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Site Profiles for Atomic Weapons Employers that Refined Uranium and Thorium - Appendix D United Nuclear Corp.	
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3/14/2008	3/14/2008	0	Appendix to Battelle-TBD-6001 describing the use of the TBD for claims at United Nuclear Corp.
4/30/2010	4/30/2010	1	Update to reflect additional site specific bioassay and external dose data and change in facility covered dates.

D.1 Introduction

This document serves as an appendix to Battelle-TBD-6001, Site Profiles for Atomic Weapons Employers that Refined Uranium and Thorium. This appendix describes the results of document research specific to this site. Where specific information is lacking, research into similar facilities described in the body of this Site Profile is used.

D.2 Site Description

Throughout its history, United Nuclear Corporation (UNC) located in Hematite, Missouri, manufactured uranium metal and uranium compounds from natural and enriched uranium for use as nuclear fuel. The fuel was manufactured for use by the federal government and government contractors and by commercial and research reactors. Research and development was also conducted at the plant as well as uranium scrap recovery¹.

Mallinckrodt Chemical Works built the plant which became operational in July 1956. The plant initially produced uranium products for use in the naval nuclear fuel program. Ownership transferred to United Nuclear Corporation in May of 1961. In 1970 the operator became Gulf United Nuclear Fuels Corporation, which was a joint venture between UNC and Gulf Nuclear Corporation. The facility was closed in 1973 and sold to Combustion Engineering in May of 1974. In 1989 Asea Brown Boveri (ABB) began

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operating the facility as ABB Combustion Engineering. In April of 2000, Westinghouse purchased the nuclear operations of ABB which included the Hematite facility¹.

Most of the operations at the facility involved manufacturing reactor fuel for the naval nuclear program or commercial reactors rather than weapons work. However, the Department of Energy has determined that the facility performed uranium scrap recovery operations for the weapons program between 1958 and 1973². Therefore, all sources of radiation exposure during this time period must be accounted for, including that from commercial and naval reactor work. After 1973 only radiation dose caused by contamination remaining from weapons related work must be accounted for (residual contamination).³

D.3 Occupational Medical Dose

No documentation regarding occupational medical dose specific to UNC was found. Information to be used in dose reconstructions for which no specific information is available is provided in ORAUT-OTIB-0006, the dose reconstruction project technical information bulletin covering diagnostic X-ray procedures.

D.4 Occupational External Dose

Personnel photon exposures at UNC were directly related to work with enriched uranium, primarily from operations in the Red Room, Blue Room, Green Room, and Item Plant. These processes are known to have been a routine component of plant operations during the entire covered period from 1958-1973 based on a review of plant radiation safety manuals and procedures, plant correspondence and operations logs, and film badge results. Uranium in various forms and compounds was used. Accordingly, external exposures to photon radiation would have resulted from radionuclides in the uranium decay chain. Exposure to beta sources at UNC would have resulted principally from uranium decay products and was likely encountered during routine scrap recovery operations and other production activities at the site. Potential exposure to neutrons would have resulted from operations with enriched uranium.

D.4.1 Co-worker gamma and beta

Dosimetry data for monitored workers at United Nuclear Corporation were evaluated. According to the 1962 UNC Health Physics Manual, all personnel assigned to the plant were issued a film badge upon entry to the facility. The available personnel monitoring data include data that represent the maximally-exposed work group and work scenario over the operational period. Film badge data was available for the operational years (1958-1973). Data for 1958 through 1960 exists only in reports summarizing exposures. Dose estimates based on coworker exposures were developed for use in the evaluation of external dose for certain claimants potentially exposed to workplace radiation, but with no or limited monitoring data. The Limit of Detection (LOD) is 40 mrem for beta and 40 mrem for gamma in accordance with NIOSH technical information bulletins and procedures.^{4,5,6}

The external dosimetry data for 1958 indicates there was a combination of weekly and monthly exchanges.⁷ The average reported dose received during 1958 was 80 mrad beta and 36 mrad gamma. The maximum reported dose received was 2525 mrad beta and 380 mrad gamma (for 6 months).⁷ A yearly missed dose based on 20 mr (LOD/2) per week

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was added to the average values to derive a geometric mean (GM) for both the beta dose and the gamma dose. The yearly missed dose was also added to the maximum values (adjusted to an annual dose) to produce values at the 95th percentile.

