NIOSH Responses to SC&A Review of ORAUT-TKBS-0041, Rev. 02: *Site Profile for Nuclear Materials and Equipment Corporation, Apollo and Parks Township, Pennsylvania*

Response Paper
Rev. 0-A

National Institute for Occupational Safety and Health

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OVERVIEW

In April of 2013, S. Cohen & Associates (SC&A) completed a review of the NIOSH Site Profile for Nuclear Materials and Equipment Corporation (NUMEC) Parks Township and Apollo Sites (ORAUT 2012a) at the request of the Advisory Board on Radiation and Worker Health (Advisory Board). Twenty one findings were reported in the SC&A report. Responses to these findings are provided in this response paper.

NIOSH RESPONSES TO SC&A PRIMARY FINDINGS

The following presents SC&A's primary findings for the NUMEC Apollo and Parks Township site profile, ORAUT-TKBS-0041 (ORAUT 2012a) and NIOSH responses.

FINDING 1: CLARIFICATION IS NEEDED ABOUT THE START AND END DATES OF PARKS TOWNSHIP SITE OPERATIONS.

NIOSH Response: The dates provided in the site profile have been reviewed and updates will be made in the next revision of the Technical Basis Document (TBD). The following table indicates the proposed revisions (changes underlined).

Table 2-3. Parks Township site area descriptions.

Building or area	Operations	Radionuclides	Operation period
Building A–Plutonium processing facility	Fabrication of plutonium reactor fuel pellets, blankets, rods	PuO_2; Pu nitrate and oxalates (AmBe, PuBe, 1959–1970); alpha, beta, and gamma sources	1961–1980
A–Fab 1	Plutonium conversion	Plutonium nitrate, plutonium oxide, depleted UO_2	1961–1980
	Fuel fabrication for FFTF	Plutonium nitrate, plutonium oxide, depleted UO_2	1972–1980
A–East Side of Fab 1	Routine repair and maintenance of contaminated equipment	All	1961–1980
A–Fab 2	Fuel fabrication for ZPR-III	Plutonium nitrate, plutonium oxide, depleted UO_2	1964–1966
	Fuel fabrication for ZPPR	Plutonium nitrate, plutonium oxide, depleted UO_2	1966-1970
A–Fab 3	Manufacturing operations, metallography	Plutonium nitrate, plutonium oxide	1963–1980
	Quality control of FFTF fuel	Plutonium nitrate, plutonium oxide	1972–1980
A–Fab 4	Alpha, beta, gamma, and neutron source fabrication	AmBe double encapsulated, PuBe compacted powder, Ir-192, Cs-137, Be-7, Po-210, Co-60, Am-241, Pu-238, Pu-239, PuO_2, plutonium and americium metal	1963–1980

Building or area	Operations	Radionuclides	Operation period
A–Fab 5	Scrap recovery	Plutonium – various forms	1963–08/01/1967
	Analytical laboratory work	All, small quantities of radioactive samples	1979–1980
A–Fab 6	Scrap recovery	Plutonium – various forms	1968–1980
A–Fab 7	Fuel rod quality control tests, nonradioactive processes	All, clean and contaminated items	1968–1980
A–Fab 8	Storage	All, clean and contaminated items	1970–1980
A–Fab 9	FFTF fuel pin finishing	Encapsulated nuclear material	1970–1980
Building A–Hot Cell Room	Examination of irradiated samples, high-activity source fabrication	PuBe compacted powder, Co-60, Ir-192	1961–1969
	Storage of sources	Sealed sources, Ir-192, Cs-137, Be-7, Po-210, Co-60, Am-241, Pu-238, Pu-239, PuO_2,	1969–1980
Plutonium Facility Trailer Storage Area	Storage of large quantities of plutonium and uranium	Plutonium (nonpyrophoric), DU, NU, and EU (to 5% ^{235}U)	1961–1980
Building B–Multipurpose fabrication building	DU, NU, thorium, plutonium	DU metal or alloy, U_3O_8, Pu-238, ThO_2	1961–1980
Building B Hafnium Plant	Metal production	Hafnium and Zr-Be alloy (nonradioactive)	1961–1980
Building B Plutonium Annex	Conversion of Pu-238 nitrate to oxide	Pu-238 nitrate, Pu-238 oxalate, PuO_2 powder or alloys	1963–1980
Building B–Hot Cell Room	Large source production: Postirradiation examination of test capsules and fuel pins	Co-60, Cs-137, Ir-192, and PoBe, irradiated, uranium and plutonium and other TRU elements and FPs	1961–1980
Building B–Metals Plant	First floor: small-scale metals production, fuel pellet production, materials testing	First floor: DU, UO_2, U_3O_8, UF_4, fully clad U-233, U-235, and Pu-239	1963–1974
	Second floor: Pu-238 pacemakers	Second floor: Pu-238-powered heart pacemakers	1963–1970
Building B–Machine Shop	Occasional machining of clad or unclad uranium, and clad plutonium and U-233; fabrication and repair of new and contaminated equipment from Parks Township and Apollo; machining of DU	Fully clad U-233 and Pu-239 and clad or unclad U-235 (any enrichment), primarily DU contamination; could include HEU, plutonium, thorium, and mixed FPs	1964–1980
Building C, Type II Facility or T-2 Plant	HEU processing to form sintered product	HEU (1973–1978), soluble chloride/oxide complexes, SNM oxides (UO_3, UO_2 and U_3O_8)	1973–1978
Outdoor Scrap Storage Area	Storage	UF_6 cylinders	1971–1980

The updates were based on review of current and new information obtained in the February 2015 data capture effort. Reports from the data capture effort include Shields 1990, Scarlata 1996, and B&W 1995.

