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A REVIEW OF ORAUT-OTIB-0080, REV. 00: INTERNAL COWORKER DOSIMETRY DATA FOR AREA IV OF THE SANTA SUSANA FIELD LABORATORY AND THE DeSOTO AVENUE FACILITY

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Disclaimer

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ABBREVIATIONS AND ACRONYMS

Advisory Board or ABRWH	Advisory Board on Radiation and Worker Health
AETR	Advanced Epithermal Thorium Reactor
AMAD	Activity Mean Aerodynamic Diameter
BZ	breathing zone
CEDE	Committed Effective Dose Equivalent
CEP	Controls for Environmental Pollution, Inc.
Ci	curie
cm^2	square centimeter
CMD	count median diameter
cpm	count per minute
d	day
DOE	(U.S.) Department of Energy
dpm	disintegrations per minute
ETEC	Energy Technology Engineering Center
EU	enriched uranium
FP	fission product bioassay analysis method
g/cm^3	gram per cubic centimeter
GB	gross beta bioassay analysis method
GIF	Gamma Irradiation Facility
gm	grams
GM	geometric mean
GSD	geometric standard deviation
HEPA	high efficiency particulate air
ICRP	International Commission on Radiological Protection
IMBA	Integrated Modules for Bioassay Analysis
kg	kilogram
L or l	liter
LETF	Laser Experimental Test Facility
LMEC	Liquid Metal Engineering Center
MDA	minimum detectable amount

MFP	mixed fission product bioassay analysis method
MFPB or MFP(B)	mixed fission product for beta counting bioassay analysis method
MFPG	mixed fission product for gamma counting bioassay analysis method
MPC	maximum permissible concentration
µCi/cc	microcurie per cubic centimeter
µg	micrograms
µm	micrometer
mL	milliliter
MPM	Maximum Possible Mean
mrem	millirem
nCi	nanocuries
NIOSH	National Institute for Occupational Safety and Health
NMDF	Nuclear Materials Development Facility
NOCTS	NIOSH/OCAS Claims Tracking System
NRC	(U.S.) Nuclear Regulatory Commission
OCAS	Office of Compensation Analysis and Support
OPOS	One Person – One Sample
ORAUT	Oak Ridge Associated Universities Team
ORNL	Oak Ridge National Laboratory
PuO_2	plutonium oxide
RMDF	Radioactive Materials Disposal Facility
SC&A	S. Cohen and Associates (SC&A, Inc.)
SEC	Special Exposure Cohort
SRDB	Site Research Database
SRE	Sodium Reactor Experiment
SSFL	Santa Susana Field Laboratory
TBD	Technical Basis Document
TRUMP-S	Transuranic Management by Propartitioning-Separation (TRUMP-S)
U-Al_x	uranium aluminide
U_3O_8	triuranium octoxide
UO_2	uranium dioxide
UR	uranium bioassay analysis method
UX	unknown unit

EXECUTIVE SUMMARY

The Santa Susana Field Laboratory (SSFL) is located approximately 30 miles northwest of downtown Los Angeles, California. SSFL consists of 2,850 acres in the Simi Hills of Ventura County. SSFL is divided into four administrative and operational portions based on ownership and operations. Department of Energy (DOE) operations are conducted in Rockwell International-owned and DOE-owned facilities in an area designated as Area IV. SSFL was initially established by North American Aviation in 1947 as a field test laboratory to static-fire rocket engines. In 1953, Area IV was established as a nuclear research and development facility with it being managed by Atomic International starting in 1955. Atomic International conducted activities in Area IV on development of civilian nuclear power, and the Liquid Metal Engineering Center (LMEC) that focused on research and testing of non-nuclear components to liquid metals. LMEC was renamed the Energy Technology Engineering Center (ETEC) in 1978. Through various mergers, the SSFL is now managed by Boeing.

Between 1954 and 1980, several small nuclear reactors and critical test assemblies were built, tested, and operated in Area IV. These research facilities focused on development and operation of homogeneous water boiler-type reactors, sodium-cooled graphite-moderated reactors, and uranium-zirconium hydride reactors. Starting in 1956 and continuing through 1996, operations also supported the manufacture, management, and disassembly of nuclear reactor fuel, as well as the operation of nuclear waste management facilities. Fuel manufacturing included the assembly of fuel elements for the Sodium Reactor Experiment (SRE), a plutonium fuel manufacturing facility, and a uranium carbide fuel pilot plant. There was also a Fuel Storage Facility for storage of special nuclear materials used to make the fuels. Other research programs addressed reprocessing of used reactor fuel, operation of small particle accelerators, and use of radioisotopes.

Three other facility locations are associated with SSFL operations: Downey, Canoga Park, and DeSoto. Downey and Canoga Park were operational prior to 1960 and are not considered in this report.[1] At DeSoto, radiological operations occurred from 1959 into the mid-1990s. These operations included a Nuclear Regulatory Commission (NRC)-licensed research reactor, fuel fabrication for the Advanced Test Reactor, the Gamma Irradiation Facility that used sealed Cs-137 and Co-60 sources, and a mass spectroscopy laboratory. Some workers worked interchangeably between DeSoto and SSFL.

The technical information bulletin ORAUT-OTIB-0080 titled, *Internal Coworker Dosimetry Data for Area IV of the Santa Susana Field Laboratory and the De Soto Avenue Facility* (ORAUT 2014a), presents an overview of the available internal monitoring data and derived coworker intake rates to be used at SSFL. SC&A was tasked by the Advisory Board on Radiation and Worker Health to review this document and associated coworker data for the feasibility of reconstructing unmonitored internal exposures. This report documents the results of that review.

[1] Special Exposure Cohort (SEC) Petitions 00093 and 00156 were designated in 2009 and 2010 (respectively) for the Santa Susana operational period extending from January 1, 1955, through December 31, 1964, based on the inability to reconstruct internal doses.

Section 1 of this report provides an overview of the available dataset provided to SC&A, as well as the general characteristics of the applicable internal monitoring data, while Section 2 summarizes how these data are distributed among occupations. Sections 3, 4, and 5 individually focus on the three contaminants of interest analyzed in ORAUT-OTIB-0080: "plutonium," "fission products," and "uranium," respectively. Section 6 provides an evaluation of the need to model other radionuclides that would not have been monitored using analytical methods for uranium, plutonium, or mixed fission products. SC&A's review focuses on the completeness and adequacy of the available data for the purposes of assigning coworker intake rates to unmonitored workers. In general, the "completeness" of a dataset refers to the extent to which the available data provide sufficiently representative and/or bounding temporal, work area, and job type coverage. "Adequacy" of the coworker dataset refers to the extent to which the numerical values used in intake calculations accurately reflect and/or bound the internal exposure potential experienced by the unmonitored worker population.

Based on this review, SC&A has produced 15 findings as listed below:

Finding 1: It is evident that the "reported" values with the "<" designation represent the volume-adjusted minimum detectable activity (MDA), and that the "calculated" values are much less than the censorship level. In these instances, NIOSH should utilize the "reported" value as a censored result, which is claimant favorable and in accordance with the Maximum Possible Mean (MPM) methodology in ORAUT-RPRT-0053 (ORAUT 2012).

Finding 2: SC&A was unable to reconcile apparent differences between the effective bioassay dates listed in Table A-1 of ORAUT-OTIB-0080 and the known start and end dates of "statistically usable" plutonium data.

Finding 3: Combining multiple years for the purpose of calculating the 50^{th} and 84^{th} percentile One Person – One Sample (OPOS) results must be scientifically justified in the context of the potential for historical changes in operations and associated intakes for monitored workers.

Finding 4: More definitive evidence is warranted to establish that exposure potential to plutonium was lower in 1965–1966 than it was in 1967–1968 to validate the back extrapolation of plutonium results to the earlier period.

Finding 5: SC&A was not able to duplicate the plutonium OPOS values presented in Table A-1 to within a reasonable level of precision. The source of the discrepancies should be identified and resolved prior to the calculation of coworker intakes.

Finding 6: SC&A's research into the historical use of plutonium at SSFL indicates that programs or campaigns in plutonium facilities did not extend for more than 1 or 2 years in most cases [at least in the case of the primary plutonium facility, Nuclear Materials Development Facility (NMDF)] and reinforces the need for Findings 3 and 4 to be addressed. Therefore, averaging plutonium data across multiple years for the purposes of calculating coworker intakes may not be appropriate.

Finding 7: Based on the number of monitored workers tabulated in Tables 13 and 14, it is likely that NIOSH did not adhere to the inclusion guidelines outlined in ORAUT-OTIB-0080. It appears that all samples labelled "FP" or "MFP" in the "Nuclide/Analysis" column were included regardless of the actual designated method of measurement.

Finding 8: SC&A was not able to duplicate the "FP" OPOS values presented in Table A-3 of ORAUT-OTIB-0080 to within a reasonable level of precision for most years. The source of the discrepancies should be identified and resolved prior to the calculation of coworker intakes.

Finding 9: The coworker model would benefit from a substantive discussion providing the rationale for combining all 27 years into a single intake regime. This might include a discussion on the variability in the calculated measurements observed by year, changes in the geometric standard deviation (GSD) indicating a larger spread of potential exposures, the actual effect on the calculated intakes in relation to dose, changes in site operations, etc. Such a discussion would reinforce the assumption that a single chronic intake regime is appropriate over a relatively long period of time.

Finding 10: SC&A's attempt to duplicate the uranium OPOS values presented in Table A-2 of ORAUT-OTIB-0080 was off by an order of magnitude in 4 of the 25 years analyzed by NIOSH (1984 and 1985 were not analyzed). The source of the discrepancies should be identified and resolved prior to the final calculation of coworker intakes.

Finding 11: It is important to understand the reasons for the drastic decrease in observed sampling for uranium from 1984 to 1985. Concurrently, because the paucity of the data makes it impossible to make quantitative statements about the available bioassay data from 1984 to1985, it should be established that conditions and exposure potential are sufficiently similar to (or bounded by) surrounding years to validate the use as surrogate data.

Finding 12: Uranium intake rates appear to be underestimated, based on the practice of collecting urinalysis samples on Monday morning after 2 days of no exposure. Although not all the available bioassay data were collected on Monday morning, NIOSH should apply an adjustment factor to the calculated intakes as a claimant-favorable and bounding assumption.

Finding 13: NIOSH should discuss how uranium samples taken from workers in the uranium aluminide ($U\text{-}Al_x$) powder operations are interpreted in the coworker analysis, and whether a separate assessment of coworker doses in the Powder Building may be warranted. In the latter case, it would be necessary to establish with sufficient accuracy which workers were potentially exposed to $U\text{-}Al_x$ in the Powder Building during the relevant period.

Finding 14: The internal coworker model should specify the types of incidents that are assumed to be covered by the model, and when incidents would be expected to be reviewed separately with an incident-specific model.

Finding 15: The internal dose coworker model is incomplete in that it does not address other radionuclides that were present on site and that could have substantively contributed to worker intakes. NIOSH should ascertain their ability to reconstruct intakes of thorium, tritium, americium, cesium, strontium, polonium, and cobalt.

In addition, SC&A identified the following 10 pertinent observations:

Observation 1: The assumption that database entries that do not contain a specific sample volume and are in units of total activity are reflective of a daily excretion rate has been validated by examination of handwritten logbooks contained in individual claimant bioassay records.

Observation 2: SC&A noted that when the "reported" value was not designated as "<", the difference between the reported and calculated values was simply that the "reported" result was normalized to 1,500 mL. Since NIOSH is already making the volume correction for the "calculated" result, this has a negligible effect on the coworker calculations.

Observation 3: SC&A noted that the value reported in the "Remarks" box was not always transferred to the ORAUT-created electronic database. NIOSH needs to ensure that all reported doses in the hardcopy of the dose records are appropriately transferred to the electronic database.

Observation 4: To better understand what the overall monitoring data represent, research should be performed to determine why no monitoring data are available for certain occupations in the late 1980s and early 1990s.

Observation 5: Page 9 of ORAUT-OTIB-0080 states, "Sufficient plutonium bioassay data were available to perform a statistical analysis for 1965 through 1986." Page 11 of ORAUT-OTIB-0080 states, "Insufficient [plutonium] bioassay data were available for 1965 and 1966 to perform a statistical analysis… The 1967–1968 data was used to back-extrapolate for 1965 and 1966, when the potential plutonium exposure was less." It appears based on the analysis and discussion performed that the latter statement is correct and that the former statement should be revised.

Observation 6: The applicable plutonium data contained two results with units of weight (2.69 gm and 0 µg), both marked as "urine" results. Both results were set to zero dpm/1.4L. In the case of the former result, it is unclear why the sample was set to zero; however, the effect of this adjustment is likely negligible. Regardless, ORAUT-OTIB-0080 specifies on page 8 that samples given in units of mass are assumed to be fecal samples and are not to be used. However, these two samples were marked as "Y" in the "Use" column of the database.

Observation 7: Bioassay data in available literature pertaining to the monitoring of plutonium intakes support the assumption made in the internal dose model that the intake rate in the late 1980s and 1990 is representative (or even bounding) of plutonium intakes that could have occurred in the 1990s.

Observation 8: While there are temporal gaps in the job title coverage for mechanics and technical staff, less than ¼ of the monitored workers could be identified with a particular occupation. Of the monitored workers for which job title information exists, mechanics and technical staff constitute 65% of the population. Therefore, it is reasonable to assume that these higher risk job types were not systematically excluded and were likely targeted for plutonium monitoring.

Observation 9: Note that the last effective bioassay date in Table 15 is March 1, 1991. Examination of the available data indicates that "FP" bioassay data effectively end in mid-April. This comports with an effective bioassay date for the evaluation period of January 1, 1991, to roughly April 30, 1991. Since the end of April 1991 marks the end of the "FP" bioassay evaluation, the calculated intakes found in ORAUT-OTIB-0080 Table 5-6 should also end in April 1991 and not December 1991. However, it is not clear if changing the date of the end of the chronic intake calculation would significantly affect the calculated intakes.

Observation 10: It is important that assumptions that cross-cut among models (e.g., internal, external, ambient) are consistent. Unless there is justification to do otherwise, the length of time of exposure of coworkers should be assumed to be the same across all models. Because use of overtime is often a common practice among operations personnel, a 2,000 work-year may not bound the dose to a worker.

1.0 CHARACTERIZATION OF AVAILABLE COWORKER DATASET

SC&A requested and was provided with the underlying dataset containing the internal monitoring data utilized in ORAUT-OTIB-0080 to calculate coworker intake rates.[2] The database contains 37,323 total entries covering various types of internal monitoring, including urinalysis, whole body counts, wound monitoring, fecal results, and smear samples. The available internal monitoring results also cover a myriad of different contaminants and radiation types. In the database, 118 distinct "Nuclide/analysis" types were identified, although several of these entries were redundant, such as "GB," "Beta," Gross Beta." For a full listing of internal monitoring entries and the number of samples associated with each, refer to Attachment 1.

Many of the database entries were not germane towards calculating coworker intake rates for uranium, plutonium, and mixed fission products, and therefore had to be removed. The specific steps that SC&A utilized to remove unwanted samples are documented in Table 1 and comport with the analysis steps discussed in ORAUT-OTIB-0080 and Arno 2013. As seen in Table 1, after the steps to remove unwanted internal monitoring entries were performed, there were 902 plutonium results, 3,378 mixed fission product results, and 8,669 uranium results available for coworker analysis. Each of these sub datasets is evaluated and discussed in further detail in Sections 3-5, respectively.

Table 1. Description and Rationale for the Removal or Extraction of Entries from the Database for Coworker Analysis

Total Entries Remaining for Analysis	Rationale for Items Removed/Extracted
37,322	No entries removed; this represents the total number entries in the original database.
26,515	Remove all entries marked as "N" in the "Use" column (Column AU) indicating the sample should not be used in coworker analysis. Remove samples prior to 1/1/1965 or after 8/4/1991. The end date was chosen to mirror coworker modeling dates and exclude samples analyzed by the laboratory "Controls for Environmental Pollution, Inc." (CEP) which has been determined to be an unreliable bioassay vendor.
25,546	Remove entries with a blank field in the "Nuclide/Analysis" column (column T). Remove entries with a blank result in both the "Reported result" and "Calculated result" columns (columns AC and AD, respectively) in accordance with Arno 2013.
902	*Extraction of Plutonium Coworker Records*: Extract all records with "Nuclide/Analysis Type" PU, Pu and PUA in accordance with ORAUT-OTIB-0080, pg. 9 and Arno 2013, pg. 2. Remove two additional records with in vitro units of "gm" (grams) and "µg" (micrograms).
3,378	*Extraction of Fission Product Coworker Records:* Extract all records with "Nuclide/Analysis Type: MFP" with a corresponding "Method Type: B."[3] Additionally, extract all records with "Nuclide/Analysis Type: FP" which contains a corresponding "Method Type: 3A."[4] The selected bioassay data are in accordance with ORAUT-OTIB-0080, pg. 10.
8,682	*Extraction of Uranium Coworker Records*: Extract all records with "Nuclide/Analysis Type: UR" in accordance with ORAUT-OTIB-0080, pg. 9 and Arno 2013, pg. 2. Remove two additional records with in vitro unit types "µg" (micrograms) and "UR" (unknown unit).

[2] This dataset can be found at [K:\ABRWH\AB Document Review\Santa Susana\SSFL Internal coworker data for OTIB-0080 Rev 00\SSFL_FINAL compilation 0301113_r2 internal only R3.xlsx].

[3] This includes "Nuclide/Analysis" entries "MFPB" and "MFP(B)." Additionally MFP with method types: "1B", "2B", "3B","5B", "7B"

[4] This includes method types: "3A T-U" and "3AG"

Each datapoint available for coworker analysis was normalized to the units of dpm/1.4L. SC&A noted that there was a large number of activity units used in the coworker data. Table 2 presents the distribution of observed units for each of the coworker radionuclides of interest. As seen in the table, the majority of samples for each monitored radionuclide were given in the units of "dpm." In a small percentage of cases where the unit "dpm" was observed, there was a corresponding sample volume, which can be used to normalize the result to dpm/1.4L. For example, only 22 of 666 plutonium samples, 11 of 2,611 mixed fission product samples, and 810 of the 4,986 uranium samples given in "dpm" contained a corresponding sample volume. The remaining samples given in units of dpm did not state the volume of the sample.

Table 2. Overview of Bioassay Sample Units Available for Coworker Analysis

Observed Units	# Plutonium Samples (% of Total)	# Fission Product Samples (% of Total)	Uranium
dpm	666 (73.84%)	2611 (77.29%)	4986 (57.52%)
dpm/sample	62 (6.87%)	266 (7.87%)	2925 (33.74%)
dpm/L	10 (1.11%)	Not Used	Not Used
dpm/d	133 (14.75%)	221 (6.54%)	41 (0.47%)
dpm/vol	2 (0.22%)	14 (0.41%)	4 (0.05%)
Blank	29 (3.22%)	266 (7.87%)	712 (8.21%)
cpm	Not Used	Not Used	1 (0.01%)

Also as shown in Table 2, other common units observed in the database were actually provided in dpm per unit volume (such as dpm/d or dpm/L). With the exception of the samples in which a separate sample volume was given or the units were already provided in "dpm/d" and "dpm/L," a sample volume would seemingly be required to correctly normalize the numerical result to dpm/1.4L. SC&A examined each coworker dataset to determine how many results provided enough information to effectively normalize the sample. The results are shown in Table 3.[5]

Table 3. Percentage of Entries for Each Contaminant with Known and Unknown Sample Volumes

Contaminant Type	Total Entries Available for Coworker Analysis	Percentage of Samples with a Known Volume*	Percentage of Samples with an Unknown Volume*
Plutonium	902	21.51%	78.49%
Fission Products	3,378	15.04%	84.96%
Uranium	8,669	39.75%	60.25%

*Samples listed as dpm/d or dpm/L were assumed to be 1.5 liters and 1 liter respectively

As can be seen in Table 3, the majority of samples in each contaminant category did not contain the volume of the sample analyzed. In the case of fission products, nearly 85% of the samples available for coworker analysis did not have an associated sample volume. The sample volume of a particular aliquot given only in units of total activity can have a significant effect on the normalized end result; therefore, it is essential that the volume of these results be understood.

