on the property were lower. These radon concentrations will be used to estimate bounding radon exposures at Blockson.

Court records for the Texas City Chemicals case indicate the measurements were made when the stacks had been inactive for a number of years and cited EPA's findings that radon flux from inactive phosphogypsum piles was nearly 5 times less than from active piles due to formation of crust on the pile surface. Therefore, the highest net radon concentration of 0.42 pCi/L was multiplied by 5 to get a concentration of 2.1 pCi/L that would have been present when the pile was active. This value compares reasonably well to reports by the Florida Institute for Phosphate Research (FIPR 1998). FIPR reported radon results for some outdoor areas from Florida plants that had detectable elevated radon concentrations, including phosphogypsum stacks (or piles). The results were highly variable and statistics were reported for 5 locations with elevated results. The median radon concentration for the areas ranged from 1.07 to 2.72 pCi/L. The 2.1 pCi/L value should provide a reasonable bounding estimate for exposure to radon gas from phosphogypsum at Blockson, given that workers do not typically continuously occupy waste piles.

Blockson generated phosphogypsum waste both before and after the AEC work. The 2.1 pCi/L concentration estimated for the active stack will be assumed for radon concentrations as of July 1, 1960, and all will be assumed to be attributable to AEC work. Continued operation of the phosphate plant resulted in the generation of addition radium-bearing phosphogypsum being placed on the pile, which was estimated to be 90 feet high when the plant closed. As previously noted, the formation of a crust on inactive piles was reported to reduce levels by a factor of 5. For this evaluation, the additional waste placed on the stacks is assumed to have gradually reduced the radon flux from the AEC-related waste placed on the stack in the 1950s.

Thus, for Blockson residual radon calculations, a concentration of 2.1 pCi/L will be assumed for July 1, 1960 (start of residual contamination period), and a concentration of 0.42 pCi/L will be assumed for September 1993 when the radon flux measurements were taken at the inactive stacks. It is assumed that both value the 0.42 pCi/L radon estimate for the inactive pile (when flux measurements were made in 1993) and the 2.1 pCi/L concentration as of July 1960 is all attributable to AEC work. A gradual decrease from a peak concentration in July 1960 to September 1993 (a span of 12,115 days) is calculated according the following exponential equation:

$$\frac{0.42\ pCi}{L} = \frac{2.1\ pCi}{L} * e^{-\lambda * 12{,}115 d}$$

The constant λ was determined to be 0.000133 $day^{-1}$. Average radon gas concentrations for each year from 1960 through 1993 were calculated. The exposure to the respiratory tract from radon progeny, in working levels (WL), was calculated based on an equilibrium factor of 0.4 and a conversion of 1 WL per 100 pCi/L. Annual exposures in Working Level Months (WLM) were then calculated. Results are provided in the following table.

**Table 13: Annual Radon Exposure from Residual Contamination.**

| Year | pCi/L[1] | WL[2] | WLM/yr |
|---|---|---|---|
| 1960 | 2.075 | 0.0083 | 0.050[3] |
| 1961 | 2.000 | 0.0080 | 0.096 |
| 1962 | 1.906 | 0.0076 | 0.091 |
| 1963 | 1.816 | 0.0073 | 0.087 |
| 1964 | 1.729 | 0.0069 | 0.083 |
| 1965 | 1.647 | 0.0066 | 0.079 |
| 1966 | 1.570 | 0.0063 | 0.075 |
| 1967 | 1.495 | 0.0060 | 0.072 |
| 1968 | 1.424 | 0.0057 | 0.068 |
| 1969 | 1.357 | 0.0054 | 0.065 |
| 1970 | 1.293 | 0.0052 | 0.062 |
| 1971 | 1.231 | 0.0049 | 0.059 |
| 1972 | 1.173 | 0.0047 | 0.056 |
| 1973 | 1.117 | 0.0045 | 0.054 |
| 1974 | 1.065 | 0.0043 | 0.051 |
| 1975 | 1.014 | 0.0041 | 0.049 |
| 1976 | 0.966 | 0.0039 | 0.046 |
| 1977 | 0.920 | 0.0037 | 0.044 |
| 1978 | 0.877 | 0.0035 | 0.042 |
| 1979 | 0.835 | 0.0033 | 0.040 |
| 1980 | 0.796 | 0.0032 | 0.038 |
| 1981 | 0.758 | 0.0030 | 0.036 |
| 1982 | 0.722 | 0.0029 | 0.035 |
| 1983 | 0.688 | 0.0028 | 0.033 |
| 1984 | 0.655 | 0.0026 | 0.031 |
| 1985 | 0.624 | 0.0025 | 0.030 |
| 1986 | 0.595 | 0.0024 | 0.029 |
| 1987 | 0.567 | 0.0023 | 0.027 |
| 1988 | 0.540 | 0.0022 | 0.026 |
| 1989 | 0.514 | 0.0021 | 0.025 |
| 1990 | 0.490 | 0.0020 | 0.024 |
| 1991 | 0.467 | 0.0019 | 0.022 |
| 1992 | 0.444 | 0.0018 | 0.021 |
| 1993 | 0.425 | 0.0017 | 0.020 |
| 1994-end | 0.42 | 0.0017 | 0.020 |

