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National Institute of Occupational Safety and Health

Third Supplemental Review of the Mallinckrodt Site Profile Revision 1 (Effective Date March 10, 2005)

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1.0 EXECUTIVE SUMMARY

This report is SC&A's third supplemental review of the Mallinckrodt Technical Basis Document (MCW TBD) (ORAUT-TKBS-0005, Rev. 01, Effective Date: March 10, 2005). It was prepared pursuant to actions taken by the Advisory Board on Radiation and Worker Health (Board) at its July 2005 meeting in St. Louis, Missouri, when SC&A presented its second supplemental review of the MCW TBD. At that meeting, the Board asked the National Institute of Occupational Safety and Health (NIOSH) to develop its approach to dose reconstruction for six specific priority issues (see Attachment 1). The Board also asked SC&A to comment on the work that NIOSH was doing as the issues were being addressed internally by NIOSH, and to prepare an overall review of the six issues to be submitted to the Board by August 16. As with the second supplemental review, the principal aim of this third review is to support the Board in its decision regarding the Special Exposure Cohort (SEC) petition for the outstanding period of 1949 through 1957.

Since this report is directed at the six priority issues, SC&A is not repeating the discussion regarding 42 CFR 82 and 42 CFR 83, and other details of various issues that were covered in the second supplemental review (SC&A 2005b). In order to get a complete picture of SC&A's review of the MCW TBD, this third supplemental review should be read in conjunction with SC&A's earlier reviews (SC&A 2005, SC&A 2005a, and SC&A 2005b). It should also be borne in mind that, even though SC&A has prepared this review to assist the Board in its deliberations regarding the Mallinckrodt 1949–1957 SEC Petition, the SC&A review and task is provided with respect to the changes in the dose reconstruction approach in the MCW TBD proposed by NIOSH.

It should also be noted that this review is focused mainly on methodological questions. In some instances, SC&A has made approximate checks of key NIOSH parameters, notably in the case of radon breath to Ra-226 body burden and in the case of the ratios of various trace radionuclides to Ra-226. However, SC&A has not verified the numerical details of the various distributions and their median and 95th percentile values.

1.1 OVERALL FINDINGS REGARDING MAXIMUM DOSE ESTIMATES

SC&A's findings, as summarized in Table 1, address the methods proposed by NIOSH to perform dose reconstruction. See Section 4 for comments regarding regulatory issues.

Table 1. Summary Table Comparing SC&A Understanding of NIOSH Position and SC&A Position on Priority Issues

Priority Issue	SC&A Understanding of NIOSH Position	SC&A Position	Status	Comments
1-1. Non-equilibrium radionuclide ratios (1 a. through 1 d), **workers with radon breath data**	A. NIOSH will use (1) radon breath data to determine radium body burden and (2) Fernald K-65 Silo 1 ratios to derive values for other radionuclides B. NIOSH will use 95th percentile of air concentration in AM-7 areas for Th-230, Pa-231, Ac-227 C. NIOSH will consistently use the higher dose estimate resulting from the values in A or B	Agreement in principle	S	A. For workers with radon breath data, claimant-favorable adjustments are needed B. Air concentration data in AM-7 residue areas need to be analyzed according to recommendations in prior SC&A reviews for 95th percentile values. Ratios need further checking
1-2. Non-equilibrium radionuclide ratios (1 a. through 1 d), workers - **no radon breath data**	As in 1-1, except radium body burden will be derived from 95th percentile of the radon breath value	Agreement in principle	S	Generally as in 1-1. Plant 7E bioassay data need to be compiled to ensure claimant-favorable estimates for 1955–1957
1-3. Non-equilibrium radionuclide ratios unmonitored workers	As in 1-2, except mean values will be used	Dose would represent a reasonable approach inside buildings	A	
2-1. Radon job specific data	NIOSH proposes allocating mean radon breath radium burden and related intakes	NIOSH verification shows this is claimant favorable	A	NIOSH is not using job specific approaches
2-2. Radon dose estimates for non-respiratory tract organs	NIOSH proposes to use radon breath data	NIOSH verification shows this is claimant favorable	A	
3-1. External dose correction factor – organ vs. badge geometry	Use a 2.1 correction factor for lower torso organs, and 1.0 for other organs		A	See 3-2 and 3-3 – factors must be combined multiplicatively
3-2. Angle of incidence to badge, deep dose	Board Working Group deferred to Task 3 review, Aug. Board meeting	N/A	S	Correction factor from 1.0 to 7.0 depending on photon energy and angle. For ~200 KeV and 67.5^{o}, it is ~2.1
3-3. AP, PA, Rotational, isotropic geometry	Board Working Group deferred to Task 3 review, Aug. Board meeting	N/A	S	For rotational or isotropic, a range of 1.4 to 2 is appropriate for correction factors

Table 1. Summary Table Comparing SC&A Understanding of NIOSH Position and SC&A Position on Priority Issues

Priority Issue	SC&A Understanding of NIOSH Position	SC&A Position	Status	Comments
3-4. Overall external deep dose correction factor	Not developed – see 3-2 and 3-3 above	Varies from 1.4 to 8 depending on organ, incidence angle, and geometry	S	Higher end of range (~6 to 8) for lower torso organs for ~200 KeV photons. Range or value for other organs in areas needs to be specified
4-1. Incidents	Incidents are usually covered in a claimant-favorable way by continuous intake modeling	SC&A agrees, based on limited examples	A	There may be unusual situations in which this may not apply but this is an individual dose reconstruction issue rather than a TBD issue
5-1. Unmonitored workers, office-related non-production	Median values of intakes and external dose, including Ra-226, Th-230, and other U decay products	This appears to be satisfactory provided job status can be verified	A	
5-2. Plant 1 and Plant 2 decommissioning workers	One case only with uranium bioassay, covered as other monitored workers, with 95th percentile radon breath data. NIOSH will also look at decommissioning air data for verification	This appears to be satisfactory	A	SC&A cautions in regard to air use of air concentration data apply
5-3. Maintenance mechanics, SLAPS workers	95th percentile values from bioassay and residue ratio analysis	Agreement in principle	S	Some work remains on radionuclide ratio refinement
5-4. Environmental dose due to stack releases	NIOSH proposes to assign inside building mean dose to unmonitored workers to cover this aspect of dose	NIOSH has validated its approach for routine exposures by checking air monitoring data. Approach to incidents not yet developed	S	Site specific data may be sufficient for maximizing estimates for incidents

A = agreement, which means SC&A agrees that the NIOSH approach will yield a maximum dose estimate with scientifically plausible assumptions. It does not mean that SC&A has endorsed all the numerical results presented by NIOSH, since this review is primarily methodological. See Section 1.1.
U = Unresolved issues
S = Some further methodological work is needed (see comment in the last column and the issue discussion in Part 3 of this review).
N/A = Not applicable

Notes

1. Doses for uranium derived from uranium bioassay will be applied to all workers. Correction factor for Barnes data needed (SC&A 2005b)
2. NIOSH is not using the details of job types and times spent on each job in the dose reconstruction process.

1.2 SUGGESTIONS FOR COMPLETION

SC&A's review of NIOSH's new approach to Mallinckrodt dose reconstruction indicates that the following need to be integrated into the approach before it can be used for scientifically sound, claimant-favorable dose reconstruction:

(1) **Th-230 area air concentration values:** The Th-230 air concentrations should be 95th percentile values of the measurements in the Th-230 (AM-7) areas, not the 95th percentile value of the time-weighted averages for all of Plant 6. NIOSH also needs to adjust air intake to working hours corresponding more closely to what workers have described as routinely occurring overtime.

(2) **Ionium extraction bioassay data:** NIOSH should construct a co-worker database for the Th-230 extraction in Plant 7E during the 1955–1957 period in order to develop a procedure to ensure that doses are not underestimated for workers with one or a few Th-230 bioassay samples.

(3) **Pa-231 and Ac-227 to Th-230 ratios:** NIOSH should complete this research and use the most claimant-favorable value for these ratios. The use of defensible claimant-favorable ratios is central to a claimant-favorable result for dose, which is very sensitive to the values used. NIOSH should publish the data and the analysis that go with its choice of ratios.

(4) **Types of workers exposed to non-equilibrium ratios of trace radionuclides:** Given the crucial difference that attribution of exposure to non-equilibrium values of trace radionuclides (Th-230, Pa-231, Ac-227 especially) makes to dose, NIOSH should exercise great caution to ensure that when it makes an assumption that certain workers were not exposed to non-equilibrium ratios (Section 3.1) that it is defensible and claimant favorable.

(5) **Use of radon breath data for individuals with measurements:** Missing radon breath data appear in significant measure to be due to instrument failure. This raises the issue of how the gaps should be filled in for workers with radon breath data, especially for workers with sparse radon breath data. NIOSH should use a more claimant-favorable approach, such as the 95th percentile values, to fill in missing data. Further, a check into instrument uncertainties is warranted. Finally, the documentation needs to be analyzed to investigate whether all high radon breath readings are included and repeat samples are excluded.

(6) **External dose correction factors:** NIOSH should adopt a set of correction factors for external dose that incorporate all three aspects needing correction. SC&A estimates that the range of correction factors is 1.4 to 8, with the higher end (~6 to 8) being applicable to deep external dose for lower torso area organs, such as prostate and colon. A claimant-favorable value or set of values for other organs (upper torso, head, and neck areas) remains to be established.

(7) **Environmental dose due to incidents:** NIOSH should complete the environmental dose investigation to check for the dosimetric significance of accidental releases. Available site-specific data may be sufficient for a maximizing approach to be implemented.

1.3 OVERALL CONCLUSIONS

If NIOSH amends the dose reconstruction procedure to resolve the six issues according to the recommendations in Section 1.1 above, it appears possible to estimate maximum dose with scientifically plausible worst-case assumptions under the definitions of these terms in 42 CFR 83. According to NIOSH (which proposes somewhat different terminology), such estimates would be the basis for both compensation and denial decisions by the Department of Labor under 42 CFR 82 (see Attachment 6, last paragraph, and Section 4 of this review). SC&A stresses that (1) this review is not a review of NIOSH's SEC petition evaluation report, and (2) the actual validity of the proposed NIOSH approach on the six issues, when applied to dose reconstruction for individuals, will depend centrally on the choice of radionuclide ratios and overall external dose correction factors. This is because the result is critically dependent on these parameters.

The approach to dose reconstruction being proposed by NIOSH as a result of the MCW TBD review process is significantly different than that in the TBD, as well as the efficiency approach for maximum dose reconstruction that has been used in at least some Mallinckrodt cases.[1] For instance, NIOSH has used Technical Information Bulletin (TIB) TIB-0002, Rev. 01, to determine worst-case internal dose in assessing some claims where the probability of causation was deemed likely to be less than 50%. In its second supplemental review, SC&A found that the following:

> *NIOSH has not established that this applies to worst-case data from Mallinckrodt. Further, the assumption of intakes of large amounts of radionuclides, such as Sr-90, Pu-239, and Cs-137, which were not present at Mallinckrodt in appreciable amounts, if at all, does not meet the test of a scientifically reasonable assumption for worst-case dose estimation required by 42 CFR 82.*
> (SC&A 2005b, Finding 16, p. 39)

SC&A has not evaluated the implications of the changes for the probability of causation of the new NIOSH approach, since that is beyond the scope of the task set forth by the Board. However, in view of the major differences between the new approach and existing approaches to dose reconstruction, SC&A suggests that NIOSH should publish the essentials of the new approach, including the values of ratios of radionuclides to be used in internal dose calculations, as well as the external dose correction factors, as soon as is practicable.

[1] Minimum dose reconstruction for compensation has not been an issue in the review process since SC&A concluded in the first review (SC&A 2005) that the TBD is sufficient for minimum dose reconstruction for purposes of compensation. NIOSH has indicated that it considers the proposed approach to be "a best estimate with some maximizing aspects where NIOSH cannot do better (as distinct for the efficiency approaches to maximum dose using the Hanford radionuclide list, for instance)" (Attachment 6, last paragraph).

2.0 SCOPE AND INTRODUCTION

This is SC&A's third supplemental review of NIOSH's Revision 1 of the Mallinckrodt Site Profile (ORAUT-TKBS-0005, Rev. 01, Effective Date: March 10, 2005). During its July 2005 meeting in St. Louis, Missouri, the Board directed SC&A to review NIOSH's approach to dose reconstruction on the following six priority issues:

(1) Handling of raffinate exposures
(2) Handling of radon exposures
(3) Application of correction factors for external doses to organs
(4) Assessment of intermittent incident exposures
(5) Specification of dose reconstruction methodology for unmonitored workers
(6) Example dose reconstructions for representative cases

The Board also directed SC&A and NIOSH to communicate with each other in order to facilitate and expedite the process of SC&A's review of NIOSH proposals through an open interchange and critique. The Board also asked SC&A and NIOSH to maintain a summary of the specific contacts made and the topics discussed.

The following is a list of contacts between NIOSH and SC&A, as well as the topics discussed and associated documentation:

(1) NIOSH sent three draft documents to SC&A in the July 30–August 2, 2005 period. The first showed NIOSH's radon dose calculations for non-respiratory tract organs (Attachment 2a).[2] The second contained a description of NIOSH's proposed approach to estimating the Ra-226, Th-230, Pa-231, Ac-227, Pb-210, and Po-210 intakes of MCW workers (Attachment 2b). The third document was a draft TIB describing external dose correction factors due to organ vs. badge location geometry (Attachment 2c).

(2) SC&A, Board Working Group members, NIOSH and NIOSH contractor personnel, and the Petitioner (Denise Brock) attended a meeting in Cincinnati, Ohio, on August 4, 2005, to discuss the documents provided as per Item 1 above, as well as documents prepared by NIOSH for the meeting (see Attachments 3a though 3f), such as estimation procedures for incidents and example dose reconstructions for representative cases. The meeting was announced in the *Federal Register* and was open to the public. A transcript of this meeting is included as Attachment 4.[3] The Board Working Group concluded that several issues still remained to be addressed:

[2] NIOSH also forwarded a copy of a few pages of ICRP 71 having to do with the inadvisability of using the ICRP 66 respiratory tract model for estimating radon progeny dose to non-respiratory tract organs. These pages are not reproduced in this report. See Section 3.2.

[3] The transcript of the August 4 meeting was not available at the time of preparation of this review. The references to the discussion at the August 4 meeting in this report are from personal notes and memory, except in cases where reference is made to the documents prepared by NIOSH, which are reproduced in Attachments 3a though 3f.

- Approach for resolution of missing radon breath data—since 20% to 30% of the data may be missing or not analyzed
- Analysis to indicate whether radon breath data will bound radon doses
- Approach to determining Th-230, Pa-231, and Ac-227 intakes in areas that were not expected to have significant amounts of Ra-226, because it had been removed in earlier steps
- Analytical justification for not using environmental data

(3) Subsequent to the meeting, Arjun Makhijani of SC&A and Jim Neton of NIOSH exchanged a number of e-mails to help clarify issues that remained. Besides NIOSH, NIOSH contractor personnel, and SC&A personnel, these e-mails were copied to the Board Working Group, Board Chairman Paul Ziemer, and Petitioner Denise Brock. The e-mail exchanges are reproduced in Attachment 5.

(4) A conference call was held on August 9, 2005, between SC&A and NIOSH personnel, which was announced to the Working Group of the Board, Chairman Ziemer, and Denise Brock. Summary notes from that call are presented in Attachment 6.

(5) NIOSH sent materials to SC&A following the August 9 conference call, which are reproduced in Attachments 7a though 7f.

This SC&A report is restricted to review of the six priority issues raised by the Board, as noted above, and any related sub-issues. Comments on Issue 6 (dose reconstruction examples) are based mainly on the materials and discussion provided by NIOSH for the August 4, 2005, meeting (Attachments 3a though 3f). Comments regarding other issues covered in earlier reviews are made here if they are relevant in some way to the six issues being reviewed. SC&A has reviewed underlying documentation in some areas; however, this was not uniformly the case, since this review is focused on methodological questions. The same is true of numerical estimates. SC&A has assessed the values provided by NIOSH to a degree sufficient to evaluate the methodology. As noted in the suggestions for completion of the development of the method (Section 1.1), there is still some crucial work remaining to be done before the new approach can be applied to individual dose reconstruction.

This review is organized according to the six priority issues in the order set forth by the Board (Attachment 1). This discussion is followed by observations relating to 42 CFR 82 and 42 CFR 83.

SC&A has developed a check list, which is presented in the Executive Summary, that summarizes SC&A's understanding of NIOSH's proposed approach to each priority issue, SC&A's conclusions regarding each issue, and the status of each issue according to the SC&A review.

SC&A wishes to stress that, while this review was prepared to assist the Board in its deliberations on the pending SEC petition, SC&A was not charged with evaluating the SEC Petition or NIOSH's evaluation of the petition. SC&A has focused on the six priority issues

defined by the Board as they relate to the development of a methodology for reconstructing doses other than minimum doses in the context of a review of the MCW Site Profile (Rev. 01). For definitions of SC&A's view of the various types of dose reconstruction under 42 CFR 82 and 42 CFR 83, the reader is referred to SC&A 2005b.

3.0 PRIORITY ISSUES

3.1 PRIORITY ISSUE 1: HANDLING OF RAFFINATE EXPOSURES

The essence of this issue is that there were many steps, locations, job types, and processes at Mallinckrodt that involved handling of materials in which the decay products of U-234 and U-235 were **not in equilibrium**. Specifically, there are waste and processing streams that are enriched to varying degrees in the decay products of U-234 (which is itself in the decay chain of U-238). The decay products of U-234 at issue are Th-230, Ra-226, Pb-210, and Po-210). The decay products of U-235 at issue are Pa-231 and Ac-227. Further, since the K-65 residues were themselves processed for some period of time for uranium extraction, the disequilibrium periods and locations were likely to have been more varied than those that would be inferred from an ore process flowsheet alone (see SC&A 2005b; Attachment 2b of this report, produced by NIOSH; and the MCW TBD). Such processing likely extended into the 1950s, and possibly until 1955 (SC&A 2005b).

The specific mix of radionuclides assumed for dose reconstruction is important, because bioassay data only measured uranium and no other element (with the exception of some thorium bioassay data for Plant 7E in the 1955–1957 period). On the other hand, air concentration data measured gross alpha, which does not provide any indication of the specific mix of radionuclides present in air and therefore breathed by workers. Since the dose per unit intake (dose conversion factor) is very different for the various radionuclides, the same air concentration can give very different dose estimates, depending on the specific mix that was prevalent at a particular time and place. Moreover, there is no one mix of radionuclides that would be uniformly claimant favorable for all organs. **However, the dose estimate result is, in most cases, especially sensitive to the intake values used for Th-230, Ac-227, and Pa-231 (see Attachments 3a through 3d).** In some cases, Po-210 and Pb-210 are also important. Therefore, the issue of how the radionuclide ratios (i.e., the specific mix of different radionuclides breathed by workers) are determined is a crucial issue in dose reconstruction.

The Board listed four sub-issues as part of this task for NIOSH development of dose reconstruction approach as follows (Attachment 1):

a. NIOSH should specify the radionuclide ratios (Ra-226, Th-230, Ac-227, Pa-231) for all ore processing (including non-pitchblende ores)

b. NIOSH should specify the radionuclide ratios for K-65 uranium extraction (including processing steps and filtration steps)

c. NIOSH should specify how ratios will be applied by job title (including roving workers and cases where no job titles are available)

d. NIOSH should specify approach for estimating intake when any combination of urine, air sampling and breath radon data is available. In this context NIOSH should consider the reliability of breath radon for determining Ra-226 intakes.

In order to clarify the various areas where radionuclides other than natural uranium isotopes would be expected to be present, NIOSH developed a very useful flowsheet of the uranium production process that also shows the waste streams (see Attachment 2b).

In its TBD, NIOSH had identified a ratio of 100 to 1 for Ra-226 areas with uranium decay products in disequilibrium. This did not reflect the significant amounts of the other decay products of uranium present in the various streams of residues. Moreover, the K-65 residues were also processed for uranium extraction for a period that likely extended into the 1950s, possibly until 1955. A special process called the sodium carbonate/bicarbonate process was set up for this purpose. As noted in SC&A's second supplemental review (SC&A 2005b, Finding 7, pp. 25–31), this means that materials containing significant amounts of Ra-226, Th-230, and U-235 decay products were processed in parts of Plant 6 other than those that handled only residue wastes. SC&A suggested that data from the K-65 silos might be a source of useful data on the amounts and relative proportions of the various radionuclides in some processing steps (SC&A 2005b).

The data available to estimate doses in areas containing uranium, as well as a mix of decay products, are as follows:

- Uranium bioassay data for most workers and the vast majority of production and maintenance personnel
- Job type data
- Radon breath data, indicating body burden of Ra-226 for about one-fifth of the workers (i.e., those workers thought to be at risk of radium intakes)
- Gross alpha air concentration data, which, by definition, represent the aggregate concentration of all alpha-emitting radionuclides
- Some data on concentrations of the decay products of U-234 and U-235 in various residue streams
- Radionuclide ratios from Fernald Silo 1, where the K-65 wastes from Mallinckrodt pitchblende processing were sent

Even with all this data, it appears impossible to actually calculate an accurate "best estimate" of dose for a variety of reasons:

(1) The precise amounts of time that workers spent in the various working areas are not known, even though job types are known.

(2) Dose estimates are very sensitive to the amounts of trace radionuclides (Th-230, AP-231, and Ac-227), for which there are no individual measurement data (other than some Plant 7E data for Th-230 during 1955–1957) or radionuclide-specific air concentration data. Uranium and Ra-226 tend not to dominate the dose when the other radionuclides are present out of equilibrium. For example, in three of the four NIOSH estimates in Attachment 3a, the combined dose due to inhalation of Th-230, Pb-210, Po-210, Pa-231,

and Ac-231 accounts for about 90% or more of the total, while in the remaining case, it is still about 70% of the total. The other NIOSH examples (Attachments 3b, 3c, and 3d) also show a similar pattern of very low uranium contribution to total internal dose and the domination of the combined total of dose due to Th-230, Pb-210, Po-210, Pa-231, and Ac-227 in the total dose. The uranium contribution to the total dose, derived from bioassay data, is under 5% in all cases and under 1% in some cases.

(3) The ratios of the various trace radionuclides to uranium and radium varied over time and according to location in the process. Hence the number of variables is greater than the sets of measurements, which makes maximizing assumptions necessary.

The above factors indicate that a maximum dose approach with scientifically reasonable worst-case assumptions (under 42 CFR 83) would need to be developed for demonstrating internal dose reconstruction capability. Basically, there are two different estimation problems for maximizing approaches. The first is to estimate doses in those areas where there are significant amounts of uranium and/or radium. In these cases, uranium bioassay data and radon breath data (for estimating radium intake) are available, or can be attributed to workers through a suitable claimant-favorable co-worker data set. The ratios of Th-230, Pa-231, Ac-227, Pb-210 and Po-210 to Ra-226 are based on data from Silo 1 at Fernald, where the Mallinckrodt K-65 residues were sent for storage either directly or via Lake Ontario Ordnance Works.

Since NIOSH had suggested at the July 5–7, 2005, Board meeting that it may use radon breath data for estimating body burdens and intakes of Ra-226, SC&A reviewed the proposed use of such data in preparation for reviewing NIOSH's approach. SC&A's review of the subject (prepared by Dr. Mike Thorne) concluded that the approach was, in principle, valid (see Attachment 8).

NIOSH sent a document entitled *Dose Reconstruction Approach for Mallinckrodt Uranium Process Residues* to SC&A for review prior to the August 4, 2005, meeting in Cincinnati, Ohio (see Attachment 2b). This document describes the approach that NIOSH intends to use in determining the various radionuclides present in inhaled dust, except in areas where it can be established that uranium was essentially free of the decay products of U-234 and U-235. NIOSH has stated that, "If the worker is not a radium or thorium worker, assign intakes of associated radionuclides based on the "progeny in equilibrium" ratios. For radium and thorium workers see below." (Attachment 2b.) An examination of raw radon breath data indicates that the practical problem may be more complex. NIOSH also prepared a TIB that connected radon breath measurements to body burden of Ra-226 (TIB-0025).

SC&A reviewed the documents prepared by NIOSH and is in general agreement with the approach set forth regarding the estimation of the use of radon breath data to infer Ra-226 body burden. Specifically, SC&A found the method of TIB-0025 to be scientifically sound. SC&A also found that the result of 0.25 microcuries body burden corresponding to 1 picocurie per liter of radon breath is claimant favorable. For instance, the assumption that the fraction of radon exhaled is 0.63 yields a claimant-favorable value of radium body burden and, hence, intake.

SC&A also reviewed *Dose Reconstruction Approach for Mallinckrodt Uranium Process Residues* (Attachment 2b). SC&A is in general agreement with the approach described in this document for deriving radionuclide intakes based on radon breath data.

NIOSH proposes to use Ra-226 body burden to infer Ra-226, and then to use this value to estimate the intakes of the other trace radionuclides.[4] Since Ra-226 body burden is based on individual measurements, NIOSH's proposed method is, in principle, scientifically reasonable, provided a claimant-favorable approach to unanalyzed data is developed. Specifically, NIOSH is not using the ratios of Th-230 and other trace radionuclides with uranium, as evidenced in the Fernald K-65 Silo 1 measurements, because this material represents only one part of the mix of materials to which Mallinckrodt workers were exposed. The Ra-226 body burdens are likely to be more representative of the intakes in those areas where both uranium and Ra-226 were present in significant concentrations.

SC&A is in general agreement with NIOSH's approach for determining Th-230 and other trace radionuclides from Ra-226 body burden in areas with significant Ra-226. However, SC&A notes that great care must be taken in determining the actual ratios. The dose reconstruction result is likely to be very sensitive to the following ratios:

Th-230 to Ra-226
Pa-231 to Ra-226
Ac-227 to Ra-226

In addition, in some cases, the result may also be sensitive to the Pb-210 to Ra-226 and the Po-210 to Ra-226 ratios. This sensitivity arises from the fact that the dose conversion factors for the trace radionuclides in question are generally much higher for non-respiratory tract organs compared to Ra-226.

In the review process between the July 2005 Board meeting and the preparation of this third supplemental review of the MCW TBD, SC&A has pointed to the need to determine these ratios in a claimant-favorable manner. Specifically, the ratios of Th-230 to Ra-226, Pa-231 to Ra-226, and Ac-227 to Ra-226 in Table 7 of *Dose Reconstruction Approach for Mallinckrodt Uranium Process Residues* (Attachment 2b) were underestimated. During the August 4, 2005, Working Group meeting, NIOSH indicated that the ratios indicated in Attachment 2b were rounded values meant for methodological illustration, rather than numerical precision. NIOSH has, in fact, revised the ratios in a document sent to SC&A on August 12, 2005: *Intake Determination from Radioactive Particulate – Mallinckrodt Plant 6 Process Workers and Related Workers (Waste Handlers, Ledoux Laboratory Workers, etc.)* (Attachment 7f).

The ratios in Attachment 7f, Table 1, as they relate to the radon breath case, appear to be scientifically reasonable for high Ra-226 areas in light of the data that SC&A has reviewed, notably the K-65 data. These areas include the areas where the reprocessing of K-65 residues was done.

[4] Radon gas and radon progeny are not included in this part of NIOSH's proposed method or SC&A's review – they are separately covered in Section 3.2 below.

In the processing steps where there is very little Ra-226 and only small concentrations of uranium, the problem is more complex. The ratios in the Th-230-dominated areas, including the locations where AM-7 residues were handled, may be more uncertain in the present state of research. The ratios of Pa-231 and Ac-227 in NIOSH's Table 1 of Attachment 7f appear to be roughly in accordance with an overall mass balance for these radionuclides. However, given the importance of the ratios in the dose result, a mass-balance approach is too crude and aggregated. In the notes of the conference call of August 9, 2005, NIOSH stated the following:

> *NIOSH's current research indicates that the ratio of Pa-231 and Ac-227 to Th-230 is likely to be no more than 10-to-1 and quite possibly as low as 1-to-1.* (Attachment 6, numbered item 1 under Issue 1.)

In later correspondence between SC&A and NIOSH, the "10-to-1" value was appropriately corrected to "1 to 10" (see e-mail exchange number 4, Attachment 5). However, no further explanation or data was provided to SC&A explaining the source for the possible 1 to 1 ratio, perhaps because the available remaining time was very short. In any case, an increase in the ratio of Pa-231 and Ac-227 to Th-230 from 0.134 in Table 1, Attachment 7f, to 1-to-1 or even 0.3-to-1 would make a material difference to the dose result. Generally, the higher the ratio of Pa-231 and Ac-227 to Th-230, the higher the dose to a variety of organs.[5] Hence it is critical that the issue of ratios in the locations with low radium and low uranium, such as the AM-7 and Sperry cake areas, be addressed, documented, and resolved in a claimant-favorable manner. Note also that SC&A is not commenting on the ratio of Th-230 to uranium in the AM-7 areas, because NIOSH does not propose to use this ration to derive any dose-sensitive parameters. The same applies to the ratio of Ra-226 to uranium in the high radium areas (see text above Table 1 in Attachment 7f).

Further, the uranium bioassay data would, in general, not reflect uranium intakes in the high Th-230, but low uranium and low Ra-226 work areas. Rather uranium bioassay would tend to represent intakes in areas where more uranium was present, such as ore processing and K-65 residue reprocessing. The same is true of radon breath and Ra-226 data – they would not reflect the high Th-230 area radionuclide intakes. Hence, none of the individual measurements are useful in determining internal dose in these areas (except for the Th-230 bioassay data for the period of Th-230 (ionium) extraction from 1955 to 1957 in Plant 7E). Only gross-alpha air concentration data are available. Further, the time spent in the high Th-230 (and Pa-231 and Ac-227) areas is not well defined, so that the fraction of the dose from the high-uranium/high-radium areas and those from high Th-230 areas cannot be determined.

NIOSH has proposed to estimate a dose assuming that the air concentration in the high Th-230 areas came entirely from Th-230, Pa-231, and Ac-227 (see Attachment 7f, Table 1). This is a scientifically reasonable and generally claimant-favorable assumption. NIOSH has stated that it intends to use 95th percentile values of air concentrations in these calculations. This is a sound approach in principle. However, it is not clear that the specific value that NIOSH has proposed

[5] This observation is based on a review of the committed organ doses published by EPA as part of its Federal Guidance Report 13 library of dose conversion factors (which is available only on a CD).

for the 95th percentile of the air concentration is claimant favorable. NIOSH's approach is as follows (Attachment 7f):

> *Thorium residue (i.e., AM-7 material) intakes are estimated using the 95th percentile of the time-weighted average gross alpha air concentrations in all Plant 6 process operations. The 95th percentile air concentration corresponds to 607 dpm/m^3 or approximately 9 times the maximum allowable concentration of 70 dpm/m^3. The area air concentrations used in this analysis are about a factor of two higher than the measured air concentrations/exposures in areas associated with the AM-7 raffinate.*

This approach compares the 95th percentile value of all the time-weighted air concentrations in Plant 6 with the average time-weighted value for AM-7 areas. NIOSH has proposed that this is claimant favorable, because the former number is about double the latter. However, the appropriate value to use is the 95th percentile value for each year for the AM-7 areas estimated according to the approach consistently suggested by SC&A in its prior reviews of the MCW TBD. Moreover, it is not at all clear that the 95th percentile value of all Plant 6 time-weighted averages would be greater than the 95th percentile value of AM-7 areas. Due to the very short time available after August 12, 2005, to complete the review, and due to the focus of this review on methodological questions, SC&A did not attempt to estimate the 95th percentile for AM-7 areas. NIOSH should perform such calculations, keeping in mind the various cautions that SC&A has discussed in its prior reviews, especially SC&A 2005a and SC&A 2005b.

NIOSH has not taken into account worker testimony that the number of workdays and the hours per workday were such that a routine workweek was much longer than 40 hours. In prior reviews, SC&A recommended use of a longer workday and adjustment of breathing rate upward for those parts of the day when heavy work was done. These should be factored in when estimating intakes from air concentrations in high Th-230 areas.

SC&A also reviewed NIOSH's approach to examining the validity of the raw data and analyzing the missing data. It should be noted that SC&A has unable to verify the details of the NIOSH analysis of the raw radon breath data. This is partly because much of the electronic version of the record is illegible. Further, as noted above, SC&A has approached this review as mainly a methodological review, not a verification of NIOSH calculations. The comments below should be understood in that context.

A lower limit of detection of 0.1 picocuries per liter for radon breath is clearly indicated in the raw datasheets and is reasonable. SC&A concurs with NIOSH that the dataset is usable, because its overview of the radon breath raw data do not indicate any obvious large-scale tampering or avoidance of high results in order to lower the results of the measurements. However, there are some issues that need to be addressed in regard to those individuals who were monitored.

First, the available facilities to perform radon breath analysis were insufficient, at least as of April 14, 1950, when K.J. Caplan noted that fact in a letter to the AEC:

> *Radon breath samples are taken as a periodic routine test rather than as a pre-employment test. The volume of samples required for adequate routine testing of* ***exposed personnel*** *would be approximately ten per week; analytical facilities (furnished directly by the AEC)* ***have never been adequate for more than five samples per week****. Therefore sampling programs have never been initiated for purposes other than a routine check on exposed personnel, which application was felt to be the most important. The Health & Safety Branch, AEC-NYOO, is aware of these problems concerning sample volume and has set the present sampling rates.* [Emphasis added.] (Caplan 1950, p. 2)[6]

SC&A has not investigated when or even whether these sample-volume problems were resolved, but it is clear that for a time, persons at risk of Ra-226 exposure were monitored less frequently than was deemed desirable. This problem is compounded by the frequency of instrument failures indicated in the raw data sheets. NIOSH should investigate whether the failure to analyze many samples due to instrument failure affected the accuracy of the radon breath readings. Further, for individuals with few radon breath data points, the role of measurement error may be significant. These problems would affect individuals who have sparse monitoring data considerably more than unmonitored workers (no radon breath data) to whom the 95th percentile value is assigned. Hence, taking measurement errors into account in a claimant-favorable manner could make a significant difference to some individual dose reconstruction results. (See also comments in Section 3.5.)

Another concern is that the radon breath database needs to be screened for repeat samples and documents need to be examined for samples that may not be in the single compilation that SC&A has briefly reviewed.

NIOSH proposes to compare two values of dose from uranium decay products and use the higher of the two (Attachment 7f). The values compared would be:

(1) Dose due to Ra-226 estimated from radon breath data and other radionuclides estimated from their ratios with Ra-226, as specified in Table 1 of Attachment 7f. Workers would be assumed to have spent all their time in such areas.

(2) Dose due to Th-230 and associated radionuclides in areas with little or no Ra-226, assuming that workers spent all their time in such areas. This dose would be derived from gross alpha air concentration data and radionuclide ratios based on residue measurements.

In both cases, dose inferred from uranium bioassay data would also be assigned. The two doses would be compared, and the higher value would be chosen. NIOSH has described this approach with an illustration in its August 12, 2005 paper, *Intake Determination from Radioactive Particulate – Mallinckrodt Plant 6 Process Workers and Related Workers (Waste Handlers, Ledoux Laboratory Workers, etc.)* (Attachment 7f).

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[6] Caplan also made a similar comment in regard to termination measurements – that radon breath was not being used as a termination examination (Caplan 1950, p. 3).

In order to verify to which jobs NIOSH would attribute radon breath values (and hence Ra-226 and other trace radionuclides), SC&A reviewed the raw radon breath data.[7] SC&A found that that there were significant radon breath measurements in places such as Plant 4 and Plant 6E, where disequilibrium in Ra-226 is not expected.[8] For instance, page 99 of the document indicated that a machinist in Plant 6E, where uranium tetrafluoride was produced and reduced to metal, had a significant radon breath value (0.2 picocuries per liter). Two painters in Plant 4 (p. 97), also a metal reduction plant, are also listed as having measurable radon in the breath. Values above the limit of detection were also recorded in Plant 7, which was the new green salt plant (p. 42, p. 51, and p. 101, for instance).

NIOSH has proposed that it would allocate equilibrium ratios for uranium decay products "[i]f the worker is not a radium or thorium worker" (Attachment 2b, below Table 5). A close examination of the raw data is warranted in that it may be difficult to segregate workers into those who were and those who were not "radium or thorium" workers. NIOSH has already proposed that roving workers would be classified as radium or thorium workers, and the raw data may support that decision. But the issue deserves further close scrutiny, both at the stage of writing guidelines for Mallinckrodt dose reconstruction as well as during the implementation of the guidelines in individual dose reconstruction. This is because a decision to rule out exposure to non-equilibrium concentrations of trace radionuclide would make a central difference to the dose reconstruction result by lowering it significantly. Such a decision in the context of a maximum dose reconstruction with plausible worst-case assumptions, the criterion in 42 CFR 83, should be done in a way that corresponds to its significant consequences for claimants.

