REPORT TO THE ADVISORY BOARD ON RADIATION AND WORKER HEALTH

National Institute of Occupational Safety and Health

Audit of Case #PIID* from the Huntington Pilot Plant

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S. Cohen & Associates: ***Technical Support for the Advisory Board on Radiation & Worker Health Review of NIOSH Dose Reconstruction Program*** **AUDIT OF CASE #PIID* FROM THE HUNTINGTON PILOT PLANT** Task Manager: _____ Date: 02/04/05 Hans Behling, PhD, MPH Project Manager: _____ Date: 02/04/05 John Mauro, PhD, CHP	Document No. SCA-TR-TASK4-CNPIID* Effective Date: February 4, 2005 Revision No. 1 Page 2 of 21 Supersedes: Draft Rev. 0

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SUMMARY

This report presents the results of an independent audit of a dose reconstruction performed by the National Institute of Occupational Safety and Health (NIOSH) for an individual who worked at the Huntington Pilot Plant as PIID* from PIID*, to PIID*, PIID*, at which time the worker transferred to the Huntington Administration Building.[1] From 1951 to 1963, the Huntington Pilot Plant performed work under contract to the Atomic Energy Commission (AEC). The AEC work at Huntington involved the processing of scrap nickel to produce refined nickel powder for use in the manufacture of gaseous diffusion barriers for gaseous diffusion plants. The feedstock for producing the nickel powder was uranium-contaminated nickel that originated from the Oak Ridge Gaseous Diffusion Plant (ORGDP). The contaminated nickel scrap metal was offloaded by rail at Huntington, where the nickel was loaded into a furnace, and melt refined, and in the process, the uranium contamination was separated from the nickel through a special step in the process involving the application of carbon monoxide, referred to as the nickel carbonyl process.

As a result of the worker's employment at the plant, the worker likely experienced internal exposures due to the inhalation of airborne particles of enriched uranium, and external exposure from working in the vicinity of the nickel scrap metal and the separated uranium byproduct. In addition, the worker is believed to have had routine x-rays as part of Huntington's medical surveillance program. The worker was not provided with film badge or TLD dosimeters to measure external exposures, nor were bioassays performed to estimate internal exposures. As a result, exposures experienced by the worker were estimated using the exposure matrix provided in the site profile or Technical Basis Document (TBD) prepared by NIOSH for the Huntington plant (ORAUT-TKBS-0004, January 16, 2004).

On PIID*, the employee was diagnosed with cancer of the bladder. Table 1 summarizes the results of NIOSH's reconstruction of the doses to the employee's bladder for the purpose of deriving the probability of causation (POC) using IREP. Table 1 also presents the results of the audit. The results of the audit are expressed in terms of whether we found the exposures to have been derived in a scientifically valid and claimant-favorable manner.

A review of file number PIID* (March 26, 2004) indicates some uncertainty as to the period of employment of the worker at the Huntington Pilot Plant. It appears that the employee may have worked at International Nickel Company from PIID* until PIID* which could have included work at the Huntington Pilot Plant from PIID* through PIID*.

Table 1. Summary of Internal and External Exposure to the Bladder as estimated by NIOSH, Along with the Audit Results

Exposure Scenario	NIOSH Derived Annual Doses rem)	Scientifically Valid?	Claimant Favorable?
Internal exposure from inhalation (alpha) during nickel handling and processing operations	0.4	No. An error was made in running IMBA	Yes, with some qualifications
Internal exposures from ingestion (alpha) during nickel handling and processing operations	Not Addressed	The TIB assumed that the worker stopped working when AEC operations ceased. This does not appear to be the case.	
Internal exposure from inhalation (alpha) of residual resuspended particles following the conclusion of nickel handling and processing operations	Not Addressed		
External exposures during operations			
Ground surface contamination (chronic)	1.42E-2	No	No
Stored enriched uranium (chronic)	1.38E-1	No	No
Submersion in airborne plume (chronic)	Not Addressed	Yes	NA
Diagnostic x-rays (acute)	7.4E-2	No	Yes
Chronic external exposure to residual contamination following the conclusion of nickel handling and processing operations in PIID*	Not Addressed	The TIB assumed that the worker stopped working when AEC operations ceased. This does not appear to be the case.	

