PALOMARES NUCLEAR WEAPONS ACCIDENT

REVISED DOSE EVALUATION REPORT

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Prepared For: Radiation Protection Division
Air Force Medical Operations Agency
Bolling AFB, DC 20332-7050

Prepared By: LABAT-ANDERSON INCORPORATED
8000 West Park Drive, Suite 400
McLean, VA 22102

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EXECUTIVE SUMMARY

A nuclear weapons accident occurred on January 17, 1966 over Palomares, Spain when a United States Air Force (USAF) B-52 bomber and an USAF KC-135 tanker aircraft collided. That accident lead to the release of four thermonuclear weapons. The accident damaged two of the weapons with release of radioactive contamination, leading to a three-month response effort to identify, characterize, remove, and remediate the accident site. During the response effort, some personnel were exposed to airborne dust and debris contaminated with plutonium.

Radiation monitoring efforts during the response were limited to the evaluation of exposures and their possible effects on health using principles and methods accepted at that time. However, recent interest in radiation exposure to veterans and government employees, as well as the availability of improved technology for assessing doses led the Air Force to review the data for possible use in estimating radiation exposures.

Initial Exposure Evaluation

The response effort began on the evening of January 17. A base of operations (Camp Wilson) was established, and measurements for released plutonium began on January 18. The response force peaked at about 680 U.S. personnel on January 31, and then gradually fell until the effort ceased on April 11. Approximately 1,600 personnel participated during the operation.

Response personnel provided urine and nasal swab samples while on site to assess possible intakes of plutonium and the potential effects on health. The sample results were evaluated in terms of guidelines available at the time.

The assessment program concluded that of the nearly 1,600 participants, less than 20% showed levels of plutonium in their bodies that could be detected in urine samples. Only 26 personnel showed values of 7% to 67% of the upper limit for plutonium in the body (Odland 1968a). Those 26 were followed up for a period of 18 to 24 months following the accident. A 1968 Air Force review of the follow-up program concluded that no additional information could be gained from continued sampling and recommended that further sampling effort be suspended.

Exposure and Dose Updates

The evaluations conducted during 1966 through 1968 depended on the limited understanding of plutonium's behavior under field conditions. Since then, advances in that understanding and in methods for assessing dose provided an opportunity to reexamine the monitoring data. The approach uses the concept of Committed Effective Dose Equivalent (CEDE) - a cumulative dose, weighted for the contributions of individual organs, and summed over a 50-year period – as an indicator of possible risk from the exposure. Comparisons can also be made to the annual limit on intake (ALI) of 20,000 picocuries (NRC 2000), to the 21 rem from cumulative exposure to average background radiation over 70 years, or to the 50 rem guideline for cumulative dose to workers (1 rem per year over 50 years of work).

During the project, computer programs that perform the necessary intake and dose calculations were tested. Two programs (CINDY and LUDEP) were selected because intakes they estimated agreed to within a factor of two for the majority of the test cases. That agreement was judged reasonable and acceptable for this assessment.

Available Records

The initial urine sampling that began within three days of the accident experienced some problems such as sampling for less than the desired 24-hour period, possible sample contamination from dust spread by strong winds, and use of non-clinical sample containers. Follow-up sampling was conducted on personnel with initial urine results indicating retained plutonium at 10% of the maximum permissible body burden (MPBB) or more. This second phase was implemented to assess whether sample contamination may have produced spurious urine levels, indicating a false-positive exposure.

Most of the cases involved samples collected on site that were assayed once for gross alpha radioactivity. The remaining cases involving samples collected on site were either resampled, or reanalyzed using alpha spectrometry. Finally, 26 cases were resampled for 18 to 24 months.

Analysis of all the data produced the following four groups.

- A High 26 Cases Group that included the 26 individuals who were resampled for 18 to 24 months after the initial phase of sampling in 1966.
- A Repeat Analysis Cases Group that contained 54 individuals who either had submitted initial samples that were reanalyzed using more sensitive methods (alpha spectrometry), or who were resampled.
- A Contamination Cutoff Cases Group that included 313 individuals with results that were below an assumed cutoff level of 0.1 pCi per day.
- A Remaining Cases Group that contained 1,063 individuals with records that were not otherwise evaluated because their data indicated contamination from collection on site.

Environmental measurements obtained in the Palomares vicinity for over 15 years following the accident provided a basis for preparing independent estimates of intake and dose using representative scenarios for response force activities.

Results

The CEDEs estimated from urinary bioassay were judged unrealistically high when compared with estimates prepared for other plutonium exposure cases – persons residing in the Palomares vicinity and Manhattan Project workers. The estimates of plutonium intake and CEDE from inhalation using environmental data measured in Palomares ranged up to no more than about 0.2 rem. Consequently, the estimates from urine analyses are not useful as representative intakes and doses. The detailed evaluations performed for the High 26, Repeat Analysis and Contamination Cutoff Cases represent preliminary estimates that cannot be considered as definitive. Follow-up studies are required to develop credible estimates of dose that are compatible with those calculated from environmental data.

Conclusions

Preliminary results calculated for all 26 individuals in the High 26 Cases Group, the 54 individuals in the Repeat Analysis Cases Group, and the 313 individuals in the Contamination Cutoff Cases Group proved unrealistically high. They are inconsistent with those calculated from environmental data and when compared with the experience from exposed workers. Furthermore, the urine results are inconsistent with plutonium's known behavior and are inadequate by themselves to support meaningful intake and dose evaluations without confirmatory studies, such as analysis of urine samples now using very sensitive instrumentation,

detailed review of participant medical records, participant interviews, and comprehensive assessments based on sound environmental measurements.

Recommendations

Several future actions should be considered to further refine these initial estimates.

1. Additional effort is needed to reconcile the estimated intakes and doses derived from the urinary bioassay data with the estimates from environmental measurements. A targeted effort that includes participant activities, participant interviews, urine and other appropriate plutonium analyses using current techniques, medical records review, and modeling should be considered.
2. The results of this effort should be communicated to responders, veterans organizations, and other interested parties using appropriate information that clearly confirms the conclusions of the original medical evaluation program, recognizes the difficulties in preparing updated intake and dose estimates, and outlines the options for strengthening the estimates.
3. Further contacts with the Department of Energy for comparison with evaluations of their personnel who responded to this accident could provide useful data. The effort should be summarized in a companion document that conveys the details of the project and its potential effects on health in an easily understood manner. That document should be made available to any of the responders who desire a copy.

1 INTRODUCTION

LABAT-ANDERSON INCORPORATED was awarded TASK ORDER Number TO 799BG0031 under General Services Administration Contract GS–35F-4813G to provide services to the Air Force Medical Operations Agency for evaluating the radiation exposure records of personnel who responded to past nuclear weapons accidents and incidents for the purpose of updating dose estimates. The Task Order specified the following objectives:

- To identify, locate and review the records of the incident, radiation exposure assessments, and other information pertinent to the study.
- To evaluate current methods and models for estimating radiation doses and risks from the intake of radioactive materials contained in nuclear weapons.
- To recommend a methodology for conducting the re-evaluation of the available radiation exposure information.
- To evaluate any and all radiation exposure information, such as urine bioassays, nasal swabs, air sampling information, etc. for scientific soundness and possible use in updating the radiation records of the response personnel.
- To perform the update and prepare records for input to the Air Force Master Radiation Exposure Registry.

The Task Order did not specify extensive searches of personnel records, or efforts to locate and contact the personnel involved except on a limited basis where specific information might be useful or when individuals expressed interest in the project.

The Task Order also required that the effort should begin with the nuclear weapons accident of January 17, 1966 over Palomares, Spain involving a United States Air Force (USAF) B-52 bomber and a USAF KC-135 tanker aircraft. That accident involved a mid-air collision between the two aircraft, the release of four thermonuclear weapons, damage to two of the weapons with release of radioactive components, and a three-month response effort to identify, characterize, remove, and remediate the accident site. During the response effort, personnel were exposed to airborne dust and debris contaminated with plutonium.

Substantial response efforts provided a foundation for evaluating the potential radiation effects from the exposure using accepted principles and methods of the time. However, heightened interest in radiation exposure within the Department of Energy and veterans of the 1991 Gulf War led to this effort to review the data and update radiation exposures, wherever possible, using current methods and procedures.

This report provides the results of the efforts conducted under this Task Order and includes a review of the accident details and radiation assessment efforts and results in Section 2; and a summary of the environmental measurements and review of the radiation assessment data from 1966 through 1968, an evaluation of its accuracy and usefulness, and efforts to prepare the data for re-assessment of radiation doses in Section 3. Section 4 provides a summary of radiation effects and dosimetry methods. Section 5 discusses the methods and results of preparing estimates from environmental data. Section 6 summarizes the methods and results for preparing estimates from the urinary bioassay results. Section 7 discusses the results, assesses the

implications of the results on health, and Section 8 concludes with a summary and recommendations for further evaluations of the responders to this accident.

2 BACKGROUND

At 10:30 a.m. (local time), on January 17, 1966, a U.S. Air Force B-52 bomber and a USAF KC-135 tanker collided during a refueling operation at 9.44 km (31,000 ft.) over the southeastern coast of Spain (DNA 1975). The incident released four thermonuclear weapons that fell to earth near the small coastal hamlet of Palomares, Spain. Serious damage to two of the weapons caused dispersion of their contents over a limited area. Strong winds contributed to further spread of the material and contaminated aircraft debris to the village, surrounding lands, and agricultural crops (Odland 1968a). The response to the incident to find, safeguard, recover, and return weapons contents to the United States, and to assess and mitigate effects on the local populace required significant effort involving hundreds of personnel for almost three months.

Responding personnel encountered the contaminated debris, lands, village, and crops. Although emergency protection measures were followed, responders and local citizens were exposed to the plutonium dispersed from the two weapons. Extensive efforts assessed the effects of those exposures on US military and civilian responders during a program that went on for two years after the incident. Soon after the accident, the Government of Spain represented by the Spanish Junta de Energia Nuclear (JEN) and the Government of the United States, represented at the time by the Atomic Energy Commission (now the Department of Energy), agreed to cooperative programs for extensive follow-up studies of the site and surrounding areas (DOE 2001). Those studies have produced significant understanding about the characteristics of the residual plutonium, its environmental distribution, resuspension into the air, and migration through the soil and other pathways; as well as estimates of the radiation doses to the local populace and evaluations of their health condition.

This section provides additional details about the accident itself, discusses the nature of the response, reviews the methods, procedures and operation of the health and safety assessment program, and reviews the results and limitations of the assessment.

2.1 ACCIDENT SUMMARY

Both aircraft were destroyed in the air. Four thermonuclear weapons, 11 men (four survived), and hundreds of tons of debris fell to earth in and around the barriada (Hamlet) of Palomares. Parts of the aircraft were scattered over a wide area generally between Cuevas de Almanzora and Vera along the Mediterranean Sea between Puerto Rey and Villaricos. At that time Palomares had no telephones and did not appear on maps of the area. The population of the time was estimated to be about 1200.

The first of the four nuclear weapons was found intact with its primary chute deployed on the evening of January 17, 1966 just east of Palomares. A radiation survey showed that no radioactivity escaped the weapon. The area was designated impact point 1.

The primary chutes did not open for two other weapons whose chemical high explosives detonated. One weapon was found on the morning of January 18, 1966 about one mile west of the village (impact point 2). The third weapon was found about two hours later on the eastern edge of Palomares (impact point 3) with high explosive and radioactive material scattered by

impact and explosions. The fourth weapon was finally recovered intact from the Mediterranean Sea on April 7, 1966 (Odland 1968a).

The explosions and fires around impact points 2 and 3 produced airborne clouds of plutonium-containing dust that were carried over some distance by 30 knot winds. Eventually, a total of 558 acres of soil contaminated above 5.4 micrograms per square meter ($\mu g/m^2$) were remediated by removal or plowing. These levels provided many opportunities for responders to inhale or ingest the radioactive plutonium.

2.2 RESPONSE SUMMARY

The Guardia Civil, the first representatives of the Spanish government, arrived on site about one hour after the accident. They immediately took charge, secured the accident site and informed both Spanish and American authorities. The commander of the 16th Air Force headquartered at Torrejon Air Base near Madrid and the Strategic Air Command Headquarters at the Offutt Air Force Base in Omaha, Nebraska were notified and the "Broken Arrow " response system was initiated. The commander and three staff members surveyed the accident site from the air and arrived at San Javier (195 km from Palomares) at 1:30 p.m.

OPERATION RECOVERY was initiated by deciding to bring personnel in from two Spanish bases, Moron, and Torrejon. Movement of personnel started at 0100Z on January 18 from Moron with a second convoy at 0310Z. 126 personnel were transported in six buses. The first of two convoys from Torrejon departed at 0137Z, the second at 0202Z, with 175 persons in six buses. Following a 12 to 14 hour drive to the southern coast, the first of buses arrived at 1300Z and the last arrived about 1700Z.

Another Disaster Control Team from Offutt Air Force Base in Omaha Nebraska arrived at the accident scene at 7:30 am on January 18. Members of the Joint Nuclear Accident Coordinating Center (JNACC), Sandia Corporation, and the Los Alamos Scientific Laboratory (LASL) left Albuquerque at 1800 GMT on January 17.

By the evening of January 17, 49 U.S. personnel were on site. Camp Wilson was established as a general headquarters, and measurements for released plutonium began on January 18. About 300 more airmen from the Moron and Torrejon air bases were on site by the evening of January 18. A maximum of about 680 U.S. personnel were at Camp Wilson on January 31.

By January 21, the camp moved to leveled, higher ground some 5.6 km east of the Garrucha where it remained until April 3. A helicopter pad, motor pool, and 75 tents were on firmer ground in less danger of flooding. The camp was moved again where it remained until closure on April 11.

Manning reached a peak by January 31, with 598 Air Force, 64 Army, and 19 Navy. All except some officers were housed at the camp. Those were quartered in two hotels close to the accident scene. Personnel involved with search, recovery, and decontamination generally rotated through the camp at two-week intervals. Population at the camp varied, but from the high on January 31, there was a gradual reduction until the camp closed on April 11. The first major reduction occurred on February 9 and 10 when about 50 of the clean-up personnel and the 40-man ordnance disposal team left. A slight upswing occurred from March 11 to 17 during the period of filling of 4,810 barrels with contaminated soil and crops. Other personnel at camp included 126

Guardia Civil and 39 Spanish personnel who worked in the cleanup and other activities. Overall, almost 1,600 personnel participated in the response effort at one time or another.

Response activities included performing radiation surveys, protection, and recovery of nuclear weapons, development of remediation plans, and decontamination of affected areas. These will not be discussed in this report. However, details of the efforts to assess and control radiation exposure are of vital importance to this effort and are described next.

