SC&A S. COHEN & ASSOCIATES
AN EMPLOYEE-OWNED COMPANY

July 18, 2008

Mr. David Staudt
Center for Disease Control and Prevention
Acquisition and Assistance Field Branch
Post Office Box 18070
626 Cochrans Mill Road – B-140
Pittsburgh, PA 15236-0295

Re: Contract No. 200-2004-03805, Task Order 5: Draft Report SCA-SEC-TASK5-0063, *A Focused Review of the NIOSH Evaluation Report for the Texas City Chemical Company SEC Petition*

Dear Mr. Staudt:

SC&A is pleased to submit its draft report, *A Focused Review of the NIOSH Evaluation Report for the Texas City Chemical Company SEC Petition*, SCA-SEC-TASK5-0063. This report has been reviewed for Privacy Act information and cleared for unrestricted distribution.

Should you have any questions, please contact me at 732-530-0104.

Sincerely,

John Mauro, PhD, CHP
Project Manager

cc: P. Ziemer, Board Chairperson
Advisory Board Members
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Draft

ADVISORY BOARD ON

RADIATION AND WORKER HEALTH

National Institute for Occupational Safety and Health

A FOCUSED REVIEW OF THE NIOSH EVALUATION REPORT FOR THE TEXAS CITY CHEMICAL COMPANY SEC PETITION

Contract No. 200-2004-03805
Task Order No. 5
SCA-SEC-TASK5-0063

Prepared by

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July 2008

S. COHEN & ASSOCIATES: ***Technical Support for the Advisory Board on Radiation & Worker Health Review of NIOSH Dose Reconstruction Program***	Document No. SCA-SEC-TASK5-0063 Effective Date: Draft — July 18, 2008 Revision No. 0 – Draft
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Task Manager: ________________________ Date: ___________ William C. Thurber, MS Project Manager: ________________________ Date: ___________ John Mauro, PhD, CHP	Supersedes: N/A

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EXECUTIVE SUMMARY

This report presents SC&A's review of the Texas City Chemicals, Inc. (TCC), Special Exposure Cohort Petition (SEC-00088), which was qualified on August 17, 2007, and the NIOSH SEC Petition Evaluation Report (ORAUT 2008), which was submitted to the Advisory Board on Radiation and Worker Health (Board) on January 18, 2008.

Key findings based on the review are as follows:

Finding 1: The petitioners' class definition needs to be re-examined to insure that the time period has been properly justified.

Finding 2: The use of mean annual external exposures observed among phosphogypsum workers in Idaho does not appear to be a scientifically sound or claimant-favorable surrogate for pre-operational phosphate ore handlers at TCC. Consideration should be given to using the upper 95th percentile exposures for Florida phosphate rock workers as a more appropriate claimant-favorable surrogate.

Finding 3: An alternative modeling approach based on reasonable assumptions derived from the available information suggests that external doses could be two orders of magnitude lower than those developed by NIOSH. NIOSH should reexamine its modeling approach to insure that its assumptions for calculating external doses are representative of plausible circumstances.

Finding 4: NIOSH should consider data from FIPR 1998 as an alternative source of data for estimating internal exposures.

Finding 5: The assumption that workers were exposed to yellowcake for 39 months, while claimant favorable, is not consistent with the available data and may overstate the exposure by an order of magnitude.

Finding 6. NIOSH should consider other data sources in addition to EPA 1978 to estimate internal exposures to workers outside the uranium recovery building, such as FIPR 1998.

Finding 7: The approach used in the evaluation report to reconstruct internal exposures to uranium production operations appears to be unrealistically conservative with regard to exposure duration and exposure level.

Finding 8. The methodology used to estimate inadvertent ingestion should be revised.

Finding 9: NIOSH should consider adjustments to the dataset used to calculate radon doses to fully reflect the available information. This would increase the dose from 0.112 WLM/yr to 0.56 WLM/yr. Should NIOSH determine that the value they selected is, in fact, appropriate for the reasons discussed in the main body of this report, the rationale for this conclusion should be provided in the evaluation report.

Finally, SC&A applied the four draft criteria developed by the Work Group on the Use of Surrogate Data to the surrogate data actually used by NIOSH in their SEC petition evaluation report. Using the TCC report as a "test case," SC&A concluded that the work group might wish to consider the addition of a "plausibility/fairness" criterion. When attempting to bound doses in situations where only limited data are available, the possibility exists that very conservative (perhaps implausible), but certainly claimant-favorable approaches will be used, which result in compensation. Under these circumstances, it is appropriate to consider the "plausibility" of the assumptions and exposure scenarios. We believe that this is one of the reasons why "plausibility" is explicitly required by 42 CFR Part 83.

In a related manner, the "fairness" issue could emerge in circumstances where it is apparent that the potential exposures at a given site are clearly lower than at another site. Specifically, if unrealistically conservative surrogate data are used at a given site, it is possible that implausibly high exposures could be assigned to workers at that site, resulting in compensation. Issues of fairness emerge if compensation is denied at another site that has a much higher potential for exposure because abundant data are available and realistic (yet claimant favorable) doses can be reconstructed. A fairness criterion would invoke specific consideration of such a possibility when choosing surrogate data.

1.0 INTRODUCTION

1.1 PURPOSE AND SCOPE

During the meeting of the Advisory Board on Radiation and Worker Health (the Board) held on April 7–9, 2008 in Tampa, Florida, S. Cohen & Associates (SC&A, Inc.) was directed by the Board to perform a review of the Texas City Chemicals, Inc. (TCC), Special Exposure Cohort Petition (SEC-00088), which was qualified on August 17, 2007, and the NIOSH SEC Petition Evaluation Report (ORAUT 2008), which was submitted to the Board on January 18, 2008. The purpose of this report is to provide the Board with an independent technical review of any issues raised by the petitioners and NIOSH's position regarding these issues as presented in the petition evaluation report.

Since exposure data are not available from the site, NIOSH relied on surrogate data and modeling for dose reconstruction. Particular attention is given in this report to this use of surrogate data and whether such usage meets the draft criteria developed for consideration by the Work Group on the Use of Surrogate Data. These draft criteria for surrogate data usage are included here as Appendix B.

This is the first occasion where the draft criteria related to surrogate data are being used as part of an SEC petition review. This aspect of our review is intended to help the Board in its assessment of the petition and NIOSH's evaluation report with respect to the use of surrogate data. In addition, it is also our intention to show that applying the draft criteria to a real petition and evaluation report will provide insight into aspects of the criteria that may need to be expanded, revised, or refined.

1.2 OBJECTIVE

The objective of this report is to provide the Board with complete and accurate technical information regarding the TCC SEC Petition and NIOSH's evaluation of that petition. There were no issues raised in SEC-00088 that require technical review. Consequently, this report is designed to assist the Board in determining whether radiation doses can be estimated with sufficient accuracy in plausible circumstances based on the following regulatory requirement:

> *Radiation doses can be estimated with sufficient accuracy if NIOSH has established that it has access to sufficient information to estimate the maximum radiation dose, for every type of cancer for which radiation doses are reconstructed, that could have been incurred in plausible circumstances by any member of the class, or if NIOSH has established that it has access to sufficient information to estimate the radiation doses of members of the class more precisely than an estimate of the maximum radiation dose [42CFR83.13(c)(1)].*

For members of the class of workers at TCC, NIOSH must have sufficient information to estimate the maximum radiation dose that could have been incurred in **plausible circumstances**. In the ensuing discussion, considerable attention is directed to ascertaining whether the approach proposed by NIOSH for dose reconstruction does, indeed, utilize doses that could have been incurred in plausible circumstances.

1.3 TECHNICAL APPROACH

The approach used by SC&A to perform this review follows the procedures described in the draft report prepared by SC&A entitled *Board Procedures for Review of Special Exposure Cohort Petitions and Petition Evaluation Reports*, Revision 1 (SC&A 2006) and the January 16, 2006, draft, *Report of the Working Group on Special Exposure Cohort Petition Review* (ABRWH 2006). The latter is a set of draft guidelines prepared by a Board-designated working group for evaluation of SEC petitions performed by NIOSH and the Board. The former is a draft set of procedures prepared by SC&A for the Board and approved by the Board for use by SC&A on an interim basis. The procedures are designed to help ensure compliance with Title 42, Part 83, of the *Code of Federal Regulations* (42 CFR 83) and implement the guidelines provided in the report of the working group.

The key considerations identified in the report of the working group include the following:

(1) Timeliness

(2) Fairness

(3) Understandability

(4) Consistency

(5) Credibility and validity of the dataset, including pedigree of the data, methods used to acquire the data, relationship to other sources of information, and internal consistency

(6) Representativeness and completeness of the exposure data with respect to the area of the facility, the time period of exposure, and the types of workers and processes covered by the data

The working group guidelines also recommend that NIOSH include in their SEC evaluations a demonstration that it is feasible to reconstruct individual doses for the cohort, including sample dose reconstructions.

The specific steps that SC&A usually implements in performing its SEC petition reviews include the following:

(1) Gathering and critically reviewing all documents and datasets cited by the petitioners and in the NIOSH evaluation report

(2) Meeting with former workers and petitioners to gain a richer understanding of the operations and incidents that took place during the time period covered by the petition, and to identify any additional records and data that may be pertinent to the review

Since this assignment is designated as a focused review, only step 1 was performed. Step 2 will be performed as directed by the Board or its designated working group.

NIOSH did not discover any dosimetry data or other exposure data (such as dust samples) at Texas City Chemicals (TCC) that would permit dose reconstruction at that facility using facility-specific data. Rather, NIOSH used surrogate data and very conservative estimates of exposure duration to bound the exposures of workers at TCC. As will be demonstrated, the generic approach adopted by NIOSH for performing dose reconstructions at TCC certainly appears to be bounding, but one of the important questions that we believe will need to be addressed by NIOSH and the Board is whether the approach is unrealistically conservative as applied to TCC workers and thereby raises issues regarding plausibility and fairness.

2.0 BACKGROUND INFORMATION

2.1 TEXAS CITY CHEMICALS, INC.

Texas City Chemicals, Inc. was incorporated in 1950 with the business objective of producing high-grade fertilizers and animal feed supplements from phosphate rock and recovering uranium as a by-product. TCC signed a letter contract (AT(49-1)-616) with the Atomic Energy Commission (AEC) on February 14, 1952. Under the terms of the contract, TCC was to commence construction of a fertilizer plant capable of processing 100,000 tons of phosphate rock annually, and to include a uranium recovery unit in the plant. The target completion date was October 1, 1953. Under the terms of the contract, the AEC would purchase all the uranium produced at a price based on 85% of the uranium concentrate production costs plus a fixed fee of \$3.75 per pound of contained U_3O_8. The AEC provided no financial support for the construction of the fertilizer plant and the uranium recovery unit.