1.0 INTRODUCTION

This report presents the results of an independent audit of a dose reconstruction performed by the National Institute of Occupational Safety and Health (NIOSH) for an atomic worker employee that worked at the Huntington Pilot Plant in West Virginia. This audit is one of several dose reconstruction audits being performed by S. Cohen & Associates (SC&A) on behalf of the Advisory Board on Radiation and Worker Health.

Part one of this audit report presents a summary of our understanding of the doses derived by NIOSH, along with a brief description of the basic approach and assumptions employed by NIOSH to derive the doses. This material is extracted directly from the final dose reconstruction report published by NIOSH for this case, along with supporting documentation, including the Technical Basis Document: *Basis for Development of a Exposure Matrix for Huntington Pilot Plant* (ORAUT-TKBS-0004, January 16, 2004). This section of the report is provided as a baseline for the actual audit.

Part two of the audit process consists of an attempt to independently duplicate selected doses derived by NIOSH and a discussion regarding the validity of the methods employed. The doses selected for duplication are based on the judgment of the auditors, as to the importance of the particular doses to the totality of the doses experienced by the atomic worker employee. The reason for this step in the audit process is to provide NIOSH and the Advisory Board with a level of assurance that the auditors understand how NIOSH went about deriving the doses provided in their dose reconstruction report. In the process of duplicating the NIOSH derived doses, we also provide a critical review of fundamental data, information, models, and assumptions used by NIOSH to perform the dose reconstruction. This part of the audit explores the degree to which the data are adequate to support the dose reconstruction, and whether the models and assumptions adopted by NIOSH to perform the dose reconstruction are scientifically sound and claimant favorable. Areas where the methods are found to meet these criteria, or are deemed to be inadequate with regard to these criteria, are identified and discussed. The report is not exhaustive in the review of these matters, but is limited to those areas of inquiry that are judged by the auditors to be potentially significant with respect to the dose reconstruction and the derivation of probability of causation (POC).

Methods employed by NIOSH that are found to be either scientifically inappropriate or not necessarily claimant favorable are identified, but no attempt is made to correct these deficiencies and redo the dose calculations. It is assumed that the Advisory Board and NIOSH will have an opportunity to consider the results of this audit and determine whether a revision of the dose reconstruction is needed and, if so, how to go about making the necessary revisions.

2.0 SUMMARY OF DOSES

As a subcontractor to the AEC, International Nickel Company (INCO) received uncontaminated nickel sinter and uranium-contaminated nickel scrap at their facility in Huntington, West Virginia. The material was received at a railroad siding at the facility, where it was offloaded, weighed, and placed in buckets while it was still in the steel cartons, and then charged into the furnace with the sealed cartons. After the scrap was melted, it was transferred to the nickel carbonyl chamber, where carbon monoxide gas (CO) was added to the chamber, forming two separate streams, nickel carbonyl gas and enriched uranium.

Because the contaminated nickel remained in its shipping carton until melting, NIOSH concluded that the potential for exposure of workers was low prior to melting. However, during and following melting, the potential for exposure was assumed to exist.

The atomic worker employee worked at the Huntington Pilot Plant as a PIID* from PIID*, to PIID*, at which time the worker transferred to the Huntington Administration Building. From 1951 to 1963, the Huntington Pilot Plant performed work under contract to the Atomic Energy Commission (AEC). During this time period, the facility was also referred to as the Reduction Pilot Plant because of the special nickel processing operations the facility performed on behalf of the government, which involved the handling and processing of material contaminated with enriched uranium.

In PIID*, following employment, the atomic worker employee was diagnosed with cancer of the bladder. Table 2 presents the results of NIOSH's reconstruction of the doses to the atomic worker employee's bladder for the purpose of deriving the POC using IREP.