In summary, SC&A finds that the approach outlined by NIOSH is scientifically supportable for estimating maximum dose estimates with scientifically reasonable worst-case assumptions (see Section 4). However, as noted in Section 1.1 and above, the specific values that NIOSH proposes to use need further refinement to ensure their claimant favorability in light of the paucity of some aspects of available information.

3.2 PRIORITY ISSUE 2: HANDLING OF RADON EXPOSURES

The Board asked NIOSH to address, and SC&A to review, the NIOSH approach in the following areas related to radon exposure:

a. NIOSH must resolve whether sufficient radon data are available to determine job-specific radon exposures.

b. NIOSH must specify approaches for handling job-specific radon values.

c. NIOSH / SCA must further discuss and, if possible, resolve organ doses per unit intake of radon.

[7] The reference ID for this document is 9207 in the NIOSH Site Research database. Privacy considerations prevent reproduction of the actual datasheets in a public report.

[8] The functions of the various plants and their operating dates are discussed in the MCW TBD, Section 3.

NIOSH compiled and performed a statistical analysis of radon data. NIOSH also estimated non-respiratory tract data based on data from the literature regarding mobilization of radon gas from the lung and its transport to other organs of the body. The draft NIOSH data analysis and dose estimates sent to SC&A are shown in Attachment 2a.

In this evaluation, NIOSH pointed out that it had questioned SC&A estimates presented in the July 2005 review (SC&A 2005b, Table 4, p. 18):

> *. . .primarily because of the lung absorption rates that were chosen. The SC&A doses were calculated using ICRP models for two different absorption types (type F and type M for Bi-214 and type F only for Pb-214). These absorption types were chosen based on the ICRP default assumptions for lead and bismuth. The default absorption rates defined by the ICRP are intended to be used when no other specific information is available. ICRP 71 indicates that these defaults are not applicable to short-lived radon progeny (ICRP 71, page 261).*

SC&A agrees with this NIOSH criticism regarding the method for calculating the values of organ doses in SC&A 2005b, Table 4. Nonetheless, even the NIOSH calculations show that the doses from radon to non-respiratory tract organs would be non-trivial in high radon areas. Further, there is at present no ICRP-approved model for estimating radon doses to non-respiratory tract organs.[9]

NIOSH proposes to resolve the problem by using radon breath data. There are two classes of workers for whom the use of such data as a substitute for radon and radon progeny doses for non-respiratory tract organs must be shown to be systematically claimant favorable:

(1) Workers at risk of both Ra-226 and radon exposure
(2) Workers at risk of radon exposure, but possibly not significant Ra-226 exposures

Since the half-life of radon in the body is only ~24 hours, NIOSH has observed that an attribution of a Ra-226 body burden corresponding to the mean value of radon breath data of 0.3 picocuries per liter would be comparable to a continuous radon exposure to 3,000 picocuries per liter. While the exact value of the comparable radon intake is open to debate because of the uncertainties in the approach to radon dose estimation, the value of 3,000 picocuries per liter is far higher than actual 95th percentile values for radon exposure at Mallinckrodt. Hence SC&A concurs that for the category of workers not at risk of significant radium exposure, but at risk of high radon exposure, the suggested NIOSH approach is claimant favorable.

The same conclusion also applies to workers at risk of radium exposure. NIOSH proposes to attribute all exhaled radon to a Ra-226 body burden, rather than split into a fraction due to radon inhalation and a fraction due to radium intake.

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[9] SC&A did some further work on the radon dose issue and found values somewhat different (in some cases lower and in some cases higher) than NIOSH. This is not discussed in this report, since dose reconstruction for radon gas and its short-lived progeny is not proposed. Rather NIOSH proposes to use radon breath data.

The NIOSH calculations in support of its approach were sent to SC&A and are shown in Attachment 7a. SC&A concurs with NIOSH that its approach of attributing exhaled radon to radium body burden yields higher values of dose than splitting the measured radon into a component due to radium body burden and one due to exhalation of inhaled radon. It is therefore a claimant-favorable approach to resolving the problem of estimating doses to non-respiratory tract organs from radon and radon progeny. Evidently, NIOSH should use the ratios of other trace radionuclides to Ra-226 as finally developed in the calculations it proposes to address the radon dose issue (Section 3.1).

3.3 PRIORITY ISSUE 3: APPLICATION OF CORRECTION FACTORS FOR EXTERNAL DOSES TO ORGANS

a. NIOSH must specify the approach to be used to determine organ-specific and job-specific geometry assumptions to be used in dose reconstructions

NIOSH had developed some correction factors as examples of job-specific geometry prior to the July Board meeting. The NIOSH approach was discussed in the SC&A second supplemental review (SC&A 2005b). NIOSH has further developed the method and developed a draft TIB (TIB-0012) pursuant to the Board resolution on this priority issue. This draft is shown in Attachment 2c. NIOSH proposes to use a correction factor of 2.1 for organs in the lower torso for certain categories of workers. NIOSH has performed a qualitative analysis of those categories of workers who were in jobs where the correction factor would be applicable for a significant fraction of the time.

SC&A has not done a detailed verification of the analysis of job types in Attachment 2c, but has focused on evaluating the approach to applying the correction factor. SC&A concurs with the general approach that the correction factor should be applied in a claimant-favorable manner to certain categories of workers, but need not be applied to others when the postulated geometry can clearly be ruled out for a large portion of the work year. For the examples given, the correction factor of 2.1 appears to be appropriate to address the problem of organ vs. film badge location.

In its second supplemental review (SC&A 2005b) SC&A has raised issues relating to two other aspects of external dose that require correction:

(2) The correction factors also need to be further developed to account for the dependence of the dose recorded by the film badge on the angle of incidence of the photons. Film badge response declines in a non-linear fashion as the angle of incidence of the photons deviates from the normal. The diagrams of the three sources relative to the film badges, as provided by NIOSH in support of the data presented in Table 1, indicate non-normal incidence. There is also some uncertainty in the incidence angle due to (1) the movement of the badge in the course of the work, and (2) the location of the badge on the individual.

(3) *All three examples in Table 1 are based on doses in the AP geometry. Similar correction factors would be needed for PA and rotational geometry. Once all these correction factors are developed, they would need to be combined with the coefficients in Table 18 of the TBD for AP, PA, and rotational time percentage to develop an overall factor to be applied to the recorded film badge dose. In addition, the correction factors would be photon-energy-dependent. Alternatively, NIOSH may elect to employ a more simple but bounding set of correction factors, because detailed information on the geometry of actual exposures may not always be available.* (p. 13).

At its July 2005 meeting, the Board did not explicitly ask NIOSH to address these two issues as part of the Mallinckrodt priority issues, since they are generic issues relating to external dose estimation. SC&A has raised the question of the status of these two issues here because (1) it has raised it in its second supplemental review, and (2) the members of the Board Working Group indicated at the August 4, 2005 Working Group meeting in Cincinnati that they would take up these two issues at the August 24–26, 2005, Board meeting. In order to assist the Board in its deliberations, and also to provide suggestions that may expedite pending Mallinckrodt cases, SC&A prepared a short memorandum on these two issues (Attachment 9).

Both correction factors are highly dependent on photon energy. In the case of Mallinckrodt, the dominant photon energy in most cases is expected to be from Ra-226, which would put it in the 30 to 250 KeV range. The range of correction factors indicated for the angle of incidence in this case would be 1.0 to about 2.5, depending on the angle. For the examples used by NIOSH in TIB-0012, NIOSH might evaluate using one or more values in the range of 1.4 to 2.2 or a single value at the upper end of the range.

Finally, a range of correction factors of 1.4 to 2 is indicated for isotropic and rotational geometries. This is explained in more detail in the external dose section of SC&A's Task 3 report (SC&A 2005c).

Overall, correction factors in the range ~1.4 to 8 for external deep dose are indicated, depending on the organ, angular dependence, and direction of exposure, with the higher end of the range being applicable for lower torso organs.

3.4 PRIORITY ISSUE 4: ASSESSMENT OF INTERMITTENT INCIDENT EXPOSURES

a. NIOSH should determine, for a given set of bioassay data, what the approach to estimating cumulative intake will be when there are significant / infrequent incidents (e.g. dust bag ruptures, overflow of thorium solution) given the guidelines in the site profile Rev. 1, Section 6.1, item 1.

NIOSH prepared examples of dose reconstruction assuming continuous intakes and mixtures of continuous and impulse intakes for a given set of uranium bioassay data for the August 4, 2005 Working Group meeting (Attachment 3e). SC&A has reviewed these examples and agrees with NIOSH's conclusion that an assumption of continuous intake will usually be claimant favorable.

However, it must be borne in mind that this review is based on limited examples and that very unusual events and patterns of bioassay measurements might combine to yield a different result in specific, individual cases. SC&A's conclusion should be seen in the context of a TBD review, which sets forth a general approach, but which must always be used with caution and technical discretion in accordance with the facts of individual dose records.

3.5 PRIORITY ISSUE 5: SPECIFICATION OF DOSE RECONSTRUCTION METHODOLOGY FOR UNMONITORED WORKERS

The Board defined the task for NIOSH as follows:

> *a. NIOSH needs to outline scientifically defensible approaches to be used for conducting dose reconstructions for unmonitored workers (e.g. secretarial workers, maintenance mechanics, plant 1 and 2 decommissioning workers, SLAPS workers), including specific assumptions regarding radionuclide mixture.*

In the context of the approach suggested by NIOSH at the August 4 meeting, the term "unmonitored workers" can fall into several categories:

- Workers with no monitoring data whatsoever who were clerical and office workers not routinely in areas of production and handling of radioactive materials.
- Production, maintenance, decommissioning, and other non-administrative workers who were not monitored or whose records are missing.
- Workers with partial monitoring data. The most important example of this is workers with bioassay and film badge data, but no radon breath data. Another example is workers with radon breath data for only part of the 1949–1955 or 1949–1957 period.

NIOSH proposes to use the mean values of bioassay and radon breath data for unmonitored clerical workers, and 95th percentile values of bioassay and radon breath data for unmonitored (i.e., no radon breath data and/or no uranium bioassay data) production and maintenance workers and other job categories in which workers were routinely in production areas. This appears to be a satisfactory approach. Again, it should be noted that SC&A has not actually verified the various numerical estimates that NIOSH has made for illustrative purposes. NIOSH has located some data regarding Plant 1 and 2 decommissioning, which may provide useful supplemental information. SC&A has not reviewed this data. The approach mentioned by NIOSH in e-mail exchange number 5 (Attachment 5) appears, in principle, to be satisfactory. SC&A would only re-emphasize its prior cautions in regard to estimating and using 95th percentiles of air concentration data, since these were mentioned by NIOSH in this context.

Regarding environmental dose for unmonitored personnel, NIOSH has checked the internal plant air concentration values against outside air concentration data. NIOSH concluded that the indoor values are more claimant favorable for routine exposure, especially when uranium bioassay,

Ra-226, and other trace nuclide intakes are also assigned. SC&A concurs with NIOSH's evaluation of environmental data for routine exposures (Attachment 7d).

However, NIOSH has not yet addressed the issue of short-term exposures during accidents. SC&A's review of site-specific information indicates that there may be sufficient data to make a maximizing estimate of exposure to an incident to assess its significance. Specifically, the environmental air sampling data were taken at 1,000 feet, 2,000 feet and 3,000 feet.[10] The November 1950 study of environmental air concentration of uranium contains a log-log plot of air concentration with distance from the plant. The document also contains outside air concentration data relating to Plant 4, Plant 6, and both together (Harris 1950). Another document contains data for Mallinckrodt stacks, including air flow dimensions (Caplan 1948). These data, along with the emissions data and related documents, may be sufficient to determine whether the dose from one or a few incidents lasting for a short time may be important, and whether it warrants an addition to the environmental dose for unmonitored personnel.

It will also be important to consider trace radionuclides in this context. SC&A understands that routine releases of uranium would be unlikely to have large disequilibrium components of trace radionuclides since these were wet processes, as NIOSH has noted. However, the same conclusion is more difficult for incidents. For instance, K-65 residues were brought back from SLAPS in 1949 and these cannot be presumed to be wet. The transient air concentrations under such circumstances could have non-negligible dose implications. SC&A has not investigated this issue in detail, but did raise the issue of short-term concentrations as a potential problem to be addressed during the August 4, 2005, Working Group meeting and again during the August 9, 2005, conference call (Attachment 6).

3.6 PRIORITY ISSUE 6: EXAMPLE DOSE RECONSTRUCTIONS FOR REPRESENTATIVE CASES

The Board requested NIOSH to demonstrate its proposed approach to internal dose reconstruction using examples with actual individual worker data. NIOSH agreed to do so in a manner that would protect the privacy of workers and their families. The Board's request was as follows:

> *Representative cases should include the following:*
>
> *a. Example internal dose reconstruction for plant 6 subject involving K-65 residues processing for U extraction (where urine, air and breath radon are available) for selected metabolic and non-metabolic organs.*
>
> *b. Example internal dose reconstruction for plant 6 case involving K-65 residues processing for U extraction (where urine, air data are available but no breath radon) for selected metabolic and non-metabolic organs.*

[10] Denise Brock kindly provided the documents discussed in this paragraph to SC&A during the August 4, 2005, meeting.

c. Example internal dose intake estimate for plant 7 thorium extraction worker (1955–1957).

NIOSH distributed four case studies showing how the approach to internal dose would be implemented at the August 4, 2005, meeting (Attachments 3a though 3d). These examples indicate that NIOSH's assumptions regarding allocation of intakes for unmonitored workers yield higher doses than when actual monitoring data are taken into account. Such a result is to be expected, since 95th percentile values are being allocated to workers exposed to production areas on a routine basis.

NIOSH has also located thorium bioassay samples and presented a calculation using one of these data points (Attachment 3c). The bioassay sample was dated April 8, 1955. However, the TBD lists the start of production scale thorium-230 operations as July 1955 (Table 31F, p. 228). During the August 4, 2005, Working Group meeting, NIOSH indicated that thorium production work in Plant 7E may have begun earlier. NIOSH should verify whether this earlier work was production-scale or only pilot-scale work. Adjustment of early bioassay samples prior to the start of production-scale work may be necessary if the values are projected into production-scale operations.

SC&A notes that in Case 3 (Attachment 3c), there was only one thorium bioassay sample. This is clearly an insufficient basis for claimant-favorable dose reconstruction. NIOSH needs to develop a procedure to handle cases of Th-230 extraction workers with only one or two Th-230 bioassay data points. NIOSH should construct a co-worker database for the 1955–1957 Th-230 extraction process and develop a procedure comparable to that for using co-worker radon breath data. For instance, in the case of a few samples, NIOSH should consider using the 95th percentile value of those samples. In case of a single sample, NIOSH should evaluate using the higher of the 95th percentile value of that sample as a constant input, given the measurement error, and the 95th percentile of the co-worker distribution, also as a constant.

SC&A's review of the approach used by NIOSH for cases with full or partial monitoring data indicates that the suggested approach to internal doses is sound in principle. For detailed discussion regarding cautions, see Section 3.1 above. Further, SC&A provides the following as reminders of issues that have been raised in the past:

(1) The Barnes hospital data for 1949 and early 1950 need a correction factor (SC&A 2005b).

(2) The length of the work year needs to be adjusted upward.

(3) Oro-nasal breathing needs to be considered, even for light work.

(4) While an average breathing rate of 1.2 m^3/hour over a whole year is appropriate, it needs to be augmented for short periods of known heavy work, especially as these would be more likely to be associated with higher air concentrations.

Points 2 though 4 above are only relevant if air concentrations are used in dose calculations.

4.0 MATTERS RELATED TO 42 CFR 82 AND 42 CFR 83

NIOSH has indicated that the new approach it has set forth for Mallinckrodt dose reconstruction for the 1949–1957 period "is by way of a best estimate with some maximizing aspects where NIOSH cannot do better (as distinct for the efficiency approaches to maximum dose using the Hanford radionuclide list, for instance)" (see Attachment 6, last paragraph). This statement was made in the context of how the doses estimated by NIOSH using this approach would be used by the Department of Labor (DOL) in its compensation decisions. Hence, it appears to be in the context of how decisions would be made under 42 CFR 82.

This SC&A evaluation of the approach that NIOSH has suggested has been in the context of the maximum dose with scientifically plausible worst-case under 42 CFR 83. SC&A had noted the terminological issues regarding the use of "maximum dose" in two different contexts in the regulations in its second supplemental review (SC&A 2005b). SC&A reaffirms its earlier findings that a "best-case estimate" with some claimant-favorable assumptions is not possible with the given data, as SC&A has so far interpreted this term in its TBD reviews. The basis for this view was discussed in an earlier SC&A review of the MCW TBD and has also been briefly discussed above (Section 3.1). However, it should be noted that the SC&A conclusion regarding a "best" case dose estimate was made in the context of the term as it has generally been used in the context of 42 CFR 82.

In the context of a decision regarding dose reconstruction capability under 42 CFR 83, NIOSH appears to have clarified and interpreted the term "maximum dose" with plausible assumptions as a "best estimate with some maximizing aspects where NIOSH cannot do better" which would be used by DOL for both compensation and denial of claims. Without going into further terminological discussion in the context of this TBD review, SC&A concurs with the NIOSH conclusion as to the use of the approach, should the SEC petition be denied for the 1949–1957 period. However, SC&A stresses that this review has been focused on the six priority areas of dose reconstruction methodology specified by the Board in the context of a TBD review. SC&A has not assessed the impact of these six issues for all cancers and all members of the SEC class as it is currently defined, since that was well beyond the scope of the task allocated by the Board. Finally, a number of refinements remain to be completed before the approach proposed by NIOSH can be applied to actual, individual dose reconstruction. SC&A's suggestions in regard to completion of the development of the dose reconstruction approach are summarized in Section 1.1. The technical basis for these suggestions is discussed in Sections 3.1 through 3.6.

5.0 REFERENCES

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Attachments to the
Third Supplemental Review of the
Mallinckrodt Site Profile Revision 1 (Effective Date March 10, 2005)

August 16, 2005

ATTACHMENT 1

Priority Issues for Demonstrating Feasibility of Dose Reconstruction for MCW Destrehan Street Workers for the Time Period of 1949 to 1957: List of Tasks Developed by the ABRWH July 6, 2005

The following is a list of tasks to be completed by NIOSH and delivered to the ABRWH workgroup and SCA, Inc. for resolution. To allow for an adequate amount of time for SC&A/NIOSH/Board to complete comment resolution by August 23 the following schedule is set forth:

- Workgroup conference call for status report and task clarification by July 26
- NIOSH will provide a draft report on the following tasks in consultation with SCA by July 31
- Work Group meeting between July 31 and August 8
- SCA to review the NIOSH response to the tasks and issue report to Board by August 16 (1 week before Board meeting)
- Work group conference call for comment resolution between August 16 and August 22

List of Tasks

1. Handling of Raffinate exposures
    a. NIOSH should specify the radionuclide ratios (Ra-226, Th-230, Ac-227, Pa-231) for all ore processing (including non-pitchblende ores)
    b. NIOSH should specify the radionuclide ratios for K-65 uranium extraction (including processing steps and filtration steps)
    c. NIOSH should specify how ratios will be applied by job title (including roving workers and cases where no job title are available)
    d. NIOSH should specify approach for estimating intake when any combination of urine, air sampling and breath radon data is available. In this context NIOSH should consider the reliability of breath radon for determining Ra-226 intakes.

2. Handling of Radon exposures
    a. NIOSH must resolve whether sufficient radon data is available to determine job specific radon exposures
    b. NIOSH must specify approaches for handling job specific radon values
    c. NIOSH / SCA must further discuss and if possible resolve organ doses per unit intake of radon

3. Application of correction factors for external doses to organs
    a. NIOSH must specify the approach to be used to determine organ specific and job specific geometry assumptions to be used in dose reconstructions

4. Assessment of intermittent incident exposures
    a. NIOSH should determine, for a given set of bioassay data, what the approach to estimating cumulative intake will be when there are significant / infrequent incidents (e.g. dust bag ruptures, overflow of thorium solution) given the guidelines in the site profile Rev. 1 section 6.1, item 1

5. Specification of dose reconstruction methodology for un-monitored workers
    a. NIOSH needs to outline scientifically defensible approaches to be used for conducting dose reconstructions for un-monitored workers (e.g. secretarial workers, maintenance mechanics, plant 1 and 2 decommissioning workers, SLAPS workers) including specific assumptions regarding radionuclide mixture

6. Example dose reconstructions for representative cases

Representative cases should include the following:

a. Example internal dose reconstruction for plant 6 subject involving K-65 residues processing for U extraction (where urine, air and breath radon are available) for selected metabolic and non-metabolic organs.
b. Example internal dose reconstruction for plant 6 case involving K-65 residues processing for U extraction (where urine, air data are available but no breath radon) for selected metabolic and non-metabolic organs.
c. Example internal dose intake estimate for plant 7 thorium extraction worker (1955-1957).

ATTACHMENT 2a

Dose Reconstruction Approach for Mallinckrodt Uranium Process Residues

Introduction

An extensive review of Mallinckrodt source term information was made to develop the Mallinckrodt Site Profile and its subsequent revision. Questions concerning the disequilibrium of uranium progeny in the residues, and subsequent exposures to these disequilibrium fractions of progeny have resulted in a major effort to again consider all available information to address the compositions of the radioactive source term at Mallinckrodt. This source term information will be used to estimate organ doses for the purpose of making compensation decisions under EEIOCPA. This means that where information is uncertain the interpretation that results in a higher dose will be used, although it might result in significant, although not implausible, overestimates of dose.

A number of data sources are available to address the issue of source term at Mallinckrodt. These include information on uranium ore types and composition, process information, residue information and dosimetry information. While this document provides an overview of uranium ore type, composition, and process information, the primary focus is on the identification of the residues, their radioactive components, and the available Mallinckrodt monitoring information. The focus is on the years 1949 through 1957, although data are available from other periods and were sometimes considered. A more complete treatment of the Mallinckrodt site is contained in the profile document. The goal of this document is to establish uranium progeny activity fractions relative to uranium activity and to describe how these are to be used in reconstructing doses.

RAFFINATE RADIONUCLIDE RATIOS

Uranium Ore

From 1942 to 1957, Mallinckrodt processed approximately 50,000 tons of uranium. Uranium was extracted from two main types of ore at Mallinckrodt. These included **domestic ores** from Canada and the United States, with typical uranium content in the range of 1 to 10% by weight and **African ores** with uranium mass concentrations in the range of 20 to 80%. In addition Mallinckrodt received **ore concentrates**, which were residues from ores that were chemically processed typically to remove precious metals and/or radium. Some of the African and Canadian ores were sometimes referred to as pitchblende. The uranium mass fractions in the concentrates typically were higher than the uranium concentrations in the unprocessed ore and the progeny may have been in significant disequilibrium, especially if the ore was processed to remove radium or other constituents prior to shipment. This would, of course, result in lower doe per unit intake than the un-concentrated pitchblende ores. In later years, AEC specified acceptable uranium concentrate characteristics, and the specified uranium mass concentrations were similar to the range for African ores.

Uranium ore contains some Th-232. In 1967, AEC reported that carnotite and pitchblende had, respectively 30 and 8 times as much Th-232 as Th-230 by mass residues. A comparison of the respective activity fractions shows that Th-232 would be more than a factor of 2E-4 less than Th-230, and therefore Th-232 and its progeny (including Th-228) would not significantly contribute to the total thorium dose. A review of uranium ore concentrates information related to uranium and thorium (mass) percentages supports the conclusion that accounting for Th-232 separately will not result in a significant change (less than 3% and most likely less than 0.03%). Typical methods of measuring thorium in urine and on air samples during the period of Mallinckrodt operations would not differentiate between thorium isotopes, and so would account for Th-232, Th-230 and Th-228 activity. Spectrometry techniques were available, but their use would have been unusual for analysis of routine urine and workplace samples.

Uranium Process Information

Attachment A shows flow diagrams of the typical uranium processes as well as residue endpoints.

The feed material was prepared for uranium extraction, as follows. Milled ore or ore concentrate was taken from storage, thawed if necessary in an enclosed “thaw oven” in the thaw house, and dried. It was then digested in nitric acid in a digestion tank. Sulfuric acid was added to the solution in the vessel to precipitate the radium and lead as sulfates. The uranium was left in solution as uranyl nitrate and the precipitate was filtered out; the filter cake arising from this process was the K-65 or "gangue lead" cake (GLC). This cake contained nearly all of the radium and lead that had been in the feed material.

Next, a slurry of barium carbonate was added to the uranyl nitrate solution to remove the rest of the sulfates, including the residual radium. To remove the solids, the mixture was run in a continuous solid-bowl centrifuge. The uranium remained in solution as uranyl nitrate and the precipitate formed a barium sulfate cake, AJ-4.

The uranyl nitrate solution was then boiled to concentrate it, calcium nitrate was added, and the solution was filtered to remove any solids formed. In the two-step ether extraction process, diethyl ether was added to the solution in extraction columns; the purpose of the double extraction was to remove the acid-insoluble molybdenum. The first step was the acid ether extraction, including ether addition, nitric acid addition, and re-extraction into water (also called a “water wash”). The second step was a neutral ether extraction followed by re-extraction into water. In the extraction process, the isotopes of thorium and protactinium were left in the “aqueous phase” while the uranium was stripped off by the solvent (ether) into the organic phase. The organic phase was boiled to form uranyl nitrate hexahydrate; the aqueous phase was limed and filtered, with the resulting cake being the raffinate or AM-7 cake.

At times, the precipitate from the first extraction column was dewatered using a Sperry filter press, producing a supernate and the so-called Sperry cake (later used as a source of Pa-231 by Mound). The supernate from this press was added to and treated with the aqueous phase of the extraction, as described above.

Uranium Process Residues

As Attachment A shows there are number of places in the uranium process where residues were created. The residues were referred to as raffinates, precipitates, cakes or wastes, and were given name and/or code designations. Although the code was used to describe specific materials, the informal names did not always mean the same thing in various reports. This could be due to lack of a designator for a particular residue, non-specificity of the designator in a particular program, and misapplication of the designator by some writers. The designator conventions for the residues used in this document are described later in this report. The differences in term usage in the many reviewed reports will not significantly affect the conclusions in this report.

Descriptions of the residues follow, as does a summary of the activity fractions of the progeny relative to uranium. It should be noted that the residues would have been wet at the time of initial generation, and descriptions of most of the wastes note moisture contents. The presence of moisture in the residues is likely to reduce generation of airborne material. The data used to derive the activity fractions are shown in Attachment B.

Radium-Bearing Residues, Including K-65 and Barium Sulfate Cake (AJ-4)

As noted earlier, the radium and lead were separated from the uranium in an early processing step. In 1948, a change in process resulted in more efficient separation of the uranium from the radium-bearing K-65, which resulted in a higher radium:uranium ratio in the residue and a lower ratio in the bulk of uranium-bearing material. The residue from processing African pitchblende (for the most part Q-11), was code-named **K-65** and contained a large fraction of the original radium activity in the ore. African Metals retained ownership of the K-65, which was drummed and shipped offsite, because of its increased radiological hazard. When the more efficient uranium recovery process was developed, the K-65 from prior processing was brought back from storage and reprocessed from early 1948 to about the middle of 1949 to recover additional uranium. In April 1949, the K-65 was shipped in drums to Lake Ontario Ordnance Works (LOOW) for storage, and in 1952 was subsequently transferred to the Feed Materials Production Center in Fernald, Ohio, where Mallinckrodt K-65 was also received directly and placed into storage Silo 1. Mallinckrodt (Anonsen 1953) noted in December 1953, "Changes in the raw materials used in the C-3 process [the enhanced uranium recovery process] have been made on the basis of October experience so as to reduce the uranium concentration in the K-65 [still further]".

Other ores and ore concentrates produced some radium-bearing residues. Over time, processing became more efficient at recovering uranium, thus changing the ratios of radium to uranium. In fact some of the K-65 was reprocessed from early 1948 to about the middle of 1949 to recover additional uranium. This reprocessing would have resulted in increased radium:uranium ratios in the feed as well as the residues.

Estimates of radionuclide concentrations are made for the K-65 and for the reprocessed K-65 residue. To estimate progeny to uranium ratios for K-65, a number of different data sets were reviewed.

Fernald (1993) included the most comprehensive listing of radionuclides in K-65, which were derived from sampling the residues in Silo 1, which originated from Mallinckrodt. A record review makes it clear that at least 75% (18,000 drums) of the material in Silo 1 was brought directly to Fernald from Mallinckrodt; the other 25% (6,000 drums) was material that was originally transported to Lake Ontario Ordnance Works (LOOW) for storage and later brought to Fernald. Documentation supports the assumption that the drums from LOOW contained K-65 that originated from Mallinckrodt. An additional 7,000 K-65 Mallinckrodt drums went into Silo 2, but this Silo also included K-65 from Fernald, which although its composition was likely similar to that from Mallinckrodt, subtle differences in composition may exist. (Silo 3 at Fernald included metal oxides, and no K-65.)

The barium sulfate residue, **AJ-4**, was also a radium concentrate. The barium sulfate processing was referred to as leaching. Leaching appears to have been an optional step, because there are references to leached and unleached residue. Reprocessing of some residues occurred in late 1949 to the mid-1950s, but only as time permitted. (There is a 1953 memo that says that barium sulfate reprocessing (leaching) could not be done one month because the regular production needs were so high.)

Other Residues

Residues, other than K-65, were sometimes referred to generically as the raffinates, and this term usually did not included residues with elevated radium concentrations. However, it appears that "raffinate" workers could refer to workers, who handled any of the residues. The non-K-65 residues were sometimes generically referred to as **airport cake**, because of their storage at the St. Louis Airport Site. Later, some of these residues were eventually transferred to Cotter Corporation and were later referred to as the **Cotter Concentrates**.

The supernate from the barium carbonate was processed further to obtain **AM-7**, a material that had elevated concentrations of thorium. As part of an effort to obtain ionium (Th-230), about 350,000 tons of AM-7 were processed at Mallinckrodt from 1955 to 1957

Sperry cake was obtained from the first wash of the Sperry filter. The washing of the Sperry filter occurred every few days and the cake was referred to as white and pasty with a 50% moisture content, and was known to be a good source of protactinium-231.

Summary of Source Term Information

Based on review of the source terms at Mallinckrodt, it was determined that three source terms, plus radon dominated internal radiation exposures at Mallinckrodt from 1949 through 1957. A fourth source term, uranium without its progeny also existed at Mallinckrodt, but is not included in this analysis because it will result in lower exposures when starting from bioassay data. The source terms in this document are all presented as activity ratios relative to uranium activity, unless otherwise noted.

Uranium in equilibrium with its progeny is found in ores and some ore concentrates and was probably the dominant source of internal exposures at Mallinckrodt. Although some ore

concentrates and processed uranium were depleted in radium, to expedite this analysis, progeny are assumed to be in equilibrium. Table 1 shows those radionuclides that dominate internal dose from the natural uranium series source term.

Table 1. Activity Fractions Relative to Uranium in Mallinckrodt Source Terms

Source	Progeny in equilibrium	Radium enriched	Thorium enriched
Uranium	1	1	1
Th-230	4.89E-01	50	100
Ra-226	4.89E-01	400	-
Pb-210	-	200	-
Po-210	4.89E-01	200	-
Pa-231	2.28E-02	5	10
Ac-227	2.28E-02	5	10

Radium activity dominates the residue source term at the front end of uranium ore processing. Because the radium residue source term generally results in higher doses than residues produced later in the uranium process, it is the default residue source term for 1949 to 1957. In addition, reprocessing of K-65 and later barium sulfate between 1948 through 1950 would have resulted in increased exposure to radium at the feed end of the uranium process. The radium source term shown in Table 1 is based primarily on analysis of K-65 residue in Fernald's Silo 1.

Thorium activity dominates the AM-7 source term. No measurement have been found to indicate directly what fraction of protactinium -231 might be in the AM-7, but there is some information regarding protactinium-231, as well as actinium-227 in the related Sperry Cake. The thorium source term is based on a consideration of a combination of information about AM-7 and Sperry Cake. Some workers were involved in reprocessing AM-7 during the period 1955 through 1957. The thorium residue source term is shown in the Table below. It should be noted that although the thorium is listed as Th-230, Th-232 and Th-228 were also considered in the ratio when information identified these latter isotopes.

INTAKE DETERMINATION

Overview

In accordance with 42 CFR 82.18, intakes are to be estimated from individual bioassay information, when available. In accordance with 42 CFR 82.17, coworker bioassay data are considered when individual bioassay data are unavailable. For Mallinckrodt dose reconstruction, characterization of the source term (above and in the site profile), coworker bioassay information, and workplace monitoring (in the form of air monitoring) are used to supplement individual bioassay.

For most Mallinckrodt radioactive material processing and storage areas, some of the uranium progeny will add a considerable fraction of the internal dose to some organs. The exception is for areas where only uranium metal was handled. Three source terms are considered for assigning intakes from progeny:

- Uranium in equilibrium with progeny
- Radium residue source term (based primarily on K-65 waste in Silo 1)
- Thorium residue source term (based on airport cake, including consideration of AM-7 and Sperry cake residues)

The information that follows includes very brief information regarding parameters related to uranium, radium and thorium intakes based on individual bioassay. This is followed by analysis of the source term resulting in worker internal exposures versus the source term in the residues. Finally information on assigning intakes when there are no individual bioassay, and information on worker categorization is provided.

Uranium

When **uranium urinalyses** results are available, they should be used to estimate uranium intake(s). The detection threshold is assumed to 10 ug/L of uranium. If uranium urinalysis results are not available for a worker or for an extended period of employment, uranium intake rates estimated for coworkers based on compiled bioassay results can be used.

Thorium

Thorium urinalysis results have been located for some workers, and should be used for dose reconstruction. Most results are reported as a value followed by a +/- and another value, which is assumed to be similar to a one-sigma error. Twice the assumed error will provide an estimate of the detection threshold. It is suspected that most workers who had thorium bioassay results were **ionium (Th-230)** workers, engaged in recovery of thorium from AM-7. An electronic tabulation of thorium bioassay results is not currently available. As soon as these electronic data are tabulated, they will be reviewed for possible use in determining coworker intakes for individuals associated with the ionium process area, but who have no thorium bioassay.

Radium

The **radon breath analyses**, when available for an individual, should be used to estimate Ra-226 body activity using the method in ORAUT-OTIB-0025. Although there are some factors that could cause radon breath results to be underestimated, the available information indicates that the collection of these samples would likely result in reasonable estimates of radon in breath for the purpose of this program, in other words, no information has been found to indicate that these results would have been underestimated. The records indicate that a number of Mallinckrodt samples were collected in areas of elevated radon, which resulted in the breath radon emanations from radium within the body to be overestimated. In addition, breath radon samples should not have been collected within one to two hours after a meal. (Eating appears to increase the radon breath concentration and waiting would reduce the body emanation concentration by up to a factor of 2 and thus allow for appropriate application of the factor to convert from radon breath concentration to radium activity concentration within the body.)

Radon

Area radon data are used to assign radon exposure. CER radon data for Plant 6 and subsets of the Plant 6 data by areas were considered. A set of low exposure data will also be considered to assign radon exposure to workers who did not enter areas associated with uranium and associated processing or storage. Data for SLAPS were also analyzed. Based on the SLAPS preliminary results, additional data from identified as Scalehouse and Ore room will be analyzed before finalizing radon exposure assignments.

The CER data were analyzed by assuming the data are lognormally distributed. Data were ranked from low to high. The z-scores of the ranks and the transformed data were fit to a line for determination of the 50th percentile and geometric standard deviation of each distribution in accordance with ORAUT-OTIB-0019 methods. Results that were "less than" results were ranked, but not fit. No attempt was made to sort data prior to analysis to remove those data that were not representative of occupied areas, for instance, measurements taken in a drum head space or in the thawing furnace would likely have been included. For some areas, the large geometric standard deviations provide some indication that two different concentration distributions are being measured. The analysis results are shown in Attachment C, and a summary is shown in Table 2.

Table 2. 1949-1957 Range of Radon Exposure Distribution Parameters Based on Data in CER

Location	50th (pCi/L)	GSD	84th (pCi/L)	95th (pCi/L)
Plant 6	3.36-18.9	3.3-6.6	20.4-72.2	58.9-244
C-3	4.93-34.7	2.7-9.9	32.7-231.6	78.9-1012
FEinc	3.57-28.06	2.2-5.3	13.9-66.5	33.3-181
K-65 Centrifuge	3.14-13.3	2.2-8.3	14.6-59.5	23.9-192
SLAPS (1948-1949)	53.3-84.7	4.3-7.3	231-622	595-2249
Summary	**All < 100**	**-**	**Most less than 100**	**Most less than 200**

To determine exposure due to a specified radon concentration (assumed to be Rn-222), the concentration C in pCi/L was converted to potential alpha energy exposure (PAEE) per year units of working level months (WLM)/y using:

$$\mathbf{PAEE = C * F* M / (100\ pCi/L\ per\ WL)}$$

where the equilibrium factor F was taken as 0.4, a value recommended by the ICRP (1981) and the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR 1993) and M is the number of months of exposure, 12, in a year, where a month is assumed to be 170 work-hours.