During 1959 all weekly exchanges were changed to monthly. The data for 1959 indicates the maximum beta dose received was 240 mrem in a month. The average beta was 90 mrem in a month. The maximum gamma was 15 mrem in a month.⁷ Since it is not clear when in 1959 the exchange frequency changed, missed dose was assigned based on a weekly exchange frequency for the entire year. Missed dose was added to the maximum recorded values and used to represent the 95th percentile dose. Missed dose was added to the average beta dose and used to represent the GM beta dose. The GM gamma dose was assigned based on the GM beta dose adjusted using the average beta to gamma ratio between 1961 and 1965.

The data for 1960 is limited. One report indicated the maximum gamma dose received prior to 1961 was 100 mrem in a week⁷ while the typical reading was below 50 mrem. The maximum value was assumed to occur every week and assigned as the 95th percentile. The typical reading was also assumed to occur every week and assigned as the GM. A Beta to gamma ratio based on 1961 to 1965 data was used to derive the annual beta doses.

The data for 1961 through 1965 was reported as separate beta and gamma readings. This data was used to assign gamma and beta dose for each year at the GM and the 95th percentile.

For 1966 through 1973 only the total dose (combining beta and gamma) was reported. Using the 1961 through 1965 data, a ratio of beta to gamma was calculated for each of those years, and the average of that ratio was applied to the 1966 through 1973 data to derive annual beta and gamma dose at the GM and the 95th percentile.

Unmonitored workers that were routinely involved with hands on uranium work will be assigned the 95th percentile doses. Unmonitored workers routinely in the area but not routinely handling uranium directly (supervisors, laborers, etc.) will be assigned the dose distribution assuming a lognormal distribution. Those not routinely in the areas where uranium was handled will be assigned the GM of the distribution as a constant. The annual external dose values to be applied each year for unmonitored workers are listed in Tables D2 and D3.

D.4.2 Co-worker neutron

A routine external monitoring program for neutron exposure did not exist at UNC, but there was a potential for personnel neutron exposures from the uranium-enrichment operations.⁷ The site received and processed uranium compounds containing fluorine and oxygen. The site was authorized by the AEC to receive enriched uranium hexafluoride and typically requested quantities of 50 to 100 kilograms (kg) on a quarterly basis with enrichments ranging from approximately 20% to 93%.

The maximum amount of 20% enriched uranium handled at any one time is assumed to be 100 kg.⁷ The maximum amount of 93% enriched uranium being handled at any one time is assumed to be 50 kg due to criticality concerns. Conversion factors from ORAUT-OTIB-0024 for naturally enriched uranium without progeny were used to estimate the neutron dose rate. The conversion factors were adjusted based on the

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specific activity of naturally enriched uranium compared to the specific activity of 20% and 93% enriched uranium.

Operators are assumed to be working within 1 foot of the enriched uranium fluoride $\frac{1}{2}$ the time (1000 hours per year). Supervisors are assumed to be working within 3 feet of the enriched uranium fluoride $\frac{1}{2}$ the time (1000 hours per year). Both are assumed to be working with 20% enriched uranium and 93% enriched uranium for equal amounts of time. Others will be assumed to receive an annual dose equal to half the supervisors' dose. Annual dose values are listed in Table D.4. These values represent a bounding estimate and should be considered a constant.

D.4.3 Measured Dose

If external dosimetry data is available for the employee, the data should be used to reconstruct external dose received. The external dosimetry data was reported as a mix of monthly and yearly exposures. For dose reconstruction purposes, unless otherwise indicated, exchange frequencies are considered to be weekly for 1958 and 1959 and monthly for all other years.