FINDING 2: THE SITE PROFILE SHOULD PROVIDE GUIDANCE ON WHAT LEVEL OF URANIUM ENRICHMENT SHOULD BE ASSUMED FOR THOSE URINE BIOASSAY RESULTS THAT ARE EXPRESSED IN UNITS OF MICROGRAMS (μG/L) PER LITER.

NIOSH Response: A new section (Section 5.2.2.4, *Uranium Fluorimetric Bioassay Evaluations*) will be added to the site profile to provide guidance on conversion from mass units to activity units, as follows.

5.2.2.4 Uranium Fluorimetric Bioassay Evaluations

The evaluation of intakes based on uranium fluorimetric bioassays requires conversion from units of mass to activity. This requires information on the uranium-235 enrichment level because the specific activity of the uranium material varies with enrichment. If the enrichment is known for the exposed worker, the specific activity for that enrichment should be used as defined in Table 5-2. If the enrichment is not known, an estimate of activity that is favorable to claimants can be made by assuming the material to be highly enriched (93%). This is a reasonable approach because highly enriched uranium was used frequently at the Apollo facilities in the early years when urine bioassays were performed by fluorimetric analysis.

If the bioassay results contain both fluorimetric and radiometric results, the radiometric results should be used as the radiometric analysis method provided a more sensitive estimate of uranium activity and no conversion of units is required.

FINDING 3: SOME GUIDANCE IS NEEDED ON HOW TO PERFORM DOSE RECONSTRUCTIONS PRIOR TO 1959, AND WHAT APPROACH SHOULD BE USED FOR MISSED AND UNMONITORED EXPOSURES.

NIOSH Response: The evaluation of external and internal dose for NUMEC sites has been addressed in the Special Exposure Cohort (SEC) and was found to be infeasible. Therefore, evaluation of missed and unmonitored exposures is not feasible. Partial dose reconstructions for non-presumptive claims are assigned based on the guidance in the TBD.

FINDING 4: URANIUM INHALATION RECOMMENDATIONS FOR THE APOLLO SITE NEED TO TAKE INTO CONSIDERATION THE METHOD DISCUSSED BY DAVIS AND STROM (2008) FOR DEALING WITH UNCERTAINTIES IN DAILY WEIGHTED EXPOSURE (DWE). THIS TECHNIQUE WAS EVALUATED AND FOUND TO BE APPROPRIATE IN THE SITE PROFILE REVIEW FOR THE FERNALD FEED MATERIALS PRODUCTION CENTER (FERNALD).

NIOSH Response: Davis and Strom (2008) evaluated several HASL reports and concluded that the daily weighted exposures have a lognormal distribution with a geometric standard deviation of 5. HASL data is available for the Apollo site and has been used to evaluate exposure concentrations. As discussed in Finding 18, the air concentration used for the beginning of the residual period has been re-evaluated and is now based on General area (GA) monitoring rather than daily weighted average exposures (see response to Findings 5 and 18 below). The

geometric standard deviation (GSD) for the revised air concentrations has been set to 5.0 based on Davis and Strom (2008).

FINDING 5: INADEQUATE INFORMATION IS GIVEN TO REPLICATE NIOSH'S DETERMINATION OF MEDIAN INHALATION CONCENTRATION OF URANIUM. THE NIOSH RESULT COULD NOT BE REPLICATED, AND IT APPEARS THAT RELEVANT INFORMATION HAS BEEN OMITTED FROM THE HEALTH AND SAFETY LABORATORY (HASL) STUDIES REPORTED IN APPENDIX A TO THE SITE PROFILE.

NIOSH Response: All data listed in the HASL tables has been used in the NIOSH evaluation (reported in the Table II entries in the HASL reports). The median concentration was evaluated as described in the Site Profile based on the assumed 95% (maximum = 6300 dpm/m^3) and 5% (minimum = 7 dpm/m^3) values. The median is evaluated as the square root of the product of the maximum and minimum values. This method assumes the distribution to be lognormal. The actual distribution is skewed to high values above the median. A re-evaluation of all of the data in the HASL reports indicates the estimated value of 210 dpm/m^3 is slightly smaller than the arithmetic mean (286 dpm/m^3) but much greater than the median based on a review of all data, using actual 95% (910 dpm/m^3) and 5% values (9 dpm/m^3) which gives a median of 90 dpm/m^3. Using the 1 sigma values, a median value of 59 dpm/m^3 is obtained. Therefore, the value reported is favorable to claimants. When using a lognormal distribution, the correct value to use is the median and not the arithmetic mean.

The SC&A discussion indicated that a concentration of 563 dpm/m^3 (Schnell 1966) is not reflected in the NIOSH median air concentration of 210 dpm/m^3. The value of 563 dpm/m^3 was taken on September 21, 1966 (most likely date), and represents a small period. The memo by Schnell was written to demonstrate that the air concentrations in the locker room were becoming too high. Data presented indicated that from late August to late September 1966, the locker room air concentration increased from 16 dpm/m^3 on August 25, 1966, to 321 dpm/m^3 on September 20, 1966, with the 563 dpm/m^3 value being reported in the memo of September 22, 1966. The memo suggested that the high air concentrations were a result of workers tracking contamination into the locker room on clothing. Remedial actions were suggested to reduce the locker room air concentrations. The value of 563 dpm/m^3 is a value that occurred over a short period of time and does not represent an average that would be representative of the general air concentration. Also, the NIOSH value of 210 dpm/m^3 and GSD of 7.91 can be used to estimate the 84th percentile of the distribution as 1661 dpm/m^3, which is much higher than the Schnell value. Therefore, the Schnell value is well represented in the NIOSH analysis when the lognormal distribution is considered.