To assign radon exposures, workers are categorized into three exposure categories as shown in Table 3.

Table 3. Radon Exposure Assignments

Category	Description	Exposure (WLM/y)	Basis
General Plant	Did not work in or routinely enter process or associated radioactive material storage/waste areas.		95th percentile of non-process and non-storage area results
Process Areas	Worked in on near process areas, but did not encounter ore products, residues or associated storage areas directly.	1949 10.9pCi/L 6.62GSD 1950 18.9 3.57 1951 17.0 3.39 1952 11.1 4.30 1953 9.41 4.11 1954 8.41 5.65 1955 9.74 4.68 1956 4.99 4.48 1957 3.37 6.04	Lognormal distribution of the Plant 6 includes the possibility of lower and higher exposures
Process Workers	Includes operators, ore samplers, residue samplers, laborers, maintenance workers, etc. All SLAPS workers would be included in this category, including guards.	1949 244 1950+ 200	95th percentile of …

Unmonitored workers, work periods or radionuclides

This section provides estimations of uranium and radium intakes based on coworker data, whose analyses are shown in Attachment D. Workers and work periods for which non-uranium exposures would dominate internal dose are also identified. Source term information is provided in the form of a table **estimating intakes of other radionuclides relative to uranium**. (Note, that generic source term information only applies if there are no bioassay to assess an intake of a particular radionuclide.) General periods of exposure for certain source terms are provided, but claim-specific information supercedes the provided information. To calculate intakes from air concentration data, when no information regarding work hours us available, an exposure period of 2000 hours or an air intake of 2400 m^3/workyear is assumed.

Air concentrations

Air sampling information is useful to check the reasonableness of calculations from other methods of intake estimating, but is generally reserved for use in estimating intakes when no other data are available. When time-weighted air concentrations are available for an individual, these could provide temporal information for exposure assumptions. Most air sample measurements were made using gross alpha techniques. Gross alpha measurements would detect all alpha activity and the result would be apportioned to the different alpha emitters. (Fluorimetry was sometimes used to detect radioactivity on air samples. This was a less commonly used technique, but in the case of fluorimetry measurements, the concentration would

be assigned to uranium, and other radionuclides would be accounted by using their ratio of activity relative to uranium activity.) To reiterate, the primary use of the air concentration information is qualitative, with a major exception being area radon concentrations, which are used to assign worker exposures.

Consideration of Bioassay, Air Concentration and Source Term Data in Estimating Intakes and Intake Ratios for Unmonitored Radionuclides

CER uranium urinalyses and radon breath analyses were used to determine Ra:U intake ratios. Ratios based on the 50th percentile intake regimes indicates that for type M uranium, Ra:U is 39 in 1949, and falls to 28 in later years. For type S uranium, Ra:U is less than 5. The ratio is lower for the 95th percentile regimes, about 18 for type M, and again less than 5 for the type S regimes. It is notable in 1949 that the Ra:U ratio is less than 1.

The air concentration data have also been used to provide an assessment of the reasonableness of the intake estimates of the non-uranium radionuclides based on source term fractions. The average CER time-weighted air concentration for 1949 through 1952 was 8.28 MAC (maximum allowable concentration), which would represent an alpha intake rate of 1716 pCi/d. The maximum result in the CER database during the period 1942 to 1952 was 196 MAC (ore room in 1949), which would equate to an alpha intake rate of 40,600 pCi/d.

Assuming that the maximum time weighted air concentration is bounding, the derived intake rate from that air concentration shows that the non-uranium alpha intake ratio to uranium could not exceed 150 for a type M uranium intake, and that the radium intake would not exceed 92 (assuming the ratios for the radium enriched source term, and that the fraction of radium alpha activity to other alpha activity is about 400/650). For a type S uranium intake, the effect is more dramatic, indicating that non-uranium alphas would not exceed a factor of 4 larger than U, and that Ra:U would be 2.3. For the 95th percentile uranium intakes the maximum Ra:U ratio is 8.3, and an intake of radium (or any other alpha emitter) in addition to type S uranium, is not supported by the air data. It should also be noted the 95th percentile radium intake and the 95th percentile uranium intake for 1949 are both larger than the intake determined from the maximum time-weighted air concentration in the CER database.

Intakes for several workers, who had uranium urinalyses as well as radon breath analyses and/or thorium urinalyses are being assessed to assure the validity of parameter considerations. Initial analyses indicate that the largest Ra:U intake ratio is 69, and that the largest Th:U intake ratio is 101, based on consideration of bioassay results only.

The conclusion from this assessment is that the Ra:U and Th:U ratios based on source term information appear to represent maximum intake ratios. This appears to indicate especially for the radium source term, that workers were not exposed to a source term including only K-65 during any given year. This is also consistent with information that job rotations were required for some jobs primarily to keep external and radon exposures below limits. Based on bioassay, air concentrations, and source term considerations, a maximum Ra:U ratio of 100 appears to be more supportable; however, because there are coworker bioassay it is recommended that radium intakes be assigned based on analyses results of coworker data. Ratios of the radium-associated

radionuclides are reduced by a factor of 4 relative to uranium based on the maximum Ra:U ratio, but assignment of intakes of these radionuclides should be based on their ratio to radium. The reconsidered radium-associated ratios are shown in Table 4.

Coworker data for thorium bioassay have not yet been compiled and analyzed yet, but the intake ratios based on thorium source term, as well as initial bioassay analyses appear to be reasonable. It was not certain that the maximum air concentration of 196 MAC could also be applied to the 1955 through 1957 period, and so no comparison with air data was made for the thorium source term. However based on greater concerns about thorium exposure and the general downward trend in exposures throughout the 1950s at most AEC contractors, it is suspected that 196 MAC would represent a maximum annual exposure concentration.

Table 4. Activity Fractions Relative to Uranium for Mallinckrodt Dose Reconstruction

Source	Type	Progeny in equilibrium	Radium enriched	Thorium enriched
Uranium	M, S	1	1	1
Th-230	M, S	0.489	12.5	100
Ra-226	M	0.489	See coworker data	-
Pb-210	F	-	50	-
Po-210	M	0.489	50	-
Pa-231	M, S	0.0228	1.25	10
Ac-227	M, S	0.0228	1.25	10

Assignment of Uranium Intakes for Unmonitored Workers or Unmonitored Periods, and Assignment of Uranium-Associated Intakes

For workers with no associated bioassay, the coworker uranium intakes based on the CER data provide a basis for uranium dose determination. The CER data is a set of worker and workplace monitoring data compiled by ORAU for epidemiological studies. The compiled worker information only includes data for white males. Reviews of the accuracy of the compiled data for urinalysis results, performed by individuals from SC&A and NIOSH, revealed small differences between the original data and the CER data. ORAUT did find a small number of differences in dates, results and type of data entered, but does not feel these few differences will grossly change summary analysis results of the CER data. (These differences were readily apparent in the claim file data, and so the appropriate data are readily available for use in individual dose reconstruction.) The lognormal distribution parameters were determined from the CER uranium urinalysis results for the years 1949 through 1957 using methods in ORAUT-OTIB-0019.

If uranium urinalysis results are not available for a worker or for an extended period of employment, uranium intake rates estimated for coworkers based on compiled bioassay results may be used. The lognormal intake distribution information derived from the coworker data is used to estimate organ dose distributions for workers, who were unlikely to have had assignments associated with uranium and associated radioactive material processing operations, as determined from claim specific information. For workers, who were likely to be associated

with process work or process work areas, the 95th percentile of the CER derived intake distribution is used to determine organ doses and is input into IREP as a constant distribution. These uranium intakes are shown in Table 5.

The source term is natural uranium, but U-234 can be used for intake and dose calculations once results are converted to activity. Either type M or type S uranium is chosen, but not both. The IREP dose distribution type is either lognormal for the 50th percentile intakes or constant for 95th percentile intakes.

Table 5. Uranium Intakes Based on Lognormal Distribution, and 95th Percentile CER Urinalysis Data

			50th percentile intake rate (pCi/d)	GSD	95th percentile intake rate (pCi/d)
Type	**Start**	**End**	**Workers outside process, storage and waste areas**		**Process workers and workers in process buildings**
M	1/1/1949	1/1/1950	270	4.15	2,810
	1/1/1950	1/1/1958	95.8	2.79	517
S	1/1/1949	1/1/1950	8440	3.87	78,000
	1/1/1950	1/1/1958	762	1.87	2,130

If the worker is not a radium or thorium worker, assign intakes of associated radionuclides based on the "progeny in equilibrium" ratios. For radium and thorium workers see below.

Assignment of Radium-226 Intakes for Unmonitored Workers or Unmonitored Periods, and Assignment of Radium-Associated Intakes

If radon breath analysis results are not available for a worker categorized as a radium worker, radium intake rates estimated for coworkers, summarized in Table 6 and based on compiled bioassay results from CER may be used to assign radium intakes.

For radium workers with no associated bioassay, the coworker radium 95th percentile intakes based on the CER data provide a basis for radium dose determination. The IREP dose distribution type is constant.

Table 6. Radium Intakes Based on 95th Percentile CER Radon Breath Analysis Data

Type	Start	End	Intake rate (pCi/d)
M	1/1/1949	1/1/1950	49,600
	1/1/1950	1/1/1958	8,780

If the worker was not a thorium worker during the radium period of intake, assign intakes of associated radionuclides based on the ratios in Table 7. If the worker is also a thorium worker during the radium intake period, assign only the radium-associated intakes from Pb-210 and Po-210 in this section.

Table 7. Intake Ratios Relative to Radium

Source	Intake ratios relative to radium	Comments
Th-230	0.125	Do not assign during Th work periods
Pb-210	0.5	
Po-210	0.5	
Pa-231	0.0125	Do not assign during Th work periods
Ac-227	0.0125	Do not assign during Th work periods

Assignment of thorium-230 intakes for unmonitored workers or unmonitored periods, and assignment of thorium-associated intakes

Thorium intakes can be derived from bioassay or calculated based on an intake of thorium. If no bioassay are available for a thorium worker, either an individual or coworker uranium intake is used to estimate the intake of thorium and associated radionuclides based on the ratios in the second column of Table 8. For thorium intakes derived from bioassay, the third column is used to assign intakes of Pa-231 and Ac-227.

Table 8. Thorium Intake Ratios Relative to Uranium and Thorium

Source	Intake ratios relative to uranium	Intake ratios relative to thorium
Th-230	100	1
Pa-231	10	0.1
Ac-227	10	0.1

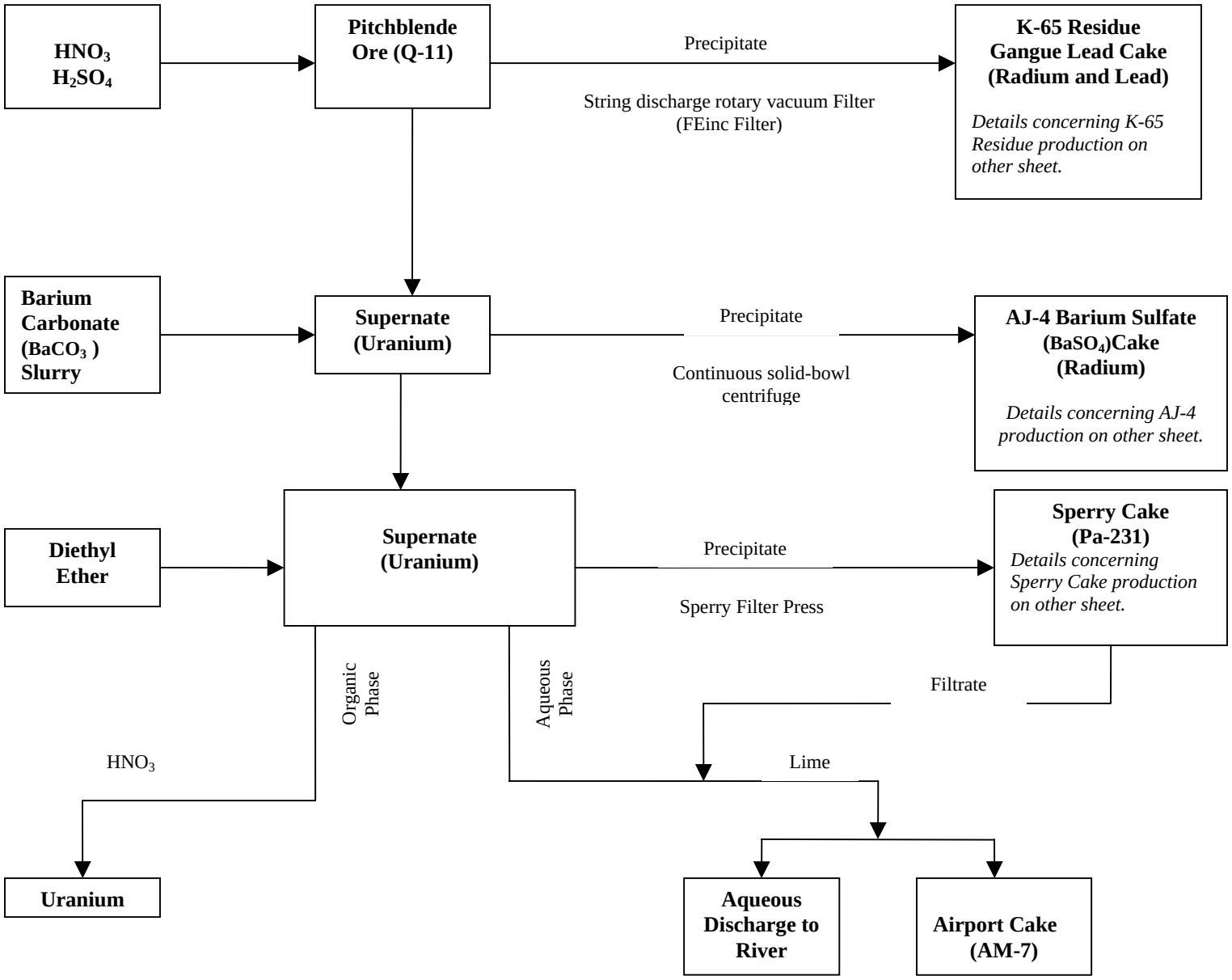
Summary of Intake Determinations and Worker Categories

A summary of intake assignment based on coworker data applies when there are no individual bioassay for a worker is shown in Table 9. Non-uranium intakes are considered for four worker categories: General worker, residue worker, radium reprocess worker and thorium or ionium reprocess worker.

Table 9. Summary of Internal Dose Reconstruction Assignments by Worker Category

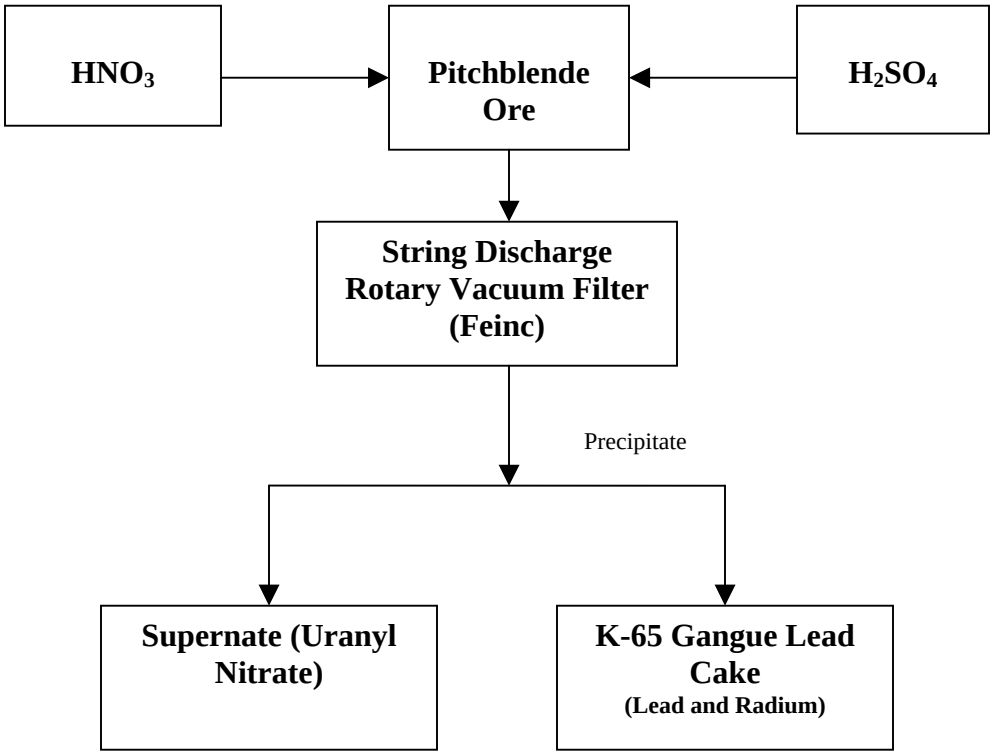
Category	Description	How to Identify	Period of Applicability	Dose reconstruction
General	Probably had no contact with process area.		1/1/1949-1/1/1958	**Coworker data:** assume lognormally distributed uranium intakes. **Other radionuclides:** In equilibrium.
General	Had contact with process area.		1/1/1949-1/1/1958	**Coworker data:** assume constant distribution and use 95th percentile uranium intakes. **Other radionuclides:** In equilibrium ratios.
Residue worker	Handled or worked primarily in areas where residues rather than uranium dominated the source term by activity.		1/1/1949-1/1/1958	**Coworker data:** assume constant distribution and use 95th percentile radium results. **Other radionuclides:** Radium ratios
Reprocess worker (Ra-226)	Reprocessed K-65 or AJ-4	Job title, external exposure results, radon breath samples taken	1/1/1949-1/1/1956	**Coworker data:** assume constant distribution and use 95th percentile radium results. **Other radionuclides:** Radium ratios
Reprocess worker (Th-230)	Reprocessed AM-7	Plant 7E, ME, Thorium bioassay, ionium bioassay	1/1/1955-1/1/1958	**Coworker data:** assume constant distribution and use 95th uranium percentile results. **Other radionuclides:** Thorium ratios

Attachment A: Mallinckrodt Plant 6 High-Grade Pitchblende Ore (Radium-bearing)

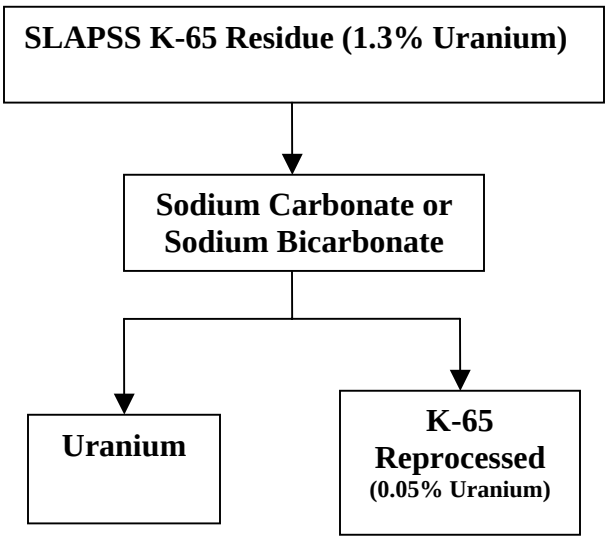


Attachment A: Mallinckrodt Plant 6 High-Grade Pitchblende Ore (Radium-bearing)

Formulation of K-65

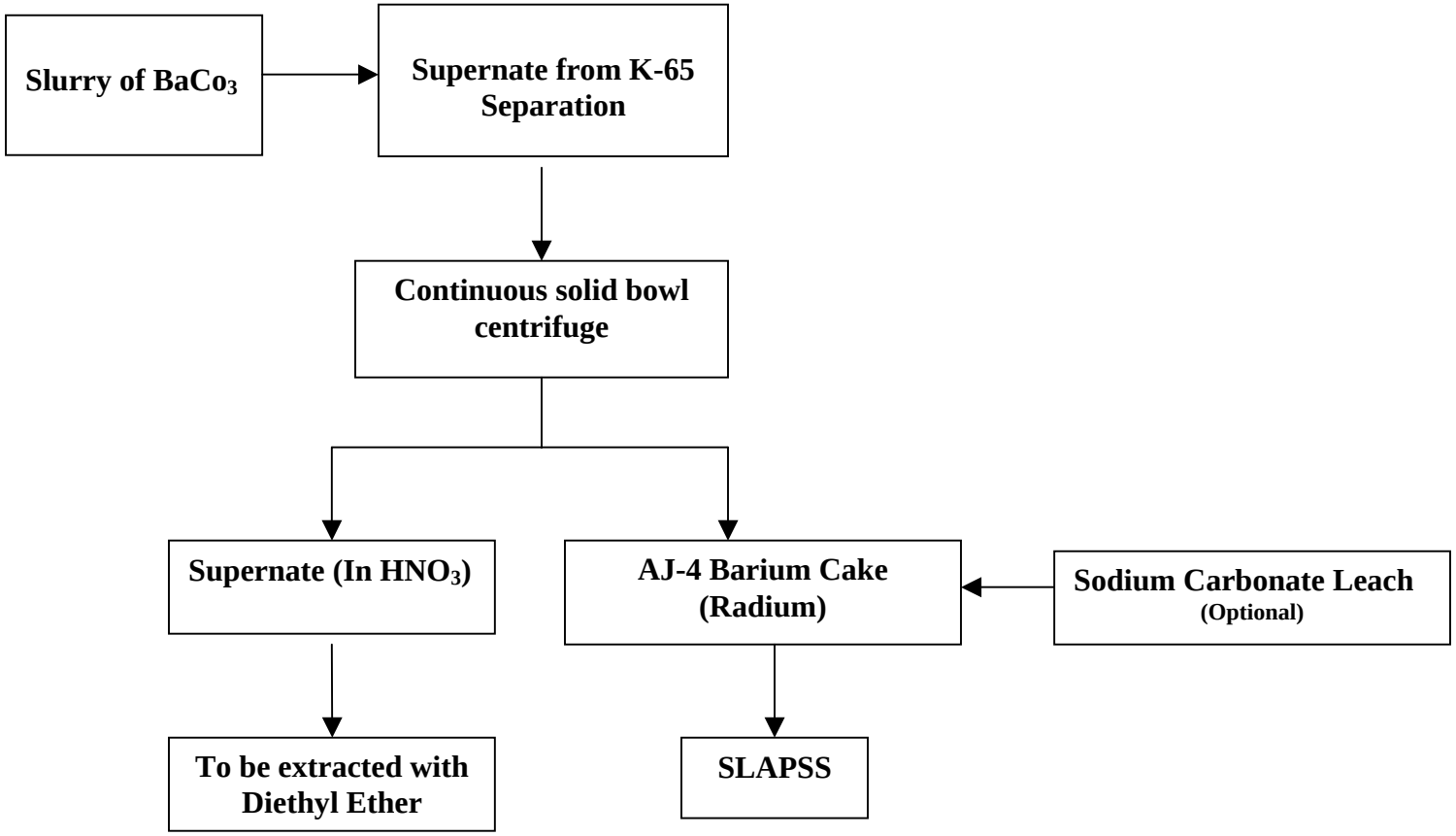


Reprocessing SLAPSS to Extract K-65
March 1948 thru September 1949



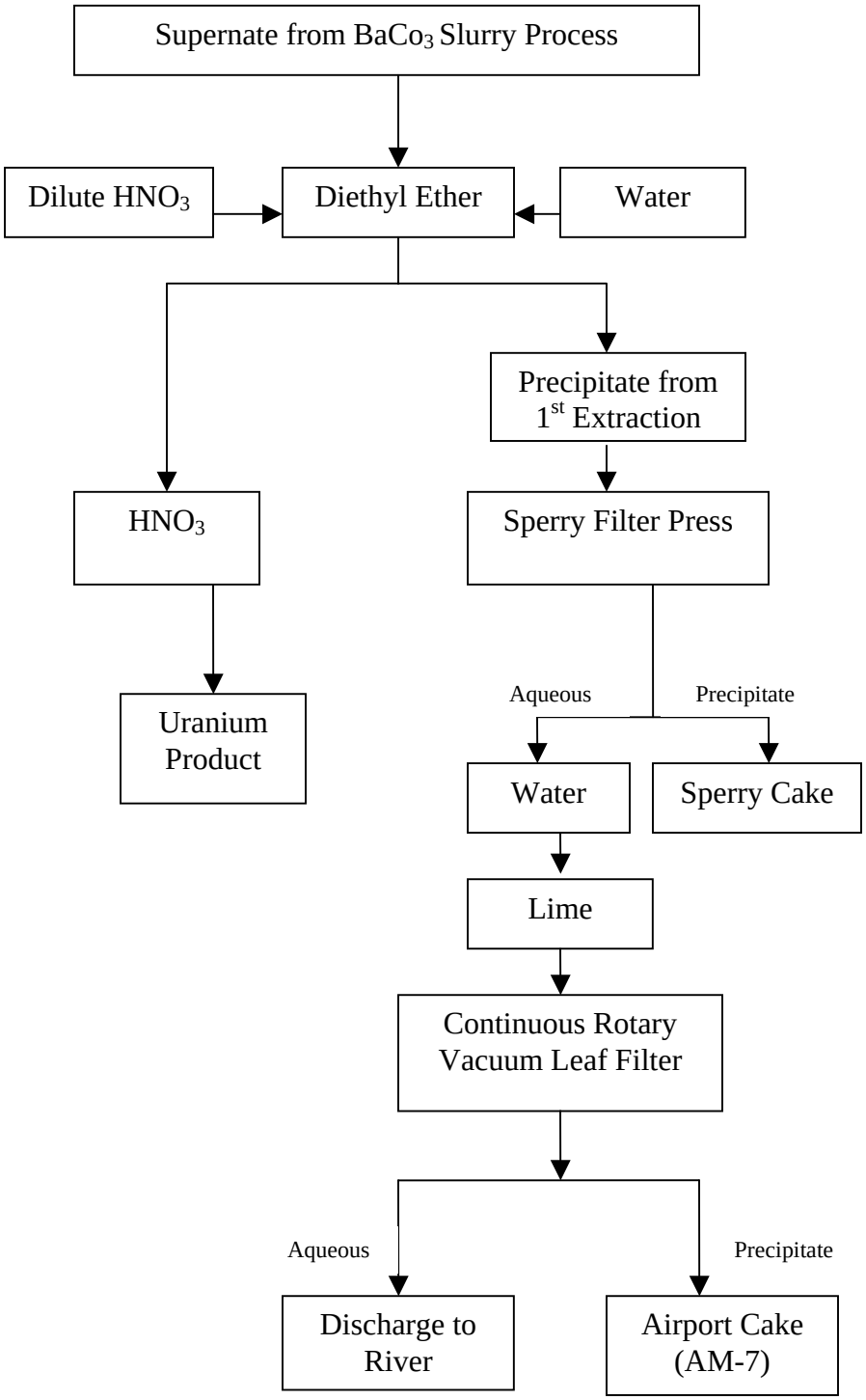
Attachment A: Mallinckrodt Plant 6 High-Grade Pitchblende Ore (Radium-bearing)

AJ-4 Residue



Attachment A: Mallinckrodt Plant 6 High-Grade Pitchblende Ore (Radium-bearing)

Sperry Cake & Airport Cake (AM-7)



Attachment B
Radionuclides in Ores and Residues

This attachment summarizes the information gathered to estimate uranium progeny activity fractions relative to uranium. Because fractions could not be calculated in cases, because only one type of data was reported, data were also presented as activity mass concentrations to show the variation in different materials.

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Activity Fraction	Ratio of activity of radionuclide or element to activity of uranium								
Uranium							1.00E+00	1.00E+00	1.00E+00
U-238							4.34E-01	4.13E-01	
U-234							5.41E-01	5.55E-01	
U-235							2.57E-02	3.22E-02	
Thorium									
Th-232							2.85E-01	1.36E+00	
Th-230							4.05E+01	4.10E+01	
Radium									
Ra-228									
Ra-226							2.64E+02	2.84E+02	5.27E+02
Rn-222									
Po-218									
Pb-214									
At-218									
Bi-214									
Po-214									
Tl-210									
Pb-210							1.11E+02	1.20E+02	
Bi-210									
Po-210							1.64E+02	1.67E+02	
Tl-206									
Th-231									
Pa-231									
Ac-227							4.03E+00	4.57E+00	
Th-227									
Fr-223									
Ra-223									
Rn-219									
Po-215									
Pb-211									
At-215									
Bi-211									
Po-211									
Tl-207									
Waste location	MCW	TOA	NLO	from LOSA	MCW and MSP	from MCW to LOOW	Silo 1	Silo 1 95th percentile	LOOW
Comments							Average		

This worksheet uses the results in the K-65 worksheet and calculates activity concentrations. Activity fractions are shown on beginning on row ..

K-65 RESIDUE										
Units	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g	pCi/g
Uranium						2.13E+03	1.48E+03	1.68E+03	3.41E+02	
U-238							6.42E+02	6.93E+02		
U-234							8.00E+02	9.32E+02		
U-235							3.80E+01	5.40E+01		
Thorium										
Th-232							4.22E+02	2.28E+03		
Th-230							6.00E+04	6.89E+04		
Radium	5.51E+05	8.26E+05	2.20E+05	7.07E+05	2.75E+05					
Ra-228										
Ra-226							3.91E+05	4.77E+05		1.80E+05
Rn-222										
Po-218										
Pb-214										
At-218										
Bi-214										
Po-214										
Tl-210										
Pb-210							1.65E+05	2.02E+05		
Bi-210										
Po-210							2.42E+05	2.81E+05		
Tl-206										
Th-231										
Pa-231										
Ac-227							5.96E+03	7.67E+03		
Th-227										
Fr-223										
Ra-223										
Rn-219										
Po-215										
Pb-211										
At-215										
Bi-211										
Po-211										

Tl-207										
	MCW	TOA	NLO	from LOSA	MCW and MSP	from MCW to LOOW	Silo 1	Silo 1	LOOW	LOOW
Waste location										
Comments							Average	95th percentile		

Attachment C
CER Radon Analysis Results

FOUR-LOCATION SUMMARY--Plant 6, C3, FEINC, K65

Site **Mallinckrodt** **Area Radon Concentration** **pCi/L**

Plant 6

Data for fitting				Data Statistics									
Measurement Year	**GM (50th) pCi/L**	**GM*GSD (84th) pCi/L**	**# of samples**	**# of employees**	***Slope(m)***	***Intercept (b)***	***GM***	***GSD***	***R^2***	***50th (normal)***	***84th normal***	**95th**
1948	33.562	166.247	386	N/A	1.600	3.513	33.562	4.95	0.980	38.000	167.800	466.6263
1949	10.903	72.206	507	N/A	1.891	2.389	10.9	6.62	0.968	10.000	63.040	244.4232
1950	18.914	67.585	380	N/A	1.273	2.940	18.9	3.57	0.963	22.000	57.720	153.6667
1951	17.022	57.765	721	N/A	1.222	2.835	17.0	3.39	0.980	19.000	55.000	127.0371
1952	11.149	47.898	708	N/A	1.458	2.411	11.1	4.30	0.980	12.000	47.000	122.6442
1953	9.410	38.719	613	N/A	1.415	2.242	9.41	4.11	0.961	9.000	40.000	96.42309
1954	8.406	47.464	714	N/A	1.731	2.129	8.41	5.65	0.976	7.000	51.000	144.9607
1955	9.738	45.597	621	N/A	1.544	2.276	9.74	4.68	0.975	10.000	49.000	123.4165
1956	4.995	22.387	215	N/A	1.500	1.608	4.99	4.48	0.908	<3	27.760	58.91212
1957	3.367	20.352	210	N/A	1.799	1.214	3.37	6.04	0.960	<3	25.000	64.94608
Average							12.7	4.781				167.177

Site **Mallinckrodt** **Area Radon Concentration** **pCi/L**

--C-3 Locations

Data for fitting				Data Statistics								
Measurement Year	**GM (50th) pCi/L**	**GM*GSD (84th) pCi/L**	**# of samples**	**# of employees**	***Slope(m)***	***Intercept (b)***	***GM***	***GSD***	***R^2***	***50th (normal)***	***84th normal***	**95th**
1948	58.680	295.204	21	N/A	1.616	4.072	58.680	5.031	0.974	59.000	375.400	836.9012
1949	23.509	231.582	98	N/A	2.288	3.157	23.509	9.851	0.973	25.000	195.400	1012.737
1950	25.286	79.689	64	N/A	1.148	3.230	25.286	3.152	0.948	27.000	109.200	167.088
1951	34.667	100.866	134	N/A	1.068	3.546	34.667	2.910	0.971	40.000	93.320	200.8716
1952	20.786	78.431	130	N/A	1.328	3.034	20.786	3.773	0.966	25.500	67.080	184.7019
1953	19.134	52.440	115	N/A	1.008	2.951	19.134	2.741	0.936	23.000	46.280	100.4797
1954	13.008	49.721	138	N/A	1.341	2.566	13.008	3.822	0.945	14.500	49.080	118.0689
1955	15.428	51.837	138	N/A	1.212	2.736	15.428	3.360	0.968	17.500	47.080	113.2717
1956	8.365	32.733	49	N/A	1.364	2.124	8.365	3.913	0.846	5.000	35.920	78.91881
1957	4.928	39.293	54	N/A	2.076	1.595	4.928	7.973	0.936	<3	39.040	149.9147
Average							22.379	4.652				

Site **Mallinckrodt** **Area Radon Concentration** **pCi/L**

-- FEINC Locations

Year **1945-57**

Data for fitting				Data Statistics								
Measurement Year	**GM (50th) pCi/L**	**GM*GSD (84th) pCi/L**	**# of samples**	**# of employees**	***Slope(m)***	***Intercept (b)***	***GM***	***GSD***	***R^2***	***50th (normal)***	***84th normal***	**95th**
1948	31.141	115.898	87	N/A	1.314	3.439	31.141	3.722	0.953	38.000	93.960	270.5576
1949	14.083	66.500	154	N/A	1.552	2.645	14.083	4.722	0.964	16.000	68.680	180.9808
1950	28.058	61.607	85	N/A	0.786	3.334	28.058	2.196	0.956	33.000	56.560	102.3392
1951	17.644	51.655	226	N/A	1.074	2.870	17.644	2.928	0.964	20.000	52.000	103.3015
1952	13.747	53.299	221	N/A	1.355	2.621	13.747	3.877	0.981	15.000	48.600	127.7272
1953	14.807	55.075	164	N/A	1.314	2.695	14.807	3.720	0.970	15.500	50.840	128.5317
1954	14.797	66.386	189	N/A	1.501	2.694	14.797	4.487	0.967	16.000	58.760	174.8402
1955	16.451	66.167	155	N/A	1.392	2.800	16.451	4.022	0.963	16.000	79.080	162.3676
1956	3.866	20.669	64	N/A	1.676	1.352	3.866	5.346	0.922	<3	23.920	60.93537

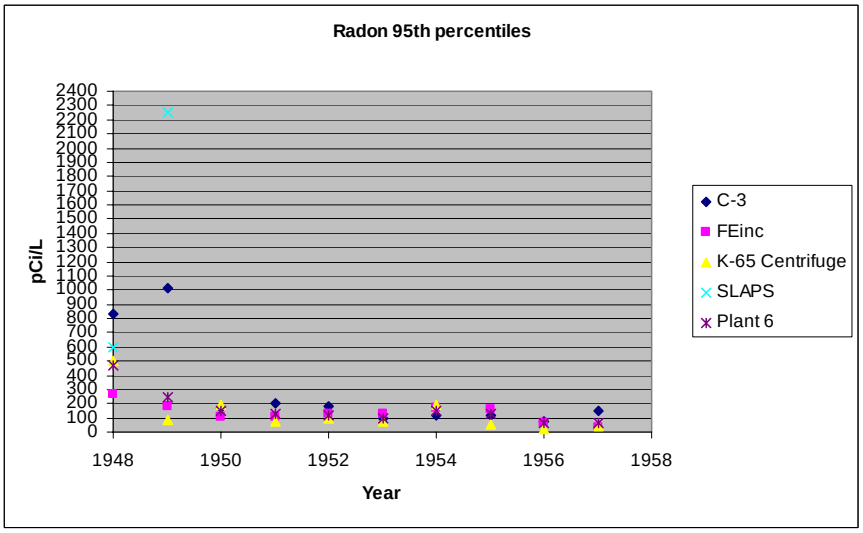
1957	3.569	13.873	67	N/A	1.358	1.272	3.569	3.887	0.937	<3	13.440	33.3014
Average							15.816	3.891				

