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National Institute for Occupational Safety and Health

EVALUATION OF URANIUM DOSE RECONSTRUCTION METHODS AT CPP (1953–1962)

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ABBREVIATIONS AND ACRONYMS

Ac	actinium
Am	americium
ANL-W	Argonne National Laboratory – West
Cm	curium
CPP	Chemical Processing Plant
DCAS	Division of Compensation Analysis and Support
dpm/d	disintegrations per minute per day
FAP	fission and activation products
ICPP	Idaho Chemical Processing Plant
IMBA	Integrated Modules for Bioassay Analysis
INL	Idaho National Laboratory
MDA	minimum detectable activity
$\mu Ci/cm^3$	microcurie per cubic centimeter
NIOSH	National Institute for Occupational Safety and Health
NOCTS	NIOSH/OCAS Claims Tracking System
Np	neptunium
Pa	protactinium
pCi	picocurie
Pu	plutonium
SEC	Special Exposure Cohort
Sr	strontium
TBD	technical basis document
Th	thorium
TLD	thermoluminescent dosimeter
TWOPOS	time weighted one-person-one-statistic
U	uranium

1 INTRODUCTION AND BACKGROUND

As part of the Idaho National Laboratory/Argonne National Laboratory-West (INL/ANL-W) Work Group's review of the National Institute of Occupational Safety and Health (NIOSH) evaluation report for Special Exposure Cohort (SEC) Petition SEC-000219 (NIOSH 2017), SC&A was tasked to investigate potential SEC issues related to other areas and time periods apart from the proposed class at the Chemical Processing Plant (CPP), which is defined as follows:

> *All employees of the Department of Energy, its predecessor agencies, and their contractors and subcontractors who worked at the Idaho National Laboratory (INL) in Scoville, Idaho, and (a) who were monitored for external radiation at the Idaho Chemical Processing Plant (CPP) (e.g., at least one film badge or TLD dosimeter from CPP) between January 1, 1963 and February 28, 1970; or (b) who were monitored for external radiation at INL (e.g., at least one film badge or TLD dosimeter) between March 1, 1970 and December 31, 1974 for a number of work days aggregating at least 250 work days, occurring either solely under this employment, or in combination with work days within the parameters established for one or more other classes of employees in the Special Exposure Cohort.* [NIOSH 2017]

As part of SC&A's review tasking, a focused review of the CPP prior to 1963 was performed. In a joint effort by the Work Group, NIOSH, and SC&A, several data captures and interviews with former INL workers were conducted between January and December 2016. These activities included:

- January 25, 2016, to January 27, 2016: In-person interviews with former INL workers
- January 28, 2016: INL onsite data capture
- February 16, 2016: Telephone interviews with former INL workers
- March 15, 2016, to March 16, 2016: INL onsite data capture
- April 5, 2016: Telephone interviews with former INL workers
- November 8, 2016, to November 10, 2016: In-person interviews with former INL workers
- December 15, 2016: Telephone interviews with former INL workers

In July 2017, SC&A delivered its evaluation of dose reconstruction feasibility at CPP prior to 1963, which focused on the ability to reconstruct doses to alpha-emitting radionuclides during this period (SC&A 2017). That report contained five findings and five observations related to the ability to reconstruct doses to alpha-emitting material (primarily isotopes of uranium) that was no longer comingled with fission and activation products (FAP), which are beta/gamma-emitting material. This is of particular import because current methods for reconstructing doses to alpha-emitting material for workers who were not directly monitored for alpha contaminants is to use a ratio to the individual worker's intake based on gross beta and/or gross gamma urinalysis. The summary conclusion of SC&A report, as stated in the Executive Summary of that document, is as follows:

> ***Summary Conclusion:*** *SC&A identified several example locations and time periods for which alpha contamination was identified and was not directly comingled with FAP. Reconstruction of internal exposures to alpha material by ratioing to calculated intakes of FAP material would not be technically appropriate for at least some workers, activities, and locations within CPP.* [SC&A 2017, p. 7]