Based on Finding 18, the concentration used for the residual period should be based on general air monitoring and not on breathing zone values (as was used for the HASL studies). Based on this finding, the air concentration has been re-evaluated using GA monitoring results (see Finding 18). The revised air concentration will be used to update the site profile.

FINDING 6: THE SITE PROFILE WOULD BENEFIT FROM A DISCUSSION DEMONSTRATING THAT THE HANFORD SITE FUEL GRADE MIX, AS OPPOSED TO THE WEAPONS-GRADE OR COMMERCIAL-GRADE PLUTONIUM, IS LIMITING FOR THE FULL RANGE OF PLUTONIUM MIXES AND AGES THAT WERE USED AT NUMEC. IN ADDITION, GIVEN THE COMPLEXITY OF THIS SUBJECT, A REVIEW OF ACTUAL DOSE

RECONSTRUCTIONS WOULD PROVIDE GREATER INSIGHT INTO HOW THIS MATTER IS ACTUALLY BEING ADDRESSED.

NIOSH Response: The Fast Flux Test Facility (FFTF) program used only 12% plutonium-240 fuel, so that fuel grade should be used when it is known that the worker was involved with FFTF fuel fabrication. The Zero Power Plutonium Reactor (ZPPR) fuel program used primarily 12% plutonium-240 (11,500 fuel plates), but also used some 27% plutonium-240 (700 fuel plates). The 27% plutonium-240 material was not received on the site until about April of 1969 based on shipping receipt records (NUMEC 1968–1970).

Additional information was obtained from a review of new documents provided by BWTX in February 2015. A fuel fabrication contract with Japan (PNC abbreviation in NUMEC documents) was performed from 1966 to 1970. A fuel fabrication contract with Argonne National Laboratory for preparation of fuel wafers for the Zero Power Reactor (ZPR-III abbreviation) was performed from 1964 to 1966. Both of these contracts involved fuel with a plutonium-240 content of 8.1% to 8.5%. Based on this information, Table 5-3 has been revised to include the following four plutonium types: Hanford 6% weapons grade, DOE 8.5% material, Hanford 12% fuel grade, and 27% commercial fuel. The Hanford and commercial specific activity data is from Table 5-3 through Table 5-4 of the Internal Dosimetry section of the Hanford TBD. The specific activity of the 8.5% material was based on the reported activity of 91.47% for the sum of the plutonium-239 and -241 activities (fissile material). The activities of plutonium-238, -241, and -242 were estimated by ratio of the isotopic content between the 6% and 12% inventory values. The plutonium-240 content was set at 8.3% as 100%–91.47%. The activity of americium-241 was based on the ratio of americium-241 to plutonium-241 in the Hanford fuels. This ratio was a constant for the two Hanford fuels, as expected because the americium-241 is a decay product of the plutonium-241.

Table 7-4 will be updated to include the commercial reactor-grade fuel mix fractional activity data. Guidance will be added regarding selection of the appropriate inventory component for evaluation of internal doses based on available information.

The revised plutonium isotopic table is as follows.

Table 5-3. Activity composition of plutonium mixtures.

Mixture designation	**Hanford 6% Weapons-grade Plutonium Specific activity (Ci/g)[a]**				
Years of aging[b]	**0**	**5**	**10**	**15**	**20**
Pu-238	8.56E-03	8.23E-03	7.91E-03	7.60E-03	7.31E-03
Pu-239	5.77E-02	5.77E-02	5.77E-02	5.77E-02	5.77E-02
Pu-240	1.36E-02	1.36E-02	1.36E-02	1.36E-02	1.36E-02
Pu-241	8.24E-01	6.48E-01	5.09E-01	4.00E-01	3.15E-01
Pu-242	1.97E-06	1.97E-06	1.97E-06	1.97E-06	1.97E-06
Am-241	0	5.83E-03	1.04E-02	1.39E-02	1.66E-02

Mixture designation	DOE 8.5% Plutonium Specific activity (Ci/g)[c]				
Years of aging[b]	**0**	**5**	**10**	**15**	**20**
Pu-238	1.22E-02	1.17E-02	1.12E-02	1.08E-02	1.04E-02
Pu-239	5.55E-02	5.55E-02	5.55E-02	5.55E-02	5.55E-02
Pu-240	1.94E-02	1.94E-02	1.94E-02	1.94E-02	1.94E-02
Pu-241	1.78E+00	1.40E+00	1.10E+00	8.64E-01	6.80E-01
Pu-242	2.80E-06	2.80E-06	2.80E-06	2.80E-06	2.80E-06
Am-241	0	1.26E-02	2.25E-02	3.00E-02	3.58E-02
Hanford 12% Fuel-grade Plutonium Specific activity (Ci/g)[a]					
Years of aging[b]	**0**	**5**	**10**	**15**	**20**
Pu-238	1.71E-02	1.64E-02	1.58E-02	1.52E-02	1.46E-02
Pu-239	5.26E-02	5.26E-02	5.26E-02	5.26E-02	5.26E-02
Pu-240	2.72E-02	2.72E-02	2.72E-02	2.72E-02	2.72E-02
Pu-241	3.09E+00	2.43E+00	1.91E+00	1.50E+00	1.18E+00
Pu-242	3.93E-06	3.93E-06	3.93E-06	3.93E-06	3.93E-06
Am-241	0	2.19E-02	3.89E-02	5.22E-02	6.24E-02
Commercial 27% Fuel-grade Plutonium Specific activity (Ci/g)[a]					
Years of aging[b]	**0**	**5**	**10**	**15**	**20**
Pu-238	1.71E-01	1.64E-01	1.58E-01	1.52E-01	1.46E-01
Pu-239	3.41E-02	3.41E-02	3.41E-02	3.41E-02	3.41E-02
Pu-240	5.90E-02	5.89E-02	5.89E-02	5.89E-02	5.89E-02
Pu-241	1.34E+01	1.05E+01	8.28E+00	6.51E+00	5.12E+00
Pu-242	1.97E-04	1.97E-04	1.97E-04	1.97E-04	1.97E-04
Am-241	0	9.49E-02	1.69E-01	2.26E-01	2.79E-01

a. Source: ORAUT (2012b).
b. Time since separation of Am-241 from the plutonium mix.
c. NUMEC records (Gerrish 1965).