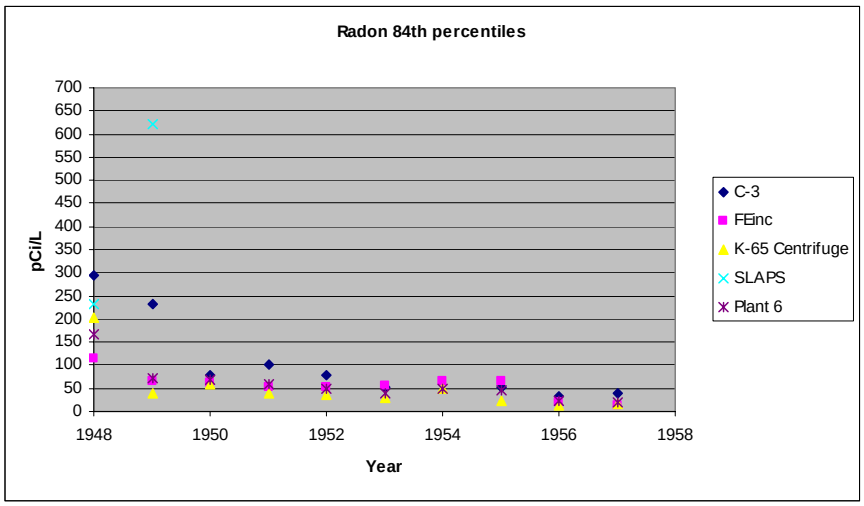
Site **Mallinckrodt** **Area Radon Concentration -- K-65 CENTRIFUGE Locations** **pCi/L**

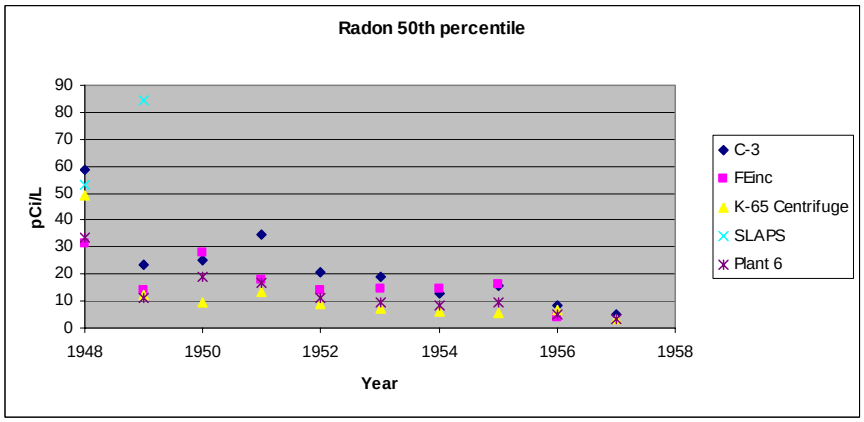
Data for fitting				Data Statistics								
Measurement Year	**GM (50th) pCi/L**	**GM*GSD (84th) pCi/L**	**# of samples**	**# of employees**	***Slope(m)***	***Intercept (b)***	***GM***	***GSD***	***R^2***	***50th (normal)***	***84th normal***	**95th**
1948	48.980	201.993	101	N/A	1.417	3.891	48.980	4.124	0.984	48.000	235.000	503.7549
1949	12.250	40.630	113	N/A	1.199	2.506	12.250	3.317	0.944	0.000	43.160	88.04267
1950	9.692	59.545	91	N/A	1.815	2.271	9.692	6.144	0.962	11.000	47.800	192.0388
1951	13.261	37.687	181	N/A	1.044	2.585	13.261	2.842	0.914	16.000	34.000	73.92243
1952	8.670	36.912	178	N/A	1.449	2.160	8.670	4.258	0.986	9.000	34.720	93.96659
1953	7.213	30.575	144	N/A	1.444	1.976	7.213	4.239	0.965	4.000	30.000	77.61571
1954	5.897	48.822	69	N/A	2.114	1.774	5.897	8.279	0.993	6.000	46.360	190.8558
1955	5.441	22.915	56	N/A	1.438	1.694	5.441	4.212	0.925	5.000	18.000	57.92863
1956	6.762	14.564	18	N/A	0.767	1.911	6.762	2.154	0.888	6.000	16.280	23.89066
1957	3.142	16.123	14	N/A	1.635	1.145	3.142	5.131	0.877	<3	16.200	46.29322
Average							12.131	4.470				

Site **Mallinckrodt** **Area Radon Concentration -- SLAPS** **pCi/L**

Data for fitting				Data Statistics								
Measurement Year	**GM (50th) pCi/L**	**GM*GSD (84th) pCi/L**	**# of samples**	**# of employees**	***Slope(m)***	***Intercept (b)***	***GM***	***GSD***	***R^2***	***50th (normal)***	***84th normal***	**95th**
1948	53.25155445	231.0564125	41	N/A	1.4676349	3.975026996	53.25155	4.338960898	0.9746	55	242.6	595.4315
1949	84.68397958	621.7637545	25	N/A	1.9936338	4.438926441	84.68398	7.342165042	0.8967	110	593.2	2249.479
Average							24.473	5.132				


Radon 95th percentiles
pCi/L
Year
C-3
FEinc
K-65 Centrifuge
SLAPS
Plant 6

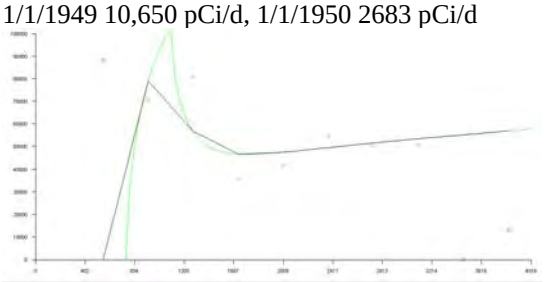

Radon 84th percentiles
pCi/L
Year
C-3
FEinc
K-65 Centrifuge
SLAPS
Plant 6


Radon 50th percentile
pCi/L
Year
C-3
FEinc
K-65 Centrifuge
SLAPS
Plant 6

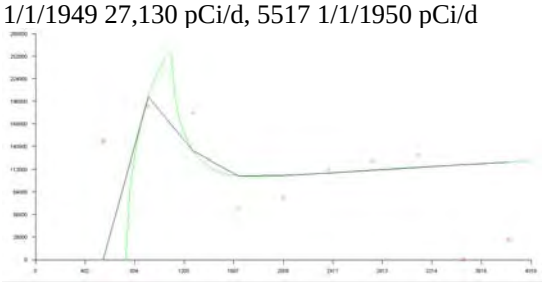
Attachment D
Mallinckrodt Preliminary Analysis of CER Coworker
Uranium Urinalyses and Radon Breath Analyses

CER data were analyzed to determine annual 50th and 84th radon breath analysis and urine analysis results in accordance with ORAUT-OTIB-0019. These results were assigned to July 7 of their respective years and used to determine inhalation intake rates for uranium and radium. Data (in red) prior to 1949 and the single 1956 and five 1957 radon breath analysis results from were excluded from the fit. For the purpose of this assessment, to compare radium:uranium ratios, two intake periods, 1/1/1949 to 1/1/1950 and 1/1/1950 to 1/1/1958 were chosen. Uranium intakes were assumed to be type M or S, radium intakes were type M. On the graphs, the x-axis is the number of day since 1/1/1947, and the y-axis is the bioassay result in units of pCi for radium and pCi/d for uranium. The black line shows the point to point fit, and the green line shows the bioassay projection. Radium to uranium ratios are noted at the bottom of this page.

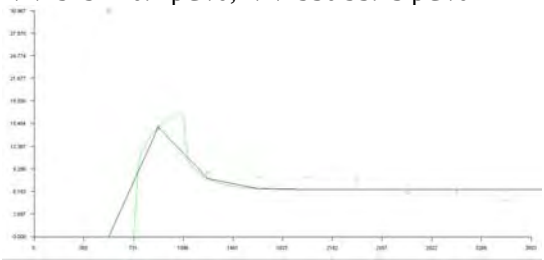
Radium whole body Chronic Type M 50th
1/1/1949 10,650 pCi/d, 1/1/1950 2683 pCi/d



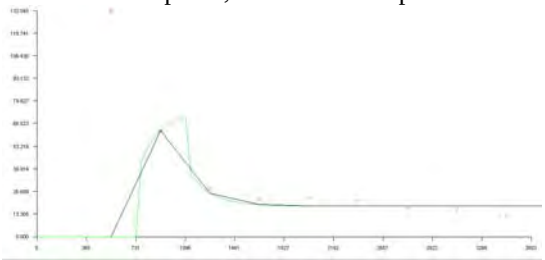
Radium whole body Chronic Type M 84th
1/1/1949 27,130 pCi/d, 5517 1/1/1950 pCi/d



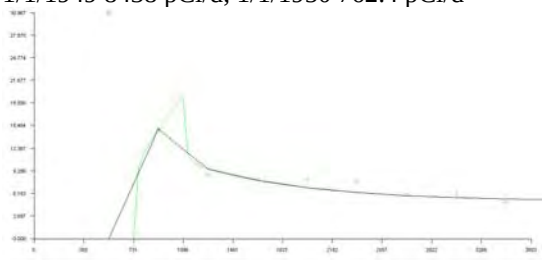
Uranium urinalysis Chronic Type M 50th
1/1/1949 270.1 pCi/d, 1/1/1950 95.75 pCi/d



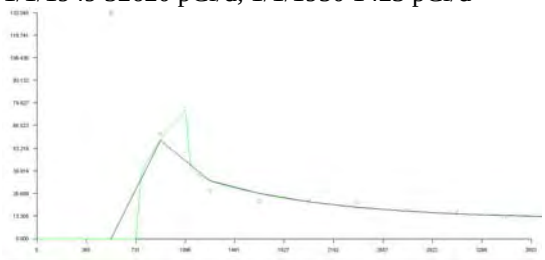
Uranium urinalysis Chronic Type M 84th
1/1/1949 1122 pCi/d,1/1/1950 267.0 pCi/d



Uranium urinalysis Chronic Type S 50th
1/1/1949 8438 pCi/d, 1/1/1950 762.4 pCi/d



Uranium urinalysis Chronic Type S 84th
1/1/1949 32620 pCi/d, 1/1/1950 1423 pCi/d



50th Ra:U Type **M** 1st intake: **39**, 2nd intake: **28**; Type **S** 1st intake: **1.3** 2nd intake: **3.5**
95th Ra:U Type **M** 1st intake: **18**, 2nd intake: **17**; Type **S** 1st intake: **0.6** 2nd intake: **4.1**

ATTACHMENT 2c

Office of Compensation Analysis and Support Technical Information Bulletin	Document Number: OCAS-TIB-0012 Effective Date: Revision No. Draft
Draft - Geometric Exposure Considerations for External Dose Reconstruction of Mallinckrodt Workers	**Page 1 of 10**
Approval:_____ Date: **J.W. Neton, Health Science Administrator**	**Supersedes:** **None**

RECORD OF ISSUE/REVISIONS

ISSUE AUTHORIZATION DATE	EFFECTIVE DATE	REV. NO.	DESCRIPTION
		0	First release of document

1.0 Purpose

The purpose of this technical information bulletin (TIB) is to provide guidance on the application of geometry-based dose correction factors to external dosimetry badge data for Mallinckrodt workers in particular job classifications. The factors are to be used for the periods of time where the individual's work history places him/her in a job for which a correction may be necessary. Specific job classifications for which these factors apply are discussed in section 3.0.

2.0 Geometric Exposure Considerations

Consideration must be given to the role that geometry plays in dose reconstruction for Mallinckrodt workers. Jobs at the facility ranged from those that required continuous hand or near-hand contact during a work day to those jobs where contact was only at a distance and of short duration. The distance and time factors could result in an underestimation of the reconstructed dosimeter and missed dose to organs located in the lower torso region of the body (e.g., stomach, liver, bladder, prostate, ovaries, testes, etc…) for high contact jobs. The degree of underestimation, or in some cases overestimation, is also dependent upon the shielding between the worker and source of radiation.

2.1 Exposure Geometries

Exposure geometry is a special consideration in dose reconstruction of energy employees who worked with uranium metal, powders and residues. An underestimation of the dose could occur if the energy employee wore his/her dosimeter on the lapel and not the center area of the chest or on the waist. This underestimation could result due to the difference in relative distance between the external radiation source, the organ of interest, and the dosimeter. Organs in the lower torso are

affected most. The dose to the lung is considered to be reasonably approximated by the lapel dosimeter, at least to within the dosimeter uncertainty, while the dose to the face and head could be lower than the dose measured by the dosimeter worn on the lapel.

2.2 ATTILA Radiation Transport Software

The Attilla Radiation Transport Software[1] was used to evaluate several near hand exposure scenarios. Attila is a multi-group deterministic radiation transport program that is capable of modeling complex geometries efficiently and accurately to solve large 3-D problems.

Attila can solve for neutron and photon transport equations and accounts for the same transport effects as Monte Carlo but is faster as no variance reduction is required. Attila directly solves the differential form of the Boltzmann transport equation. For charged particles, the Boltzmann-Fokker-Planck transport equation is solved. It discretizes in space, angle and energy to solve for flux as a function of angle, energy and particle type, at every location in the computational domain.

3.0 Methodology

3.1 ATTILA Model Development

A review of table 18, Job titles and classifications with geometry factor, from the *"Basis for Development of an Exposure Matrix for the Mallinckrodt Chemical Company St. Louis Downtown Site and the St Louis Airport Site, St. Louis, Missouri, Period of Operation: 1942–1958"*[2] was made to determine the type of scenario/s for which ATTILA could be used to develop general correction factors for the lower torso of an energy worker. The lower torso is defined as those organs below the lungs in figure 1. These organs include but are not limited to the stomach, liver, kidney, gall bladder, ureter, pancreas, small intestine, large intestine, rectum, ovaries, uterus, urinary bladder, and prostate.

Using Attila, three near hand job exposure scenarios were evaluated that exemplified the range of body/source orientations for which badge correction factors could be determined. These were pitchblende/cleanup activities, ingot/machining activities and de-nitration pot activities. For development of the geometry correction factor, only the photon contribution was analyzed, since only the relative values between body locations and the photon components were of interest. For the model development, the human figure was divided into head, upper torso, badge, lower torso and hands. Dose factors were developed based on flux/dose at the badge relative to the other body locations.

It should be re-iterated that the purpose of this ATTILA model was to demonstrate that, due to geometry, a relative flux factor exists between the badge and different points of the

body for each of 3 source term scenarios. Because of this it was necessary to create a realistic model of a human body or provide the dose to human tissue at those points.

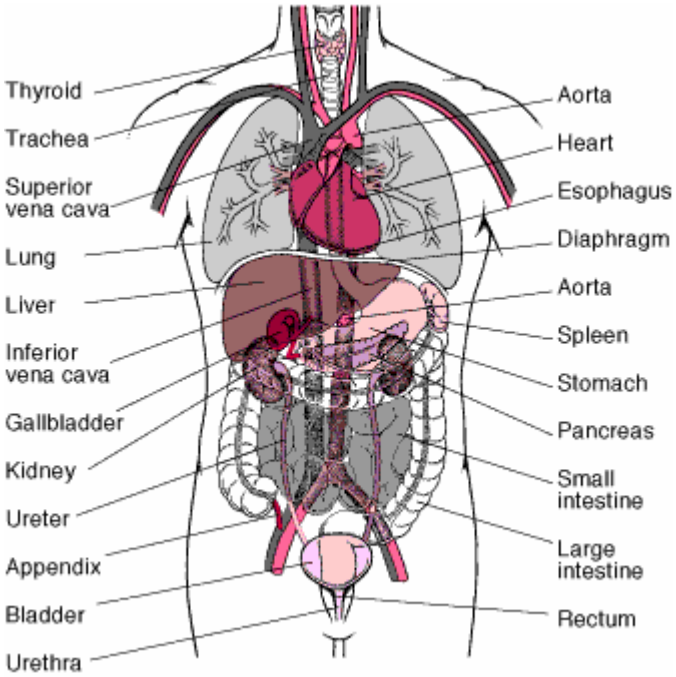


Figure 1: Diagram of Human Torso [3]

The results for pitchblende/cleanup activities, ingot/machining activities and de-nitration pot activities are provided for in figures 2, 3 and 4 respectively. Each geometric correction factor is relative to the dosimeter/badge worn on the upper chest. The factors, as calculated, are provided in Table 1. Based on the ingot and pitchblende exposure scenario results, a factor of 2.1 should be applied to the lower torso. The detail of each scenario follows in the next three sections.

Table 1 Geometry Correction Factors for Specific Jobs

Body Location	Attila Exposure Scenario		
	Ingot / Machining	Pitchblende Clean-up	De-nitration Pot
Head	0.4	0.41	0.57
Badge (Upper Chest)	1.00	1.00	1.00
Lower Torso	2.13	2.11	0.03
Hands	3.65	1.20	1.67

3.1.1 Pitchblende/Cleanup scenario

Cleanup activities at Mallinckrodt occurred on a frequent basis in several areas. The purpose of this scenario was to simulate the positional relationship between the worker and a pitchblende/cleanup spill. The material was spread in a circular pattern on the floor. This was seldom a hands on activity.

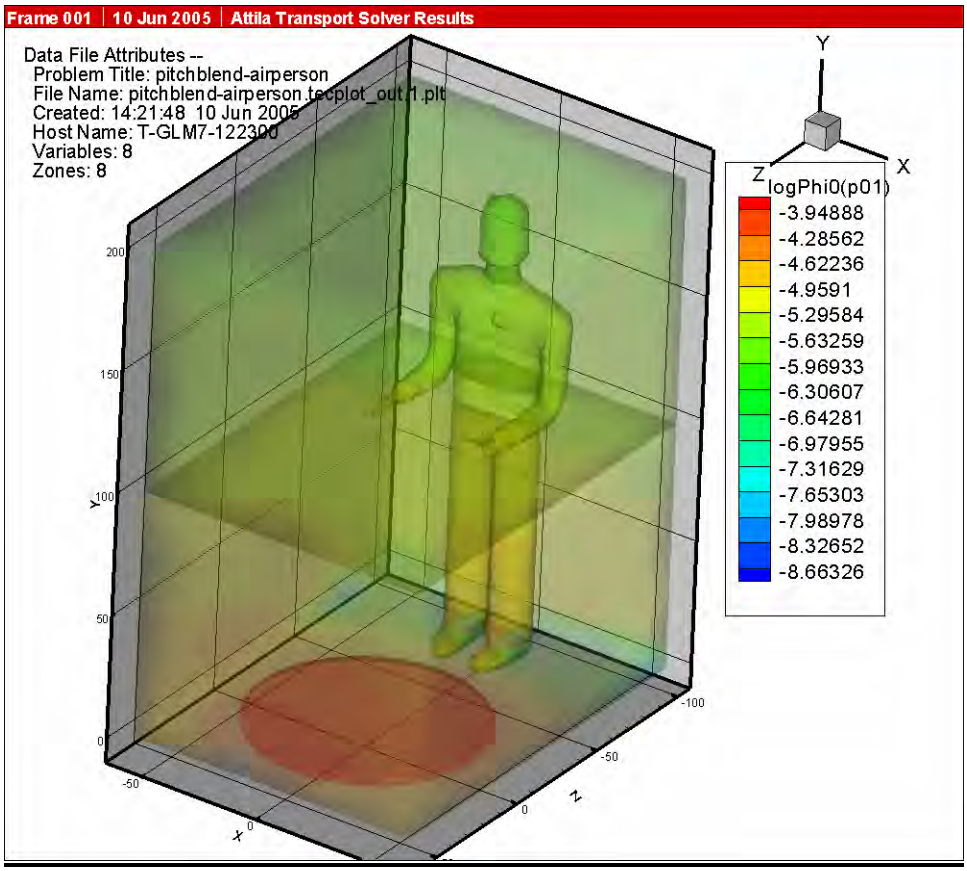


Figure 2: Pitchblende/Cleanup

3.1.2 Uranium Ingot Scenario

In some situations workers would come in close contact with uranium metal during machining or sawing. This scenario demonstrated the positional relationship between an ingot at waist level in close association to the worker's body.

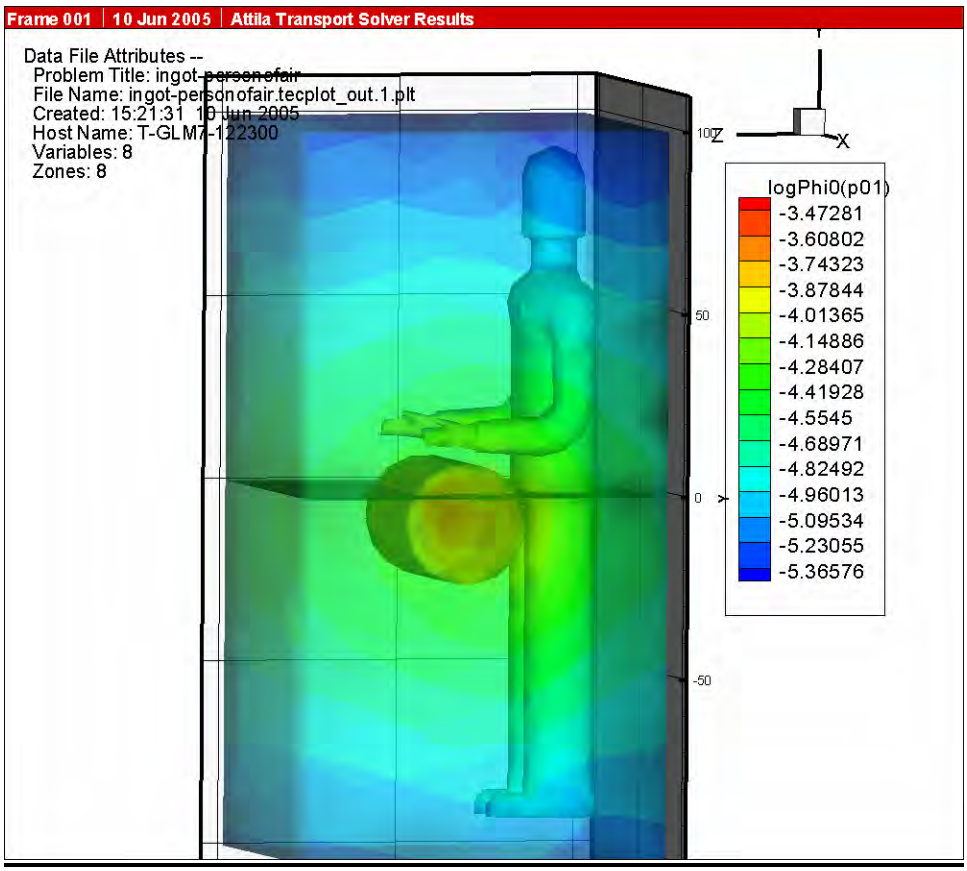


Figure 3: Ingot/Machining

3.1.3 De-nitration Pot Scenario

Some activities would place a worker near a uranium de-nitration pot or drum. The individual would have the lower torso shielded by metal or some other material with the majority of photon flux being delivered to the hands, upper torso and head.

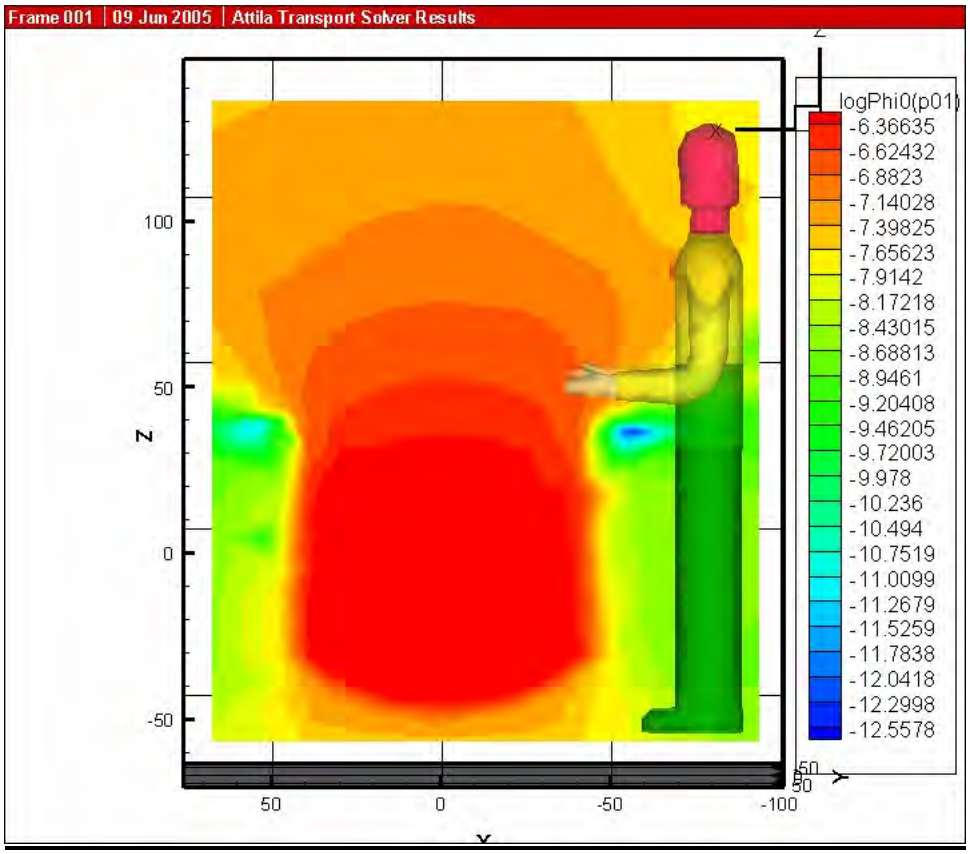


Figure 4: Cut away exposure view of De-nitration Pot

3.2 Application of Geometry Correction Factor

It should be noted that there is a significant reduction in dose to the head for all scenarios. In addition, due to shielding provided by the De-nitration Pot, there was also a significant reduction in dose to the lower torso. Assigning the badge dose to these body locations is claimant favorable. However, as illustrated in Table 1, there are some organs in certain exposure scenarios that could be underestimated.

In a review of selected work histories and dust study estimates of internal exposure, it is clear that most operators conducted various different tasks over their career. As a result the application of a single factor over their entire work history is also considered claimant favorable as some tasks would involve near-hand exposures while others would not.

As noted there are numerous different work stations (tasks) for which a geometry correction factor should be applied to accurately estimate the dose to organs in the lower torso from a film badge worn on the lapel. A review of the claimant work history information in conjunction with the job descriptions (Tasks) in Table 18 of the Mallinckrodt technical basis document indicates that many Production and Chemical Operators worked at various stations some of which require a geometry correction and some do not. Time estimates used in the dust studies could be used to estimate the fraction of time a worker would be in one geometry versus another, however, the external dose cannot be reasonably partitioned based time since the external dose rate changes from location to location. In addition, trades and crafts workers conducted work in and around radioactive materials throughout the facility. As with chemical and production operators, time motion dust study data could be used to estimate the general time they were in a specific area, however, the external dose rates changed such that it is not practical to partition the dose measured on the lapel dosimeter accordingly.

As a result, the additional claimant favorable assumption is made that the geometry correction factor should be applied to all measured and missed dose for the general occupations listed in Table 2.

Table 2 Job Types for which the geometric correction factor should be applied.

General Category	Specific Job Titles
Operators and Material Handlers	Chemical Operator, Production Operator, Material Handler
Trades and Crafts	Pipefitters, Carpenters, Welders, Sheet Metal Workers, Electricians, Foremen, etc…

It should be noted, that occupations other than those listed above should be evaluated on a case by case basis to determine if a correction factor should be applied. A review of the current active Mallinckrodt claims indicates that the application of a geometric correction factor would not be appropriate for a significant fraction (≈43%) of the current Mallinckrodt claimant population. Figure 5 provides the general occupation distribution of the current claims received for Mallinckrodt workers.

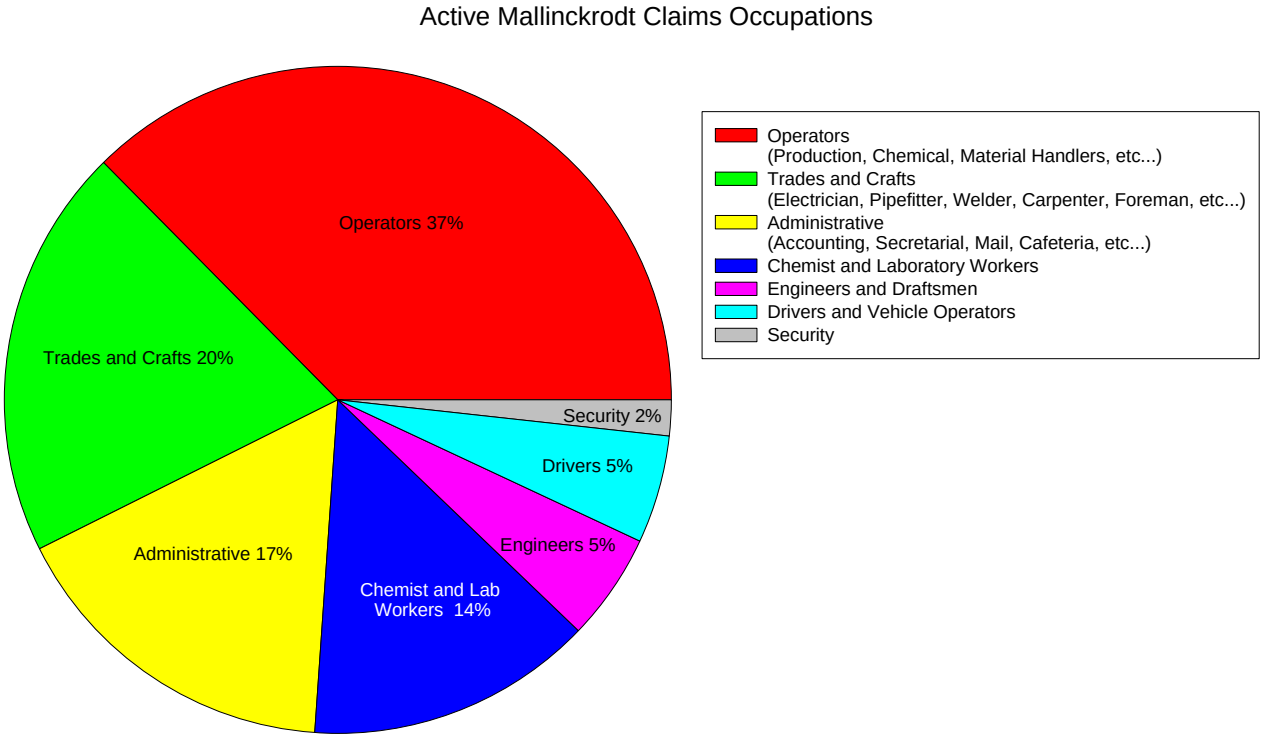


Figure 5: Breakdown of Active Mallinckrodt Claims

4.0 Comparison of Attila Model Estimate to Dosimetry Data

Limited ring badge dosimetry data was used to validate the Attila modeled geometry ratio. In 1949, for a 5 week period, selected Mallinckrodt workers were issued ring dosimeters in addition to their whole body dosimeter.[4] At the current time, a full comparison of this data is not possible since the detailed dosimetry for most of the study participants has not yet been obtained from DOE. However, for three individuals with a non-zero dose, a comparison of the ring dosimeter to whole dosimeter was possible. Table 3 provides the data comparison and the calculated geometric correction ratio. Figure 6 provides this limited data on a probability plot to indicate the range of values with the three Attila point estimates.

Table 3 Comparison of Ring Dosimeter to Whole Body Dosimeter

Study Subject	Radiation Type	Ring Dose	Whole Body Dose	Geometric Correction Ratio
1	Beta	195	85	2.29
2	Beta	95	30	3.17
1	Gamma	440	150	2.93
3	Gamma	560	275[(a)]	2.04
4	Gamma	460	275[(a)]	1.67
	Maximum Attila Estimated Hand Ratio			3.65

(a) Estimated Midpoint – the data indicate whole body dosimeter was between 251 mR and 300 mR.

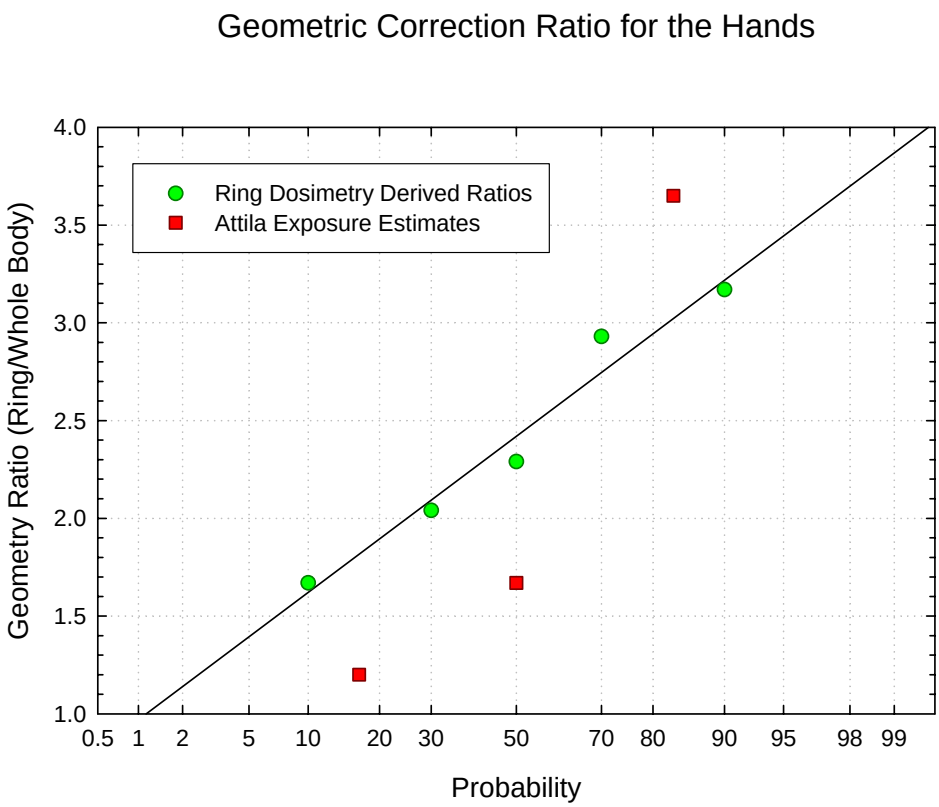


Figure 6: Probability Plot of Attila modeled point estimates and limited ring dosimetry comparison data.

The limited dosimetry data clearly indicate that the Attila model reasonably predicts the geometric correction factor. As a result, the dose reconstruction to organs in the lower torso should multiply the measured and the missed dose by the largest lower torso geometry correction factor ensure claimant favorability. This factor of 2.1 was effectively equivalent for both the pitchblende clean-up and ingot machining models.

4.1 Reasonable Claimant Favorable Assumptions

Listed below are several claimant favorable assumptions used in the development of this methodology that could result in an overestimate of the actual dose.

- Claimant favorable model design
- Assignment of badge results where actual reduction due to geometry occurs
- Assignment of correction factor for all tasks within the job classification

Obtaining data to precisely evaluate each of these parameters requires extensive time and research on an individual basis and in some instances it is not known if the information can be obtained (i.e. facility has undergone D&D or other significant modification over time).

5.0 Summary

This Technical Information Bulletin provides guidance for dose reconstruction to organs located in the lower torso. For dose reconstruction to organs in the lower torso the measured and missed dose should be multiplied by the factor 2.1 for all members of the general job categories of Table 2.

6.0 References

1. McGhee and Wareing (2004) Attila™ Version 4: User's Manual, Radion Technologies Los Alamos, NM

2. ORAUT (2005), *Basis for Development of an Exposure Matrix for the Mallinckrodt Chemical Company St. Louis Downtown Site and the St Louis Airport Site, St. Louis, Missouri, Period of Operation: 1942–1958*. ORAUT-TKBS-0005, Rev 1.

3. Merck (2004), The Merck Manual – Second Home edition, illustration obtained at the following website. http://www.merck.com/mrkshared/mmanual_home2/fg/fg001_1.jsp

4. MCW Memorandum (1949), Memo from A Piccot to K. Caplan providing results of the analysis of the film rings worn beg9ining April 25, 1949 through May 30, 1949.

ATTACHMENT 3a

Dose Reconstruction Example
Case 1
Residue worker, uranium urinalysis and radon breath

Selection Criteria

radon breath data

Cancer Description

prostate (185), colon (153.9)

Employment (Destrehan St.)