The letter contract was superseded by a definitive contract (AT(49-1)-647) on May 12, 1953. Under the terms of the definitive contract, TCC was not obliged to produce more than 50,000 lb of U_3O_8 per year unless the AEC agreed to purchase any excess above that amount. The maximum price was limited to \$25 per lb of U_3O_8. The contract duration was for 5 years from the start of regular production or September 30, 1958, whichever came first.

With regard to safety, the contract specified the following:

> *3. Safety. The Contractor shall initiate and take all reasonable steps and precautions to protect health and minimize danger from all hazards to life and property and shall make all reports and permit all inspections as required by the Commission and shall conform to all minimum health and safety regulations and requirements of the Commission.*

An AEC memorandum noted that shakedown operations began on October 5, 1953 (Johnson 1953). A subsequent AEC production report stated that, because of production problems, the plant was shut down from January 1954 to December 1955 (AEC Monthly, December 1955). There is no evidence that operations resumed after December 1955. Prior to shutdown (i.e., from October 5, 1953, through December 1953), the plant had produced 303 lb of U_3O_8 from "intermittent shake-down operations." At some point (start date unknown), the AEC awarded TCC a development contract (AT(49-6)-910) that expired on September 30, 1955. Apparently, there was also another contract (AT(05-1)-481) whose terms and purpose are unknown, since the contract was destroyed (ERDA 1976). TCC filed for bankruptcy in July 1956 and was subsequently acquired by Smith-Douglass. Smith-Douglass did not pursue uranium recovery (Powers 1979).

At some unknown date, the uranium recovery building was demolished and all that remained was a concrete pad about 19 × 36 yards. The disposition of the demolition debris and building contents are unknown. The site was surveyed on November 17, 1977, by Oak Ridge National Laboratory (ORNL) at the request of the Department of Energy (DOE). At the time of the survey, the pad was used for phosphogypsum storage, but the facility was not operating then

(ORNL 1980). The survey found that gamma-radiation levels and Ra-226 concentrations in the soil were higher than local background levels. The gamma radiation levels were about the same as at other phosphate products plants where uranium recovery was not part of the process. The highest observed gamma exposure rate was 120 μR/hr at 1 meter above the surface. The highest observed beta-gamma dose rate at 1 cm above the surface was 0.25 mrad/hr. One soil sample contained Ra-226 at a level of 170 pCi/g. Uranium-238 in soil samples ranged from 4.5 to 15.3 pCi/g (ORNL 1980).

On November 20, 1985, DOE notified Amoco Chemical Company (then owner of the TCC site) that, although some residual contamination (elevated Ra-226 levels) existed at the site, the site did not qualify for clean-up under the Formerly Utilized Sites Remedial Action Program (FUSRAP 1985). The contamination was determined to be associated with fertilizer production, rather than AEC-related uranium recovery (Fritz 1985).

NIOSH uncovered no exposure data that would assist in dose reconstruction (ORAUT 2008), and consequently was forced to rely on surrogate data and conservative modeling assumptions.

2.2 FLORIDA INSTITUTE FOR PHOSPHATE RESEARCH ANALYSES

A major resource for obtaining surrogate data is the comprehensive study on exposures to technically enhanced naturally occurring radioactivity materials (TENORM) in the phosphate industry published in 1998 by the Florida Institute for Phosphate Research (FIPR 1998). Since this study will be frequently referenced in the ensuing discussions, some of the key aspects are summarized here.

2.2.1 External Exposures

To characterize external exposures, FPIR made a series of measurements using personal dosimeters on workers involved in six types of operations at five phosphate producers in Florida. The measurement approach involved co-badging workers with aluminum oxide carbon dosimeters (Landauer X9) and LiF thermoluminescent dosimeters (TLDs). The workers wore the LiF detectors for a period of 3 months. At the same time, three consecutive deployments of X9 dosimeters were made using the same workers. At least two of the X9 campaigns were concurrent with the LiF TLD campaign. Comparison of the X9 dosimeters results with those for LiF TLDs indicated that the X9 readings were consistently higher than the TLD readings by a factor of 1.22. Since the LiF TLD was judged to be the industry "gold standard," the X9 readings were adjusted accordingly. Results from these external dosimetry measurements are summarized in Tables 2-1, 2-2, 2-3 and 2-4 (FIPR 1998, Tables 14, 15, 16, and 17).

Table 2-1. Summary Statistics for X9 Dosimeter Deployment #1

Work Location	GM (mrem/yr)	GSD	Min. (mrem/yr)	Max. (mrem/yr)	Count
All personnel	15.0	2.7	4.1	184.4	151
Dry product areas	10.3	2.2	5.1	78.4	30
Shipping Areas	13.7	2.5	5.3	97.2	21
Mine Areas	9.6	2.1	4.1	32.1	21
Phosphoric Acid Areas	34.4	2.5	6.9	102.7	32
Rock Areas	24.3	3.1	5.3	141.3	16
Service Companies	10.2	2.4	4.8	184.4	31

Table 2-2. Summary Statistics for X9 Dosimeter Deployment #2

Work Location	GM (mrem/yr)	GSD	Min. (mrem/yr)	Max. (mrem/yr)	Count
All personnel	12.8	3.1	3.5	163.2	147
Dry product areas	8.1	2.7	3.6	80.9	28
Shipping Areas	12.2	2.9	3.7	124.1	20
Mine Areas	9.5	2.2	4.2	51.5	22
Phosphoric Acid Areas	34.8	2.7	4.1	163.2	32
Rock Areas	21.2	3.6	3.6	135.3	16
Service Companies	6.6	2.1	3.5	116.5	29

Table 2-3. Summary Statistics for X9 Dosimeter Deployment #3

Work Location	GM (mrem/yr)	GSD	Min. (mrem/yr)	Max. (mrem/yr)	Count
All personnel	14.7	2.5	3.6	186.3	133
Dry product areas	13.1	2.3	4.2	95.2	26
Shipping Areas	19.2	2.5	3.6	179.5	19
Mine Areas	11.4	2.3	6.2	186.3	21
Phosphoric Acid Areas	26.4	2.4	5.1	172.0	25
Rock Areas	19.4	2.9	6.2	119.7	13
Service Companies	8.7	1.8	5.7	103.8	29

Table 2-4. Summary Statistics for LiF TLD Deployment

Work Location	GM (mrem/yr)	GSD	Min. (mrem/yr)	Max. (mrem/yr)	Count
All personnel	20.9	2.0	6.5	209.9	148
Dry product areas	17.1	1.9	10.6	209.9	29
Shipping Areas	19.7	1.8	12.4	66.8	21
Mine Areas	16.8	1.5	8.1	56.5	22
Phosphoric Acid Areas	31.1	2.0	11.3	82.6	29
Rock Areas	30.0	2.4	6.5	128.4	16
Service Companies	17.5	1.7	8.6	166.5	31

2.2.2 Inhalation Exposures

In the FIPR study, 86 air samples were collected from production areas, including mine (16), rock (16), phosphoric acid (18), dry products (20), shipping (11), and service (5). The samples were analyzed for gross alpha [lower limit of detection (LLD) – 1 μCi/ml] and gross beta (LLD – 1.2 μCi/ml). Using the LLD in data analysis is conservative. It will overstate the mean, because values that are actually approaching zero are assigned minimum values equal to the LLD. The radionuclide content of the dust was determined by analyzing 17 samples of deposited dust in various locations with an HPGe detector. Based on the distribution of the results and dose conversion factors (DCFs) from International Commission on Radiological Protection (ICRP) Report No. 68 (ICRP 1994), FIPR used Crystal Ball® to develop probability distribution functions for inhalation exposures in the various work areas. Examples from the rock areas and phosphoric acid areas are presented in Figures 2-1 and 2-2.

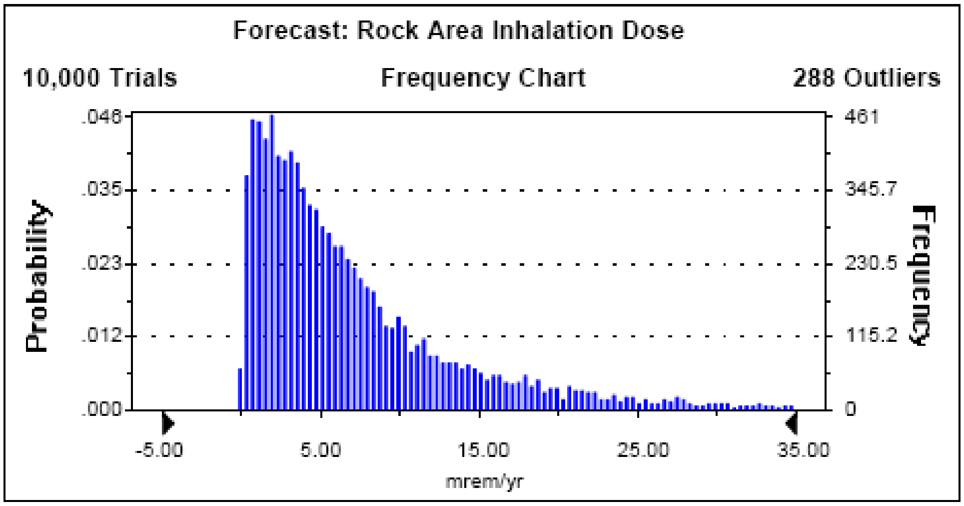

Figure 2-1. Rock Area Inhalation Dose Distribution (FIPR 1998, Figure 33)

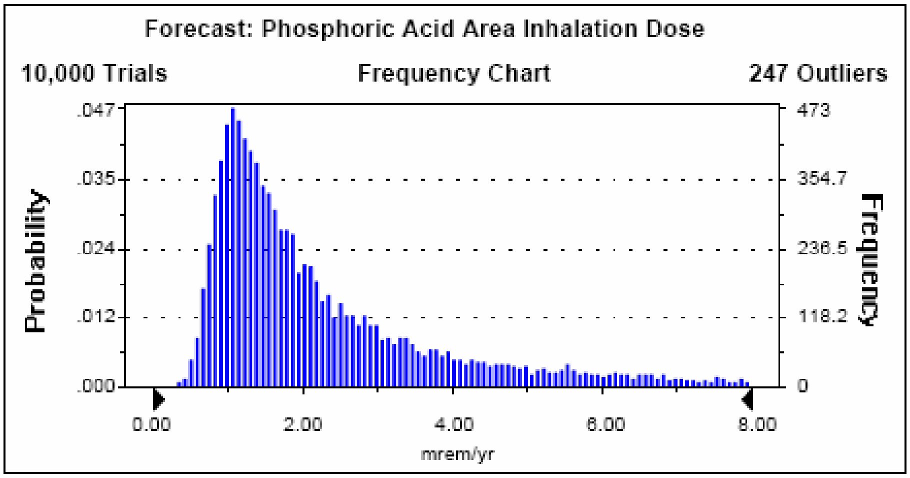

Figure 2-2. Phosphoric Acid Area Inhalation Dose Distribution (FIPR 1998, Figure 34)

Maximum inhalation exposures (excluding outliers) are about 35 mrem/yr for workers in the rock areas and 8 mrem/yr for workers in the phosphoric acid areas.