FINDING 7: THE MINIMUM DETECTABLE ACTIVITIES (MDAS) FOR AM-241 LUNG COUNTING ARE VERY LOW. THE COUNTING METHOD SHOULD BE FURTHER EXPLORED IN ORDER TO GIVE THEM CREDIBILITY.

NIOSH Response: The values listed for americium-241 are consistent with contemporary minimum detectable activity (MDA) values for other sites, such as Hanford. For example, both sites used sodium iodide crystal detectors in the late 1960s; the Hanford americium-241 MDA is given as 0.33 nCi and the value for NUMEC is listed as 0.13 to 0.38 nCi.

The MDA values in the NUMEC site profile are listed as minimum and maximum values. Because the detection limit is a function of body size and chest wall thickness, several data points are needed to estimate a reasonable upper bound on the detection limit. The MDA to be used in dose reconstructions (when a detection limit is not provided in the bioassay records) should account for large persons and provide an intake estimate that is favorable to the claimant. The variation in MDA with chest wall thickness that was reported for the University of Pittsburg system in 1975 is as follows (Author unknown 1977):

Chest wall thickness	MDA Pu-239 (nCi)
1 cm	3–4
2 cm	9–10
3 cm	17–20

The review of the February 2015 BWXT data capture documents for MDA values for lung counting resulted in minor changes to the previous data. The changes are indicated in the following proposed revision to Table 5-9 in the NUMEC site profile (AEC 1974a, 1974b, Caldwell 1969a).

Table 5-9. In vivo MDAs for plutonium-239 and americium-241.[a]

Year	Pu-239 MDA (nCi)			Am-241 MDA (nCi)		
	Minimum	Maximum	Counts	Minimum	Maximum	Counts
1968	NR[b]	NR[b]	NR[b]	0.13	0.38	17
1969	NR[b]	NR[b]	NR[b]	0.16	0.16	1
1970	10	10	1	NR[b]	NR[b]	NR[b]
1971	NR[b]	NR[b]	NR[b]	NR[b]	NR[b]	NR[b]
1972	9.0	11.5	3	0.13	0.13	1
1973	5.6	15.6	46	0.11	0.21	28
1974	5.44	21.3	122	0.06	0.22	96
1975	4.8	19.9	133	0.11	0.21	104
1976	5.0	20.3	109	0.11	0.19	91
1977	4.4	19.6	113	0.09	0.19	88
1978	4.7	19.0	132	0.10	0.19	100
1979	5.16	24.3	168	0.08	0.26	132
1980	5.03	28.2	132	0.09	0.21	94
1981	7.21	27.8	55	0.12	0.20	31
1982	7.12	34.3	77	0.12	0.21	44
1983	9.41	15.6	6	0.12	0.16	4
1984	8.67	22.32	9	0.12	0.15	5
1985	8.84	31.07	31	0.11	0.22	29

a. From a review of worker dosimetry records (AEC 1974a, 1974b, Caldwell 1969a, Boyd 2006a, 2006b, 2006c, 2006d, 2006e, 2006f). Values for 1968 through 1971 are based on the Helgeson system, with remaining values for the University of Pittsburgh system.

b. NR = none reported.

Because the data is sparse for the early years, the upper bound of the MDA is not likely reflected in the data. Based on the above information, guidance will be added to the site profile directing use of the MDA values in the following table when the bioassay records do not provide the MDA.

Period	Pu-239 MDA (nCi)	Am-241 MDA (nCi)
1968–1985	35	0.40

FINDING 8: THE SITE PROFILE WOULD BENEFIT FROM A MORE THOROUGH DISCUSSION OF THE POSSIBLE USE OF AIR SAMPLING DATA TO RECONSTRUCT INTERNAL PLUTONIUM EXPOSURES AND TO TAKE INTO CONSIDERATION THE ADDITIONAL DATA PROVIDED BY CROSBY 1967 AND NUMEC 1967.

NIOSH Response: The two references give results of air sampling in the plutonium scrap recovery area of Parks Township. The evaluation of internal dose for NUMEC sites has been addressed in SECs and was found to be infeasible. Therefore, evaluation of internal plutonium exposures for claims with no bioassay monitoring data is not feasible. Internal plutonium exposures can only be included in dose reconstructions when valid bioassay monitoring data is available. Air sampling data can only be used when it represents sampling for the energy employee being evaluated in a claim. Such data has been used in past dose reconstructions.

The records typically provide the exposure time and air concentration which can be used to estimate an intake. The internal dose from the intake can be evaluated using the IMBA program. Partial dose reconstructions for non-presumptive claims are assigned based on the guidance in the site profile.