The notations used in Table 2 to present the doses include the year in which the dose was received by the organ of interest, the statistical distribution that was used, and the key parameters for the distribution. For example, for exposure period number 1 in Table 2 (1951), a lognormal distribution was employed with a geometric mean of 5.12E-3 rem and a geometric standard deviation of 4.3 as the estimate of the internal alpha dose to the bladder due to chronic inhalation of airborne uranium particles. A discussion of various types of statistical distributions and other parameters used as input to NIOSH-IREP is provided in NIOSH (2002). The external and internal doses to the organ of interest were determined by NIOSH to be 2.5 and 9.81 rem, respectively. The POC was determined by NIOSH to 39.18% at the 99% confidence interval. On this basis, the claim was denied.

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3.0 INDEPENDENT DUPLICATION AND REVIEW OF SELECTED NIOSH-DERIVED DOSES

This section describes the dose reconstruction methods employed by NIOSH, followed by a series of hand and computer calculations that attempt to duplicate selected doses derived by NIOSH. In so doing, we will have confirmed that we understand how NIOSH performed the dose reconstruction and whether or not the calculations are correct, given the models and assumptions employed by NIOSH. In the process of attempting to duplicate the doses, we also discuss and critically review the data, models, and assumptions employed by NIOSH to reconstruct the doses.

3.1 INTERNAL DOSE FROM INHALATION

The starting point that NIOSH used to derive internal doses from inhalation was historical air particulate data collected at Huntington, which revealed concentrations ranging from 0.01 to5 mg Ni/m^3, depending on location in the plant. These data are summarized in Table 4 of the technical basis document (TBD). Using these data, NIOSH assumed that the airborne dust loading throughout the plant was lognormally distributed with a median of 0.05 mg Ni/m^3 and a geometric standard deviation of 4.3. This distribution appears to be claimant friendly, given that the worker was a PIID* PIID*. Specifically, Table 4 of the TBD indicates that PIID* worked plant-wide and the measured airborne dust loadings throughout the plant were about 0.01 mg Ni/m^3. However, we are concerned with two aspects of this analysis. First, the data in Table 4 includes both historical and more recent measurements. If only the historical data were considered applicable to historical dose reconstructions, the median dust loading would be 0.2 mg/m3 instead of 0.05 mg/m3. The second concern is that the dust loading in the breathing zone of a worker is often higher than the dust loading determined using general area air samplers. A discussion of this issue is provided in ICRP Publications 35 and 75. It is suggested that NIOSH investigate the degree to which these two issues could impact the reconstructed inhalation doses and the associated probability of causation (POC) for employees at the Huntington Pilot Plant.

An additional concern that does not apply to this case, but may apply to other Huntington cases, is that the TBD provides direction that the median dust loading is appropriate for use in all cases that may arise from the Huntington Pilot Plant. We believe that a more claimant-favorable approach is to assume that, lacking information regarding a worker's specific responsibilities at the plant, it should be assumed that claimants were members of the critical population groups at the facility, where the assumed average or median exposures were toward the high end of the distribution, perhaps the 90^{th} or 95^{th} percentile value. At the 90^{th} percentile level, the dust load is estimated to be 5.57 mg Ni/m^3 and the 95^{th} percentile level is 7.15 mg/m^3 using the data in Table 4 of the TBD. It is suggested

that NIOSH investigate the degree to which these two issues could impact the reconstructed inhalation doses and the associated POC for employees at the Huntington Pilot Plant.

The level of uranium contamination on the nickel starting material, as reported in the TBD, is as follows:

Uranium Isotope Ratio to Ni (pCi/mg)

U-235 0.425 U-234 10.901 U-238 0.102

This level of contamination is based on the assumption that the residual contamination of uranium on the nickel was 39% enriched and had the following composition:

Isotope	Weight Percent	Activity Percent
U-234	0.3471	94.9106
U-235	39	3.6914
U-236	0.1794	0.5102
U-238	60.4735	0.8877

In the first draft of this audit report, SC&A expressed some concerns with the assumptions employed in the TBD that the nickel scrap received from the gaseous diffusion plants was contaminated only with uranium isotopes and their progeny. Various literature sources have documented that nickel scrap generated in the same general time frame was also contaminated with isotopes of Tc, Np and Pu. Appendix A presents a description of the basis for this concern.

