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 8 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

Table 10, cont.

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 April 1, 1962. Thereafter, airborne radioactivity from resuspension of contamination in the facility and corresponding intakes are assumed to decrease according to an exponential model described below.

Extensive radiological surveys were performed starting in March 1978. Airborne radioactivity concentration at 16 years post operation was estimated by assuming that the facility was uniformly contaminated at the level of the maximum alpha smear result of 640 dpm/100 cm^2 as reported in the 1978 survey (DOE 1983). This value was multiplied by a resuspension factor of 1E-06 m^{-1} (NRC 2002c), which results in an estimated maximum residual 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 assumption of are equal amounts of U-238, U-234, and Th-230 and that they account for all alpha particles, results in a U-238 intake of about 0.092 pCi/day. The 1962 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 April 1, 1962
I_0	=	daily intake on April 1, 1962
e	=	base of the natural logarithms
λ	=	exponential constant

The derived U-238 median intake of 13 pCi/day on April 1, 1962, 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, 1962. The time between April 1, 1962, and April 1, 1978 is 5844 days. This resulted in the following equation to calculate the exponential constant λ.

$$\frac{0.092\ pCi}{d} = \frac{13\ pCi}{d} * e^{-\lambda * 5844\, d}$$

The constant λ was determined to be 0.000847 day^{-1}.

Average daily inhalation intake rates for each year from 1962 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
1962	12	0.54	0.38
1963	8.9	0.42	0.29
1964	6.5	0.31	0.22
1965	4.8	0.22	0.16
1966	3.5	0.16	0.12
1967	2.6	0.12	0.085
1968	1.9	0.089	0.062
1969	1.4	0.065	0.046
1970	1.0	0.048	0.034
1971	0.75	0.035	0.025
1972	0.55	0.026	0.018
1973	0.40	0.019	0.013
1974	0.29	0.014	0.010
1975	0.22	0.010	0.007
1976	0.16	0.0075	0.005
1977	0.12	0.0055	0.0038
1978	0.085	0.0040	0.0028
1979	0.063	0.0029	0.0021
1980	0.046	0.0022	0.0015
1981	0.034	0.0016	0.0011
1982	0.025	0.0012	0.0008
1983	0.018	0.0009	0.0006
1984	0.013	0.0006	0.0004
1985	0.0098	0.0005	0.0003
1986	0.0072	0.0003	0.0002
1987	0.0053	0.0002	0.0002
1988	0.0039	0.0002	0.0001
1989	0.0028	0.0001	0.0001
1990	0.0021	0.0001	0.0001
1991	0.0015	0.0001	0.0001
1992	0.0011	0.0001	0.00004
1993	0.0008	0.00004	0.00003
1994	0.0006	0.00003	0.00002
1995	0.0004	0.00002	0.00001
1996	0.0003	0.00002	0.00001

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. 1962 intakes from residual contamination start on April 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 11b 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.

$$I_t = \frac{I_0 \ pCi}{d} * e^{-0.000847\, d^{-1} * t}$$

where:

I_t = daily ingestion intake rate at time t
I_0 = bounding ingestion intake on April 1, 1962
t = days since April 1, 1962

The above equation was used to derive average daily ingestion intake rates for each year between 1962 and 1978. Results are presented in Table 11b. 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**
1962	37	1.8	1.2
1963	29	1.3	0.94
1964	21	1.0	0.69
1965	15	0.72	0.51
1966	11	0.53	0.37
1967	8.3	0.39	0.27
1968	6.1	0.29	0.20
1969	4.5	0.21	0.15
1970	3.3	0.15	0.11
1971	2.4	0.11	0.08
1972	1.8	0.08	0.06
1973	1.3	0.06	0.043
1974	1.0	0.045	0.031
1975	0.70	0.033	0.023
1976	0.51	0.024	0.017
1977	0.38	0.018	0.012
1978	0.28	0.013	0.0091
1979	0.20	0.010	0.0067
1980	0.15	0.0070	0.0049
1981	0.11	0.0051	0.0036
1982	0.080	0.0038	0.0026
1983	0.059	0.0028	0.0019
1984	0.043	0.0020	0.0014
1985	0.032	0.0015	0.0010
1986	0.023	0.0011	0.0008
1987	0.017	0.0008	0.0006
1988	0.013	0.0006	0.0004
1989	0.0092	0.0004	0.0003
1990	0.0067	0.0003	0.0002
1991	0.0050	0.0002	0.0002
1992	0.0036	0.0002	0.0001
1993	0.0027	0.0001	0.0001
1994	0.0020	0.0001	0.0001
1995	0.0014	0.0001	0.0000
1996	0.0011	0.0000	0.0000

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. 1962 intakes from residual contamination start on April 1.
3. The f1 values are the same as those in Table 4b.

Radon exposure from residual contamination is assumed to have occurred; these values drop over time to be lower than that received on site from natural radioactivity or other operations on site. A gradual decline in radon in Building 55 from AEC activities is also assumed to have occurred coinciding with the end of operations in 1962. The decline is assumed to be at the same rate as the particulate contamination discussed above, starting with the bounding annualized operational exposure of 0.112 WLM specified in Table 5. The annual radon exposures are listed in Table 12.

Table 13: Annual Radon Exposure from Residual Contamination.

	Lungs	**Non-respiratory tract tissues**
Year	**WLM/yr[1]**	**rem/yr, alpha**
1962	9.95E-02	0.002
1963	7.63E-02	0.002
1964	5.62E-02	0.001
1965	4.11E-02	0.001
1966	3.02E-02	0.001
1967	2.21E-02	0.001
1968	1.63E-02	0.0004
1969	1.19E-02	0.0003
1970	8.75E-03	0.0002
1971	6.42E-03	0.0002
1972	4.73E-03	0.0001
1973	3.46E-03	0.0001
1974	2.54E-03	0.0001
1975	1.86E-03	<0.0001
1976	1.37E-03	<0.0001
1977	1.00E-03	<0.0001
1978	7.37E-04	<0.0001
1979	5.41E-04	<0.0001
1980	3.98E-04	<0.0001
1981	2.91E-04	<0.0001
1982	2.14E-04	<0.0001
1983	1.57E-04	<0.0001
1984	1.15E-04	<0.0001
1985	8.45E-05	<0.0001
1986	6.20E-05	<0.0001
1987	4.55E-05	<0.0001
1988	3.35E-05	<0.0001
1989	2.45E-05	<0.0001
1990	1.80E-05	<0.0001
1991	1.32E-05	<0.0001
1992	9.71E-06	<0.0001

Table 13, cont.

1993	7.11E-06	<0.0001
1994	5.22E-06	<0.0001
1995	3.83E-06	<0.0001
1996	2.82E-06	<0.0001

1. ET1 and ET2 doses are to be calculated from WLM values using conversion factors in NIOSH 2006b.

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 1962 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 March 31, 1962. Residual contamination doses start on April 1, 1962. Building 55 was demolished in 1996 and residual doses from that building should be assigned through that year.

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 March 1962	Table 3 or Table 4a/4b Table 5
Residual Period	April 1962 through December 1996	Table 12a/12b Table 13
External Dose	**Dates**	**Dose Tables**
Operational Period	March 1951 through March 1962	Table 8 Table 9
Residual Period	April 1962 through December 1996	Table 11
Medical X-rays	Annual 1951 through 1962	Table 10

7.0 Attributions and Annotations

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

Sam Glover, PhD, of NIOSH/OCAS 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 OCAS management have been incorporated.

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