Photons

Dose Reconstructors should include an uncertainty factor of 1.3 multiplied by the measured photon dose, assume a constant distribution, and utilize the Exposure (R) to Organ Dose (HT), dose conversion factors from IG-001.⁸ The photon radiation should be considered 50% in the 30-250 keV energy range and 50% in the greater than 250 keV energy range.

Electrons

From 1961 through 1965 beta and photon exposures were measured separately. Beginning in 1966 the photon dose and the beta dose were reported as a sum of the beta and photon dose. For combined doses, 25% should be considered photon dose and 75% considered beta dose. This is based on the ratio of photon to beta dose from the 1961-1965 data. The guidance contained in ORAUT-OTIB-0017⁵ for assignment of skin dose from penetrating and nonpenetrating radiation should be followed. The nonpenetrating radiation is predominantly in the $E > 15$ keV energy range.

D.4.4 Missed dose

Missed dose applies to workers who were monitored but had results below the Limit of Detection (LOD) of their personal radiation monitor. An acceptable, favorable to claimant estimate of missed dose is one-half the LOD multiplied by the number of zero dose results for dose results below one-half the LOD. A GSD of 1.52 should be applied to missed dose. For summary data, recorded dose should be assumed to have occurred during only one monitoring period and missed dose assigned for the remaining periods.

D.5 Occupational Internal Dose

Bioassay and air monitoring data from UNC identified the potential for internal radiation exposures to workers. The primary source of this exposure was deposition of alpha-emitting materials via inhalation and ingestion of airborne uranium and thorium (and progeny).

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D.5.1 Uranium

There is urinalysis data for workers at UNC available for most years from 1958 through 1973. This data, however, generally does not separate people into job categories. The available samples were analyzed on an annual basis to determine the parameters of a lognormal distribution. The geometric mean for each year was compared to the urine concentrations predicted from the original intake calculations in revision 0 of this appendix. The predicted urine concentrations for operators overestimated most annual results for both type M and type S intakes. The predicted concentrations for supervisors were at the low end of the annual results for both solubility types. This is consistent with data that has known job categories. Results from the operators are consistently higher than those from the supervisors and other job categories. Since the original intake estimate was able to separate operators from others and appears to be consistent with additional data, the original intake estimate, as described below, will be used.

If bioassay data is available for the employee, the data should be used to estimate intakes.

Coworker Internal Dose from Uranium

Urinalysis samples starting in late 1962 revealed some operators had higher intakes than previously believed based on air sampling⁷. Several improvements were made and the samples were reduced by mid 1963. Therefore, the urinalysis results were broken into four categories. The categories were operators prior to June 1963 and after June 1963 as well as "others" prior to June 1963 and after. All the samples in each group were used to determine parameters of a lognormal distribution of urine samples. From these distributions, intake rates were determined assuming first type M solubility material and then type S material for each of the four categories. In determining the intakes, a constant chronic intake was assumed from 1/1/1958 (the first day of the covered work) to 6/13/1963 (the day before the sample date showing a significant decrease in results). The second period assumed a constant chronic intake from 6/14/1963 to 12/31/1973 (the last day of the contract period).

The resulting intakes are the geometric mean of a lognormal distribution. The intake rates along with the geometric standard deviations (GSD) are listed in Table D.1. When the calculated GSD was less than three, the value of 3.0 was substituted in order to account for the uncertainty in the biokinetic models used in dose estimation. The geometric mean of the supervisor intake will be used for people categorized as other. Since this category is reserved for people that do not routinely enter the production areas, this value represents a bounding estimate and should be considered to be a constant.