During the factual review meeting with NIOSH on January 12, 2005, NIOSH concurred that issues regarding recycled uranium need to be explicitly addressed in these types of analyses. However, NIOSH also pointed out that, in this particular case, NIOSH employed the assumption that all the uranium was 39% enriched in U-235, when, in fact, the enrichment level was much lower. This assumption, coupled with the fact that the starting point for the dose reconstruction was measured airborne dust loadings expressed in units of mass, the reconstructed doses were derived in a highly conservative manner, which more than accounted for any small incremental doses associated with trace levels of other radionuclides that may have been present in recycled uranium. We concur with NIOSH with regard to this matter. Nevertheless, it is suggested that in the future, NIOSH investigate whether isotopes of Tc, Np, and Pu may have also been present in the feedstock, and if so, evaluate the potential significance of this additional contamination on the reconstructed doses. NIOSH could then provide the rationale for not explicitly addressing this particular contribution to worker exposures.

Accepting the uranium contamination levels at face value, the median airborne concentration of uranium dust is estimated in the TBD as 0.58 pCi/m^3. The annual dose to the bladder was derived by NIOSH using IMBA, and assuming a breathing rate of 1.2 m^3/hr, 2,000 hours per year, (or 1,400 pCi/yr as the inhalation rate), Absorption Type M, and 5 micron AMAD. As indicated in Table 2, beginning in 1951, the exposure rate to

the bladder continually increased as the body burden increased. The exposures leveled off at about 0.4 rem/yr and then, after 1962, the time when processing of the contaminated nickel ceased, the exposures gradually declined due to the clearance of the uranium from the body.

An independent check on this calculation confirms the uranium inhalation rate of 1,400 pCi/yr, with the following isotopic distribution; U-234 = 1308 pCi/yr, U-235 = 51 pCi/yr, U-238 = 12.24 pCi/yr, and U-236 = 7.1 pCi/yr. Using this inhalation rate for enriched uranium, we were not able to duplicate the annual doses to the bladder of about 0.4 rem/yr as reported in the TBD. Our dose estimate using IMBA and another internal dosimetry code resulted in substantially lower doses (i.e., <1 mrem/yr versus 400 mrem/yr to the bladder). Inspection of the IMBA file contained in the claimant file package indicates that an error was made in the input for the uranium inhalation rate. Instead of inputting the inhalation rate in pCi/day, as required by IMBA, the value that was input into the pCi/day field appears to be the **total quantity of uranium inhaled by the worker** for the full duration of employment. This error corresponds to about a 4,000-fold overestimate in assigned internal dose. It would appear that, at least in this case, there was a breakdown in quality control that needs to be investigated.

The TBD indicates that exposures terminated after AEC nickel processing operations ceased in 1963, and no additional intake occurred between 1963 and when the employee stopped work in 1972. Apparently, as indicated in File No. PIID* (March 26, 2004), the employee continued to work at the Huntington Pilot Plant until PIID*. The TBD assumed that he stopped working at the plant in PIID*. This does not appear to be the case. If he did in fact continue to work at Huntington from PIID* to PIID*, he may have experienced additional internal and external exposures associated with residual radioactivity at the facility because, according to the TBD, the facility was not decontaminated until 1978. As a result, the worker may have experienced additional exposures that are not accounted for in his dose reconstruction. During our meeting with NIOSH on January 12-13, 2005, NIOSH indicated that they are re-evaluating the potential that this individual continued employment at the Huntington Pilot Plant through PIID*.