2.2.3 Radon Exposures

A number of radon studies were summarized in FIPR 1998, including prior studies (Appendix A, Table A-3 and Tables A-7 through A-14) and E-perm electrit ion chamber measurements taken as part of the then-current study (Appendix C, Table C-8). These summary tables are also included in Attachment B of ORAUT 2006. For all except Table A-3, where radon doses are presented in units of WLM/yr, the measurements are presented in terms of pCi/L. FIPR does not discuss radon equilibrium factors other than to note that "EPA assumes 50% equilibrium of daughters and thus the conversion of 4 pCi/L to 0.02 WL." According to UNSCEAR 2000 (Annex B, Section 123), **"**The range of the equilibrium factor for outdoor radon is from 0.2 to 1.0, indicating a degree of uncertainty in the application of a typical value to derive equilibrium equivalent concentrations." They further note "that a rounded value of 0.6 may be more appropriate for the outdoor environment than the previous estimate of 0.8." With regards to indoor equilibrium factors, it is noted in Annex B, Section 127 of UNSCEAR 2000 that, "Recent determinations of the equilibrium factor for radon indoors generally confirm the typical value of 0.4 previously assessed by the Committee. Indoor measurements show a range from 0.1 to 0.9, but most are within 30% of the typical value of 0.4."

3.0 NIOSH APPROACH TO DOSE RECONSTRUCTION

NIOSH was forced to rely on numerous bounding assumptions and surrogate data because, as stated in Section 6.0 of ORAUT 2008, "NIOSH did not find any TCC personnel or workplace monitoring records for the period under evaluation." While the bounding approach taken by NIOSH is clearly claimant favorable, the approach raises several questions, including the following:

- Does the approach meet the regulatory standards set forth in 42 CFR 83.13(c)(1)?
- Is the chosen approach "fair" from the perspective of dose reconstruction at other sites?
- Is the use of surrogate data consistent with draft criteria being developed by the Work Group on the Use of Surrogate Data?

These questions are considered in the subsequent sections. To the extent that surrogate data are used in the proposed dose reconstruction, an evaluation of this use of surrogate data against the Board criteria is presented in Section 4.

3.1 CHRONOLOGY

The SEC petition requested that "all laborers who worked in all areas at Texas City Chemical, Inc. from January 1, 1952 to December 31, 1956" be assigned to the petitioners' class. NIOSH amended the class definition to include all employees, instead of all laborers. Using the information discussed in Section 2.1 above, NIOSH assumed that from January 1, 1952, through December 31, 1952, employees would receive no exposure, since the initial AEC letter contract was not signed until February 14, 1952, and plant construction was assumed to be ongoing for the balance of the year. NIOSH further assumed that a pre-operational period occurred from January 1, 1953, until October 4, 1953. During this pre-operational period, employees might be exposed to radiation from raw phosphate rock delivered to the site in anticipation of the start-up of production operations.

Based on the start of intermittent shake-down operations on October 5, 1953, NIOSH assumed that this date delineated the start of the operational period. The operational period was assumed to continue through December 31, 1956, the end date requested by the SEC petitioners. NIOSH provides no justification for assuming that the operational period continued to December 31, 1956, even though there is no evidence that any yellowcake was produced after December 31, 1953 — 3 years earlier. This assumption is particularly important, since NIOSH assumed that exposure to yellowcake would be the dose-limiting process.

NIOSH did not consider the possibility of a post-operational period when workers might be exposed to residual contamination. Results of a post-operational site survey taken in 1977 are summarized in Section 2.1 above.

3.1.1 Class Definition

As noted in Section 3.1, NIOSH defined the petitioners' class as comprising all employees who worked in all areas at TCC from January 1, 1952, to December 31, 1956. Based on the available facts, we believe that the proposed period for the petitioners' class is open to question. These facts are summarized below:

(1) The SEC petition requested that "all laborers who worked in all areas at Texas City Chemical, Inc., from January 1, 1952, to December 31, 1956" be assigned to the petitioners' class. NIOSH amended the class definition to include all employees, instead of all laborers.

(2) TCC signed a letter contract (AT(49-1)-616) with the AEC on February 14, 1952. Under the terms of the contract, TCC was to commence construction of a fertilizer plant capable of processing 100,000 tons of phosphate rock annually and to include a uranium recovery unit in the plant. The target completion date was October 1, 1953. Under the terms of the contract, the AEC would purchase all the uranium produced at a price based on 85% of the uranium concentrate production costs plus a fixed fee of $3.75 per pound of contained U_3O_8. **The AEC provided no financial support for the construction of the fertilizer plant and the uranium recovery unit**.

(3) The letter contract was superceded by a definitive contract (AT(49-1)-647) on May 12, 1953. Under the terms of the definitive contract, TCC was not obliged to produce more than 50,000 lb of U_3O_8 per year unless AEC agreed to purchase any excess above that amount. The maximum price was limited to $25 per lb of U_3O_8. The contract duration was for 5 years from the start of regular production or September 30, 1958, whichever came first.

(4) At some point (start date unknown), the AEC awarded TCC a development contract (AT(49-6)-910) of unknown scope that expired on September 30, 1955.

(5) From October 5, 1953, through December 1953, the plant had produced 303 lb of U_3O_8 from "intermittent shake-down operations." There is no documentation that the AEC actually purchased this material.

(6) Because of production problems, the plant was shut down from January 1954 to December 1955. There is no evidence that production resumed after December 1955.

(7) TCC filed for bankruptcy in July 1956, and was subsequently acquired by Smith-Douglass. Smith-Douglass did not pursue uranium recovery.

The question that needs to be carefully examined is, **What should be the period for the SEC petitioners' class?**

NIOSH and petitioners say the period should be January 1, 1952 through December 31, 1956. No justification for the end date was provided. Alternatively, should it be the period of the AEC contracts – February 14, 1952 through September 30, 1958? Or since the contact only provides that AEC purchase yellowcake, should it be the period during which yellowcake was produced from intermittent shakedown operations—October 5, 1953 through December 31, 1953? Or should it be the period from February 14, 1952 (the date of the letter contract) until July 1956, since Smith-Douglass says they were not involved in uranium production? We also note that NIOSH did not consider post-operational exposures.

We believe these and perhaps other alternatives should be considered in defining the duration of potentially compensatory activities at TCC.

Finding 1: The petitioners' class definition needs to be re-examined to insure that the time period has been properly justified.

3.2 EXTERNAL EXPOSURE

3.2.1 External Doses Away from Uranium Recovery Area

3.2.1.1 Pre-Operational External Exposures

NIOSH assumes that workers could have received external exposures from phosphate rock prior to the start of operations on October 5, 1953, since some inventory build-up of raw materials was likely prior to start-up. Exposure to phosphate rock was assumed to occur from January 1, 1953, to October 4, 1953, at an annual dose rate of 70 mrem (ORAUT 2008, Section 7.3.4.1). Photon energies were assumed to have the range of 50% 30-250 keV and 50% >250 keV. The annual dose rate of 70 mrem is taken from ORAUT-OTIB-0043 (ORAUT 2006, Section 4.1.1), which is based on an NCRP report (NCRP 1993) citing exposure to phosphogypsum stacks for 2,000 hours per year. In OTIB-0043, the annual dose rate of 70 mrem is the assumed geometric mean (GM) of a lognormal distribution with a geometric standard deviation (GSD) of 2.00 and a 95th percentile value of 220 mrem/yr. Tracing this datum to the NCRP report (NCRP 1993, Section 9.6.3.5), we find the following:

> *Personnel working on phosphogypsum piles for 2,000 h per y would have annual gamma radiation doses not exceeding 0.7 mGy per y; annual doses would be proportionally lower for lower occupancy times and would be considerably lower for work with small amounts.*

First, we observe a discrepancy in the units—0.7 mGy corresponds to 70 mrad, not mrem. OCAS-IG-001 (Taulbee 2002) lists DCFs for external exposure to photon radiation based on four types of measured or calculated values: (1) deep dose equivalent, $H_p(10)$; (2) ambient dose equivalent, $H^*(10)$; (3) exposure; or (4) air kerma. The first two quantities may be expressed in mrem, the third in mR, and the fourth in mrad. Dose reconstructors need clear guidance as to which set of DCFs to employ for a given exposure scenario.

It is possible that raw phosphate rock was on the premises at TCC from the beginning of the pre-operational period (January 1, 1953, through cessation of TCC operations in July 1956), creating a continuing source of external exposure. During the operational period, NIOSH assumed that exposure to yellowcake was limiting. Since no phosphogypsum stacks existed during the pre-operational period and only limited fertilizer production occurred during the operational period (Wilkinson 1976), a picture of dose rates under more plausible circumstances could be derived from the extensive database developed by the FIPR (FIPR 1998) and discussed in Section 2.2.

For a more representative source of surrogate data for the pre-operational period than phosphogypsum stacks, one could consider selecting relevant data from Tables 2-1 through 2-4. For example, one could select dose rate data from the rock areas as a possible surrogate for ore handling operations. As shown in these tables, the GMs were 24.3, 21.2, 19.4, and 30.0 mrem/yr for the rock areas. Use of the rock area data is clearly claimant favorable, since the GM values are higher than for any other operational area except phosphoric acid, and phosphoric acid was not produced during the pre-operational period. Furthermore, as was argued above, use of the GM as the metric in dose reconstruction may not be appropriate, given how little is known about TCC processing. The 95th percentile dose rates calculated from the data in Tables 2-1 through 2-4 are 156, 174, 112, and 127 mrem/yr, respectively. The 95th percentile for the entire population of X9 dosimeter measurements from the rock areas is 150 mrem/yr.

Finding 2: The use of mean annual external exposures observed among phosphogypsum workers in Idaho does not appear to be a scientifically sound or claimant-favorable surrogate for pre-operational phosphate ore handlers at TCC. Consideration should be given to using the upper 95th percentile exposures for Florida phosphate rock workers as a more appropriate claimant-favorable surrogate.

3.2.1.2 Operational External Exposures

NIOSH notes in Section 5.2.3.1 of ORAUT 2008 that during the operational period, workers inside the uranium recovery building were routinely exposed to full drums of uranium concentrate, even though only 303 pounds (less than ½ of a 55-gallon drum) of product were produced at TCC.