[5] dpm/d and dpm/L were assumed to represent sample volumes of 1.5 and 1 liter, respectively.

ORAUT-OTIB-0080 states the following on page 9 concerning unknown sample volumes:

> *Some of the records have individual fields that are blank or illegible but contain sufficient information to proceed with the analysis. The following rules were used to adjust for blank or unusable fields: … Volumes were assumed to be 1,500 ml.*

Additionally, Arno 2013 provides similar instructions:

- Uranium: *"If no volume is provided, assume 1500 mL."* (pg. 2)
- Plutonium: *"If no volume is provided, with units of dpm or dpm/sample, assume results are dpm/day."* (pg. 2)
- Fission Products: *"If no volume is provided, assume 1500 mL."* (pg. 3)

The technical justification for the statements in ORAUT-OTIB-0080 was annotated as follows:

> *Arno, Matthew. ORAU Team. Dose Reconstructor. January 21, 2011. These rules were developed based on examination of the data for trends in how the data was recorded, review of the internal dose technical basis document for SSFL (ORAUT 2010) to determine default, and/or assumed analytical techniques and reporting conventions.*

The Technical Basis Document (TBD) for Occupational Internal Dose, ORAUT-TKBS-0038-5 (ORAUT 2010) contains the following statements about the bioassay program in the 1967–1974 period:

- *The detection limit* [for Uranium Radiometric (UR)] *was 0.5 dpm/sample. At SSFL the standard sample volume per day was 1,500 mL. The result was the total activity."* (pg. 19)
- *The detection limit* [for plutonium] *was 0.05 dpm/sample. At SSFL the standard sample volume per day was 1,500 mL.* (pg. 19)

No direct references appear to be provided in ORAUT 2010 for the above two statements and no mention of standard sample volumes was provided for mixed fission products. Information on the standard sample volume was not stated before or after the 1967–1974 period in the TBD. SC&A examined the data available for coworker analysis to assess the extent to which the data itself lend credence to the assumption that the standard samples were 1.5 liters. The percentage of records that list a corresponding sample volume are shown by year in Figure 1.

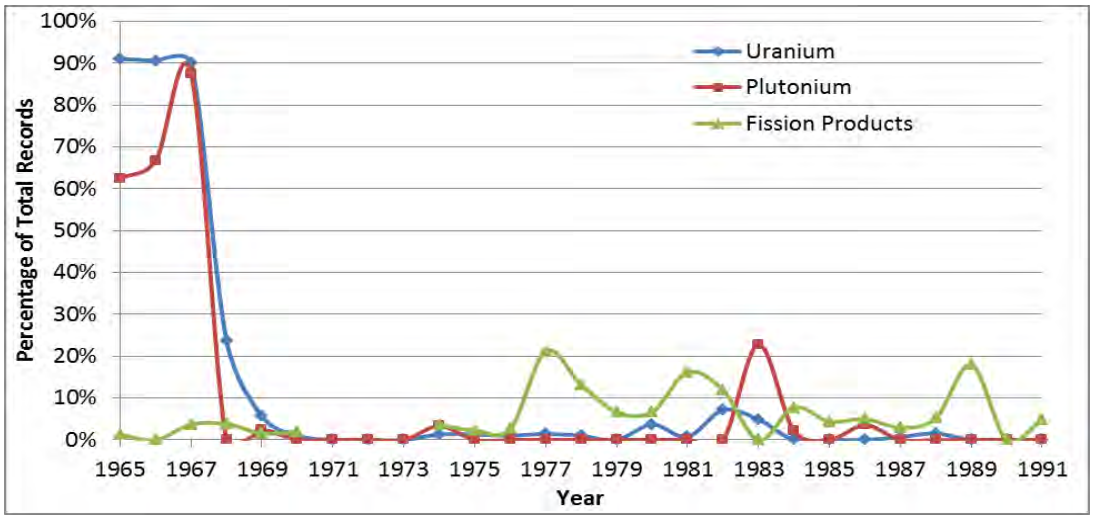


Figure 1. Percentage of Records Listing a Sample Volume for Uranium, Plutonium and Fission Products

As seen in Figure 1, it is clear that the majority of results for uranium and plutonium contained a listed sample volume in the 1965–1967 timeframe. After this time, generally 90% of the available samples for uranium and plutonium did not contain a sample volume. One possible conclusion for plutonium and uranium is that after 1967, the actual sample volume was not recorded, but was taken into account in the reported result. Another possibility is that the actual sample volumes were simply not entered into the available database. There does not appear to be a discernible trend for fission product bioassay results.

In order to verify the assumption that samples given in units of total activity were, in fact, representative of daily excretion rates, SC&A examined the individual claimant monitoring records contained in the NIOSH/OCAS Claims Tracking System (NOCTS) database for additional information to affirm or refute the assumed sample volumes. SC&A found several cases where handwritten logbooks were included in the DOE-supplied monitoring records that indicated the sample was in the units of "dpm/day" (nominally dpm/1,500 ml). These same numerical values could be matched up to the electronic record, which either contained no units or contained units of total activity (dpm). One example of this can be seen in Figures 2 and 3, which show the handwritten log and associated database entry.

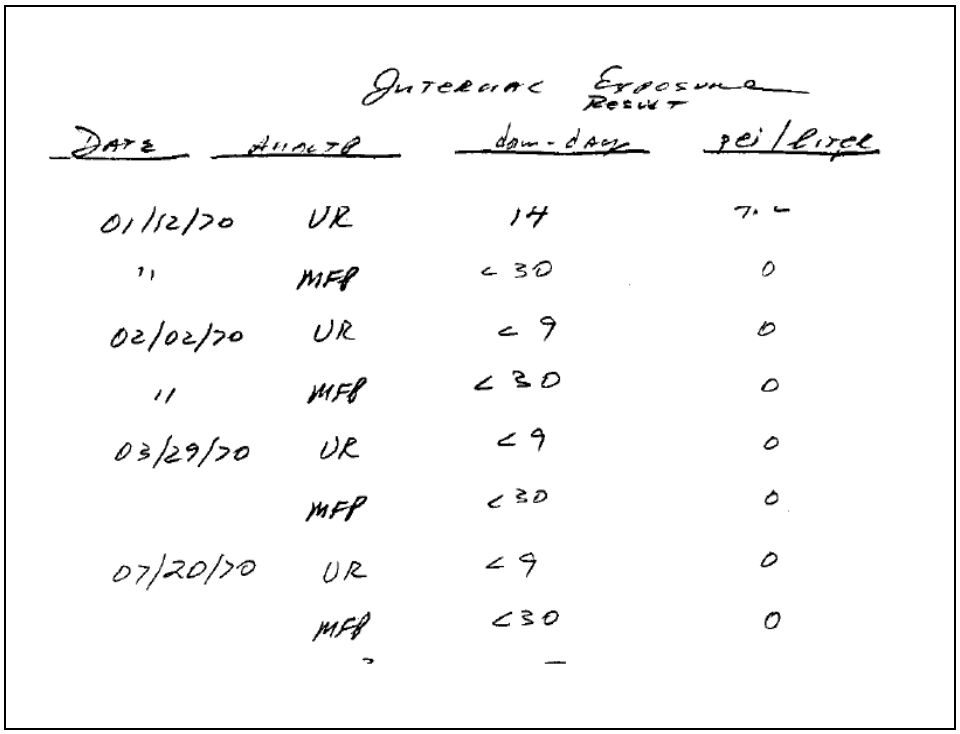


Figure 2. Example Logbook Contained in Claimant Hardcopy File Showing Units of "dpm/day"

	U	Y	Z	AA	AB	AC	AD	AG
1	Nuclide/ Analysis	Analyzed Volume	volume units	Sample Date (Collection)	Date Received	<	Reported result (in box)	in-vitro units
25563	MFP			1/12/1970		<	30	dpm
25564	UR			1/12/1970			14	dpm
25565	MFP			2/2/1970		<	30	dpm
25566	UR			2/2/1970		<	9	dpm
25567	MFP			3/29/1970		<	30	dpm
25568	UR			3/29/1970		<	9	dpm
25569	MFP			7/20/1970		<	30	dpm
25570	UR			7/20/1970		<	9	dpm

Figure 3. Corresponding Entry in Coworker Database: Entries Showing Units of "dpm" with a Blank Field for "Analyzed Volume"

Observation 1: The assumption that database entries that do not contain a specific sample volume and are in units of total activity are reflective of a daily excretion rate has been validated by examination of handwritten logbooks contained in individual claimant bioassay records.

Finally, as seen in Figure 4, there are two sources of urinalysis activity values present in the SSFL records (see columns AC and AD in the figure). The column AC header is titled "Reported Result (in box)," while the column AD header is titled "Calculated Result (above box)." In some cases there is a numerical value for each entry, but in most cases a numerical value is present in only one of the columns.

T	U	W	X	Y	Z	AA	AB	AC	AD	AE	AF
Nuclide/ Analysis	Method	Type	Analyzed Volume	volume units	Sample Date (Collection)	Date Received	<	Reported result (in box)	Calculated result (above box)	error	in-vitro units
UF		urine	1	mL		11/3/1966	<	2.00E-04	1.60E-04		ug/sample
UR		urine	95	mL		11/3/1966	<	0.57	0.24		dpm/sample
UR		urine	95	mL		12/14/1966		3.43		0.87	dpm/sample
UF		urine	1	mL		12/15/1966		2.50E-04		4.50E-05	ug/sample
UF		urine	1	mL		1/13/1967	<	2.00E-04	8.00E-05		ug/sample
UR		urine	95	mL		1/13/1967		3.5		0.95	dpm/sample
UR		urine	100	mL		2/8/1967			6.33	1.27	dpm/sample
UF	1A	urine			2/9/1967			0			
UR		urine	1500	mL	2/9/1967			55			dpm

Figure 4. Example Database Entry Showing "Reported" and "Calculated" Urinalysis Activity Results

Arno 2013 instructs the modeler to always use the "calculated result" when present and only to use the "reported result" when no calculated result is available. Appendix B of ORAUT 2010 provides descriptions of the different bioassay forms encountered at Santa Susana. An illustrative example of a bioassay form containing both a calculated and reported value can be found in Figure 5.

The figure highlights the formula used to reach the calculated value, as well as the reported value and sample volume. Notably, the reported value is quantitatively higher and is labelled as a censored result (i.e., "less than" or "<"). The reported value is likely indicative of the MDA for the given measurement and sample volume.

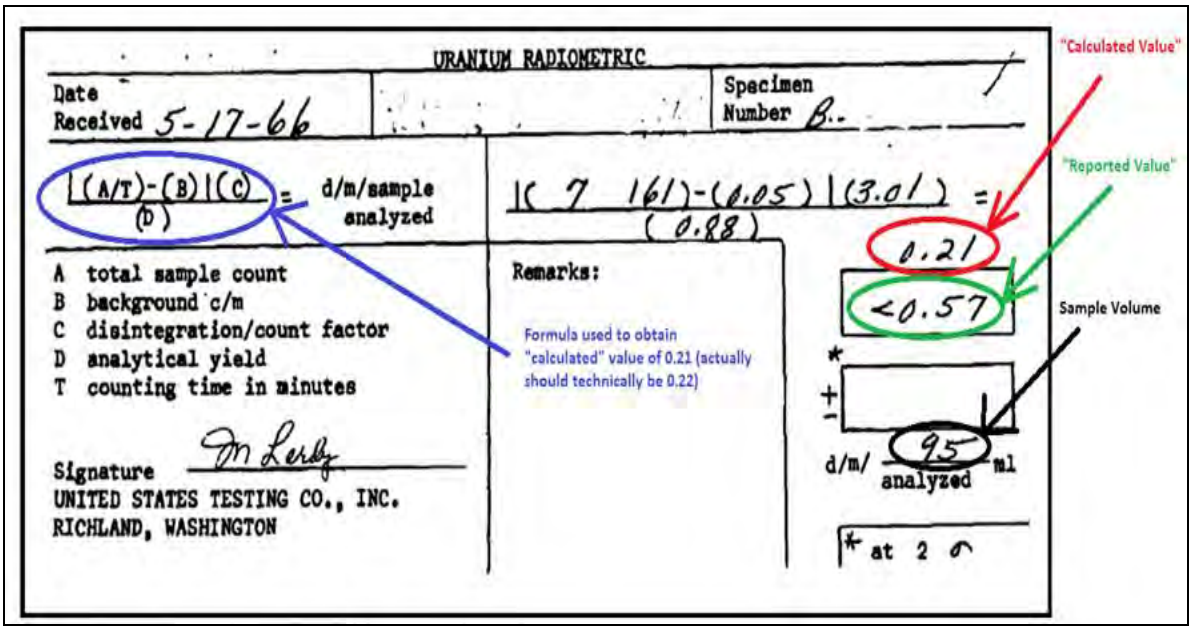


Figure 5. Example Bioassay Form Showing Both a "Calculated" Value and a "Reported" Value

The uranium coworker data contained 1,719 entries that listed a numerical value for both the "reported" and "calculated" entries (roughly 20% of the total coworker samples). SC&A closely examined these paired results for evidence of a numerical relationship and it appears these values are in fact MDA values that have been prorated to the specific volume of the sample. Figure 6 plots the sample volume versus the reported result for samples labelled with "<." While there is some scatter present in the figure, there certainly appears to be a linear relationship between the "reported" value and the sample volume.

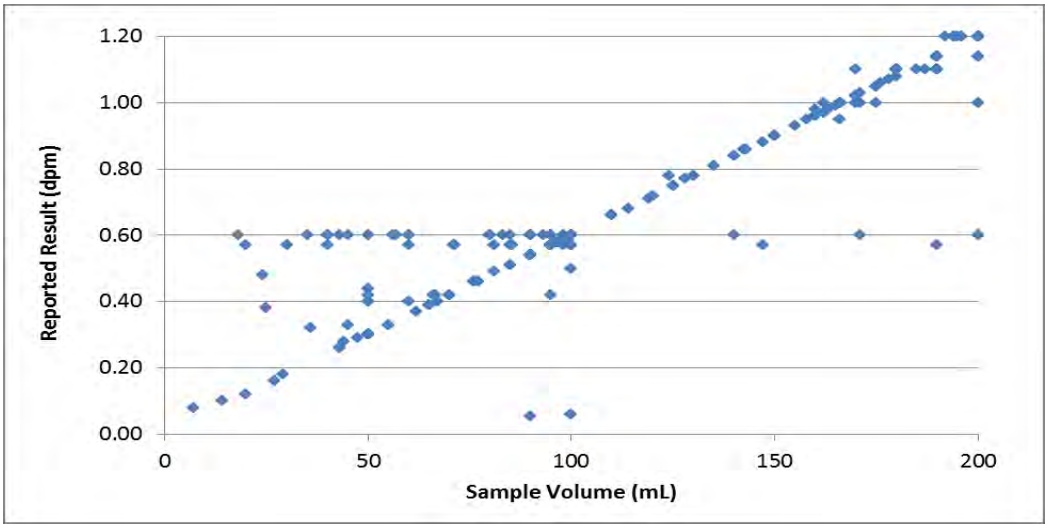


Figure 6. Less than "Reported" Results versus the Associated Sample Volume

Additionally, Table 4 characterizes the "reported" result and "calculated" result quantities for the most commonly observed sample volumes.[6] Column 2 of the table shows that approximately 78% of the paired samples for which the "reported" result was censored fell into one of the three stated sample volumes. Columns 3 and 4 show the most common "reported" value by sample volume as well as the percentage of entries reflecting that value. Columns 5–7 characterize the spread of "calculated" values for that same sample. As an example, of the 135 samples of 90 mL, 96% of the "reported" values were <0.54 dpm. The "calculated" result for these same samples ranged from 0 to 0.52 with an average of 0.14. It is clear from Table 4 that the reported values have been truncated at a censoring level, while the "calculated" values span a wide range of values below this censorship level.

Table 4. Characterization of Paired Results in which the "Reported" Value was Designated as "Less Than"

Sample Volume (mL)	# of Total Entries (% of Total)	Most Common "Reported" Value (dpm)	% of "Reported" Samples Exhibiting Most Common Value for Indicated Sample Volume	Minimum "Calculated" Value	Average "Calculated" Value	Maximum "Calculated" Value
90	135 (8%)	<0.54	96%	0	0.14	0.52
95	642 (39%)	<0.57	99%	0	0.17	0.6*
100	508 (31%)	<0.60	98%	0	0.20	0.8**

*Only 1 of the 642 "calculated" values exceeded the "reported" result of 0.57.
**Only 3 of the 508 "calculated" values exceeded the "reported" result of 0.60.

Finding 1: It is evident that the "reported" values with the "<" designation represent the volume-adjusted MDA and that the "calculated" values are much less than the censorship level. In these instances, NIOSH should utilize the "reported" value as a censored result, which is claimant favorable and in accordance with the MPM methodology in ORAUT-RPRT-0053 (ORAUT 2012).

Observation 2: SC&A noted that when the "reported" value was not designated as "<," the difference between the reported and calculated values was simply that the "reported" result was normalized to 1,500 mL. Since NIOSH is already making the volume correction for the "calculated" result, this has a negligible effect on the coworker calculations.

Observation 3: SC&A noted that the value reported in the "Remarks" box was not always transferred to the ORAUT-created electronic database (for example, see the dose records for claimant [redacted]). NIOSH needs to ensure that all reported doses in the hardcopy of the dose records are appropriately transferred to the electronic database.

[6] There were 80 different sample volumes present in the database for paired results with a "<" reported result. The three most common were 90 ml (~8% of the entries, 95 ml (~39% of the entries), and 100 ml (~31% of the entries).

2.0 OVERVIEW OF MONITORED JOB TYPES CONTAINED IN ORAUT-OTIB-0080 DATABASE

The electronic dose record database prepared by ORAUT was examined to determine what occupations were primarily monitored. The database contained a data field titled 'Job Title.' This data field contained the job title of an individual contained on their hardcopy dose record if it was identified and legible. Of the 37,323 dose records in the database, 7,152 (19%) had job titles identified. To increase the number of records with identifiable job titles, the NOCTS database was compared to the dose records database, and job titles (occupational names) were added to the electronic database based on individuals' social security numbers. If the job title identified in the dose record database was different than the job title identified in NOCTS, then the NOCTS database replaced the dose record database job title information. In addition, for some individuals in NOCTS, their occupation changed during the course of their career. This change in occupational status was reflected in the dose record database such that an individual may be counted more than once when totaling the number of individuals monitored for each occupation. In addition, because many different job titles were used, for the purposes of this analysis, job titles were grouped into more generalized occupational titles. Attachment 3 shows how different job titles were grouped together.

The addition of information from NOCTS increased the number of dose records with an identifiable occupation to 9,149 records. Records prior to 1965 and between January 1991 and mid-June 1993 were removed, per the procedure applied in the internal dose model, which reduced the total number of records to 8,856. Deleting records with the 'Use' designation of 'N' reduced the number of records to 6,550. Removing all records with no 'Nuclide/Analysis' identified and no 'Reported Result' and 'Calculated Result' reduced the number of records to 6,300.