2.3 SUMMARY OF HEALTH ASSESSMENT ACTIVITIES

This accident represented one of the first times that plutonium had been dispersed on and around civilian property outside the United States. Furthermore, the response placed a significant number of military and civilian personnel resources at risk. Procedures for assessing and controlling contamination from the materials in these weapons were available and used. However, there were many questions about the behavior of inhaled and ingested plutonium under field conditions.

2.3.1 On-Site Sampling

Urine sampling, recognized as a reasonable method for assessing exposure to plutonium, was begun within three days of the accident. Urine sample collection on site was subject to collection of less than the desired 24-hour specimen and possible sample contamination. Samples were shipped by the most expedient means to the USAF Radiological Health Laboratory (USAF RHL) at Wright-Patterson AFB, Ohio for analysis. Two sampling phases were used – an initial phase and a resample phase.

2.3.2 Interpretation of Urine Results

The results were evaluated in terms of the maximum permissible body burden (MPBB, see Appendix A) of ^{239}Pu as recommended by the National Bureau of Standards (NBS) in Handbook 69 (NBS 1959). The NBS recommendations were based in part on Publication 2 of the International Commission on Radiation Protection, *Recommendations of the International Commission on Radiological Protection, Report of Committee II on Permissible Dose for Internal Radiation*, published in 1959 (ICRP 1960). The MPBB for ^{239}Pu considers the bone as the "critical organ" or the organ that is most susceptible to radiation from plutonium and is the basis for developing protection limits. The body burden is defined as that portion of ^{239}Pu distributed by systemic circulation. It does not include that amount fixed in the lungs. The MPBB was 0.044 microcurie (μCi) of ^{239}Pu.

The MPBB was developed as an operational tool for limiting dose to a critical organ over a working lifetime. The dosimetry model used assumed uniform deposition of the radionuclide in the organ, energy emitted equals energy absorbed, and the characteristics of the model could be represented by "Standard Man" data. The concept was designed to provide adequate protection over a 50-year working lifetime and as such applied to continuous intake of radionuclides over the entire period. Thus for a material like plutonium, the limit would allow for continuous intake for 50 years while keeping the dose to the bone (the critical organ) below the limit.

An individual's body burden was estimated from the measured urinary gross alpha radioactivity for initial samples. The following equation was used, taken from Langham (Langham 1956):

$$D_r = 435\ U\ t^{0.76}$$

where:

D_r = retained systemic body burden (pCi or Bq); meaning the amount retained in the body "t" days after exposure

U = ^{239}Pu activity (pCi or Bq) in a 24-hour sample

t = time in days from exposure to sampling

The analysis required assumptions about the type of exposure (acute or continuous), and about whether samples represented true 24-hour urine outputs. This calculation applies to a single acute exposure. The individuals responding to the incident were generally on site for two weeks, some more and some less. Others remained for almost the entire period of operations. The beginning date for the exposure was assumed as the midpoint of time an individual arrived on site until ceasing activities (departing). Odland (Odland 1968b) reported that "When the 12-hour volume was less than 1.2 L, calculations were so adjusted as to express the total activity had the output been 1.2 L. When the volume exceeded 1.2 L, the actual value for calculating systemic body burden was used."

2.3.3 Resampling Program

The Air Force conducted a resampling program at 90 to 150 days after collection of the initial sample. This resampling applied to individuals whose gross alpha results for initial samples suggested a systemic body burden of 10% or more of the MPBB.

The program established procedures to identify and quantify the isotope of interest in the urine – ^{239}Pu.

2.4 SUMMARY OF RESULTS

Odland reported that the USAF RHL processed almost 1600 urine samples during the initial phase (Odland 1968a). Table 1 gives the distribution of the samples in relation to the systemic body burdens they represent. Those results indicate that 20 individuals potentially exceeded the MPBB and 442 samples exceeded 10% of the MPBB and required resampling. However, the possibility for contamination of the initial samples collected on site introduced uncertainty about that conclusion. This potential for sample contamination in and around Palomares was also recognized by the Spanish Junta de Energia Nuclear (JEN), which transferred urine sample collection and medical examination of local residents from Palomares to Madrid in 1967 (Iranzo 1987).

Table 1. Initial Urine Samples (Percentage of one MPBB).

Number Analyzed	1586
BB greater than 100%	20
BB 9% to 99%	422
BB 0.9% to 9%	537
BB less than 0.9%	607

A resampling program began shortly after on-site operations ended. Originally, samples were desired at two-month intervals; however, this became impractical and samples were collected primarily at the discretion of the individuals. Table 2 contains the results of the resampling program (Odland 1968a). The laboratory processed 422 samples during the resampling phase. Of those, only six exceeded 10% of the MPBB with slightly less than half of those resampled (203) showing results below 1% of the MPBB.

Table 2. Urine Resampling Program Results.

BB greater than 10%	6
BB 1 to 10%	213
BB less than 1%	39
BB zero	164
Total	422

A small specimen of lung tissue, obtained at time of necropsy from an early responder who died from heart disease, contained 2.8 pCi of ^{239}Pu; or about 0.00034 microcuries (about 2% of estimated maximum permissible lung burden) when extrapolated to the total mass of the lung. Early urine analyses for the individual indicated a rapid decrease in gross alpha radioactivity that was attributed to contamination. However, early behavior of inhaled plutonium was not excluded as a possibility (Odland 1968a). Nevertheless, if correct, the quantity in the lung of this individual represents a small fraction of the MPBB after 9 months following exposure.

In summary, the assessment program indicated that of the nearly 1,600 participants, less than 20% indicated systemic body burdens of plutonium that could be detected by urine bioassay, and only 25 showed values in the range 7% to 67% of the MPBB guideline (Odland 1968a). Those 25 and one additional individual were followed up for a period of 18 to 24 months following the accident.

2.5 PLUTONIUM DEPOSITION REGISTRY BOARD

The Air Force recognized that the consequences of possible exposure to plutonium from the Palomares Broken Arrow required in-depth and credible assessment, provisions for long-term maintenance of the records, and possible follow-up of those exposed. To satisfy that need, representatives of the U.S. Air Force Medical Service met in Omaha, Nebraska in March 1966 and identified the need for a detailed and long-range program to provide follow-up and treatment, when required. The concept of a special board to satisfy those needs was developed into a Plutonium Deposition Registry and Board with the following purposes as stated in the proceedings of the first meeting (Odland 1966):

(1) It would provide adequate follow-up of personnel with internal deposition of plutonium, in order that any possible biological injury would be detected at the earliest date, and it would provide, when required, the best possible treatment to reduce body burdens of Plutonium-239.

(2) It would provide the government with complete factual data upon which to evaluate claims for compensation that might subsequently arise.

(3) It would provide the medical profession with additional urgently needed data with which to manage medical problems at future Broken Arrows or laboratory accidents of a similar nature.

The Plutonium Deposition Registry Board met first on October 26-28, 1966 (Odland 1966) to establish the Board; to review progress to date, and to set policy for further follow-up. The Board reflected a tri-service nature as well as an interagency flavor with participation by the Atomic Energy Commission, the Veterans Administration, and the Defense Atomic Support Agency. Additionally, several recognized experts in plutonium medical effects participated as Board Members or as Consultants (Odland 1966). Board deliberations produced recommendations in the following areas:

- Samples should be collected from all that departed the accident scene without submitting a sample, or whose initial samples suggested a systemic body burden greater than 9%.
- No further sampling of individuals whose **initial** urine results suggested a systemic body burden of less than 9%.
- Sampling should be continued on members whose results on resampling were in the top 10% of the resampling group and showed systemic body burdens of 1-10%.

The Board also discussed the use of whole-body counting as an additional assessment tool and the use of ^{239}Pu to ^{241}Am ratios in the weapon components, soil and urine as possible method for determining ^{239}Pu in the lungs; however, no specific recommendations were developed.

On January 16, 1968, the Air Force Logistics Command Surgeon issued a letter report that reviewed progress of the follow-up effort (Wallace 1968). The report summarized the results of resampling of the 26 individuals whose initial urine samples showed the highest ^{239}Pu content suggesting systemic body burdens of 7% to 67%. The report concluded that little additional information could be gained from continuing the effort. Finally, the report announced that the Surgeon General of the Air Force had concurred with canceling the Board meeting scheduled for 1967 and that further activities would be limited to analyzing tissue specimens, as they became available. As a practical matter, this letter report suspended activities of the Board in the matter of the Palomares accident. Research during this project identified no evidence of additional testing efforts or results.

Our review of the urinary levels reported during the assessments conducted in 1966 and 1967 indicated that the initial intakes could exceed the current annual limit on intake (ALI) recommended by the ICRP (ICRP 1979). Consequently, a repeat evaluation of the urinary data seemed warranted to provide assessments using currently accepted methods for analysis and management of radiation risks. The remainder of this report discusses the detailed approach for performing those assessments.

3 ASSESSMENT OF AVAILABLE DATA

The response effort discussed in Section 2 above included a health evaluation program that generated records of the possible doses to those who responded to the accident. Locating those records involved contacts with the Air Force Medical Operations Agency (AFMOA) at Bolling AFB, DC and the Air Force Institute for Environmental, Safety and Occupational Health Risk Analysis (AFIERA) at Brooks AFB, TX. Those records required detailed review to understand

the data they contained and the processes that produced the data; an analysis of the consistency and reliability of the contents; and possible adjustments to estimate intake and dose equivalent.

In addition, the Government of Spain, in collaboration with the U.S. Department of Energy, has conducted extensive studies of the environmental characteristics of the residual contamination in the Palomares area. In particular, air sampling, particle size characteristics, and resuspension factors have been determined from data collected for more than 15 years. These data provide a valuable source for independent intake and dose estimates.

3.1 ENVIRONMENTAL DATA

Studies of the environment around Palomares have included air sampling at four locations, and estimates of the resuspension of plutonium particles from the surface into the air for subsequent inhalation by the local populace. Those studies used air samplers placed in four locations representing possible sources of plutonium. Samplers were located near the impact points of the two destroyed weapons, at another contaminated area, and in the town of Palomares. From 1966 to 1980, the highest annual average air concentration was measured at 11.9 fCi/m^3 (442 μBq/m^3) in 1967. The highest average for a weekly measurement period occurred in March 1967 with a concentration of 292 fCi/m^3 (10.8 mBq/m^3) (Iranzo 1987). Measurements during other periods were lower than these, but demonstrated some variation over time.

Studies at Palomares have also estimated the resuspension of plutonium at and around Palomares from the same air sampling data combined with knowledge of the plutonium surface contamination levels. Resuspension is a process that represents the air concentration of a material above a surface contaminated with the same material. The resuspension factor (in units of m^{-1}) is the ratio of the air concentration (expressed in units of pCi/ m^{-3} or Bq/m^{-3}) to the surface contamination (in units of pCi/ m^2 or Bq/m^{-2}). The studies at Palomares indicate that the resuspension factors initially were 10^{-7} m^{-1} initially, dropped to values on order of 10^{-9} m^{-1} months later, and to 10^{-9} m^{-1} to 10^{-10} m^{-1} after several years (Iranzo 1994). The air concentrations were determined in areas where the surface contamination ranged from 3.2 μCi/m^3 (0.118 MBq/m^2) to 32 μCi/m^3 (1.18 MBq/m^3).

Both the air sampling and the resuspension results represent credible efforts that can be used as the basis for estimates of intake and dose.

3.2 AIR FORCE BIOASSAY DATA

During the initial contact, AFIERA and AFMOA provided records in the form of:

- Air Force Forms with laboratory analytical and exposure details of the nasal swipe and urine samples submitted and processed.
- Complete case files for the 26 individuals identified for follow-up in 1966 and commonly referred to as the "High 26".
- A Microsoft Excel spreadsheet prepared by AFIERA staff that contained the data from those Air Force Forms, and some data related specifically to the 26 individuals (referred to as the "High 26" who were considered as having the highest exposures).
- Copies of the accident response reports, USAF RHL documents on the evaluation of exposures by urinalysis, and selected publications from journals and conference proceedings.

Appendix B contains a detailed discussion of the information collected, an evaluation of the information's suitability for a dose evaluation, and adjustments made to the data for performing intake and dose calculations. The record prepared and maintained by the Air Force consisted of forms, computer spreadsheets, and written correspondence and reports of activities.

The data were evaluated to assess the availability of the elements required by the internal dosimetry models. Review indicated that the exposure date or dates, sample date, and results were not completely recorded for all cases. Substantial numbers of samples lacked one or more important pieces of data. Data forms for 115 individuals apparently represented a repeat analysis of a sample or a follow-up sample for an individual. Sample collection proceeded for only 12 hours for many samples collected at Camp Wilson, indicating a correction to 24 hours would be needed. Our review indicated that 12-hour samples were clearly designated in only 42 of the samples. Lacking any other recorded information, sample volumes were assumed to represent 24-hour output unless specifically designated as 12-hour samples.

Urine sampling, begun within three days of the accident, was subject to several compromises, including: collection limited to 12 hours or less for operational requirements; sample contamination from strong winds; non-uniform decontamination procedures; make-shift sample containers, and frequently contaminated storage areas.

Records for 122 nasal swab reviewed indicated that only 13 contained a result (8 were 0 pCi, 4 had values all below 1.5 pCi, and 1 was reported as NDA). Therefore, the nasal swab records were not used in this analysis.

The majority of available records contained results from the gross alpha method on samples collected on site. Most of the records for samples collected on site raised serious questions about estimates derived from them. Records for the 26 individuals in follow-up contained multiple samples collected up to two years after the incident. Unfortunately, the pattern of results for samples collected during the resampling phase often did not follow the pattern expected for Class Y (Type S) plutonium. However, treatment of the records for the 26 served as the model for the other cases. A second group of records contained repeated analyses using the more sensitive alpha spectrometry and provided a reasonably well-defined set of cases for analysis. These two groups were designated the High 26 Group and the Repeat Analysis Group, respectively. Appendix E provides additional details of the bioassay data evaluation and grouping of cases.

The remaining results generally fell into two categories: those with the results of some resampling; and those with one sample and often very high results. Careful review of the group of data indicated that processing all of the cases would produce unrealistic estimates that would be based on potentially contaminated samples. Gross alpha results from samples collected on site produced intake estimates and doses that seemed unreasonably high. Contamination of samples collected at the accident site continued to impact the evaluation as it did at the time of the accident. However, review of those data also indicated a substantial number of cases with urinary results that were essentially below the detection limit or were quite low. Their data were reviewed again to determine whether a reasonable lower cutoff could be determined. Analysis of the processes (Appendix E) supported a cutoff limit at 0.1 pCi/day for gross alpha activity. This was similar to the detection limit of 0.74 mBq/d (0.02 pCi/d) used in studies by the Government of Spain from 1966 to 1985 (Iranzo 1987). Consequently, 0.1 pCi/day was selected as a cutoff limit, and cases in that category were designated the Contamination Cutoff Group.