NIOSH did not develop an external exposure model for workers not involved in uranium production during the operational period, since they assumed that exposure to uranium concentrates would be limiting and applied to all production workers. However, if one takes an alternative approach to bounding yellowcake exposures, external exposures from operations outside of uranium recovery, such as phosphoric acid production, could become more significant. In the survey discussed in Section 2.2, FIPR measured external exposures to phosphoric acid production workers and determined that the GM values from X9 dosimetry were 34.4 mrem/yr (32 samples), 34.8 mrem/yr (32 samples), and 26.4 mrem/yr (25 samples). The equivalent 95th percentile values are 155, 178, and 111 mrem/yr, respectively. The 95th percentile value for the entire population of X9 dosimeter measurements in phosphoric acid areas is 151 mrem/yr—essentially the same as for the rock areas.

3.2.2 External Doses in Uranium Recovery Area

NIOSH used the same approach to estimate external exposures in the uranium recovery building as was proposed for Blockson Chemical (ORAUT 2004a). To calculate the median dose, NIOSH assumed that a worker spent 400 hours per year 1 foot from a drum of yellowcake with a U_3O_8 density of 6 g/cm^3. (A 55-gallon drum filled with U_3O_8 at a density of 6 g/cm^3 would contain about 2,750 lb of yellowcake.) NIOSH calculated an air kerma rate from primary photons (gamma rays and characteristic x-rays) plus bremsstrahlung of 6.43 mrad/hr, resulting in a median air kerma of 2.572 rad/y. NIOSH used an exposure duration of 2,000 hours per year to calculate the 95th percentile air kerma, which results in 12.86 rad/y. Such a distribution has a GSD of 2.7.

To explore the range of possible exposures, SC&A considered an alternative scenario to model external exposure at TCC. Since only 303 lb of U_3O_8 was produced at TCC, we assume that this amount was stored in a 30-gal steel drum. Assuming a density of 2 g/cm^3, this amount would fill a little more than one-half of the drum.

We assume that the uranium isotopes are present in the ratios of their natural abundance. We assume that U-238 is in secular equilibrium with its short-lived progeny, that U-235 is in equilibrium with its entire decay chain, and that the activity of Ra-226, which is in secular equibrium with its entire decay chain, is equal to 10% of the activity of U-238. We further assume that the activity of Th-232, which is in secular equilibrium with its entire decay chain, is equal to 3.3% of the activity of U-238.

We calculated the air kerma rate at a distance of 1 ft (30.48 cm) from the edge of the drum, at a height of 1 m above the floor, a height that corresponds to the lowest position of most major organs. The resulting value is 0.65 mrad/h, which is about 10% of the value reported by NIOSH for a full 55-gal drum at a lower height above the floor. We also calculated the skin dose from non-penetrating (beta) radiation at the same location to be 0.85 rad/h. A detailed description of this analysis is presented in Appendix C.

We further assume an exposure duration of 12 workdays (1 day per week for the 3-month intermittent shakedown period), or 96 hours. The total doses during this period would be 62 mrad air kerma and 82 mrad beta dose to the skin, substantially lower values than calculated by NIOSH.

Finding 3: An alternative modeling approach based on reasonable assumptions derived from the available information suggests that external doses could be two orders of magnitude lower than those developed by NIOSH. NIOSH should reexamine its modeling approach to insure that its assumptions for calculating external doses are representative of plausible circumstances.

3.3 PROCESS-RELATED INTERNAL DOSES AT TCC

To reconstruct internal doses, information must be available on exposure rate and duration. With regard to exposure duration, NIOSH assumed that, "For the purposes of this evaluation, the Operational Period is assumed to be continuous from October 5, 1953 through December 31, 1956" (ORAUT 2008, Section 7.2.1). They also assumed a pre-operational period from January 1, 1953, to October 4, 1953. They did not assume a post-operational period where workers could be exposed to residual contamination.

3.3.1 Pre-Operational Internal Exposures

For the pre-operational period from January 1, 1953, through October 3, 1953, NIOSH assumed that workers would be exposed to dust from ore handling operations. This is a reasonable, claimant-favorable premise. For this period, NIOSH assumed that workers were exposed to a dust level of 5.43 mg/m^3 (ORAUT 2008, Section 7.2.1). This value, for a single air sample taken during a 1975 EPA study of a phosphate plant in Idaho, was indicative of dust loadings during ore unloading and storage (EPA 1978). NIOSH notes that the Idaho facility used a wet process for phosphoric acid production similar to that used by TCC. Using this dust loading and assuming that the phosphate rock contained 0.014% U (Stoltz 1958), one can calculate that the inhalation rate from U-238 (as well as U-234, Ra-226, Th-230, Pb-210, and Po-210) was 1.64 pCi per calendar day (5.43 mg ore/m^3 × 1.4×10^{-4} mg U/mg ore × 1 g U/1,000 mg U × 1.2 m^3/hr × 8 hr/work day × 250 work days/365 calendar days × 3.3×10^{-7} Ci/g U-238 × 10^{12} pCi/Ci × 0.997gU-238/gU = 1.64 pCi/calendar day). Based on a committed effective dose equivalent for Type M U-238 of 9,620 mrem/μCi, from ICRP Report No. 68 (FIPR1998, Table 12), this exposure would result in a dose rate of 6 mrem/yr for that radionuclide (1.64 pCi/calendar day × 365 calendar days/yr × 9,620 mrem./μCi × 1 μCi/10^6 pCi).

As noted above, this estimate is based on a single dust loading measurement. The broader database available in FIPR 1998 could have been used to establish confidence that the upper bound exposure had been captured. For example, data from the rock areas shown in Figure 2-1 might provide a more realistic surrogate reflecting exposure variability.

Finding 4: NIOSH should consider data from FIPR 1998 as an alternative source of data for estimating internal exposures.

3.3.2 Operational Internal Exposures

During the operational period defined by NIOSH, intermittent shakedown operations from October 5, 1953, to December 1953, resulted in production of about 300 lb of U_3O_8. The plant was shut down at least from January 1954 through December 1955, and possibly through July 1956. According to Wilkinson (1976), problems were encountered not only with uranium by-product recovery, but also with the basic phosphate plant. For some portion of that shutdown period, development work may have been done, but no uranium concentrates were reported to have been produced, and the nature and scope of the development work are unknown. It is likely that all uranium-related activities ceased by July 1956, when TCC declared bankruptcy. Even though NIOSH defines the operational period as about 39 months, the limited record provided indicates that workers were actually exposed to quantifiable amounts of uranium concentrates (yellowcake) for only 3 months. It appears that the proposed duration of yellowcake exposure is more than a factor of 10 higher than experienced by TCC workers.

Finding 5: The assumption that workers were exposed to yellowcake for 39 months, while claimant favorable, is not consistent with the available data and may overstate the exposure by an order of magnitude.

3.3.2.1 Internal Exposures Outside the Uranium Recovery Building

During the operational period, workers outside the uranium recovery plant were subject to exposure to dust from uranium-bearing phosphate ores generated from processes such as ore handling. One source indicates that phosphate ore arrived by barge and was trucked (by contractor) to TCC, where it was stored in hoppers (Witt 2000). According to Witt (2000), hauling may have occurred over a period of 2 ½ years. Conceivably, this could cover the pre-operational period and a portion of the operational period (e.g., from January 1, 1953, through June 30, 1955). There is no evidence to indicate that any pre-treatment of the ore was required prior to digestion in sulfuric acid.

NIOSH assumes that exposure to workers outside the uranium recovery building would be higher during the operational period than during pre-operational period, as a result of operations such as crushing of the phosphate rock. While it is reasonable to assume that dust levels were higher during the operational period, NIOSH has provided no evidence that rock crushing was performed at TCC. At Blockson Chemical, the ore was "pulverized" after calcining (Stoltz 1958), but there is no information that similar unit operations were used at TCC. Flow diagrams presented in FIPR 1998 show that crushing, grinding, and sizing operations produce rock concentrates and high-grade rock that are the feedstocks for the wet acid process. It is possible that TCC purchased these feedstocks from Florida and no further size reduction was required at TCC. While the assumption that rock crushing with associated higher dust levels is claimant favorable, it may not be representative of processing at TCC.

Since there was no exposure information for outside operators available at the TCC site, NIOSH assumed that the dust levels would be similar to those experienced by operators at a phosphate plant in Idaho that was surveyed by the Environmental Protection Agency (EPA) in 1975 (EPA 1978). NIOSH proposed the same approach for TCC as used at Blockson Chemical (ORAUT 2004). They calculated a dust level of 50.4 mg/m^3, which was associated with workers involved in calcining (EPA 1978, Table 12), a unit operation not used at TCC, but selected by NIOSH as a claimant-favorable bounding value. As stated in ORAUT-2007, "The highest dust concentration in that study was from Calciner #3, which was 50.4 mg/m^3, indicative of an operation with likely visible dust." To obtain this dust loading, NIOSH used the mass of dust collected (0.2742 g) during the flow of 5.44 m^3 of air through a particulate sampler. As was the case for estimating internal exposures during the pre-operational period, this is based on a single sample. The U-238 exposure associated with this dust loading was 1.0 pC/m^3 (EPA 1978, Table 12). In fact, the highest uranium exposure rate reported in EPA 1978 (Table 14) was associated with the triple super phosphate (TSP) dryer where a U-238 concentration of 1.4 pCi/m^3 was measured.

To develop an exposure rate from the dust level of 50.4 mg/m^3, NIOSH further assumed that the phosphate ore contained 0.014% U, and that outside workers, with a normal breathing rate of 1.2 m^3/hr, were exposed for 2,000 hours per year. The uranium concentration in the ore was based on an estimate developed by Stoltz (1958) for Blockson Chemical. While this concentration is probably a reasonable value for ore handled at TCC, there is no information to support that assumption. On this basis, NIOSH estimated that the exposure rate to U-238 (as well as U-234, Ra-226, Th-230, Pb-210, and Po-210) for outside workers was 16 pCi per calendar day (ORAUT 2008, Table 7-1). (50.4 mg ore/m^3 × 1.4×10^{-4} mg U/mg ore × 1 g U/1000 mg U × 1.2 m^3/hr × 8 hr/work day × 250 work days/365 calendar days × 3.3×10^{-7} Ci/g U-238 × 10^{12} pCi/Ci × 0.997gU-238/gU = 15.3 pCi/calendar day).

