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Site Profiles for Atomic Weapons Employers that Worked Uranium Metals

Appendix CD – Seymour Specialty Wiring Company

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7/16/2007	7/16/2007	0	Appendix to Battelle-TBD-6000 describing the use of the TBD for claims at Seymour Specialty Wiring
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SEYMOUR SPECIALTY WIRE COMPANY

CD.1 Introduction

This document serves as an appendix to Battelle-TBD-6000, Site Profiles for Atomic Weapons Employers that Worked Uranium Metals (Battelle 2011). This Site Profile presents site-specific information for the Seymour Specialty Wire Company located in Seymour, Connecticut. Sufficient information has been found to provide more appropriate estimates of worker radiation dose than provided for in the technical basis document (TBD). Where specific information is lacking, research into similar facilities described in the body of this Site Profile is used.

CD.2 Site Description

The Bridgeport Brass Company performed Atomic Energy Commission (AEC) contract work at its facilities in Seymour Connecticut from 1962 through 1964. This work was moved to the Seymour site from the Havens Laboratory in Bridgeport Connecticut. The work was developmental work which included extrusion and machining and metallurgical laboratory analysis of uranium rods. In late 1964 all Bridgeport Brass work was consolidated at the Bridgeport Brass Reactive Metals site in Ashtabula, Ohio known at Reactive Metals Inc. or RMI. The Seymour site was later taken over by employees and eventually became the Seymour Specialty Wire Company. In 1992 operations ceased and DOE performed remediation at the site in 1992 and 1993.

CD.2.1 Site Activities

On March 13, 1962 the Bridgeport Brass Company wrote a letter to the AEC informing them that the building that housed the development group (Havens Laboratory) was being sold and the operations were going to be relocated to Seymour Connecticut (Bridgeport Brass 1962). The AEC approved the move in a May 15, 1962 letter (AEC 1962). The exact date when operations at Seymour began is unclear but a summary of radiation exposures at Bridgeport Brass indicates the dose summary for Havens Laboratory is actually Seymour Connecticut after 6/12/1962 (Bridgeport Brass 1963). To account for the possibility that there may have been some operations at both locations during a short transition period, this dose estimate will assume operations started at Seymour on 5/15/1962, the day the AEC approved the move.

In 1964 the Seymour operation was moved to Ashtabula Ohio. A summary of radiation exposures indicates that the Seymour facility was closed after 9/30/1964 (RMI 1962-1964, pg. 212). A final survey of the facility was conducted by the Health and Safety Lab of the AEC on 10/21/1964 (Breslin 1964). This dose estimate will assume operations ended on 10/21/1964.

CD.3 Occupational Medical Dose

No information regarding occupational medical dose specific to the Seymour site was found. Information to be used in dose reconstructions for which no specific information is

available is provided in ORAUT-OTIB-0006 (ORAU, 2011), the dose reconstruction project technical information bulletin covering diagnostic x-ray procedures.

CD.4 Occupational Internal Dose

Periodic urine bioassay samples were collected from individuals working at the Havens laboratory and the practice continued when operations were moved to the Seymour site. The samples were analyzed for uranium and reported in units of mg/L. Samples were collected from some employees on 14 different days. A total of 25 different employees were sampled during the operational period at Seymour (AEC 1962-1964). Table 1 below provides the sample results in units of micrograms per liter.

Table 1 – Urinalysis Uranium Results from Bridgeport Brass, Seymour (ug/L)

Employee	10/8/1962	11/19/1962	12/17/1962	2/18/1963	5/23/1963	6/10/1963	9/30/1963
A		2	35	4			
B	7	5	11		53		
C	3	2	4	8			
D					1	4	2
E	4	2	5	2	4		3
F	2	4	19	6	5	1	2
G		5	3	4			
H	3	3	5	5		0	1
I	6	5	7	8	2	6	
J	3	2	3				2
K							
L	6	13	30	7	7	11	14
M						0	
N				3	0	2	
O		4	13				
P					1		
Q							1
R	5	3	4	2			
S					0		
T	3	4	4	4			
U							
V							1
W					4	0	
X							
Y	1	2220	12	5		1	

Table 1 - Continued

Employee	12/9/1963	12/13/1963	1/7/1964	1/10/1964	1/13/1964	2/4/1964	2/11/1964
A							
B							
C							
D	1	0	10	14	4	23	14
E	4	20	18	17	80	2	17
F		7	20	14	4	2	14
G							
H	0	10	21	6	2	14	6
I		16				7	
J	5		7			2	
K	34	6			18	20	
L	2	4	36	22	10		22
M							
N	0	2	32	42	2	4	
O							
P							
Q	6	8	41	8	2	2	8
R							
S							
T							
U			23	13	8	20	13
V	3	8			2		
W							
X	4		41	20	2	1	20
Y							

To estimate the uranium intake at Seymour, an intake of uranium was calculated for 21 of the 25 individuals sampled. The 4 individuals removed from the data set had either none or only one sample greater than zero. Also, one sample was discarded as erroneous. The 11/19/1962 sample for employee Y indicated 2200 ug/L while samples on 13 other people that day indicated between 2 and 13 ug/L. Also, no other sample on any day exceeded 80 ug/L. Lastly, this individual was sampled again 28 days after this sample and the new result indicated 12 ug/L. The biokinetics of uranium would not allow it to clear out of the body that fast. It is much more likely the sample was contaminated either during collections or later in the laboratory or that some other error occurred in the laboratory. Therefore, this result is considered erroneous and discarded.