3.2 EXTERNAL DOSE

Enriched uranium contains U-238, U-234, U-235, and U-236, along with their short-lived progeny, which grew in after the uranium was separated from the ore. The short-lived progeny include primarily Th-234 ($T_{1/2}$ = 24 days) and Pa-234m ($T_{1/2}$ = 1.17 minutes) from the U-238 series, and Th-231 from U-235. These radionuclides emit mostly weak gammas, x-rays, and beta particles with their associated bremsstrahlung radiation. In addition, Pa-234m decays to Pa-234, which emits a strong gamma with a branching fraction of 0.16%.

The TBD evaluated several sources of potential exposure to external radiation, including (1) exposure to surfaces contaminated with deposited uranium dust, (2) exposure due to

proximity to the carbonyl process residue, and (3) exposure to contaminated skin. For the employee, however, beta doses to skin and near-surface organs and tissue were ignored, because the betas (and associated bremsstrahlung) do not have sufficient penetrating power to expose the bladder.

Table 3 presents the external exposures experienced by the organ of interest as reported in the TBD.

Table 3. External Exposure to the Bladder

External Exposure Scenario	Annual Dose to Bladder (rem)	
	E= 30-250 keV	E>250 keV
During Ni Processing (1951-1963)		
Ground surface contamination (chronic)	6.43E-3	7.76E-3
Birdcages (chronic)	8.09E-2	5.74E-2
Diagnostic x-rays (acute)	7.4E-2	
Submersion	Negligible	Negligible
Chronic residual exposures following decon (after PIID*)	6.43E-3	7.76E-3

3.2.1 External Exposure to Uranium Deposited on Surfaces During Operations

Two separate methods were used to estimate the external exposures due to uranium deposited on surfaces. The first method involved multiplying the airborne dust loading described above by the deposition velocity. The deposition velocity was assumed to be 0.002 m/s, as cited in CRC 1986 for 5 micron particles with a density of 2 g/cm^3 in air at 25°C and 1 atm. This deposition rate was assumed to continue for 10 years without any removal due to building ventilation, cleanup, or natural weathering. Based on this buildup, the external dose rate was derived using the dose conversion factors provided in *Federal Guidance Report No. 12*. This approach resulted in an external dose rate of <1 mrem/yr, which was low when compared to the actual external dose rates measured in a survey performed in 1981. Hence, the 1981 survey results were used to derive external exposures during and following operations.

Though the deposition velocity approach was not used in this case for deriving external exposures from deposited activity, it is worth noting that actual measurements reveal that this approach is not claimant favorable, at least not in this case. The reason for bringing this up is, in other cases, NIOSH may have employed this method for estimating exposures from residual deposited activity, believing it to be claimant favorable. The implications of the survey results are that the deposition velocity approach may not be claimant favorable. We suspect that the reason for this is residual contamination on surfaces may have come about from spills and from the deposition of particles much larger than 5 micron, which may have had a much higher deposition velocity than those that apply to 5 micron particles.

According to the 1981 survey report, gamma radiation readings ranged from 9-12 µR/hr,

with a maximum of 23 µR/hr at 1 meter above the floor. The TBD assumed that the exposure rates were lognormally distributed, with a median of 12 µR/hr and a geometric standard deviation of 1.9 (which corresponds to a 95% confidence level of 35 µR/hr). To convert the exposure rate to the dose rate to the organ of interest, the Exposure (R) to Organ Dose (H_T) photon dose conversion factors provided in Appendix B of NIOSH *External Dose Reconstruction Implementation Guideline* (NIOSH 2002a) were used. The exposure geometry was assumed to be isotropic, and the dose rate was divided equally between photons with E=30-250 keV and photons with E>250 keV.
Since the external gamma readings were obtained in 1981, which was several years after AEC operations ceased and after decontamination of the facility took place in 1978, the use of the 1981 survey data for determining exposures to deposited radioactivity during operations is highly questionable. It is suggested that NIOSH revisit this issue using a strategy for estimating exposures to deposited radioactivity that more realistically represents the conditions at the time of operations. One approach may be to obtain data on the surface contamination of nickel dust at other nickel-melting operations.