Start: 1951
End: 1958

Work History

NOCTS : 'chemical operator'
Dosimetry Data: 'pot room, MFG 6, ORE room, Cleanup, unload, GLC, MGX, RAF, Dingot KB2, slag, warehouse, fork truck operator, furn rm, saw, NA house

Data Summary:

External			Uranium					
Start (yr)	End (yr)	Deep Total (rem)	Start (yr)	End (yr)	Count	min (mg/l)	max (mg/l)	Avg (mg/l)
51	58	10	51	57	16	0.000	0.073	0.027

Thorium					
Start	End	Count	min (dpm/l)	max (dpm/l)	Avg (dpm/l)

Radon					
Start	End	Count	min % tol	max % tol	Avg % tol
51	54	7	10	30	18

Data

Uranium Urine

Sample Date	Reported Result (mg/l)	Normalized Result (pCi/d)
07/28/51	0.017	16.2
08/28/51	0.041	39.1
02/21/52	0.043	41.1
05/15/52	0	<9.5
08/07/52	0.023	22
04/15/53	0.046	43.9
07/08/53	0.035	33.4
09/30/53	0.045	43
01/28/54	0.032	30.6
07/14/54	0.073	69.7
08/24/54	0.034	32.5
10/06/54	0.027	25.8
09/21/55	0.001	<9.5

Radon Breath

Samp Date		% tolerance*	% tolerance*	Ra WB (pCi)
02/05/1951	10	30	20	50400
06/11/1951	10	20	15	37800
08/20/1951	30		30	75600
11/05/1951	20		20	50400
02/18/1952	10		10	25200
06/02/1952	10		10	25200
03/23/1953	20		20	50400
05/25/1953	20		20	50400
10/26/1953	10		10	25200
08/16/1954	30		30	75600

*tolerance = 1 pCi/L

Approach:

Uranium intake from uranium bioassay data. Radium intake from radon breath monitoring data (using OTIB -025). Progeny intake normalized to radium intake, based on ratios:

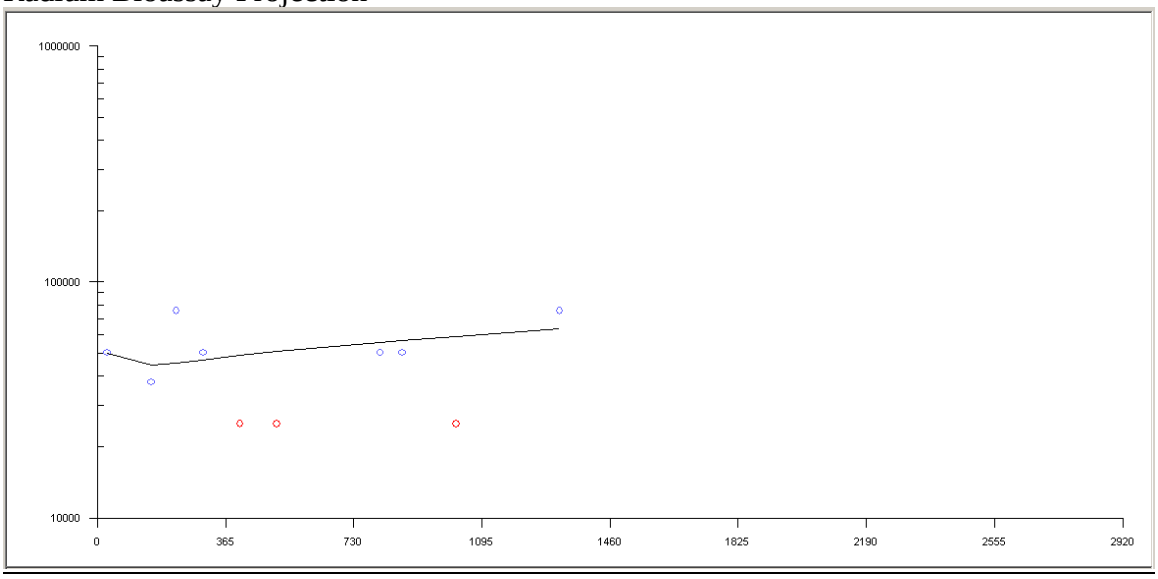
Ratios:

	Th-230	Po-210	Pa-231	Ac-227
X/Ra	0.125	0.500	0.013	0.013

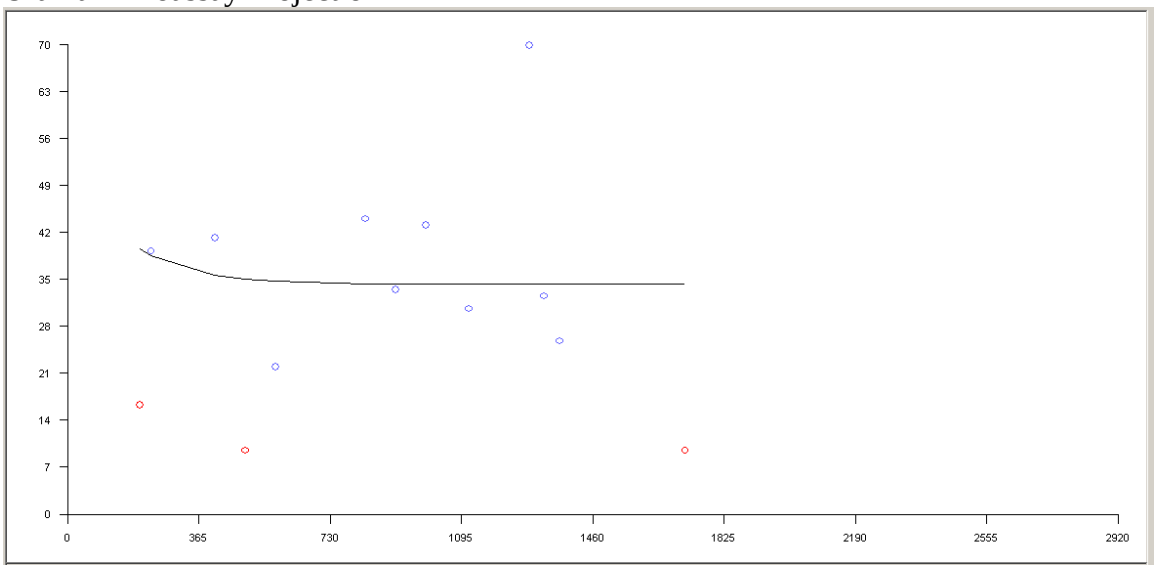
Intake:

	Intake #	U	Ra-226	Th-230	Po-210	Pa-231	Ac-227
1/9/51 – 6/6/1958 (chronic) (pCi/d)	1	5.10E+02	4.36E+03	5.45E+02	2.18E+03	5.45E+01	5.45E+01
1/9/1951 (acute) (pCi)	2	1.82E+05	7.05E+05	8.81E+04	3.53E+05	8.81E+03	8.81E+03

Radium Bioassay Projection



Uranium Bioassay Projection



Dose:

Nuclide	Intake #	Dose colon, rem (through diagnosis)	Dose HNM, rem (through diagnosis)
U-234	1	4.04E-01	4.30E-01
Ra-226	1	3.22E+00	1.36E+00
Pa-231	1	1.11E-02	4.69E-03
Ac-227	1	3.05E-02	1.14E-01
Po-210	1	6.14E+00	6.67E+00
Th-230	1	1.50E+00	8.07E+00
U-234	2	5.41E-02	5.69E-02
Ra-226	2	1.79E-01	7.56E-02
Th-230	2	8.31E-02	4.92E-01
Po-210	2	3.83E-01	3.70E-01
Pa-231	2	6.37E-04	2.94E-04
Ac-227	2	1.80E-03	6.54E-03
Total		**1.20E+01**	**1.77E+01**

POC:

ICD-9	Organ	POC
153.9	colon	26.48%
185	HNM	20.60%
Total		**41.63%**

Alternative Organ Doses (using data of diagnosis)

Nuclide	Intake #	Dose Liver, rem	Dose Bone Surface, rem	Dose Kidney, rem
U-234	1	1.84E+00	1.30E+01	5.44E+00
Ra-226	1	5.75E+00	3.56E+02	1.32E+00
Pa-231	1	1.64E-01	1.26E+03	2.57E-01
Ac-227	1	2.12E+03	8.76E+03	4.56E-02
Po-210	1	2.76E+01	6.67E+00	5.34E+01
Th-230	1	8.24E+01	5.91E+03	1.02E+02
U-234	2	2.33E-01	1.64E+00	6.67E-01
Ra-226	2	3.19E-01	2.05E+01	7.37E-02
Th-230	2	5.02E+00	3.66E+02	6.12E+00
Po-210	2	1.53E+00	3.70E-01	2.96E+00
Pa-231	2	9.11E-03	8.36E+01	1.42E-02
Ac-227	2	1.24E+02	5.19E+02	2.68E-03
Total		**2.37E+03**	**1.73E+04**	**1.73E+02**

ATTACHMENT 3b

Dose Reconstruction Example
Case 2
Residue worker, uranium urinalysis only
(no radon breath or thorium bioassay)

Selection Criteria

Hypothetical residue worker without radon breath monitoring

Cancer Description:

Prostate (185), colon (153.9)

Employment (Destrehan St.)

Start: 1951
End: 1958

Work History

NOCTS : 'chemical operator'
Dosimetry Data: 'pot room, MFG 6, ORE room, Cleanup, unload, GLC, MGX, RAF, Dingot KB2, slag, warehouse, fork truck operator, furn rm, saw, NA house

Data Summary

External			Uranium					
Start (yr)	End (yr)	Deep Total (rem)	Start (yr)	End (yr)	Count	min (mg/l)	max (mg/l)	Avg (mg/l)
51	58	10	51	57	16	0.000	0.073	0.027

Thorium					
Start	End	Count	Min (dpm/l)	max (dpm/l)	Avg (dpm/l)
No Data					

Radon					
Start	End	Count	Min % tol	max % tol	Avg % tol
No Data					

Data

Uranium Urine

Sample Date	Reported Result (mg/l)	Normalized Result (pCi/d)
07/28/51	0.017	16.2
08/28/51	0.041	39.1
02/21/52	0.043	41.1
05/15/52	0	<9.5
08/07/52	0.023	22
04/15/53	0.046	43.9
07/08/53	0.035	33.4
09/30/53	0.045	43
01/28/54	0.032	30.6
07/14/54	0.073	69.7
08/24/54	0.034	32.5
10/06/54	0.027	25.8
09/21/55	0.001	<9.5

Approach:

Uranium intake from uranium bioassay data. Radium intake from 95th percentile of CER radon breath analysis data. Progeny intake normalized to radium intake, based on ratios:

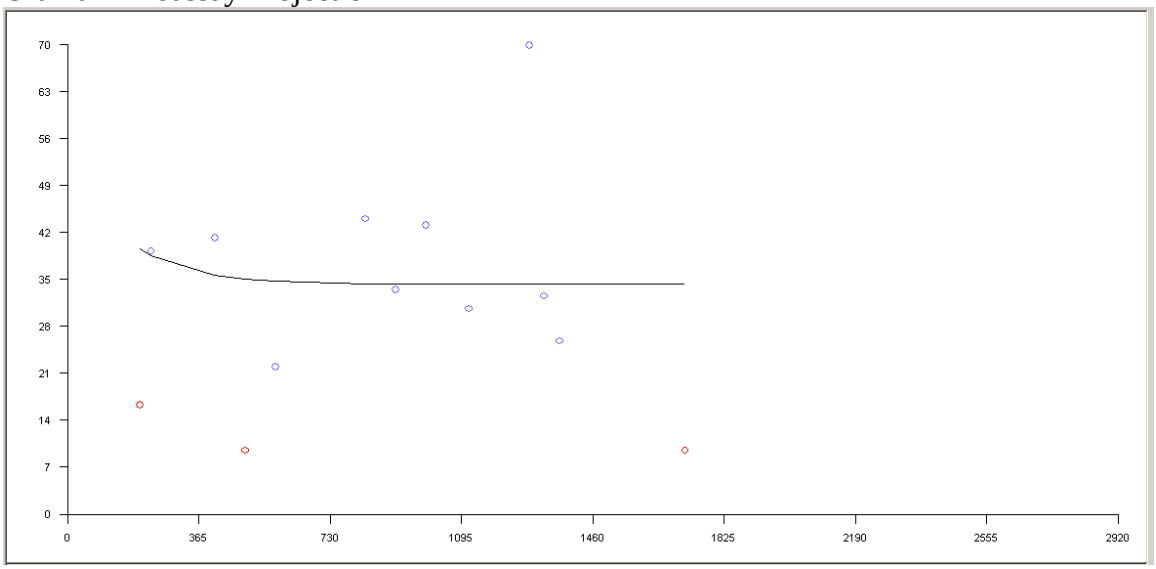
Ratios:

	Th-230	Po-210	Pa-231	Ac-227
X/Ra	0.125	0.500	0.013	0.013

Intake:

	Intake #	U	Ra-226	Th-230	Po-210	Pa-231	Ac-227
1/9/51 – 6/6/1958 (chronic) (pCi/d)	1	5.10E+02	8.78E+03	1.10E+03	4.39E+03	4.39E+03	1.10E+02
1/9/1951 (acute) (pCi)	2	1.82E+05					

Uranium Bioassay Projection



Dose:

Nuclide	Intake #	Dose colon, rem (through diagnosis)	Dose HNM, rem (through diagnosis)
U-234	1	4.04E-01	4.30E-01
Ra-226	1	6.49E+00	2.74E+00
Pa-231	1	2.24E-02	9.46E-03
Ac-227	1	6.15E-02	2.31E-01
Po-210	1	3.02E+00	1.34E+01
Th-230	1	1.24E+01	1.63E+01
U-234	2	5.41E-02	5.69E-02
Total		**2.24E+01**	**3.32E+01**

POC:

ICD-9	Organ	POC
153.9	colon	40.15%
185	HNM	32.71%

Alternative Organ Doses (using diagnosis date):

Nuclide	Intake #	Dose Liver, rem	Dose Bone Surface, rem	Dose Kidney, rem
U-234	1	1.84E+00	1.30E+01	5.44E+00
Ra-226	1	1.16E+01	7.16E+02	2.66E+00
Pa-231	1	3.32E-01	2.55E+03	5.18E-01
Ac-227	1	4.28E+03	1.77E+04	9.21E-02
Po-210	1	5.56E+01	1.34E+01	1.08E+02
Th-230	1	1.66E+02	1.19E+04	2.07E+02
U-234	2	2.33E-01	1.64E+00	6.67E-01
Total		**4.51E+03**	**3.29E+04**	**3.24E+02**

ATTACHMENT 3c

Dose Reconstruction Example
Case 3
Thorium worker, uranium and ionium (thorium) urinalysis and radon breath

Selection Criteria

ionium bioassay

Cancer Description

pancreas (157.0)

Employment (Destrehan St.)

Start: 1949
End: 1958+ (break in employment 1953)

Work History

NOCTS : 'laboratory technician'
Dosimetry Data: 'LAB', 'CHEM', 'chemist M.E. Process (sp) plant 7E', 'TECH M.E. Process(sp) plant 7E'

Data Summary

External			Uranium					
Start (yr)	End (yr)	Deep Total (rem)	Start (yr)	End (yr)	Count	min (mg/l)	max (mg/l)	Avg (mg/l)
49	59	8.2	50	59	15	0.000	0.035	0.009

Thorium					
Start	End	Count	min (dpm/l)	max (dpm/l)	Avg (dpm/l)
55	55	1	1.4	1.4	1.4

Radon					
Start	End	Count	min % tol	max % tol	Avg % tol
50	54	7	10	60	27

Data

Uranium Urine

Sample Date	Reported Result (mg/l)	Normalized Result (pCi/d)
06/22/50	0	<9.5
10/11/50	0.016	15.3
12/19/50	0.008	<9.5
03/21/51	0.003	<9.5
09/18/51	0.035	33.4
03/04/52	0.024	22.9
10/06/54	0.005	<9.5
03/10/55	0.02	19.1
04/07/55	0.004	<9.5
06/15/55	0.004	<9.5
11/29/55	0.004	<9.5
09/12/56	0.004	<9.5
08/13/57	0.005	<9.5
03/10/58	0.002	<9.5
03/30/59	0.002	<9.5

Radon Breath

Samp Date	% tolerance*	% tolerance*	% tolerance*	Ra WB (pCi)
01/03/1950	10		10	25200
3/12/1950	20	20	20	50400
7/24/1950	40		40	100800
6/25/1951	20		20	50400
10/18/1954	60		60	151200

*tolerance = 1 pCi/L

Thorium Urine

Sample Date	Reported Result (dpm/l)	Normalized Result (pCi/d)
4/8/1955	1.4 +/- 0.7	0.88

Approach:

Uranium intake from uranium bioassay data. Radium intake from radon breath monitoring data (using OTIB -025). Polonium intake normalized to radium intake, based on ratios. Thorium intake from thorium bioassay. Protactinium and actinium intake normalized to thorium intake, based on ratios:

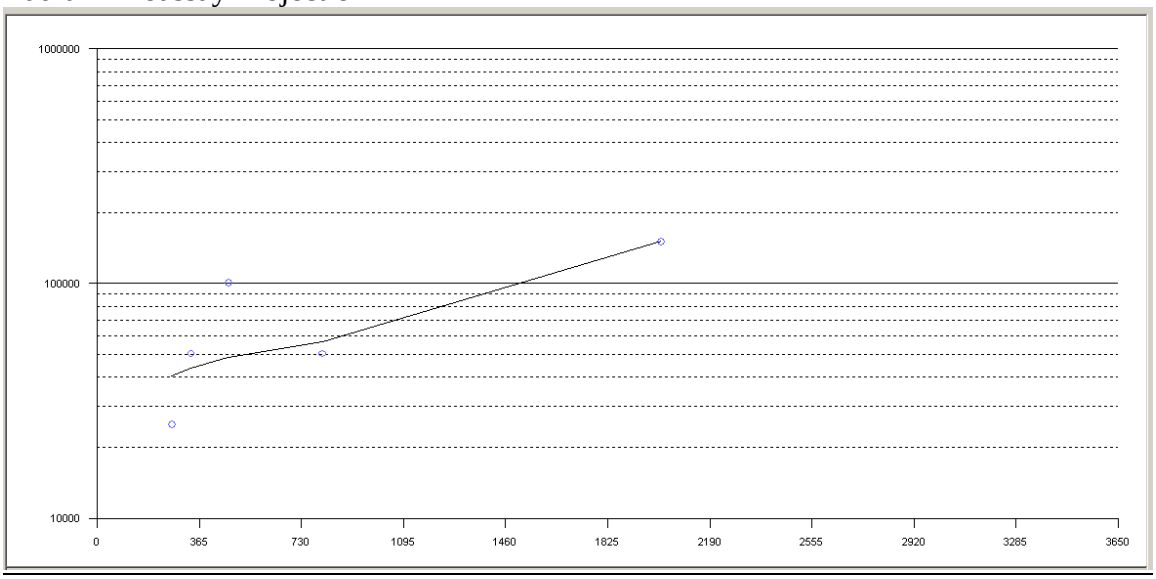
Ratios:

	Th-230	**Po-210**	**Pa-231**	**Ac-227**
X/Ra	0.125	0.500	0.013	0.013
X/Th		0.000	0.100	0.100

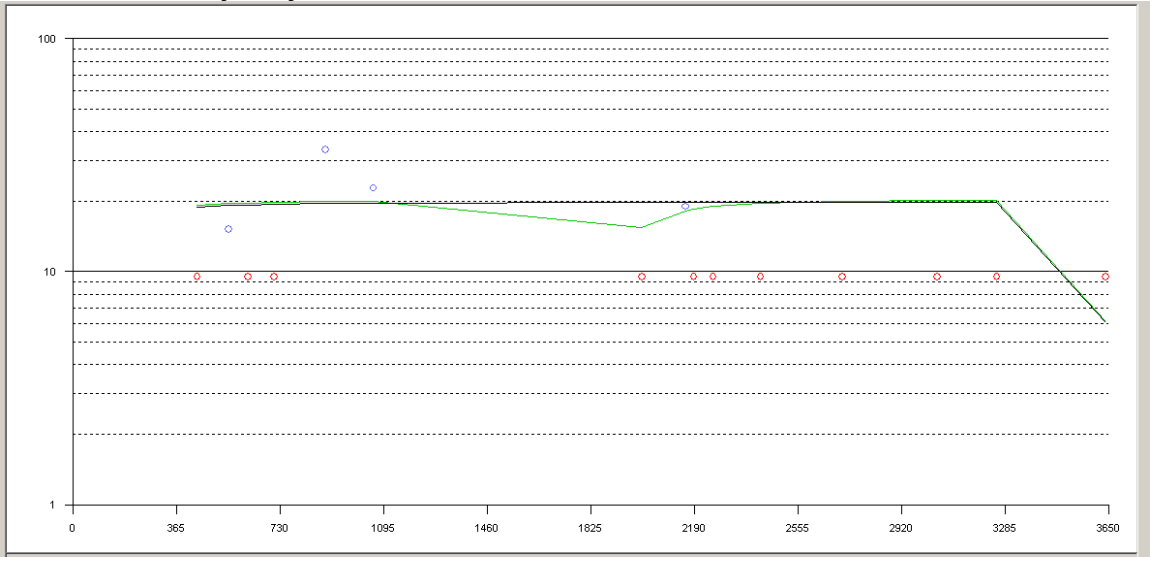
Intake:

	Intake #	**U**	**Th-230**	**Ra-226**	**Po-210**	**Pa-231**	**Ac-227**
1949 – 1952	1	3.00E+02	5.83E+02	4.66E+03	2.33E+03	5.83E+01	5.83E+01
1954	2	3.00E+02	2.59E+03	2.08E+04	1.04E+04	2.59E+02	2.59E+02
1955 – 1957	3	3.00E+02	1.25E+04	2.08E+04	1.04E+04	1.25E+03	1.25E+03
1958	4	3.00E+02	2.59E+03	2.08E+04	1.04E+04	2.59E+02	2.59E+02

Radium Bioassay Projection



Uranium Bioassay Projection



Dose:

Nuclide	Intake #	Dose pancreas, rem (through diagnosis)	Contribution (%)
U-234	1	1.60E-01	0%
Th-230	1	5.79E+00	5%
Ra-226	1	7.21E-01	1%
Po-210	1	6.25E-01	1%
Pa-231	1	2.19E-03	0%
Ac-227	1	3.47E-02	0%
U-234	2	3.79E-02	0%
Th-230	2	5.98E+00	5%
Ra-226	2	8.03E-01	1%
Po-210	2	6.96E-01	1%
Pa-231	2	2.23E-03	0%
Ac-227	2	3.78E-02	0%
U-234	3	1.10E-01	0%
Th-230	3	8.29E+01	75%
Ra-226	3	2.41E+00	2%
Po-210	3	2.09E+00	2%
Pa-231	3	3.07E-02	0%
Ac-227	3	5.42E-01	0%
U-234	4	3.53E-02	0%
Th-230	4	5.47E+00	5%
Ra-226	4	8.01E-01	1%
Po-210	4	6.96E-01	1%
Pa-231	4	2.01E-03	0%
Ac-227	4	3.69E-02	0%
Total		**1.10E+02**	

POC:

ICD-9	Organ	POC
157.0	Pancreas	50.19%

Alternative Organ Doses:

Nuclide	Intake #	Dose HNM, rem	Dose liver, rem	Dose Bone surface, rem	Dose kidney, rem
U-234	1	1.60E-01	6.11E-01	4.46E+00	1.62E+00
Th-230	1	5.79E+00	6.66E+01	5.16E+03	7.68E+01
Ra-226	1	7.34E-01	3.08E+00	2.23E+02	7.23E-01
Po-210	1	3.57E+00	1.47E+01	3.57E+00	2.85E+01
Pa-231	1	3.61E-03	8.87E-02	1.47E+03	1.38E-01
Ac-227	1	6.65E-02	1.38E+03	6.04E+03	3.03E-02
U-234	2	3.79E-02	1.52E-01	1.10E+00	4.05E-01
Th-230	2	5.98E+00	7.05E+01	5.42E+03	8.19E+01
Ra-226	2	8.17E-01	3.44E+00	2.45E+02	8.05E-01
Po-210	2	3.98E+00	1.64E+01	3.98E+00	3.18E+01
Pa-231	2	3.67E-03	9.83E-02	1.50E+03	1.53E-01
Ac-227	2	7.26E-02	1.51E+03	6.55E+03	3.30E-02
U-234	3	1.10E-01	4.52E-01	3.26E+00	1.22E+00
Th-230	3	8.29E+01	9.93E+02	7.59E+04	1.16E+03
Ra-226	3	2.45E+00	1.03E+01	7.28E+02	2.42E+00
Po-210	3	1.19E+01	4.94E+01	1.19E+01	9.56E+01
Pa-231	3	5.05E-02	1.42E+00	2.05E+04	2.22E+00
Ac-227	3	1.04E+00	2.16E+04	9.35E+04	4.73E-01
U-234	4	3.53E-02	1.50E-01	1.07E+00	4.05E-01
Th-230	4	5.47E+00	6.65E+01	5.05E+03	7.80E+01
Ra-226	4	8.15E-01	3.44E+00	2.40E+02	8.03E-01
Po-210	4	3.98E+00	1.64E+01	3.98E+00	3.18E+01
Pa-231	4	3.29E-03	9.81E-02	1.34E+03	1.53E-01
Ac-227	4	7.09E-02	1.47E+03	6.35E+03	3.22E-02
Total		**1.30E+02**	**2.73E+04**	**2.30E+05**	**1.60E+03**

ATTACHMENT 3d

Dose Reconstruction Example
Case 4
Thorium worker, uranium and radon breath, plant 7E worker without thorium bioassay

Selection Criteria

'plant 7E' on bioassay card / record

Cancer Description

pancreas (157.0)

Employment (Destrehan St.)

Start: 1949
End: 1958+ (break in employment 1953)

Work History

NOCTS : 'laboratory technician'
Dosimetry Data: 'LAB', 'CHEM', 'chemist M.E. Process (sp) plant 7E', 'TECH M.E. Process(sp) plant 7E'

Data Summary

External			Uranium					
Start (yr)	End (yr)	Deep Total (rem)	Start (yr)	End (yr)	Count	min (mg/l)	max (mg/l)	Avg (mg/l)
49	59	8.2	50	59	15	0.000	0.035	0.009

Thorium					
Start	End	Count	min (dpm/l)	Max (dpm/l)	Avg (dpm/l)
No data					

Radon					
Start	End	Count	min % tol	max % tol	Avg % tol
50	54	7	10	60	27

Data

Uranium Urine

Sample Date	Reported Result (mg/l)	Normalized Result (pCi/d)
06/22/50	0	<9.5
10/11/50	0.016	15.3
12/19/50	0.008	<9.5
03/21/51	0.003	<9.5
09/18/51	0.035	33.4
03/04/52	0.024	22.9
10/06/54	0.005	<9.5
03/10/55	0.02	19.1
04/07/55	0.004	<9.5
06/15/55	0.004	<9.5
11/29/55	0.004	<9.5
09/12/56	0.004	<9.5
08/13/57	0.005	<9.5
03/10/58	0.002	<9.5
03/30/59	0.002	<9.5

Radon Breath

Samp Date	% tolerance*	% tolerance*	% tolerance*	Ra WB (pCi)
01/03/1950	10		10	25200
3/12/1950	20	20	20	50400
7/24/1950	40		40	100800
6/25/1951	20		20	50400
10/18/1954	60		60	151200

*tolerance = 1 pCi/L

Approach:

Uranium intake from uranium bioassay data. Radium intake from radon breath monitoring data (using OTIB -025). Polonium intake normalized to radium intake, based on ratios. Thorium, protactinium, and actinium intake normalized to uranium intake, based on ratios:

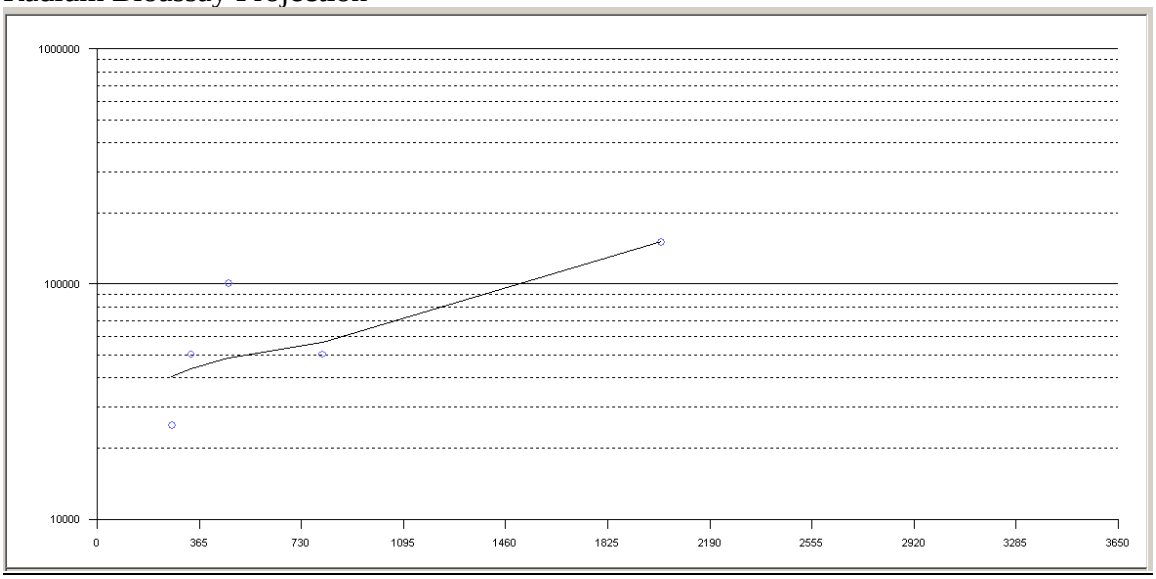
Ratios:

	Th-230	Po-210	Pa-231	Ac-227
X/Ra	0.125	0.500	0.013	0.013

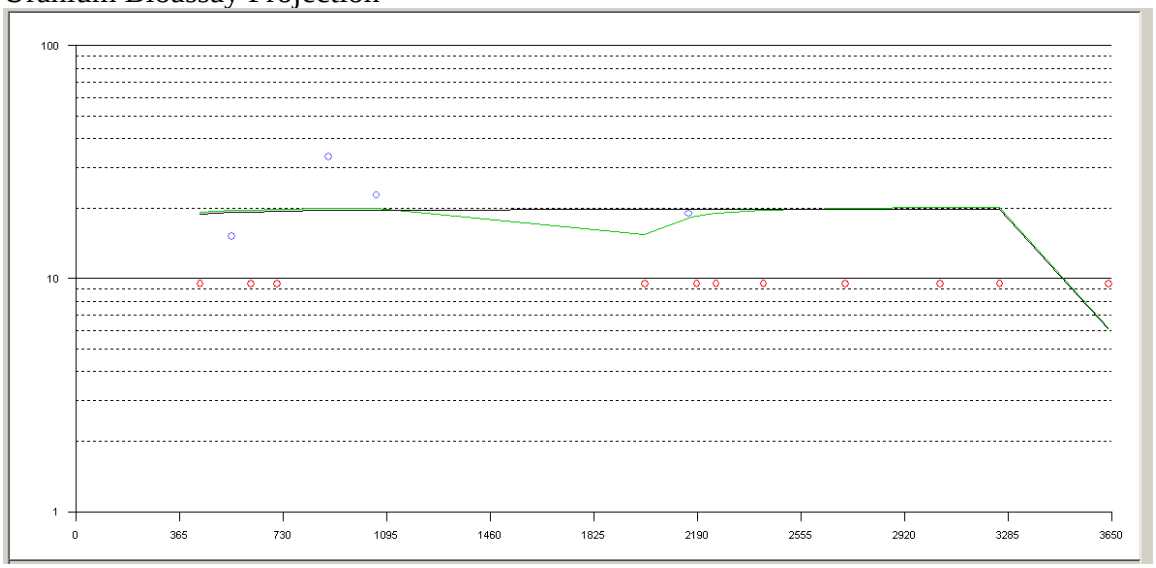
Intake:

	Intake #	U	Th-230	Ra-226	Po-210	Pa-231	Ac-227
1949 – 1952	1	3.00E+02	5.83E+02	4.66E+03	2.33E+03	5.83E+01	5.83E+01
1954	2	3.00E+02	2.59E+03	2.08E+04	1.04E+04	2.59E+02	2.59E+02
1955 – 1957	3	3.00E+02	3.00E+04	2.08E+04	1.04E+04	3.00E+03	3.00E+03
1958	4	3.00E+02	2.59E+03	2.08E+04	1.04E+04	2.59E+02	2.59E+02

Radium Bioassay Projection



Uranium Bioassay Projection



Dose:

Nuclide	Intake #	Dose pancreas, rem	Contribution (%)
U-234	1	1.60E-01	0%
Th-230	1	5.79E+00	3%
Ra-226	1	7.21E-01	0%
Po-210	1	6.25E-01	0%
Pa-231	1	2.19E-03	0%
Ac-227	1	3.47E-02	0%
U-234	2	3.79E-02	0%
Th-230	2	5.98E+00	3%
Ra-226	2	8.03E-01	0%
Po-210	2	6.96E-01	0%
Pa-231	2	2.23E-03	0%
Ac-227	2	3.78E-02	0%
U-234	3	1.10E-01	0%
Th-230	3	1.99E+02	88%
Ra-226	3	2.41E+00	1%
Po-210	3	2.09E+00	1%
Pa-231	3	7.37E-02	0%
Ac-227	3	1.30E+00	1%
U-234	4	3.53E-02	0%
Th-230	4	5.47E+00	2%
Ra-226	4	8.01E-01	0%
Po-210	4	6.96E-01	0%
Pa-231	4	2.01E-03	0%
Ac-227	4	3.69E-02	0%
Total		**2.27E+02**	

POC:

ICD-9	Organ	POC
157.0	Pancreas	67.19%

Alternative Organ Doses:

Nuclide	Intake #	Dose HNM, rem	Dose liver, rem	Dose Bone surface, rem	Dose kidney, rem
U-234	1	1.60E-01	6.11E-01	4.46E+00	1.62E+00
Th-230	1	5.79E+00	6.66E+01	5.16E+03	7.68E+01
Ra-226	1	7.34E-01	3.08E+00	2.23E+02	7.23E-01
Po-210	1	3.57E+00	1.47E+01	3.57E+00	2.85E+01
Pa-231	1	3.61E-03	8.87E-02	1.47E+03	1.38E-01
Ac-227	1	6.65E-02	1.38E+03	6.04E+03	3.03E-02
U-234	2	3.79E-02	1.52E-01	1.10E+00	4.05E-01
Th-230	2	5.98E+00	7.05E+01	5.42E+03	8.19E+01
Ra-226	2	8.17E-01	3.44E+00	2.45E+02	8.05E-01
Po-210	2	3.98E+00	1.64E+01	3.98E+00	3.18E+01
Pa-231	2	3.67E-03	9.83E-02	1.50E+03	1.53E-01
Ac-227	2	7.26E-02	1.51E+03	6.55E+03	3.30E-02
U-234	3	1.10E-01	4.52E-01	3.26E+00	1.22E+00
Th-230	3	1.99E+02	2.38E+03	1.82E+05	2.78E+03
Ra-226	3	2.45E+00	1.03E+01	7.28E+02	2.42E+00
Po-210	3	1.19E+01	4.94E+01	1.19E+01	9.56E+01
Pa-231	3	1.21E-01	3.42E+00	4.93E+04	5.33E+00
Ac-227	3	2.50E+00	5.19E+04	2.24E+05	1.13E+00
U-234	4	3.53E-02	1.50E-01	1.07E+00	4.05E-01
Th-230	4	5.47E+00	6.65E+01	5.05E+03	7.80E+01
Ra-226	4	8.15E-01	3.44E+00	2.40E+02	8.03E-01
Po-210	4	3.98E+00	1.64E+01	3.98E+00	3.18E+01
Pa-231	4	3.29E-03	9.81E-02	1.34E+03	1.53E-01
Ac-227	4	7.09E-02	1.47E+03	6.35E+03	3.22E-02
Total		**2.48E+02**	**5.89E+04**	**4.96E+05**	**3.22E+03**

ATTACHMENT 3e

8/2/2005
DRAFT
Evaluation of Chronic vs. Acute Intake Assumptions

Background

When fitting urinalysis measurement to determine intakes, the intake regime is always an important consideration. Barring any indication of unusual conditions, a chronic exposure is often chosen as the initial assumption when assessing cases. To explore the adequacy of this assumption, the following analysis was performed.

Table 1 provides a list of urine samples collected from an individual who was employed at Mallinckrodt Chemical Works from 1947 through 1958. The samples, which were reported in units of mg U/L, were converted to pCi/day using the uranium specific activity of natural uranium. For this analysis, a detection limit of 0.007 mg/L was assumed.