Using the EPA reported exposure of 1 pCi/m^3 from the same measurements that were used by NIOSH to calculate a dust loading of 50.4 mg/m^3, one can calculate a U-238 dust exposure rate of 6.6 pCi/calendar day. The difference between 6.6 and 15.3 pCi/day cannot be explained based on differences in ore composition. EPA measured the U-238 activity in ore discharged from the calciner as 38 pCi/g (EPA 1978, Table 3). Based on the specific activity of U-238 of 3.3×10^{-7} Ci/g, the uranium content of the ore discharged from the calciner would be 0.012% (38 pCi/g U-238 × 1 g U-238/3.3×10^{-7} Ci × 1 Ci/10^{12} pCi × 1 g U/0.993 g U-238 ×100). This suggests that the ore processed at the Idaho phosphate plant had a U content similar to that at Blockson (0.012% vs. 0.014%).

In its review of the Blockson technical basis document, SC&A did not take issue with this use of surrogate data because internal exposures were dominated by those received by ongoing yellowcake production (SC&A 2007). However, in the case of TCC, use of the same surrogate data needs to be examined more carefully with regards to meeting the requirement that radiation doses can be estimated with sufficient accuracy, as required by 42 CFR 83.13(c)(1).

In addition, use of a single measurement for the dust loading raises the question as to whether the expected range of exposures has been captured. An alternate approach might take advantage of the information presented in FIPR 1998. In that study, 86 air samples were taken from all operations associated with phosphate rock processing, including mining, beneficiation, phosphoric acid production, and shipping (FIPS 1998, Table C-6). After eliminating mining from the dataset as non-applicable, the maximum measured value for gross alpha was 2.3×10^{-11} µCi/ml in the rock area. This can be converted to a total alpha exposure of 151 pCi/day (2.3×10^{-11} µCi/ml × 10^6 ml/m^3 × 10^6 pCi/µCi × 1.2 m^3/hr × 8 work hours/workday × 250 workdays/365 calendar days). Assuming that six longer-lived alpha emitters (U-238, U-234, Ra-226, Po-210, Pb-210, Th-230) in the U-238 decay chain contribute to the gross alpha measurement, the U-238 exposure rate would be about 25 pCi/calendar day. This value is slightly higher than the value of 16 pCi/calendar day developed by NIOSH, but was taken from a more comprehensive dataset representative of a range of operations at several phosphate fertilizer plants.

This analysis neglects the contribution of Th-232 and its progeny. For high-grade phosphate rock with an activity 26.37 pCi/g for U-238, the comparable value for Th-232 is 1.91 pCi/g (FIPR 1998, Table 21).

The U-238 exposure rate of 25 pCi/calendar day can be converted to a dose rate of 88 mrem/yr based on a DCF of 9,620 mrem/µCi for U-238. Similar dose rates can be developed for other alpha-emitting radionuclides in the U-238 decay chain based on the assumption of secular equilibrium for rock concentrates. This approach uses the maximum measured value from the gross alpha sampling in the rock area of 2.3×10^{-11} µCi/ml reported by FIPR.

Another alternative approach would be to use the dose rate data developed by FIPR. Results for two work areas are illustrated in Figures 2-1 and 2-2 above, which take into account the statistical distributions from particulate air sampling. FIPR developed a statistically based approach using Crystal Ball® to calculate annual dose rates using air sampling data, time and motion studies, etc. The probability distribution function for the rock areas is shown in Figure 2-1. This probability distribution function has a maximum indicated value of 35 mrem/yr total effective dose equivalent (TEDE). These examples are provided as illustrations of how other data might be used to bound exposures at TCC using a broad database, rather than a single measurement.

Finding 6. NIOSH should consider other data sources in addition to EPA 1978 to estimate internal exposures to workers outside the uranium recovery building, such as FIPR 1998.

3.3.2.2 Internal Exposures Inside the Uranium Recovery Building

As described by NIOSH, production operations began on October 5 1953, and were shut down in December 1953, due to processing problems (ORAUT 2008, Table 5-1). During this period, only 303 lb of U_3O_8 were produced. At some unknown point in time, AEC issued a development contract, possibly to address the production problems, and this contract expired on September 30, 1955 (ORAUT 2008, Section 5.1). Another contract of unknown scope and duration was also issued by AEC. NIOSH assumed that all AEC-related activity ceased on December 31, 1956 (the petitioner-requested SEC class end date).

Since no exposure data were available, NIOSH assumed that inhalation exposures in the uranium recovery building could be quantified using surrogate data from Christofano and Harris 1960. These authors summarized the results from a large number of surveys of various unit operations in the uranium fuel cycle, including uranium ore handling and sampling, uranium concentrate handling and sampling, acid digestion, solvent extraction, denitration, oxide reduction, etc. NIOSH reasoned that, since maximum inhalation exposures at TCC would occur when the yellowcake was dried and drummed, the concentrate handling and sampling data from Christafano and Harris 1960 would provide an acceptable surrogate from concentrate handling at TCC. Table 2 of Christafano and Harris 1960 reported daily average exposures ranging from 40 to 190 dpm/m^3, with an average from the various surveys of 140 dpm/m^3. NIOSH chose the upper limit of these daily average exposures (190 dpm/m^3) and converted this to a U-238 exposure of 281 pCi/calendar day, based on a breathing rate of 1.2 m^3/hr and 2,000 hr per year (190 gross alpha dpm/m^3 × 1.2 m^3/hr × 8 hr/work day × 1 pCi/2.22 dpm × 0.5 U-238 dpm/gross alpha dpm × 250 work days/365 calendar days = 281 U-238 pCi/calendar day). The same exposure was expected from U-234, Po-210, Th-230, and Pb-210. Ra-226 was assumed to have been removed during ore processing and, therefore, was not an important contributor to internal dose.

The Christofano and Harris surveys of concentrate sampling were taken at three facilities, two of which were operating in 1960 and one was in a standby mode. One of these sampling plants was located at the Fernald Environmental Management Project (FEMP). Operations at FEMP are described as follows (ORAUT 2004b, Section 2.2.2.1):

> *The sampling operation included a number of supporting operations. Several large-scale systems existed for crushing, grinding, and blending solid materials. These systems had a combined capacity of more than 10 tons per hour. Major equipment included hammer mills, ring-roll mills, and falling stream samplers. Some of this equipment was shielded for handling radioactive materials. Special dust collecting and ventilating equipment permitted the processing of toxic and radioactive materials. Enriched uranium slag and selected recycle materials*

> *were processed through a ring-roller mill for reuse in the production of uranium derby metal or for chemical processing to UO_3 in the refinery. This equipment could reduce particulate size to 95% minus 325-mesh at a rate of up to 9.1 tons per day.*

It is questionable that the use of FEMP data based on large-scale production meets the regulatory test of estimating "maximum doses … incurred in plausible circumstances" for an operation that produced only a few hundred pounds of yellowcake. If surrogate data must be used, a possible alternative approach would be to consider the urinalysis data available from uranium recovery operations at Blockson Chemical (ORAUT 2004a). Blockson production operations began in August 1952 and continued through 1962. During the period September 1952 through June 1960, Blockson produced 118.3 tons of U_3O_8. For these data, the 95th percentile exposure was calculated to be 82 pCi/d (50% U-238 and 50% U-234) (ORAUT 2007). Use of the 95th percentile Blockson data as a surrogate for TCC uranium exposures is bounding, claimant favorable, and scientifically more robust than using data from uranium concentrate sampling operations. The method adopted by NIOSH would assume that TCC workers experienced substantially higher exposures than workers at Blockson, which does not appear to be plausible, given the relatively small quantity of yellowcake produced at TCC. It could also be argued that the exposures at Blockson are overly conservative as applied to TCC.

Finding 7: The approach used in the evaluation report to reconstruct internal exposures to uranium production operations appears to be unrealistically conservative with regard to exposure duration and exposure level.

3.3.3 Inadvertent Ingestion

The estimates of internal exposure via inadvertent ingestion at TCC are based on OCAS-TIB-0009 (Neton 2004). SC&A has previously taken issue with this methodology. We believe that this methodology is neither scientifically correct nor claimant favorable. Since the rates of inadvertent ingestion cannot be rigorously quantified, we suggest that NIOSH consider adopting the value recommended by EPA in the "Exposure Factors Handbook" (EPA 1997). The value recommended for the assessments of adults exposed to contaminated soil is 50 mg/day. By comparison, the methodology prescribed by Neton yields a value of 1 mg per 8-hour day for a dust loading of 5 mg/m^3, a high-end dust loading equal to the Occupational Safety and Health Administration (OSHA) permissible exposure limit (PEL) for concentrations of respirable nuisance dusts. This is only 2% of the EPA-recommended value. Lower dust loadings would yield correspondingly lower ingestion rates. Since the settling of airborne dust is only one mechanism that leads to intake by inadvertent ingestion, there is no simple relationship between the ingestion rate and airborne concentrations, contrary to the assumptions made by Neton.

Finding 8. The methodology used to estimate inadvertent ingestion should be revised.

3.4 RADON EXPOSURE

NIOSH states in the TCC evaluation report that, "elevated radon concentrations are assumed to be present throughout the facility" (ORAUT 2008, Section 5.2.3.2). To quantify this premise, NIOSH assumes that 95th percentile radon exposure values were 0.112 WLM/yr for both the pre-operational and operational periods. This value was obtained from an analysis presented in ORAUT 2006. NIOSH used radon measurements reported by FIPR (1998), selecting a subset of about 130 values ascribed to chemical plant operations. The subset was selected to be representative of operations relevant to uranium extraction. As such, mining and rock loading operations were excluded. The GM of the data subset was 0.751 pCi/L with a GSD of 1.989, fixing the 95th percentile value at 2.33 pCi/L (ORAUT 2006, Section 4.2). Using an equilibrium factor of 0.4, NIOSH calculated a mean radon exposure of 0.036 WLM/yr and a 95th percentile value of 0.112 WLM/yr.

As discussed in Section 2.2.3, the outdoor equilibrium factor suggested by UNSCEAR is 0.6. Since radon exposures during the pre-operational period were likely to have been incurred by outside workers, adjustment of exposures using 0.6 rather than 0.4 would be more claimant favorable during the pre-operational phase.

For the pre-operational period, NIOSH used radon exposures expected from chemical plant operations. Since activities during the pre-operational period were more likely to have involved rock handling, an alternative approach to estimating radon exposures during this period would be to use exposures reported in FIPR 1998 and ORAUT 2006 for wet rock loading. Table B-2 in ORAUT 2006 summarizes data based on 78 measurements taken by several companies. The 95th percentile of these values is 0.244 WLM/yr. (See Appendix A, Section A.3 of this report for details of the calculations.)