In April 2018, NIOSH provided a white paper response (NIOSH 2018a) to SC&A's evaluation of alpha exposures at CPP during the pre-1963 era. That white paper provided detailed responses to each of SC&A's five findings and five observations and came to the following conclusion:

> *In summary, based on further evaluation of the findings and observations from* Draft Review of Internal Alpha Exposure Potential at CPP Prior to 1963, *DCAS still believes it can reconstruct doses with sufficient accuracy for workers at CPP before 1963. This conclusion was largely reached due to the comprehensive Health Physics at CPP, which included tight contamination controls, airborne radioactivity monitoring, attention to changes in source term based on process knowledge with appropriate monitoring, and a large* in-vitro *bioassay program. There was a small subset of the CPP workforce that was on a routine uranium bioassay program due to internal exposure potential to uranium without fission products present. Internal monitoring for most of the CPP workforce for alpha-emitting material was limited, as such monitoring was typically incident-based, as the actual exposure potential was likely restricted to certain operations and analytical laboratory personnel. Special bioassay monitoring was identified in interviews with former CPP workers, as well as in bioassay records and incident reports.* [NIOSH 2018a, p. 6]

As there was no discussion in NIOSH 2018a of alternate dose reconstruction methods for exposure to alpha-emitting material that was not comingled with FAP material, the logical implication is that the Division of Compensation Analysis and Support (DCAS) plans to continue to use the ratio method to derive bounding intakes of alpha-emitting material when uranium or other bioassay data are not available for an individual energy employee.

To evaluate whether the ratio method would indeed bound potential intakes of alpha-emitting material (specifically uranium), SC&A compiled the gross beta and gross gamma urinalysis data for the 32 claimants discussed in Appendix A of SC&A 2017. These 32 claims were identified in SC&A 2017 as working in CPP and having job titles most likely to be associated with laboratory areas and the final product stages where alpha-emitting material not comingled with FAP material would be present. These job titles included [redacted], [redacted], [redacted], [redacted], [redacted], and [redacted]. SC&A then used the gross beta and gross gamma bioassay for these claims to calculate intakes of FAP material, which can be used in conjunction with the ratios presented in the INL/ANL-W internal dose technical basis document (TBD), ORAUT-TKBS-0005-5, Revision 03 (NIOSH 2010a[1]), to arrive at intakes of uranium. The intakes of uranium using the ratio method can then be compared to alternate methods of calculating uranium intakes to determine if they are indeed bounding.

Section 2 of this report presents the evaluation of uranium intakes using the ratio method and available claimant bioassay compiled from the 32 claimants discussed in SC&A 2017. Both the individual claimant intakes are evaluated (see Section 2.2), as well as a simulated coworker approach in which all of the data for the 32 individuals are combined (see Section 2.3). Section 3 presents alternate methods of calculating intakes of uranium based on air sampling data (Section 3.1), and uranium urinalysis (Section 3.2). Section 4 summarizes the comparison of derived uranium intakes using the different methods discussed in the previous two sections and provides SC&A's summary recommendation.

[1] It should be noted that the ratios presented in NIOSH 2010a are currently under review by the INL/ANL-W Work Group and are the subject of the SC&A white paper, *SC&A's Evaluation of the NIOSH Evaluation Report Proposed Use of Fission-Activation Product Bioassay Indicator Radionuclides (in Conjunction with ORAUT-OTIB-0054 and ORAUT-TKBS-0007-5) for Assessment of Fission-Activation Product and Actinide Intakes at Idaho National Laboratory* (SC&A 2015).