Two intakes were calculated for the 21 individuals assuming they were exposed to uranium with solubility type M and S. For each solubility, the distribution of intakes was analyzed and found to fit a lognormal distribution reasonably well. The geometric mean (GM) and the geometric standard deviation (GSD) of the lognormal distributions are recorded in Table 2.

Table 2 – Operational Period Uranium Inhalation Rate

5/15/1962 – 10/21/1964	Geometric Mean (pCi/day)	GSD
Type S uranium	3277	1.77
Type M uranium	124	1.79

Internal dose estimates should be calculated using both solubility types and one that produces the higher dose used. The doses should be entered into IREP as a lognormal distribution with the applicable GSD from Table 2.

CD.5 Occupational External Dose

No individual film badge data was located for the Seymour site. However, annual summaries indicated no one exceeded 1 rem of gamma exposure from 1962 through 1964 (RMI 1962-1964).

Since operations at Seymour were moved from the Havens Lab and since the reason for the move was not related to any change in operation (the building was being sold), the external exposures at Havens lab should provide a reasonable estimate for the external exposures at Seymour. The Technical Basis Document for the Havens Lab (ORAUT-TKBS-0030) provides a full time annual external dose of 1.225 rem penetrating (gamma) and 2.932 rem nonpenetrating (beta). These value are the 95th percentile of a distribution of film badge results at the Havens Lab. As such, they are considered bounding and organ doses derived from them should be entered into IREP as a constant distribution.

Table 3 – Operational Period Annual External Dose

5/15/1962 – 10/21/1964	Annual Dose (rem/yr)
Gamma radiation	1.225
Beta radiation	2.932

CD.6 Dose from Residual Contamination

After the operations at Seymour were moved to Ashtabula, the Seymour facility was cleaned and surveyed. The initial survey was a very detailed survey performed by the company using a grid pattern for each process area (Breslin 1964). Following that survey, on 10/21/1964, HASL performed a survey of randomly selected locations from the grid pattern. Based on that survey, the area was released for unrestricted use.

In 1977, a new survey was conducted for the Department of Energy. That survey was followed up by an additional survey in 1980. All four surveys included direct alpha contamination measurements; however the 1977 survey did not include removable contamination measurements while the other three did. Also, the detailed results of the two 1964 surveys were included in reports but only summary statements of the results were made in the 1977 and 1980 survey reports.

The results of the surveys in 1964 compared to 1977 and 1980 do not appear to be significantly different. The maximum removable contamination from each survey was 112 dpm/100cm^2, 90 dpm/100cm^2 and 70 dpm/100cm^2 (there were no removable

measurements taken in 1977). The contact beta plus gamma measurements in 1964 were generally higher overall but the maximum results of the later surveys were higher than the 1964 surveys. Therefore, this TBD will assume no change in the dose rates or contamination levels over the residual contamination period.

Internal Dose
The 1980 survey report indicated the removable contamination was less than 10 dpm/100 cm^2 except for one location where it was 70 dpm/100 cm^2. The maximum removable contamination from the 1964 survey conducted by the company was 112 dpm/100 cm^2 while the maximum found by HASL was 90 dpm/100 cm^2.

The maximum value found (112 dpm/100 cm^2) is relatively low so that value will be used for this dose estimate. Since these values represent easily removable contamination, a resuspension factor of 10^{-5} m^{-1} will be applied resulting in an airborne value of 0.112 dpm/m^3. Assuming a 2000 hour work year, this results in an inhalation rate of 0.736 dpm/calendar day.

Since the intake rate in the operational period was based on bioassay, ingestion did not have to be accounted for separately. The intake rate in the residual period, however, is based on airborne and surface contamination values and must therefore include an estimate of the ingestion rate. The ingestion rate is calculated by assuming the removable surface contamination (112 dpm/100 cm^2) is ingested at a rate of 1.1E-4 m^2/hr (NUREG/CR 5512). This results in an ingestion rate of 6.75 dpm/calendar day.

Since both the inhalation and ingestion values are based on the maximum removable surface contamination measurement, they are considered bounding and doses calculated from these should be entered into IREP as a constant. The values are summarized in Table 4 below.