FINDING 9: IT DOES NOT SEEM APPROPRIATE TO USE ORAUT-OTIB-0054 (ORAUT 2007a) TO RECONSTRUCT THE INTERNAL EXPOSURES OF WORKERS AT NUMEC WHO MIGHT HAVE BEEN EXPOSED TO MIXED FISSION PRODUCTS BECAUSE ORAUT-OTIB-0054 STATES THAT ITS GUIDANCE "DOES NOT APPLY TO DETERMINATION OF INTAKES WHERE RADIONUCLIDES HAVE BEEN PURPOSELY EXTRACTED AND CONCENTRATED AS FOR HEAT GENERATION SOURCES, MEDICAL USES, OR WASTE HANDLING OPERATIONS THAT CAUSED SIGNIFICANT ALTERATION TO THE SOURCE TERM TO WHICH WORKERS WERE EXPOSED." FOR EXAMPLE, TABLE 5-1 OF THE SITE PROFILE INDICATES THAT THE FISSION PRODUCTS HANDLED AT NUMEC ARE SOURCES USED FOR VARIOUS RESEARCH AND OPERATIONS PURPOSES AND ARE NOT ACTUALLY FUEL OR SPENT FUEL AND WOULD NOT NECESSARILY BE PRESENT IN THE SAME RATIOS AS IN REACTOR FUEL. ALSO, THE FISSION PRODUCT MIX GIVEN IN ORAUT-OTIB-0054 DOES NOT CONTAIN THE SAME RADIONUCLIDES AS THE FISSION PRODUCT MIXES GIVEN FOR THE NUMEC LAUNDRY IN THE 1975 EFFLUENT RELEASE REPORT (WILLIAMS 1967) AND FOR IN-VIVO COUNT RESULTS IN CALDWELL 1969B. THE NUMEC MIXES INCLUDE CO-60, WHICH THE ORAUT-OTIB-0054 MIX OMITS, FURTHER DEMONSTRATING THE INAPPLICABILITY OF ORAUT-OTIB-0054 TO NUMEC.

NIOSH Response: Use of OTIB-0054 (ORAUT 2015) is appropriate when the source of activity is from reactors. As indicated in this finding, the NUMEC laundry processed materials from many sources in addition to the commercial reactor laundry. The site profile will be modified to remove guidance on use of OTIB-0054 for evaluation of dose from associated fission products and activation products. The guidance will indicate that doses are to be evaluated only for the radionuclides included in the bioassay results.

FINDING 10: INTERNAL DOSE RECONSTRUCTIONS PERFORMED FOR NUMEC PERSONNEL MIGHT NEED TO BE REVISITED IN LIGHT OF CHANGES TO THE FERNALD SITE PROFILE (ORAUT 2004) WITH RESPECT TO RECYCLED URANIUM (RU). ALSO, ADDITIONAL DIRECTION IS NEEDED WITH RESPECT TO WHICH WORKERS OR OPERATIONS SHOULD BE ASSIGNED RU INTAKES.

NIOSH Response: Changes to the Fernald site profile have not been finalized at this time so changes to guidance on evaluation of contaminant activity for RU are not yet necessary. The source of the uranium used at NUMEC is not known for many of the activities, so the assumption of RU for operations involving enriched or depleted uranium is reasonable. Operations involving natural uranium would not be expected to have RU contaminants. Current guidance for NUMEC is to include the components associated with recycled uranium (plutonium-239, neptunium-237, and technicium-99) using activity fractions in Table 5-12 of the current (ORAUT 2004) Fernald internal dose TBD. The NUMEC site profile will be updated to indicate the need to include all contaminants in the comparison of types F, M, and S uranium materials in order to decide which absorption type of uranium is most favorable to the claimant. The most claimant-favorable absorption type will also vary, depending on the organ of interest. The dose reconstructor is instructed to follow guidance in ORAUT-OTIB-0060 (ORAUT 2014a) for which material types to use when assigning recycled uranium contaminants. These

contaminants must be added to all uranium intakes regardless of the monitoring method (urine, chest counts, or breathing zone samples).

FINDING 11: NIOSH SHOULD EXPLAIN WHETHER THE CONCERNS EXPRESSED IN THE PANTEX SITE PROFILE (ORAUT 2007d) ABOUT THE HELGESON CHEST COUNT DATA MIGHT ALSO APPLY TO CHEST COUNT DATA AT NUMEC PERFORMED BY HELGESON FOR NUMEC WORKERS.

NIOSH Response: The in vivo counting at NUMEC was performed by Helgeson only during 1968, 1971, and once in 1975. The Helgeson counts included primarily plutonium-239 with a few results for uranium and fission products. The majority of in vivo counting was performed at the University of Pittsburg hospital. The issue with Pantex was that the counts for uranium were biased high and represented false positives (Brake 1989). The current Pantex internal dose TBD has eliminated all reference to Helgeson in vivo measurements. Using the Helgeson results for uranium is favorable to the claimant because of the high bias in the reported values. There was no issue identified related to Helgeson in vivo counting for fission products, plutonium, or americium. No changes to the NUMEC site profile are planned at this time related to the Helgeson chest counts for uranium.

FINDING 12: TABLE 6-2 AND THE ASSOCIATED TEXT IN SECTION 6.3.2 OF THE SITE PROFILE SHOULD BE REVIEWED AND MODIFIED AS NEEDED TO CORRECT ANY OVERSIGHTS, INCONSISTENCIES, OR ERRORS.

NIOSH Response: The indium foil criticality dosimeters were not included in Table 6-2 because they were not used for routine workplace exposures. They were included only to determine dose from a criticality in the event that a criticality occurred. No criticality incidents were reported at the NUMEC facilities. The text was reviewed as suggested and the information in the text is consistent with information provided in Tables 6-2 and 6-3..