3.2.2 External Exposures to Uranium in Birdcages During Operations

Following melting and the carbonyl separation process, the separated uranium was decanted into containers referred to as "birdcages." Birdcages are made up of 5 by 5 arrays of cylinders 4.25 inches in diameter and 9.75 inches high, each containing no more than 350 grams of U-235. These 5 by 5 arrays were stacked in no more than 2 layers to avoid criticality. It was assumed that a worker was located 1 meter from the 5 by 5 double-stacked array for 2,000 hours per year, resulting in an annual exposure rate of 0.13 R. The upper 95th percentile value of 0.76 R/y was calculated assuming 2,000 hours per year of exposure at 1 foot away. To convert the exposure rate to the dose rate to the organ of interest, the Exposure (R) to Organ Dose (H_T) photon dose conversion factors provided in Appendix B of NIOSH *External Dose Reconstruction Implementation Guideline* (NIOSH 2002a) were used. The exposure geometry was assumed to be anterior-posterior and the dose rate was divided equally between photons with E=30-250 keV and photons with E>250 keV. In order to confirm these values, we first estimated the quantity of uranium that may have been produced per year, and compared that quantity to the quantity of uranium housed in the birdcages. We then independently estimated the air exposure rate using MicroShield. Appendix B presents a description of our MicroShield analysis.[2] The birdcages are 5 by 5 double-stacked containers, each containing a maximum of 350 grams of U-235. Accordingly, the total inventory of enriched uranium in the 50 containers is 44.9 kg. According to the TBD, up to 2,587,000 pounds of nickel were shipped from ORGDP to INCO per year from 1950 to 1961. Given that the starting material contained 11.428 pCi U/mg Ni (i.e., about 5e-7 g U/mg Ni), the annual production rate of enriched uranium was about 582 kg per year, or enough to fill up about 13 arrays per year.
Using MicroShield, we estimated the air exposure rate 1 meter and 1 foot from a 5 by 5 double array to be 0.91 R/y and 2.38 R/y, respectively. These air exposure rates correspond to a bladder dose of 0.97 rem/y at 1 meter and 2.53 rem/yr, respectively. The dose to the bladder (at 1 meter away) reported in the TBS is 0.138 rem/y. The difference

between our derived bladder doses and those reported in the TBD are large enough to raise some question regarding the accuracy of the values in the TBD. [2] The MicroShield analysis described in Appendix B is considered to be a reasonable approximation of the exposure. A more precise analysis would involve the use of MCNP.

3.2.3 Annual X-Ray Exposures

NIOSH also assumed that the workers received annual chest x-ray examinations as a part of routine medical surveillance of workers. The air kerma skin dose, which was assumed to be 0.108 R for each diagnostic chest x-ray (Scalsky 2003), was multiplied by the Kerma (Ka) to Organ Dose (Ht) photon dose conversion factors for 30-250 keV photons provided in Appendix B of DHHS 2002 to obtain the annual dose to the bladder of 0.074 rem. Unlike the other exposures, these exposures were assumed to be acute and the uncertainty in the x-ray exposures was assumed to be normally distributed, with a standard deviation of 30%.

Given that the bladder is outside the primary beam of a chest x-ray, it is questionable whether the high bladder dose assumed in the TBD is scientifically appropriate. A more appropriate approach would have been to employ the method described in ORAUT-OTIB-006, *Technical Information Bulletin: Dose Reconstruction from Occupationally Related Diagnostic X-ray Procedures*, December 29, 2003. Secondly, in accordance with the guidance and lacking information to the contrary, it would have been appropriate to assume that at least some examinations included photofluorography, because such examinations were a matter of standard practice prior to 1960. Table 4.0-1 of the guideline recommends a dose to the bladder of 0.025 rem for each photofluorography. Using this approach, as opposed to the approach used in the TBD, the medical exposures would be lower by about a factor of three.

3.2.4 External Dose After Cessation of AEC Operations

NIOSH assumed that the worker terminated employment at the facility in PIID*. However, as described above, it appears that employment at the Huntington Pilot Plant continued until PIID*. As a result, external exposures may have continued after PIID* due to residual radioactivity at the facility, until employment was terminated in PIID*. These exposures should be added to the worker's reconstructed doses if the worker did, in fact, continue to work at the facility after PIID*.