Table 5 shows the results of the occupational analysis. The most prevalent occupations that were monitored were technical staff (30% of the monitored individuals) and mechanics (26% of the monitored individuals). This would be expected, because the mechanics would have performed the majority of the hands-on activities associated with nuclear operations with the support of the engineers and scientists who comprise the technical staff and who would have been responsible for the design, construction, and operation. The cross-section of occupations represented by the dose records also includes occupations not directly related to performance of nuclear operations, but who may have been periodically present. These support personnel represent occupations such as management, office personnel, security, and inspectors, as shown in Table 5. The distribution of workers shown in Table 5 appears to be representative of a monitoring program that focuses on those occupations at highest risk of exposure (technical staff and mechanics), but also reflective of support personnel in the area.

Note that there are seven firemen that were monitored, all of whom had recorded dose in at least one monitoring period. Survivors of workers stated in the February 9, 2010, Advisory Board meeting that they were concerned that firemen would not be compensated, because there was no proof they worked in Area IV. While it is not possible to verify the presence of specific individual firemen in Area IV unless they are shown in the electronic database, this occupational analysis shows that at least some firemen were monitored at both DeSoto and Area IV. The

occupational doses that these monitored firemen received would be included in the internal coworker model, though the question of how well the available data represent or bound intakes received by unmonitored firemen remains open.

SC&A notes that monitoring data for many of the occupations end in 1982–1983. This can be expected for an occupation such as Reactor Operator, because the last operating reactor ceased operations in 1980. However, it is unclear why other occupations, such as Health Physics and Maintenance, do not have monitoring data continuing into the early 1990s, consistent with several other occupations, despite the continued operation of other SSFL radiological facilities, such as the Hot Lab Facility (operational through 1988), the Fuel Storage Facility (operational through 1993) and the Radioactive Materials Handling Facility (currently operational). For the two most heavily monitored occupations (Technical Staff and Mechanics), there are data available for every year through 1991, although not necessarily data for every contaminant in each year (see Sections 3.4, 4.3 and 5.4).

Observation 4: To better understand what the overall monitoring data represent, research should be performed to determine why no monitoring data are available for certain occupations in the late 1980s and early 1990s.

Table 5. Occupations Monitored at SSFL[a]

Occupational Group	Number of Records	Range of Years	Number of Individuals[b]	Percentage of Individuals in an Occupational Group
Construction	25	1965–1997	[redacted]	2%
Electrician	131	1965–1988	10	3%
Fireman	33	1965–1974	[redacted]	2%
Health and Safety/ Health Physics[c]	281	1964–1988	12	4%
Inspector	38	1975–1994	[redacted]	1%
Janitor	310	1965–1983	12	4%
Machinist	112	1965–1983	[redacted]	3%
Maintenance[d]	203	1965–1983	17	5%
Management	208	1965–1986	[redacted]	2%
Mechanic[e]	2,446	1963–1993	81	26%
Office Personnel	153	1965–1989	12	4%
Operator	58	1965–1976	[redacted]	2%
Quality Assurance	63	1970–1986	[redacted]	1%
Radiographer	24	1975–1982	[redacted]	<1%
Reactor Operator	85	1965–1983	[redacted]	2%
Security	51	1965–1989	[redacted]	3%
Shipping	8	1965–1968	[redacted]	<1%
Technical Staff[f]	1,531	1965–1993	96	30%
Technician	185	1949 –1997	[redacted]	3%
Truck Driver	86	1966–1986	[redacted]	1%
Unknown[g]	241	1965–1981	[redacted]	1%
Welder	51	1965–1982	[redacted]	1%
Total	6,323	-	316	-

[a] Excludes dose records prior to 1965 and between January 1991 and mid-June 1993.
[b] An individual may be represented more than once if their occupation changed over the course of their monitoring.
[c] Includes medical staff, and health physicists and their support staff
[d] Includes individuals with 'maintenance' as part of their job title, utility services staff, and painters
[e] Includes individuals with 'mechanic' somewhere in their job title, and those who describe themselves as reactor and reactor fuel assemblers/disassemblers.
[f] Includes individuals with 'engineer' as part of their job title, physicists, chemists, and analysts
[g] Either a job title was not specified and marked as unknown, or it is unclear what the job title represents (for example, 'Y&S', 'FMS').

More specific occupational information related to plutonium, fission product, and uranium monitoring is provided in Sections 3.4, 4.3 and 5.4, respectively, of this report.

3.0 REVIEW OF PLUTONIUM DATA AND ASSOCIATED COWORKER INTAKES

Section 3 provides a detailed review of the plutonium coworker model presented in ORAUT-OTIB-0080. Specifically, the OPOS values are examined for applicability and appropriateness in coworker modeling (Section 3.1). The raw data were also used to try and recreate the OPOS results presented in Attachment 1 of ORAUT-OTIB-0080 (see Section 3.2 below). A discussion of plutonium operations throughout the period of interest can be found in Section 3.3. Section 3.4 presents an analysis of the occupations monitored for internal plutonium exposures.

3.1 REVIEW OF CALCULATED OPOS EXCRETION RESULTS

Calculated urinary excretion rates (OPOS values) from ORAUT-OTIB-0080 are presented in Table 6.

Table 6. Plutonium Excretion Rates (dpm/1.4L)

Effective Bioassay Date	50th Percentile	84th Percentile	No. of employees
1/1/1968	0.064	0.328	34
1/1/1970	0.026	0.097	31
1/1/1974	0.013	0.040	28
1/1/1976	0.037	0.049	48
1/1/1980	0.022	0.034	30
1/1/1984	0.040	0.053	48
1/1/1986	0.020	0.031	53

Source: ORAUT-OTIB-0080, Attachment A, Table A-1

As seen in Table 6, the calculated 50th and 84th percentile OPOS values were not analyzed over the standard period of 1 year, as can be seen in the choice of effective bioassay dates (column 1). The effective bioassay date represents the midpoint of a given OPOS evaluation period. For example, if the evaluation period was the standard "1 year" period, the effective bioassay date for that period would be on July 1st (approximately halfway through the evaluation period). In the case of plutonium, NIOSH did not begin to use the plutonium data until 1967; specifically, ORAUT-OTIB-0080 states:

> *Insufficient* [plutonium] *bioassay data were available for 1965 and 1966 to perform a statistical analysis… The 1967–1968 data was used to back-extrapolate for 1965 and 1966, when the potential plutonium exposure was less.*

Therefore, it can be assumed that the first OPOS evaluation period began on January 1, 1967, with its midpoint on January 1, 1968, and its endpoint on December 31, 1968 (or alternately January 1, 1969). Similarly, the OPOS evaluation of available plutonium data ended in 1986 and has an "effective bioassay date" of January 1, 1986. Therefore, the final evaluation period would have started on January 1, 1985. Using this logic, SC&A attempted to match the effective bioassay dates shown above in Table 6 with the period under evaluation. Table 7 demonstrates this attempt to match up the effective bioassay dates with appropriate analysis intervals.

Table 7. Plutonium OPOS Analysis Period Reconstruction Based on Effective Bioassay Dates Shown in ORAUT-OTIB-0080 Table A-1

ORAUT-OTIB-0080 Effective Bioassay Date	SC&A Assumed Start Date	SC&A Assumed End Date	Years Included in OPOS Result
1/1/1968	1/1/1967	12/31/1968	2
1/1/1970	1/1/1969	12/31/1970	2
1/1/1974	1/1/1971	12/31/1976	6
1/1/1976	**Unknown**	**Unknown**	-
1/1/1980	1/1/1977	12/31/1982	6
1/1/1984	1/1/1983	12/31/1984	2
1/1/1986	1/1/1985	12/31/1986	2

The method SC&A used in Table 7 is the same as the example described above. The assumptions were that the starting date for the first acceptable plutonium analysis period was January 1, 1967, and the ending date for the final acceptable plutonium analysis period was December 31, 1986. As one can see in the table, the middle evaluation period highlighted in red does not comport with the surrounding evaluation periods. In fact, the effective bioassay date of January 1, 1976, appears to be covered by the previous evaluation period, which spans from January 1, 1971, to December 31, 1976. If the highlighted entry in Table 7 was removed, then the remaining evaluation periods and effective bioassay dates cover the entire period seamlessly.

Finding 2: SC&A was unable to reconcile apparent differences between the effective bioassay dates listed in Table A-1 of ORAUT-OTIB-0080 and the known start and end dates of "statistically usable" plutonium data.

Additionally, it should be noted that whatever the correct evaluation periods should be, there will necessarily be evaluation periods that cover more than 1 year. It is assumed that these OPOS evaluation periods were chosen to reach the recommended "30 workers" per period in accordance with ORAUT-RPRT-0053, *Analysis of Stratified Coworker Datasets* (ORAUT 2012). RPRT-0053, however, also states:

> *Data from multiple years (usually no more than 3) can be combined to achieve this minimum if the conditions in the workplace are reasonably constant over the period in question.* (ORAUT 2012, pg. 9)

Similarly, SC&A found in its review of that report that combining periods of bioassay for the purposes of calculating OPOS values must be validated and cannot be based solely on gaining the desired number of samples. It must be demonstrated that the combination of longer periods is appropriate and defensible from an exposure potential standpoint. On the other hand, it has been discussed at recent SEC Work Group meetings that a given evaluation period in which less than 30 OPOS results are available may be acceptable if the exposed population was very small (i.e., bench scale operations in which one would expect a low number of monitored workers).

Neton 2014a has the following statements regarding the requirements of sample sizes for coworker analysis:

> *The amount of the available monitoring data should also be evaluated to determine if there are sufficient measurements to ensure that the data are either bounding or representative of the exposure potential at the facility. Facilities with the potential for internal and/or external exposure to a large percentage of the workforce would require many more samples than one in which the potential for exposure was limited to just a few workers. In addition, the variability of the exposure potential should be considered. It has been observed, for example, that some National Laboratories conducted work under many different experimental configurations, resulting in a wide variety of exposure potentials. In this case, it might not be possible to generate a single coworker model that adequately captures all categories of unmonitored worker doses.*
>
> *Although there is no hard and fast rule for the minimum number of data points required to represent a given time interval, approximately fifteen values has* [sic] *been cited as a reasonable number for performing statistical tests on censored datasets (Singh et al. 2010). Because our program estimates parameters from the data, a default minimum of 30 values is recommended per modeled interval. The minimum number of samples should, of course, be considered in light of the number of worker's* [sic] *potentially expose* [sic] *to the airborne source-term.* (Neton 2014a, pg. 2)

SC&A understands that these global policy discussions are ongoing and many post-date the release of this coworker model. However, these newer developments should be considered in the discussions and issues resolution process for this internal dose reconstruction method going forward.

Finding 3: Combining multiple years for the purpose of calculating the 50th and 84th percentile OPOS results must be scientifically justified in the context of the potential for historical changes in operations and associated intakes for monitored workers.

ORAUT-OTIB-0080 determined that there was not a sufficient number of plutonium bioassays in the first 2 years of evaluation (1965–1966) to perform statistical evaluation. As stated previously in this section, the rationale for back extrapolation was that plutonium exposure potential was less in 1965–1966 than it was in 1966–1967, and therefore the back extrapolation of intakes is claimant favorable. Specifically, ORAUT-OTIB-0080 (pg. 11) states:

> *The Nuclear Materials development facility began operation in 1967, which led to an increase in potential plutonium exposure and therefore plutonium urinalysis data. The 1967-to-1968 data was used to back-extrapolate for 1965 and 1966, when the potential plutonium exposure was less. This assumption is favorable to the claimants.*

SC&A agrees with NIOSH that the data are very limited during those first 2 years of potential evaluation, with only 14 bioassay samples identified. While these samples may not be useful for the standard statistical analysis, they do contain valuable information to indicate exposure potential during those years. SC&A performed a simple rank ordering of the available 1965–

1966 data for comparison with the calculated OPOS excretion rates at the 50th and 84th percentile shown in Table A-1 of ORAUT-OTIB-0080. This comparison is shown in Figure 7. The blue dots represent the actual plutonium bioassay data for 1965–1966, the red dots represent the calculated excretion data from ORAUT-OTIB-0080, and the green dots represent the linear interpolation between the actual 1965–1966 data to obtain an approximate 50th and 84th percentile value.

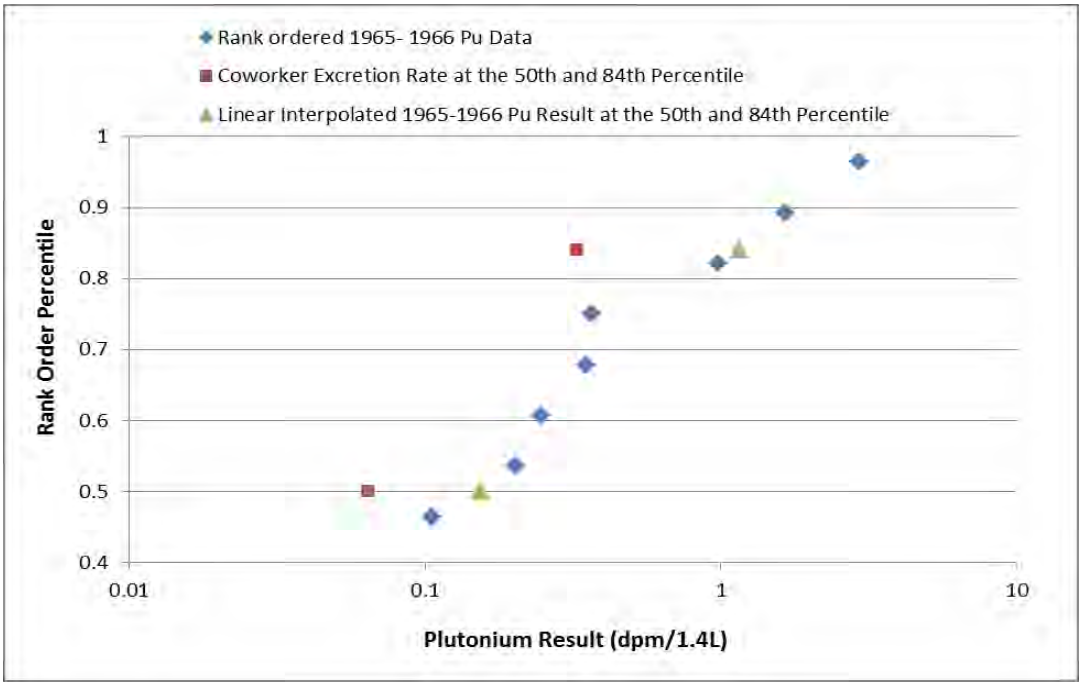


Figure 7. Comparison of Available 1965 and 1966 Plutonium Urinalysis Values Including Interpolated Values at the 50th and 84th Percentile with Calculated OPOS Results from 1967–1968 at the 50th and 84th Percentile

Though the data are limited, the calculated 50th and 84th percentile excretion rates shown for 1967–1968 do not bound the rank-ordered data for the earlier period. In order to justify the back extrapolation of one period to earlier period, one must clearly demonstrate that the exposure potential is bounded by the later period. This can be accomplished through work place monitoring, BZ/daily weighted exposure studies, source term characterizations, etc. SC&A concedes that the increase in plutonium monitoring likely mirrors an increase in the number of workers exposed to plutonium. However, it cannot be assumed a priori that this indicates an increase in intake potential to the exposed worker.

Finding 4: More definitive evidence is warranted to establish that exposure potential to plutonium was lower in 1965–1966 than it was in 1967–1968 to validate the back extrapolation of plutonium results to the earlier period.

SC&A also had the following observation pertaining to the text of ORAUT-OTIB-0080:

Observation 5: Page 9 of ORAUT-OTIB-0080 states, "Sufficient plutonium bioassay data were available to perform a statistical analysis for 1965 through 1986." Page 11 of ORAUT-OTIB-0080 states, "Insufficient [plutonium] bioassay data were available for 1965 and 1966 to perform a statistical analysis… The 1967–1968 data was used to back-extrapolate for 1965 and 1966, when the potential plutonium exposure was less." It appears based on the analysis and discussion performed that the latter statement is correct and that the former statement should be revised.

3.2 REPLICATION OF CALCULATED PLUTONIUM OPOS VALUES

SC&A was provided with the raw database used by ORAUT in calculation of plutonium OPOS values shown in Table A-1 of ORAUT-OTIB-0080. SC&A extracted the appropriate sample results based on the criteria described in Section 1 of this report. SC&A then calculated OPOS values based on the assumed evaluation period shown in Table 6 in Section 3.1. As was noted in that section, SC&A was not able to determine the exact evaluation periods utilized by ORAUT for the purposes of calculating OPOS, and there is some confusion as to how the middle evaluation periods were obtained. Nonetheless, it appears that comparison of the early evaluation periods and later evaluation periods has some merit based on the total number of monitored workers. Table 8 contains the results of SC&A's OPOS calculation as compared to the ORAUT Table A-1 values.

Table 8. Comparison of SC&A-Calculated OPOS Results and ORAUT-OTIB-0080 Table A-1 OPOS Results

Effective Bioassay Date	SC&A Assumed Start Date	SC&A Assumed End Date	# SC&A OPOS	# NIOSH OPOS	SC&A 50th Percentile OPOS (dpm/1.4L)	NIOSH 50th Percentile OPOS (dpm/1.4L)	SC&A 84th Percentile OPOS (dpm/1.4L)	NIOSH 84th Percentile OPOS (dpm/1.4L)
1/1/1968	1/1/1967	12/31/1968	33	34	0.0922	0.064	0.585	0.328
1/1/1970	1/1/1969	12/31/1970	31	31	0.0691	0.026	0.207	0.097
1/1/1974	1/1/1971	12/31/1976	77	28	0.0235	0.013	0.050	0.040
1/1/1976*	N/A	N/A	N/A	48	N/A	0.037	N/A	0.049
1/1/1980	1/1/1977	12/31/1982	46	30	0.0234	0.022	0.049	0.034
1/1/1984	1/1/1983	12/31/1984	48	48	0.0277	0.04	0.047	0.053
1/1/1986	1/1/1985	12/31/1986	53	53	0.021	0.02	0.036	0.031

* SC&A did not calculate an OPOS result for this effective bioassay date entry because it appeared to fall within the assumed evaluation dates of surrounding entries (see Section 3.1).

It is not surprising that NIOSH and SC&A obtained different OPOS results, considering the confusion concerning what evaluation periods were actually used. Other reasons might also account for the differences, such as data selection/extraction. Interestingly, the number of workers involved in the calculation of the OPOS for the first two periods and the last two periods were remarkably similar. However in several cases, the actual magnitude of the calculated excreta result was higher in SC&A's calculation. The exact nature and source of these discrepancies is unknown, but should be resolved to assure that the calculated excreta values used in calculating coworker intakes are accurate.

Finding 5: SC&A was not able to duplicate the plutonium OPOS values presented in Table A-1 to within a reasonable level of precision. The source of the discrepancies should be identified and resolved prior to the calculation of coworker intakes.

SC&A also had the following observation concerning the use of the dose information in the electronic database:

Observation 6: The applicable plutonium data contained two results with units of weight (2.69 gm and 0 µg), both marked as "urine" results. Both results were set to zero dpm/1.4L. In the case of the former result, it is unclear why the sample was set to zero; however, the effect of this adjustment is likely negligible. Regardless, ORAUT-OTIB-0080 specifies on page 8 that samples given in units of mass are assumed to be fecal samples and are not to be used. However, these two samples were marked as "Y" in the "Use" column of the database.

3.3 DISCUSSION OF HISTORICAL PLUTONIUM PROCESSES

To more fully explore whether Findings 3 and 4 are valid concerns, SC&A examined the operational history of plutonium processes at SSFL.

Facility Operations

Operations at SSFL have involved the use of plutonium radionuclides over much of the site's operational period. The most prominent facility that used plutonium was the Nuclear Materials Development Facility (NMDF), which operated from 1967 through 1979. The NMDF was constructed specifically to work with plutonium. Its operating history is summarized in Table 9. Based on this table, it appears that process campaigns typically did not extend for more than 1 or 2 years, which does not agree with the more lengthy OPOS periods assumed by NIOSH in Table 7 of this report, which SC&A attempted to replicate in Table 8.