As described above, NIOSH used a subset of about 130 measurements to calculate the mean and GSD of the radon measurements. Examination of Table B-3 of ORAUT 2006 shows that data reported as 8 individual measurements were actually mean values of 593 total measurements. Since the possibility exists that treating the means as single values could distort the statistics for the population, SC&A expanded the dataset using other statistics included in Table B-3. Details of these calculations are summarized in Appendix A, Sections A.1 and A.2. The 95th percentile of the expanded population is 7.78 pCi/L, rather than 2.33 pCi/L calculated by NIOSH for the reduced dataset. Using the value of 7.78 pCi/L and an equilibrium factor of 0.6, one would obtain an alternative bounding estimate of radon exposures to outdoor workers during the operational period of 0.56 WLM/yr. NIOSH may wish to consider this approached to deriving surrogate radon exposures.

It should be pointed out that these suggestions were provided to the working group in an SC&A white paper related to Blockson and discussed during a Blockson working group meeting held on June 5, 2008. During that meeting, NIOSH acknowledged that the recommended default value of 2.33 pCi/L of radon is more representative of the 95th percentile of a set of means of individual measurements than the 95th percentile of a set of individual measurements. However, NIOSH provided convincing arguments that, since no one individual would be expected to be **continually** exposed to the upper 95th percentile value, it is more appropriate to use the 95th percentile of the mean, as opposed to the 95th percentile of the individual values. SC&A concurred with this philosophy, but stated that ORAUT-OTIB-0043 would benefit from a thorough discussion of this matter. We believe that this same consideration should be given to the TCC site profile and evaluation report.

Finding 9: NIOSH should consider adjustments to the dataset used to calculate radon doses to fully reflect the available information. This would increase the dose from 0.112 WLM/yr to 0.56 WLM/yr. Should NIOSH determine that the 95th percentile of the means is, in fact, a more appropriate surrogate value, the rationale for this conclusion should be provided in the site profile.

4.0 EVALUATION OF THE SURROGATE DATA USED FOR TCC DOSE RECONSTRUCTION AGAINST THE DRAFT SURROGATE DATA CRITERIA DEVELOPED BY THE BOARD

Because no site-specific data on exposures and doses exist for TCC, NIOSH relied on surrogate data to develop dose reconstruction guidelines. Surrogate data usage, discussed in detail in Section 3, is summarized in Table 4-1.

Table 4-1. Summary of Surrogate Data Used In TCC Dose Reconstruction

Exposure Pathway	Exposure Period	Exposure Location	Dose Reconstruction Data
Internal	Pre-operational	All	1.6 pCi/day for U-238, U-234, Ra-226, Po-210, and Pb-210. This is based on a single dust loading measurement of 5.43 mg/m^3 taken during ore handling at an Idaho phosphate plant.
	Operational	Outside U recovery bldg.	16 pCi/day for U-238, U-234, Ra-226, Po-210, and Pb-210. This is based on a single dust loading measurement of 50.4 mg/m^3 at the calciner at an Idaho phosphate plant.
	Operational	Inside U recovery bldg.	190 dpm/m^3. This is the maximum reported daily average for plants sampling ore concentrates.
External	Pre-operational	All	70 mrem/yr. This is the GM photon dose based on exposure to phosphogypsum stacks for 2,000 hours per year.
	Operational	Outside U recovery bldg.	Same as inside U recovery bldg.
	Operational	Inside U recovery bldg.	5.53E-03 to 6.47E-03 rad/hr total dose rate based on exposure at 30 cm from drum of yellowcake.
Radon	Pre-operational/ Operational	All	0.112 WLM. This is the 95th percentile value for “chemical plant” operations at Florida phosphate producers based on an equilibrium factor of 0.4.

These uses of the surrogate data were tested against the draft criteria developed by the Work Group on the Use of Surrogate Data (see Appendix B). An evaluation of each use is presented below, based on the information summarized in Table 4-1.

4.1 INTERNAL EXPOSURES

4.1.1 Internal Pre-Operational Exposures

- Hierarchy of data – This criterion specifies that surrogate data at the same hierarchy level be used “only after the appropriate adjustments have been made to reflect the uncertainty of this substitution.” No adjustments were made in the proposed dust level to reflect uncertainties related to the substitution. The selected surrogate datum was a single measurement of airborne dust in the ore handling area.
- Exclusivity constraints – This criterion states that, in cases where little or no monitoring data are available, the use of surrogate data “would need to be stringently justified.” It is not clear that the use of the EPA data were stringently justified. NIOSH did not consider other data sources, such as the extensive reporting by the FIPR (e.g., FIPR 1998)

- Site or process similarities – The selected dust loading was based on measurements taken during ore handling. This operation should be similar to activities at TCC.
- Temporal considerations – Surrogate data should be taken from the same general period as that for which doses are to be reconstructed. The selected dust loading was based on measurements taken about 20 years later than TCC operations. However, Birky (2005) of the FIPR has observed that there have been no changes in the construction of wet process acid plants since the time of Atomic Weapons Employer (AWE) operations.

4.1.2 Internal Operational Exposures – Outside Uranium Recovery Building

- Hierarchy of data – This criterion specifies that surrogate data at the same hierarchy level be used "only after the appropriate adjustments have been made to reflect the uncertainty of this substitution." No adjustments were made in the proposed dust level to reflect uncertainties related to the substitution. The selected surrogate datum was a single measurement of airborne dust at an ore calcining operation.
- Exclusivity constraints – This criterion states that in cases where little or no monitoring data are available, the use of surrogate data "would need to be stringently justified." It is not clear that the use of the EPA data was stringently justified. NIOSH did not consider other data sources, such as the extensive reporting by the FIPR (e.g., FIPR 1998).
- Site or process similarities – The selected dust loading was based on measurements taken during operation of a calciner at an Idaho phosphate plant. There is no information to suggest that a calciner was used at TCC. Use of FPIR data as an alternative could have resulted in setting a value for U-238 of 25 pCi/d/calendar day, rather than a value of 16 pCi/calendar day. While the difference is probably not significant, it can be argued that use of the FIPR data would be more consistent with this work group draft criterion.
- Temporal considerations – Surrogate date should be taken from the same general period as that for which doses are to be reconstructed. The selected dust loading was based on measurements taken about 20 years later than TCC operations. However, Birky (2005) of the FIPR has observed that there have been no changes in the construction of wet process acid plants since the time of AWE operations.

4.1.3 Internal Operational Exposures – Inside Uranium Recovery Building

- Hierarchy of data – This criterion specifies that surrogate data at the same hierarchy level be used "only after the appropriate adjustments have been made to reflect the uncertainty of this substitution." To reflect uncertainty in the substitution, NIOSH selected the maximum reported activity level of 190 dpm/m^3 from a range of values measured during sampling of ore concentrates.
- Exclusivity constraints – This criterion states that in cases where little or no monitoring data are available, the use of surrogate data "would need to be stringently justified." It is not clear that the use of the data from sampling ore concentrates on a production scale as a surrogate for limited yellowcake handling at TCC was stringently justified.

- Site or process similarities – The selected exposure level was based on measurements taken during concentrate sampling operations at AEC production facilities where lidding and delidding of drums was a major source of exposure (Christofano and Harris 1960). The similarities between these operations and an operation that produced less than one drum of product are few. Bioassay data from Blockson Chemical workers are an alternative that NIOSH might consider as a better fit with regard to process similarities. However, even that application is strained, given the differences in the processes and throughput between Blockson and TCC.

- Temporal considerations – Surrogate date should be taken from the same general period as that for which doses are to be reconstructed. The selected exposure measurements were taken over the period 1948–1958, which is contemporaneous with the TCC operations.

4.2 EXTERNAL EXPOSURES

4.2.1 External Exposures – Pre-Operational Period

- Hierarchy of data – This criterion specifies that surrogate data at the same hierarchy level be used "only after the appropriate adjustments have been made to reflect the uncertainty of this substitution." No adjustments were made in the proposed dose rate to reflect uncertainties related to the substitution. Although the assumed GM value of the surrogate data was selected, one could argue that use of the 95th percentile value should be used to reflect uncertainty in the substitution.

- Exclusivity constraints – This criterion states that in cases where little or no monitoring data are available, the use of surrogate data "would need to be stringently justified." It is not clear that the use of the NCRP data was stringently justified. NIOSH did not consider other data sources, such as the extensive reporting by the FIPR (e.g., FIPR 1998).

- Site or process similarities – The selected dose rate was based on measurements taken for workers in the vicinity of phosphogypsum stacks. Since there were no stacks created during the pre-operational phase, the goal of selecting surrogate data based on process similarities was not achieved.

- Temporal considerations – Surrogate data should be taken from the same general period as that for which doses are to be reconstructed. It is not clear when the surrogate data were collected, but since the relevancy of the selected data is questionable, temporal considerations are moot.

4.2.2 External Exposures during Operational Period

- Hierarchy of data – This criterion specifies that surrogate data at the same hierarchy level be used "only after the appropriate adjustments have been made to reflect the uncertainty of this substitution." Application of this criterion is difficult, since NIOSH used modeling data based on straightforward dose calculations.

- Exclusivity constraints – This criterion states that in cases where little or no monitoring data are available, the use of surrogate data "would need to be stringently justified."

NIOSH did not justify that the modeling of exposure to drums of yellowcake as developed for Blockson was appropriate for TCC.

- Site or process similarities – The modeled dose rate was based on exposures of workers in close proximity to drums of yellowcake. While the modeled doses are qualitatively similar to what workers at TCC might have received, the assumed exposure duration is not consistent with the available information.
- Temporal considerations – Surrogate data should be taken from the same general period as that for which doses are to be reconstructed. This criterion is not applicable, since doses were modeled.

4.3 RADON EXPOSURES

- Hierarchy of data – This criterion specifies that surrogate data at the same hierarchy level be used "only after the appropriate adjustments have been made to reflect the uncertainty of this substitution." Recognizing the uncertainty underlying use of radon measurements from the Florida phosphate industry for workers at TCC, NIOSH proposed the use of 95^{th} percentile values to insure that the uncertainty was captured. This approach is consistent with the criterion.
- Exclusivity constraints – This criterion states that in cases where little or no monitoring data are available, the use of surrogate data "would need to be stringently justified." It is not clear that the use of the FIPR data was stringently justified. NIOSH did not present a review of the various sources of available data.
- Site or process similarities – The selected dose rate was based on measurements taken for workers involved in "chemical plant" operations. Selection of chemical plant operations was based on the argument that phosphate chemical plant operations would be similar to those for uranium extraction from phosphoric acid. While this approach is reasonable for some operations, specific consideration should also be given to workers involved in processes prior to digestion of the phosphate rock in sulfuric acid.
- Temporal considerations – Surrogate data should be taken from the same general period as that for which doses are to be reconstructed. The selected radon measurements were taken about four decades after TCC operations. However, Birky (2005) of the FIPR has observed that there have been no changes in the construction of wet process acid plants since the time of AWE operations.