Table 4 – Residual Period Intake Rates

Time Period	Inhalation (dpm/calendar day)	Ingestion (dpm/calendar day)
10/22/1964 – 1993	0.736	6.75

External Dose
Each of the four surveys included contact measurements of the beta plus gamma radiation levels. These contact measurements were sometimes reported as being 1 cm from the surface. These were the only type of external radiation measurements made during the 1964 surveys but the report included all the results. The 1977 survey also included gamma dose rate measurements at 1 cm and 1 m from the surfaces but only reported the highest measurements in each of three areas. The 1980 survey also measured gamma dose rates at 1 cm and 1 m but the report only listed 1 gamma contact result and 3 beta plus gamma contact results. The report did mention that the remaining measurements were below background readings which it listed as 5 to 10 uR/hr gamma and 0.02 mrad/hr beta plus gamma.

With no beta (or beta plus gamma) measurements at 1 meter, the available contact dose rates must be corrected to a 1 meter height for use in a dose estimate. The maximum contact beta plus gamma reading was that of a crack 1 inch wide and almost 13 feet long measured in the 1977 survey and verified in the 1980 survey. This can be approximated as a line source which means as a geometry correction, the dose rate would vary linearly with the distance above the crack. Therefore, the 1 cm dose rates can be divided by 100 to estimate the 100 cm (1 meter) dose rate. This correction would not account for the additional attenuation by air so it would slightly overestimate the 1 m dose rate. The two contact readings on the crack would then represent a 1 meter dose rate of 0.0175 mrad/hr and 0.011 mrad/hr.

The beta plus gamma surface measurements in 1964 were extensive so the contact dose rates can be corrected from 1 cm to 1 meter height assuming an area source. This correction would be based on the contamination being evenly distributed throughout the area so the average values from the surveys would be the most appropriate value. The average value of these readings were 0.0726 mrad/hr and 0.0593 mrad/hr for the company and the HASL surveys respectively. To correct the dose rate, the area of the contamination must be known. According to maps from the company's survey, the 4 rooms surveyed were 50' x 24', 40' x 32', 48' x 18' and 24' x 100' (the ' symbol representing feet). These equate to 1200 square feet, 1280 square feet, 864 square feet and 2400 square feet for an average of 1436 square feet. 1436 square feet can be represented by a circle with a radius of 21.4 feet. The dose rate geometry correction from 1 cm to 100 cm for a 21.4 foot radius area is approximately 3.4 times lower than the 1 cm reading. The two corrected values would then be 0.0174 and 0.0214 mrad/hr at 1 meter.

The 4 estimates of the 1 meter beta plus gamma dose rate are in reasonably good agreement (0.0175, 0.011, 0.0174, and 0.0214). This estimate will use the highest of the four rounded up to 0.022 mrad/hr. This appears to be conservative since the 1980 survey reported that with the exception of three small locations, all the beta plus gamma readings were not significantly different than the 0.020 mrad/hr background.

There were no gamma dose rates (gamma only) taken during the 1964 surveys. The 1980 survey indicated all the 1 meter gamma dose rates were background which it indicated was 0.005 to 0.010 mR/hr. The 1977 survey only reported the three highest 1 meter gamma dose rates of 0.009, 0.006 and 0.008 mR/hr. For this dose estimate, 0.010 mR/hr will be used as the general area 1 meter dose rate. The beta dose rate will then be beta plus gamma dose rate minus this gamma dose rate which results in a beta dose rate of 0.012 mrad/hr.

In both cases, employees will be assumed to be exposed to these dose rates continuously for 2000 hours per year. This results in an annual dose rate of 20 mR gamma and 24 mrad beta. The beta plus gamma estimate used favorable assumptions that compared favorably with the estimate from the maximum readings in later years. Therefore this will be considered a bounding estimate. The gamma estimate was based on maximum

reported dose rates. Therefore both the beta and the gamma dose estimates will be considered bounding and have no uncertainty assigned to them.

Table 5 – Residual Period Annual External Dose

	Gamma Dose (mR/yr)	Beta Dose (mrad/yr)
10/22/1964 – 1993	20	24

CD.7 References

AEC 1962, Letter from John W. Ruch to D. R. Jefferson, *Health and Safety Inspection – Seymour Site*, June 13, 1962, SRDB 9588 pp. 375-376.

AEC 1962-1964, Urine Sample Data Sheets from Bridgeport Brass, Seymour, various dates in 1962-1964, SRDB 9885 pp. 185-188, 193,194, 197, 200, 201, 204, 205, 208, SRDB 9895 pp. 6-9, 12, 15-18, 22, 23, 25, 26, 28, 29.

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Breslin 1964, Letter from A. J. Breslin to John W. Ruch, *Contamination Survey at Reactive Metals, Inc., Seymour Connecticut*, October 30, 1964, SRDB 10851 pp. 21-32.

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NUREG/CR 5512 Vol. 1, *Residual Radioactive Contamination From Decommissioning*, Prepared for the Nuclear Regulatory Commission, October 1992.

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RMI 1962-1964, *AEC Summary of Whole Body Radiation Exposures*, submitted by Reactive Metals, Inc., 1964, SRDB 25304 pg 211, 212, 214, 218.