Table 9. Operations at the Nuclear Materials Development Facility Operating Period Operation

Operating Period	Operation
1967–1968	Development of Analysis Technologies for uranium-plutonium oxide fuels
4/1968–6/1969	Recycle of scrap uranium-plutonium fuel
7/1968–6/1970	Development of technologies to mix tungsten into uranium-plutonium carbide fuel
4/1970–9/1970	Preparation of samples for uranium-plutonium oxide irradiation studies
9/1970–3/1974	Idle
1974–1975	Bench scale tests-recovery of plutonium from simulated waste
1975 – 5/1977	Mixed uranium-plutonium carbide fuel fabrication
5/1977–11/1978	Partial decontamination and clean-up
11/1978–11/1979	Fabrication of depleted uranium carbide fuel
11/1979–10/1982	Idle
10/1982–10/1986	Decontamination and decommissioning
7/1987	Released for unrestricted use

Source: Rucker 2009

Although plutonium fuel was manufactured at SSFL Area IV, according to operational descriptions, none of the reactors or criticality experiments operated at SSFL Area IV used

plutonium as a fuel. The manufactured plutonium fuel was actually a mixture of plutonium and uranium oxide pellets. The plutonium would have been comprised primarily of the isotope Pu-239, with lesser amounts of Pu-238, Pu-240, and Pu-241 (Rucker 2009).

The following processes were used in about 1975 to 1977 (Rockwell 1975) and do not appear to be representative of earlier operations based on the description of activities in Table 9 of this report:

1. Receipt, sampling, analysis, and storage of raw feed materials, including ceramic-grade PuO_2 powder, enriched UO_2 powder, and depleted UO_2 powder
2. Batch weighing, blending, and agglomeration of PuO_2 powder, UO_2 powder, and graphite
3. Carbothermic reduction of oxide powders and graphite to plutonium-uranium carbide
4. Crushing, milling and agglomeration
5. Chemical analysis of carbide sample
6. Pressing of carbide into pellets
7. Sintering of carbide pellets
8. Product analysis
9. Chemical analysis of selected pellets
10. Metallographic examination of selected pellets
11. Pellet dimensional and weight measurements
12. Batching pellets for fuel pin loading
13. Load pellets and pin hardware into fuel pins
14. Settle fuel pellets and hardware into sodium
15. Weld fuel pin end closure
16. Centrifugally bond fuel pins
17. Fuel Pin Inspection
18. Eddy current inspection
19. Alpha monitor
20. Leak test
21. Autoradiograph
22. X-ray welds and pin internal hardware
23. Gamma scan
24. Dimensional
25. Penetrate Xenon capsule
26. Package and ship fuel pins
27. Prepare, package, and ship scrap fuel material to recovery vendor

The Hot Lab (Building 4020), which operated from 1959 to 1990, was used to examine reactor fuel assemblies and other test specimens to determine how they were performing. This involved handling and examining highly radioactive items. This facility was also used to support disassembly, fuel removal, and size reduction of items generated when a reactor was

decommissioned. In addition, the Hot Lab was used to manufacture and leak-check sealed sources, and machining operations involving Co-60. As a result of this work, the interior of the Hot Cells in the lab and associated equipment became contaminated with uranium, plutonium, thorium, and fission and activation products (ORAUT 2006a). No information was found in the Site Research Database (SRDB) that would indicate how operations may have changed over time in the Hot Lab.

The Hot Cave in the Engineering Test Building (Building 4003) supported testing of nuclear fuel reprocessing from SSFL reactors and other off-site reactors as described in ORAUT-TKBS-0038-2 (ORAUT 2006a), and operated from 1954 to 1964. The used fuel assemblies from nuclear reactors contain unused fissionable material and transuranic products, primarily consisting of plutonium. The facility was decontaminated and decommissioned in 1975 (ORAUT 2006a).

Another facility associated with plutonium was the Fuel Storage Facility (Building 4064). This facility, which operated from 1958 to 1993, was used to provide safe, secure storage of fissile fuel material such as plutonium (Rucker 2009).

Non-reactor-related research programs were performed at SSFL. One research program that required the use of plutonium was the TRUMP-S program, but this program was transferred to the University of Missouri. Seventy-five (75) grams of depleted uranium, 5 grams of plutonium, and 4 grams of neptunium, were received at SSFL Area IV and stored in the Fuel Storage Facility before being shipped to the University of Missouri sometime after 1990 (Rucker 2009).

Plutonium Exposure Data

SC&A reviewed available literature to determine if there were facility-specific or individual-specific plutonium exposure data available that would provide some indication of the prevalence and duration of plutonium-related activities.

Some limited data are available that show how many NMDF workers received bioassays during short time periods in the early 1980s. Table 10 shows, by quarter, the number of people receiving bioassays, the number of samples taken, the number of analyses performed and number of positive results. No positive bioassay results were documented during these quarterly reporting periods. Based on Table 10, it appears that operations occurred intermittently, since no workers were monitored during some of the monitoring periods.

Table 10. Bioassay Data for the NMDF

Quarter	Number of Individuals Monitored	Number of Samples	Number of Analyses Performed	Number of Positive Results
1980 Q2	3	3	4	0
1980 Q3	3	3	6	0
1980 Q4	0	0	0	0
1981 Q2	0	0	0	0
1982 Q2	3	3	8	0
1982 Q3	1	2	3	0

Source: Moore 1980a, 1980b, 1981a, 1981b, 1982a, 1982b

In another report (Barnes 2000), a screening analysis was performed to support regulatory compliance efforts at SSFL. In this study, a screening of all positive bioassay results between the years 1989 through 1999 was conducted. In the majority of cases, the screening demonstrated that Committed Effective Dose Equivalents (CEDEs) were less than 10 mrem. In certain cases, the screening results suggested that CEDEs greater than 100 mrem had been received. In these cases, formal dose assessments were conducted. Table 11 summarizes the results related to plutonium exposures. This report was the only one found that tied names of individuals to exposure results. The report did not identify the facilities in which these exposures took place. As can be seen, the exposures occurred over a short time span in 1989 and 1990, and did not extend through 1999. However, given that all documented plutonium activities had wound down by the late 1980s, as shown in Table 9, it is reasonable to assume that exposures monitored in 1989 and 1990 would have been higher than those in the mid to late 1990s due to generally accepted removal processes that deplete residual contamination.

Table 11. Plutonium Bioassay Results from 1989–1999

Name	Date of Intake	Isotope (Class)	CEDE (mrem)	Method of Calculation
[redacted]	2/20/90	Pu-239 (W)	745	Assessment
[redacted]	12/19/89	Pu-239 (W)	45	Assessment
[redacted]	4/18/90	Pu-239 (W)	<10	Assessment
[redacted]	7/14/90	Pu-239 (W)	<10	Assessment
[redacted]	12/18/89	Pu-239 (W)	85	Screen
[redacted]	9/15/90	Pu-239 (W)	85	Screen
[redacted]	4/16/90	Pu-239 (W)	139	Assessment
[redacted]	7/14/90	Pu-239 (W)	26	Assessment
[redacted]	4/17/90	Pu-239 (W)	45	Screen
[redacted]	3/18/90	Pu-239 (W)	<10	Assessment
[redacted]	5/14/90	Pu-239 (W)	284	Assessment
[redacted]	7/15/90	Pu-239 (W)	11	Assessment

Source: Barnes 2000

Incidents Involving Plutonium

The following two incidents were found in available literature in the SRDB that involved possible exposures to plutonium. No incidents were found in the DOE's Occupational Reporting and Processing System, which contains incident reports since 1990. These incident descriptions provide a better understanding of the hazards that were encountered and protective measures used.

> *On February 7, 1983, while disassembling a balance in glove box 3A in the NMDF, an employee accidentally jammed a small screwdriver through the box glove and surgeon's glove, resulting in a puncture wound at the base of his left index finger. He immediately notified Radiation and Nuclear Safety. A survey of the wound and finger indicated approximately 3500 dpm alpha. His surgeon's glove indicated approximately 30,000 dpm alpha. The total activity in the wound was estimated to be about 7 nCi. Surgery removed any remaining activity with bioassays showing negative results (Rockwell 1986).*

> *On July 15, 1983, an incident involving irradiated plutonium fines and fission product radioactivity occurred at the Decon Room 4 alpha glove box in the Hot Lab. While preparing to replace a leaded glove with a plastic bag in a glove box, the glove box operator checked his gloved hands and detected alpha contamination. Personnel performed a survey that indicated gross alpha contamination on the hood, mask, lab coat of the operator (>50,000 c/m), as well as the glove port and the immediate surrounding area. Evacuation was ordered and additional support requested. Wearing respiratory protection, gloves, etc., two Radiation and Nuclear Safety representatives carefully stripped contaminated articles from the operator, leaving his full-face respirator until last. Activity levels up to 50,000 c/m alpha were detected, with 5,000 c/m alpha on the right lower part of the pant leg. The gross activity was removed (to less than 200 c/m alpha). His mask was then removed. Nasal smears of involved personnel indicated no significant radioactivity. Urine and fecal samples were requested and submitted for analyses. The urinalysis results were negative; the fecal specimen was lost at the laboratory during processing (Rockwell 1986).*

SC&A's review of plutonium exposure data and incident descriptions leads to the following finding and observation:

Finding 6: SC&A's research into the historical use of plutonium at SSFL indicates that programs or campaigns in plutonium facilities were fairly diverse and often did not appear to extend past 1 or 2 years in most cases (at least in the case of the primary plutonium facility NMDF). Therefore, averaging plutonium data across multiple years for the purposes of calculating coworker intakes may not be appropriate.

Observation 7: Bioassay data in available literature pertaining to the monitoring of plutonium intakes support the assumption made in the internal dose model that the intake rate in the late 1980s and 1990 is representative (or even bounding) of plutonium intakes that could have occurred in the 1990s.

3.4 OCCUPATIONS MONITORED FOR PLUTONIUM INTAKE

Occupational information was analyzed by cross-referencing dose records that specify an analytical method for detecting plutonium with occupational information developed as described in Section 1.0. From Section 1.0, SC&A determined that the monitoring program appeared to target the workers most at risk (technical staff and mechanics), while also covering a cross-section of other occupations that could also be exposed. Table 12 shows the number and percentage of workers in each occupation monitored for plutonium intakes, as compared to the total number of workers monitored for plutonium, and the years records are available.

Based on Table 12, it appears that monitoring for plutonium among the different occupations mirrors the overall site data presented in Section 1.0, with the Mechanics (38%) and Technical Staff (27%) receiving the most plutonium monitoring while other, but not all, occupations were monitored. It is noteworthy that no monitoring data are available for Mechanics and Technical Staff in the mid-1970s despite plutonium operations occurring at the NMDF, per Table 9.

Observation 8: While there are temporal gaps in the job title coverage for mechanics and technical staff, less than ¼ of the monitored workers could be identified with a particular occupation. Of the monitored workers for which job title information exists, mechanics and technical staff constitute 65% of the population. Therefore, it is reasonable to assume that these higher-risk job types were not systematically excluded and were likely targeted for plutonium monitoring.

Table 12. Occupations Monitored for Plutonium Intakes at SSFL[a]

Occupational Group	Number of Individuals[b]	Years Monitoring Data Available	Percentage of Individuals in an Occupational Group
Construction	[redacted]	1975	2.1%
Electrician	[redacted]	1970–1971, 1975	6.3%
Fireman	0	-	0%
Health and Safety/Health Physics[c]	[redacted]	1965–1976, 1984–1985	10.4%
Inspector	[redacted]	1987–1988, 1990	2.1%
Janitor	0	-	0%
Machinist	0	-	0%
Maintenance[d]	0	-	0%
Management	0	-	0%
Mechanic[e]	18	1967–1973, 1978–1990	37.5%
Office Personnel	[redacted]	1983, 1986	2.1%
Operator	0	-	0%
Quality Assurance	[redacted]	1985	2.1%
Radiographer	0	-	0%
Reactor Operator	0	-	0%
Security	0	-	0%
Shipping	0	-	0%
Technical Staff[f]	13	1967–1976, 1984	27.1%
Technician	[redacted]	1987	2.1%
Truck Driver	[redacted]	1983–1984	2.1%
Unknown[g]	[redacted]	1968–1971, 1974–1979, 1983–1985, 1987, 1989	6.3%
Welder	0	-	0%
Total	48	-	100%

[a] Excludes dose records prior to 1965 and between January 1991 and mid-June 1993.
[b] An individual may be represented more than once if their occupation changed over the course of their monitoring.
[c] Includes medical staff, and health physicists and their support staff
[d] Includes individuals with 'maintenance' as part of their job title, utility services staff, and painters
[e] Includes individuals with 'mechanic' somewhere in their job title, and those who describe themselves as reactor and reactor fuel assemblers/disassemblers.
[f] Includes individuals with 'engineer' as part of their job title, physicists, chemists, and analysts
[g] Either a job title was not specified and marked as unknown, or it is unclear what the job title represents (for example, 'Y&S', 'FMS').

4.0 REVIEW OF FISSION PRODUCT DATA AND ASSOCIATED COWORKER INTAKES

Section 4 provides a detailed review of the fission product coworker model presented in ORAUT-OTIB-0080. Specifically, the OPOS values are examined for applicability and appropriateness in coworker modeling (Section 4.1). The raw data were also used to try and recreate the OPOS results presented in Attachment 1 of ORAUT-OTIB-0080 (Section 4.2). Section 4.3 presents an analysis of the occupations monitored for internal fission product exposures.

4.1 REVIEW OF CALCULATED OPOS RESULTS

The calculated OPOS excretion rates for fission products were calculated on an annual basis for the purpose of coworker intake modeling from 1965–1991. SC&A was provided with the raw database used by ORAUT in calculation of fission product OPOS values shown in Table A-3 of ORAUT-OTIB-0080. SC&A extracted the appropriate sample results based on the criteria described in Section 1 of this report. It was immediately clear that the number of monitored workers was significantly different between the two compilations in the earlier years. A comparison of the number of OPOS results per evaluation period is shown in Table 13. As is evident from the table, the two compilations of data do not agree until about 1975, at which point the two compilations become nearly identical as far as how many monitored workers exist.

Table 13. Comparison of SC&A and NIOSH OPOS Totals by Year

Year	SC&A OPOS Total	NIOSH OPOS Total	Year	SC&A OPOS Total	NIOSH OPOS Total
1965	52	321	1979	84	84
1966	24	216	1980	79	82
1967	21	221	1981	55	55
1968	136	154	1982	65	65
1969	111	116	1983	45	45
1970	44	125	1984	65	65
1971	0	84	1985	67	67
1972	0	75	1986	78	78
1973	0	66	1987	63	63
1974	51	55	1988	65	65
1975	83	84	1989	50	50
1976	67	67	1990	33	33
1977	75	75	1991	27	27
1978	118	118			

ORAUT-OTIB-0080 states the following concerning which records to use for fission product coworker analysis:

> *The analysis type designations that were used were 'MFP' (mixed fission product, chemical separation of alkaline earths and rare earths including strontium)* ***if the method type was "B" for beta counting*** *(which excludes gamma-counting data), 'MFPB', 'MFP(B),' and 'FP'* ***if the method type was '3A'*** *(mixed fission products less cesium and volatiles, assumed to indicate strontium). Sufficient*

bioassay data were available to perform a statistical analysis for 1965 through 1991. [Emphasis added.]

In its OPOS calculation, SC&A also considered the following sample types as "fitting" the stated criteria for inclusion:

- Nuclide/Analysis: MFP, Method Type: 7B → 1 sample
- Nuclide/Analysis: MFP, Type 5B → 1 sample
- Nuclide/Analysis: MFP, Type 3B → 1 sample
- Nuclide/Analysis: MFP, Method Type: 2B (this constituted the vast majority of samples with a Nuclide/Analysis: MFP) →1170 samples
- Nuclide/Analysis: MFP, Method Type: 1B →1 sample
- Nuclide/Analysis: FP, Method Type: 3AG → 2 samples
- Nuclide/Analysis: FP, Method Type: 3A T-U → 1 sample

SC&A noted that additional codes with a "Nuclide/Analysis" type of either FP or MFP were found that did not fit the above list of criteria for inclusion. These samples included the following "Nuclide/Analysis" type and "Method Type" as shown in Table 14.

Table 14. Additional Sample Types Not Meeting the Criteria for Inclusion per ORAUT-OTIB-0080

"Nuclide/Analysis" Type	"Method" Type	Number of Samples
FP	3B	2,737
MFP	Blank	2,388
MFP	3A	10
FP	Blank	9
MFPG	Blank	5
MFP	G	3
FP-3AG-Co-60	Blank	1
MFP	1D	1
MFP(G)	Blank	1

SC&A added the datapoints in Table 14 to the original datapoints that fit the ORAUT-OTIB-0080 criteria and compared the number of workers by year for this expanded dataset. The comparison of the number of monitored workers by year for the expanded SC&A dataset versus the NIOSH compilation in Table A-3 is shown in Table 15. With the exception of minor differences in a handful of years, the expanded SC&A dataset matches the number of monitored workers in each year very well. Note that the total number of workers in the earlier years (left side of the table) now matches exactly, where this was the biggest discrepancy shown in Table 13. Conversely, years that had matched perfectly in Table 13 in the later years (right side of the table) are now off by one or two workers. It is likely that not all of the samples tabulated in Table 13 as being "additional" were actually used in NIOSH's calculation.

Finding 7: Based on the number of monitored workers tabulated in Tables 13 and 14, it is likely that NIOSH did not adhere to the inclusion guidelines outlined in ORAUT-OTIB-0080. It

appears that all samples labelled "FP" or "MFP" in the "Nuclide/Analysis" column were included, regardless of the actual designated method of measurement.

Table 15. Comparison of Expanded SC&A OPOS Totals and NIOSH OPOS Totals by Year

Year	SC&A OPOS Total	NIOSH OPOS Total	Year	SC&A OPOS Total	NIOSH OPOS Total
1965	321	321	1979	85	84
1966	216	216	1980	82	82
1967	221	221	1981	55	55
1968	154	154	1982	67	65
1969	116	116	1983	45	45
1970	124	125	1984	66	65
1971	84	84	1985	68	67
1972	75	75	1986	79	78
1973	66	66	1987	63	63
1974	55	55	1988	65	65
1975	84	84	1989	51	50
1976	67	67	1990	33	33
1977	75	75	1991	27	27
1978	118	118			

4.2 REPLICATION OF CALCULATED OPOS RESULTS

SC&A attempted to recreate the NIOSH OPOS values contained in ORAUT-OTIB-0080 Table A-3 based on the expanded dataset described above. The resulting comparison of the calculated geometric mean (GM) and 84th percentile is shown in Table 16. It is clear from this table that SC&A-calculated OPOS results and NIOSH-calculated values do not match up for most years. In the early years, the NIOSH OPOS result is generally higher, while in the latter years the SC&A result is higher. Sometimes the difference between the calculated 50th percentile by year was a factor of 3–6.

Finding 8: SC&A was not able to duplicate the FP OPOS values presented in Table A-3 of ORAUT-OTIB-0080 to within a reasonable level of precision for most years. The source of the discrepancies should be identified and resolved prior to the calculation of coworker intakes.