The TBD provides estimates of post-operational exposures due to residual radioactivity, which persisted until decontamination of the facility in November 1978. NIOSH was unable to find any pre-decontamination survey data. However, as described above, post-decontamination surveys revealed exposure rates of about 9 to 12 µR/hr, with a maximum of 35µR/hr. As a means to estimate the pre-decontamination

exposure rates, NIOSH assumed that the median exposure rate was 35 µR/hr and the upper 95% confidence level was 250 µR/hr, which corresponds to the radiation exposure limit of 500 mrem/yr recommended by the NCRP at that time. The annual dose rate to the bladder was then derived using the Exposure (R) to Organ Dose (rem) photon dose conversion factors from Appendix B of NIOSH 2002a. The exposure geometry was assumed to be isotropic, and the dose rate was divided equally between photons with E=30-250 keV and photons with E>250 keV.

This approach to deriving the post-PIID* external exposure rates appears to be somewhat arbitrary. There is no reason to believe the highest air exposure rate observed post-decontamination bears any relationship to the exposure rate prior to decontamination. NIOSH may want to revisit this issue and develop a dose reconstruction methodology that has a better scientific foundation.

REFERENCES

NIOSH 2002, "NIOSH-Interactive Radioepidemiological Program (NIOSH-IREP) Technical Documentation," Final Report. June 18, 2002.

NIOSH 2002a, "External Dose Reconstruction Implementation Guideline," OCAS-IG-0001, Rev 1, August 2002.

ORAUT-TKBS-0004, "Technical Basis Document: Basis for Development of an Exposure Matrix for Huntington Pilot Plant," January 16, 2004.

ORAUT-OTIB-0006, Kathren, R.L., Shockley, V.E., Thomas, E.M., Taulbee, T.D., Technical Information Bulletin: Dose Reconstruction from Occupationally Related Diagnostic X-ray Procedures, 2003.

APPENDIX A: RADIOACTIVE CONTAMINATION IN NICKEL FEEDSTOCK

Radioactivity in the gaseous diffusion plants consists of U-234, U-235, and U-238, which occur naturally in uranium ores,[1] their short-lived progeny, and Tc-99, U-236, and trace quantities of Np-237 and Pu-239. This contamination was introduced into the plants with reprocessed uranium fuel from plutonium production reactors at Hanford and Savannah River (NAS 1996). Eighty-eight percent of the reprocessed uranium was shipped to and introduced into the uranium enrichment facilities at the PGDP, and 12% initially went to the K-25 Site. Of the contamination in the reprocessed uranium feed shipped to K-25, only about 85% of the Tc-99 and about 25%t of the Np-237 and Pu-239 actually entered the cascade, because variable yields of different elements occurred when the uranium (as UO_3) is converted to UF_6, and when the UF_6 is vaporized prior to being fed into the cascade (Smith 1984). (For example, during feedstock vaporization, PuF_6 and NpF_6 would tend to remain in the storage cylinders.) Similarly, at K-25, only 25% of the Np-

237 and 1.5% of the Pu-239 entered the cascade. Smith (1984) estimated that 539 kg of Tc-99, 4.6 kg of Np-237, and 0.1 g of Pu-239 were fed into the Paducah cascades from 1953 until September 11, 1975, when use of reactor return feeds was discontinued. Much of the contamination that had been deposited in the uranium enrichment facilities prior to 1975, when this feed source was eliminated, was removed during the cascade improvement and upgrade programs (CIP/CUP) in the 1980s (NAS 1996). Some of this contamination is also present at the Portsmouth plant.

[1] Natural uranium contains 0.712% U-235, 99.282% U-238, and trace amounts of U-234 (0.006%).

REFERENCES

NAS 1996. National Academy of Sciences/National Research Council, "Affordable Cleanup? Opportunities for cost reduction in the decontamination and decommissioning of the nation's uranium enrichment facilities." National Academy Press, Washington DC.