These assumed intakes were compared with a sampling of the individuals for which both air sample data and urinalysis data exist⁴. The geometric mean of the assumed intakes underpredicted urinalysis results for the people with higher sample results but overpredicted those with lower results. This indicates good agreement with the distribution. However, the calculated intakes in Table D.1 for type S material were considerably higher than the air sample results for these people. The intakes calculated for type M material did, however, match the air sample values reasonably well. The reports containing these values imply the most significant difficulty they had in controlling intakes involved the metal reduction area. This area would contain both type M and type S material and it is likely the highest urinalysis results came from type M exposures. However, it is not possible to determine if the urine sample results from 1962-1964 involved only type M material or less soluble type S material. Therefore,

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since both were available at the site, intakes rates for both were determined and the dose reconstructor should apply the more favorable of the two.

Internal Dose from Uranium Using Bioassay Results

Beginning in 1958, UNC operating personnel submitted urinalysis samples on a three- to six-month basis dependent on specific work assignments. Sample analysis incorporated gross activity counting methods to determine uranium activity levels. UNC evaluated bioassays by different analytical protocols, including enriched and fluorometric uranium techniques. Total uranium concentrations were measured and reported with a minimum detectable concentration (MDC) of 10 dpm per liter.

D.5.2 Thorium

Approximately nine tons of natural thorium was on site during the covered period for a single, specific project conducted in 1964 in the Pellet Plant.⁷ Thorium dioxide powder was blended with uranium dioxide powder to produce Th-U fuel pellets to be used in fuel assemblies for breeder reactors.⁷ Each pellet consisted of a mixture of 97 % ThO₂ and 3% UO₂, compressed and sintered (heated) in a furnace. A total of 210 air samples, designated specifically as “ThO₂” (thorium dioxide), were collected throughout the Pellet Plant project in 1964 and the results were reported in uCi/ml. These samples were used to determine parameters of a lognormal distribution. From this distribution, intake rates were determined. Operators who routinely handled thorium or operated thorium processing equipment will be assigned the 95th percentile of this distribution. Supervisors, laborers and others who routinely entered the processing area but did not routinely handle thorium will be assigned the full distribution of intakes. Others who did not routinely enter the area, will be assigned the geometric mean of the distribution.

The Technical Information Bulletin, “Estimation of Ingestion Intakes”¹¹ was also utilized to determine additional internal intake due to ingestion with the f_1 value the same as that used for inhalation. The intake rates along with the geometric standard deviations (GSD) are listed in Table D.1. For internal dose estimates, the thorium will be considered to be type M or S solubility, whichever produces the highest dose. No thorium intakes will be assigned after 1964 since indications are that a cleanup of residual thorium dioxide was conducted following the project.

D.5.3 In vivo

Direct (*in vivo*) whole-body counting (WBC) was conducted during the years 1963, 1964, and 1965 for several workers, using total body counters available at the Y-12 plant in Tennessee.⁷ Emphasis was placed primarily on several workers who had received elevated internal exposures in the Red Room (exceeding 10 C.F.R. pt. 20 limits). The analysis was performed to assay the amount (micrograms) of U-235 in the workers’ lungs. Other employees were also assayed for uranium at Y-12 during these years but not on a consistent basis. *In vivo* WBC was also conducted during the years 1968-1973. UNC contracted Helgeson Nuclear Services, Inc., to conduct WBC for evidence of U-235 intakes.⁷

D.6 Residual Contamination

Residual contamination from weapons related work may have been present at United Nuclear after weapons work ended in 1973. Only exposure from weapons related work is addressed after 1973 even though additional exposure from non-weapons related work may have been received.³

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In order to estimate this residual contamination, the highest intake rate from table D.1 was converted to an air concentration and assumed to settle to the floor and accumulate for an entire year. The surface contamination resulting from this was then assumed to expose an individual for 2000 hours per year. Both external exposure and intakes from resuspension were calculated based on this value.

A letter to the AEC⁴ describes some of the measures taken in 1960 to reduce airborne concentrations. These measures included wet mopping at frequent intervals not only the floors but overhead, piping and duct work. It also describes various type of ventilation used to remove airborne contaminants. Therefore, the assumption that all the airborne contamination settled out and was never removed is considered a bounding estimate and will be used as a constant in dose estimates.