Table 16. Comparison of SC&A-Calculated OPOS Values Using Expanded Fission Product Data to NIOSH OPOS Values contained in Table A-3 of ORAUT-OTIB-0080

Year	Effective Bioassay Date	SC&A 50th Percentile OPOS (dpm/1.4L)	NIOSH 50th Percentile OPOS (dpm/1.4L)	SC&A 84th Percentile OPOS (dpm/1.4L)	NIOSH 84th Percentile OPOS (dpm/1.4L)
1965	7/1/1965	15.82	26.06	51.63	58.88
1966	7/1/1966	18.94	21.04	38.84	38.26
1967	7/1/1967	19.64	20.99	44.30	44.65
1968	7/1/1968	5.26	19.63	14.35	30.48
1969	7/1/1969	1.49	10.07	9.25	24.71
1970	7/1/1970	18.44	19.34	29.40	24.40
1971	7/1/1971	29.13	16.54	32.37	28.26
1972	7/1/1972	28.06	14.67	30.04	23.58

Table 16. Comparison of SC&A-Calculated OPOS Values Using Expanded Fission Product Data to NIOSH OPOS Values contained in Table A-3 of ORAUT-OTIB-0080

Year	Effective Bioassay Date	SC&A 50th Percentile OPOS (dpm/1.4L)	NIOSH 50th Percentile OPOS (dpm/1.4L)	SC&A 84th Percentile OPOS (dpm/1.4L)	NIOSH 84th Percentile OPOS (dpm/1.4L)
1973	7/1/1973	27.79	12.90	31.76	24.28
1974	7/1/1974	20.23	4.65	37.65	22.27
1975	7/1/1975	21.20	12.90	33.39	20.51
1976	7/1/1976	17.98	18.72	26.77	29.14
1977	7/1/1977	21.24	6.98	43.84	31.58
1978	7/1/1978	20.59	14.25	42.32	30.33
1979	7/1/1979	20.10	9.06	39.43	22.97
1980	7/1/1980	15.30	5.54	27.34	17.07
1981	7/1/1981	17.24	17.75	27.74	33.77
1982	7/1/1982	24.65	14.67	44.33	33.51
1983	7/1/1983	20.66	15.42	36.01	23.90
1984	7/1/1984	24.17	3.59	44.38	12.29
1985	7/1/1985	26.37	15.89	42.08	26.69
1986	7/1/1986	27.66	16.15	41.77	27.12
1987	7/1/1987	26.16	21.92	43.38	38.70
1988	7/1/1988	33.90	24.52	43.25	29.45
1989	7/1/1989	32.86	22.79	45.78	37.83
1990	7/1/1990	23.47	7.67	47.77	45.77
1991	3/1/1991	28.06	8.08	47.24	35.30

Observation 9: Note that the last effective bioassay date in Table 15 is March 1, 1991. Examination of the available data indicates that FP bioassay data effectively end in mid-April. This comports with an effective bioassay date for the evaluation period of January 1, 1991, to roughly April 30, 1991. Since the end of April 1991 marks the end of the FP bioassay evaluation, the calculated intakes found in ORAUT-OTIB-0080 Table 5-6 should also end in April 1991, not December 1991. However, it is not clear if changing the date of the end of the chronic intake calculation would significantly affect the calculated intakes.

NIOSH has assumed a single chronic intake regime over all 27 years of exposure. SC&A acknowledges that the choice of intake regimes involves some professional judgment on the part of the dose modeler. However, if the intake regimes had been split up into smaller parts, the calculated intake rates could be significantly higher (such as during the periods of 1965–1968 or 1985–1989). In general, the longer the period of time one assumes for the chronic intake regime, the lower the overall calculated intakes will be. The justification for grouping certain years together into a single intake regime is often that the magnitude of the bioassay results are sufficiently similar that the exposure potential during that time is also similar.

Figure 8 plots the 50th percentile NIOSH OPOS value by year; as can be seen, the range in magnitude of the 50th percentile bioassay value can vary significantly. For example, the difference in the calculated result in 1984 (3.59 dpm/1.4L) and surrounding years is approximately a factor of 4. The difference of a factor of 4 is not dissimilar to the plutonium excretion differences in the first and second intake regimes in Table 6 (0.064/0.026 = factor of 2.5), which warranted a change in intake regime. SC&A admits that this is not an issue that is

restricted to the SSFL dataset, but has programmatic implications as well. It would be beneficial to provide the rationale in the context of historical site operations, exposure potential, and observed OPOS variability to lend scientific credibility to the chosen approach of a single intake regime for mixed fission products.

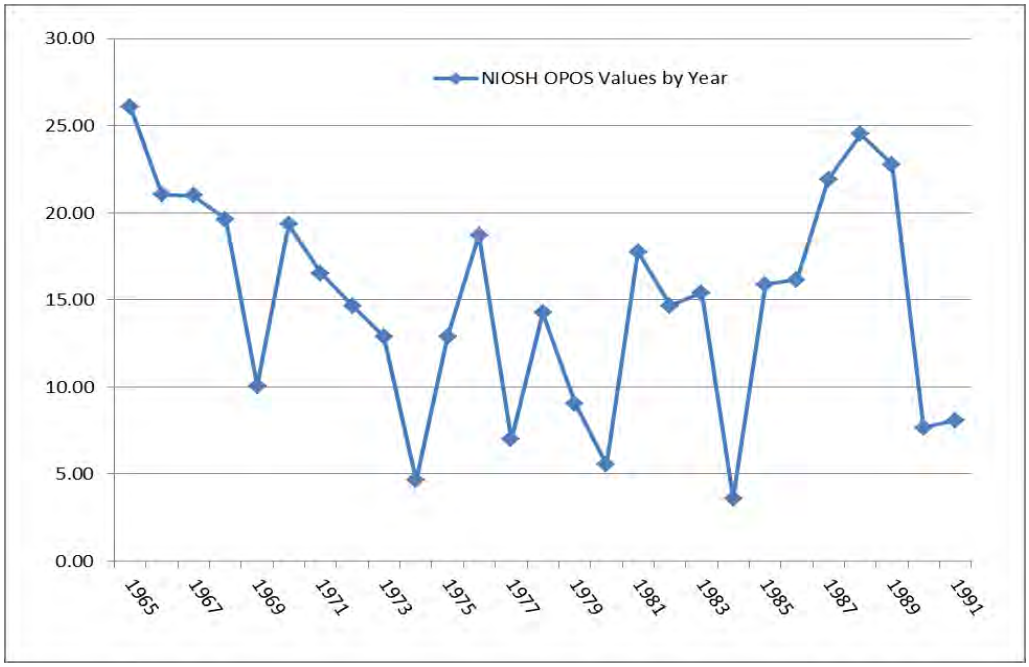


Figure 8. Calculated NIOSH OPOS Values by Year

Finding 9: The coworker model would benefit from a substantive discussion providing the rationale for combining all 27 years into a single intake regime. This might include a discussion on the variability in the calculated measurements observed by year, changes in the GSD indicating a larger spread of potential exposures, the actual effect on the calculated intakes in relation to dose, and changes in site operations. Such a discussion would reinforce the assumption that a single chronic intake regime is appropriate over a significantly long period of time.

4.3 OCCUPATIONS MONITORED FOR FISSION PRODUCT INTAKE

Occupational information was analyzed by cross-referencing dose records that specify an analytical method for detecting mixed fission products with occupational information developed as described in Section 2.0. From Section 2.0, SC&A determined that the monitoring program appeared to target the workers most at risk (technical staff and mechanics), while also covering a cross-section of other occupations that could also be exposed. Table 17 shows the number and percentage of workers in each occupation monitored for mixed fission product intakes as compared to the total number of workers monitored for mixed fission products.

Table 17. Occupations Monitored for Mixed Fission Product Intakes at SSFL[a]

Occupational Group	Number of Individuals[b]	Years Monitoring Data Available	Percentage of Individuals in an Occupational Group
Construction	[redacted]	1975	<1%
Electrician	[redacted]	1974–1976, 1980	3%
Fireman	[redacted]	1968–1970, 1974	3%
Health and Safety/ Health Physics[c]	[redacted]	1968–1970, 1974–1977, 1979–1980, 1984–1985, 1988	4%
Inspector	[redacted]	1988–1990	<1%
Janitor	[redacted]	1968–1970	4%
Machinist	[redacted]	1965–1966	<1%
Maintenance[d]	[redacted]	1968–1969	3%
Management	[redacted]	1965–1969, 1977–1981, 1986	3%
Mechanic[e]	41	1965–1970, 1974–1991	34%
Office Personnel	[redacted]	1968, 1970, 1975, 1979–1982, 1985–1989	2%
Operator	[redacted]	1968–1969	2%
Quality Assurance	[redacted]	1985–1986	<1%
Radiographer	0	-	0%
Reactor Operator	[redacted]	1965–1969, 1981, 1983	3%
Security	[redacted]	1968–1970, 1986, 1989	4%
Shipping	[redacted]	1968	<1%
Technical Staff[f]	32	1965–1970, 1974–1991	27%
Technician	[redacted]	1978–1981, 1984–1989	3%
Truck Driver	[redacted]	1975–1979, 1981–1986	<1%
Unknown[g]	[redacted]	1965, 1977–1991	3%
Welder	[redacted]	1977–1978, 1982	2%
Total	120	-	100%

[a] Excludes dose records prior to 1965 and between January 1991 and mid-June 1993.
[b] An individual may be represented more than once if their occupation changed over the course of their monitoring.
[c] Includes medical staff, and health physicists and their support staff
[d] Includes individuals with 'maintenance' as part of their job title, utility services staff, and painters
[e] Includes individuals with 'mechanic' somewhere in their job title, and those who describe themselves as reactor and reactor fuel assemblers/disassemblers.
[f] Includes individuals with 'engineer' as part of their job title, physicists, chemists, and analysts
[g] Either a job title was not specified and marked as unknown, or it is unclear what the job title represents (for example, 'Y&S', 'FMS').

Based on Table 17, it appears that monitoring for mixed fission products among the different occupations mirrors the overall site data presented in Section 1.0, with the Mechanics (38%) and Technical Staff (27%) receiving the most plutonium monitoring while other, but not all, occupations were monitored. The distribution of monitoring among the occupations also closely resembles the distribution of occupations monitored for plutonium when compared to Table 12. The annual distribution of monitoring among Mechanics and Technical Staff appears to correspond to NMDF operations, which would explain the lack of monitoring data between 1970 and 1974.

5.0 REVIEW OF URANIUM DATA AND ASSOCIATED COWORKER INTAKES

Section 5 provides a detailed review of the uranium coworker model presented in ORAUT-OTIB-0080. Specifically, the OPOS values are examined for applicability and appropriateness in coworker modeling (Section 5.1). The raw data were also used to try and recreate the OPOS results presented in Attachment 1 of ORAUT-OTIB-0080. Section 5.2 presents an analysis of the assumptions associated with uranium uptake, and whether SSFL bioassay collection and analysis practices were fully considered. Section 5.3 presents an analysis of the occupations monitored for internal uranium exposures.

5.1 REVIEW AND REPRODUCTION OF CALCULATED OPOS RESULTS

The OPOS excretion rates for uranium bioassay were calculated on an annual basis for the purpose of coworker intake modeling from 1965–1988. SC&A was provided with the raw database used by ORAUT in calculation of uranium OPOS values shown in Table A-2 of ORAUT-OTIB-0080. SC&A extracted the appropriate sample results based on the criteria described in Section 1 of this report. SC&A attempted to replicate the calculated OPOS values in Table A-2 of ORAUT-OTIB-0080; the results are contained in Table 18.

It is evident from the table that the number of monitored workers contained in both SC&A's and NIOSH's OPOS analysis are identical in all years but two (1967 and 1987). The actual OPOS results between the two calculations are rather close in most years; however, some years display over an order of magnitude difference (1968, 1969, 1975 and 1982). It is not clear from the available data where the discrepancies arise.

Table 18. Comparison of SC&A-Calculated OPOS Results to NIOSH-Calculated OPOS Results Contained in ORAUT-OTIB-0080 Table A-2

Year	Effective Bioassay Date	SC&A OPOS	NIOSH OPOS	SC&A 50th Percentile OPOS (dpm/1.4L)	NIOSH 50th Percentile OPOS (dpm/1.4L)	SC&A 84th Percentile OPOS (dpm/1.4L)	NIOSH 84th Percentile OPOS (dpm/1.4L)
1965	7/1/1965	470	470	2.22	2.90	7.77	8.33
1966	7/1/1966	298	298	4.74	4.52	11.97	10.83
1967	7/1/1967	336	338	5.65	5.47	15.39	14.69
1968	7/1/1968	269	269	0.70	6.22	6.06	14.05
1969	7/1/1969	153	153	0.35	2.50	3.19	8.6
1970	7/1/1970	164	164	6.63	3.41	10.42	7.79
1971	7/1/1971	135	135	8.50	2.35	8.80	5.09
1972	7/1/1972	122	122	8.83	1.68	10.05	5.67
1973	7/1/1973	88	88	6.67	4.99	9.19	7.09
1974	7/1/1974	86	86	4.96	5.95	7.74	7.53
1975	7/1/1975	134	135	4.37	0.23	7.44	1.13
1976	7/1/1976	166	166	6.04	5.77	8.46	8.03
1977	7/1/1977	95	95	5.89	4.52	8.72	6.68
1978	7/1/1978	147	147	4.16	4.89	7.39	14.14
1979	7/1/1979	124	124	4.96	5.14	7.49	6.52

Table 18. Comparison of SC&A-Calculated OPOS Results to NIOSH-Calculated OPOS Results Contained in ORAUT-OTIB-0080 Table A-2

Year	Effective Bioassay Date	SC&A OPOS	NIOSH OPOS	SC&A 50th Percentile OPOS (dpm/1.4L)	NIOSH 50th Percentile OPOS (dpm/1.4L)	SC&A 84th Percentile OPOS (dpm/1.4L)	NIOSH 84th Percentile OPOS (dpm/1.4L)
1980	7/1/1980	109	109	2.91	1.76	6.00	3.66
1981	7/1/1981	104	104	2.61	1.34	3.44	2.08
1982	7/1/1982	74	75	2.27	0.22	3.40	0.86
1983	7/1/1983	57	57	2.43	3.17	3.44	3.41
1984*	7/1/1984	12	Not Listed	2.52	Not Listed	4.15	Not Listed
1985*	7/1/1985	2	Not Listed	Insufficient Data	Not Listed	Insufficient Data	Not Listed
1986	7/1/1986	34	34	2.90	2.05	3.58	3.18
1987	7/1/1987	58	59	2.52	1.48	3.54	2.29
1988	7/1/1988	63	63	2.46	1.44	3.48	2.24

*These years were not included in the ORAUT-OTIB-0080 coworker analysis

Finding 10: SC&A's attempt to duplicate the uranium OPOS values presented in Table A-2 of ORAUT-OTIB-0080 was off by an order of magnitude in 4 of the 25 years analyzed by NIOSH (1984 and 1985 were not analyzed). The source of the discrepancies should be identified and resolved prior to the final calculation of coworker intakes.

As evidenced in Table 18 and also from Section 4.4 of ORAUT-OTIB-0080, NIOSH did not include the years 1984 and 1985 in the coworker analysis. Additionally, it was determined that 1975 results were not contemporaneous with the surrounding years and so were excluded. Specifically, ORAUT-OTIB-0080 states:

> *In 1974,[7] the excretion rates from the statistical analysis were inconsistent with the contemporaneous years and were excluded to be favorable to claimants. In 1984 and 1985, the results are zero (statistical analysis not possible) and were similarly excluded.*

It is assumed that the above statement meant to indicate 1975 as not being contemporaneous with surrounding years (1974 - 5.95 dpm/1.4L; 1975 - 0.23 dpm/1.4L; 1976 - 5.77 dpm/L). It should be noted that SC&A calculated an OPOS value of 4.37 dpm/1.4L for this year, which is contemporaneous with the surrounding years.

With regard to 1984 and 1985, SC&A agrees that it is likely not statistically appropriate to analyze the 1984 and 1985 data for the purposes of a coworker model because of the paucity of available samples (12 and 2 OPOS results per year). The 1984 and 1985 data are tabulated in Table 19 for reference. Only one worker was sampled twice in that 2-year period.

[7] This appears to be a typo and should be "1975"

Table 19. Tabulation of Available 1984 and 1985 Data

Date of Sample	Sample Result (dpm/1.4L)	Date of Sample	Sample Result (dpm/1.4L)
1/1/1984	< 3.547	7/1/1984	< 3.267
2/13/1984	< 3.547	8/8/1984	< 0.215
2/14/1984	< 3.547	11/19/1984	< 3.547
2/14/1984	< 3.547	11/28/1984	< 3.547
2/14/1984	< 3.547	11/28/1984	< 3.547
2/21/1984	0	2/11/1985	0
4/1/1984	< 3.547	4/18/1985	< 3.5
4/9/1984	< 3.547		

Based on the data shown in Table 19, there does not appear to be any discernible reason why samples were taken on the dates shown (such as in reaction to some unforeseen exposure or perhaps a trip to another site). The more important question might be, why there are so few uranium urinalysis data for these 2 years? SC&A was not able to identify any obvious rationale for why there are so few samples taken in 1984 and 1985. A chart of the number of samples and monitored workers for plutonium, uranium, and fission products is shown in Figures 9 and 10.

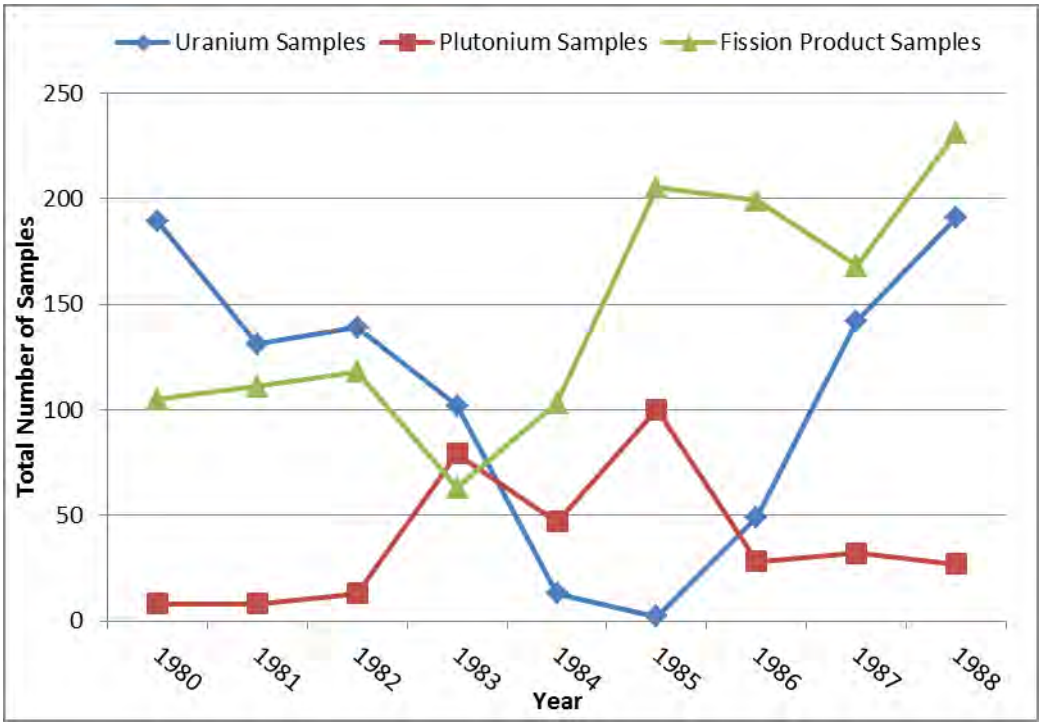


Figure 9. Comparison of the Number of Samples for Coworker Radionuclides during the 1980s

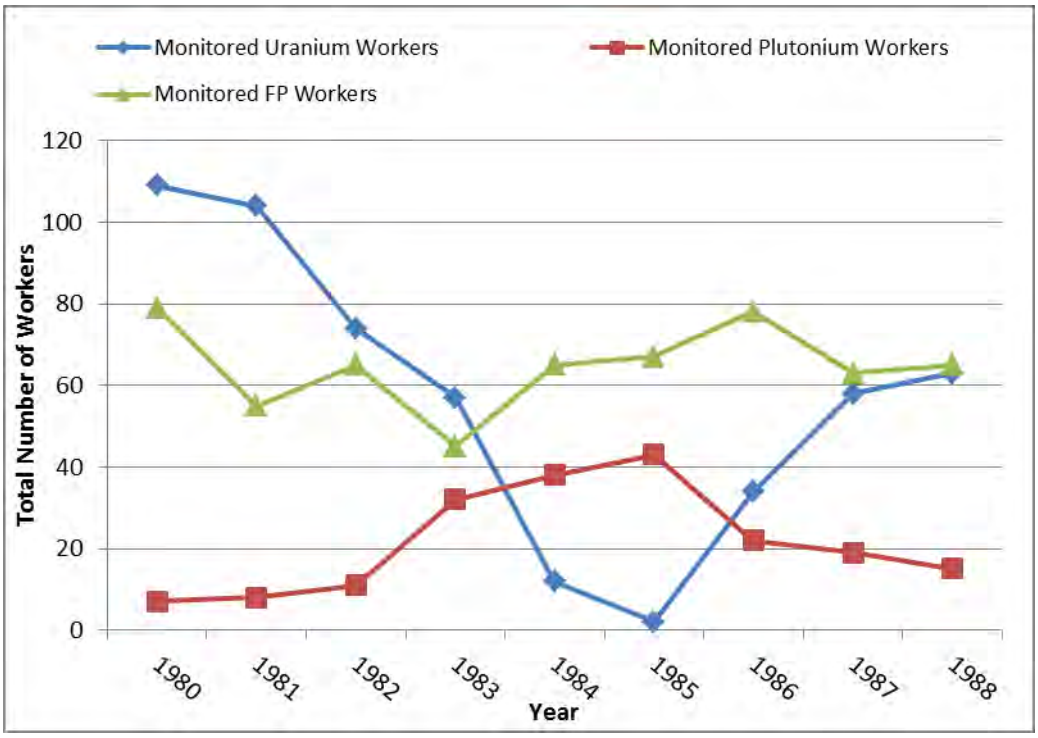


Figure 10. Comparison of the Number of Monitored Workers for Coworker Radionuclides during the 1980s

During the 1980s, plutonium sampling was at its highest from 1983–1985. Fission product sampling increased incrementally from 1983 to 1985. Based on Figure 10, it appears that the number of monitored plutonium and fission product workers was also increased during 1984 and 1985. The trend of little-to-no monitoring results for uranium was not mirrored by the monitoring for other contaminants of concern.