Applying a cutoff to urinary excretion to individual cases does not precisely impact all samples equally. A fixed value for the cutoff concentrations produces higher estimated intakes and correspondingly higher dose equivalent values for samples taken at longer times after the exposure, especially for Class Y (Type S) plutonium.

After applying the cutoff, 1,219 samples for 1,063 individuals had urine concentrations above 0.1 pCi/d that were classified in the Remaining Cases Group. These were not evaluated further.

4 RADIATION EFFECTS AND DOSIMETRY METHODS

Responders to the Palomares accident encountered sources of possible exposure from plutonium-contaminated aircraft debris, contaminated lands, and agricultural crops, and dust produced by winds. Evaluation of the potential radiation effects requires estimates of the exposure and associated radiation dose, and comparison with knowledge about the effects of radiation on human health. Furthermore, these evaluations must take into account current knowledge and apply accepted methods for estimating the radiation exposure and dose. The approach to accomplishing these estimates is guided by recommendations of both international and national scientific bodies concerned with radiological protection. These bodies, primarily the International Commission on Radiological Protection (ICRP) have published recommendations on the relevant guidelines for limiting radiation effects and exposure, and estimating doses from radioactive materials that may enter the body, as plutonium does.

This section summarizes the current understanding of radiation effects, in general, and plutonium, specifically, on health, and the guidelines to protect workers and the public from those effects. It also summarizes updated internal dosimetry methods relevant to evaluating plutonium exposures.

4.1 SUMMARY OF RADIATION EFFECTS

This study of exposure to plutonium at Palomares and calculation of possible doses to internal organs raises questions about the possible health effects that may be associated with them. This section provides a brief summary of our understanding of the possible health effects from ionizing radiation and plutonium in particular, some of the guidelines for limiting exposure to it, and some basic information about the possibility that a certain dose could cause some kind of effect on health.

4.1.1 General Radiation Effects

In discussing health effects relating to ionizing radiation, the term "dose" is used. "Dose" comes from the early medical use of x-rays, much as a dose of medicine is measured in grains or ounces. It refers to the amount of radiation energy absorbed by an organ, tissue, or cells, measured in rem (or Sv). Today, the average American receives a dose of 0.3 rem (0.003 Sv) every year from natural sources—radioactive materials in rocks and soil, cosmic radiation, radon, and radioactivity in our bodies. Over a 70-year lifetime, the cumulative background dose averages 21 rem (0.21 Sv). In some areas of the world, people receive much higher doses from background radiation. For example, in areas of India and Brazil the ground is covered with monazite sand, a radioactive ore. Radiation exposure rates there are many times the average background levels elsewhere. People who live in these areas receive doses of up to about 0.7 rem

(0.007 Sv) each year from the gamma radiation alone (NAS 1990). These levels combined with the other sources of background radiation (cosmic rays, radon, etc.), cause average doses that are about three times more than the U.S. average. Yet these people show no unusual rates of cancer or other diseases linked to radiation.

The effects of ionizing radiation can be categorized as either prompt or delayed, based on the time frame in which the effects are observed. Prompt effects, like rapid death, occur when high doses are received in a short period of hours to weeks. Delayed effects, such as cancer, can occur when the combination of dose and dose rate is too small to cause prompt effects. Both animal experiments and human exposures to high levels of radiation show that ionizing radiation can cause some cancers (NAS 1990). All of the observed effects of ionizing radiation in humans occur at relatively high doses. At the low doses that are of interest to radiation workers and the general public (that is, below a few rem), studies to date are inconclusive (NAS 1990). Although adverse health effects have not been observed at low doses, the carcinogenic nature of ionizing radiation makes it wise to limit the dose.

For low-doses, there are no conclusive data that relate dose to health effects or showing a threshold, or minimum, level for cancer. Because of this, experts who study radiation effects have decided that the results from high-dose, high-dose-rate studies must be used to control the low-dose, low-dose-rates experienced by workers and the public. A convenient way to do this is to assume that no effects occur at zero dose. In addition, since the rate at which effects occur is extrapolated from higher doses, it is also assumed that the effect increases linearly with dose. These two assumptions are known as the "linear-dose-response, non-threshold" (LNT) hypothesis. This implies that the same number of additional cancers would occur from exposing 100 persons to 100 rem (1 Sv), or 10 thousand persons to 1 rem (0.01 Sv), or 10 million persons to 0.001 rem (0.00001 Sv). No prompt effects have ever been reliably observed in humans below about 10 rem (0.1 Sv). Reports from the Japanese atomic bomb survivor studies conclude that the location and reality of such a threshold, if one does exist, are difficult to assess. Nevertheless, the Health Physics Society (HPS 1996) has stated that "Below 10 rem (which includes occupational and environmental exposures), risk of health effects are either too small to be observed or are non-existent."

Within the first 30 years after the discovery of x-rays, standards were developed for the measurement of radiation. At about the same time, acceptable levels of dose were set. The first level, known as the 'tolerance dose', or that amount of radiation that could be tolerated, was set at one-tenth of a unit (about 0.1 rem (0.001 Sv) in today's units) per day for 300 days a year, which amounts to 30 rem (0.3 Sv) in a year.

From World War II to the early 1980s, radiation dose limits were adjusted downward in response to increased concern about radiation effects, the increased uses of radiation, and because improved radiation protection technologies appeared. The National Council on Radiation Protection and Measurements (NCRP, established in the 1930s) developed the recommended changes for the United States. During that time, the dose limit was reduced from three-tenths of a rem in a six-day period in 1946 to 5 rem (0.05 Sv) per year in the mid-1950s. In addition, a limit for the public was set at one-tenth of the worker limit to provide an additional margin of safety.

Research does not show a clear threshold dose for cancers from radiation, so the small risk per person at low doses had to be considered in relation to the large number of workers who were receiving those doses (NCRP 1993b).

The NCRP adopted three radiation protection principles: (a) no practice shall be carried out unless it produces a positive net benefit (sometimes called justification); (b) all exposures shall be kept as low as reasonably achievable (ALARA), economic and social factors being taken into account (called optimization); and (c) the dose equivalent to individuals shall not exceed the recommended limits (called limitation). These principles work together to protect against both prompt and delayed effects in large groups of workers and the public.

In 1993, the NCRP released a new set of national recommendations based on International Commission on Radiation Protection's 1990 recommendations. Those limits for non-threshold effects differ slightly from the earlier recommendations: 50 rem (0.5 Sv) per year to any tissue or organ and 15 rem (0.15 Sv) to the lens of the eye to avoid cataract formation. The recommended occupational limits on whole-body doses (total effective dose equivalent), first set at 5 rem (0.05 Sv) per year in 1958, are now set at no more than 5 rem (0.05 Sv) in any one year and a lifetime average of no more than 1 rem (0.01 Sv) per year (NCRP 1993).

Occupational radiation exposure limits for federal agencies are currently established in "Radiation Protection Guidance to Federal Agencies for Occupational Exposure," 52FR 1717, signed by President Reagan on January 20, 1987. The Nuclear Regulatory Commission implemented that guidance in its regulations on radiation protection (Title 10, Code of Federal Regulations, Part 20). These limits apply to all licensed uses of radioactive material under NRC's jurisdiction. Similarly, other Federal agencies as a matter of policy and directive, including the DoD in DODI 6055.8, Occupational Radiation Protection Program, also observe this guidance.

The current established protection standards are (NRC 1999):

- 5 rem in a year for workers (to protect against cancer).
- 50 rem in a year for workers to any organ (to protect against threshold effects, such as radiation burns, etc.).
- 50 rem in a year to the skin or to any extremity.
- 15 rem in a year to the lens of the eye (to protect against cataracts).
- 0.1 rem in a year (70-year lifetime) for members of the public.

These limits are in addition to the radiation doses a person normally receives from natural background, medical testing and treatment, and other sources.

The protection standards mentioned above provide regulatory guidelines to be used primarily for designing radiation protection programs and facilities. Their intent is to limit dose to a worker so that risk is limited to levels that are similar to so-called "safe industries." Limits for the public perform the same purpose but generally include additional margins of safety to account for a wider range of ages (childhood to aged), more diverse health condition, and individual sensitivities. Their primary purpose is to prevent exposures that are associated with risks exceeding the established guides.

These guidelines also offer usable comparisons for evaluating the possible effects of exposures. For example, the occupational limit of 5 rem (0.05 Sv) in a year provides one such value. Since 5 rem (0.05 Sv) represents an acceptable risk, any exposure below 5 rem (0.05 Sv) should be considered acceptable. NCRP recommends that the average dose equivalent per year for workers should not be more than 1 rem (0.01 Sv) a year over 50 years or work. That is the same as 50 rem (0.5 Sv) in 50 years. Therefore, 50 rem (0.5 Sv) provides a reasonable guide for an exposure

from radioactive materials in the body, such as plutonium. Since these guides are set with margins of safety, receiving a higher dose does not mean that one will be harmed. However, it would mean that further evaluation might be needed to determine whether the exposure was a one-time incident or one that could recur.

An alternate approach to evaluating the possible effects of an exposure considers the possibility that an exposure will lead to health effects, such as cancer or hereditary effects. The NCRP has provided risk factors for the probability that a certain dose equivalent from radiation will cause an effect. Those factors for workers are 0.0004 per rem (0.04 per Sv) for fatal cancer, 0.00008 per rem (0.008 per Sv) for non-fatal cancer, and 0.00008 per rem (0.008 per Sv) for hereditary disorders for a total of 0.00056 per rem (0.056 per Sv) (NCRP 1993a). For members of the entire population, these factors are 0.0005 per rem (0.05 per Sv) for fatal cancer, 0.0001 per rem (0.01 per Sv) for non-fatal cancer and 0.00013 per rem (0.013 per Sv) for hereditary disorders, for a total of 0.00073 per rem (0.073 per Sv).

4.1.2 Health Effects of Plutonium

Plutonium, discovered in 1941, is radioactive and can be dangerous when it gets into the human body. Some have even referred to plutonium as "the most toxic substance known to man". Early concerns about the health risks of plutonium arose from knowledge of the effects of radium, discovered by Marie Curie in 1899. With its half-life of 1620 years, radium-226 presents an intense and constant radiation source for hundreds of years. Early uses of radium exposed workers to significant doses with acute cases ending in rapid death, and lower exposures leading to infections of the jawbones, pathological bone fractures, or cancers of the bone.

The National Bureau of Standards addressed radium's dangers by developing an occupational standard for radium, adopted in May 1941, about two months before the discovery of plutonium. Scientists on the Manhattan Project then recognized the potential hazards of plutonium, which is similar to radium. They estimated that plutonium would be roughly as dangerous as radium when comparing equal masses.

Plutonium gives off alpha particles that produce heavy ionization and give up their energy more quickly than the lighter beta particles, or x-rays and gamma rays. In air, alphas travel only 3 to 5 centimeters and in living tissue only about 30 micrometers. That distance is less than the thinnest part of the dead layer of external skin cells (called the epidermis), or the thickness of a piece of paper (about 100 micrometers). Because of this low penetrating power, materials that give off alpha particles present no hazard when kept outside the body.

Unfortunately, when they get inside the body, alpha emitters come into very close contact with the body tissues and irradiate cells. Plutonium can be inhaled, ingested, or passed into the blood stream through a wound. When that happens, about 90 percent eventually goes to the lung, liver, or bones.

The half-life of plutonium-239 is 24,065 years. This half-life is short enough that 1 microgram of material will undergo more than 2000 decay events per second, but it is long enough to allow that microgram to decay at an approximately constant rate for thousands of years.

No one has ever died from an acute plutonium uptake. But, researchers have estimated lethal doses from studies on dogs, rats, and mice, which indicate that a few milligrams of plutonium per kilogram of tissue is a lethal dose. Extrapolated to humans, an intravenous injection of about

22 milligrams into an average human (70 kilograms; about 154 pounds) would be lethal within about 30 days to half the people exposed. Inhalation would require about four times more or 88 milligrams.

Recognizing the similarity of plutonium to radium, scientists worked to develop exposure standards that would limit the risks to workers, especially on the important war-time effort of developing a plutonium-implosion bomb. Beginning in 1945, those efforts have evolved into a set of radiation protection recommendations that have received international acceptance. In 1977, the ICRP published major revisions in those recommendations that based radiation protection for plutonium on dose rather than deposition in the body. Those recommendations, known as ICRP 30, have been largely adopted in the United States. In 1991, the ICRP published new recommendations (ICRP 60), which reduced the recommended annual occupational limit to 2 rem (20 millisieverts) per year. Thus far, these recommendations have not been adopted in the United States, however, they are considered in most radiation protection assessments.

Plutonium absorption in the body depends mainly on the plutonium compound and how it enters the body. The body generally absorbs the soluble forms (nitrates, citrates, and certain oxides) more readily than insoluble forms. Plutonium absorption through intact skin is usually quite low, but deposits in tissues through puncture wounds, cuts, and somewhat less through skin burns. Soluble plutonium begins movement throughout the body within minutes or hours of the uptake and may move to the lymph nodes near the wound; remaining for years. Some insoluble plutonium gets into the blood circulation quickly, but most remain at the site and are slowly redistributed over weeks and months. About 90 percent of the systemic burden deposits in the liver and bones. The kidneys excrete plutonium in urine that represents the concentration of the plutonium in the blood making plutonium measurements in urine a convenient indicator of plutonium in the body.

Ingesting plutonium is perhaps the least likely means for plutonium to enter the body. But even if plutonium is ingested, the gastrointestinal tract provides a natural barrier, and in adults only about 0.05 percent of the soluble plutonium compounds and a mere 0.001 percent of the insoluble ones enter the blood stream. The rest of the plutonium simply moves out of the body in feces.

Inhalation of plutonium dust provides the most likely entry route for plutonium. Particle size affects plutonium absorption. Smaller particles are more likely to be retained. Particles over 10 micrometers in diameter (considered large) are filtered out in the nose and upper respiratory region, swallowed, and eventually passed out through the gastrointestinal tract. Particles less than 10 micrometers in diameter (called respirable particles), deposit on the mucus layer of the bronchial tubes. Through a process, known as lung clearance, hair-like structures of the lining (called celia) transport the mucus layer and dust particles up to the throat, removing much of the foreign material deposited in the bronchial tubes.