2 SUMMARY OF CLAIMANT INTAKE EVALUATIONS OF URANIUM USING GROSS BETA URINALYSIS

As noted in the introduction to this report, NIOSH's currently proposed method for reconstructing doses to uranium involves evaluating the energy employee's available gross beta and/or gross gamma urinalysis data and applying a ratio to obtain intakes to alpha-emitting material. A full description of this proposed dose reconstruction method can be found in Section 5.5.2 of ORAUT-TKBS-0005-5, Revision 03 (NIOSH 2010a).

Appendix A of SC&A 2017 identified 32 claimants who worked at CPP during the period of interest and had job titles or other information to indicate work in and around the CPP laboratory facilities or final product staging areas. These are the plant areas most likely to contain source terms in which alpha-emitting uranium material could be found separated from the fission and activation products that were considered waste material. To characterize the magnitude of assigned uranium intakes based on the currently proposed dose reconstruction methodology, SC&A used the Integrated Modules for Bioassay Analysis (IMBA) program to evaluate the available gross beta urinalysis data[2] for these claims and assuming the activity was entirely strontium-90 (Sr-90). Fifteen of the 32 claimants had sufficient gross beta urinalysis to effectively model a chronic intake of Sr-90. The derived daily intake rates of strontium were then used with ratios developed in the INL internal dose TBD to establish an associated uranium intake rate.

In addition to using IMBA to model the chronic intakes for each of the individual workers, SC&A simulated the development of a coworker model for the 32 claims using all available gross beta urinalysis for those claims (20 of 32 had at least some gross beta urinalysis results). Section 2.1 describes the assumptions that SC&A used in both the individual and simulated coworker intake calculations. Section 2.2 summarizes the results for the individual chronic intake calculations, and Section 2.3 discusses the simulated coworker results. For more detailed information on the claimant-specific intake calculations, see Attachment A.

2.1 Assumptions Used in Intake Calculations

The following subsections describe the assumptions SC&A employed in developing the intake calculations using IMBA. These assumptions include:

- Development of intake evaluation regimes that reflect the assumed starting and ending dates for the modeled chronic intake period (see Section 2.1.1)
- Intake mode and solubility type (see Section 2.1.2)
- Ratio used to arrive at an intake of uranium relative to the modeled intake of Sr-90 (see Section 2.1.3)
- Treatment of urinalysis results that were less than the limit of detection, including results that had a reported activity of zero (see Section 2.1.4)

[2] Gross beta urinalysis was chosen for analysis over gross gamma because there were over three times more gross beta urinalysis samples among the 32 reviewed claims.

2.1.1 Intake Evaluation Regimes

The start date for each intake evaluation was assumed to be the first covered employment date during the period of interest, which is defined as February 1, 1953, through December 31, 1962. The start date of the period of interest is based on the fact that the first hot run at CPP occurred in February 1953 (NIOSH 2010b, p. 25). If the start of covered employment for a given claim occurred prior to February 1, 1953, then the start of the evaluated intake period was assumed to coincide with the first hot run at CPP. The end date for each intake evaluation was assumed to be the date of the final gross beta urinalysis result during the period of interest (February 1, 1953–December 31, 1962).

2.1.2 Intake Mode and Solubility Type

For the purposes of calculating an intake from the available gross beta urinalysis, it was assumed that the material was strontium Type F and the material was inhaled. SC&A evaluated Type F strontium because of the following statement in the INL internal TBD:

> *Because the available information on the INEL Site does not indicate that strontium titanate ($SrTiO_3$) was ever present at the site and because strontium titanate was an uncommon strontium compound, strontium only needs to be assessed as type F material.* [NIOSH 2010a, p. 16]

Therefore, only Type F Sr-90 was considered in this evaluation.

2.1.3 Ratio of Uranium to Strontium

Section 5.5.2 of the INL internal TBD (NIOSH 2010a) provides ratios of various actinide contaminants to gross beta and gross gamma urinalysis. SC&A's review of the internal dosimetry files for the 32 claimants described in Appendix A of SC&A 2017 determined that gross beta urinalysis was the most common and routinely found bioassay method during the period of interest; therefore, this is the bioassay method analyzed in this section. Table 5-22 of NIOSH 2010a provides the actinide-to-Sr-90 ratios for use in dose reconstruction; these ratios are shown in Table 1 below for convenience.