Finding 11: It is important to understand the reasons for the drastic decrease in observed sampling for uranium from 1984 to 1985. Concurrently, because the paucity of the data makes it impossible to make quantitative statements about the available bioassay data from 1984 to 1985, it should be established that conditions and exposure potential are sufficiently similar to (or bounded by) surrounding years to validate the use as surrogate data.

5.2 REVIEW OF URANIUM MONITORING PRACTICES AND INCORPORATION INTO INTERNAL MODEL

SC&A examined how uranium exposures were calculated by NIOSH in relation to SSFL bioassay sample collection and analysis practices.

Based on information in ORAUT-TKBS-0038-5, Rev. 01 (ORAUT 2010, pg. 14), routine urine samples were collected after 48 hours' absence of work (it is unclear whether employees were requested to fill the samples on Sunday during the entire SSFL operational history):

By 1959, routine urine samples were requested on Fridays, and each monitored employee was required to submit the first voiding on Monday morning (after an absence from work of 48 hr or more). The time of the previous voiding was recorded to determine the excretion rate. If the Monday morning sample was verified as positive, a series of 24-hr samples was collected to determine the body burden. Employees were requested to fill these samples on Sunday. Appropriate adjustments to this schedule were made for weekend work, etc.

Table 20 provides a description of the sampling protocol at Santa Susana (from ORAUT 2010, Table 5-7.)

Table 20. Description of Bioassay Sampling Protocol at SSFL

Media	***Sample***
Urine	*Routine samples. Single voiding collected on Monday morning before returning to work. Collected in single bottle. Also referred to as rate samples. Positive samples were followed up with additional rate sample for verification. Time of sample collection and previous voiding were recorded. Normally collected in a 16-oz container. By 1999, 1-L polyethylene containers were used. It appears that one full container (900–1,000 mL) was collected and collection time noted.*
Urine	*24-hr samples. Used for follow-up to verify rate samples or for incidents. One or more samples could be requested. Single samples were collected at home on Sunday. Collected in 32-oz or 1-L polyethylene containers.*
Urine	*Spot samples. For follow-up to incidents, spot sample could be collected as soon as possible. This sample was probably collected in 16-oz container.*
Feces	*Could be requested in conjunction with urine samples as follow-up to incidents. No descriptions of historical fecal kits were located. By 1999, single voiding samples were collected in 83-oz polyethylene containers. The minimum mass that was considered adequate was 30 g.*

Source: ORAUT 2010, Table 5-7

SC&A examined the available data to determine how sample collections were distributed during individual weeks; the results are shown in Table 21. Since the "Date Received" column in the dose records represents when the sample was received by the lab, only samples that had a date listed in the "Sample Date (collection)" column were analyzed in Table 21. Bioassay results with "Date Received" are analyzed in Table 22.

Table 21. Distribution of Bioassay Results by the Day of the Week Sample is Collected

Day of the Week	Uranium	Plutonium	Fission Products
Sunday	489 (9.23%)	345 (39.75%)	424 (12.68%)
Monday	3,029 (57.18%)	227 (26.15%)	1,566 (46.82%)
Tuesday	765 (14.44%)	70 (8.06%)	482 (14.41%)
Wednesday	359 (6.78%)	75 (8.64%)	276 (8.25%)
Thursday	271 (5.12%)	65 (7.49%)	244 (7.29%)
Friday	321 (6.06%)	57 (6.57%)	284 (8.49%)
Saturday	63 (1.19%)	63 (7.26%)	69 (2.06%)
Total	5,297 (100%)	868 (100%)	3,345 (100%)

As shown in Table 21, between 60% and 66% of the available bioassay data that provided a sample collection date was taken on either Sunday or Monday. This reinforces the notion that the majority of samples were taken after approximately a 48-hour absence from work.

For completeness, Table 22 presents the distribution of samples (by day) for bioassay entries that only provide a "Received Date." As seen in the table, the majority of samples for uranium and plutonium were received by the lab on Wednesday or Thursday (~80% for uranium and ~76% for plutonium). This trend was not observed for fission products, where the majority of samples were "received" on Monday. ORAUT-OTIB-0080 states the following concerning the "sample received" date:

> *The sample receipt date on the laboratory data form is usually a few days after the sample collection date on the facility form.*

This is consistent with a bioassay protocol in which the majority of samples were collected on Monday and received a few days later at the analytical laboratory (Wednesday or Thursday).

Table 22. Distribution of Bioassay Results by "Sample Received" Date

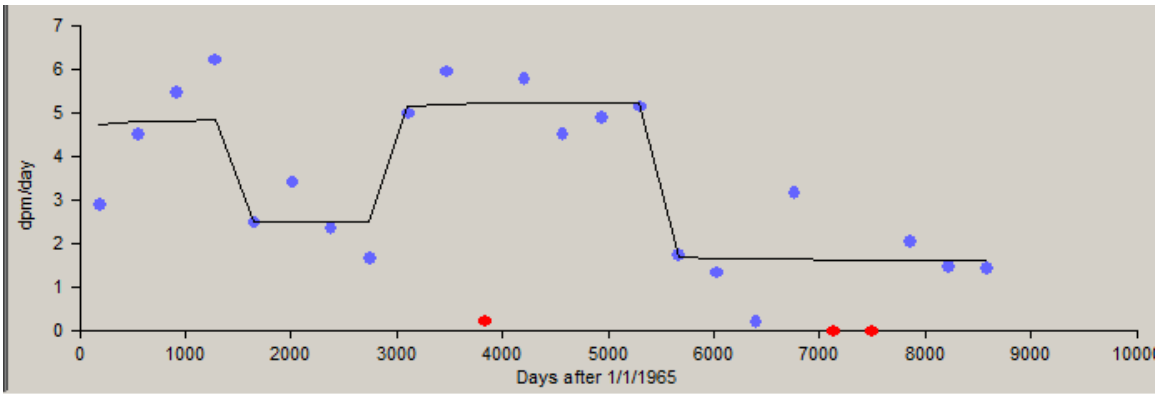
Day of the Week	Uranium	Plutonium	Fission Products
Sunday	11 (0.33%)	3 (8.82%)	5 (15.15%)
Monday	152 (4.51%)	3 (8.82%)	18 (54.55%)
Tuesday	145 (4.30%)	0 (0.00%)	2 (6.06%)
Wednesday	1,179 (34.94%)	11 (32.35%)	2 (6.06%)
Thursday	1,521 (45.08%)	15 (44.12%)	2 (6.06%)
Friday	363 (10.76%)	1 (2.94%)	3 (9.09%)
Saturday	3 (0.09%)	1 (2.94%)	1 (3.03%)
Total	3,374 (100%)	34 (100%)	33 (100%)

In ORAUT-OTIB-0080, Rev. 00, NIOSH assumes a continuous intake pattern to calculate uranium coworker intake rates. If it is assumed that all urine samples were collected after a 48-hour absence from work, the approach used by NIOSH is underestimating the intake rates. For Type F uranium, NIOSH underestimates the intake rate by a factor of around 4. For Type M uranium, the underestimation is a factor of around 2, and for Type S the underestimation factor fluctuates from 1.6 to 2.0. A description of SC&A's analysis for each type of uranium follows.

For Type F uranium, NIOSH calculated GM intake rates to be (ORAUT-OTIB-0080, Table 5-3):

1965–1968	17.6 dpm/d
1969–1972	8.91 dpm/d
1973–1979	18.87 dpm/d
1980–1988	5.666 dpm/d

Figure 11 shows the predicted uranium bioassay results calculated using these derived GM intakes and their comparison with the 50th percentile bioassay results presented in ORAUT-OTIB-0080.



Source: ORAUT-OTIB-0080, Figure A-13

Figure 11. Predicted Uranium Bioassay Results Calculated using IMBA-derived Uranium Intake Rates (Line) compared with Bioassay Results (Dots), 50th Percentile, All Years, Type F

In Table 23 (2nd and 3rd columns), SC&A reproduced the values of Figure 11 using the assumption of a continuous intake. In addition, Table 23 shows the predicted uranium bioassay values (column 4) for samples taken on Monday mornings (after a 48-hour absence from work). As shown in column 5, the predicted excretion rates on Monday mornings are a factor of about 4 smaller than the ones predicted using a continuous intake rate. This means that the intake rates calculated for Type F uranium are underestimated by a factor of about 4.

Tables 24 and 25 are equivalent to Table 23, using uranium compounds of Type M and Type S, respectively.

Table 23. Predicted Urine Excretion Rates Corresponding to NIOSH's Derived GM Intake Rates of Type F Uranium Compounds, Assuming a Continuous Exposure as compared to a 5-day Per Week Exposure, with Monday Morning Sample Collections

Year (OTIB-0080 Intake rate)	50th percentile urine (dpm/d)	Predicted 24-Hour Urine (dpm) Assuming a Continuous Intake	Predicted 24-Hour Urine (dpm/d) Assuming Monday Morning Collection (dpm/d)	Underestimation of Intake*
1965 (17.6 dpm/d)	2.90	4.78	1.10	4.35
1966 (17.6 dpm/d)	4.52	4.80	1.11	4.32
1967 (17.6 dpm/d)	5.47	4.81	1.13	4.26
1968 (17.6 dpm/d)	6.22	4.83	1.13	4.27
1969 (8.91 dpm/d)	2.50	2.42	0.56	4.35
1970 (8.91 dpm/d)	3.41	2.43	0.56	4.31
1971 (8.91 dpm/d)	2.35	2.44	0.57	4.28
1972 (8.91 dpm/d)	1.68	2.44	0.57	4.25
1973 (18.87 dpm/d)	4.99	5.12	1.18	4.34
1974 (18.87 dpm/d)	5.95	5.15	1.19	4.33
1975 (18.87 dpm/d)	0.23	5.16	1.21	4.26
1976 (18.87 dpm/d)	5.77	5.18	1.22	4.25
1977 (18.87 dpm/d)	4.52	5.19	1.23	4.22
1978 (18.87 dpm/d)	4.89	5.20	1.23	4.23
1979 (18.87 dpm/d)	5.14	5.21	1.24	4.20
1980 (5.666 dpm/d)	1.76	1.52	0.35	4.29

Table 23. Predicted Urine Excretion Rates Corresponding to NIOSH's Derived GM Intake Rates of Type F Uranium Compounds, Assuming a Continuous Exposure as compared to a 5-day Per Week Exposure, with Monday Morning Sample Collections

Year (OTIB-0080 Intake rate)	50th percentile urine (dpm/d)	Predicted 24-Hour Urine (dpm) Assuming a Continuous Intake	Predicted 24-Hour Urine (dpm/d) Assuming Monday Morning Collection (dpm/d)	Underestimation of Intake*
1981 (5.666 dpm/d)	1.34	1.53	0.36	4.26
1982 (5.666 dpm/d)	0.22	1.53	0.36	4.23
1983 (5.666 dpm/d)	3.17	1.54	0.37	4.22
1984 (5.666 dpm/d)	N/A	1.54	0.37	4.18
1985 (5.666 dpm/d)	N/A	1.54	0.37	4.16
1986 (5.666 dpm/d)	2.05	1.55	0.37	4.17
1987 (5.666 dpm/d)	1.48	1.55	0.37	4.14
1988 (5.666 dpm/d)	1.44	1.55	0.38	4.12

* NIOSH-derived geometric mean intake rates should be multiplied by the values under the column "Underestimation of Intakes" in order to reproduce NIOSH's predicted bioassay results which were derived by NIOSH as the best fit to the 50th percentile bioassay results, given in Table A-2 and Figure A-13 of OTIB-0080 Rev. 0 (ORAUT 2014a).

Table 24. Predicted Urine Excretion Rates Corresponding to NIOSH's Derived GM Intake Rates of Type M Uranium Compounds, Assuming a Continuous Exposure as compared to a 5-day Per Week Exposure, with Monday Morning Sample Collections

Year (OTIB-0080 Intake rate)	50th percentile urine (dpm/d)	Predicted 24-Hour Urine (dpm) Assuming a Continuous Intake	Predicted 24-Hour Urine (dpm/d) Assuming Monday Morning Collection (dpm/d)	Underestimation of Intake*
1965 (17.6 dpm/d)	2.90	4.77	2.14	2.23
1966 (17.6 dpm/d)	4.52	5.03	2.33	2.16
1967 (17.6 dpm/d)	5.47	5.07	2.36	2.15
1968 (17.6 dpm/d)	6.22	5.09	2.37	2.15
1969 (8.91 dpm/d)	2.50	2.19	0.98	2.23
1970 (8.91 dpm/d)	3.41	2.31	1.07	2.16
1971 (8.91 dpm/d)	2.35	2.33	1.08	2.16
1972 (8.91 dpm/d)	1.68	2.34	1.09	2.15
1973 (18.87 dpm/d)	4.99	4.90	2.20	2.23
1974 (18.87 dpm/d)	5.95	5.17	2.39	2.16
1975 (18.87 dpm/d)	0.23	5.22	2.42	2.16
1976 (18.87 dpm/d)	5.77	5.23	2.44	2.14
1977 (18.87 dpm/d)	4.52	5.25	2.45	2.14
1978 (18.87 dpm/d)	4.89	5.26	2.45	2.14
1979 (18.87 dpm/d)	5.14	5.27	2.46	2.14
1980 (5.666 dpm/d)	1.76	1.38	0.62	2.23
1981 (5.666 dpm/d)	1.34	1.45	0.67	2.16
1982 (5.666 dpm/d)	0.22	1.47	0.68	2.16
1983 (5.666 dpm/d)	3.17	1.47	0.69	2.15
1984 (5.666 dpm/d)	N/A	1.47	0.69	2.14
1985 (5.666 dpm/d)	N/A	1.48	0.69	2.14
1986 (5.666 dpm/d)	2.05	1.48	0.69	2.14
1987 (5.666 dpm/d)	1.48	1.48	0.69	2.14
1988 (5.666 dpm/d)	1.44	1.48	0.70	2.13

* NIOSH-derived geometric mean intake rates should be multiplied by the values under the column "Underestimation of Intakes" in order to reproduce NIOSH's predicted bioassay results which were derived by NIOSH as the best fit to the 50th percentile bioassay results, given in Table A-2 and Figure A-13 (ORAUT 2014a).

Table 25. Predicted Urine Excretion Rates Corresponding to NIOSH's Derived Geometric Mean Intake Rates of Type S Uranium Compounds, Assuming a Continuous Exposure as compared to a 5-day Per Week Exposure, with Monday Morning Sample Collections

Year (OTIB-0080 Intake rate)	50th percentile urine (dpm/d)	Predicted 24-Hour Urine (dpm) Assuming a Continuous Intake	Predicted 24-Hour Urine (dpm/d) Assuming Monday Morning Collection (dpm/d)	Underestimation of Intake*
1965 (17.6 dpm/d)	2.90	2.83	1.21	2.34
1966 (17.6 dpm/d)	4.52	4.38	2.32	1.89
1967 (17.6 dpm/d)	5.47	5.50	3.11	1.77
1968 (17.6 dpm/d)	6.22	6.32	3.70	1.71
1969 (8.91 dpm/d)	2.50	1.33	0.567	2.35
1970 (8.91 dpm/d)	3.41	2.05	1.08	1.90
1971 (8.91 dpm/d)	2.35	2.57	1.46	1.76
1972 (8.91 dpm/d)	1.68	2.96	1.73	1.71
1973 (18.87 dpm/d)	4.99	2.25	0.97	2.33
1974 (18.87 dpm/d)	5.95	3.49	1.84	1.90
1975 (18.87 dpm/d)	0.23	4.37	2.48	1.76
1976 (18.87 dpm/d)	5.77	5.03	2.94	1.71
1977 (18.87 dpm/d)	4.52	5.52	3.30	1.67
1978 (18.87 dpm/d)	4.89	5.91	3.58	1.65
1979 (18.87 dpm/d)	5.14	6.23	3.80	1.64
1980 (5.666 dpm/d)	1.76	0.69	0.30	2.33
1981 (5.666 dpm/d)	1.34	1.07	0.57	1.89
1982 (5.666 dpm/d)	0.22	1.34	0.76	1.76
1983 (5.666 dpm/d)	3.17	1.54	0.91	1.70
1984 (5.666 dpm/d)	N/A	1.70	1.01	1.68
1985 (5.666 dpm/d)	N/A	1.82	1.10	1.65
1986 (5.666 dpm/d)	2.05	1.91	1.17	1.63
1987 (5.666 dpm/d)	1.48	1.99	1.22	1.63
1988 (5.666 dpm/d)	1.44	2.06	1.27	1.62

* NIOSH-derived geometric mean intake rates should be multiplied by the values under the column "Underestimation of Intakes" in order to reproduce NIOSH's predicted bioassay results which were derived by NIOSH as the best fit to the 50th percentile bioassay results, given in Table A-2 and Figure A-13 of OTIB-0080 Rev. 0 (ORAUT 2014a).