Smaller particles, especially those under 1 micrometer in diameter, are carried down into the tiniest airways of the lung and into alveoli (also known as air sacs). These structures have no effective lung-clearance mechanisms, but scavenger cells called phagocytes, engulf the inhaled plutonium particles, and transport them into lymph nodes or into lung tissues.

Autopsy studies reveal that, initially, plutonium is mostly deposited on the bone surfaces. Less than 5 percent of the plutonium is typically found within the bone marrow. Based on this this pattern of deposition, the primary carcinogenic risk from plutonium in the skeleton is bone

cancer. There is no conclusive evidence that plutonium increases the risk for leukemia, which is the unchecked proliferation of certain blood cells produced in the bone marrow.

Plutonium in the bone remains there for a very long time, gradually being redistributed throughout the bone. Current models (based on observation of exposed persons and autopsy data) estimate a half time of about 50 years for plutonium retention.

The plutonium deposited in the liver is eventually transformed from relatively soluble forms in hepatic cells into insoluble forms (hemosiderin deposits), which are sequestered in the cells that form the linings of liver ducts (reticuloendothelial cells). The retention half time for the plutonium deposited in the liver is approximately 20 years.

To date, there have been only few epidemiological studies of workers exposed to plutonium. Studies of workers at Los Alamos National Laboratory (Wiggs 1994) and Rocky Flats (Wilkinson 1987) are the only ones in the United States to have used quantitative measurements of plutonium exposures, but they involved few workers: 303 at Los Alamos and 1450 at Rocky Flats. These two studies showed no evidence of statistically increased rates of lung, liver, and bone cancers, which are shown in animal experiments to be the highest-risk cancers due to plutonium exposure. Another study (Reyes 1984) indicates that an increased brain-cancer rate in Rocky Flats workers was not caused by plutonium exposure or external radiation.

A study (Voelz 1983) involving 224 males exposed to plutonium between 1944 and 1974 who had plutonium deposition greater than 0.16 microgram (0.01 microcurie) found no cases of bone or liver cancer. By 1980, the final year of the study, only one person had died of lung cancer indicating risks were much lower than predicted by some nuclear-industry critics. Another study looked at 26 chemists, metallurgists, and technicians at Los Alamos, who were accidentally exposed to plutonium between 1944 and 1946.Their plutonium body burdens ranged from 50 Bq to 3,180 Bq when estimated by analysis of their urine (Voelz 1997). Interestingly, the mortality rate of these men has been lower than that of the population in general, and in 1996, 19 of them were still living.

Of those who are no longer alive, one died of lung cancer in 1989, at the age of 66, and two died of prostate cancer and congestive heart failure, respectively, but both had lung cancer at the time of death. All three men were very heavy smokers. Significantly, three cases of lung cancer are consistent with the national cancer incidence rate, over the same period, in U.S. white males of the same age. Another subject, who had an estimated plutonium deposition of 0.245 microgram, developed a rare bone cancer 43 years after exposure and died in 1990. This finding is statistically significant for the small group of 26, but in the Los Alamos study (Wiggs 1994) of 303 workers, this same individual remained the only one to have developed bone cancer. That one death from bone cancer in this larger group could well be due to chance and is not statistically significant. Finally, three more died of causes unrelated to cancer.

Overall, data from the several studies of persons exposed to low levels of plutonium radiation in the United States do not show a relation- ship between dose and effect. They merely indicate that such a relationship does not exist or cannot be confirmed. If plutonium is harmful at these low levels, its health risks are so small that, given the small number of workers involved, epidemiological methods cannot differentiate between effects triggered by plutonium radiation and variations in a group of people unexposed to such radiation.

Although studies on plutonium workers in the United States did not demonstrate the risk from plutonium radiation, there are data from much higher doses to which Russian plutonium workers have been exposed. Russian scientists have recently published two studies (Tokarskaya et al.1997, Koshurnikova et al.1998) of workers who had been exposed to plutonium at the Mayak Plant. The authors demonstrate that an increased risk for lung cancer is associated with higher exposures. Although both studies investigate this risk on many of the same workers, their conclusions about the relationship between dose and risk are different.

In one study, (Koshurnikova 1998) analyzed data from a cohort of 1479 workers who had been exposed to high doses of various types of radiation, including plutonium radiation, between 1948 and 1993. The control group was composed of 3333 other workers at Mayak who had also been exposed to radiation but within occupational limits. The study found a linear relationship between lung doses from 0.5 to 30 sieverts (or 50 to 3000 rem) and standardized mortality ratios. While this result found no threshold for effects, the trend of increasing rates with increasing dose is impressive.

The second study (Tokarskaya 1997) found a nonlinear threshold relationship between dose and lung cancer risk in a case-control study devoted to 162 plutonium workers who developed lung cancer between 1966 and 1991 and a control group of 338 Mayak workers who, during the same period, did not. The authors found no lung cancer risk up to a threshold dose of 16 sieverts, corresponding to about 1.6 micrograms of plutonium deposited. Above this threshold value, however, the risk rises rapidly. The two Russian studies are very different in the dose response relationships reported. However, the data demonstrate that lung cancer risk does indeed increase with higher doses.

A recently reported study to estimate the mortality risk per unit dose from exposure to plutonium produced results that compare well with estimates derived by other workers. This study developed the estimates using four independent approaches – epidemiologic studies of workers exposed to plutonium; epidemiologic studies of persons exposed to low-LET radiation combined with a relative biological effectiveness factor (RBE) for alpha particles appropriate to the cancer site; epidemiologic studies of persons exposed to alpha-emitting radionuclides other than plutonium; and controlled studies of animals exposed to plutonium and other alpha-emitting radionuclides extrapolated to humans (Grogan 2001). That work reported mortality risk per unit dose of 0.13 per Gy for lung, 0.057 per Gy for liver, 0.0013 per Gy for bone, and 0.013 per Gy for bone marrow (leukemia). Calculations of the risk for a unit intake compared well with estimates prepared by other workers.

It has been almost six decades since plutonium was first made. No doubt, the dangers of plutonium are real. However, plutonium has been handled in different chemical forms, fabricated as a metal, machined, and used successfully primarily because standards and procedures were established early. Because of this, there has been no instance of acute death from taking plutonium into the body.

4.2 REVIEW OF INTERNAL DOSIMETRY METHODS

Exposure to radiation can occur from sources of penetrating radiation outside the body, such as x-ray machines or industrial radiography sources, or from sources of radioactive materials, such as plutonium or uranium, that enter the body, locate in an internal organ or organs, and irradiate the tissues of those internal organs. The problem of calculating the dose depends on many factors

such as the shape of the organ, the type of radiation, the amount of the deposit, and the distribution of the deposit. Each of these individual factors is subject to considerable variability and difficulty in determining accurately. Once a dose is calculated, effectively communicating the possible effect of the dose on health requires additional skill and effort.

The current approach to limiting radiation exposure in the United States is derived from recommendations in ICRP Publications 26 and 30. The ICRP approach uses the concept of Committed Effective Dose Equivalent (CEDE) - a cumulative dose, weighted for the contributions of individual organs, and summed over a 50-year period for workers. Quantities derived from the CEDE such as the Annual Limit on Intake (ALI) and the Derived Air Concentration (DAC) provide operational limits for workers so that the overall guidelines will not be exceeded. The ALI is the activity of a radionuclide that would irradiate a person to the limit set by the ICRP for each year of occupational exposure. The DAC is found by dividing the ALI by the volume of air inhaled (2,400 m^3) in a working year (2,000 hours) (ICRP 1979).

For internal exposures, determining the dose requires knowledge of the following questions:

- How does the material get into the body?
- Once in the body, how quickly does the material move to other organs?
- Does the material in the initial organ leave the organ or does some remain?
- Once in an organ, how does the material irradiate the organ and other organs?
- Once in an organ, how does the material move to other organs?
- Finally, how does is the material eliminated from the body if at all?

Answers to these provide the basis for developing an approach to calculate the dose to organs, the effective dose equivalent to the body, and interpreting the effects of the dose.

4.2.1 Internal Dosimetry Methods

The methods for estimating organ dose from internal radionuclides have evolved since radioactive materials were discovered and used. Until 1979, ICRP Publication 2 provided the guidelines and methodology. In 1979, ICRP Publications 26 and 30 changed the basic approach to limiting radiation, and for internal radionuclides in particular. ICRP Publications 54, 60 and 66 provided revised recommendations and updated models on the behavior of radionuclides in the body.

ICRP-2 assumed that a single organ could be considered the critical organ; that the organ retention could be represented by a single exponential term; that the physical characteristics, such as intake parameters, transfer functions, and tissue size and weight, could be represented by "Standard Man" data; that organs could be assumed to be spherical; and that scattered radiation could be ignored. Intakes of radionuclides were controlled by limiting "Maximum Permissible Concentration" (MPC) values in air and water for workers so that the annual dose limit to the critical organ would not be exceeded.

ICRP Publication 26 revised the system of dose limitation to one based on risk. This approach acknowledged the availability of sufficient information about the effects of radiation to estimate risk for fatal cancer from a unit dose equivalent in exposed people and in the risk of serious disease to offspring of exposed people. The basic recommendations addressed both stochastic

effects and non-stochastic effects. For stochastic effects, such as cancer and hereditary effects, risks are assumed to be directly related to dose equivalent with no threshold, meaning that the probability of the effect occurring, rather than the severity, is related to the dose equivalent. The severity of non-stochastic effects, such as cataracts and erythema, varies with dose, usually above a threshold or minimum dose.

ICRP Publication 30 provided revised dosimetry models that assume organ retention is represented by one or more exponential expressions, the critical organ concept no longer applies, the dose in an organ must consider radiation emitted by other organs in the body, and the physical characteristics are represented by "Reference Man" data in ICRP Publication 23 (ICRP 1975).

Under the revised system, dose equivalent limits are intended to prevent non-stochastic effects and to limit stochastic effects to acceptable levels. To meet this end, an annual occupational limit of 50 rem (0.5 Sv) to any organ was established (ICRP 1979). For stochastic effects, the limit on risk is the same whether the whole body is irradiated or organs are non-uniformly irradiated. This is accomplished by assigning organ weighting factors, w_t, that represent the ratio of the risk for the effect in an organ to the risk for whole body irradiation. The limit on risk to the whole body – called committed effective dose equivalent (CEDE) is then determined by summing the contributions for each irradiated organ and is limited to 5 rem (0.05 Sv). The committed dose equivalent (CDE) is the total dose equivalent averaged over a tissue (T) in the 50 years following intake and is limited to 50 rem (0.5 Sv). Table 3 contains the organ weighting factors from ICRP-30.

The dosimetry model calculates the absorbed dose averaged over the organ mass during 50 years following intake. It considers each radiation type and applies a radiation weighting factor, sometimes called the quality factor, which has the following value:

Q=1 for beta particles, electrons and all electromagnetic radiation.

Q=10 for fission neutrons emitted in spontaneous fission and protons.

Q=20 for alpha particles from nuclear transformations, for heavy recoil particles, and for fission fragments.

Table 3. ICRP-30 Tissue weighting factors, w_T (ICRP 1979).

Tissue	Weighting Factor, w_T
Gonads	0.25
Red Marrow	0.12
Lung	0.12
Breast	0.15
Thyroid	0.03
Bone Surface	0.03
Remainder 0.06 for the organs with the five highest dose.	0.30

The ICRP further refined its basic recommendations and updated certain models for the respiratory tract and the biokinetics of deposited materials. The ICRP's 1990 recommendations

(ICRP 1991) provide weighting factors for tissues that were part of the remainder in the 1979 recommendations of ICRP-26 (ICRP 1979). Table 4 compares the tissue weighting factors of ICRP-26 and ICRP-60 and include a reduction in the bone surface and breast factors by three times, a 67 percent increase in the thyroid factor, and assignment of factors for additional organs, including the skin of the whole body.

Table 4. Tissue Weighting Factors (ICRP 1991).

	ICRP Recommendations	
Tissue or organ	1979	1990
Gonads	0.25	0.20
Red Marrow	0.12	0.12
Colon		0.12
Lung	0.12	0.12
Stomach		0.12
Bladder		0.05
Breast	0.15	0.05
Liver		0.05
Esophagus		0.05
Thyroid	0.03	0.05
Skin		0.01
Bone Surface	0.03	0.01
Remainder	30[1]	.05[2]

[1] A value of 0.06 is applicable to each of the five remaining organs or tissues receiving the highest equivalent doses.
[2] The remainder is composed of the following tissues or organs: adrenals, brain, small intestine, kidney, muscle, pancreas, spleen, thymus and uterus.

The differences between the two ICRP models for the respiratory tract could be expected to produce differences in estimated doses. During development of the updated respiratory tract model, its performance was tested in detail to determine the affects of various parameters taken alone and in combination. Some examples of the performance of both systems provide useful information about likely differences in estimating both equivalent dose and effective dose equivalent.

One such evaluation, reported by James (James 1994) compared the lung dose equivalent and effective dose for several categories of radionuclides, including insoluble alpha emitters, such as plutonium at Palomares. In those illustrations, James compared doses for intakes of 1 μm activity median aerodynamic diameter (AMAD) particles although ICRP recommends 5 μm AMAD for workers. For 1 μm AMAD, Type S (Class Y) ^{239}Pu, the ICRP-30 and ICRP-66 equivalent dose per unit intakes were 320 μSv/Bq and 84 μSv/Bq, respectively. The ICRP-66 equivalent dose was lower by about a factor of 3.8. For 5 μm AMAD particles, ICRP-66 estimated 50 μSv/Bq, or about 6 times lower. Calculating effective dose for the same conditions, ICRP-30 produced 60 μSv/Bq and ICRP-66 produced 16 μSv/Bq for 1 μm AMAD particles and 9.1 μSv/Bq for 5 μm AMAD particles, representing reductions of about 3.7 and 6.5, respectively. Thus, other factors being equal, the ICRP-66 respiratory tract model can produce equivalent doses that are roughly 3

to 6 times lower for the same intake than the ICRP-30 model. This difference, attributed to the modified model for lung deposition and clearance and revised tissue weighting factors must be recognized in evaluating methods for this project.

Determining the amount of material taken into the body during an exposure forms the basis for estimating the amount of material that is transferred to the blood stream and internal organs as well as the amount that clears from the body. Estimates of the organ dose from internal emitters generally follows from an intake assessment, which is usually based on measures of the material in the body or excreted from the body. Common methods include in-vitro bioassay of the amount of the material excreted, measurements of body or organ content, or estimates from air or water concentrations. For this case, estimates of the intakes from environmental plutonium concentrations provide the best available method for assessing the intake. The large collection of urinary analyses were evaluated and used to estimate intakes and doses; however, those were judged unrealistically high. The methods and models used for accomplishing the estimates from urine analyses are discussed in Appendix D.