4.4 UTILIZATION OF THE DRAFT CRITERIA

This is the first time that an attempt has been made to use the draft criteria for surrogate data in the review of a NIOSH report. In considering the criteria, we would suggest that the work group consider adding a "plausibility" and/or "fairness" criterion. When attempting to bound doses in situations where only limited data are available, the possibility exists that very conservative approaches will be taken, approaches that could be considered implausible. Though this approach would result in doses that are very claimant favorable, workers in another cohort where more data are available could be characterized as receiving lower doses and, therefore, would not be eligible for compensation. A fairness criterion would invoke specific consideration of such a possibility when choosing surrogate data.

Another related criterion that warrants consideration by the working group is whether the estimates of maximum doses are representative of those that could have been incurred under plausible circumstances. Several examples have been provided in Section 3, where surrogate data have been selected to make bounding, claimant-favorable assumptions, but convincing arguments were not made that the surrogate data represented exposures under plausible circumstances.

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APPENDIX A: REVIEW OF RADON EXPOSURE ESTIMATES DERIVED FROM SUMMARY DATA IN ORAUT-OTIB-0043 APPENDIX B

A.1. Verification of OTIB-0043 Radon Exposure Estimates Based on the Equal Weight Assumption

The summary data on radon exposures presented in Appendix B of ORAUT-OTIB-0043 (ORAUT 2006) were used by NIOSH to estimate the geometric mean (GM), geometric standard deviation (GSD), and 95th percentile of worker radon exposures. The OTIB calculations were performed with an assumption of equal weighting for all reported data points in the Appendix B tables that are not grayed out. The applicable data points include many values that are the mean of a set of measurements, as well as many individual measurements. Some of the means represent large datasets, while other mean values represent smaller datasets.

The assumption of equal weighting is not applicable when the means represent very different sample sizes. However, the equal-weight assumption was used to develop the estimated radon exposures reported in Table 4-4 of OTIB-0043. In this section of the current report, the equal-weight assumption is applied to confirm the NIOSH estimates. In the following section, the mean values in Table B-3 of OTIB-0043 are weighted proportionally to the sample sizes represented by each mean value to develop more appropriate weighted estimates of worker radon exposure.

Although OTIB-0043 states that 130 data values are contained in the non-shaded rows of the Appendix B tables, only 128 values are listed there, including 7 values in Table B-1 that are below the limit of detection (LOD) reported as "<0.5 pCi/L." Two methods are used here to estimate a lognormal distribution using the equal-weight assumption; the maximum likelihood method (MLM) and a graphical method based on normal scores. The maximum likelihood method requires that surrogate values be imputed for the 7 LOD values. The midpoint of the uncertainty range (LOD/2 = 0.25 pCi/L) was used for these 7 values. In the graphical method, these 7 points are not included on the graph, but their ranks are used when computing the normal scores.

The results of these two analyses are compared with the OTIB-0043 estimates in Table A-1. As shown in the table, the lognormal distributions estimated using the MLM and graphical methods have a GM and GSD that are very similar to the GM and GSD reported in OTIB-0043 for the Appendix B data. The estimated mean (arithmetic), standard deviation, and 95th percentiles also closely match.

The normal score plot used in the graphical method is shown in Figure A-1. Note that the fitted regression line underestimates the high readings in the upper tail of the distribution. As shown in Table A-1, the 95th percentile of the empirical distribution (i.e., the dotted "line" in Figure A-1) is significantly higher than the 95th percentile estimated from the fitted lognormal distribution—2.9 pCi/L versus the 2.558 pCi/L—which, in turn, is higher than the estimate of 2.33 pCi/L reported in OTIB-0043.

Table A-1. Characteristics of Lognormal Distributions for Radon Measurements in OTIB-0043 Appendix B

Source of Estimates	Lognormal Parameters mu	Lognormal Parameters sigma	N	GM (pCi/L)	GSD	95th Percentile (pCi/L)	Mean (pCi/L)	Standard Deviation (pCi/L)
Maximum Likelihood	-0.323	0.738	128	0.724	2.092	2.438	0.951	0.757
Graphical Method	-0.351	0.784	128	0.704	2.191	2.558	0.958	0.790
OTIB-0043 Estimates	-0.286	0.688	130	0.751	1.989	2.330	0.951	0.731
				Median				
Empirical Distribution	--	--	128	0.700	--	2.900	1.006	1.169

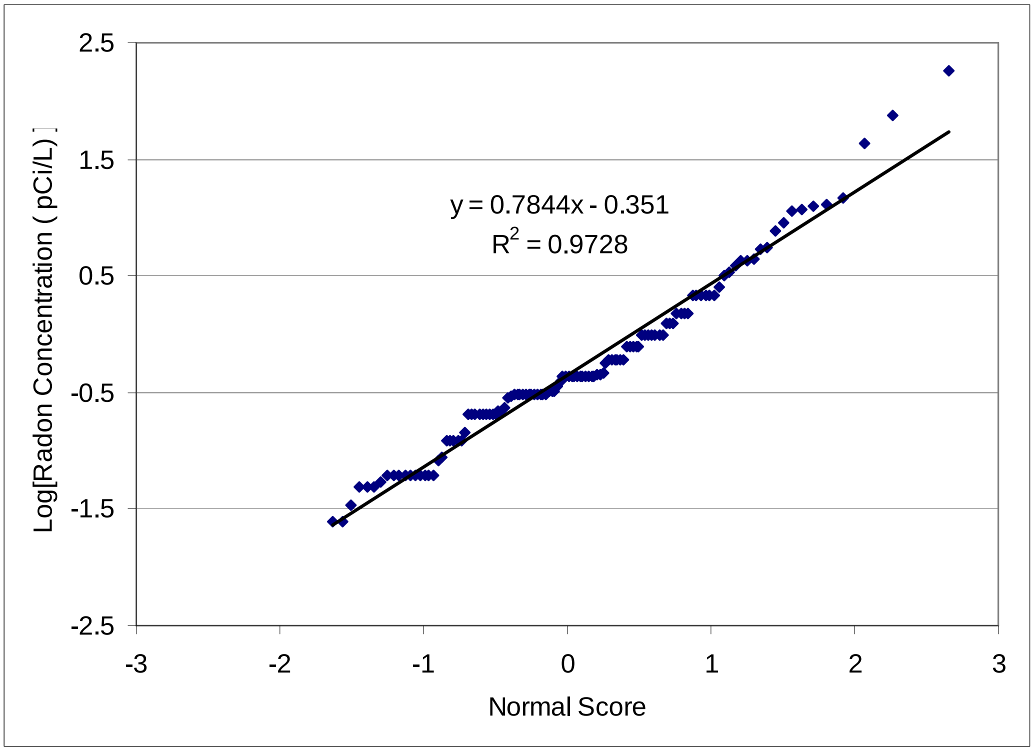

Figure A-1. Normal Score Plot of Appendix B Radon Measurements

A.2. Radon Exposure Estimates Based on the Expanded Table B-3 Datasets

Many of the mean values reported in the Appendix B tables do not include detailed information on the underlying dataset. However, the mean values reported in Table B-3 of OTIB-0043 are accompanied by other statistics that describe the underlying dataset, such as the number of samples and the variance of the sample values. The sample sizes for these eight datasets range from 24 to 118, indicating that a wide range of weights should be considered. The smallest dataset in Table B-3 (labeled gypsum stack flux test) has a mean value that is the second highest value contained in the entire dataset (6.52 pCi/L). Other larger datasets summarized in Table B-3 also have relatively high mean values when compared to the mean value of approximately 1 pCi/L for the equally weighted data shown in Table A-1. Although other tables in Appendix B include mean values, only Table B-3 contains additional information on the variance of the underlying datasets from which the means were derived. The sample variance provides a measure of the spread of the values in each dataset about the mean. When the means are treated as single values, this information is lost. As a result, any estimate of the 95th percentile that does not include this information will be biased to the low side.

The mean values in Table B-3 were assigned weights in proportion to the size of sample each mean represents. All other valid data values in the Appendix B tables were assigned a weight of 1, as in the equally weighted approach. A weighted mean was calculated using these weights. The variances reported for each dataset in Table B-3 and the sample sizes were used to estimate the sum of squared deviations of the data values when determining the weighted standard deviation. The estimated weighted mean and standard deviations were then used to calculate the parameters of a weighted lognormal distribution that represents the population of exposures in Appendix B. The required calculations are shown in Table A-2. The weighted mean is estimated as 1482.49/713 = 2.079 pCi/L, approximately twice as high as the unweighted mean value. The weighted standard deviation is estimated as sqrt(17834/713) = 5.01 pCi/L.

Table A-2. Derivation of Weighted Lognormal Distribution Using Expanded Data for Table B-3 Means

Values for analysis:	Count	Mean	Terms to sum for weighted mean	Sum of Squares about Weighted Mean	Sum of Squares about Mean	Sample variance
NE gypsum stack well	90	2.43	218.7	1671.57	1660.5	18.45
Auto shop SE fence	56	2.89	161.84	1365.69	1328.88	23.73
SW of plant	31	0.35	10.85	99.52	6.82	0.22
Burn area fence	118	1.89	223.02	3236.25	3232.02	27.39
Liming station ladder	105	1.9	199.5	2666.17	2662.8	25.36
Environmental monitoring well	101	2.6	262.6	5608.65	5581.26	55.26
Gypsum stack flux test	24	6.52	156.48	1061.77	588.48	24.52
Cooling pond hand rail	68	2.08	141.44	1825.80	1825.8	26.85
All other Appendix B Tables	120	0.9005	108.06	298.67	131.946	
Total	713		1482.49	17834.10		

The weighted lognormal estimates are compared with the unweighted estimates from Section A-1 in Table A-3. Although both methods yield similar estimates for the GM, the weighted estimates have a GSD that is twice as large as the unweighted estimate. This results in a much higher estimate of the 95th percentile when the weighted approach is used (7.78 versus 2.33 pCi/L). A graphical comparison of the GM, GSD, and 95th percentiles is shown in Figure A-2.