Smith, R.F 1984. "Historical Impact of Reactor Tails on the Paducah Cascade," KY/L-1239, Paducah Gaseous Diffusion Plant, March 19, 1984.

APPENDIX B: EXTERNAL EXPOSURE TO AIR FROM BIRDCAGE ARRANGEMENT OF URANIUM CYLINDERS: MICROSHIELD ANALYSIS

In order to simulate the "birdcage" container (described in Figure 2 of the Huntington Pilot Plant TBD) for MicroShield, we assumed that the amount of uranium in the 50 cylinders that make up a birdcage is evenly distributed within the volume of the entire container. The volume of the container is calculated to be 5,243,860 cm^3. Using the maximum mass control of 350 g of U-235, along with a maximum of 39% by weight of U-235, the mass of enriched uranium in 50 cylinders is calculated to be 44.9 kg. The density of the material is therefore 8.55E-3 g/cm^3. The activities of the uranium isotopes and their short-lived progeny are calculated based on the weight percentages presented in Table 2 of the TBD and the maximum mass control of 350 g of U-235. The short-lived progeny is assumed to be in equilibrium with its parents. Those calculations are presented Table B-1.

Enriched Uranium in 50 cylinders (birdcage)

Table B-1. Characteristics of the Enriched Uranium in the Birdcages Nuclide	**Weight % in EU**	**Weight (g)**	**pCi/g of EU**	**Ci in birdcage**
U234	0.3471%	156	2.16E+07	9.71E-01
U235	39%	17500	8.43E+05	3.78E-02
Th231				3.78E-02
U236	0.1794%	81	1.14E+05	5.21E-03
U238	60.4735%	27136	2.02E+05	9.14E-03
Th234				9.14E-03
Pa234m				9.14E-03
Pa234				1.46E-05
Total EU		44872	2.28E+07	1.02

The geometry dimensions used to simulate the external dose from the birdcage are presented in the figure below. Simulations were performed at both 1 foot and 1 meter away.

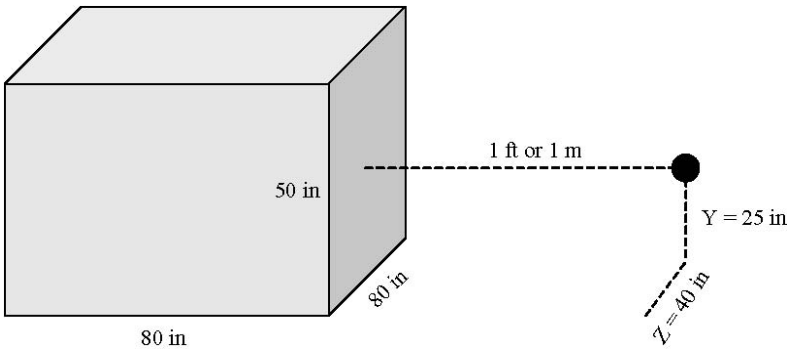

Figure B-1. Schematic of the Exposure Geometry

The analysis was also performed with and without the short-lived progeny and with and without shielding. The shield is assumed to be the equivalent of a 55-gallon carbon steel drum, with a wall thickness of 1.5189 mm (ORNL No. 100-1A-2-0006) and a density of 7.8 g/cm^3 (*Handbook of Health Physics and Radiation*, 3rd ed., Table 5.4). The calculated activities and the described geometries were used as input into MicroShield. Table B-2 presents the results.

Table B-2. MicroShield Results

Distance from source	**Air Exposure with short-lived progeny (R/yr assuming 2000 hrs/yr exposure)**	
	No shield	**with shield**
1 meter	1.08	0.91
1 foot	2.93	2.38

Distance from source	**Air Exposure without short-lived progeny (R/yr assuming 2000 hrs/yr exposure) U-234, U-235, U-236, U-238 only**	
	No shield	**with shield**
1 meter	0.90	0.78
1 foot	2.46	2.0