4.2.2 Computer Models Investigated

Many computer programs have been developed and are available for performing the calculations of the models discussed above. Currently more programs implement the ICRP-30 system than the ICRP-66 model. This comes as no surprise since the ICRP-30 system remains the current system for regulation of the doses from radioactive materials in the United States. However, one objective for this project included the evaluation and recommendation or the best calculation method. Since ICRP provisions are usually adopted in the U.S., investigating at least one software program that implemented the most recent approach seemed reasonable. After some review of the available software, three programs were selected for further study – the Radiological Bioassay and Dosimetry Program (RBD) as modified for the Air Force, Code for Internal Dosimetry (CINDY), and Lung Dose Evaluation Program (LUDEP ver 2.06). Testing of program performance and selection for use are described in Appendix D.

4.3 MODEL ADOPTION

RBD/AF, CINDY, and LUDEP all provide acceptable performance on estimating intake, calculating dose, and providing compatibility with the available data. LUDEP is somewhat less convenient for manipulating large numbers of cases and for generating outputs that can be used in other manipulations; however it implements the current ICRP respiratory tract model.

CINDY and RBD/AF implement the current regulatory system of the NRC and DOE for radiation protection, while LUDEP offers the alternative for applying the respiratory tract model and other features of recent ICRP recommendations. CINDY provides somewhat more flexibility in setup, estimating intakes, and reporting. Consequently, CINDY was chosen as the primary method for assessing the Palomares cases. LUDEP was retained as a reasonable alternate that provides complementary assessments for interesting cases and offers a much-needed point for comparison of results.

5 ESTIMATES FROM ENVIRONMENTAL MEASUREMENTS

5.1 METHODS

The environmental studies summarized in Section 3.1 reported values for the annual average air concentration and the highest weekly measurement obtained with air samplers located near the impact point for weapon number 2. These were selected as reasonable values for air concentrations that response force personnel could have experienced. Those values were combined with dose conversion factors for Type S plutonium calculated using LUDEP. Since breathing rates affect the intake – the more air one breathes in, the more plutonium that enters the lungs – the calculations were performed for standard workers (breathing rate of 1.2 m^3 per hour) and for heavy workers (1.688 m^3 per hour). Also, the calculations were performed for particle sizes of 1 micrometer and 5 micrometers AMAD. Smaller particle sizes tend to produce higher deposition in the lung and consequently higher doses. Previous recommendations of the ICRP (ICRP-30) recommended 1 micrometer AMAD; however, recent recommendations (ICRP-66) favor 5 micrometers AMAD as more representative of worker exposures.

5.2 RESULTS

Calculations of intake and dose were performed for three exposure scenarios. The first assumed that response force workers were on site for two weeks, and worked 6 days per week for 12 hours a day. This would represent many of the responders who rotated at two-week intervals. The second scenario used 4 weeks on site under the same work conditions to represent those who stayed somewhat longer. Finally, the last scenario assumed that responders could have been exposed for 11 weeks, which essentially represents the entire response effort; i.e. from just after the accident until March 31, 1966. Those estimates are shown in Table 5 and indicate that even the highest scenario produces much less than 1 rem whole body committed effective dose equivalent.

The resuspension factors described in Section 3.1 were used to calculate air concentrations, intakes, and doses (CEDE) for the same scenarios described above. The results listed in Table 6 indicate that even the highest dose (0.312 rem) is well below a significant amount. Furthermore, these estimates differ significantly from the intakes and dose estimates derived from urine analysis, and demonstrate the need to refine the analysis with follow-up studies.

6 RESULTS FROM URINALYSIS DATA

The response to the Palomares nuclear accident involved hundreds of personnel working toward the common purpose of recovering vital materials, protecting themselves and the local populace, and restoration of the accident scene to useable and safe conditions. The accident itself released plutonium during explosions and fires that followed the impact of two of the nuclear weapons with the ground. The plutonium was released primarily as airborne dust and as residues from fire that contaminated the ground. Since the fires essentially were out long before serious response efforts started, the main source of exposure arose from activities such as vehicle movement, handling debris during recovery, plowing fields to mix the contaminant into the soil, and vehicle movement. Persistent winds also contributed to the resuspension of contaminated soils from the

Table 5. Intake and dose estimates from air concentrations.

Average Air Concentration 0.000442 Bq/m^3
Maximum Air Concentration 0.0108 Bq/m^3

					Dose Conversion Factor (Sv/Bq)		Average Air Concentration			Maximum Air Concentration		
								CEDE (Sv)/ CEDE (rem)			CEDE (Sv)/ CEDE (rem)	
Scenario	Worker Type	Breathing Rate (m^3/hr)	Particle Size (um)	Exposure Time (hours)	ICRP-26	ICRP-60	Intake (Bq)	ICRP-26	ICRP-60	Intake (Bq)	ICRP-26	ICRP-60
2 weeks	Standard	1.2	1	144	1.946E-05	1.531E-05	0.076	1.486E-06	1.169E-06	1.87	3.632E-05	2.857E-05
6 days per week								0.00015	0.00012		0.0036	0.0029
12 hours per day	Standard	1.2	5	144	1.084E-05	8.647E-06	0.076	8.279E-07	6.604E-07	1.87	2.023E-05	1.614E-05
								0.00008	0.00007		0.0020	0.0016
	Heavy	1.688	1	144	1.975E-05	1.571E-05	0.107	2.122E-06	1.688E-06	2.63	5.185E-05	4.124E-05
								0.00021	0.00017		0.0052	0.0041
	Heavy	1.688	5	144	1.227E-05	1.010E-05	0.107	1.318E-06	1.085E-06	2.63	3.221E-05	2.651E-05
								0.00013	0.00011		0.0032	0.0027
4 weeks	Standard	1.2	1	288	1.946E-05	1.531E-05	0.153	2.973E-06	2.339E-06	3.73	7.263E-05	5.714E-05
6 days per week								0.00030	0.00023		0.0073	0.0057
12 hours per day	Standard	1.2	5	288	1.084E-05	8.647E-06	0.153	1.656E-06	1.321E-06	3.73	4.046E-05	3.227E-05
								0.00017	0.00013		0.0040	0.0032
	Heavy	1.688	1	288	1.975E-05	1.571E-05	0.215	4.244E-06	3.376E-06	5.25	1.037E-04	8.248E-05
								0.00042	0.00034		0.0104	0.0082
	Heavy	1.688	5	288	1.227E-05	1.010E-05	0.215	2.637E-06	2.170E-06	5.25	6.442E-05	5.303E-05
								0.00026	0.00022		0.0064	0.0053
Full Response	Standard	1.2	1	792	1.946E-05	1.531E-05	0.420	8.175E-06	6.431E-06	10.3	1.997E-04	1.571E-04
11 weeks								0.00082	0.00064		0.0200	0.0157
6 days per week	Standard	1.2	5	792	1.084E-05	8.647E-06	0.420	4.554E-06	3.632E-06	10.3	1.113E-04	8.876E-05
12 hours per day								0.00046	0.00036		0.0111	0.0089
	Heavy	1.688	1	792	1.975E-05	1.571E-05	0.591	1.167E-05	9.283E-06	14.4	2.852E-04	2.268E-04
								0.00117	0.00093		0.0285	0.0227
	Heavy	1.688	5	792	1.227E-05	1.010E-05	0.591	7.250E-06	5.968E-06	14.4	1.772E-04	1.458E-04
								0.00073	0.00060		0.0177	0.0146

Table 6. Intake and dose estimates from resuspension.

Minimum Resuspension Factor	1.29E-09	m^{-1}
Minimum Surface Contamination	1.18E+05	Bq/m^2
Minimum Air Concentration	1.522E-04	Bq/m^3

Maximum Resuspension Factor	1.00E-07	m^{-1}
Maximum Surface Contamination	1.18E+06	Bq/m^2
Maximum Air Concentration	0.118	Bq/m^3

					Dose Conversion Factor (Sv/Bq)		Minimum Air Concentration			Maximum Air Concentration		
								CEDE (Sv)/ CEDE (rem)			CEDE (Sv)/ CEDE (rem)	
Scenario	Worker Type	Breathing Rate (m^3/hr)	Particle Size (um)	Exposure Time (hours)	ICRP-26	ICRP-60	Intake (Bq)	ICRP-26	ICRP-60	Intake (Bq)	ICRP-26	ICRP-60
2 weeks	Standard	1.2	1	144	1.946E-05	1.531E-05	0.026	5.119E-07	4.027E-07	20.4	3.968E-04	3.122E-04
6 days per week								5.119E-05	4.027E-05		0.0397	0.0312
12 hours per day	Standard	1.2	5	144	1.084E-05	8.647E-06	0.026	2.851E-07	2.274E-07	20.4	2.210E-04	1.763E-04
								2.851E-05	2.274E-05		0.0221	0.0176
	Heavy	1.688	1	144	1.975E-05	1.571E-05	0.037	7.308E-07	5.813E-07	28.7	5.665E-04	4.506E-04
								7.308E-05	5.813E-05		0.0566	0.0451
	Heavy	1.688	5	144	1.227E-05	1.010E-05	0.037	4.540E-07	3.737E-07	28.7	3.519E-04	2.897E-04
								4.540E-05	3.737E-05		0.0352	0.0290
4 weeks	Standard	1.2	1	288	1.946E-05	1.531E-05	0.053	1.024E-06	8.054E-07	40.8	7.936E-04	6.244E-04
6 days per week								1.024E-04	8.054E-05		0.0794	0.0624
12 hours per day	Standard	1.2	5	288	1.084E-05	8.647E-06	0.053	5.703E-07	4.549E-07	40.8	4.421E-04	3.526E-04
								5.703E-05	4.549E-05		0.0442	0.0353
	Heavy	1.688	1	288	1.975E-05	1.571E-05	0.074	1.462E-06	1.163E-06	57.4	1.133E-03	9.012E-04
								1.462E-04	1.163E-04		0.1133	0.0901
	Heavy	1.688	5	288	1.227E-05	1.010E-05	0.074	9.080E-07	7.474E-07	57.4	7.039E-04	5.794E-04
								9.080E-05	7.474E-05		0.0704	0.0579
Full Response	Standard	1.2	1	792	1.946E-05	1.531E-05	0.145	2.815E-06	2.215E-06	112.1	2.182E-03	1.717E-03
11 weeks								2.815E-04	2.215E-04		0.2182	0.1717
6 days per week	Standard	1.2	5	792	1.084E-05	8.647E-06	0.145	1.568E-06	1.251E-06	112.1	1.216E-03	9.697E-04
12 hours per day								1.568E-04	1.251E-04		0.1216	0.0970
	Heavy	1.688	1	792	1.975E-05	1.571E-05	0.204	4.019E-06	3.197E-06	157.8	3.116E-03	2.478E-03
								4.019E-04	3.197E-04		0.3116	0.2478
	Heavy	1.688	5	792	1.227E-05	1.010E-05	0.204	2.497E-06	2.055E-06	157.8	1.936E-03	1.593E-03
								2.497E-04	2.055E-04		0.1936	0.1593

ground or contaminated dusts from the surfaces of accident debris, local buildings, or agricultural crops.

Ingestion by hand to mouth transfer is a second possible route of entry. However, that route is very inefficient. Furthermore, the fraction of plutonium that enters the bloodstream from the intestines is very small (0.00001 for Type S). For reasons discussed in Appendices D and E, the ingestion route is not considered further.

The type of exposure was assumed a single acute exposure. This assumption accommodates the long time for removal of plutonium oxides from the human body. The response activity occurred from January 18, 1966 until April 3, 1966 when activities were moved from Camp Wilson to another location. Personnel on site reached a maximum in late January; tapered off during February, and then increased slightly in mid-March during the packaging of contaminated debris, soil and other wastes for disposal. Most departed the site by late March 1967. The nominal length of assignment was about two weeks. However, records indicate that some personnel stayed much longer.

6.1 HIGH 26 CASES

The responders were assigned to four groups of cases, as discussed above – the High 26 Cases Group, the Repeat Analysis Cases Group, the Contamination Cutoff Cases Group and the Remaining Cases Group. The High 26 Cases Group offered the best collection of urinary measurement data to develop an overall understanding of the relationship between the measurement results and possible intake of plutonium. Therefore, substantial effort was applied to evaluating these cases. Then, that understanding was applied to the remaining three groups of cases. As discussed above, however, the quality of the data set limited the preparation of reasonable estimates. This section describes the approach to evaluating each group, the results obtained, the relationship between the estimated dose equivalents and effects.

6.1.1 Methods and Results

The High 26 Cases Group represents the collected measurement data from 26 responders who were identified for follow-up after the initial phase of sampling in 1966. This group provided 127 urine samples during their on-site and resampling activities. Most of those samples (102 of 127) produced ^{239}Pu measurements from alpha spectrometry. Appendix E provides detailed discussions of the data evaluation, results of model fitting, and estimated intakes and doses.

6.1.1.1 Methods

Careful evaluation of the results revealed several difficulties with the reported results. These included differences in the reporting of confidence levels for the results. The reported errors for gross alpha measurements represented the 95% confidence level while the reported errors for alpha spectrometry measurements represented the 68% confidence level. Since the criterion for reporting a result as no detectable activity was based on the 95% confidence limit, some alpha spectrometry results may have been reported as positive when the estimated errors did not support that conclusion. In addition, some alpha spectrometry reports contained calculated values although the reported results indicated NDA. Those results were used in these estimates when recorded on the individual data cards.

Laboratory measurements experienced some difficulties with reproducibility also. In several samples with multiple analyses, differences in reported concentrations of two to three times were observed.

The urine analysis results for the High 26 Cases Group indicated that those cases with several measurements for samples collected over the entire initial and resampling efforts could provide the best data for testing. The data and CINDY and LUDEP program setup were varied in several ways. Assumptions were developed for the date of exposure, the use of gross alpha results and the use of NDA results. For the programs, the main adjustment involved the method for weighting results during intake assessment using CINDY and LUDEP. Generally, the date of exposure was assumed as the first day on site, gross alpha measurements were rejected, and values were developed for NDA reports. These variations are discussed in detail in Appendix E.