Table A-3. Comparison of Unweighted and Weighed Distributions for Appendix B Data

Lognormal models	mu	sigma	GM (pCi/L)	GSD	95th Percentile (pCi/L)
OTIB-0043	-0.286	0.688	0.75	1.99	2.33
Weighted Distribution	-0.226	1.384	0.80	3.99	7.78

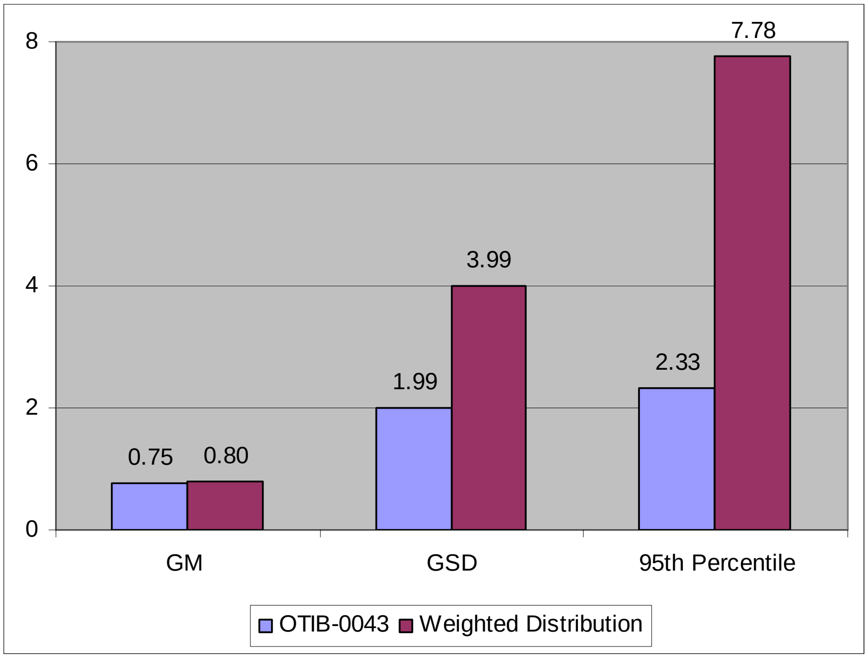

Figure A-2. Comparison of OTIB-0043 Estimates with Lognormal Parameter Estimates Using Expanded Data in Table B-3

A.3. Weighted Mean Estimates for Table B-2 in OTIB-0043

Table B-2 in OTIB-0043 contains mean values for worker radon exposures during wet rock loading at 15 companies. The table also includes sample size and the minimum and maximum data values. In this case, information on the variance is not provided. The mean, minimum, and maximum values were used to compute the lognormal parameters for the distribution of exposures in the worker populations at each company.

A weighted mean was estimated using the sample sizes at each company as weights. The minimum and maximum values were used to estimate the variance of the sub-population of workers in each company. The normal score of the minimum is z^1 = Normal^{-1}(p^1) where p^1=(1-0.5)/N, and the normal score of the maximum is z^N = Normal^{-1}(p^N) where p^N=(N-0.5)/N where Normal^{-1} represents the inverse cumulative distribution function (CDF) of the normal distribution. The minimum is $X^1 = \exp(\mu + z^1\sigma)$ and the maximum is $X^N = \exp(\mu + z^N\sigma)$, hence $\ln(X^N/X^1)=\sigma(z^N - z^1)$ which may be solved for σ in terms of the minimum, maximum, and the sample size N. This provides an estimate of the variance of each sub-population that contains more than one sample.

The sub-population variances were combined to estimate the sum of squared deviations from the overall weighted mean. The sum of squares provides an estimate of the standard deviation of each population of workers about the weighted mean, as in the previous section. The standard deviation and the weighted mean were used to estimate the lognormal distribution parameters mu and sigma for the combined population in Table B-2. The lognormal parameters provide estimates of the GM, GSD, and 95th percentile of the combined worker population shown in Table A-4.

Table A-4. Weighted Distribution Estimates of Worker Radon Exposure for OTIB-0043 Table B-2 Data

Weighted mean	0.0997	WLM/yr
Standard deviation	2.939	WLM/yr
mu	-5.690	
sigma	2.602	
95th Percentile	0.244	WLM/yr

APPENDIX B: DRAFT CRITERIA FOR THE USE OF SURROGATE DATA

For the purposes of this report, the term "surrogate data" will refer to the use of exposure data from one site for individual dose reconstruction for workers at another site. This has become a common practice in the NIOSH dose reconstruction program because of the lack of complete and comprehensive exposure monitoring records for many of the workers at the sites covered by the program. It is especially common for sites during the early years of these DOE facilities, because of the lack of reliable monitoring methods, the urgency of developing production capabilities, and other reasons.

This report will suggest a number of criteria that need to be considered in determining whether the specific use of surrogate data for individual dose reconstruction is scientifically sound for that particular application.

(1) Hierarchy of Data – It should be assumed that the usual hierarchy of data would apply to dose reconstructions for that site. Individual worker monitoring data are preferable to workplace-monitoring data, etc. The use of surrogate data should also follow this hierarchy. In general, surrogate data should not be used to replace available data from the site in question that is at a higher level in the hierarchy. It should only be used to replace data at the same level in the hierarchy if the surrogate data has some distinct advantages over the available data, and then only after the appropriate adjustments have been made to reflect the uncertainty of this substitution.

(2) Exclusivity Constraints – In many cases, surrogate data are used to supplement the available monitoring data from a site. In those cases, the surrogate data is usually used to justify certain assumptions about the distribution or range of possible exposures or assumptions about the source terms. In those cases, no special justification is necessary beyond the usual scientific evaluation. However, in some cases, there are no or very little monitoring data available. In those cases, the use of the surrogate data as the basis for individual dose reconstruction would need to be very stringently justified. This judgment needs to take into account not only the amount of surrogate data being relied on relative to data from the site, but also the quality of the surrogate data relative to data available for the site in question.

(3) Site or Process Similarities – One of the key criteria for judging the appropriateness of the use of surrogate data would be the similarities between the site (or sites) where the data were generated and the site where the surrogate data are being utilized. The application of any surrogate data to an individual dose reconstruction at a site should include a careful review of the rationale for utilizing that source of data (why that site(s) - similarity of the production processes, monitoring methods, factors affecting exposures, etc.). Are there other sources that were not used and why? Do these other potential sources contradict or undermine the application of the data from the selected site? Are there adequate data characterizing the site being used that would help support its application to other sites? Does the surrogate data reflect the type of operations and radiation protection practices in use at the facility in question? Surrogate data should not

be used if the equivalence or claimant favorability of working conditions, source terms, and processes of the surrogate facility to the one for which dose reconstructions are being done cannot be established with appropriate scientific or technical certainty.

(4) Temporal Considerations – Consideration also needs to be given to the period in question, since working conditions and processes varied in different periods. Surrogate data should belong in the same general period as the period for which doses are sought to be reconstructed.

APPENDIX C: EXTERNAL EXPOSURE TO 30-GALLON DRUM OF YELLOWCAKE

To explore the range of possible external exposures, SC&A considered an alternative scenario to model external exposures. We calculated the dose rate in air (air kerma) due to external exposure to a drum of yellowcake, as well as the dose to the skin from non-penetrating (beta) radiation. Our analysis utilized the Los Alamos Monte Carlo code MCNP5 (LANL 2003). Given the wide variability in the chemical composition of yellowcake, we modeled the radiation source as U_3O_8, and assumed it was stored in a 30-gal drum. According to a specification of the Oak Ridge National Laboratory for carbon steel open-head drums (ORNL 2002, Specification No. 100-1A2-0006), 30-gal drums have a nominal thickness of 1.2141 mm. According to The Cary Company (2008), 30-gal drums have an inside height of 27.5 inches and an 18.25-in inside diameter. The bulk density of U_3O_8 has a range of 1.5–4 g/cm^3 (ANL n/d). Since the freshly produced material would most likely be loosely packed, we assumed the U_3O_8 to have a bulk density of 2 g/cm^3.

The drum is constructed of ASTM A 366 steel, as cited by ORNL 2002. The steel is assigned a nominal density of 7.86 g/cm^3. The average elemental composition of the alloy is listed below.

Elemental Composition of Steel Drum

Element	Mass fraction
C	8.5e-4
P	1.5e-4
S	1.75e-4
Mn	0.003
Fe	0.996
Total	1.000

The drum is located in the center of a stylized cylindrical room, 3 m high, with a radius of 5 m, which is filled with moist air. The walls, floor, and ceiling are concrete which is optically thick. The dose rates are calculated at 30 cm from the exterior of the drum at a height of 1 m above the floor.

The yellowcake is assumed to have the isotopic ratio of natural uranium, listed below. The specific activities of the three uranium isotopes are calculated from the isotopic composition and the uranium fraction of U_3O_8. We assume that U-238 is in equilibrium with its short-lived progeny: Th-234, Pa-234m, and Pa-234. The 6.7-hour Pa-234 is the product of the isomeric transition of Pa-234m, which occurs in 0.16% of the disintegrations. We assume that U-235 is in secular equilibrium with its entire decay chain. We also assume that Th-232, in secular equilibrium with its entire decay chain, has an activity concentration that is 3.3% of the activity of U-238, as listed by Tomes and Glover (2007, Table 6). We assume that Ra-226, in full secular equilibrium with its short-lived progeny, has an activity concentration of 10% of the activity of U-238. This assumption is based on the following statement of Hull and Burnett (1996): “During acidulation of Florida phosphate rocks, the great majority of Ra-226 . . . typically 90–100%, is fractionated to the [phosphogypsum].” This implies that as much as 10%

could report to the phosphoric acid. Since Hull and Burnett report approximately equal activities of Ra-226 and U-238 in the phosphate rock, we chose 10% as a reasonable upper-end value.

Specific Activity of Yellowcake

Nuclide	Specific Activity (Bq/g)
U-234	10557
U-235	482
Th-231	482
Pa-231	482
Ac-227+D[a]	482
U-238	10470
Th-234	10470
Pa-234m	10470
Pa-234	17
Ra-228+D	349
Th-228+D	349
Ra-226+D	1047

[a] In secular equilibrium with progeny

MCNP Analysis

We initially performed two sets of calculations, comparing the dose rates from gamma rays and characteristic x-rays emitted by the yellowcake in an open 30-gal drum, as well as from the same amount of material in the bottom of a 55-gal drum. The 55-gal drum, like the 30-gal drum, was modeled on the specification of ORNL (2002) and the description by The Cary Company (2008). The dose point in each case was located 1 ft (30.48 cm) from the edge of the drum at a height of 1 m above the floor. The 30-gal drum produced the higher dose rate, and was therefore adopted for this analysis.

The air kerma was calculated by applying the conversion coefficients listed by ICRP (1996, Table A.1) to the flux of gamma rays and x-rays calculated by MCNP5. The air kerma from beta bremsstrahlung x-rays was calculated in a similar manner, and the results from the two calculations were added together to produce the air kerma from direct penetrating radiation. We also calculated the skin dose from beta rays at the same location, applying the conversion coefficients listed by ICRP (1996, Table A.43) to the electron flux computed by MCNP5.

The results show an air kerma rate of 0.65 mrad/h from direct penetrating radiation and a dose rate to the skin of 0.85 mrad/h.

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