Specific to the Chemical Processing Plant, Section 5.5.2 of NIOSH 2010a states the following:

> *For the ICPP, the actinide ratios for the aluminum fuels likely provide a reasonable overestimate of the actinides present at the ICPP before 1971, because the non-aluminum fuels that were reprocessed at the ICPP before 1971 had much lower burnups than what was assumed for the aluminum fuels.* [page 41]

Therefore, SC&A assumed the Sr-90-to-uranium ratio for aluminum reactor fuels as shown in italicized in red in Table 1.

Table 1. Actinide-to-Sr-90 Ratios Presented in NIOSH 2010a (Table 5-22)

Actinide	Aluminum Reactor Fuel Type Ratio (Isotope to Be Used)	Zirconium Reactor Fuel Type Ratio (Isotope to Be Used)	Stainless-Steel Reactor Fuel Type Ratio (Isotope to Be Used)	Maximum Reactor Fuel Type Ratio (Isotope to Be Used)
Actinium (Ac)	8.0E-12 (Ac-227)	1.3E-11 (Ac-227)	2.3E-10 (Ac-227)	2.3E-10 (Ac-227)
Thorium (Th)	2.4E-08 (Th-228)	6.4E-08 (Th-228)	2.3E-07 (Th-228)	2.3E-07 (Th-228)
Protactinium (Pa)	1.2E-10 (Pa-231)	1.1E-10 (Pa-231)	3.8E-09 (Pa-231)	3.8E-09 (Pa-231)
Uranium (U)	***5.6E-05 (U-234)***	6.2E-06 (U-236)	1.4E-03 (U-234)	1.4E-03 (U-234)
Neptunium (Np)	3.4E-06 (Np-237)	3.7E-06 (Np-237)	6.8E-07 (Np-237)	3.7E-06 (Np-237)
Plutonium (Pu)	8.7E-03 (Pu-238)	1.5E-02 (Pu-238)	3.7E-03 (Pu-239)	1.5E-02 (Pu-238)
Americium (Am)	1.4E-04 (Am-241)	3.9E-06 (Am-241)	9.0E-08 (Am-241)	1.4E-04 (Am-241)
Curium (Cm)	4.9E-05 (Cm-244)	1.8E-06 (Cm-244)	1.1E-10 (Cm-242)	4.9E-05 (Cm-244)

2.1.4 Treatment of Zero and Less than Minimum Detectable Activity

Section 5.2.1 of NIOSH 2010a lists the minimum detectable activity (MDA) for gross beta urinalysis as 24,000 disintegrations per minute per day (dpm/d) for the years 1951–1953 and 26,000 dpm/d for the years 1954 through 1960. Of the 15 claimants included in this internal dose assessment, only 5 (or 33%) had gross beta bioassays with reported results above ½ of the applicable MDA. Only 7 of the 293 gross beta bioassay samples (~2.4%) included in the analysis were greater than ½ the MDA. Only a single bioassay out of the 293 evaluated samples had results above the full MDA. To account for this, SC&A used two different analytical approaches to estimate the intake rate of Sr-90 from gross beta urinalysis for the individual workers. These analytical methods can be summarized as follows:

1. **Fitted Sr-90 Intake Rate:** Each non-zero analytical result was included in the IMBA fitting calculation regardless of whether it was greater than ½ of the MDA. Bioassay values that were reported as zero were excluded from the calculation but can be seen in the IMBA-generated urinalysis figures shown in Attachment A.

2. **Adjusted Sr-90 Fitted Intake Rate:** all bioassay results that were greater than ½ of the MDA were used "as is" in the IMBA fitting calculation. If the first and/or last bioassay results in the intake evaluation were less than ½ of the MDA, they were set to ½ of the MDA. All other bioassay results were excluded.