It is worthwhile to note that NIOSH uses a continuous intake to derive coworker intake rates. The continuous intake assumes 365 days chronic exposure without rest. In reality, the worker has a schedule that consists of periodic breaks that would be characteristic of a particular facility. The most common work schedule is a 5-day workweek followed by a 2-day break (weekend). The assumption of a chronic nonstop 365-day work period overestimates the worker's committed doses by a factor of 1.4, if in reality the worker has a 5-day workweek (7 days versus 5 days per week). In the case of Santa Susana, the committed doses calculated from the radioactivity in urine samples taken on Monday mornings after a 48-hour absence from work underestimates the dose for Type F and Type M uranium, even if doses were calculated using the 5-days per week schedule. It is claimant favorable to assume a continuous intake of 365 work days per year, as is assumed for all coworker-derived intakes and assign that value to the unmonitored worker. Thus, the uranium intake rates derived by NIOSH should be multiplied by an adjustment factor to account for the SSFL protocol of Monday morning sampling.

The higher underestimate is due to Type F compounds. SC&A calculated 10, 15, 20, 25, 30, 40, and 50 years committed equivalent doses for all uranium types (F, M, S), assuming workers were

exposed continuously to uranium during the periods 1965–1968, 1969–1972, 1973–1979, and 1980–1988, with each period of time being analyzed independently. For each period of time, the intake rates were the ones defined as GMs by NIOSH in Tables A-6, A-7 and A-8 of OTIB-0080. Type S is predominantly the most claimant-favorable type to be chosen, if the correction factors for Monday morning samples are not applied.

Finding 12: Uranium intake rates appear to be underestimated, based on the practice of collecting urinalysis samples on Monday morning after 2 days of no exposure. Although not all the available bioassay data was collected on Monday morning, NIOSH should apply an adjustment factor to the calculated intakes as a claimant-favorable and bounding assumption.

SC&A would also like to note that it is typically assumed that, for exempt employees, there are 2,000 work hours per year when determining final intakes; however, there are situations in which overtime may be taken into account. In the procedure for determining ambient dose, ORAUT-PROC-0060, Rev. 01 (ORAUT 2006b), the Maximizing Methodology stipulates that 2,600 hours per year be used, and the Best Estimate Methodology stipulates that 2,500 hours be used. The same exposure time should be used across all models when determining dose unless justification is provided.

Observation 10: It is important that assumptions that cross-cut among models (e.g., internal, external, ambient) are consistent. Unless there is justification to do otherwise, the length of time of exposure of coworkers should be assumed to be the same across all models. Because use of overtime is often a common practice among operations personnel, a 2,000 work-year may not bound the dose to a worker.

5.3 EXPOSURES TO URANIUM ALUMINIDE

Attachment A of the Occupational Internal Dose TBD for SSFL (ORAUT 2010) states the following concerning uranium aluminide ($U\text{-}Al_x$):

> *SSFL began fabricating reactor fuel elements in the fall of 1965. SSFL fabricated fuel for the ATR and the ETR from 1966 to 1968 at the De Soto Facility. In Room 1110-62 of Building 101 (known at the time as Building 001), briquettes of an alloy of 93% EU and aluminum known as uranium aluminide (UAl_x) were formed in an electric arc melting furnace. These briquettes were crushed to form a powder, which was cold-pressed into compacts that became the cores of the fabricated fuel plates. The room where these activities took place was known as the powder room.*
>
> *In 1967 after 15 mo of operation, urine bioassay data indicated that the material was probably insoluble and, therefore, that the air activity was not being compared to the appropriate MPC. Although the uranium was more than 93% ^{235}U by weight, ^{234}U accounted for more than 96% of its activity. The insoluble MPC for ^{234}U was a factor of 6 lower than the soluble MPC. This led to the conclusions that the regulatory standard (weekly average MPC) had been exceeded on a number of occasions and that equipment and procedures for controlling the airborne uranium were insufficient.*

> *An internal investigation determined that the primary reason for the ineffective confinement of the uranium was leakage from the crushing glovebox seal, from the fume hood of weighing station 2, and from the arc furnace. Temporary measures were put in place until more permanent controls could be implemented. These changes were put in place and the project was completed. Along with the engineering changes, workers were required to wear full-face respirators and lapel air samplers (Saxe 1967). At some point, operations were relocated to the new powder room.*
>
> *…*
>
> *A particle size study consisting of two general area air samples from the powder room indicated that the particles were less than 1-µm count median diameter (CMD) (Alexander 1967). The actual CMD and geometric standard deviation were not provided in the reference; therefore, the AMAD could not be determined. In the only other particle size reference found, Baurmash (1967) measured the particle size distribution for a UO_2 grinding operation in the "processing room in Bldg. 1." The CMD reported for this operation was 0.195 µm with a geometric standard deviation of 1.66. The mass median diameter was calculated as 0.42 µm. Using a density of 10.97 g/cm^3 for UO_2, the AMAD for this aerosol is 1.39 µm. While this information is extremely limited, it points out that small particle sizes were at least possible during uranium fuel fabrication. Therefore, dose reconstructors should consider a 1-µm AMAD particle size for these operations in addition to the default assumption.*

In Attachment H of ORAUT 2010, NIOSH adopts the proposal of Leggett, Eckerman and Boice (Leggett et al. 2005), in which mechanical clearance parameters and dissolution parameters of the ICRP 66 Human Respiratory Tract Model (ICRP 1994) are modified to fit the bioassay data collected from workers who were exposed to U-Al_x in the powder room. This proposal states that more than 60 persons worked in the powder room from 1966 to the early 1970s, when operations were moved to an improved facility.

SC&A was able to identify 38 workers that were exposed to U-Al_x in the powder room during 1966 and 1967 and who might have worked in exposure conditions above the permissible levels of the time. Many of those workers were restricted from work in areas subject to potential airborne radioactive material for a certain period of time (Atomics International 1967).

In ORAUT 2014a, NIOSH appears to include the uranium bioassay results from these identified workers that were exposed in the powder room. In Neton 2014b, NIOSH observes:

> *…at a number of sites different classes of workers during the same time period may have had monitoring programs that were conducted for different purposes. For example, construction and building trade workers, who worked intermittently in radiological areas, may have been monitored only when an incident was expected to have occurred, while those employees involved in routine process operations would have been routinely monitored on a frequency commensurate with their exposure potential. In this case, it would not be appropriate to combine the monitoring data for these two groups of workers into a single coworker model*

> *that assumes a chronic exposure pattern. Rather, the default in this case should be to consider separate coworker models.* (Neton 2014b)

In the case of the workers that were exposed to U-Al_x, the frequency of monitoring increased when workers were found contaminated. In addition, workers were exposed to uranium compounds with an AMAD of 1 µm instead of the generally assumed 5 µm AMAD. In addition, the compounds of U-Al_x handled in the powder room during this period showed a distinctly different excretion rate pattern than what one would expect for the classical solubility Types F, M, and S uranium compounds. As explained by Leggett et al. 2005, several months after the start of the U-Al_x fuel fabrication program, it became evident from available monitoring data that the behavior of inhaled U-Al_x differed from that of other forms of uranium that had been handled at this facility, and intake and lung retention were being underestimated based on the standard models and assumptions. For example, it was observed that the urinary excretion rate actually began to increase after workers ceased their internal exposure potential to radioactive materials. In these workers who had been removed from exposure, the rate of urinary excretion of uranium would continue to rise over a period of months before reaching a peak excretion rate, which was followed by a sharp decline. This pattern of excretion rate was observed by SC&A in the coworker data for workers who were listed as being restricted from work in the Powder Building (Atomics International 1967).

ORAUT 2014a uranium intake rates were aggregated during these periods: 1965–1968, 1969–1972, 1973–1979, and 1980–1988. The 1965–1968 period of time is not appropriate for aggregating the results from workers exposed to uranium aluminide, as many high uranium excretion results were obtained in a period of time when the workers were restricted from work in areas subject to potential airborne radioactive material. Additionally, "less than detectable" or "zero" samples may have been included in the coworker model, which would not accurately reflect exposure to U-Al_x, owing to the non-monotonic nature of the associated excretion pattern.

Finding 13: NIOSH should discuss how uranium samples taken from workers in the U-Al_x powder operations are interpreted in the coworker analysis, and whether a separate assessment of coworker doses in the Powder Building may be warranted. In the latter case, it would be necessary to establish with sufficient accuracy which workers were potentially exposed to uranium aluminide in the Powder Building during the relevant period.

5.4 OCCUPATIONS MONITORED FOR URANIUM INTAKE

Occupational information was analyzed by cross-referencing dose records that specify an analytical method for detecting uranium with occupational information developed as described in Section 2.0. From Section 2.0, SC&A determined that the monitoring program appeared to target the workers most at risk (technical staff and mechanics) while also covering a cross-section of other occupations that could also be exposed. Table 26 shows the number and percentage of workers in each occupation monitored for uranium intakes compared to the total number of workers monitored for uranium.

Based on Table 26, it appears that monitoring for uranium among the different occupations mirrors the overall site data presented in Section 1.0, with the Mechanics (38%) and Technical Staff (27%) receiving the most uranium monitoring while other, but not all, occupations were

monitored. These two occupation classes were monitored throughout SSFL's operational history for uranium intakes, except there are no data available from the early-to-mid 1980s, as was noted in Finding 11. Other occupations were intermittently monitored as would be expected.

Table 26. Occupations Monitored for Uranium Intakes at SSFL[a]

Occupational Group	Number of Individuals[b]	Years Monitoring Data Available	Percentage of Individuals in an Occupational Group
Construction	[redacted]	1965–1966, 1968–1970	1%
Electrician	[redacted]	1965, 1971–1981, 1983	3%
Fireman	[redacted]	1965–1970	2%
Health and Safety/ Health Physics[c]	11	1965–1976, 1978–1980, 1983, 1988	5%
Inspector	[redacted]	1975–1977, 1981, 1988	1%
Janitor	11	1965–1983	5%
Machinist	[redacted]	1965–1968, 1970–1976, 1978, 1980–1983	3%
Maintenance[d]	14	1965–1976, 1978–1983	6%
Management	[redacted]	1965–1983	3%
Mechanic[e]	62	1965–1982, 1986–1989	26%
Office Personnel	11	1965–1968, 1970, 1975–1980, 1982–1983, 1985–1988	5%
Operator	0	1965–1976	0%
Quality Assurance	[redacted]	1970–1981, 1983	<1%
Radiographer	[redacted]	1975–1978, 1981–1982	<1%
Reactor Operator	[redacted]	1965–1967, 1970, 1983	2%
Security	[redacted]	1965–1970, 1986	3%
Shipping	[redacted]	1965, 1966, 1968	<1%
Technical Staff[f]	63	1965–1982, 1986–1988	27%
Technician	[redacted]	1967–1968, 1987 - 1988	1%
Truck Driver	[redacted]	1973–1976, 1979, 1982	<1%
Unknown[g]	[redacted]	1965–1966, 1971–1974, 1986–1988	2%
Welder	[redacted]	1965–1969, 1980–1982	1%
Total	235	-	95%[h]

[a] Excludes dose records prior to 1965 and between January 1991 and mid-June 1993.
[b] An individual may be represented more than once if their occupation changed over the course of their monitoring.
[c] Includes medical staff, and health physicists and their support staff
[d] Includes individuals with 'maintenance' as part of their job title, utility services staff, and painters
[e] Includes individuals with 'mechanic' somewhere in their job title, and those who describe themselves as reactor and reactor fuel assemblers/disassemblers.
[f] Includes individuals with 'engineer' as part of their job title, physicists, chemists, and analysts
[g] Either a job title was not specified and marked as unknown, or it is unclear what the job title represents (for example, 'Y&S', 'FMS').
[h] Total percentage does not equal 100% due to rounding.

One exposure aspect not discussed in the internal coworker model is modeling for incidents. In the case of uranium, uranium fires were not uncommon occurrences. In a review of the SRDB, SC&A found 53 incidents involving uranium fires. It must be noted that most, but not all, of these documented fires occurred prior to 1965 and only two of the incidents appeared to result in bioassays being performed (post-1965). Nonetheless, incidents involving uranium fires in the

period of interest should be addressed. A summary of the incidents can be found in Attachment 2.

Finding 14: The internal coworker model should specify the types of incidents that are assumed to be covered by the model, and when incidents would be expected to be reviewed separately with an incident-specific model.

6.0 EVALUATION OF NEED FOR MODELING OTHER CONSTITUENTS

The internal dosimetry model addresses uranium, plutonium, and mixed fission products, as analyzed in the previous sections of this report, but is silent regarding other radionuclides potentially inhaled and/or ingested by workers. These radionuclides include isotopes of thorium, cesium, strontium, americium, polonium, and cobalt, as well as tritium. All but one of these radionuclides cannot be detected using the monitoring methods for uranium, plutonium, or mixed fission products.

A number of research and production activities are associated with thorium and tritium. Cesium and strontium are mixed fission products that would be important contributors to dose. Some machining of cobalt may have occurred. While exposures to cobalt generated as a fission product from a reactor would normally be monitored using analytical methods associated with mixed fission products, it is possible that this monitoring approach was not used for the unique activity of machining unsealed cobalt, and it may be necessary to look at its isotopes separately when modeling exposures to workers found to have performed these activities.

Bioassays for all of these radionuclides were specifically performed at some point during SSFL operations and cleanup, which may lend credence to their importance in determining radiation dose to workers.

Isotopes associated with cesium, strontium, europium, and promethium can generally be estimated using activity ratios described in ORAUT-OTIB-0054 Rev. 02 (ORAUT 2014b) and do not need to be separately addressed in dose reconstructions. However, no mention is made of this in ORAUT-OTIB-0080. NIOSH should additionally demonstrate the feasibility of reconstructing exposures to thorium, tritium, and americium.

Finding 15: The internal dose coworker model is incomplete in that it does not address other radionuclides that were present on site and that could have substantively contributed to worker intakes. NIOSH should ascertain their ability to reconstruct intakes of thorium, tritium, americium, cesium, strontium, polonium, and cobalt.

More detailed information is provided below regarding how each of these isotopes was used or was present at SSFL, and to what extent, based on the available dose records. Information was taken from the site profile unless otherwise referenced.

Thorium

Facility Operations

The Sodium Reactor Experiment (SRE) reactor, which operated from 1957 to 1964, used two cores over its lifetime, the second core containing thorium. The thorium fuel was manufactured in the Engineering Test Building (Building 4003). The operation consisted of loading of uranium and thorium metal slugs into metal tubes, filling the interstices with sodium metal, and

sealing the tubes (Rucker 2009, ORAUT 2006a). The fuel loaded in 1960 was 7.6% (weight) Th-232 with 93% enriched uranium (EU) (ORAUT 2010).

The other reactor that used fuel containing thorium was the Advanced Epithermal Thorium Reactor (AETR), which operated from 1960 to 1974. The AETR was built to study and test reactor core configurations for thorium- and uranium-fuel reactors. The AETR facility was expanded in 1961 to include the Fast Critical Experiment Lab (Building 4100) for epithermal and fast neutron criticality tests, in which 20 different reactor core configurations were studied. The AETR's first 9 core configurations (through 1965) contained various amounts of U-233 and Th-232 and were driven by 93% EU fuel. The AETR and Fast Critical Experiment Lab supported the development of reactors for the Southwest Atomic Power Association. For both the SRE and AETR, the thorium isotopes present in the fuel would have consisted mainly of Th-232 and its decay progeny Th-228 in partial equilibrium. Thorium-232 is also an activation product that was generated from operation of the reactors at SSFL (Rucker 2009, ORAUT 2006a, ORAUT 2010).

Research activities were conducted at SSFL regarding the reprocessing of used nuclear fuel. Rockwell International developed a process to effect a partial separation of used fuel, removing part of the mixed fission products, so that the material could be used again as reactor fuel. Tests were performed in a well-shielded Hot Cave in the Engineering Test Building. These experiments used up to one-kilogram quantities of un-irradiated uranium and thorium, and up to 100-gram quantities of highly irradiated materials (Rucker 2009).

In 1979, about 540 thorium plates containing about 200 kg of thorium were chamfered, removing an estimated total of 150 grams of thorium using a surface grinder (Tuttle 1979a; Tuttle 1979b). An enclosure around the head of the grinder and a chip catcher were used to capture the release of material ground off the plates. A vacuum cleaner with HEPA filter was attached to the enclosure (Tuttle 1979a). However, an adequate air flow could not be maintained and wet grinding was used to complete the chamfering (Begley 1979a). This operation was conducted in Building 001. The machinist was required to wear rubber gloves, shoe covers, and coveralls, and was required to undergo urinalysis testing to monitor internal exposure to thorium (Tuttle 1979a). The machinist was W.C. Woods, who was assisted by G. Iskiyan. The supervisor was D.W. Hebdon. The work was conducted from February 28 to March 12, 1979 (Begley 1979b).

Exposures to Thorium

No facility-specific or individual-specific information was found in the SRDB regarding the amount of exposure received by workers to thorium. In the unscreened electronic dose records database, there are 49 records associated with specific nuclide analyses for thorium, all from 1970, 1971, and the 1990s. When matched with occupational data, it is found that "mechanics" was the only occupation class monitored for thorium. The machinist that performed the grinding operation summarized above and the supervisor are in the electronic dose record database, but not for the year the grinding operation was performed. The assistant is not in the database.

Incidents Related to Thorium

No incidents involving thorium were found in the SRDB or the DOE's Occupational Reporting and Processing System, which contains incident reports since 1990.

Tritium

A safety review report was issued in 1992 that summarizes the uses and generation of tritium at SSFL (Tuttle 1992). Small amounts of tritium were formed by the reactors. Most of the tritium generated by the SSFL reactors remained bound within the reactor fuel because most of the fuel was metallic, or within the coolant; this is especially true of the SRE, where the coolant was a sodium-potassium compound. The tritium found in the groundwater at SSFL was most likely generated due to the interaction of neutrons in the reactor biological shielding (concrete) and with naturally occurring lithium within the molecular structure of the shielding. Tritium would then leach from the shielding into the surrounding groundwater.

There were five other activities where tritium was used.

1. Several thousand curies of tritium were used at the Hot Lab from 1964 to 1967 to investigate the properties of uranium-zirconium hydride systems. This tritium would have been used in small quantities and released through the Hot Lab exhaust system as a gas and not as tritiated water.
2. Titanium and zirconium targets containing 1 to 120 Ci of tritium were used in a variety of neutron generators to produce high-energy neutrons for research and development projects. Small amounts of tritium outgas during use of these targets were to be trapped by oil in vacuum pumps or discharged to the atmosphere.
3. Gas chromatographs were used for extremely sensitive analysis of organic compounds in water and other materials. An important part of the instrument is the detector, which, in some models, uses a titanium-tritium foil of roughly 0.2 Ci as a source of electrons. Essentially no tritium is released from these foils.
4. A high-powered laser using a deuterium-fluorine reaction was operated at the Laser Experimental Test Facility (LETF) in Area I. The deuterium gas, furnished by Oak Ridge National Laboratory (ORNL), contained a small amount of tritium as an impurity, and this was released to the atmosphere during operation of the laser.
5. Based on the SSFL Site Profile Site Description (ORAUT 2006a), reactor-generated tritium targets were bombarded by a Van de Graaff generator to produce neutrons. These accelerators were removed in 1962.

There are 429 records in the unscreened electronic dose database where tritium was specifically analyzed. Health and safety personnel, mechanics, and technical staff were occupations that were monitored for tritium.

Americium

Small quantities of americium (4 grams) were used as part of the Transuranic Management by Propartitioning-Separation (TRUMP-S) Program. This material was stored in the Fuel Storage Facility and was ultimately shipped offsite. Americium was generated as a byproduct of the use of plutonium at SSFL and would be a potential source of exposure in conjunction with plutonium.

There are 24 records in the unscreened electronic dose database where americium was specifically analyzed from the late 1960s through 1993. Health and safety personnel, mechanics, an electrician and an inspector were occupations that were monitored for americium.