FINDING 13: GIVEN OUR UNDERSTANDING THAT IT IS NIOSH'S POSITION THAT EXTERNAL EXPOSURES AT THE PARKS TOWNSHIP SITE CAN BE RECONSTRUCTED WITH SUFFICIENT ACCURACY, IT APPEARS THAT THE DESCRIPTION OF THE SOURCES AND CIRCUMSTANCES RESPONSIBLE FOR EXTERNAL EXPOSURES NEED TO BE BETTER DEVELOPED, IF POSSIBLE.

NIOSH Response: External dose can be evaluated only when dosimetry records exist. There was one dosimetry department for all NUMEC facilities that provided dosimetry for both the Parks Township and Apollo facilities. The limitations stated for the Apollo facilities also apply to the Parks Township facilities (see response to Finding 16).

FINDING 14: THE SITE PROFILE SHOULD PROVIDE JUSTIFICATION FOR WHY ADJUSTMENT FACTORS ARE NOT REQUIRED FOR NEUTRON EXPOSURES ESTIMATED USING NUCLEAR TRACK EMULSION TYPE A (NTA) FILM, CONSIDERING THAT IT APPEARS THAT THE NEUTRON ENERGY SPECTRUM LIKELY EXTENDED TO WELL BELOW 1 MEGA-ELECTRONVOLT (MEV). FOR EXAMPLE, TABLE 6-8 OF THE SITE PROFILE INDICATES THAT THE ENERGY RANGE OF NEUTRON EXPOSURES EXTENDED FROM 0.1 TO 2 MEV.

NIOSH Response: NUMEC used NTA film to monitor for neutron exposure until approximately June 1968 when thermoluminsecent dosimeters (TLDs) were introduced at the site. A study was performed in 1975 using TLDs to determine the photon and neutron doses while working in the FFTF fuel fabrication area of the Parks Township Plutonium Facility. Additionally, dosimeters were placed at various fixed locations. The results of this study resulted in the following estimate of neutron to photon ratio (Author unknown 1977):

Neutron-to-Photon Ratio on Personnel Dosimeters	
Geometric Mean	0.34
GSD	1.71

This ratio is supported in the Parks Township Plutonium Facility 1979–1981 Health and Safety ALARA Reports (Corridoni 1982) which indicate neutron to photon ratios varied from 0.23 to 0.42 with an average of 0.33.

Neutron-to-Photon Ratio Glovebox Dosimeters[a]	
Geometric Mean	1.00
GSD	1.49

a. The difference between the ratios is likely a result of shielding on the gloveboxes that would reduce the photon component reaching the dosimeter.

In addition, information was obtained from a September 1968 event where a worker involved with manufacturing neutron sources had a neutron-to-photon ratio of 2.33 (determined using estimated neutron dose values) (Caldwell 1968). A review of neutron-to-photon ratios from neutron sources was evaluated and a broad range of ratios were found. Without additional information to indicate otherwise, the project determined ratio of 2.33 will be used.

It is proposed that NUMEC workers, prior to 1969, with neutron dosimetry receive a neutron dose assigned using the ratio most appropriate for their work and job location (i.e., use a ratio of 0.34 for typical workers, a ratio of 1.00 for glovebox workers, and a ratio of 2.33 for workers involved with manufacturing sources). This dose would be assigned using a lognormal distribution with the GSD provided, except for the manufacturing source ratio which would be applied as a constant. If the worker's recorded neutron dose is higher than the neutron dose calculated from applying the ratio to the photon dose, the recorded neutron dose should be assigned.

This approach will be included in the NUMEC site profile.

FINDING 15: THE MARKEDLY DIFFERENT PHOTON ENERGIES ASSOCIATED WITH THE OPERATIONS AT NUMEC WOULD INDICATE THE POSSIBLE NEED FOR ADJUSTMENT FACTORS FOR THE RESULTS OF FILM BADGE DOSIMETERS, WHICH ARE NOT PROVIDED IN THE SITE PROFILE. IT IS UNCLEAR WHAT THE AUTHORS INTENDED BY DESCRIBING ALL BETA ENERGIES FOR AM-241 AS BEING >15 KEV, AS AM-241 AND U-233 ARE ALPHA EMITTERS.

NIOSH Response: Film badge dosimeters, while over responding to radiation recorded in the open window, may under respond to low energy photons (16 keV and 59 keV photons are a particular concern) (Wilson et al. 1990). Although the films and filters at NUMEC were different than the dosimeters discussed in the reference, a reasonable comparison between the film

dosimeters is expected (AEC 1955). The site acknowledged this deficiency in a 1966 report (Caldwell and Judd 1966) and made corrections to the dosimetry to account for this under response. Prior to the report being issued (i.e., prior to 1966), to account for under response of film dosimetry to low energy photons, the result in the open window should be assigned as <30 keV photons for workers at plutonium facilities while the deep dose response is assigned in accordance with the worker location.

This approach will be included in the NUMEC site profile.

Americium-241 and uranium-233 have no beta decay but do emit many internal conversion electrons and Auger electrons which are essentially beta particles. Therefore, reference to beta radiations is appropriate in Table 6-5. While some of the electron emissions are less than 15 keV, the low energy electrons are not as likely as the higher energy electrons to reach the skin and cause dose to the skin. Therefore, assigning the electron dose as 100% >15 keV is appropriate, except as noted above for plutonium workers.

FINDING 16: NIOSH SHOULD CONSIDER DEVELOPING A UNIVERSAL COWORKER MODEL BASED ON NUMEC CLAIMANT RECORDS, OR SPECIFY A MORE CONSISTENT BASIS FOR ASSIGNING EXTERNAL DOSES BEYOND THE MEDICAL X-RAYS ASSOCIATED WITH THE SITE.