For the coworker simulation presented in Section 2.3, it was necessary to first calculate the time-weighted one-person-one-statistic (TWOPOS) value for each year under evaluation in accordance with the instructions in ORAUT-RPRT-0053, Revision 02, *Analysis of Stratified Coworker Datasets* (NIOSH 2014). Each of these TWOPOS values was non-zero and thus no adjustments were made in the coworker simulation.

2.2 Summary of Daily Intake Rates of Sr-90 and Uranium Using Individual Claimant Gross Beta Bioassay

Table 2 presents the results of the IMBA calculations for 15 claimants with extensive gross beta bioassay results during the period of interest. In addition to the IMBA-derived estimates of Sr-90 intake, Table 2 shows the resulting intake rate of uranium based on the ratio method in NIOSH 2010a. As seen in the table, estimates of individual uranium intake rates using non-zero claimant bioassay data (analytical method 1 as described in the previous section) ranged from 0.25 picocuries per day (pCi/d) to 1.08 pCi/d with an average value of 0.57 pCi/d. Adjusting the IMBA calculation to only include values greater than ½ of the MDA and assuming the starting and ending bioassay was equal to ½ of the MDA (analytical method 2 as described in the previous section) yielded slightly larger estimates of the daily intake rate ranging from 1.36 to 1.96 pCi/d with an average of 1.48 pCi/d.

Table 2. Summary of IMBA Fitted Intake Rates and Resulting Uranium Intakes Using the Ratio Method

Case ID**	Total Gross Beta Bioassay	IMBA Fitted Sr-90 Type F Intake Rate (pCi/d)	Resulting Uranium Intake (pCi/d)	Adjusted IMBA Fitted Sr-90 Type F Intake Rate (pCi/d)	Adjusted Uranium Intake Rate (pCi/d)
A	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
B	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
C	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
D	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
E	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
F	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
G	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
H	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
I	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
J	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
K	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
L	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
M	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
N	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
O	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Average	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]

* Due to a break in the covered employment, these intake rates represent the weighted average of two evaluated intake periods.
** Case ID is an arbitrary designation that does not have any direct connection to the claimant. Attachment B provides the actual NOCTS claim numbers associated with each Case ID designation for reference.

While only Type F Sr-90 intakes were evaluated due to the instructions in NIOSH 2010a and discussed in Section 2.1.2 of this report, it should be noted that scoping calculations indicate that

Type M and Type S Sr-90 would increase the associated intakes by an average factor of approximately 3.5 and 47, respectively.

2.3 Daily Intake Rates of Uranium Using TWOPOS/Coworker Simulation on Claimant Gross Beta Bioassay

In addition to the individual claimant intake assessment presented in Section 2.2, SC&A also analyzed all gross beta urinalysis data among the group of 32 claimants originally identified in Appendix A of SC&A 2017. The gross beta urine data were analyzed to calculate the TWOPOS for each year in a simulation of how coworker data would be treated per the instructions in ORAUT-RPRT-0053 (NIOSH 2014). Table 3 displays the results of the TWOPOS calculation. Due to the relatively low number of TWOPOS results by year, SC&A did not attempt to fit the data to a distribution and simply used the arithmetic average TWOPOS value for intake analysis. For the IMBA calculation, each annual TWOPOS value was assumed to occur on July 1 of the year of analysis, and the inhalation intake period was assumed to extend from January 1, 1953, through December 31, 1960. Based on these assumptions, the daily intake rate of Type F Sr-90 was calculated to be 6.5×10^3 dpm/d with a corresponding uranium intake of 0.164 pCi/d.