D.7 References

1. RefID 30008 (pp. 16-17) Westinghouse, 2003, *Hematite Former Fuel Cycle Facility Decommissioning, Historical Site Assessment*, May 20, 2003
2. DOE Office of Health, Safety and Security, EEOICPA web site.
<http://www.hss.energy.gov/healthsafety/fwsp/advocacy/faclist/findfacility.cfm>
3. NIOSH, 2007, *Radiation exposures covered for Dose Reconstructions under Part B of the Energy Employees Occupational Illness Compensation Program Act, Rev. 0*, OCAS-IG-003, National Institute for Occupational Safety and Health, Office of Compensation Analysis and Support, Cincinnati, Ohio.
4. ORAUT (Oak Ridge Associated Universities Team), ORAUT- OTIB-0010, *Technical Information Bulletin: A Standard Complex-Wide Methodology for Overestimating External Doses Measured with Film Badge Dosimeters, Rev 01*, June 5, 2006.
5. ORAUT (Oak Ridge Associated Universities Team), ORAUT- OTIB-0017, *Technical Information Bulletin: Interpretation of Dosimetry Data for Assignment of Shallow Dose, Rev 01*, October 11, 2005.
6. ORAUT (Oak Ridge Associated Universities Team), ORAUT-PROC-0006, *External Dose Reconstruction, Rev 01*, June 5, 2006.
7. NIOSH, *SEC Petition Evaluation Report for the United Nuclear Corporation, Petition SEC-00116*, Rev 1, January 28, 2010.
8. NIOSH (2007) *External Dose Reconstruction Implementation Guideline, Rev 3*, OCAS-IG-001, National Institute for Occupational Safety and Health, Office of Compensation Analysis and Support, Cincinnati, Ohio.
9. RefID 11724 (pp. 22-25) J. A. Lindberg, United Nuclear Corporation, 1963, *Letter to Charles Keller, U. S. Atomic Energy Commission*, October 28, 1963.
10. RefID 17606 Frank Zeitlin, United Nuclear Corporation, 1960, *Letter to H. L. Price, U. S. Atomic Energy Commission*, December 16, 1960

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11. OCAS (Office of Compensation Analysis and Support), OCAS-TIB-009, *Estimation of Ingestion Intakes, Rev 0*, April 13, 2004.

Table D.1 INTERNAL DOSE PATHWAYS - Inhalation of Airborne Radionuclides**Assumptions:**

Intakes are in dpm per calendar day.

Dose entered into IREP as alpha radiation

Intake value is the geometric mean of a lognormal distribution except where GSD is listed as constant. Those intake values are constants (no distribution)

Job Category	Year	Operation Phase	Relevant Nuclide	Solubility Type	Intake (dpm/d)	GSD	TBD Reference or Research Justification
Operators	1/1/1958 - 6/13/1963	Operations	U234	S	12590	3.29	Bioassay results
Supervisors	1/1/1958 - 6/13/1963	Operations	U234	S	4784	3.0	Bioassay results
Other	1/1/1958 - 6/13/1963	Operations	U234	S	4784	Constant	Bioassay results
Operators	6/14/1963 - 12/31/1973	Operations	U234	S	7662	3.0	Bioassay results
Supervisors	6/14/1963 - 12/31/1973	Operations	U234	S	2311	3.0	Bioassay results
Other	6/14/1963 - 12/31/1973	Operations	U234	S	2311	Constant	Bioassay results
Operators	1/1/1958 - 6/13/1963	Operations	U234	M	871.9	3.29	Bioassay results
Supervisors	1/1/1958 - 6/13/1963	Operations	U234	M	331.2	3.0	Bioassay results
Other	1/1/1958 - 6/13/1963	Operations	U234	M	331.2	Constant	Bioassay results
Operators	6/14/1963 - 12/31/1973	Operations	U234	M	560.2	3.0	Bioassay results
Supervisors	6/14/1963 - 12/31/1973	Operations	U234	M	169	3.0	Bioassay results
Other	6/14/1963 - 12/31/1973	Operations	U234	M	169	Constant	Bioassay results
All	1/1/1974 – 07/31/2006	Residual	U234	S	10.34	Constant	Bioassay results
All	1/1/1974 – 07/31/2006	Residual	U234	M	0.72	Constant	Bioassay results