6.1.1.2 Results

For the 26 cases, the preliminary intake estimates varied from 34,000 pCi to 570,00 pCi from CINDY and 19,000 pCi to 2,600,000 pCi from LUDEP with the gross alpha results excluded in all the cases. Estimates of committed effective dose equivalent ranged from 10 rem to 170 rem (0.1 to 1.7 Sv) from CINDY and 1.3 to 180 rem (0.013 to 1.8 Sv) from LUDEP. LUDEP ranged from –83% to +150% of CINDY results. The range of differences between LUDEP results and CINDY results seems reasonable considering the variation in the data and the complexities of the assessment. In addition to the intakes and CEDE estimates, 50-year committed dose equivalents were calculated for organs using CINDY. Those results are discussed in Appendix E, however, when compared with independent estimates from environmental data and with the results of other exposure cases, these estimates seem unreasonably high.

6.2 REPEAT ANALYSIS CASES GROUP

Palomares responders were placed in the Repeat Analysis Cases Group if they met one or both of the following conditions:

- They submitted an initial urine sample while on site that was analyzed for gross alpha radioactivity and then reanalyzed by alpha spectrometry for ^{239}Pu; or
- They submitted an initial sample while on site that was analyzed by gross alpha counting and then submitted one or more follow-up samples after returning to their base of assignment for analysis by alpha spectrometry.

6.2.1 Methods and Results

From January 17, 1966 to June 22, 1966, this group provided 82 urine samples from 54 individuals that produced usable results. The gross alpha and alpha spectrometry measurements are primarily greater than 0.1 pCi/d and the two types of measurements are interspersed among one another. Most of the samples were characterized by a gross alpha measurement followed by reanalysis by alpha spectrometry in an attempt to identify the radionuclide responsible for the gross alpha result. In most cases, the alpha spectrometry result was lower than the gross alpha measurement. Unfortunately, resampling was not accomplished for those in this group.

6.2.1.1 Methods

The Repeat Analysis Cases Group had exposure dates that extended over a broader range of dates than the High 26 Cases Group. However, many were among the initial responders who arrived in January 1966. Because the time on site seemed shorter and better recorded for this group, the exposure date was assumed as the midpoint of the time at Camp Wilson. In general, gross alpha results for samples collected on site were excluded from the analysis, gross alpha results reported as NDA were assigned a value of 0.009 pCi/d, numerical results recorded on alpha spectrometry records reported as NDA were used, and some alpha spectrometry results were excluded when they did not fit the expected urinary excretion pattern. Method details are provided in Appendix D.

6.2.1.2 Results

For the 54 cases, the estimated intakes varied from 2,900 pCi to 1,300,000 pCi from CINDY and 11,900 pCi to 5,240,000 pCi from LUDEP with the gross alpha results excluded in all the cases. Estimates of committed effective dose equivalent ranged from 0.9 rem to 400 rem (0.009 to 4.0 Sv) from CINDY and 0.8 to 367 rem (0.008 to 3.67 Sv) from LUDEP. LUDEP results ranged from –238% to +94% of CINDY results. In addition to the intakes and CEDE estimates, annual dose equivalents and committed dose equivalents were calculated for organs using both CINDY and LUDEP. Details of these results are discussed in Appendix E. As for the High 26 Group, these estimates are unrealistic when compared with the estimates from environmental measurements.

6.3 CONTAMINATION CUTOFF CASES GROUP

The Contamination Cutoff Cases Group of analyses was created to calculate estimated intake and dose equivalent for those whose urine measurement results indicated potentially contaminated samples collected at the accident site but were below a reasonable minimum level that did not represent unusually high exposures. While the data for this group were not especially robust, this approach offered an opportunity to evaluate additional cases. As discussed in Appendix E, a level of 0.1 pCi/d was adopted as reasonable maximum level for cases included in the Contamination Cutoff Cases Group.

6.3.1 Methods and Results

6.3.1.1 Methods

The procedures for analysis of the High 26 Cases Group were applied to the Contamination Cutoff Cases Group, except that the intakes and dose equivalents were calculated using only the CINDY program. The group had exposure dates that began over a similar range of dates to the Repeat Analysis Cases Group. Many of this group stayed on site for one to two weeks, with some up to a month. The exposure date was assumed as the midpoint of the time at Camp Wilson. See Appendix E for additional details of this group's analyses.

6.3.1.2 Results

For the 313 individuals in the Contamination Cutoff Cases Group, the estimated intakes varied from 1,500 pCi to 110,000 pCi. Estimates of committed effective dose equivalent ranged from 0.46 rem to 34 rem (0.0046 to 0.34 Sv). The higher estimated intake and dose were produced by a urine sample, taken at 25 days after the assumed exposure date, with a result of 0.099 pCi/d of gross alpha activity. According to the excretion function derived, the urinary content on day 25 represents approximately 9×10^{-7} of the inhalation intake. This case illustrates how urine concentrations that are even slightly above delectability can lead to sizeable estimated intakes and dose equivalents. This further illustrates the difficulty in obtaining realistic estimates from sparse data at or near the analytical methods detection limit.

6.4 REMAINING CASES GROUP

The cases that were not included in one of the previous three groups were placed in the Remaining Cases Group. These samples included those from individuals who submitted only one sample, or from cases where some follow-up was attempted but results were inadequate because of low or no chemical recovery or laboratory error. This group contains sample measurements on 1,063 individuals for 1,219 samples. For discussion purposed, the lowest and the highest urine results of 0 and 237.9 pCi/d of gross alpha radioactivity were input to CINDY, and produced estimated intakes of 75,000 pCi to 20,000,000 pCi corresponding to CEDEs of about 23 rem to 6,000 rem (0.23 to 60 Sv). These results are clearly unrealistic, not supported by the air concentrations observed at Palomares and require careful evaluation.

7 DISCUSSION

The preliminary intake and dose equivalent estimates for the Palomares response personnel used the available data to the best extent possible. The approach involved reasonable assumptions about the type of activities that the responders performed and about the length of time, they may have been exposed. Detailed assignment records on the personnel were not available, nor was any significant effort expended to determine the details. Written accounts of the accident and response, correspondence in the records of some High 26 Cases Group personnel, and personal conversations with some of these individuals provided a reasonable description of the situation during the response.

Results obtained in environmental characterization programs around Palomares for over 15 years following the accident provided an alternative route to assessing intakes and doses. Those estimates are much more realistic when compared with the estimated intakes and doses for other plutonium exposures to workers or members of the public.

7.1 RESULTS FROM ENVIRONMENTAL MEASUREMENTS

The estimated intakes and doses for three scenarios of worker activity indicate that the exposures are well below recommended limits for workers and a small fraction of the dose (10 rem) for which health effects have been reliably demonstrated in humans. The estimates are limited, however, because they represent evaluations using representative scenarios. They do not represent the exposures to any specific individual responder. Additional information on responder activities, time exposed, conditions of exposure, use of personal protective equipment,

and factors that influence intake are needed to develop case-specific assessments. Nevertheless, these estimates form serious concerns about the reliability of estimates from the urinary bioassay data. As a matter of fact, the difficulty in extrapolating urinary concentrations determined at the limits of detection of the analytical methods are well known and are most likely a major contributor to the disparity in the two approaches.

The estimates from the environmental data are very consistent with the results obtained for residents of the Palomares area and with results for Manhattan Project workers. These comparisons lend credibility to the bounds of estimates from the environmental data and support conclusions about the significance of the exposures reached in 1966 through 1968.

7.2 RESULTS FROM URINARY BIOASSAY

The estimated intakes and doses for all groups were unrealistically high as discussed above. Nevertheless the implications of these estimates for effects on health are included to provide some interpretation for what are likely to be upper bound estimates. Furthermore, comments on the analytical methods, case specific information, and other inconsistencies in the data are presented as background for possible reevaluations in the future.

7.2.1 Assessment of Possible Effects

Characterizing the preliminary estimates of intakes and dose are useful only to indicate that many individual cases represent significant to very serious situations when compared to accepted guidelines for management of radiation exposures. About half the estimates exceeded the cumulative dose that would be experienced by anyone in the United States from lifetime exposure to the average background dose (roughly 21 rem (0.21 Sv) over 70 years). Fortunately, the estimates derived from environmental data (Section 6.1.3.2 above), using very conservative scenarios and assumptions, provide upper bound estimates that are well below accepted guidelines and are more consistent with the exposure experience of the local populace on site at Palomares and of industrial plutonium workers. All, but the extreme cases of the estimates, are below the recommended average radiation exposure for members of the public in one year.

7.2.2 Comments on the Estimates

Substantial experience and useful observations arose from the attempts at preparing estimates of plutonium intake and dose from the urinary bioassay data. Those observations and comments are discussed below for each of the groups.

7.2.2.1 High 26 Cases

The intakes and doses discussed in the previous section represent conservative estimates of the intakes and dose equivalents for the High 26 Cases Group. Additional comments are required to put the estimates into perspective. Those comments address the quality of the urine bioassay measurements, assumptions about the type and duration of exposure, the class (type) of material involved and specific details of the duties performed by each individual. Without further details and possible confirmation, permanent assignment of these intakes and doses to the individuals may be premature.

The laboratory analyses performed in 1966 and 1967 represent a comprehensive effort to assess the possible exposure to plutonium. At the time, the urine results used the best available models for estimating body burden. However, methods for estimating intake and deposition of plutonium in the lungs were not well understood. Progress since then allows better estimates to be made now. In fact, deposition in the lungs and the associated dose is the major contributor to the annual dose in the first few years after the exposure. Unfortunately, a very small amount of plutonium in urine can be associated with an intake that produces sizeable doses.

For the cases evaluated, the amount of plutonium in the urine after about one month is more than one million times less than the amount of the intake. That fraction decreases slowly, but steadily, thereafter. The sensitivity of the analytical methods limit the ability to confirm the amount deposited. Samples were collected out to about 15 months following the accident yet the expected excretion curve implies that plutonium excretion would continue beyond that time for actual intakes. More sensitive techniques are now available that could provide new analyses of urine samples.

At 34 years after the accident, the amount excreted per day would be about two million times less than the initial intake. The feasibility of obtaining useful assessments of plutonium uptake by sampling urine now depends mainly on the sensitivity of the analytical techniques and on the ability of the available models to represent human excretion of the plutonium in urine.

Analytical techniques currently available that provide potentially adequate sensitivity include alpha spectrometry, neutron induced fission track analysis (FTA), and mass spectrometry (Wrenn 1994). Alpha spectrometry, which cannot distinguish between ^{239}Pu and ^{240}Pu has nominal sensitivity for both of about 0.02 pCi per sample. That is about the same level available during the resampling conducted in 1966 and 1967. Most mass spectrometry techniques provide about the same sensitivity as alpha spectrometry. Thermal Ionization Mass Spectrometry (TIMS) offers sensitivities of about 0.005 pCi per sample but is tedious and costly. Neutron induced FTA provides sensitivities of about 0.00003 pCi per sample, or about 1,000 times better than alpha spectrometry and routine mass spectrometry. However, FTA is performed at only one or two laboratories.

The biokinetics and urinary excretion models available in ICRP-30 and from Jones (Jones 1985) vary in their ability to model the available data on human excretion at long times after exposure. The Jones model corresponds quite well as recently discussed (Luciani 2000). At 34 years after exposure, the model predicts that the daily urinary excretion would be 10^{-5} of the amount transferred to the blood. As an example, a urine sample with a measured ^{239,240}Pu content of 0.00003 pCi/L would translate into an uptake of 4.2 pCi to the blood from the original inhalation intake. For Class Y plutonium, about 5 percent of the inhaled plutonium transfers to blood. Therefore, the intake would be 84 pCi, which is well below one ALI of 13,500 pCi. Follow-up sampling and analysis using the most sensitive techniques available today, offers a reasonable potential for obtaining useful information. A decision to use the approach should also consider other factors, such as cost, ability to locate and obtain cooperation of response personnel, and limited laboratory availability.

Assumptions were made concerning the type of exposure (single, acute inhalation) and dates of the exposure. For some individuals, this assumption may represent up to several weeks of difference in determining the elapsed time between exposure and collection of samples. The elapsed time is one of the primary parameters for estimating the intake.

The assessment also assumed that the plutonium was PuO_2 and represented by lung Class Y (Type S). All (100%) of the intake was assumed to be from this material. Limited tests were also performed using CINDY assuming a mixed material (50% Class W and 50% Class Y). Those attempts produced lower estimated intakes and doses, however, difficulties with reconciling the approach with experimental confirmation of typical plutonium at Palomares are problematic. In addition, as discussed in Section 3, the cases of mixed plutonium forms also demonstrate a long-term excretion component that is not observed for the data. Never the less, the estimates obtained with the 100% Class Y assumption are higher and therefore conservative.

Finally, these estimates were performed with limited information about the specific activities and times that the individuals were on the site. Efforts to perform a comprehensive search of all records and information, including interviews, were beyond the scope of this effort. Some additional refinement might be possible from an expanded search for more specific information. However, the cost of such an effort should be balanced with the possible benefits from confirmatory measurements of urinary content. Ultimately, credible estimates of intake and dose will depend on an expensive, multi-phased approach involving:

- Urinalysis of selected individuals using highly sensitive techniques to assess the presence of plutonium in their urine.
- Detailed interviews with individuals to develop the details of their exposure circumstances as well as they can recall them.
- Research and evaluation of all available information, especially that collected during the recovery and response phases of the incident, including records available at DOD's Defense Threat Reduction Agency, the Air Force Safety Agency, the Department of Energy, and possibly the appropriate representatives of the Government of Spain.

7.2.2.2 Contamination Cutoff Cases

The intakes and doses discussed in the previous section represent conservative estimates of the intakes and dose equivalents for the Contamination Cutoff Group. The estimates are considered conservative because the methods and data selected tend to overestimate the actual intakes and doses. The additional comments made regarding the High 26 Cases Group apply to these cases as well. Furthermore, confirmation of the possible exposures for this group are very important because this group did not have any measurements taken in late 1966 or 1967, when alpha spectrometry measurements were more commonly used.

7.2.2.3 Repeat Analysis Cases

The intakes and doses discussed in the previous section represent conservative estimates of the intakes and dose equivalents for the Repeat Analysis Cases Group. The estimates are considered conservative because the methods and data selected tend to overestimate the actual intakes and doses. The additional comments made regarding the High 26 Cases Group apply to these cases as well. Furthermore, confirmation of the possible exposures for this group are very important because this group did not have any measurements taken in late 1966 or 1967, when alpha spectrometry measurements were more commonly used.

7.3 COMPARISON OF INTAKES AND DOSES TO OTHER PLUTONIUM EXPOSURE CASES

The results can be evaluated for reasonableness by comparing them to other plutonium exposure situations. Two such reported cases are the evaluation of the citizens of Palomares by a Joint Spanish-United States effort since the accident, and the follow-up of Manhattan Project workers who received exposures to plutonium at Los Alamos. In addition, measurements of environmental plutonium at Palomares provide data for performing independent estimates of the intakes and doses for the accident response force.