Footnotes for Table 13
1. Dose to non-respiratory tract tissues from inhalation of radon gas is less than 0.002 rem to all tissues in all years.
2. ET1 and ET2 doses are to be calculated from WLM values using conversion factors in NIOSH 2006b.
3. 1960 annual WLM was prorated for 6 months (July through December).

## 6.0 Dose Reconstruction Summary

As indicated in the discussion in the previous sections, Blockson processed materials that contained Naturally Occurring Radioactive Materials (NORM), with the uranium work for the AEC being a byproduct of the existing operations (Blockson 1958). Therefore, employees had the potential to receive radiation dose to some extent both before and after the uranium extraction operations in Building 55. During the 1951 through 1960 AEC operational period industrial doses workers may have received in the covered facilities are included in dose reconstructions. Therefore, bounding internal and external doses have been considered for the entire Blockson site during those years.

This document evaluated two possible scenarios to determine bounding doses received by workers during the AEC operational period: dose inside Building 55 to bound doses received by workers involved with uranium extraction work, and dose outside of Building 55 to bound doses received elsewhere at the plant. The claimant-favorable modeling of external dose in Building 55 bounds dose from the entire site.

For internal dose the bounding intakes from calcination of phosphate rock listed in Table 3 are lower than the intakes from Building 55 listed in Tables 4a and 4b. However, the intakes from the calciner may result in higher dose to some organs, e.g., lungs, due to solubility types likely to be present. Therefore, bounding internal doses for the site should be selected based on the values that result in the higher organ doses, whether it is from Table 3 or Table 4a and 4b.

For EEOICPA dose reconstruction purposes, exposure starts in March 1951, or the first date the employee has covered employment at Blockson, whichever is later. The end of the operational period is June 30, 1960. Residual contamination doses start on July 1, 1960. Table 14 contains a summary of the tables in this document to use for dose reconstructions.

**Table 14: Dose Reconstruction Summary Table**

| **Internal Dose** | **Dates** | **Dose Tables** |
|---|---|---|
| Operational Period | March 1951 through June 1960 | Table 3 or Table 4a/4b<br>Table 5 |
| Residual Period | July 1960 to end of employment | Table 12a/12b<br>Table 13 |
| **External Dose** | **Dates** | **Dose Tables** |
| Operational Period | March 1951 through June 1960 | Table 8<br>Table 9 |
| Medical X-rays | Annual 1951 through 1960 | Table 10 |
| Residual Period | July 1960 to end of employment | Table 11 |

## 7.0 Attributions and Annotations

Thomas Tomes of NIOSH/DCAS was the principal author of this document. He also performed the internal dose evaluations and evaluated dose from residual contamination.

Sam Glover, PhD, of NIOSH/DCAS performed the MCNP external dose modeling based on assumptions stated in the document.

Some site information and external dose evaluation was retained from the original version of the site profile developed by the ORAU team under contract to NIOSH. Jeri Anderson, PhD, was the subject expert of the original version developed by ORAU. This revision incorporated additional information from former Blockson workers that was not previously available. Newly obtained data from DOE archives and from the Olin Corporation was also included in this revision.

Chemical modeling was added to account for potential doses from the various progeny present and to assess its potential impact on worker dose. Supporting information and evaluation was provided by George Vargo, PhD, of the ORAU team, who contracted Alan Elzerman, PhD, of Clemson University to evaluate certain Blockson chemical processes. The conclusions and recommendations in this document are those of the principal author. The conclusions have been reviewed, and comments by peers and DCAS management have been incorporated.

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