Job Category	Year	Operation Phase	Relevant Nuclide	Solubility Type	Intake (dpm/d)	Ingestion (dpm/d)	GSD	TBD Reference or Research Justification
Operators	1964	Operations	Th-232	M or S	4130	126	Constant	Air sample data
Supervisors	1964	Operations	Th-232	M or S	165	5.02	7.09	Air sample data
Other	1964	Operations	Th-232	M or S	165	5.02	Constant	Air sample data

Table D.2 EXTERNAL DOSE PATHWAYS - Whole Body

Based on monthly film badge results and 12 months per year of work.

Values are the geometric mean of a lognormal distribution except where GSD is listed as constant. Those intake values are constants (no distribution)

Operation Phase	Year	Others (rem/year)	Operators (rem/year)	GSD	Supervisors (rem/year)	GSD
Operations	1958	1.016	1.74	Constant	1.016	1.390
	1959	0.680	1.16		0.680	1.384
	1960	0.620	1.240		0.620	1.524
	1961	0.346	0.550		0.346	1.325
	1962	0.511	1.349		0.511	1.805
	1963	0.467	1.070		0.467	1.655
	1964	0.538	1.255		0.538	1.674
	1965	0.769	3.494		0.769	2.509
	1966	0.181	0.382		0.181	1.576
	1967	0.160	0.271		0.160	1.376
	1968	0.177	0.386		0.177	1.605
	1969	0.181	0.371		0.181	1.546
	1970	0.179	0.514		0.179	1.897
	1971	0.134	0.176		0.134	1.179
	1972	0.151	0.390		0.151	1.779
1973	0.172	0.385	0.172	1.633		

Operation Phase	Year	Whole body (mr/yr)	Job Category	GSD
Residual	1/1/1974 – 07/31/2006	11.6	All	Constant

Table D.3 EXTERNAL DOSE PATHWAYS - Skin

Assumptions:

Based on monthly film badge results and 12 months per year of work

Doses are entered into IREP as Electron >15 keV

These doses are in addition to the Whole Body doses in Table D.2

Values are the geometric mean of a lognormal distribution except where GSD is listed as constant. Those intake values are constants (no distribution)

Operation Phase	Year	Others (rem/year)	Operators (rem/year)	GSD	Supervisors (rem/year)	GSD
Operations	1958	1.060	6.030	Constant	1.060	2.877
	1959	2.060	3.860		2.060	1.465
	1960	1.880	3.760		1.880	1.524
	1961	0.840	3.169		0.840	2.242
	1962	2.166	10.902		2.166	2.671
	1963	3.276	9.670		3.276	1.931
	1964	2.566	7.413		2.566	1.906
	1965	0.904	3.102		0.904	2.115
	1966	0.548	1.158		0.548	1.576
	1967	0.485	0.820		0.485	1.376
	1968	0.537	1.170		0.537	1.605
	1969	0.549	1.123		0.549	1.546
	1970	0.543	1.556		0.543	1.897
	1971	0.407	0.534		0.407	1.179
	1972	0.458	1.181		0.458	1.779
1973	0.520	1.166	0.520	1.633		

Operation Phase	Year	Skin (mr/yr)	Job Category	GSD
Residual	1/1/1974 – 07/31/2006	186	All	Constant

Table D.4 EXTERNAL DOSE PATHWAYS - Neutron

Assumptions:

Doses are entered into IREP as Neutrons 100 keV – 2 MeV

Values are constant (no distribution)

Operation Phase	Year	Others (rem/year)	Operators (rem/year)	Supervisors (rem/year)	GSD
Operations	1958 - 1973	0.114	2.06	0.228	Constant