7.3.1 Dose Estimates for Residents of Palomares

Since the accident, the Government of Spain has conducted a program to monitor the residual radioactivity at the accident site. That effort has included measurements of air concentrations of plutonium, soil contamination levels, and assessment of intakes and doses to the population.

During 1966, 59 people provided urine samples on three occasions. Those samples indicated the possibility of contamination (Iranzo 1997). In 1967 samples were collected in Madrid under controlled conditions. Of those, 23 exceeded the minimum detectable level of 0.02 pCi/ day. During the ensuing years, additional samples have been collected from a larger group of Palomares citizens and analyzed. The results indicate that 45 individuals who may have received intakes during the initial clean-up showed intakes that represented committed effective dose equivalents of 2 rem to 20 rem (0.02 to 0.2 Sv) (Iranzo 1987). That range includes the lower portion of the results obtained for responders. In addition, the early concerns for sample contamination and efforts to mitigate the possibility support similar concerns for the Air Force urine sampling. Although, the Air Force resample effort was conducted away from the accident site, the possibility exists that samples provided in mid to late 1966 and early 1967 may have been influenced by continued sample contamination.

7.3.2 Manhattan Project Worker Evaluations

During the Manhattan Project, 26 white, male adult workers received intakes of plutonium primarily by inhalation. Reports of follow-up studies of that group have indicated continuing refinement of the estimates of their plutonium deposition. A recent report provided the results of 50 years of follow-up. The report indicated that the depositions for the 26 individuals ranged from 1.35 nanocuries (50 Bq) to 85.86 nanocuries (3,180 Bq) (Voelz 1997). The corresponding effective doses ranged from 10 rem to 720 rem (0.1 to 7.2 Sv). If those exposures occurred by inhalation, the intake would have been approximately 20 times higher than the deposition or 27 nanocuries to 2.3 microcuries. Although the range of exposures is similar to the preliminary estimates for Palomares response personnel, the responders' exposures were unlikely to approach those of the Manhattan Project workers. Responders would have handled the much different (lower) quantities and forms of plutonium for much shorter times than the Manhattan Project workers. Those workers performed continuous, industrial operations on a daily basis over several years under what have been called "primitive conditions".

The results of follow-up of citizens of Palomares and Manhattan Project workers indicate the range of doses from exposures received under field conditions and those received in laboratory or industrial conditions. It seems reasonable to consider the results for the Palomares citizens as more representative of the kind of exposure conditions experienced by the response personnel

because both were exposed to the same or similar sources, while the Palomares residents were exposed for many years. Consequently, the results for responders that exceed even a fraction of the upper range of CEDE (20 rem/0.2 Sv), may well represent sample contamination or other artifacts. If that is the case, additional sampling and analysis of a carefully selected subset of the response force today offers an attractive approach to confirming the deposition and associated doses.

8 CONCLUSIONS AND RECOMMENDATIONS

Records of urinary ^{239}Pu and gross alpha radioactivity of samples collected from responders to the Palomares nuclear weapons accident were evaluated for possible use in calculating estimate radioactivity intakes and committed effective dose equivalent using accepted models. Data were reviewed and individuals assigned to four groups according to the amount and reliability of the data. The groups included:

- The High 26 Cases Group that included 26 individuals identified for resampling for 18 to 24 months after the initial phase of sampling in 1966.
- A Repeat Analysis Cases Group that included 54 individuals who either had submitted samples that were reanalyzed using more specific methods (alpha spectrometry), or who were resampled.
- A Contamination Cutoff Cases Group that included 313 individuals with results that were below a reasonable, assumed cutoff level of 0.1 pCi per day.
- A Remaining Cases Group that contained 1,063 records that were not otherwise evaluated and that were strongly suspected of contamination from collection on site.

Two current computer methods were tested and used to estimate intakes of plutonium by acute inhalation exposure. One method (CINDY) employed the ICRP-30 system for limiting internal dose. The other method (LUDEP) implemented the new respiratory tract model described in ICRP-66 and the organ/tissue weighting factors of ICRP-60.

Plutonium intake and dose values were estimated for all of the High 26 Cases Group, the 54 individuals in the Repeat Analysis Cases Group, and 313 individuals in the Contamination Cutoff Cases Group. The intakes and doses ranged from below annual occupational limits to more than the 50 rem (0.5 Sv) guideline for cumulative dose for workers. Some doses ranged as high as several hundred rem. However, when compared with estimates derived from environmental measurements, dose estimates for Palomares citizens, and dose estimates for Manhatten Project workers, these preliminary estimates seen unreasonably high in many cases. Additional efforts are needed to reconcile the results from the urine data with the levels that can be reasonably supported by the environmental data and experience with other exposed people.

Several future actions should be considered to further refine these initial estimates.

1. Additional effort is needed to reconcile the estimated intakes and doses derived from the urinary bioassay data with the estimates from environmental measurements. A targeted effort that includes participant activities, participant interviews, urine and other appropriate plutonium analyses using current techniques, medical records review, and modeling should be considered.

2. The results of this effort should be communicated to responders, veterans organizations, and other interested parties using appropriate information that clearly confirms the conclusions of the original medical evaluation program, recognizes the difficulties in preparing updated intake and dose estimates, and outlines the options for strengthening the estimates.

3. Further contacts with the Department of Energy for comparison with evaluations of their personnel who responded to this accident could provide useful data. The effort should be summarized in a companion document that conveys the details of the project and its potential effects on health in an easily understood manner. That document should be made available to any of the responders who desire a copy.

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APPENDIX A

GLOSSARY

APPENDIX A GLOSSARY

Absorbed Fraction (AF) – The fraction of energy emitted as a specified radiation, R, in a specified source tissue, S, which is absorbed in a specified target tissue, T. [AF (T←S)R]

Accuracy – The comparison of a measurement to the true value of a parameter. It is a function of both bias and precision.

Activity Median Aerodynamic Diameter (AMAD) - The diameter in an aerodynamic particle size distribution for which the total activity above and below this size are equal. A log-normal distribution of particle sizes is assumed.

Activity Median Thermodynamic Diameter (AMTD)– The particle Diameter, D_{tn} (thermodynamically classified) for which 50 percent of the total airborne activity, is associated with particles of thermodynamic diameter is greater than the AMTD.

Aerodynamic Diameter (d_{ae}) – The diameter (μm) of a unit density (1 g cm^{-3}) sphere that has the same terminal settling velocity in air as the particle of interest. Same as AED.

Aerodynamic Equivalent Diameter (AED) - The diameter of a sphere, in μm, of unit density (1 g cm^{-3}) that has the same terminal settling velocity in air as the particle of interest (a 1 μm AED particle has 1000 times the volume of a 0.1 μm AED particle).

Aerosol – A suspension of fine solid or liquid particles in a gaseous medium.

Airborne Concentration – The activity of particulate matter or material in a unit volume of aerosol, usually expressed in μCi cm^{-3}, μCi mL^{-1} or μCi m^{-3}.

Annual Limit on Intake (ALI) – The activity in μCi of a radionuclide which taken alone would irradiate a person represented by reference man, to a limit established by a regulatory agency for each year of occupational exposure.

Becquerel (Bq) – the International System of Units adopted unit for radioactivity. One Bq is equal to a radioactivity of 1 nuclear transformation per second (ntps).

Biokinetic Model – A set of mathematical relationships formulated to relate the intake of a material to the uptake, distribution, and retention of the material or radionuclide in various organs and tissues of the body. Some models include subsequent excretion from the body by various pathways.

Breathing Zone – The region adjacent to a worker's nose and mouth from which air is drawn into the lungs while he/she performs the assigned work.

CINDY – Code for Internal Dosimetry is a computer program that addresses the radiation protection aspects of Department of Energy orders and Nuclear Regulatory Commission regulations by implementing the approach described in ICRP Publication 30.

Class – The lung or inhalation classification scheme, developed in ICRP Publication 30, for inhaled material according to its rate of clearance from the pulmonary region of the lung. Materials are classified as D, W, or Y, which applies to a range of clearance half-times: for Class D (days) of less than 10 days; for Class W (weeks) from 10 days to 100 days; and Class Y (years) of greater than 100 days.

Clearance Pathway – The route by which material that is deposited in the lungs can move into the blood, lymph nodes or bronchi.

Committed Dose Equivalent (CDE) ($H_{T,50}$) – The dose equivalent to organs or tissues (targets) of reference (T) that will be received from an intake of radioactive material by an individual during the 50 year period following the intake.

Committed Effective Dose Equivalent (CEDE) ($H_{E,50}$) – The sum of the products of the tissue weighting factor and the radiation weighting factor or quality factor applicable to each of the body organs or tissues that are irradiated and the committed dose equivalent to these organs or tissues. [$H_{E,50} = \sum w_t (H_{t,50})$].

Curie (Ci) – A unit of radioactivity. One Ci is equal to that quantity of radioactive material in which there are 3.7×10^{10} nuclear transformations per second (ntps) or 3.7×10^{10} Becquerels (Bq). One microcurie (1μCi) is equal to 3.7×10^4 ntps or one-millionth part of a Ci.

Derived Air Concentration (DAC) – The concentration of a radionuclide in air, which breathed or inhaled alone for 1 work yr (2000 hrs) would irradiate reference man to the radiation safety limit for occupational exposure. The DAC equals the ALI of the radionuclide divided by the volume of air inhaled by reference man in a working year (i.e., 2.4×10^3 m^3).

Detriment – The identification and where possible the quantification of all the deleterious effects of exposure to ionizing radiation. Total detriment is the sum of the contributions due to fatal cancers, non-fatal cancers, and severe hereditary disorders weighted for life lost.

Disintegrations per Minute (dpm) – A rate of spontaneous emission of particles and energy from the unstable nucleus of an atom. The curie is a unit of activity quantifying this process of radioactive decay.

Dose Assessment – The process of assessing/estimating the radiological dose and associated uncertainty based on best available information. Included in this dose estimate, through use of exposure scenarios, source term data, bioassay results, monitoring or radiological survey data, and pathway analysis.

Effective Half-Life – The time required for the amount of a contaminant deposited in a living organism to be diminished to 50 percent as a result of the combined action of radioactive decay and biological elimination.

Elimination – The removal of material from the body via urine, feces, sweat or exhalation. Excretion usually refers to elimination via urine or feces.

Equilibrium, Radioactive – The state that prevails in radioactive series when the ratios between the activities of two or more successive members of the series remains constant.

Equivalent Diameter – The diameter of the sphere that would have the same value of a particular physical property as that of the irregular particle.

Exposure – The act of being exposed to a contaminant.

Exposure Assessment – The process of assessing/estimating the exposure to a contaminant and associated uncertainty, based on best available information. Included in this exposure estimate, through use of exposure scenarios, source term data, bioassay results, monitoring data, and pathway analysis.

Extrathoracic Fraction – The mass fraction of the inhaled particles which do not or fail to penetrate beyond the larynx.

Geometric Standard Deviation – For a log-normal distribution, the exponential of the standard deviation of the associated normal distribution (always ≥ 1).

Inhalabilty – The ratio of the number concentration of particles with a particular diameter inspired through the nose or mouth to the number concentration of particles with the same aerodynamic diameter present in the inspired volume of ambient air.

Inhalable Fraction – The mass fraction of the total airborne particles which are inhaled through the nose and mouth.

Intake – The total amount of material that enters the body through the principal exposure routes of inhalation, ingestion, or skin wounds.

Log-Normal Distribution – A distribution in which the logarithms of a variable (such as particle size) is normally distributed.

Lower Limit of Detection (LLD) – The smallest amount of mass or radioactivity that yields a statistically significant net result above the laboratory method background.

LUDEP – Lung Dose Evaluation Program is a computer program developed by the National Radiological Protection Board of the United Kingdom that implements the Respiratory Tract Model recommended by the ICRP's Task Group on Lung Dynamics as adopted in 1993 and published in ICRP Publication 66.

Mass Concentration – The mass of particulate matter or material in a unit volume of aerosol, usually expressed in $\mu g\ m^{-3}$, $mg\ m^{-3}$, or $g\ m^{-3}$.

Mass Median Aerodynamic Diameter (MMAD) – The aerodynamic diameter of a particle having a median mass i.e., the masses of particles above and below this vehicle are equal.

Maximum Permissible Concentration (MPC) – A concentration for a radionuclide (established in ICRP-2) in air or water set to keep dose to the critical organ from exceeding the annual limit. The annual limit applied over an intake period of 50 years.

Maximum Permissible Body Burden (MPBB) – A limit associated with Maximum Permissible Concentration that was the amount of material in the body that would not cause an organ dose to exceed the annual limit to the critical organ.

Metabolic Model – A mathematical description of the behavior of inhaled or ingested radionuclides in the metabolic process of cells, tissues, organs and organisms (humans). It is most frequently used to describe its distribution among tissues/organs and elimination/excretion.

Micrometer (μm) – A unit of measure. One micrometer (1 μm) is one millionth of a meter (1×10^{-6} m).

Non-Stochastic Effects –Those effects for which the severity of the effect varies with the dose received and for which a threshold may exist. The following are examples of non-stochastic somatic effects that are specific to particular tissues; cell depletion in the bone marrow causing hematological deficiencies, and gonadal cell damage leading to impairment of fertility. For these changes to occur, the severity of the effect depends on the magnitude of the dose received, and there is a threshold of dose below which no detrimental effects are observed.

Parent – A radionuclide that, on nuclear transformation (disintegration), forms a specified nuclide either directly or as a later member of a radioactive series.

Particle Density – The mass of the particle itself per unit volume, usually expressed in g cm^{-3}, mg m^{-3}.

Particle Dissolution Rate – The rate at which the change of a particle from a solid to a liquid form takes place.

Particle Transport – The process that clear material from the respiratory tract to the gastrointestinal tract and to the lymph nodes, and move material from one part of the respiratory tract to another.

Precision – The repeatability or reproducibility of a measurement. Precise results have small random errors.

Progeny – The decay product or products resulting after a radioactive decay or a series of radioactive decays of the parent radionuclide. The progeny can also be radioactive, and the decay chain will continue until a stable nuclide is formed.

Rad - The special unit of absorbed dose. One rad is equivalent to an absorbed dose of 0.01 J kg^{-1} or 0.01 Gray (Gy).

Reference Man – A male individual between 20 to 30 years of age weighting 154 pounds (70 kg) is 5.6 feet (1.7 m) in height, and lives in a climate with an average temperature of 50°F to 68°F (10°C to 20°C). He is a Caucasian and is a Western European or North American in habitat and custom (ICRP Publication No. 23; updated by ICRP Publication No. 66 and ICRP Publication No. 70).