NIOSH Response: The NUMEC sites are already covered by an SEC based on external and internal infeasibility. Partial dose reconstructions for non-presumptive claims are assigned based on the guidance in the NUMEC site profile.

Both SEC evaluation reports state that external doses are assigned to partial DRs in cases where monitoring data is available. The Apollo Evaluation Report states that external dose reconstruction is not feasible, but that in some cases available dosimetry data can be used. The Parks ER states that external data is generally available, however, it also states that the SEC decision was based NIOSH's determination that internal dose reconstruction was not feasible (83.14) and that the external feasibility was not evaluated. This is reasonable because the internal and external monitoring programs were performed under the guidance of the same occupational safety organization for both sites. The Parks ER also states that any available external dosimetry data will be used for partial dose reconstructions.

There are 89 post SEC claims for Apollo and 27 for Parks, some of these have concurrent employment at both sites. Thirty randomly selected claims were reviewed for Apollo and 25 for Parks (all post SEC period) for inconsistencies in their dose reconstruction approach. No inconsistencies were noted. In cases where no external dosimetry is available during the SEC period, no external dose is assigned, only medical X-rays. In cases where dosimetry is available, external dose is assigned. However, note that current guidance is to not assign medical X-ray dose based on OTIB-0079 (ORAUT 2011).

There is an issue with potential external unmonitored dose occurring at NUMEC as outlined in the Apollo ER. There are claims with internal dosimetry that clearly show that the energy employee worked in a radiation area resulting in detectable intakes but no external dosimetry data is available. However, this issue was recognized during the SEC evaluation process and contributed to the finding that it is not feasible to reconstruct external doses for Apollo (Parks was not evaluated, but the issue is most likely the same as was the case with internal).

Therefore, NIOSH advises against any effort to develop a coworker model for unmonitored workers for NUMEC.

FINDING 17: THE SITE PROFILE SHOULD INCLUDE GUIDANCE FOR DERIVING NON-PENETRATING DOSES TO SKIN AND OTHER ORGANS FROM BETA EMITTERS ASSOCIATED WITH SURFACE CONTAMINATION DURING THE RESIDUAL PERIOD.

NIOSH Response: The dose values presented in the site profile for exposure to surface contamination are currently assigned as all photon dose (30–250 keV). While this provides a favorable to claimant estimate of probability of causation, it does not include consideration of the beta radiation contribution to the dose to breast, testes, and penis as described in OTIB-0017 (ORAUT 2005). The dose conversion factors used in generation of Table 7-1 include all radiation. The doses to organs other than skin are primarily from photons and represent the deep dose equivalent in rem. This is in contrast to the dose factors presented in BMI 2011 which are exposure dose factors with 71.8% of the dose from photons <30 keV. The skin dose factor from Table 7-1 is primarily from electron radiation and is much higher than the photon dose. In the revision to the site profile in response to Finding 20, the external dose from residual thorium will be added to Table 7-1. In the revision, Table 7-1 entries will be limited to whole body exposure and skin dose. This will make application of the chronic lymphocytic leukemia (CLL) evaluations in the workbooks practical because that calculation is performed using one dose value and then applying the isotropic exposure dose conversion factors. The value given for whole body exposure in the revised Table 7-1 will be such that application of the isotropic exposure dose conversion factors provides a correct estimate of organ doses for the CLL evaluation and other cancer organs and does not underestimate the dose to any organ.

The method for assignment of external dose from residual materials at the Apollo Plant will be as indicated in the following table. The following is based on guidance in the technical information bulletin for shallow dose (ORAUT 2005) and the procedure for occupational onsite external dose (ORAUT 2006). This method includes application of the exposure isotropic dose conversion factors and results in a best estimate of dose.

Organ(s)	Dose assignment	Radiation
All except skin, breast, testes, and penis	Listed whole body dose times organ isotropic exposure dose conversion factor	Photons 30–250 keV
Skin	Listed skin dose	Electrons >15 keV
Breast, testes, and penis	Listed whole body dose times organ isotropic exposure dose conversion factor	Photons 30–250 keV
	30% of listed skin dose	Electrons >15 keV

FINDING 18: GENERAL AIR (GA) SAMPLES, AS OPPOSED TO BREATHING ZONE (BZ) SAMPLES, SHOULD BE USED AS THE STARTING POINT FOR RECONSTRUCTION OF RADIONUCLIDE INTAKE RATES DURING THE RESIDUAL PERIOD.

NIOSH Response: GA data rather than the BZ data is appropriate to use as a starting point for the residual period. The BZ values are generally taken when employees were working with radioactive materials and do not represent an average over all periods. The BZ values are therefore skewed to high values. Evaluation of the median GA based on the 5 HASL reports gives a value of 33 dpm/m^3 (GSD = 5.45). Using the 95% value gives a medium of about

60 dpm/m^3 (GSD = 4.14). The guidance in the site profile will be updated using additional information extracted from site reports. The most appropriate GA data is that collected using continuous air monitors that include all variations in air concentration over the year. Such data was found for the year 1967 for several of the operations at the Apollo uranium facility. Limited data was also found for 1966 for the Hammer Mill, and for a few periods in 1962 and 1963 for the CRP-3 beryllium glove box area and process area. The results from these measurements are summarized in the following table.