Cesium

Cesium was generated by the SSFL reactors and criticality test facilities as a fission product. The Gamma Irradiation Facility (GIF) used Cs-137 sources for radiation hardening tests of electrical components and food irradiation research. Cesium was present in nuclear support facilities like fuel disassembly and examination facilities, such as the Hot Lab and the Radioactive Materials Disposal Facility (RMDF), and waste management facilities. Cesium was also an environmental contaminant at the SSFL in the burn pit and the RMDF, and two small areas of contamination (Rockwell 1996) were found and remediated near the Fuel Storage Facility (referred to as T064 in the reference).

There are 26 records in the unscreened electronic dose database where cesium was specifically analyzed, all in the 1990s. Mechanics was the primary occupation monitored for cesium, although one construction worker was also monitored.

Strontium

Strontium was generated by the SSFL reactors and criticality test facilities as a fission product. Strontium was present in nuclear support facilities, such as fuel disassembly and examination facilities and waste management facilities. Strontium-90 sealed sources were used at SSFL. Strontium contamination was found at the RMDF and it was a primary radiological contaminant at the burn pit.

There are 68 records in the unscreened electronic dose database where strontium was specifically analyzed, from the 1960s through the 1990s. Mechanics was the primary occupation monitored for strontium, although one construction worker was also monitored, which is consistent with cesium as would be expected.

Polonium

A Polonium-Beryllium neutron source was used at SSFL (Heine 1970). The prevalence of polonium at SSFL, however, would have been primarily because it is in the uranium-238 decay chain. There are five records in the unscreened electronic dose database from 1966 for one worker where polonium was specifically analyzed. The worker's occupation is unknown.

Cobalt

Cobalt was not listed in SC&A's comments in Issue 10 of the September 14 issues matrix update. Cobalt is a fission product from reactor operations that would have been monitored as such. Cobalt did have specific uses at SSFL. Cobalt-60 sources were used in the GIF for material and food irradiation research (ORAUT 2006a). In addition, "the Hot Lab was also used for manufacturing sealed sources, for leak checks on sources, and for cutting and machining operations that involved cobalt-60" (ORAUT 2009). It is not known whether the workers in the Hot Lab would have been monitored specifically for cobalt-60 during these cutting and machining operations or if they would have generally been monitored for mixed fission products. There are 18 records in the unscreened electronic dose database from 1986 and 1993 where cobalt was specifically analyzed. The primary occupation monitored was mechanics.

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ATTACHMENT 1: BREAKDOWN OF INTERNAL MONITORING ENTRIES CONTAINED IN THE "NUCLIDE/ANALYSIS" COLUMN OF THE COWORKER DATABASE

Nuclide/Analysis Entry	Total Entries (% of Total)	Nuclide/Analysis Entry	Total Entries (% of Total)	Nuclide/Analysis Entry	Total Entries (% of Total)
UR	12,883 (34.519%)	Tot U	11 (0.029%)	alpha, beta	3 (0.008%)
UF	7,593 (20.345%)	U-Tot	11 (0.029%)	GB Beta	3 (0.008%)
FP	6,004 (16.087%)	U-238	9 (0.024%)	Iso U	3 (0.008%)
MFP	4,735 (12.687%)	Ca-45	8 (0.021%)	MFP(G)	3 (0.008%)
Blank or "XX"	1,322 (3.542%)	UF	8 (0.021%)	alpha, beta, gamma	2 (0.005%)
GA	880 (2.358%)	Ba-140	7 (0.019%)	B	2 (0.005%)
GB	745 (1.996%)	beta gamma	7 (0.019%)	GB	2 (0.005%)
PUA	644 (1.726%)	Ce-141	7 (0.019%)	HG	2 (0.005%)
PU	534 (1.431%)	Ce-144	7 (0.019%)	M	2 (0.005%)
H-3	429 (1.149%)	Cs-134	7 (0.019%)	UFP	2 (0.005%)
GB(H)	167 (0.447%)	Fe-59	7 (0.019%)	GB Beta	1 (0.003%)
Alpha	148 (0.397%)	Gamma	7 (0.019%)	EU	1 (0.003%)
Beta-gamma	121 (0.324%)	GB	7 (0.019%)	FP-3AG-Co-60	1 (0.003%)
GB(L)	106 (0.284%)	Gross beta (L)	7 (0.019%)	G	1 (0.003%)
Beta	103 (0.276%)	Mn-54	7 (0.019%)	GAL	1 (0.003%)
AP	100 (0.268%)	Ra-226	7 (0.019%)	gamma, beta	1 (0.003%)
Gamma	88 (0.236%)	Ru-103	7 (0.019%)	GAZ	1 (0.003%)
Sr-90	49 (0.131%)	Ru-106	7 (0.019%)	GAZA	1 (0.003%)
U	32 (0.086%)	Zn-65	7 (0.019%)	GI	1 (0.003%)
U-ISO	32 (0.086%)	Co-58	6 (0.016%)	GR-B-K-39	1 (0.003%)
Th	28 (0.075%)	GR-B-K-40	6 (0.016%)	GR-B-K40	1 (0.003%)
Am-241	24 (0.064%)	Zr-95	6 (0.016%)	Gross -Beta	1 (0.003%)
MFPG	23 (0.062%)	Cm-242	5 (0.013%)	K	1 (0.003%)
U-total	22 (0.059%)	Cm-244	5 (0.013%)	MFP	1 (0.003%)
SR-90 (D)	21 (0.056%)	Gross alpha	5 (0.013%)	MFP,UR	1 (0.003%)
Cs-137	20 (0.054%)	LowBeta	5 (0.013%)	NDA	1 (0.003%)
MFPB	20 (0.054%)	MFP(B)	5 (0.013%)	P	1 (0.003%)
Pm-147	18 (0.048%)	Midbeta	5 (0.013%)	PA	1 (0.003%)
CFP	16 (0.043%)	Np-237	5 (0.013%)	Pu UF	1 (0.003%)
GRB(-K40)	14 (0.038%)	Po	5 (0.013%)	Pu/MFP	1 (0.003%)
U-234	13 (0.035%)	Pu-242	5 (0.013%)	Pu238/239	1 (0.003%)
Co-60	12 (0.032%)	Th-230	5 (0.013%)	Pu-238/239	1 (0.003%)
I-131	12 (0.032%)	Th-232	5 (0.013%)	Pu-238-239	1 (0.003%)
Th-228	12 (0.032%)	Topbeta	5 (0.013%)	Sr	1 (0.003%)
U-235	12 (0.032%)	Be-7	4 (0.011%)	Sr 90	1 (0.003%)
Ac	11 (0.029%)	GB-K40	4 (0.011%)	Tri	1 (0.003%)
K-40	11 (0.029%)	GB-K-40	4 (0.011%)	U, GB	1 (0.003%)
Pu-238	11 (0.029%)	Pu-23	4 (0.011%)	UR	1 (0.003%)
Pu-239	11 (0.029%)	UISO	4 (0.011%)	UTOT	1 (0.003%)

ATTACHMENT 2: LIST OF URANIUM FIRE INCIDENTS

The following table describes instances of uranium fires that have occurred at SSFL, as found in the SRDB.

SRDB Number	Date	Number of people involved	Location	Number of Bioassays post-1964	Description
39492	6/30/1960		Room 426-13	-	1 gallon can containing uranium turnings immersed in water-soluble machining coolant.
39493	1/10/1978	[redacted]	Building 028	-	Small uranium fire. Small amount of contamination on lip of arc furnace measuring 60 dpm/100 cm^2 alpha. Full face respiratory was being worn.
39494	10/26/1982	[redacted]	ATR area, Room 119-36	-	UALx fire, 1888 grams U-235. Powder ignited as it was being poured into a blender in a glovebox. No personal contamination survey performed.
39496	11/2/1960	[redacted]	001, Room 1155	-	Magnesium flashes while opening induction furnace. 100 kg of 1.945 enriched uranium, daughter products, and oxides. Respirators worn. Bioassays neg. 8.37E-10 µCi/cc beta gamma air sample.
39497	11/30/1960	[redacted]	Pluto area #1244	-	279 grams 93% UX in hydrogen sintering furnace. No abnormal air contamination. Operator smeared. No bioassay.
39498	12/3/1966		HQ1, 111D-62	-	File contains air sampling results
39499	6/20/1960	[redacted]	HQ1, 1063	-	2,300 grams of 10% enriched uranium not handled correctly, creating U_3O_8 which was put in a can. Hydrogen buildup blew up can. No bioassays.
39500	12/19/1963	[redacted]	Building 4, 424-68	-	40-50 grams of uranium carbide involved with 30 grams being recovered. Smears on 3 personnel showed no contamination.
39501	11/2/1961	793-1416, 793-226, 793-413	Fed area, Room 1026	-	1 gram of UO_2 powder. Personnel contaminated with 2 receiving bioassays. 500 dpm alpha and 1200 dpm beta-gamma contamination.
39502	3/20/1962	788-10 752-[redacted] 779-28	FEPF, 1253	-	3 16% enriched U-Mo Hallam slugs. 156 grams UX, 4,300 grams U. Surface contamination 500 dpm alpha, 1533 dpm beta+F12-gamma.
39503	8/21/1951		Downey Room A-126	-	Uranium chips caught fire
39504	12/10/1966		HQ, Bldg 004	-	Two fires. One was a mixture of Na and Cs-137. Other was uranium metal chips. No significant inhalation exposures.

SRDB Number	Date	Number of people involved	Location	Number of Bioassays post-1964	Description
39505	2/13/1967		UCPFF	-	Small amount of UC in a metal exhaust duct ignited. No unfiltered release occurred.
39506	1/30/1967		UCPFF	-	Uranium fire in retention tanks of vacuum system. Ducts burned through but no large scale release and no personnel exposure. Nasal smears taken of 15 men. 1 fireman had contaminated clothing.
39507	1/27/1967		Aisle 1110 and Room 110-65	-	JAERI fuel (20% uranium). No internal exposure.
39508	3/10/1967			-	Uranium metal chips ignited from sparks from cutting metal duct. Fireman and maintenance man scheduled for bioassays due to contamination on face and hair.
39509	5/21/1967		On truck	-	Explosion in a 55-gal drum containing 1-gal cans of UC sludge. Vermiculite and uranium contamination 900 to 5,000 dpm/ 100 cm^2 beta.
39510	5/21/1967		RMDF storage yard	-	55-gal drum containing uranium metal under $CaCO_3$ powder discovered burning. Air samples, nasal swabs and contamination surveys revealed no significant release or contamination.
39511	6/13/1970	[redacted]	HQ001, Room 119-42	5	Oven drying of 20 lbs of uranium chips that caught fire. Floor contaminated <50 dpm/100cm^2 alpha and 30 dpm/100cm^2 beta. Airborne was 1.6E-10 µCi/cc. Bioassays taken. All beta-gamma below dpm/100cm^2 of nasal, clothing, skin. Highest alpha was 15 dpm/100cm^2 nasal, 20 dpm/100cm^2 clothing, 28 dpm/100cm^2 skin.
39512	8/26/1976	[redacted]	SNM Vault weigh room	5	Sealed transfer can of uranium fines was dropped, causing autoignition. Floor contaminated with 200 dpm/100cm^2 alpha. One person suffered finger cut with 40 dpm alpha which was deconned. Nasal smears indicated max of 11 dpm alpha.
39513	4/27/1977	[redacted]	ATR-QA lab	-	Fire in lab. Substances burned not specified. All nasal, hands, shoes, socks below 20 dpm alpha. No internal or external exposures.
39514	1/9/1978	[redacted] and several others	ATR Powder Room	-	UALx fire in crusher glove box. All smears on glove box <20 dpm alpha.
39515	9/14/1965	[redacted]	Bldg 1, SS Vault Pad	-	Fire in 1-gal can of DU from ESADA Project. No contamination found and nasal smears negative.

SRDB Number	Date	Number of people involved	Location	Number of Bioassays post-1964	Description
39516	6/12/1965	[redacted]	FED II filter plenum	-	Involved uranium carbide. Significant contamination near filter bank. Minor personnel contamination noted. One individual had up to 25 dpm alpha nasal. Up to 155 dpm alpha hair, up to 675 dpm alpha trousers, up to 300 dpm alpha skin, up to 390 dpm alpha shirt, up to 500 dpm alpha shoes.
39517	6/26/1964	[redacted]	Canoga 1, Room 1126	-	4.91% enriched UC cuttings dispersed during vacuuming. Five people contaminated. Up to 64 dpm alpha and 22 dpm beta-gamma nasal, 50 dpm alpha and 25 dpm beta-gamma body (face, hair, hands, arms).
39518	6/24/1964	[redacted]	Building 1, Room 2021	-	1 gram UC on fire in desiccator. One individual had 4 dpm alpha and 15 dpm beta-gamma nasal. Shoes had 30 dpm alpha.
39519	2/13/1964	[redacted]	Bldg 1, Room 1126	-	Fire in vacuum cleaner with unknown amount of 4.9% enriched UC. One individual received first aid. Breathing apparatus worn. Air samples indicated 1.8E-11 µCi/cc alpha and 8.4e-10 µCi/cc beta-gamma. Floor smears showed 1,000 dpm alpha and 1,400 dpm beta-gamma.
39520	9/10/1959	[redacted]	Room 807	-	Two one-gal cans of U-Nor/Zirconium chips. 12 dpm beta-gamma on neck.
39521	1/6/1964	[redacted]	Bldg 1, Room 1126	-	Fire in vacuum cleaner with unknown amount of UC. All received bioassays. Breathing apparatus worn. Air samples indicated 1.8E-11 µCi/cc alpha and 1.0e-10 µCi/cc beta-gamma. Floor smears showed 400 dpm alpha and 1000 dpm beta-gamma.
39522	3/4/1964		HQ-1, Rooms 1131, 1114	-	UC fire in ducting between rooms. Breathing apparatus worn. Air samples indicated 6.8E-11 µCi/cc alpha and 2.8e-10 µCi/cc beta-gamma. Floor smears showed 1,000 dpm alpha and 2,300 dpm beta-gamma.
39523	8/31/1959	[redacted]	Room 901	-	UC slug ignited in desicator.
59524	3/15/1961	[redacted]	HQ-1, Room 1059	-	Up to 900 grams UC, 95.1% uranium, 3.02% enriched. Burned rubber pants.
59525	3/16/1961	[redacted]	Room 1059, Cresta grinder	-	UC grindings. No increase in air activity.
59526	6/23/1961	[redacted]	Room 1059	-	13 grams UC pellet caught fire.
59527	7/8/1967		UCPFF	-	Peeled paint on retention tank indicated a fire occurred at one time.

SRDB Number	Date	Number of people involved	Location	Number of Bioassays post-1964	Description
39544	5/5/1961	[redacted]	SS Vault, Room 1142	-	3.25% enriched UC. Explosion in can while being held. Dropped it and fire came out of can. Up to 18 dpm alpha and 18 dpm beta-gamma on nose, 2 dpm alpha and 30 dpm beta-gamma face, 2 dpm alpha and 55 dpm beta-gamma on neck, and 12 dpm alpha in hair.
39545	7/29/1964	[redacted]	Room 1131, Furnace 3	-	Implosion of furnace. Up to 32 dpm alpha face and arms.
39546	2/22/1961	[redacted]	Room 1059	-	UC fire in grinding operation.
39547	6/28/1976	[redacted], including [redacted]	ATR Powder Room	-	3 kg of U-AL_x briquet dropped. All received nasal swipes, with one receiving 20 dpm/100cm^2. Up to 4,000 dpm/100cm^2 alpha found on clothing.
39548	7/5/1961	[redacted]	Building 009, SanSu	-	Two 1-gal containers containing 30 UC slugs. No personnel contamination.
39549	8/26/1964	[redacted]	Room 1059	-	Small amount of normal and enriched UC caught fire in Cresta grinder.
39574	9/23/1961	[redacted]	SNAP, room 1273	-	Fire while machining uranium on a lathe.
39575	8/25/1964	[redacted]	Room 1273	-	UC zirconium and UX material ignited. Smears and nasal smears all showed background.
39598	5/13/1965	[redacted]	Bldg 022 AMF evaporator	-	Overflow of chemically treated contaminated liquid. Received splashes to hard hat, lab coat, and goggles. No contamination found.
39599	8/21/1963	[redacted]	SNAP, Room 1252	-	U-Zr fire - sandblasting unit. Breathing apparatus worn. Surface contamination up to 3,500 dpm alpha and 4,200 dpm beta-gamma. Room 1256 also contaminated. Fireman's pants confiscated. Nasal smears positive.
39609	8/9/1962	[redacted]	Room 1261	-	Degreaser overheated causing fire and TCE fumes. 2 firemen responded, using self-contained breathing units.
39617	8/20/1959		Room 807	-	U-Zr shavings ignited.
39621	9/23/1964	[redacted]	HQ-1, Room 1114	-	Lid on a one-quart paint can containing 5,366 grams of 4.91% enriched UC vacuum dust blew off. No personnel contamination. Surface smears showed 2,000 dpm/100cm^2 alpha and 2,400 dpm beta-gamma.
39622	2/27/1963	[redacted]	Room 1059	-	Fire in R/A exhaust system. Air concentration of 3.4E-11 alpha and 5.0E-11 beta-gamma.

SRDB Number	Date	Number of people involved	Location	Number of Bioassays post-1964	Description
39623	10/1/1964	[redacted]	HQ-1, Room 1126	-	Small amount of UC burned. Worker was not wearing shoe covers or lab coat. Shoes had 30 dpm alpha, pants 45 dpm alpha and 90 dpm beta-gamma.
39624	1/12/1961	[redacted]	Bldg. 4, Room 423-29	-	3% enriched UC ignited while being welded. Nasal smear showed no significant contamination. Lab coat caught fire and was used to smother fire. Surface smears showed 300 dpm alpha and 500 dpm beta-gamma.
39625	10/3/1961	[redacted]	Room 1063	-	1 pint of Hexane solution in a bread pan ignited from a spark created by striking 2 UC slugs against one another.

ATTACHMENT 3: OCCUPATIONAL CATEGORIZATION

The following table shows how occupational names were condensed down to a more generalized list.

Construction			
Construction	Carpenter		
Mechanic			
Mechanic	Air Refrigeration Mechanic	Instrument Mechanic	Instrumentation Mechanic
Lab mechanic/reactor assembly	Maintenance Mechanic	Mechanic AER	Mechanic AER Lab
Mechanic/Nuclear Materials Worker	Mechanic/Reactor Operator	Reactor Assembly	Reactor Assemblyman
Reactor Fuel Disassembler	Structure Mechanic		
Technical Staff			
Analyst-Production	Chemist	Physicist	Associate, Test Development
Engineer	Engineering Lab Assistant	Engineering Technician	Lab Assistant
Metallurgical Engineer	Research Analyst	Sr Technical Specialist	Tech for Nuclear Research
Test Engineer	Eng. Office Personnel		
Health and Safety/Health Physics			
Associate Health Physicist	Health and Safety Rep	Health Analyst	Health Physicist
Health Rep	Nurse		
Office Personnel			
Office Personnel	Clerk		
Operator			
Chemical Operator	Control Center Operator	Sr. Production Control	
Electrician			
Electrician	Maintenance Electrician		
Management			
Manager	Management Analyst	Manager Research	Manufacturing Liaison
Quality Assurance			
Manager – QA	QA		
Machinist			
Milling Machine Machinist	Lathe Machinist	Pipe Shop	
Maintenance			
MTS III, IV, V	Ops/Maintenance Technician	Plant	Maintenance Painter
Utility Man – Machine Shop	Utility Man – General	Plumber	
Shipping			
Ore Shipping			
Inspector			
Inspector	Reactor Assembly Inspector	ISI Inspector	
Janitor			
Janitor	Custodian		
Radiography			
X-ray Tech			