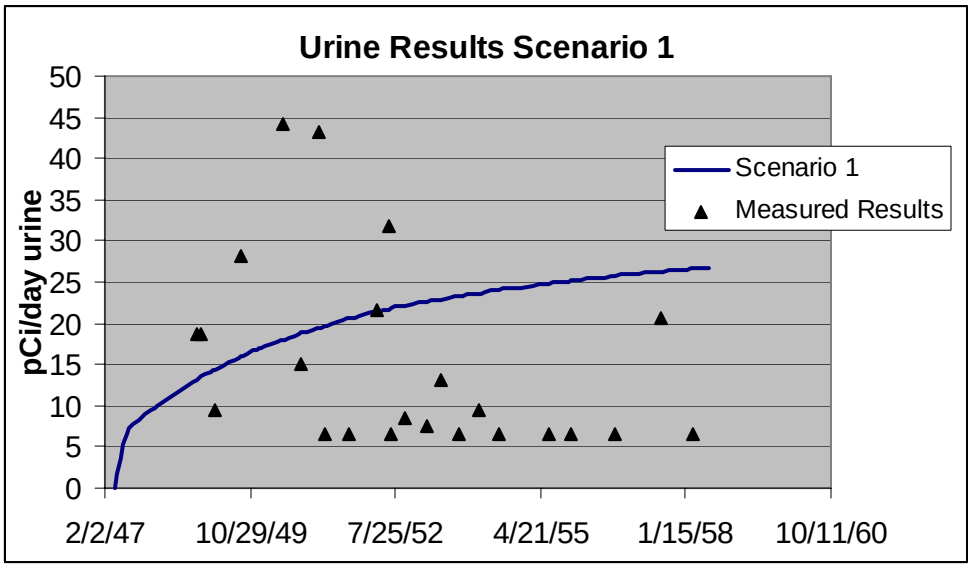
Table 1. Urinalysis Results

	mg/L	pCi/day	pCi/day
10/27/1948	0.02	18.76	18.76
12/1/1948	0.02	18.76	18.76
3/2/1949	0.01	9.38	9.38
8/26/1949	0.03	28.14	28.14
6/6/1950	0.047	44.09	44.09
10/17/1950	0.016	15.01	15.01
2/13/1951	0.046	43.15	43.15
3/28/1951	0.007	6.57	<6.57
9/11/1951	0.004	3.75	<6.57
3/14/1952	0.023	21.57	21.57
6/17/1952	0.034	31.89	31.89
7/1/1952	0.003	2.81	<6.57
9/26/1952	0.009	8.44	8.44
3/5/1953	0.008	7.50	7.50
6/6/1953	0.014	13.13	13.13
10/6/1953	0.004	3.75	<6.57
2/16/1954	0.01	9.38	9.38
7/13/1954	0.002	1.88	<6.57
6/16/1955	0.004	3.75	<6.57
11/15/1955	0.007	6.57	<6.57
9/20/1956	0.002	1.88	<6.57
7/22/1957	0.022	20.64	20.64
2/28/1958	0.002	1.88	<6.57

For this evaluation, a number of intake scenarios were modeled to determine the effect of assumed exposure mode on the estimated total intake.

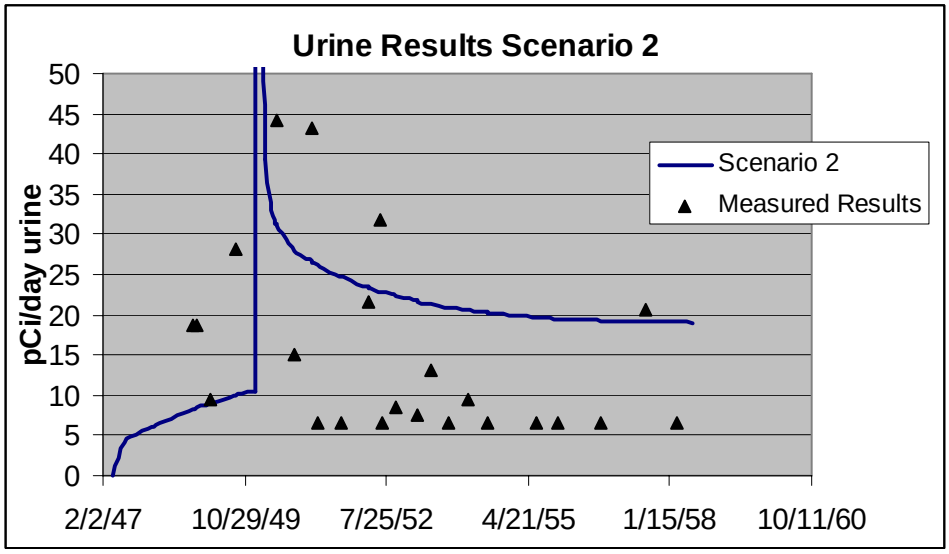
Scenario 1

The first scenario modeled was that of a chronic intake through the entire employment period. When the urine data were fit to this scenario, the resulting intake was determined to be 4735 pCi/day for the each calendar day of the 4091 day monitoring period which results in a total intake of 1.94E7 pCi. A graph of the urine results below shows the actual samples along with the results from this intake scenario.



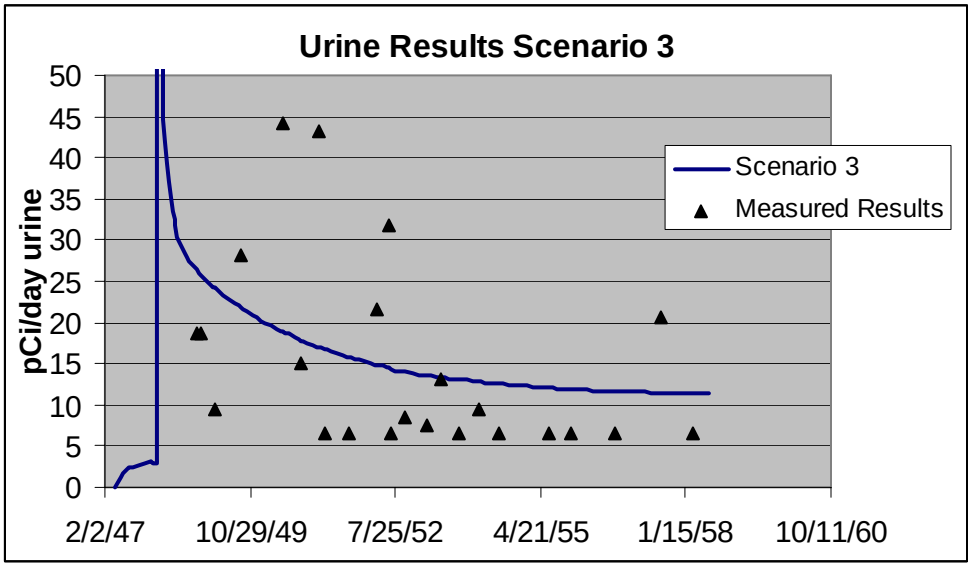
Scenario 2

The second scenario investigated included the chronic intake from the first scenario, but also included an acute intake on 1/15/1950 which is the mid-point between the highest sample on 6/6/50 and the previous sample on 8/26/49. The fit to this intake scenario resulted in a 2931 pCi/day intake for the chronic period plus 5.65E6 pCi for the acute intake. That results in a total intake of 1.76E7 pCi. A graph of the urine results below shows the actual samples along with the results from this intake scenario. The lower chronic intake in scenario #2 is caused by the fact that the acute intake assumption accounts for the highest samples allowing the chronic intake to only account for the remaining (lower) samples.



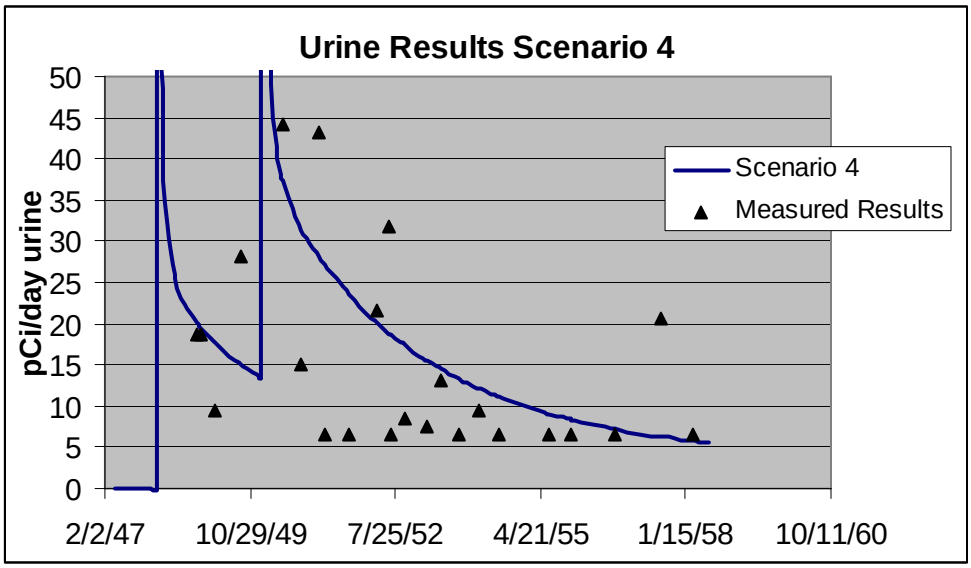
Scenario 3

The third scenario investigated included the chronic intake from the first scenario but also includes an acute intake on 1/22/1948. This date is the mid point between the first sample and the beginning of employment. When the urine data were fit using this scenario, the resulting intake was determined to be 1544 pCi/day for the chronic period plus 7.66E6 pCi for the acute intake. This yields a total intake of 1.40E7 pCi. A graph of the urine results below shows the actual samples along with the results from this intake scenario.



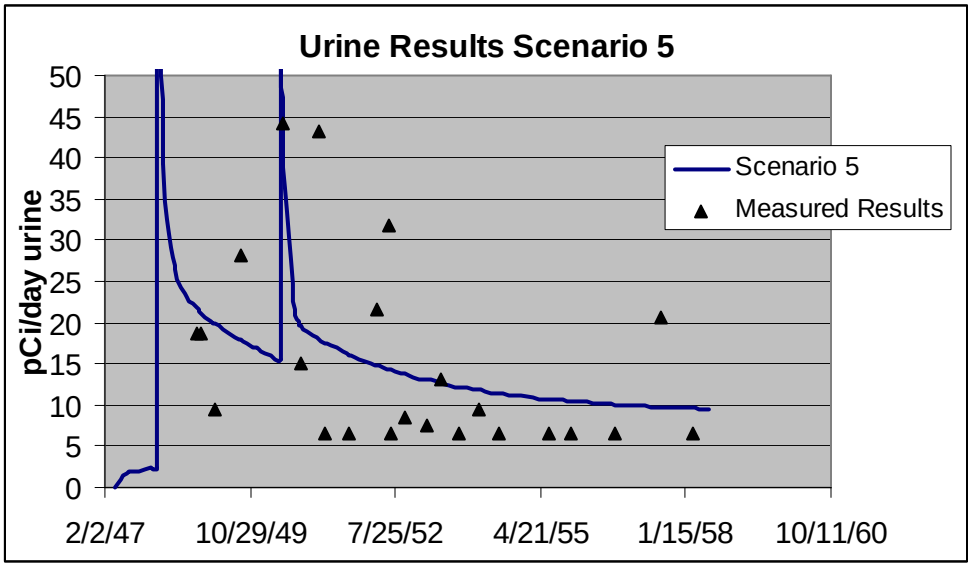
Scenario 4

The fourth scenario investigated included the chronic intake from the first scenario but also includes two acute intakes on the days from scenarios 2 and 3. When the urine data were fit using this intake scenario, the resulting intake was determined to be 1188 pCi/day for the chronic period plus 6.40E6 pCi for the first acute intake and 1.42E6 pCi for the second acute intake. That results in a total intake of 1.27E7 pCi. A graph of the urine results below shows the actual samples along with the results from this intake scenario.



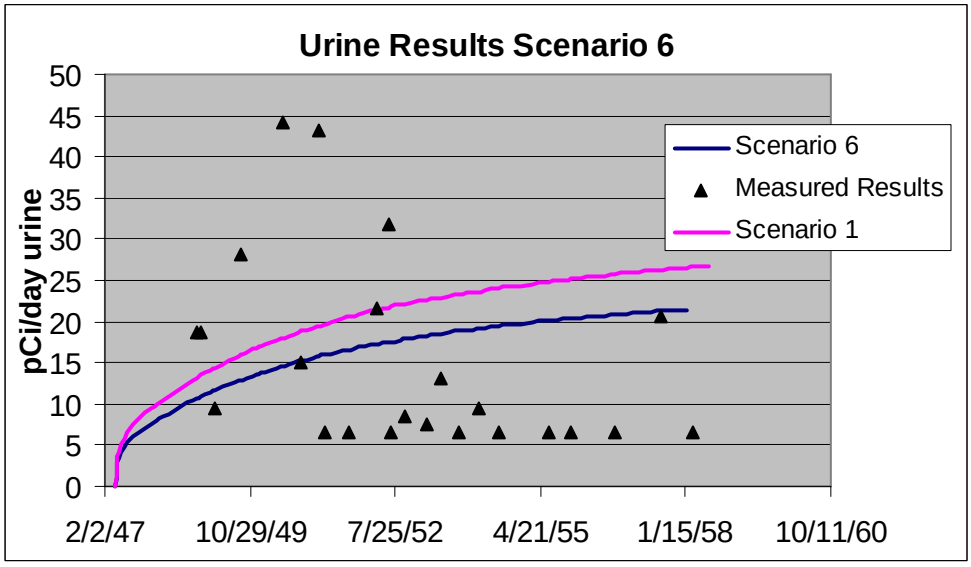
Scenario 5

The fifth scenario investigated included the chronic intake from the first scenario and also includes two acute intakes. The first acute intake is the same as scenario 3, but the second acute intake was moved closer to the date of the highest sample. This was done to bring the estimated urine results more inline with the sample result following the highest result. When the urine data were fit using this intake scenario, the resulting intake was determined to be 0 pCi/day for the chronic period plus 6.96E6 pCi for the first acute intake and 7.20E6 pCi for the second acute intake. That results in a total intake of 1.42E7 pCi. A graph of the urine results below shows the actual samples along with the results from this intake scenario.



Scenario 6

The last scenario investigated deals with the assumption that the individual was not monitored during some of his greatest intakes. For this scenario, one chronic intake was assumed just as it was in scenario 1, however, the highest two measurement results were excluded. This is intended to simulate a data set missing the worst case samples. When the urine data were fit using this intake scenario, the resulting chronic intake is smaller than the one determined in scenario 1. The new intake is 3830 pCi/day compared to 4735 pCi/day in scenario 1. The total intake is then 1.57E7 pCi compared to 1.94E7 pCi for scenario 1. This total intake, however, is still larger than all of the better fits even though the better fits used the highest samples. A graph of the urine results below shows the actual samples along with the results from this intake scenario. The original chronic assumption is also included.



Area Under the Curve

If the actual urine concentrations are plotted and connected by a line, the area under that line represents the total activity eliminated over the time period. In this case the time period is the individual's entire career. This total amount eliminated is proportional to the total intake and thus the dose. Since the individual was not sampled continuously, only an estimate of this elimination can be obtained. The simplest estimate is obtained by simply drawing a line that connects the sample results and then determining the area under that line. This analysis was performed for the measured results as well as the 6 scenarios described above. The result of this analysis as well as a summary of the total intake from each scenario is listed in Table 2.

Table 2. Summary of Results

	Total Intake (pCi)	Estimated Elimination (pCi)	Goodness of Fit
Measured results		5.52E+04	
Scenario 1	1.94E+07	8.58E+04	70.47
Scenario 2	1.76E+07	7.66E+04	62.15
Scenario 3	1.40E+07	6.71E+04	51.36
Scenario 4	1.42E+07	6.55E+04	47.35
Scenario 5	1.27E+07	6.61E+04	42.53
Scenario 6	1.57E+07	6.94E+04	62.96[a]

a Fit based on all samples. Fit was 48.41 based on the samples used for this scenario.

It can be seen from this table that, in general, the lower the estimated total intake is for the given scenario, the lower the area under the line. A subjective review of the plots of the individual scenarios indicates the difference in estimated intakes may be caused by the difference in how well the estimate fits to the data. In all fits, the scenario predicts a result that is higher than some measured data and lower than other data.

In order to explore this, a goodness of fit parameter was used. This parameter was simply the square root of the sum of the squares of the difference between the predicted value and the measured value. The parameter is included in the table above. The best two fits are scenario #4 and #5. These two estimates are within approximately 12% of each other. These are also the only two scenarios with three intake regimes. Both had a chronic intake plus two acute intakes.

This illustrates that as better fits are obtained, the total elimination (area under the line) estimated by the intake scenario tends to approach the total elimination calculated from the measured data. Therefore, when an intake amount is determined from the data, the total resultant intake is not very sensitive to the intake scenario chosen. In fact, a subjective review of a plot of the data versus the measured data can be sufficient to determine the adequacy of the intake estimate.

It can also be surmised that if all the sample data has relatively similar results, a chronic intake scenario will determine the total intake with reasonable accuracy. In the case

illustrated here, the highest sample is approximately 6.7 times the higher than the lowest result. However, the chronic intake estimate is only approximately 1.5 times higher than the lowest intake estimate.

Conclusion

From this analysis, the following conclusion can be drawn.

An assumption that acute intakes occurred in between measurement periods does not necessarily add additional dose. In fact, this often lowers dose since the addition of acute intakes will almost definitely cause the estimate to better fit the data.

Also, incidents that were not followed by samples can be accounted for by fitting existing urinalysis. Since samples taken after the incident will contain some uranium from the incident, this quantity has to be accounted for in the assumed intakes. Without an assumed acute intake, the samples are accounted for with a chronic intake which often over predicts the actual total intake.

ATTACHMENT 3f

ORAU TEAM
Dose Reconstruction
Project for NIOSH

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DRAFT

Statistical Analyses

of

Airborne Radon and Coworker Bioassay and External Dose Data

at the

Mallinckrodt Chemical Works (1945-57)

Prepared by

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Jennifer Hoff, PhD
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August 2, 2005

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1.0 PURPOSE

This document provides information to assist in demonstrating that usable coworker statistics can be performed using the results of statistical analyses for Mallinckrodt Chemical Works (MCW), Destrehan Street, tabulated data in the CER databases. Data include results from 1945-1957 for external (penetrating and shallow) radiation exposures, and results from 1948-1957 for uranium in urine, airborne radon, and breath radon. Although not included in this document, analyses of uranium in air are also planned. Further additions, changes, and improvements in the data, such as analyses by individual plant locations, are anticipated. Therefore, this information is not intended for dose reconstruction until all analyses are completed and the Subject Matter Expert has fully accepted and approved the results.

2.0 INTRODUCTION

Personnel and workplace monitoring data from MCW Destrehan Street provided through the ORAU Center for Epidemiologic Research were used for these statistical analyses.

ORAUT-OTIB-0019 is a higher-tier document that describes the overall responsibilities and methods used in this report. ORAUT-PROC-0095 further describes the statistical techniques used for this analysis.

3.0 ANALYTICAL RESULTS

This section provides a summary of the results of each of the Mallinckrodt statistical analyses for

- external penetrating radiation dose
- external shallow radiation dose
- uranium in urine
- breath radon
- area radon

Each summary includes the assumptions used for that particular analysis. All data sets were assumed to be log-normally distributed. This assumption was confirmed by the goodness-of-fit statistic labeled "R^2" in the summary tables. An R^2 value of 1.0 is a perfect fit to a log-normal distribution. All data sets were analyzed through 1957. The column labels from left to right have the following meanings:

- Year—year for which the data were analyzed
- Calculated 50th % (unit)—geometric mean of data set determined by equation
- Measured 50th % (unit)—geometric mean of data set determined by data rank
- Calculated 84th % (unit)—84th percentile of data set determined by equation
- Measured 84th % (unit)—84th percentile of data set determined by data rank
- # of records—number of data points used in analysis
- Equation slope—slope of linear equation fitted to log-transformed data
- Equation intercept—intercept of linear equation fitted to log-transformed data
- GSD—geometric standard deviation of the log-transformed data
- R^2—goodness-of-fit test for the log-transformed data and the linear equation

Jennifer Hoff, PhD, and Joseph Lochamy, CHP independently performed all analyses and compared their results. Therefore, the results have been validated for *computational* errors.

3.1 External Penetrating Radiation Dose

Data for external penetrating dose were available beginning with 1945. All results are normalized to a **weekly dose in mrem**, since many records were not full years of monitoring. Database entries with no monitoring period and no dose data were deleted, accounting for approximately 21 records. Blanks (There were no "0" entries.) in the dose columns of the database (but with monitoring periods) were changed to 0 mrem, initially, to facilitate inserting “missed dose.” “Missed dose” was accounted for by the methods described in OCAS-IG-001, External Dose Reconstruction Implementation Guide, and is not described in detail here. The “missed dose” is based on the Mallinckrodt Technical Basis Document (TKBS-0005) statement that the film badges had a 50-mrem sensitivity. Common practice is to substitute half of such values during monitoring periods where no dose is recorded. A weekly administrative limit of 300 mrem was assumed for this analysis. Tables 1 and 2, respectively, provide the statistical results for the condition were the “missed dose” is applied to every “zero” monitoring period and to 50% of the monitoring periods.

Table 1. Summary Statistics of External Penetrating Dose, mrem/week (1945-57)
(Including 100% of “Missed Dose”)

Year	Calculated 50th %, mrem/wk	Measured 50th %, mrem/wk	Calculated 84th %, mrem/wk	Measured 84th %, mrem/wk	# of records	Equation Slope	Equation Intercept	GSD	R^2
1945	109.96	98.00	182.99	155.76	39	0.51	4.70	1.66	0.90
1946	60.53	50.00	123.61	130.76	170	0.71	4.10	2.04	0.93
1947	108.12	90.00	205.73	249.48	285	0.64	4.68	1.90	0.95
1948	65.58	58.00	123.48	128.88	484	0.63	4.18	1.88	0.93
1949	44.76	38.00	73.34	78.20	506	0.49	3.80	1.64	0.87
1950	37.64	31.00	58.16	66.00	608	0.44	3.63	1.55	0.87
1951	40.58	37.00	55.79	54.00	729	0.32	3.70	1.37	0.87
1952	32.10	28.00	43.98	40.00	759	0.31	3.47	1.37	0.77
1953	30.20	28.00	41.65	39.00	781	0.32	3.41	1.38	0.91
1954	27.35	26.00	38.96	37.00	785	0.35	3.31	1.42	0.96
1955	21.33	20.00	30.56	30.00	945	0.36	3.06	1.43	0.94
1956	19.40	18.00	25.73	25.00	1040	0.28	2.97	1.33	0.93
1957	16.77	16.00	20.93	21.00	1024	0.22	2.82	1.25	0.95

Table 2. Summary Statistics of External Penetrating Dose, mrem/week (1945-57)
(Including 50% of “Missed Dose”)

Year	Calculated 50th %, mrem/wk	Measured 50th %, mrem/wk	Calculated 84th %, mrem/wk	Measured 84th %, mrem/wk	# of records	Equation Slope	Equation Intercept	GSD	R^2
1945	99.89	90.00	180.39	149.76	39	0.59	4.60	1.81	0.90
1946	46.99	41.00	117.34	123.88	170	0.92	3.85	2.50	0.96
1947	96.43	80.00	202.14	247.48	285	0.74	4.57	2.10	0.96
1948	52.21	47.50	114.83	121.60	484	0.79	3.96	2.20	0.96
1949	31.29	26.50	61.54	69.00	506	0.68	3.44	1.97	0.91

Year	Calculated 50th %, mrem/wk	Measured 50th %, mrem/wk	Calculated 84th %, mrem/wk	Measured 84th %, mrem/wk	# of records	Equation Slope	Equation Intercept	GSD	R^2
1950	25.29	20.00	45.87	56.00	608	0.60	3.23	1.81	0.88
1951	28.17	25.00	44.10	43.00	729	0.45	3.34	1.57	0.91
1952	19.79	16.00	31.61	29.00	759	0.47	2.99	1.60	0.82
1953	20.25	19.00	31.65	30.00	781	0.45	3.01	1.56	0.93
1954	20.25	20.00	32.24	31.00	785	0.46	3.01	1.59	0.97
1955	15.14	14.00	25.13	24.00	945	0.51	2.72	1.66	0.97
1956	13.06	12.00	19.89	19.00	1040	0.42	2.57	1.52	0.97
1957	10.92	11.00	15.46	16.00	1024	0.35	2.39	1.42	0.97

3.2 External Shallow Radiation Dose

Data for external shallow dose were available beginning with 1945. All results are normalized to a **weekly dose in mrem**, since many records were not full years of monitoring. Database entries with no monitoring period and no dose data were deleted, accounting for approximately 21 records. Blanks (There were no "0" entries.) in the dose columns of the database (but with monitoring periods) were changed to 0 mrem, initially, to facilitate inserting “missed dose.” “Missed dose” was accounted for by the methods described in OCAS-IG-001, External Dose Reconstruction Implementation Guide, and is not described in detail here. The “missed dose” is based on the Mallinckrodt Technical Basis Document (TKBS-0005) statement that the film badges had a 50-mrem sensitivity. Common practice is to substitute half of such values during monitoring periods where no dose is recorded. A weekly administrative limit of 600 mrem was assumed for this analysis. Tables 3 and 4, respectively, provide the statistical results for the condition were the “missed dose” is applied to every “zero” monitoring period and to 50% of the monitoring periods.

Table 3. Summary Statistics of External Shallow Dose, mrem/week (1945-57)
(Including 100% of “Missed Dose”)

Year	Calculated 50th %, mrem/wk	Measured 50th %, mrem/wk	Calculated 84th %, mrem/wk	Measured 84th %, mrem/wk	# of records	Equation Slope	Equation Intercept	GSD	R^2
1945	92.63	81.67	169.54	143.14	39	0.60	4.53	1.83	0.94
1946	40.37	33.89	111.83	116.64	170	1.02	3.70	2.77	0.98
1947	86.41	71.00	192.55	245.09	285	0.80	4.46	2.23	0.96
1948	35.60	36.66	113.02	113.93	484	1.16	3.57	3.17	0.98
1949	15.43	15.89	54.35	59.07	506	1.26	2.74	3.52	0.99
1950	10.87	8.92	36.03	47.98	608	1.20	2.39	3.31	0.98
1951	14.58	13.91	33.11	31.87	729	0.82	2.68	2.27	0.99
1952	6.47	6.52	23.53	25.00	759	1.29	1.87	3.64	0.99
1953	9.48	9.65	24.28	21.88	781	0.94	2.25	2.56	0.99
1954	12.65	12.85	26.49	24.99	785	0.74	2.54	2.09	0.99
1955	8.72	9.38	21.41	20.55	945	0.90	2.17	2.46	0.99
1956	6.81	6.83	15.85	15.72	1040	0.85	1.92	2.33	1.00
1957	5.31	5.19	12.10	12.24	1024	0.82	1.67	2.28	0.99

Table 4. Summary Statistics of External Shallow Dose, mrem/week (1945-57) (Including 50% of "Missed Dose")

Year	Calculated 50th %, mrem/wk	Measured 50th %, mrem/wk	Calculated 84th %, mrem/wk	Measured 84th %, mrem/wk	# of records	Equation Slope	Equation Intercept	GSD	R^2
1945	74.67	80.00	247.10	279.56	39	1.20	4.31	3.31	0.92
1946	64.64	81.50	169.15	172.00	170	0.96	4.17	2.62	0.95
1947	159.92	172.00	314.64	321.12	285	0.68	5.07	1.97	0.94
1948	91.56	94.00	186.79	177.72	484	0.71	4.52	2.04	0.97
1949	63.90	51.00	130.49	126.20	506	0.71	4.16	2.04	0.92
1950	39.44	36.00	88.09	94.88	608	0.80	3.67	2.23	0.95
1951	31.90	28.00	60.47	66.52	729	0.64	3.46	1.90	0.95
1952	30.87	27.00	59.15	67.00	759	0.65	3.43	1.92	0.96
1953	28.81	24.00	58.33	68.00	781	0.71	3.36	2.02	0.97
1954	24.97	21.00	54.58	65.00	785	0.78	3.22	2.19	0.95
1955	23.48	22.00	52.92	58.96	945	0.81	3.16	2.25	0.97
1956	19.06	18.00	39.90	46.00	1040	0.74	2.95	2.09	0.96
1957	15.48	14.00	30.27	33.00	1024	0.67	2.74	1.95	0.96

3.3 Uranium in Urine

Data for uranium excretion in urine are available beginning with 1948. All results are in units of milligrams/liter of urine. Zeros were left as such and used. Page 78 of TKBS-005 says that a detection threshold of 10 µg/L (0.01 mg/L) should be assumed. However, all data years include significant quantities of data at 0.001-0.01 mg/L, as well as some zero entries. The analyst concludes that the *actual* results measured were recorded, regardless of statistical significance. Therefore, the analysts did not modify the activity data by substituting other values for zeros or any other data. Where there were blanks in the year column with database comments stating the (partial) year of the analysis, the year described was inserted and the data used. If a year was blank and the comments spanned two years, the year with the longest span was used. Two blank-year entries spanned more than two years; thus, the data for those records were not used. A few of the larger plots for 1953-54 (particularly 1954) do not fit the assumed distribution well. However, these data do not affect the analysis results and were included in the analyses. It is likely that statistical analyses over a period less than a year (such as quarterly or monthly) could yield better results for dose reconstruction. The maximum measured value is also provided. Table 5 below provides the statistical results.

Table 5. Summary Statistics of Uranium in Urine, milligrams/liter (1948-57)

Year	Calculated 50th %, mg/L	Measured 50th %, mg/L	Calculated 84th %%, mg/L	Measured 84th %%, mg/L	# of records	Equation Slope	Equation Intercept	GSD	R^2	Max mg/L
1948	0.03	0.04	0.14	0.13	292	1.46	-3.43	4.30	0.93	2
1949	0.02	0.02	0.06	0.06	638	1.43	-4.17	4.17	0.94	1.23
1950	0.01	0.01	0.03	0.03	1091	1.17	-4.69	3.24	0.95	0.8

Year	Calculated 50th %, mg/L	Measured 50th %, mg/L	Calculated 84th %%, mg/L	Measured 84th %%, mg/L	# of records	Equation Slope	Equation Intercept	GSD	R^2	Max mg/L
1951	0.01	0.01	0.02	0.02	1279	0.99	-4.76	2.69	0.98	0.2
1952	0.01	0.01	0.02	0.02	1910	1.03	-4.76	2.79	0.97	0.57
1953	0.01	0.01	0.02	0.02	2421	1.00	-4.81	2.71	0.98	1.014
1954	0.01	0.01	0.02	0.02	1943	1.05	-5.07	2.85	0.98	10.13
1955	0.01	0.01	0.02	0.02	1897	0.99	-5.05	2.69	0.97	0.912
1956	0.01	0.01	0.01	0.01	802	0.94	-5.25	2.56	0.98	0.22
1957	0.00	0.00	0.01	0.01	1038	1.31	-5.61	3.71	0.98	0.142

3.4 Breath Radon

Data for breath radon are available beginning with 1948. All results are in units of **percent of the 1-pCi/liter tolerance limit**, *even though the database lists its units as pCi/liter*. There was only one data point in 1956, so there are no statistical values for that year. Except for 1949 the data in the first "Result (pCi/L)" column in the database were used. For the 20 blank results in the first data column of the database in 1949, the recorded % tolerance limit values were used. For all “<10 pCi/L” data points and blank entries, the data were linearly transformed between 0 and 10 before being log transformed. Values in the second "Result (pCi/L)" data column of the database were ignored since very little data appear in that column. Table 6 below provides the statistical results.

Table 6. Summary Statistics of Radon in Breath, % of 1-pCi/L Tolerance Limit (1948-57)

Year	Calculated 50th %, Tol. Lmt %	Measured 50th %, Tol. Lmt %	Calculated 84th %%, Tol. Lmt %	Measured 84th %%, Tol. Lmt %	# of records	Equation Slope	Equation Intercept	GSD	R^2
1948	35.02	33.50	58.29	58.00	118	0.51	3.56	1.66	0.99
1949	28.10	32.00	75.15	63.24	187	0.98	3.34	2.67	0.96
1950	32.14	35.00	72.05	70.00	175	0.81	3.47	2.24	0.96
1951	14.21	10.00	24.89	30.00	469	0.56	2.65	1.75	0.81
1952	16.61	20.00	30.36	30.00	390	0.60	2.81	1.83	0.87
1953	21.61	20.00	44.60	40.00	388	0.72	3.07	2.06	0.92
1954	19.98	20.00	48.51	40.00	402	0.89	2.99	2.43	0.94
1955	20.11	20.00	51.27	40.00	240	0.94	3.00	2.55	0.94
1956	No Data	10.00	No Data	10.00	1	No Data	No Data	No Data	No Data
1957	5.21	6.00	9.81	8.72	5	0.63	1.65	1.88	0.95

3.5 Airborne Radon

Data for airborne radon are available beginning with 1948. All results appear to be in units of pCi/liter, as stated in the database. "N" was assumed to be a "non-detect," and, during the periods when "N" was used, the detection limit was <3 pCi/L. For all "0" results (1948-1949), <3 pCi/L was assumed. In later years, the detection limit was <1.0 pCi/L. For all "N" results (1948-1951), where a blank appeared in the first data column, but a 3 appeared in the second data column, <3 pCi/L was assumed. *Blank* entries occurred in both the first and second data columns of the database for 3 cases

in 1955. The analysts assumed a cut off of <1 pCi/L. After adjusting the database as described above, all values <=3 were set to zero and included in the statistical *ranking* but not included in the *line equation* determination. Table 7 below provides the statistical results.

Table 7. Summary Statistics of Airborne Radon, pCi/liter (1948-57)

Year	Calculated 50th %, pCi/L	Measured 50th %, pCi /L	Calculated 84th %%, pCi/L	Measured 84th %%, pCi/L	# of records	Equation Slope	Equation Intercept	GSD	R^2
1948	33.18	37.50	150.14	160.00	386	1.51	3.50	4.53	0.97
1949	11.50	10.00	72.43	63.00	507	1.84	2.44	6.30	0.98
1950	18.82	22.00	65.01	57.36	380	1.24	2.93	3.45	0.96
1951	16.75	19.00	57.55	54.80	721	1.23	2.82	3.44	0.98
1952	10.83	12.00	52.66	48.88	708	1.58	2.38	4.86	0.97
1953	9.25	9.00	39.05	40.00	613	1.44	2.22	4.22	0.97
1954	8.61	7.00	45.88	50.92	714	1.67	2.15	5.33	0.97
1955	9.56	10.00	47.91	50.80	621	1.61	2.26	5.01	0.98
1956	5.23	<1	22.41	27.00	215	1.45	1.65	4.28	0.91
1957	3.36	<1	20.01	25.00	210	1.78	1.21	5.96	0.96

ATTACHMENT 4

Transcript of August 4, 2005 Working Group Meeting in Cincinnati, Ohio

The transcript was not available on the date of publication of this review. It will be posted on the NIOSH OCAS website when it is available.

ATTACHMENT 5

E-Mail Communications between Jim Neton, NIOSH and Arjun Makhijani, SC&A

Note: Compiled from the e-mail file of Arjun Makhijani. E-mails transmitting documents are not included. All documents transmitted are included as attachments in this SC&A report. E-mails are shown in six groups for convenience of reading.

E-Mail Exchange Number 1

-----Original Message-----
From: Arjun Makhijani [mailto:arjun@ieer.org]
Sent: Friday, August 05, 2005 12:06 PM
To: Neton, Jim
Cc: Wade, Lewis; griffonm@comcast.net; Wanda Munn; James Melius; Michael Gibson; Gen Roessler; jmauro@scainc.com; kbehling@scainc.com; joyce; dunstana@globo.com; anigstein@verizon.net; Anigstein@cs.com; mikethorneltd; Denise Brock; Denise Brock
Subject: Query and protocol

Hi Jim

Just to open up communications and set forth a protocol for keeping track of them.

I talked with John this morning about who to copy on SC&A communications

with NIOSH regarding completion of SC&A review of the MCW TBD. I will be copying the Board Working Group, Dr. Wade, Denise Brock, and the SC&A

team working on the Mallinckrodt review. Would you please do the same? Thanks.

Dr. Wade and Board members: if you want me to copy the whole Board I will do so, of course. If you would like a different protocol, please let Jim and me know.

I have three items today:

1. I am attaching the two page 1951 document re: radon breath that Denise Brock gave me at the meeting that seems difficult of interpretation but may be important for understanding radon breath data variability. This is the document I briefly showed you yesterday afternoon.

2. In table 2 of the draft write-up, the GSDs for radon data are shown as ranges. I don't know that this means or how these values were derived. Could you please explain? How do the 84 and 95th percentiles relate to these GSD ranges?

3. Re: radon exposure and systemic organs. There were likely to have been workers exposed to high levels of radon but not correspondingly

high levels of radium, as with de-lidders and warehouse workers. How will this be handled? And to what groups of workers will the minimum detectable limit of 0.1 picocuries per liter be applied?

Thanks

Arjun
--

Jim Neton response, 8/8/2005, 12:52 p.m.

Arjun,

Thank you for sending us the radon breath document that was provided to you by Denise Brock. We are looking at it very closely to see what implications it might have regarding the interpretation of radon breath data.