Table 3. Calculated Gross Beta TWOPOS Results by Year

Year	Assumed Date of Sample	Number of TWOPOS Results	Minimum TWOPOS Value (dpm/d)	Maximum TWOPOS Value (dpm/d)	Average TWOPOS Value Used in Intake Calculation (dpm/d)
1953	7/1/1953	10	2,970	13,109	6,615
1954	7/1/1954	12	2,007	14,178	6,021
1955	7/1/1955	14	0	5,224	2,125
1956	7/1/1956	15	0	6,290	2,526
1957	7/1/1957	14	0	7,197	3,218
1958	7/1/1958	13	779	3,596	2,226
1959	7/1/1959	12	221	3,213	1,048
1960	7/1/1960	10	0	3,360	1,124

3 ALTERNATE ESTIMATES FOR POTENTIAL URANIUM INTAKE

The following two subsections demonstrate alternate methods that could be used to determine uranium intakes that do not utilize the ratio method described in Section 5.5.2 of the INL internal TBD (NIOSH 2010a). These two methods are:

1. Use of alpha air sampling data found in the Product Bottle Room and laboratory Room 216

2. Calculation of uranium intakes based on the assumed uranium MDA (also known as a "missed dose" calculation)

The intake rates described below can be compared to the intake rates derived using the ratio method described in Sections 2.2 and 2.3 above.

3.1 URANIUM INTAKES BASED ON ALPHA AIR SAMPLING

Section 3.0 and Section 5.0 of SC&A 2017 noted several instances where long-lived alpha air sampling was noted. Figure 5 of SC&A 2017 presents an air sample in the Product Bottle Room of 5×10^{-11} microcuries per cubic centimeter ($\mu Ci/cm^3$) alpha. However, as noted in NIOSH 2018a, that sample was recounted to correct for radon/thoron progeny, so the actual long-lived alpha activity was likely closer to 3×10^{-11} $\mu Ci/cm^3$. NIOSH 2018a also notes that this corresponds to the "general permissible concentration" for alpha emitters based on 1952 radiological guidelines. Assuming a standard breathing rate of 1.2 m^3/hr and a typical 8-hour work day, the daily intake of uranium can be estimated by the following formula:

$$3\times10^{-11}\ \mu Ci/cm^3 \times 100^3\ cm^3/m^3 \times 1.2\ m^3/hr \times 8\ hr/d \times 10^6\ pCi/\mu Ci = 288\ pCi/d$$

Per the information in Table 7-7 of the SEC-00219 evaluation report (NIOSH 2017), the permissible airborne conditions were lowered to 1.7×10^{-11} $\mu Ci/cm^3$ based on a 1954 CPP operating manual (NIOSH 2017). This air concentration limit would result in a daily intake rate of 163.2 pCi/d. Section 5.0 of SC&A 2017 noted several long-lived alpha activity air samples from Room 216 that ranged from 1.2×10^{-13} $\mu Ci/cm^3$ to 9.6×10^{-12} $\mu Ci/cm^3$. The upper end of these samples would correspond to a daily intake rate of approximately 92 pCi/d.

3.2 URANIUM INTAKES BASED ON THE MISSED DOSE FROM BIOASSAY

SC&A used IMBA to calculate a hypothetical missed dose for a worker who was chronically exposed from the assumed start of the assumed period of interest (February 1, 1953) to the end of the assumed period of interest (December 31, 1962). In accordance with ORAUT-OTIB-0060, Revision 02, *Internal Dose Reconstruction* (NIOSH 2018b), the intake rate was evaluated assuming that a bioassay sample was submitted on the final day of the exposure and was assumed to be ½ of the MDA. The assumed MDA for uranium was taken from Table 5-14 of the INL internal TBD and is equal to 1×10^{-5} grams of uranium per liter (or 14 micrograms of uranium per day). Assuming natural uranium[3] with a specific activity of 6.8×10^2 (pCi/mg)

[3] Although natural uranium was analyzed for this report, assuming low or highly enriched uranium would only increase the intake estimates based on the uranium urinalysis, which is on a per mass basis.

results in an activity-based urinary excretion rate of 9.56 pCi/d at the MDA (4.78 pCi/d at ½ the MDA).