Data Source	Period	Work Location	Median dpm/m^3	GSD	Mean dpm/m^3	Samples taken
HASL Reports	1959–1961	All U processing	53	6.95	196	332
Site records	1967	Depleted GB	61	2.02	80	76
Site records	1967	Hammer Mill	35	2.07	43	223
Site records	1967	Furnace GB	98	1.34	102	86
Site records	1967	Change Room	90	2.07	93	284
Site records	1967	End Shake	51	2.26	63	132
Site records	1967	Crystallizing Hood	50	1.92	56	375
Site records	1967	Dissolving Hood	45	1.73	51	311
Site records	1967	Hammer mill	38	1.97	48	303
Site records	1967	UO2 Glove Box	34	1.89	41	331
Site records	1967	Aisleway Sampler 19	53	2.01	64	645
Site records	1967	Aisleway Sampler 20	43	2.21	51	368
Site records	1966	Hammer Mill	198	3.40	307	57
Site records	1966	Hammer Mill	222[a]	NA[b]	415	~878
Site records	1962–1963	CRP-3 process area	119[a]	NA[b]	150	181
		CRP-3 Be glove box area	119[a]	NA[b]	150	95
Averages	1967	—	54	1.95	63	—
	All data	—	82	6.95 max	119	—

a. Value estimated by ratio of median/mean for other data sets.
b. NA = value not available from data provided.

Based on the data in the above table, an air concentration value that is favorable to claimants to use for the residual period is represented by the maximum median value of 222 dpm/m^3. The maximum estimated GSD for all data sets is 6.95 for the HASL GA data. Therefore, the residual activity will be based on a lognormal distribution with a median of 222 dpm/m^3 and a GSD of 5.0. This represents a slight increase in median air concentration from the previous concentration of 210 dpm/m^3 and GSD of 7.91.

FINDING 19: SC&A RECOMMENDS THAT NIOSH USE A RESUSPENSION FACTOR OF ABOUT 1E-5 PER METER TO DERIVE THE AIRBORNE DUST LOADING FOR THE BEGINNING OF THE RESIDUAL PERIOD, OR PERHAPS SIMPLY ASSUME THAT THE AVERAGE GENERAL AIR DUST LOADING OBSERVED DURING THE OPERATIONAL PERIOD IS APPLICABLE TO THE BEGINNING OF THE RESIDUAL PERIOD.

NIOSH Response: The NIOSH guidance is to use 10^{-6} m^{-1} as a resuspension factor for undisturbed areas. As stated in OTIB-0070 (ORAUT 2012c):

> *Application of resuspension factors in dose assessment has been studied by a number of authors. Generally, early conclusions of a value of 10^{-6} m^{-1} under quiescent conditions*

and a factor of 10 higher (10^{-5} m^{-1}) under conditions of moderate activity (Stewart 1964) have been supported by later analysis (Brodsky 1980).

Because the Apollo and Parks Township facilities continued to have some operations during the residual period, the use of a resuspension factor of 10^{-5} m^{-1} is reasonable for the uranium facilities at Apollo and Parks Township. The value of 10^{-6} m^{-1} is reasonable for the Parks Township plutonium facility because the facility was cleaned prior to continued use.

The resuspension rate for the Apollo facility will be changed to 10^{-5} m^{-1} and updates will be made to Tables 7-2, 7-3, and 7-5. The ingestion rates will be updated as well, but only for the change in initial air concentration. In addition, per Finding 18, the initial air concentration and deposition activity will be re-evaluated for uranium at Apollo to reflect GA data rather than BZ data.

FINDING 20: THE SITE PROFILE MAKES NO REFERENCE TO RADIONUCLIDES OTHER THAN URANIUM DURING THE RESIDUAL PERIOD AT APOLLO.

NIOSH Response: The issue has been further reviewed; however, the vast majority of activity processed at Apollo was uranium. General air concentration data for thorium at the Apollo CP-1 facilities during 1964 and 1965 were reviewed. The GA samples were not continuous, but were taken for periods from a few minutes to about an hour. The 71 data points indicated a median thorium concentration of 112 dpm/m^3, a mean of 160 dpm/m^3, and a GSD of 2.4. The samples were counted immediately and after several hours to allow for decay of natural atmospheric radon and progeny. The results used in this analysis were those taken after decay. Based on this data, residual activity for thorium can be evaluated using a lognormal distribution of 112 dpm/m^3 and a GSD of 5.0 (as per the uranium analysis indicated in the Finding 19 response.) The thorium at Apollo was received as thorium oxide which had undergone only one separation. Because the residual period is approximately 20–30 years after the initial processing of thorium at NUMEC, the thorium is assumed to be in equilibrium with all progeny at the time of exposure (ORAUT 2014b). The measured activity (1964 and 1965) is assumed to be for freshly separated thorium with a minimum activity of thorium-228 of 42% of the thorium-232 activity. This resulted in an initial activity of 79 dpm/m^3 thorium-232. The activity of thorium-232 is also assigned to progeny radium-228, actinium-228, radium-224, and lead-210 (radon is assumed to remain in the thorium matrix).

FINDING 21: THERE IS CONFLICTING GUIDANCE ON HOW AGED PLUTONIUM MIXTURES SHOULD BE TREATED DURING THE RESIDUAL PERIOD AT PARKS TOWNSHIP.

NIOSH Response: The site profile does not include a callout or discussion of Table 7-4. This will be added along with guidance to apply the appropriate activity fractions to the total alpha values in Table 7-3. In Table 7-3, the values will be indicated as total alpha. Text will be added to indicate values in Table 7-4 are to be multiplied by the isotopic fractions to get the intakes of each radionuclide. The most claimant favorable mix should be used for each analysis (by fuel type and age).

The main purpose of Table 7-4 is to account for the amount of americium-241 in the material. The dose can then be evaluated for the americium-241 and plutonium. Although the dose from the plutonium isotopes can be evaluated for each isotope, the dose is usually evaluated using

the sum of the alpha emitting isotope values (excluding plutonium-241, a beta emitter) represented as plutonium-239 as a claimant-favorable approach.

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