Our responses to the two questions raised in your e-mail are provided below.

The radon breath data presented in table 2 describes the ranges of values observed between 1949 and 1957. Although table 2 provides the data by area within plant 6, the combined analyses for all of plant 6 are provided in Appendix C. We propose using the 95th percentile values, derived from the distributions in Appendix C, for plant 6 workers and the full distribution for those not directly associated with plant 6 activities.

As far as radon breath analyses go, NIOSH will use the actual breath data to estimate radium intakes wherever possible. For plant 6 workers with no radon breath results, the 95th percentile will be assigned. Thus, for workers for whom it not possible to determine radium exposure status, we will default to the assumption that they were exposed to K-65 material. While we are still working on the calculations, it seems to us that, even for exposures at the highest radon concentrations, the radium intakes will provide more dose than the corresponding dose from the radon gas/progeny exposure alone.

We are still working on the Th-230 ratio calculations and would like to discuss our thoughts with you on this issue tomorrow afternoon. How does 1:00 tomorrow work for you?

Thanks,

Jim

E-Mail Exchange Number 2:

-----Original Message-----
From: Arjun Makhijani [mailto:arjun@ieer.org]
Sent: Monday, August 08, 2005 2:04 PM
To: Neton, Jim

Cc: Wade, Lewis; griffonm@comcast.net; Wanda Munn; James Melius; Michael Gibson; Gen Roessler; jmauro@scainc.com; kbehling@scainc.com; joyce; dunstana@globo.com; anigstein@verizon.net; Anigstein@cs.com; mikethorneltd; Denise Brock; Denise Brock; Elliott, Larry J.; Cindy Bloom and Steve LaMonica; joe guido; Allen, David [NIOSH]
Subject: Re: Query and protocol

Hi Jim

1. Tomorrow and 1 pm Eastern is fine with me. We can both keep brief notes and send them to this e-mail list. is that okay?

2. Thanks for the responses. I understand now that the range of GSDs in

Table 3 must be the range of values for the various years. Right?

3. The team that is working with me on the radon issue has also been thinking about Dave Allen's idea re: radon. Let me share a couple of questions with you. First, since the radon biological half life is very

short (~24 hours) and radon breath measurements were every few months at

best, how can the radon exposure connection to radon breath (i.e., radium body burden) be established? Second, it would seem that there would have been job categories where high radon exposure without high radium exposure was possible, as for instance with de-lidders, or in the

warehouse. So, to whom would the MDL for radon breath be allocated?

Best

Arjun
--

Jim Neton response, 8/8/2005, 3:09 p.m.

Arjun,

1. It sounds like we're on for 1:00 tomorrow. I anticipate that the call will be brief, but just so interested parties can listen in, I've arranged for a call in number. The number is 866-718-3335 (pass code = 6841661). Taking brief notes and distributing them to the e-mail list seems consistent with the guidance we received at the working group meeting last week.

2. You are correct. The ranges for the GSDs in table 2 are the ranges for the various years.

3. I think that we have a path forward for bounding radon gas/progeny exposure that is workable and will address the questions you raise. Rather than elaborate on the details in this e-mail, I propose that we discuss this as part of our conference call tomorrow.

Thanks,

Jim

E-Mail Exchange Number 3:

-----Original Message-----
From: Arjun Makhijani [mailto:arjun@ieer.org]
Sent: Wednesday, August 10, 2005 7:43 AM
To: Arjun Makhijani
Cc: Neton, Jim; Wade, Lewis; griffonm@comcast.net; Wanda Munn; James Melius; Michael Gibson; Gen Roessler; jmauro@scainc.com; kbehling@scainc.com; joyce; dunstana@globo.com; anigstein@verizon.net; Anigstein@cs.com; mikethorneltd; Denise Brock; Denise Brock; Elliott, Larry J.; Cindy Bloom and Steve LaMonica; joe guido; Allen, David [NIOSH]; pl.ziemer@insightbb.com
Subject: radon breath raw data

Hi Jim:

I thought that I should look at some of the raw radon breath data that you described on August 4 and yesterday. Could you please put in in my folder on the Secure FX drive where I have downloaded documents? Or send me sample data sheets, including the ones that show data gaps, in pdf format? Tomorrow will be fine. If they are located already in the database, could you point me to where they are?

Thanks so much.

Arjun
--

Jim Neton response

Arjun,

I'm not sure how you access our site research database, but on my computer there is a 30 MB file at:

O:/ORAU DOE SITE IMAGES/Mallinckrodt/Eml2BX8MallinckrodtBreathRadon49-56

This is a 451 page file that contains the original HASL datasheets.

Let me know if you have trouble locating this file.

Jim

--

[Subsequent to this e-mail and logistical follow up, NIOSH arranged for Jack Gibson to assist Arjun Makhijani in locating the file. Log of telephone conversation with Jack Gibson:

Arjun Makhijani phone call to Jack Gibson, 12 August 2005, about 9:40 a.m. Eastern Daylight Time, for assistance in locating radon breath raw data file. File was located.

E-Mail Exchange Number 4:

Note the summary of the conference call is Attachment 6 in this report.

-----Original Message-----
From: Arjun Makhijani [mailto:arjun@ieer.org]
Sent: Tuesday, August 09, 2005 6:44 PM
To: Neton, Jim
Cc: Wade, Lewis; griffonm@comcast.net; Wanda Munn; James Melius; Michael Gibson; Gen Roessler; jmauro@scainc.com; kbehling@scainc.com; joyce; dunstana@globo.com; anigstein@verizon.net; Anigstein@cs.com; mikethorneltd; Denise Brock; Denise Brock; Elliott, Larry J.; Cindy Bloom and Steve LaMonica; joe guido; Allen, David [NIOSH]; pl.ziemer@insightbb.com
Subject: Draft notes from conference call

Hi Jim:

Here are my draft notes. Some items are explicitly marked for your input, where I was not quite clear or things wnet [sic] y [sic]too fast for me to make notes. OIf [sic] course, please feel free to make other changes from your notes, if I did not get correctly what you are proposing.

Thanks

Arjun
--

Jim Neton response, 8/10/2005, 10:15 a.m.

Neton, Jim wrote:

Arjun,

Thanks for putting this together. In general, I think you've accurately captured the key issues and areas of agreement from our discussion yesterday. We have provided some additional text which we hope clarifies our position on a few issues.

Jim
--

-----Original Message-----
From: Arjun Makhijani [mailto:arjun@ieer.org]
Sent: Wednesday, August 10, 2005 12:18 PM
To: Neton, Jim
Cc: Wade, Lewis; griffonm@comcast.net; Wanda Munn; James Melius; Michael Gibson; Gen Roessler; jmauro@scainc.com; kbehling@scainc.com; joyce; dunstana@globo.com; anigstein@verizon.net; Anigstein@cs.com; mikethorneltd; Denise Brock; Denise Brock; Elliott, Larry J.; Cindy Bloom and Steve LaMonica; joe guido; Allen, David [NIOSH]; pl.ziemer@insightbb.com
Subject: Re: Draft notes from conference call

Hi Jim:

Thanks for providing clarifications on each of the points requested. I looked it over, corrected a typo, and accepted the edits. I am attaching this as the final summary of the conference call. Unless there are further changes from your side, I will put this in the SC&A report as the record of our call.

Of course, I have one question about the NIOSH position as now stated:

In the first numbered observation on ratios, you say that "the ratio of Pa-231 and Ac-227 to Th-230 is likely to be no more than 10 to 1 and quite possibly as low as 1 to 1". I am puzzled by this. This would give very large values for Pa-231 and Ac-227. Is this indicated by some

data? Could you please send that to me? Mass balance calculations and some K-65 silo data indicate that most Pa-231 and Ac-227 go with the Th-230 stream (that is the part in the K-65 at 6,000 pCi/g is ~0.25 of the total inventory). This indicates a Pa-231 to Th-230 ratio of 0.1 (not 10 to 1) so perhaps you meant 1 to 10 and not 10 to 1? But even with all this, I don't see where the 1 to 1 comes from. Since the Pa-231 and Ac-227 DCFs are generally higher than Th-230, this would seem

to be a crucial point.

Arjun

--

Jim Neton response, 8/10/2005, 12:36 p.m.

Arjun,

I'm sorry for the confusion. As you surmised, I really meant to state that the ratio was 1:10. Cindy Bloom is evaluating some recent data that reports lower ratios. As soon as she completes her evaluation we'll provide it in full.

Jim

E-Mail Exchange Number 5:

-----Original Message-----
From: Arjun Makhijani [mailto:arjun@ieer.org]
Sent: Wednesday, August 10, 2005 3:13 PM
To: Neton, Jim
Cc: Wade, Lewis; griffonm@comcast.net; Wanda Munn; James Melius; Michael Gibson; Gen Roessler; jmauro@scainc.com; kbehling@scainc.com; joyce; dunstana@globo.com; anigstein@verizon.net; Anigstein@cs.com; mikethorneltd; Denise Brock; Denise Brock; Elliott, Larry J.; Cindy Bloom and Steve LaMonica; joe guido; Allen, David [NIOSH]; pl.ziemer@insightbb.com
Subject: Re: Draft notes from conference call

Thanks, Jim.

I began laying out the SC&A report, and went over the six priority issues set forth by the Board today to make sure we had at least explicitly discussed NIOSH's approach to all the parts. In the process, I realized that there is one issue that has not been explicitly discussed, so I don't know how NIOSH is planning to deal with it. This is the issue of methodology for Plants 1 and 2 decommissioning workers. This is part of the un-monitored workers question. How is this going to be addressed, given that these workers are in a different category than secretarial, accounting, and other non-production office personnel who were not monitored?

Arjun

--

Jim Neton response, 8/11/2005, 10:43 a.m.

Arjun,

If there are individual uranium and/or breath radon monitoring data for a D&D worker, we will use it. If there are no individual monitoring data, we will use coworker data and assign the Plant 1 and 2 cleanup workers the 95th percentile intakes. These 95th percentiles would be applied using the two part methodology that was discussed on Tuesday. That is, we would reconstruct the dose from the radium source term using radon breath (actual or coworker) and then reconstruct the dose from the thorium source term using air data. The higher of the two results will be selected. A search of the CER database indicates that we have approximately 300 entries that are identified as cleanup or decontamination air samples. We would, of course, make sure that the 95th percentile of the plant air data is not inconsistent with the air concentrations measured at cleanup and decontamination locations.

As you know, the cleanup campaign for plants 1 and 2 was conducted over several years staring around 1949. A search of the active cases that we have for reconstruction during the SEC petition time period identified one worker as a potential D&D candidate. This person is identified as a "cleanup worker" and has uranium bioassay in 1950 and 1951.

Jim

E-Mail Exchange Number 6:

4:51 p.m., August 12, 2005

Neton, Jim wrote:

Arjun,

A draft document that summarizes our approach to handling raffinate ratios for operations in plant 6 is attached. After you've looked this over, we would be happy to participate in a conference call on Monday to discuss any areas that are in need of discussion.

<<MCW Plant 6 Approach.doc>>

Jim

--

Arjun Makhijani reply, 6:46 p.m., August 12, 2005

HI Jim

Thanks for this paper. I have taken a very quick look, and three questions occurred to me right off. I think they are quite straightforward.

1. It seems that for 1949, the Ra source is clearly the higher of the two, since the daily intake value is higher for each radionuclide listed. Hence there appears to be no further need for comparison. Is this your view? Your graph certainly shows that for the organs you evaluated.

2. The 95 percentile value for daily gross alpha intake in thorium areas appears to be low. If I take 607 dpm/m3, 10 m3 per day air intake (setting aside the issues regarding workday length, etc. for the moment), and 2.2 dpm/pCi, I get 2,649 pCi/day. I think the difference is with your 365 days per year being joined with only 2,400 m3 per year. If my calculation is right, it seems therefore that the 1950-1957 figures for the "Th source" column would need to be changed so the total of Th, Ac and Pa comes out to 2,649. Of course, I have performed no check on your 95 percentile value of 607 dpm/m3. This is just a response to your paper.

3. Are the dose numbers from an annual intake the committed dose? If not, what year do they represent?

If you are going to change this document in response to item 2, it would be useful to have it ASAP so I can include it in the Attachments. I am including the documents you've sent and our e-mail exchanges in Attachments, so the whole Board can see on what the SC&A review is based.

As to the Monday call, that is too late for SC&A to finish its work, write, review, typeset and send by 16 August. Can we not talk tomorrow, Saturday? I am resigned to working this weekend. I will be in most of the day so you can name a time. If anyone from the Working Group or Denise can join us, fine. If not, we can keep notes, and send them out very soon after the call by the process we adopted for the Aug. 9 notes. In any case we will keep notes as we did last time. It is my recollection that we were authorized to hold such a call that by the Board at the July meeting. Unfortunately, there is no transcript so I cannot check. Is that your view also? Board members, please comment. John Mauro, please comment.

Thank you very much.

Arjun

E-Mail Exchange Number 7:

-------- Original Message -------- Subject: RE: Raffinate Ratios

Date: Mon, 15 Aug 2005 07:55:23 -0400

From: "Neton, Jim" <jfn2@cdc.gov>

To: "Arjun Makhijani" <arjun@ieer.org>

CC: "Wade, Lewis" <LOW0@CDC.GOV>, <griffonm@comcast.net>, "Wanda Munn" <wimunn@aol.com>, "James Melius" <melius@nysliuna.org>, "Michael Gibson" <Mikehgibson@cinci.rr.com>, "Gen Roessler" <gnrsslr@frontiernet.net>, <jmauro@scainc.com>, <kbehling@scainc.com>, "joyce" <joycelip@terra.com.br>, <dunstana@globo.com>, <anigstein@verizon.net>, <Anigstein@cs.com>, "mikethorneltd" <mikethorneltd@aol.com>, "Denise Brock" <dbrock@alwayson-line.net>, "Denise Brock" <db_dcch@hotmail.com>, "Elliott, Larry J." <LJE1@CDC.GOV>, "Cindy Bloom and Steve LaMonica" <radbloom@comcast.net>, "joe guido" <jguido@woh.rr.com>, "Allen, David [NIOSH]" <DKA6@CDC.GOV>, <pl.ziemer@insightbb.com>, "Katz, Ted" <TMK1@CDC.GOV>

Arjun,

Our responses are interspersed among your questions.

-----Original Message-----
From: Arjun Makhijani [mailto:arjun@ieer.org]
Sent: Friday, August 12, 2005 6:46 PM
To: Neton, Jim
Cc: Wade, Lewis; griffonm@comcast.net; Wanda Munn; James Melius; Michael Gibson; Gen Roessler; jmauro@scainc.com; kbehling@scainc.com; joyce; dunstana@globo.com; anigstein@verizon.net; Anigstein@cs.com; mikethorneltd; Denise Brock; Denise Brock; Elliott, Larry J.; Cindy Bloom and Steve LaMonica; joe guido; Allen, David [NIOSH]; pl.ziemer@insightbb.com; Katz, Ted
Subject: Re: Raffinate Ratios

HI Jim

Thanks for this paper. I have taken a very quick look, and three questions occurred to me right off. I think they are quite straightforward.

1. It seems that for 1949, the Ra source is clearly the higher of the two, since the daily intake value is higher for each radionuclide listed. Hence there appears to be no further need for comparison. Is this your view? Your graph certainly shows that for the organs you evaluated.

We agree with your conclusion.

2. The 95 percentile value for daily gross alpha intake in thorium areas appears to be low. If I take 607 dpm/m3, 10 m3 per day air intake (setting aside the issues regarding workday length, etc. for the moment), and 2.2 dpm/pCi, I get 2,649 pCi/day. I think the difference is with your 365 days per year being joined with only 2,400 m3 per year. If my calculation is right, it seems therefore that the 1950-1957 figures for the "Th source" column would need to be changed so the total of Th, Ac and Pa comes out to 2,649. Of course, I have performed no check on your 95 percentile value of 607 dpm/m3. This is just a response to your paper.

To model chronic intakes in IMBA one must put in a continuous daily intake rate for the year. There is no allowance made in the software for breaks in exposure, such as would occur over the weekend. To accommodate this, all intakes are input as the average intake per calendar day. The important consideration is that the total intake for the year be the same (i.e. 250*2649=6.6E5 pCi and 365*1798=6.6E5 pCi).

3. Are the dose numbers from an annual intake the committed dose? If not, what year do they represent?

Yes, as stated in the text, these are 50 year doses. We should have also noted this on the graph as well.

If you are going to change this document in response to item 2, it would be useful to have it ASAP so I can include it in the Attachments. I am including the documents you've sent and our e-mail exchanges in Attachments, so the whole Board can see on what the SC&A review is based.

As indicated above, we believe that our annual intake for the raffinate source term is OK.

As to the Monday call, that is too late for SC&A to finish its work, write, review, typeset and send by 16 August. Can we not talk tomorrow, Saturday? I am resigned to working this weekend. I will be in most of the day so you can name a time. If anyone from the Working Group or Denise can join us, fine. If not, we can keep notes, and send them out very soon after the call by the process we adopted for the Aug. 9 notes. In any case we will keep notes as we did last time. It is my recollection that we were authorized to hold such a call that by the Board at the July meeting. Unfortunately, there is no transcript so I cannot check. Is that your view also? Board members, please comment. John Mauro, please comment.

Thank you very much.

Arjun

Neton, Jim wrote:

Arjun,

A draft document that summarizes our approach to handling raffinate ratios for operations in plant 6 is attached. After you've looked this over, we would be happy to participate in a conference call on Monday to discuss any areas that are in need of discussion.

<<MCW Plant 6 Approach.doc>>

Jim

ATTACHMENT 6

Conference call August 9, 2005 between NIOSH and SC&A

Subject of the call: Mallinckrodt issues to be resolved as per the August 4, 2005 meeting in Cincinnati, Ohio of the Board Working Group, NIOSH and its contractors, SC&A, and the Petitioner.

Final summary
10 August 2005
Draft prepared by Arjun Makhijani, SC&A. Responses to queries and clarifications on NIOSH positions provided by Jim Neton, NIOSH. Drafts circulated to Board Working Group and Petitioner

Date: August 9, 2005

Present on the call: Jim Neton, Cindy Bloom, Joe Guido, Dave Allen, John Mauro, Arjun Makhijani, Bob Anigstein, Dunstana Melo, Joyce Lipsztein, and Denise Brock.

Issue 1: Ratios of Th-230, Pa-231, and Ac-227 to uranium and or radium-226

The question of how to set the ratio of Th-230 to uranium or radium-226 was extensively discussed.

In the areas of AM-7 residues processing and of processing of residues beyond the step where almost all radium and uranium are removed, Th-230 would be expected to be the main radionuclide in terms of activity.

Observations regarding the issue:

1. The radionuclide ratios of Th-230 to total uranium based on residue analysis data or materials balance are in on the order of ~100 to 1 (different ratios were from 70 to 1 to about 180 to 1 were mentioned). There would be corresponding values for Pa-231 and Ac-227 based on residue source terms other than K-65 and $BaSO_4$. NIOSH's current research indicates that the ratio of Pa-231 and Ac-227 to Th-230 is likely to be no more than 10 to 1 and quite possibly as low as 1 to 1. NIOSH does not believe that the thorium residues contained significant quantities of Pb-210, as this would be removed in the Pb and Ba precipitation steps. NIOSH is currently considering the fraction of Po-210 that may be present in the thorium bearing residues.
2. Since workers moved from one job to the next, such high ratios would be unlikely to be experienced full time by any worker. It was also noted by NIOSH that these high ratios would not be consistent with the air dust studies that were conducted in plant 6.
3. It will be difficult to establish the times for which workers worked in various areas.

4. The Th-230 to U ratio of ~100 to 1 in the residue could be used only as an upper limit determinant just as a check, but not as a multiplier from the uranium bioassay data to estimate doses. This is because there is little uranium in the high Th-230 areas and hence the uranium in the bioassay is not expected to come mainly or largely from these areas. Further, the airborne activity measurements do not support such large Th intakes. Hence the Th-230 to U ratio is not useful for dose reconstruction in high Th-230 areas.

Two different approaches to estimating doses for all workers other than those known only to have worked with uranium free of decay products (Th-230 and below) were discussed:

1. NIOSH will use radium-226 intake estimates from radon breath data and use the ratios of Th-230/Ra-226, and Ac-227/Ra-226, Pb-210/Ra-226 and Po-210/Ra-226 from the Fernald K-65 silo data. (Pa-231/Ra-226 would be the same as Ac-227/Ra-226.) 95 percentile intake values would be used for production workers without radon breath data and mean values for unmonitored workers. Internal dose would be estimated for each of these radionuclides. Uranium dose would be estimated from bioassay.
2. 95 percentile values for time-weighted air concentration data would be estimated for the AM-7 residue processing areas.. . These would be parsed according to the radionuclide mix that would be expected to be present in these areas consisting of Th-230, Pa-231, Ac-227, and Po-210. . Uranium dose from bioassay would be added. It was not clear if radium-226 dose from radon breath would be added as well.

NIOSH proposes to compare these two approaches and use the higher of the two values for dose reconstruction. If one of them is systematically higher than the other in all cases, then it would be the only one used.

Action item: NIOSH will send a specific write up on this approach to SC&A and the rest of the e-mail list (Dr. Zeimer, the Board Working Group, and Denise Brock) on August 10, 2005.

Issue 2: Radon dose

It is claimant favorable to assume a radon breath and radium body burden value for workers only exposed to radon gas, like delidders and warehouse workers?

NIOSH proposes to assign radium intakes based on coworker radon breath measurements to workers who had no radon breath data, but who worked in areas elevated in radon. The median coworker value for radon breath of approximately 0.3 picocuries per liter would be used in the calculation. This would result somewhere in the vicinity to 2.5 rem to the least exposed organ. It corresponds to a continuous exposure to 3000 picocuries per liter of radon. So radon breath gives a higher value of dose for workers who had no radium intake but are being assigned one, instead of being assigned a radon exposure.

SC&A seemed to be in qualitative accord with this for the moment, pending a write-up and review by the SC&A tem.

Action item: NIOSH will send a write up early on Aug. 10, 2005.

Issue 3: Environmental exposure

NIOSH is working on it. There were releases of thousands of pounds per year. There was discussion of estimating exposures from an accident or incident. There was no resolution regarding what dispersion coefficients might be appropriate since site specific data are not available. It was suggested that for accidental releases in the vicinity of buildings they might be quite high. For the moment, the use of non-site specific data was thought by SC&A team members to be inappropriate given the crucial importance of site specific factors.

Denise Brock said she has environmental monitoring data (dates not known at the time of the call) that she would send NIOSH.

Action item: NIOSH will send some written material on this by close-of-business, of 10th August.

Issue 4: Missing radon breath data

NIOSH has examined almost all the raw data. There are periods in the 1950s where there are more lost data or an entire set of samples was not analyzed. There was a report on instrument malfunctions. Jim said he did not find anything definitive on the causes of the missing data but that the job categories do not appear to be skewed and that there did not appear to be systematic censoring of data.

Action item: NIOSH will get something written out on Aug. 10th.

General:

There was a discussion at the beginning of the call about the use of this new approach being developed after the July Board meeting to estimate doses, following a question about that raised by Denise Brock. Jim Neton explained that this new approach is by way of a best estimate with some maximizing aspects where NIOSH cannot do better (as distinct for the efficiency approaches to maximum dose using the Hanford radionuclide list, for instance). The new approach, if adopted, would be used to prepare dose estimates that would be used by DOL to compensate or deny, according to PoC greater than or less than 50 percent. The approaches that resulted in prior denials would be revisited to see if the new approach has changed things.

ATTACHMENT 7a

August 10, 2005
DRAFT
Radon in Breath due to Radon Gas Inhalation vs. Radium Generated Radon
Evaluation of Dose Implications

Background

Radium-226 decays to the inert gas Radon-222. Some fraction of the Rn-222 produced in the body from the decay of internally deposited Ra-226 will be exhaled. This is the basis for the radon breath measurements that have been used to assess the radium body content.

Exposures to radium, however, are almost always accompanied by exposures to environmental levels of radon gas and radon progeny. As an inert gas, environmental levels of Rn-222 can become incorporated into the body tissues. When an individual is removed from the radon environment, this diffused radon is exhaled over a period of time. It is not possible to distinguish how much of the radon in a breath sample originated from internally deposited radium versus radon gas that is saturated in body tissues from inhalation of the gas. The purpose of this analysis is to explore which assumption would result in the highest dose to organs other than the respiratory tract.

Radium Assessment

The method for assessing radium body burden using radon breath analysis is explained in OTIB-25. The OTIB results in a conversion factor of 25200 pCi of radium in the body for every 0.1 pCi/L radon-222 in the breath.

This body burden of radium can be established at a particular point in time from an individual breath analysis. From that information, the intake of radium is established by fitting the radium whole body content using the current ICRP models for radium.

Radon Breath Concentration

The radon concentration in exhaled air after removal from a radon gas environment is based on the work of Harley. In Harley's experiment, a subject was exposed to 700 pCi/L of radon gas in a chamber for 8.5 hours. This was considered sufficient time to allow the radon gas to reach a saturation condition with the tissues of the body. After the exposure, the radon in breath was measured periodically for the next 72 hours. Those data were fit with a 5 term exponential equation. This equation and parameters were used to describe the radon exhaled versus time after removal from a radon environment.

Radon Dose

The dose from inhalation of radon gas is based on radon gas saturation with body tissues. The values for this analysis are from ICRP 32 and result in a dose of 3.33E-4 mrem/hr per pCi/L air concentration.

The dose from radon progeny to organs other than the respiratory tract is based on current ICRP models for those progeny. The lung absorption rate is assumed to have a 10 hour half-life for lead and a 13 hour half life for bismuth.

When the dose from radon gas and from its progeny are combined, the annual dose from inhaling a 1 pCi/L concentration for 2000 hours per year is approximately 2 mrem for the kidneys, 1 mrem for the stomach, 0.85 mrem for the small intestine, 0.74 mrem for the upper large intestine and 0.69 for all other tissue.

Analysis

As a starting point for this analysis, a 1 pCi/L (2220 dpm/m^3) radium-226 intake is assumed. This intake is assumed to be continuous for 2000 hours per year with a breathing rate of 20 lpm. This results in an intake rate of 6575 pCi/calendar day. A chronic exposure to this intake rate results in a body burden of 6.34E4 pCi one year after the beginning of the intake and approximately double that (1.3E5 pCi), 10 years after the intake begins. These radium body burdens will result in a radon breath concentration of approximately 0.25 pCi/L and 0.5 pCi/L respectively.

The effective dose to various organs from 1 year of exposure to the above intake rate was calculated. The effective dose includes dose from additional isotopes based on an activity ratio with radium. The isotopes used and the activity ratios are shown in Table 1. The doses for a number of organs are provided in Table 2.

Table 1. Activity Ratio of Associated Isotopes to Ra-226

	Th-230	Po-210	Pa-231	Ac-227
Activity ratio	0.125	0.5	0.013	0.013

The radon air concentration necessary to obtain the same dose was calculated for each of the selected organs using the dose conversion factors described earlier. These values are also provided in Table 2. The breath radon concentration during this radon inhalation and two days after the end of exposure are also included in the table. The two days was chosen as a weekend.

Table 2

	Dose from 1 pCi/L Ra-226 and associated isotopes for 2000 hours (mrem)	Air concentration of Rn-222 to equal Ra-226 dose (pCi/L)	Breath concentration of Rn-222 during Radon exposure (pCi/L)	Breath concentration of Rn-222 two days following radon exposure (pCi/L)
Adrenals	2433	3526	1997	0.44
Urinary Bladder	2423	3512	1989	0.44
Brain	2428	3518	1993	0.44
Breast	2422	3511	1988	0.44
Gall Bladder	2428	3519	1993	0.44
Heart Wall	2426	3515	1991	0.44
Kidneys	36807	18403	10424	2.29
Liver	137112	198713	112551	24.71
Muscle	2426	3516	1991	0.44
Ovaries	20508	29722	16834	3.70
Pancreas	2428	3518	1993	0.44
Testes	20801	30147	17075	3.75
Thyroid	2424	3513	1990	0.44
R.B.M.	146330	212072	120117	26.37
Bone Surface	2820160	4087188	2314971	508.25
Stomach	2431	2431	1377	0.30
S.I.	2455	2888	1636	0.36
U.L.I.	2639	3518	2020	0.44
L.L.I.	3170	4594	2602	0.57
Skin	2423	3512	1989	0.44
Spleen	10854	15730	8909	1.96
Thymus	2423	3512	1989	0.44
Uterus	2425	3514	1990	0.44
Adrenals	2433	3526	1997	0.44

Summary

The Ra-226 and associated progeny doses in Table 2 represent an intake of radium that would result in a radon breath concentration of 0.25 pCi/L at the end of one year of exposure. An equivalent dose from radon would result in higher radon breath concentrations. This can be seen in the last two columns of Table 2. The assumption that any radon in a breath sample is solely due to Ra-226 would produce higher doses. It would therefore be favorable to ignore the radon inhalation doses when evaluating radon breath measurements.

NIOSH has indicated that it will assign the median value of the radon breath distribution to workers who were not monitored and worked in Plant 6. In this case, a worker who breathed only radon gas would be assumed to have a breath concentration of approximately 0.33 pCi/L, which is slightly higher than the 0.25 pCi/L value used to generate the doses in Table 2. From examination of Table 2, it can be seen that the lowest dose to an individual organ is

2,423 mrem which would correspond to the continuous inhalation of radon gas at a concentration of 3,500 pCi/L.

Example

The values above were based on a starting point of 1 pCi/L Ra-226 and an assumed 1 year intake. In order to produce a more realistic example, data from a MCW employee was used. Case 1 presented to the working group of the Advisory Board on 8/4/2005 was used for this purpose. In that example case, the Ra-226 body burden was determined from radon breath analysis.

For this example, it is assumed the individual was also continuously exposed to a radon gas concentration of 100 pCi/L. Also, since all the breath samples were obtained on Mondays, it is assumed they were collected after two days absence from exposure. If the radon gas exposure is considered, some fraction of the radon concentration in a breath sample would have to be subtracted and the remainder attributed to Ra-226 exposure. This causes the Ra-226 whole body content estimate to be lower than previously estimated. In the original example given to the working group, it was assumed that all the radon in breath was attributed to Ra-226 exposure and no exposure to radon gas was assumed. The radium intake was modeled as one chronic and one acute intake. The same intake regime was assumed for this analysis in order to provide a consistent comparison.

Once the radon inhalation is considered, the new chronic intake estimate is 93.9% of the original chronic intake estimate. The new acute intake estimate is 92.4% of the original acute intake estimate. Since the associated isotopes are a ratio of Ra-226 intakes, the intake estimates for these isotopes also decreases by the same percentages. The uranium intakes were left unchanged since they were determined from uranium bioassay samples. Table 3 summarizes the original dose to several organs as well as the doses adjusted for a 100 pCi/L radon exposure.

Table 3. Case 1 Doses Adjusted for 100 pCi/L Radon Exposure

	New dose (rem)	Radon dose (rem)	New + Radon (rem)	Original (rem)
Colon	1.13E+01	0.532	1.18E+01	1.20E+01
HNM	1.66E+01	0.511	1.71E+01	1.77E+01
Liver	2.22E+03	0.511	2.22E+03	2.37E+03
Bone Surf	1.62E+04	0.511	1.62E+04	1.73E+04
Kidney	1.62E+02	1.482	1.63E+02	1.72E+02

Estimate assumes samples taken after two days absence from radon exposure

There is a decrease in the overall dose to each organ if the radon inhalation is accounted for in the radon breath sample. The increase in dose caused by the radon inhalation makes up only a fraction of the decrease. It is therefore more favorable to assume the radon breath concentration is caused solely by radium body burden by ignore any dose caused by radon inhalation.

ATTACHMENT 7b

Analysis of Lost or Not Analyzed Radon Breath Samples

Introduction

A concern was expressed at the last Advisory Board meeting regarding the number of radon breath samples with results that indicated a sample collected by MCW and submitted to HASL was not analyzed. A review of the data files indicated that, for certain time periods, approximately 25% of the radon breath samples were listed as either being "Lost" or "Not Analyzed." This high percentage raises some concern that the radon breath distributions could be biased in some way. NIOSH has examined the data to determine what impact, if any, the missing data might have on their usefulness in dose reconstruction.

Evaluation

A detailed analysis of 41 "Not Analyzed" samples from August 4, 1952 through August 18, 1952 was conducted. This sample set comprises 8% of the total number of radon breath samples taken in 1952 and 32% of the "Lost" or Not Analyzed samples in 1952. The analysis consisted of evaluating whether or not the individual workers with a "Not Analyzed" sample had a subsequent re-sample (second breath radon sample) taken over the next 12 month time period. During this evaluation, it was found that approximately 40% of the workers had a re-sample (with a valid result) within 3 months and virtually all (> 95%) had a subsequent radon breath sample taken (with a valid result) within the next year. From this detailed review only 1 of the 41 workers did not have a subsequent sample within the next year. The reason for this missing person is not known and could be due to termination of employment or relocation to a different job within the plant. Figure 1 below depicts the percentage of workers with at least one resample within the 3, 6, 9 and 12 month intervals.

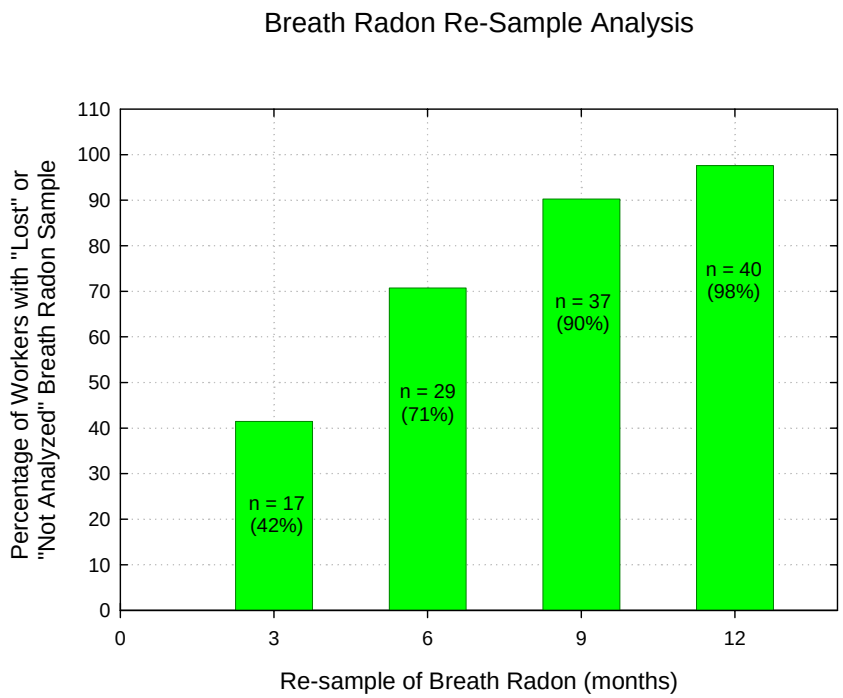


Figure 1: Percentage of workers with follow-up Breath Radon Sample

It is important to note that some workers had more than one additional sample taken as they appeared to be on routine analysis. The maximum number of additional valid samples taken within this work group was 4 in the 12-month follow-up period.

During this review it was discovered that the most frequently sampled persons (every 2-3 months) occasionally had an additional lost sample. Thus if the statistic holds that 1 out of 4 samples was “Lost or “Not Analyzed” one could expect 1 -2 samples per year would be lost for the most frequently monitored personnel.

Another potential concern is that these worker’s exposures could have been a unique group of very high exposures or an incident. In this evaluation over 80% of the workers had some form of Job/Dept information. In a review of the available job/department information for these workers, it appears that these samples were part of the routine monitoring protocol and that there was likely a problem with the instrumentation or in the shipping of the samples during this time period. This conclusion is drawn based upon the random sampling which closely matches that generally sampling job / Dept information for any given sample set. Table 1 lists the job / Dept titles for each of the 41 workers in this analysis.