Using this MDA for uranium, SC&A performed a basic missed dose calculation in which a worker was chronically exposed via inhalation to natural uranium for the period from February 1, 1953, through December 31, 1962. A single bioassay measurement at ½ the MDA was assumed to have occurred on the last day of exposure (December 31, 1962). Based on these assumptions, IMBA calculated the following uranium intake rates for each relevant solubility type:

- Type F solubility natural uranium intake rate: 17.3 pCi/d
- Type M solubility natural uranium intake rate: 70.5 pCi/d
- Type S solubility natural uranium intake rate: 869 pCi/d

4 SUMMARY AND CONCLUSION

Table 4 displays the calculated daily uranium intake rates using the methods described in Sections 2 and 3 of this report. As seen in the first 3 row entries, the current method for deriving uranium intakes, which involves using a ratio to gross beta intakes (assumed to be Sr-90), results in a daily intake rate ranging from 0.16 pCi/d to 1.48 pCi/d. The other two methods for determining uranium intake, which are based on either air samples or the available uranium bioassay limits, are at least an order of magnitude higher. Notably, the daily intake rates based on the missed dose from uranium urinalysis and assuming Type S natural uranium was over four orders of magnitude higher than the uranium intakes derived when simulating a coworker approach in conjunction with the ratio method.

Table 4. Summary of Daily Uranium Intake Rates (pCi/d) using the Gross Beta Ratio Method, Alpha Air Sampling, and Uranium Urinalysis Missed Intake Calculations

Uranium Intake Method	Description	Daily Intake Rate of Uranium (pCi/d)
Ratio to Sr-90	IMBA-calculated bioassay fit using coworker simulation approach by calculating TWOPOS annual excretion rates (see Section 2.3)	0.16
Ratio to Sr-90	Average of individual claimant assessments using IMBA and only considering non-zero gross beta urinalysis results (see Section 2.2)	0.57
Ratio to Sr-90	Average of individual claimant assessments using IMBA and assuming ½ of the MDA with the inclusion of any positive values above ½ of the MDA (see Section 2.2)	1.48
Missed Dose from Uranium Urinalysis	Solubility Type F (see Section 3.2)	17.3
Missed Dose from Uranium Urinalysis	Solubility Type M (see Section 3.2)	70.5
Alpha Air Sampling	Maximum Room 216 air sample from SC&A 2017: 9.6×10^{-12} $\mu Ci/cm^3$ (see Section 3.1)	92
Alpha Air Sampling	Permissible airborne levels of natural uranium (1.7×10^{-11} $\mu Ci/cm^3$) from a 1954 CPP operating manual (NIOSH 2017) (see Section 3.1)	163
Alpha Air Sampling	Product Bottle Room air sample from SC&A 2017 assumed to have decayed to the 1952 general permissible concentration of 3×10^{-11} $\mu Ci/cm^3$ (see Section 3.1)	288
Missed Dose from Uranium Urinalysis	Solubility Type S (see Section 3.2)	869

Summary Recommendation: Evidence suggests that the current dose reconstruction method for reconstructing uranium intakes based on a ratio of uranium intakes to intakes of FAP may not be a sufficiently bounding approach for workers who may have been exposed to uranium that was not comingled with FAP material. It would be beneficial for NIOSH to explore alternate methods of dose reconstruction for workers who were not directly monitored for uranium.

5 REFERENCES

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ATTACHMENT A: [REDACTED IN FULL]

[Attachment A is withheld in its entirety to prevent the disclosure of Privacy Act-protected information. The unredacted attachment originally occupied pages 17–32 of this report.]

ATTACHMENT B: [REDACTED IN FULL]

[Attachment B is withheld in its entirety to prevent the disclosure of Privacy Act-protected information. The unredacted attachment originally occupied page 33 of this report.]