Conclusions

Based upon this review, the following conclusions can be drawn:

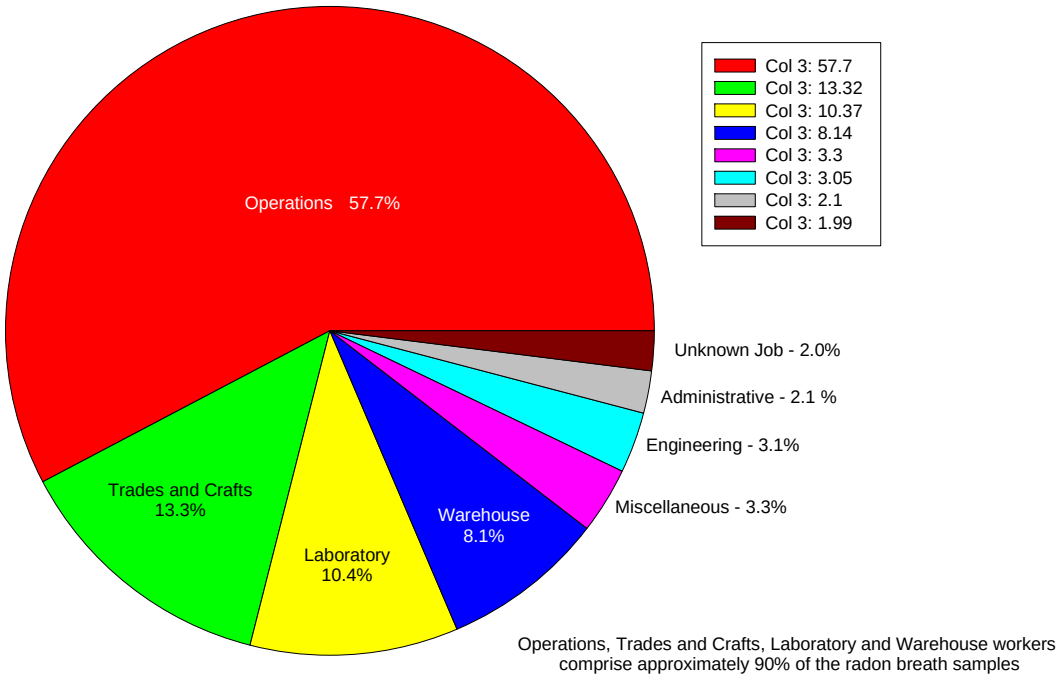
- Not analyzed of lost samples is not unique to a specific group of employees or workers at MCW.
- Approximately 40% of the workers were re-sampled within 3 months.
- Virtually all (>95%) have a re-sampled result within 1 year

Table 1. Breakdown of Job / Department and Resample Duration

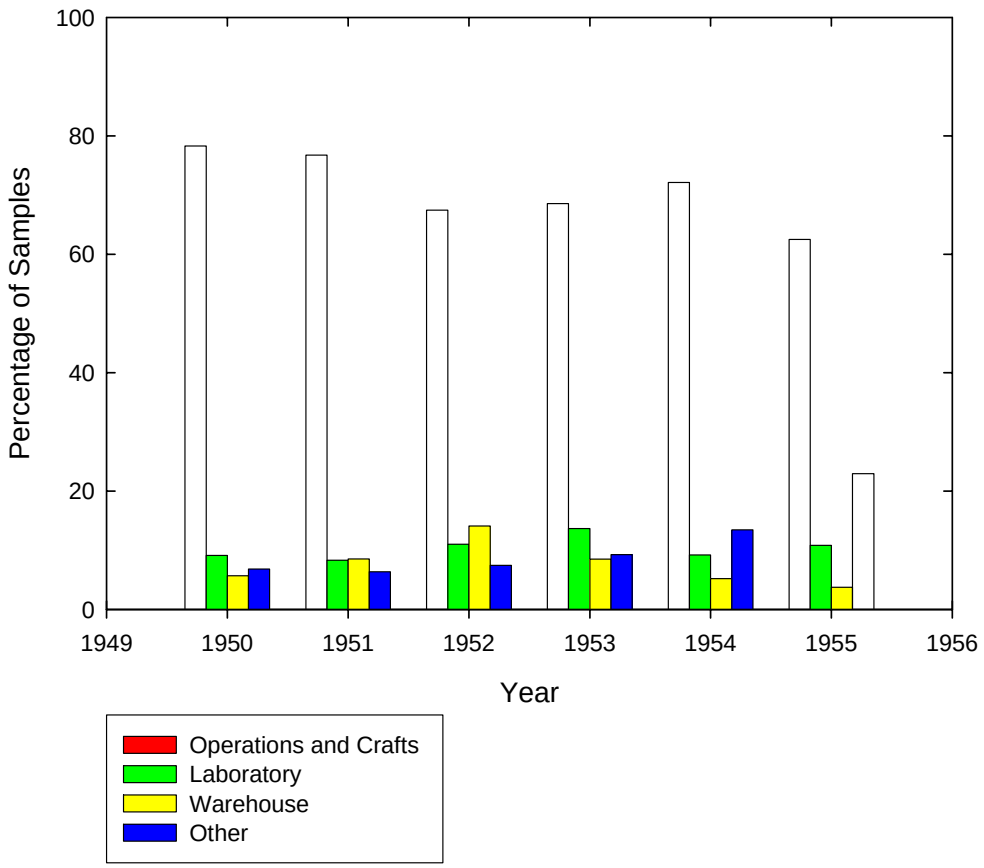
#	Job / Department	3 months	6 months	9 Months	1 year
1	Engineer Plant 6	1	1	1	1
2	Laboratory			1	1
3	Laboratory	1	1	1	1
4	Laboratory				1
5	Maintenance		1	1	1
6	Mfg - Plant 6		1	1	1
7	Pilot Plant	1	1	1	1
8	Pilot Plant 6	1	1	1	1
9	Pilot Plant 6	1	1	1	1
10	Plant 6		1	1	1
11	Plant 6	1	1	1	1
12	Plant 6		1	1	1
13	Plant 6		1	1	1
14	Plant 6			1	1
15	Plant 6	1	1	1	1
16	Plant 6			1	1
17	Plant 6			1	1
18	Plant 6		1	1	1
19	Plant 6				1
20	Plant 6			1	1
21	Plant 6	1	1	1	1
22	Plant 6		1	1	1
23	Plant 6				1
24	Plant 6		1	1	1
25	Plant 6			1	1
26	Plant 6	1	1	1	1
27	Plant 6	1	1	1	1
28	Plant 6	1	1	1	1
29	Production Office Plant 6			1	1
30	Production Office Plant 6	1	1	1	1
31	Production Office Plant 6			1	1
32	Warehouse		1	1	1
33	Warehouse	1	1	1	1
34	Unknown	1	1	1	1
35	Unknown		1	1	1
36	Unknown		1	1	1
37	Unknown	1	1	1	1
38	Unknown	1	1	1	1
39	Unknown				
40	Unknown	1	1	1	1
41	Unknown		1	1	1
	Number	**17**	**29**	**37**	**40**
	Percentage (%)	**41.5**	**70.7**	**90.2**	**97.6**

ATTACHMENT 7C

Collapsed Job Catagories of Breath Radon Data (1950-1955)



Radon Breath Data by Collapsed Job Category



Other includes the following catagories: Engineering, Administrative, Miscellaneous and Unknown

Job / Department Collapsed Categories - Sampling

Operations

Barium	Etherhouse	MGX
Breakout	Feed	Ore Room
Brushing	FEINC	Pilot Plant
Burning	FTO	Plant 6
C-3	Furnace	Plant 7
Chemical Operator	N A House	Pot Room
Cloth Room	MAG Room	Powerhouse
Digest	MFG - Manufacturing	Raffinate
Decon	M-20	Recovery

Trades and Crafts

Carpenter	Mechanic	Rigger
Electrician	Millwright	Shop Grinder
Foreman	Painter	Tinner
Insulator	Pipefitter	Welder
Machinist	Porter	
Maintenance	PFT Shop	

Laboratory

Lab	Chemist	Chem. Lab
Research Lab	Tech Lab	Con. Lab
K-65 Lab	Ladoux Lab	MX Lab
MY Lab	BA Lab	

Warehouse

Warehouse	Yard	Store Room
FTO Warehouse		

Administrative

Accounting	Office (not specified)	Printer
Drafting		

Engineering

Engineer	Office (Production)	Office (Engineering)
Mech. Engineer	Process Development	

Miscellaneous

Auto Garage	Checkers Lot	Receiving
X-ray Lab	Guards	Mechanic Garage
Laundry	Window Washer	Truck driver
Lab Truck	Packing	Health
Utility	Safety	Fork Truck Operator
Shipping	Receiving	

ATTACHMENT 7d

Review of Information Related to Exposures Outside of MCW Process Areas

NIOSH has proposed that the full distribution of production worker intakes be applied to unmonitored personnel who were not involved in production activities. A concern was raised by SC&A that workers outside the radioactive material process areas at Mallinckrodt might have higher internal exposures than the median exposed production worker. After 1949, the median uranium (type M) intake rate for Mallinckrodt process workers was about 220 dpm/calendar day, which equates to an air concentration of 34 dpm/m^3.

The atmospheric releases of uranium at the Detrehan Street Site were summarized in a letter from Shelley to the AEC. This letter, which is provided as an attached pdf file, reports releases from 1946 to 1955 for the refinery, green salt plant and metal plant. The maximum reported uranium discharge in any one year from any one plant is 2,800 pounds, and the maximum reported summed discharge in any one year is 3,630 pounds. Although sufficient data are not available to generate dispersion models to predict the magnitude of air concentrations associated with these releases, NIOSH does have access to air sampling data that suggests the assignment of median inhalation intakes of 220 dpm/calendar day to non-process workers is reasonable.

NIOSH reviewed the Mallinckrodt workplace air studies for outdoor air concentration measurements. Outdoor data were in found in air studies conducted during the years 1948, 1949, 1950, 1952, 1953, 1955, 1956, and 1958. A summary of these data are provided in Table 1. The maximum reported average alpha concentration was 32 dpm/m^3. Ninety-five percent of the samples were less than16 dpm/m^3 and most were less than 3 dpm/m^3. A 1949 emissions study by Harris reported a maximum uranium air concentration of 13 ug/m^3. Most results were less than 1 ug/m^3 at 1000 feet from Plant 4, and 1000 feet from Plant 6. The study indicated that some of the higher concentrations probably reflected sample contamination, and it was also noted the atmospheric conditions were such that these results might be biased on the low side. (Note: 1 ug is about 1.5 dpm.)

The estimated uranium median intake is about 220 dpm/day for type M uranium, which equates to a uranium air concentration during work hours of 34 dpm/m^3. The maximum reported average outdoor gross alpha concentration was 32 dpm/m^3, but most outdoor concentrations were less than 3 dpm/m^3. These values support the use of process worker uranium intake data as sufficiently bounding for non-process workers.

Table 1. Summary of Mallinckrodt Outdoor Air Monitoring Data

Plant	Location	Avg dpm/m3	No. samples	Date of samples
6	Outside areas	14	10	Oct-Nov 1948
4	Outside area (for guard DWE calculation)	16	1	Dec 1949
6	Substation	32	5	Aug 1950
6	Guard towers	8.9	3	Aug 1950
6	Yard	15	6	Aug 1950
6E	Guard towers	9	3	Sep 1950
6	Substation	7.3	2	May 1952
6	Guard towers	10.7	2	May 1952
6	Yard	14.6	6	May 1952
6	Yard	0.8	12	Jan 1953
6	Guard towers	3.4	4	Jan 1953
6	Yard	11	20	Sep 1955
7W	Outside areas East and West	1.9	8	Sep 1955
4	Yard	4	6	May 1956
4	North yard	1.4	3	Jan-Feb 1958
4	South yard	1.8	3	Jan-Feb 1958
6	East side of Pickler Building	1.7	4	Jan-Feb 1958
6	East side of Plant 6	1.2	4	Jan-Feb 1958
6E	West dock	0.9	4	Jan-Feb 1958
6E	North side of scale room	1.4	4	Jan-Feb 1958
6E	Alley between Slag Bldg and substation	1.5	4	Jan-Feb 1958
7	West dock (outside)	3.6	4	Jan-Feb 1958
7	South dock (outside)	2	4	Jan-Feb 1958
7	West HF vaults (outside)	0.6	4	Jan-Feb 1958
7W	West end	1.2	4	Jan-Feb 1958
7W	East end	2.7	4	Jan-Feb 1958
---	Boiler house, southeast steps	2.2	4	Jan-Feb 1958
---	Boiler house, sampling dock	2.2	4	Jan-Feb 1958
6	Warehouse, office	14	4	Jan-Feb 1958
6	Warehouse, smoking area	6.7	4	Jan-Feb 1958
6	Warehouse, SW bay	1.1	4	Jan-Feb 1958
6	Warehouse, NE bay	0.4	4	Jan-Feb 1958
6	Warehouse, N dock	0.7	4	Jan-Feb 1958
6	Warehouse, SW dock	1.5	4	Jan-Feb 1958
	Average	**6.6**	**166**	

ATTACHMENT 7e

February 15, 1956

Mr. F. H. Belcher
Area Manager
U. S. Atomic Energy Commission
P. O. Box 470
St. Charles, Missouri

SUBJECT: Uranium Discharged to Atmosphere

Dear Mr. Belcher:

Please refer to your letter on the above subject dated February 3, requesting the submission of certain data for use by the New York Health Group.

We have examined our records and have attached herewith our best estimates of such discharge. These estimates are those made by the Health Group as a result of intermittent studies. It should be understood that such estimates are not the result of a carefully controlled long-range program of determing this discharge and that the application of such estimates may result in considerable error.

We have made tentative plans to install permanent samplers on five of our stacks, but as yet this program has not begun.

If there is further or more detailed information you desire, please let me know.

Sincerely yours,

MALLINCKRODT CHEMICAL WORKS
URANIUM DIVISION

W. J. Shelley

WJS:lmh

Attachment

cc. Mr Mason

Ok - WJL

UNCLASSIFIED
W. J. Lemmon WJL
Critique 11/15/04
Date:
2 Pages

URANIUM DISCHARGED TO ATMOSPHERE
LBS. PER YEAR

YEAR	REFINERY	GREEN SALT	METAL PLANT
1946	2,800		
1947	2,800		
1948	1,700		
1949	1,590		
1950	1,400		520
1951	1,530	80	520
1952	1,460	100	520
1953	1,250	150	520
1954	1,270	150	1,000
1955	1,200	150	2,280

ATTACHMENT 7f

Intake Determination from Radioactive Particulate – Mallinckrodt Plant 6 Process Workers and Related Workers (Waste Handlers, Ledoux Laboratory Workers, etc.)

Overview

The following summarizes the approach to assigning intakes and dose to process workers associated with Plant 6 source terms. Although all workers are assigned uranium intakes, the source terms considered when estimating intakes from progeny are:

- Uranium in equilibrium with progeny;
- Radium residue source term (based primarily on K-65 waste in Silo 1); and,
- Thorium residue source term (based on airport cake, including consideration of AM-7 and Sperry cake residues).

The information that follows provides a brief discussion regarding parameters related to reconstructing uranium, radium and thorium intakes based on availability of individual bioassay. This is followed by analysis of the source term resulting in worker internal exposures versus the source term in the residues. Finally, information on assigning intakes when there are no individual bioassay and information on worker categorization is provided.

Uranium

Uranium intakes are determined from bioassay when available or assigned at the 95th percentile of the coworker distribution when bioassay data are not available. Uranium is assumed to be type M.

Thorium or Radium Source Term Intakes

Because the ability to accurately determine the fraction of time a worker spent in any given part of the process area can be limited, a simplified approach to assigning intakes from uranium progeny has been developed. This approach begins with the premise that a worker at any given time can be exposed to uranium source terms that are either enriched in radium or enriched in thorium. Because of the uncertainty in source term exposure over time (i.e., radium enriched or thorium enriched), uranium progeny exposure is assigned based on the highest dose delivered by either of the established source terms. For Mallinckrodt, this reconstruction approach is considered a best estimate of dose, although it is clear that in many cases this may be an overestimate of organ doses for individual workers.

The following provides information on the approach to reconstructing each of the exposure scenarios. As indicated above, the dose will be reconstructed for each source term and the value that provides the highest probability of causation will be used.

Thorium Source Term and Intake

Thorium intakes are assumed to be type M.

Thorium residue (i.e., AM-7 material) intakes are estimated using the 95th percentile of the time-weighted average gross alpha air concentrations in all Plant 6 process operations. The 95th percentile air concentration corresponds to 607 dpm/m^3 or approximately 9 times the maximum allowable concentration of 70 dpm/m^3. The area air concentrations used in this analysis are about a factor of two higher than the measured air concentrations/exposures in areas associated with the AM-7 raffinate.

From the air concentration data, a daily gross alpha intake rate of 1798 pCi/day is estimated by multiplying by an air intake of 2400 m3/y, dividing by 365 days per year and 2.22 dpm/pCi. The alpha activity is apportioned according to the source term concentrations of the alpha emitters in the raffinates, and the Ac-227 is assumed to equal the Pa-231 on an activity basis.

Although there are thorium bioassay data for workers in 1955, these don't directly apply to uranium process workers. It should be noted, however, that for some individual workers the thorium bioassay might be used to assign intakes on a case-by-case base.

Radium

Radium residue (i.e., K-65) intakes are determined from radon breath bioassay data when available or assigned at the 95th percentile of coworker data when bioassay data are not available. Radium is assigned as type M. Progeny exposures are then assigned based on the ratios established for K-65 residues.

Choosing and Assigning Non-Uranium Intakes

As noted above, workers can be exposed to uranium, thorium or uranium or a combination of these source terms over the course of a year. This section provides the guidance for choosing which of these source terms to apply based on consideration of available worker monitoring data.

To simplify the discussion, an example is included for and unmonitored worker. For the monitored worker, calculated intake rates would be substituted for coworker intake rates, and associated intake ratios would be estimated based on these calculated intake rates. The choice of scenario is made based on the result which provides the largest probability of causation.

Table 1 shows the summary of activity fractions relative to uranium, thorium or uranium. Fractions with gray backgrounds are not used, but are shown for completeness.

Table 1. Activity Fractions Relative to Uranium, Thorium or Radium

Source	Type	Uranium in equilibrium	Thorium enriched		Radium enriched	
Uranium	M	1	1	0.0134	1	0.00352
Th-230	M	0.489	74.9	1	42.4	0.149
Ra-226	M	0.489	-		284	1
Pb-210	F	0.489	-		120	0.423
Po-210	M	0.489	-		167	0.589
Pa-231	M	0.0228	10	0.134	4.57	0.0161
Ac-227	F, M	0.0228	10	0.134	4.57	0.0161

To choose the source term, intakes for uranium, thorium and radium are estimated using the approached discussed above. Progeny intakes are estimated for each of the scenarios. Intakes from uranium are assumed to be unchanged for each scenario. Table 2 shows some estimated intake values for two scenarios. The values in the column labeled thorium source were calculated assuming exposure to the 95th percentile of the plant 6 air concentrations.

Table 2. Example Intake Scenarios

1949 source term choices starting with coworker intakes (pCi/d)			
	U-source	**Th-source**	**Ra-source**
Uranium	2810	2810	2810
Th-230	1373	1580	7402
Ra-226	1373		49600
Pb-210	1373		21005
Po-210	1373		29219
Pa-231	64	212	798
Ac-227	64	212	798
Gross alpha	8366	4602	110833
1950-1957 source term choices starting with coworker intakes (pCi/d)			
	U-source	**Th-source**	**Ra-source**
Uranium	517	517	517
Th-230	253	1580	1310
Ra-226	253		8780
Pb-210	253		3718
Po-210	253		5172
Pa-231	12	212	141
Ac-227	12	212	141
Gross alpha	1539	2309	19639
Example showing elevated uranium intake and 95th % radium intake from 1949			

The intake scenario, the Th-source or Ra-source is selected by finding the scenario that produces the higher probability of causation. Sometimes this can be readily determined by inspection of the intake rates, and sometimes the doses and probability of causation calculations will need to be determined before a determination can be made. It is also conceivable that the U-source could result in a higher probability of causation, when the

radium in equilibrium is larger than either the radium from individual or coworker bioassay, but this will generally not be the case.

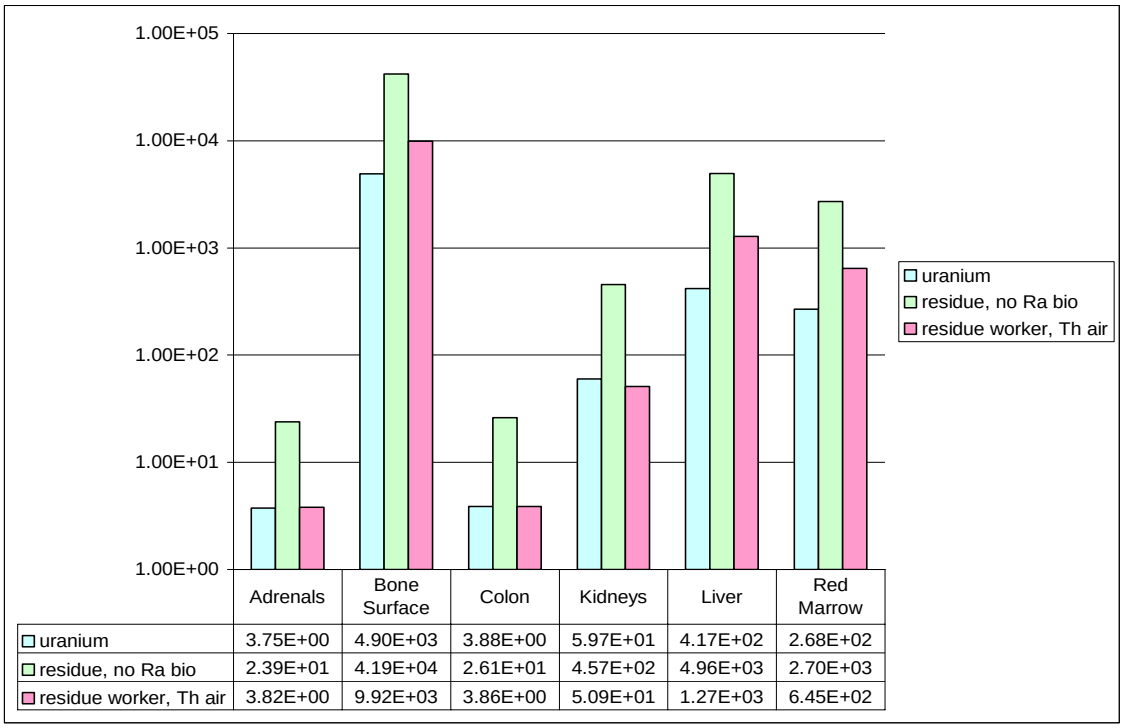
Gross alpha intake rates are shown in Table 2 to provide a sense of how the intake estimates from this method compare to the workplace measurements, which show an intake rate of 1800 dpm/m^3 based on a 95th percentile time-weighted air concentration exposures for work locations (versus people) in the process area.

To provide a sense of the dose from a one-year intake, the 1949 and the 1950-1957 scenarios based on coworker intakes are shown in the graphs below. The doses are 50-year organ doses in rem.

Note

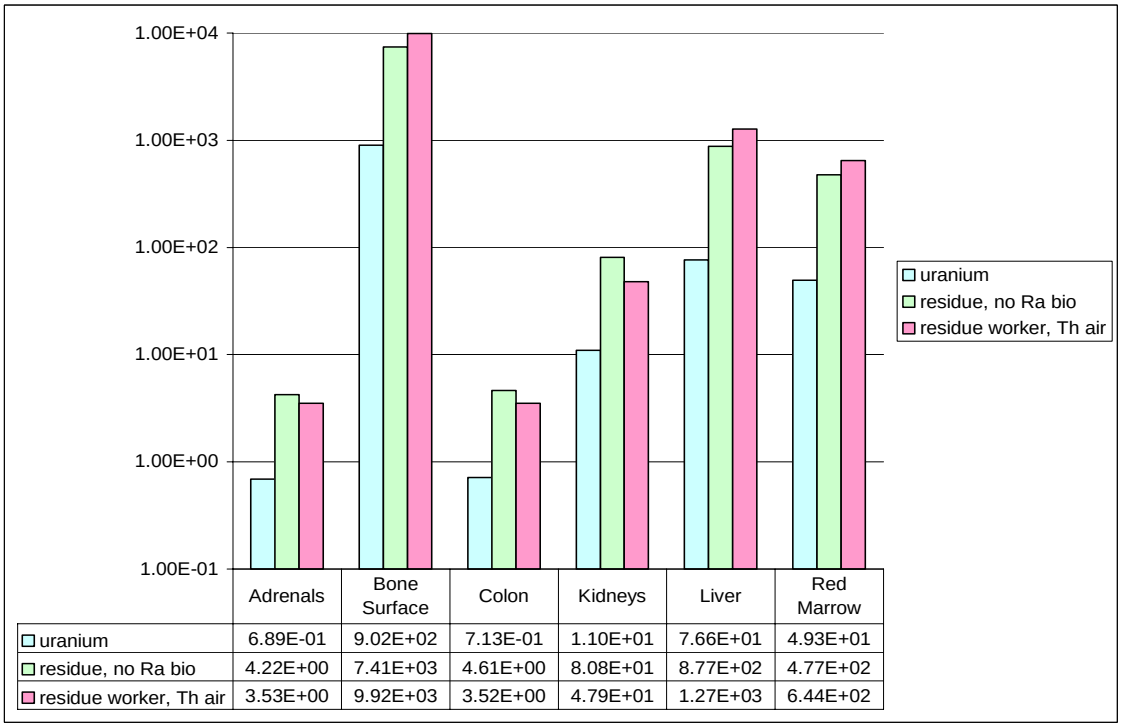
Updates are being developed for some values noted in this draft document based on additional data analyses. It is believed that most of the numbers will not change significantly, but many will change by some small fraction. These current numbers are being presented to help illustrate the proposed method of dose reconstruction, but should not be considered final.

Graph 1
1949 intake scenarios based on coworker data (doses in rem from 1 year of intake).



	Adrenals	Bone Surface	Colon	Kidneys	Liver	Red Marrow
uranium	3.75E+00	4.90E+03	3.88E+00	5.97E+01	4.17E+02	2.68E+02
residue, no Ra bio	2.39E+01	4.19E+04	2.61E+01	4.57E+02	4.96E+03	2.70E+03
residue worker, Th air	3.82E+00	9.92E+03	3.86E+00	5.09E+01	1.27E+03	6.45E+02

Graph 2
1950-1957 intake scenarios based on coworker data (doses in rem from 1 year of intake).



	Adrenals	Bone Surface	Colon	Kidneys	Liver	Red Marrow
uranium	6.89E-01	9.02E+02	7.13E-01	1.10E+01	7.66E+01	4.93E+01
residue, no Ra bio	4.22E+00	7.41E+03	4.61E+00	8.08E+01	8.77E+02	4.77E+02
residue worker, Th air	3.53E+00	9.92E+03	3.52E+00	4.79E+01	1.27E+03	6.44E+02

ATTACHMENT 8

Mike Thorne and Associates Limited

(Director: Dr M C Thorne)

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External Memorandum

Date: 26 July 2005
From: M C Thorne
To: A Makhijani
Copies:
Subject: Radon in Breath

Arjun

I understand that NIOSH is intending to use Rn-222 in breath data to place an upper bound on the Ra-226 content of the body. You described this in your e-mail of 12 July 2005 as an upper limit on radium inhalation. My more precise formulation arises because Rn-222 in breath relates to the current Ra-226 burden in the skeleton plus soft tissues.

When a Ra-226 atom decays in tissue, the Rn-222 atom that is produced can diffuse away from its site of production, enter the systemic circulation and be transferred to the lungs. Because of the long half life of Rn-222 compared with diffusion and circulation times, virtually all of the Rn-222 produced in soft tissues is expected to be exhaled. However, Ra-226 is mainly accumulated in mineral bone and its distribution within the bone changes with time. Initially, it is distributed on bone surfaces, but it then becomes distributed throughout the bone volume (though with substantial inhomogeneities of spatial distribution) on a timescale of weeks to months. Rn-222 produced within the bone volume has a more limited potential to reach the systemic circulation by diffusion. However, this potential is not zero. Stather (1990) has cited values for the fraction of Ra-226 remaining in mineral bone ranging from 9% at ten days after intake to about 40% at 27 years. He notes that the average fraction retained averaged over the 50 years following intake is estimated as 0.3 by the ICRP.

Thus, the loss of Rn-222 in exhaled air has long been considered to be a useful method for estimating the body burden of Ra-226. For example, in the early 1940s, such measurements were made on uranium miners from Colorado (http://physics.nist/gov/GenInt/Curie/1940.html). Standard methods were established at an

early stage. Thus for example, the February 1997 HASL procedure for radon in breath measurements (HAS L-300, 28th Edition, Section 4.5.4, Vol. 1) is taken from Harley *et al.* (1951).

In 1941, the National Bureau of Standards Handbook No. 27 (NCRP Report No. 5) set the permissible body burden for Ra-226 at 0.1 μCi. This was equated by the NBS and the Department of Labor to 1 pCi of Rn-222 per liter of exhaled air (http://www.orau.org/ptp/collection/ionchamber/breathanalysis.htm).

It is somewhat debatable whether standardisation on a concentration is totally appropriate. Thus, for example, Martin (1957) demonstrated that the concentration of Rn-222 in the exhaled air varied inversely with the respiratory minute volume, but that the amount of Rn-222 exhaled per minute was constant.

It is of interest to note that 0.1 μCi of Ra-226 produces 757 pCi of Rn-222 per hour. As the volume of air breathed per hour is around 1.2 m^3 under conditions of light exercise, the anticipated Rn-222 concentration in air would be around 0.63 pCi per liter, if all that which was produced was exhaled. A value of 1 pCi per liter would arise from two considerations. First the maximum permissible burden of 0.1 μCi was typically associated with mineral bone rather than the whole body, so there would have been an additional contribution from soft tissues. Secondly, breath measurements were usually made at rest rather than light exercise, so the rate of breathing of air would have been lower.

At Mallinckrodt (see Document ORAUT-TKBS-0005; Sections 5.3.7 and 6.8), a limited number of workers were subject to Rn-222 in breath analyses from June 1945. Assays were undertaken by MIT, NBS or HASL. These were appropriate laboratories, so the main concerns are likely to have been with protocols for sample collection. After some initial problems, an appropriate procedure was adopted with samples collected in a room away from the plant and only when the employee had not worked for 48 hours and was still in his street clothes. However, there were problems with enhanced background in the test area and these problems do not seem to have been recognised until 1950. These background problems meant that Rn-222 in breath and hence Ra-226 body burdens were likely over-estimated.

Although it is straightforward to estimate a lower bound for the Ra-226 body burden from Rn-222 in breath measurements (by assuming that it is all exhaled), a biokinetic model has to be invoked to partition that Ra-226 between soft tissues and mineral bone. Furthermore, for bone dosimetry an assessment has to be made of the spatial (mainly depth) distribution of Ra-226 in mineral bone. However, the uncertainties in these calculations are limited, as well calibrated models for the behaviour of alkaline earths in the human body exist. Also, an upper bound on the body burden can be estimated by assuming an exhalation fraction of 0.6 (based on 40% retention in mineral bone). In this case, partitioning the resultant burden between mineral bone and soft tissues would undoubtedly be cautious, as a retention fraction of as much as 0.4 would not be plausible for soft tissues.

In summary, I see no reason why Rn-222 in breath measurements should not be used to set an upper bound on accumulated Ra-226 and hence on doses from that radionuclide. However, as usual, attention would have to be given to the extent and quality of the data on a case-by-case basis.

Regards
Mike Thorne

References

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Martin, E J, 1957, Steady and unsteady states of breath radon in subjects with chronic radium poisoning, J. Appl. Physiol., **10**, 447-450.

Stather, J W, 1990. The behaviour, effects and radiation dosimetry of radium in man, Chapter 3-1 of IAEA Technical Reports Series No. 310, The Environmental Behaviour of Radium, Volume 2, IAEA, Vienna, 1990.

ATTACHMENT 9

DATE: August 9, 2005
TO: Arjun Makhijani, John Mauro
FROM: Hans Behling
SUBJECT: Additional External Dose Correction Factors for Consideration at MCW

This communiqué is a follow-up to comments presented at the Advisory Board's Working Group Meeting held on August 4, 2005 in Cincinnati. Among the topics discussed were correction factors needed to adjust recorded external photon doses. While NIOSH acknowledged the need to adjust the external photon dose by an "anatomic geometry" factor (which accounts for the bias of a dosimeter worn at the neck level and the higher doses received by tissues/organs below the waist), there are two additional correction factors that should be considered. These two correction factors include (1) angular dependence of the dosimeter and (2) the use of inappropriate organ dose conversion factors (DCFs). Both issues were addressed under SC&A's Task 3 Report (SCA-TR-Task3) in review of the External Dose Implementation Guideline (OCAS-IG-001) and will only briefly be summarized below.

Correction Factor for Angular Dependence

Multiple factors contribute to the uncertainty of a film dosimeter reading that can generally be defined in terms of (1) laboratory uncertainty, (2) radiological uncertainty, and (3) environmental uncertainty.

Laboratory Uncertainty. Laboratory uncertainty reflects the variable response among dosimeters subjected to a constant and highly-controlled radiation field, such as that of a dosimeter calibration laboratory. Here, dosimeters are commonly exposed to a mono-energetic radiation source at 0º angle (i.e., incident radiation is normal to the plane of the film dosimeter) and for a very brief time before processing. Under these controlled exposure conditions, variability among dosimeter responses reflect variations in (1) the film manufacturing process and composition of the film emulsion, (2) the multi-step photographic development process, and (3) the optical readout process, which quantifies the net optical density of an exposed/developed film dosimeter. OCAS-IG-001 defines film dosimeter uncertainty exclusively for "laboratory uncertainty."

Environmental Uncertainty. The response of a film dosimeter may be adversely affected by environmental conditions that define the workplace where the dosimeter is worn or stored. Factors affecting film include heat, high humidity, certain chemicals (e.g., mercury vapor), stray light, and physical stresses (e.g., pressure). While some of these factors result in fogging, others have the opposite effect of fading. However, due to its complexity, environmental uncertainty in dosimeter response is difficult to quantify and account for in assessing individual dosimeters worn in a variety of work environments.

Radiological Uncertainty. When a film badge is worn by a worker, additional uncertainties must be considered that reflect the (1) long wear period(s), (2) potentially complex radiation fields and photon energies, and (3) dosimeter's variable and reduced response at incident angles other than zero.

This last factor is known as angular dependence and has been assessed in behalf of duPont 510 film, which was commonly used as the industry standard. Table 1 identifies the relative film badge sensitivity for photon incident angles, as reported by Hine and Brownell (1956).

Table 1. Angular Sensitivity for duPont 510 Film Dosimeters

Angle of Incidence	Photon Energy		
	0.11 MeV	**0.20 MeV**	**1.2 MeV**
0º (perpendicular)	1.00	1.00	1.00
22.5º	0.87	0.92	0.97
45º	0.46	0.73	0.91
67.5º	0.33	0.45	0.92
90º	0.16	0.41	0.94

Inspection of Table 1 shows that angular dependence increases with angle and is most pronounced at low photon energies. For example, at 67.5º incidence, the dosimeter's response to 200 keV photons is reduced to 45% of its response under laboratory or 0º incidence exposure conditions.

By means of interpolation, reasonable estimates may be obtained for other angles as well as composite angles defined by isotropic and rotational exposure geometries. For example, 200 keV photons incident at 60º will likely cause a film badge to under-respond by a factor of about two.

Improperly Derived DCFs

Both Appendix B of OCAS-IG-001 and Attachment B of ORAUT-PROC-0006 contain dose conversion factors defined for different photon energies and exposures geometries. Both documents state that:

> *. . . The purpose of this section is to provide guidance on the conversion from individual monitoring data to organ dose. For photon exposures, the organ dose conversion coefficients in ICRP 74 convert from free-air KERMA to absorbed dose in the organ of interest. A conversion from monitored dose to free-air KERMA is needed to complete the organ dose conversion.*

It is my opinion that this second step (i.e., "conversion from monitored dose to free-air KERMA") was not properly performed, which resulted in DCF values that are low and clearly not claimant favorable. This conversion step must take into consideration that the "monitored dose" in most instances represents the shielded response of the dosimeter that can be generically defined as the 1,000 mg/cm^2 deep dose or $H_P(10)$ dose. Moreover, this dosimeter response was registered while the dosimeter was worn on the anterior side of the body at the chest level. In brief, it is the $H_P(10)$ dose recorded on the anterior side at chest

level of the worker that needs to be used as the reference value to which organ doses must be normalized.

The potential errors associated with DCF values are most readily seen for DCF values that convert the recorded $H_P(10)$ dose to the female breast dose. On the assumption that the dosimeter was worn at the very location of interest, then the recorded $H_P(10)$ dose can reasonably be assumed to represent the dose to the female breast. As such, the DCF values for all three energy groupings as well as all exposure geometries should have values of about 1.0.

Inspection of DCFs cited by NIOSH shows values that range from a low of 0.009 to values approaching unity. (Note: least affected are breast DCF values for the AP exposure geometry.) For the most common exposure geometries of rotational/isotropic, DCFs for 30-250 keV photons (at ~0.5) are about a factor of two low.

Errors comparable in magnitude to the female breast DCFs involve tissues/organs that are also surficial and located on the anterior side of the body. These include the skin, eyes, male testis, thymus, and thyroid.

For interior tissues/organs that are more shielded, a quantitative assessment of DCF errors is beyond the scope of this memorandum. A preliminary, ballpark estimate, however, suggests that DCF values for interior tissues/organs may be low by a factor of 1.4 to 2 for isotropic and rotational geometries involving 30-250 keV photons.

References

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