Rem - The special unit of any of the radiation quantities expressed as dose equivalent. The dose equivalent in rem is equal to the absorbed dose in rad multiplied by the quality factor or radiation weighting factor. One rem equals 0.01 sievert (Sv). (1 millirem (mrem) is 1/1000 of a rem.)

Respirable Fraction (RF) – The mass fraction of the inhaled particles which penetrate to the unciliated airways of the respiratory tract.

Respiratory Tract Clearance – The removal of material from the respiratory tract by particle transport and by absorption into blood.

Respiratory Tract Deposition – The initial process determining how much of the material in the inspired air that remains in the lungs after exhalation. Deposition of material may occur during both inspiration and inhalation.

Respiratory Tract (Lung) Model – The model that describes the behavior of particles in the respiratory tract of man. This model was developed by the ICRP's task group on lung dynamics and published in ICRP Publication 30. This model is used in the CINDY program; however, LUDEP (an alternate computer program also used) uses the ICRP's new lung model published in ICRP Publication 66.

Resuspension - The transport of particles from surfaces (inside and environmental) back into the atmosphere.

Risk – The characterization of a situation or action wherein two or more outcomes are possible, the particular outcome that will occur is unknown, and at least one of the possibilities is

undesired. Risk is also the sum of the possible alternative numbers of injuries or fatalities weighted by their probabilities.

Sensitivity Analysis – The sensitivity of the model prediction to selected perturbation of model parameters.

Solubility - The ability of a substance to form a solution with another substance. Normally lung or tissue fluid is considered the fluid of choice.

Source Tissue – Tissue (may be a body organ) that contains a significant amount of a radionuclide following an intake of that radionuclide into the body.

Specific Absorbed Fraction – The fraction of energy that is emitted as a specified radiation type (alpha, beta, electron or photons) in a source organ/tissue that is absorbed in 1 g of a target organ/tissue.

Stochastic Effect –Those effects for which the probability of an effect occurring, rather than its severity, is regarded as a function of dose. Both hereditary effects and carcinogenesis are stochastic effects.

Target Tissue – Tissue (which may be a body organ) in which radiation is absorbed.

Thoracic Fraction – The mass fraction of the inhaled particles which penetrate beyond the larynx.

Tracheobronchial Fraction – The mass fraction of the inhaled particles which penetrate beyond the larynx, but which do not or fail to penetrate to the unciliated airways of the respiratory tract.

Transfer Compartment – The compartment introduced (for mathematical convenience) into the biokinetic model to account for the translocation of radioactive material through the body fluids from where they are deposited in tissues or excreted.

Translocation – The movement of material, that has been deposited respiratory tract, by dissolution and absorption into the blood.

Transportable Half-Time – The amount of time for half of the contaminant to be transferred to a transfer compartment.

Transuranic (TRU) – An element with an atomic number greater than that of uranium. Neptunium has an atomic number of 93 and plutonium has an atomic number of 94.

Uncertainty Analysis – The analysis of the uncertainty in model prediction. The production of a Probability Density Function (PDF) that describes the confidence with which it can be claimed that some characteristic of risk (probability, severity, episodic frequency, or total number of effects) lies between two values.

Uptake – That quantity of material that is taken up or enters the body from the location of intake. The routes of entry into the body are from the respiratory tact, gastrointestinal tract, absorption through the intact skin, injection, or via a wound.

Weighting Factors –

- Organ or tissue weighting factor (w_t) – The multiplication factor by which the committed dose equivalent (CDE) in an organ/tissue is multiplied to yield the Committed Effective Dose Equivalent (CEDE). This factor represents the relative contribution of that organ or tissue to

the total detriment due to these effects resulting from uniformed irradiation of the whole body. (The w_t values are those given in ICRP Publication No. 26 for CINDY assessments and in ICRP Publication 60 for LUDEP assessments.)

- Radiation Weighting Factor (w_R) – A factor [quality factor (Q)] which is dependent on the type and energy of the radiation and is independent of the exposed organ/tissue. As used in the calculation the average quality factor is used for both external and internal radiation. (The Q values are those given in ICRP No. 26.)
- Least Squares Regression Weighting Factor (w_i) – A factor that determines the relative significance of the i^{th} data point in a least squares regression analysis. The factor is generally user selected to represent a measure of the confidence or estimated error of the data point.

APPENDIX B

AVAILABLE AIR FORCE DATA

APPENDIX B AVAILABLE AIR FORCE DATA

B.1. INTRODUCTION

This effort to re-evaluate possible doses to those who responded to the Palomares nuclear accident required a complete and careful review and assessment of available data. Since the accident occurred over 33 years ago, this review depended on the ability to identify relevant records, reports and other data to form as complete a picture of the situation as possible. Initial efforts focused on accumulating and reviewing records provided by the Air Force Medical Operations Agency (AFMOA) at Bolling AFB, DC and the Institute for Environmental, Safety, And Occupational Health Risk Analysis (IERA) at Brooks AFB, TX. IERA succeeded the USAF Radiological Health Laboratory (RHL) as the Air Force's primary radiological consultant laboratory and custodian of personnel radiation exposure records in the USAF Master Radiation Exposure Registry. Initial contact with both AFMOA and IERA identified and provided information on the availability of Palomares records. IERA and AFMOA provided their records in the form of:

- Air Force Forms with laboratory analytical and exposure details of the nasal swipe and urine samples submitted and processed.
- Complete case files for the 26 individuals identified for follow-up in 1966 and commonly referred to as the "High 26".
- A Microsoft Excel spreadsheet prepared by IERA staff that contained the data from those Air Force Forms, and some data related specifically to the 26 individuals (referred to as the "High 26" who were considered as having the highest exposures.
- Copies of reports of the accident response, RHL documents on the evaluation of exposures by urinalysis, and selected publications from journals and conference proceedings.

Those records formed the basis for significant effort: to understand what information the various records contained; to determine how the data were used in the initial evaluations; to identify data gaps, inconsistencies, and concerns with the use or interpretation of the data; and to prepare the records for input to this intake and dose assessment effort. This appendix discusses the results of this review and the modifications and assumptions made to the data for use in the dose assessment. The appendix provides specific details of the three types of records and the concerns they generated, as well as efforts to correct, improve, or interpret those records for this project.

B.2. TYPES OF RECORDS KEPT

The records prepared and maintained by the Air Force consisted of forms, computer spreadsheets, and written correspondence and reports of activities. This section provides details of the forms and the data they contained.

B.2.1. Forms

RHL, as the central laboratory for providing radiological services to Air Force units, applied their laboratory processes with some modifications to this accident. RHL, a sub-unit of the Air Force Logistics Command (AFLC) at the time, used AFLC sanctioned forms for recording the data and results of samples processed. Three series of forms were identified in the records provided: AFLC Form 1165, Internal Dosimetry Data (May 66), AFLC Form 1165, Radiological Sample Data (May 66), and AFLC Form 1165, Radiological Sample Data (Jul 67). Although similar in design and content, these three forms apparently evolved over the course of the laboratory effort on Palomares and other services at the time.

B.2.1.1 AFLC Form 1165, Internal Dose Data (May 66)

The AFLC Form 1165, Internal Dose Data contained data about the individual who submitted the sample, radiation measurement data for urine, radon (breath) (sic), and feces/blood samples. The form provides areas for recording counting data, instrument data, and other factors. For Palomares, the form primarily recorded urine sample data and results. Figure B-1 illustrates an example AFLC Form 1165.

Annotated comments (callout boxes) on Figure B-1 draw attention to several features of the form and its use for the Palomares Accident. In addition to basic identifying information (name, and Social Security Number (SSN)), the form typically contained an entry for the Air Force Serial Number (AFSN) as an additional entry. At the time, the SSN had not become an official identifier for Air Force military personnel.

Comments about certain uses of the form pertain to the review and analysis of data contained on these forms for possible use in the reassessment project. These include (identified by text in callout box on Figure B-1):

- **Basic Counting Data**: this area provides spaces for the entry of Counter Identification (N), Counter Background (cpm), Counter Efficiency (%), and other pertinent counting information. Additional data were often recorded in this area. For example, the entry for Counter background - 0.03 (900) – refers to the counts per minute (0.03) and the time the background was counted (900 minutes).
- **Notation of Elapsed Days**: this entry – t = 49 – refers to the elapsed time (in days) between the assumed exposure and the date the sample was collected. According to other records, the exposure date was generally assumed to occur on the day that was the midpoint of an individual's time on station.
- **Exposure Date Entry**: an entry with the known or estimated dates of exposure. Often this represented the actual calendar time at the site performing duties. In this case, the entry contains a range of dates.

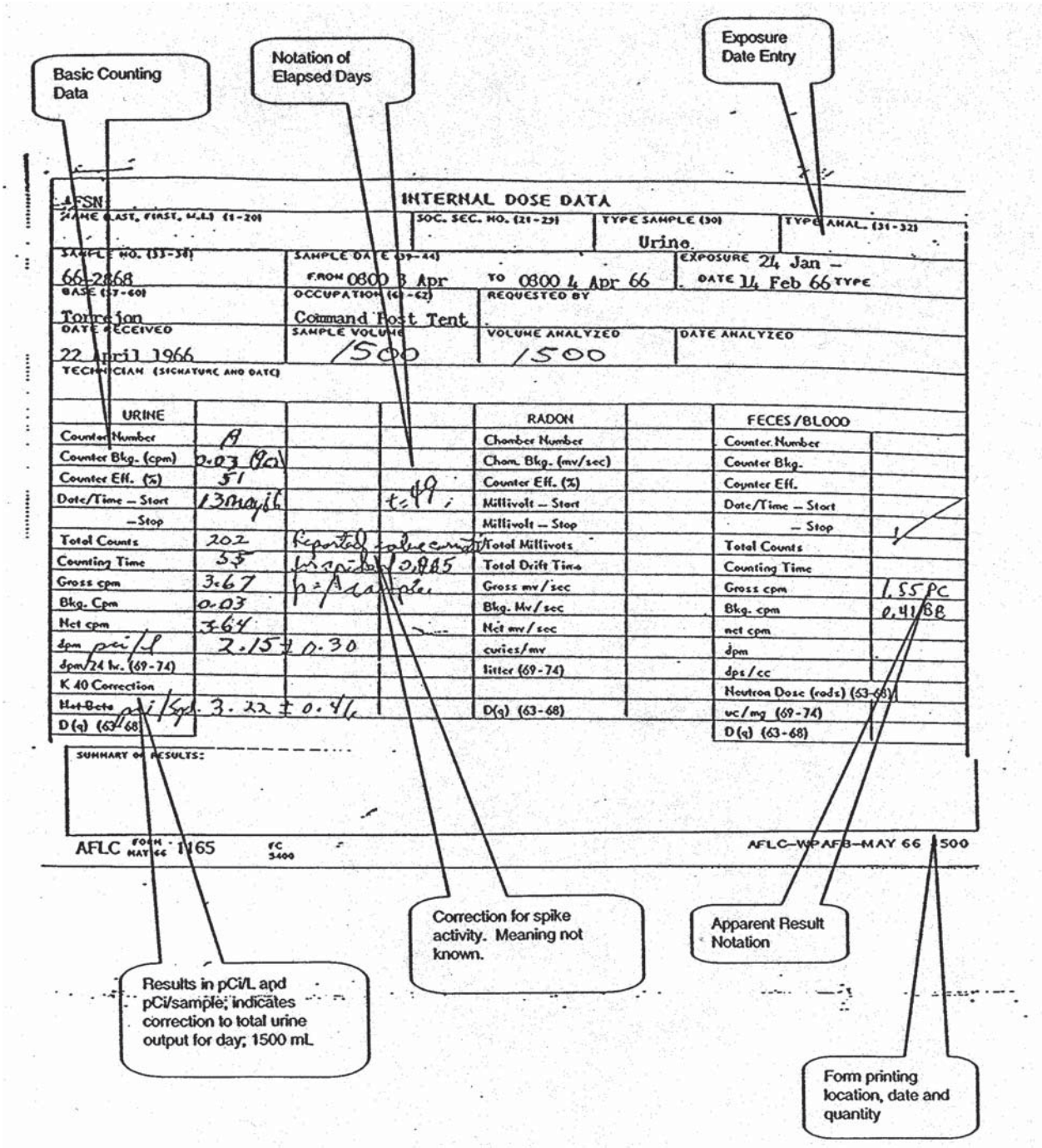

Figure B- 1. AFLC Form 1165, Internal Dose Data (May 66)

- **Results, etc.**: this section demonstrates flexibility in use of the form by hand written notations of the meaningful result. In this example, the result (2.15 ± 0.30 pCi/L) is expressed in activity per unit volume as picocuries per liter (pCi/L) and as activity per sample (pCi/spl). In this case, the pCi/spl means the total gross alpha activity excreted in one day as required by equations relating content in urine to systemic body content. In addition to the actual value, the estimated error (based on 95% confidence level of the counting data only) is also shown.
- **Correction for spike activity**: This notation apparently refers to a factor applied to correct for added ^{236}Pu radioactivity. The exact meaning of this notation has not been determined for gross alpha measurements.
- **Apparent Result Notation**: an entry in the feces/blood section that apparently represents an independent evaluation of the radioactivity content and an estimate of the fractional systemic body burden (0.44 BB).
- **Form printing location, etc.**: represents the place (WPAFB – Wright-Patterson Air Force Base), date (May 66), and quantity of forms printed (4500). This is an administrative requirement.

Figure B-2 provides a second example of an AFLC Form 1165, Internal Dose Data. For this case, three features are discussed.

- **Background counts, etc.**: this form clearly shows the entry of the counter background rate and counting time.
- **Exposure Date Entry**: this form contains one date rather than a range. Based on personal conversations with the individual, he arrived at the accident site early on 18 Jan 66 so the date of 19 Jan 66 is reasonable. Also, the individual said that he stayed at the site until close to the end of the operation. Therefore, a sample date of 18 Mar 66 could represent his last sample while on site. In fact that is the case.
- **Apparent Result Notation**: this entry refers to written notation ($D_R = 6.59 \times 10^{-3}$ μc). The notation D_R is identical to the notation for retained body burden in Langham's excretion equation for plutonium. That entry apparently denotes a retained body burden of 0.00659 microcuries or about 15%.

The previous examples provide the basis for further investigating the relevance of the data on these forms. The relevance may be particularly crucial because these forms represent data for some of the earliest samples collected; especially those collected on site at Camp Wilson that had a very high potential for sample container contamination as referred to by Odland (Odland 1968a and Odland 1968).

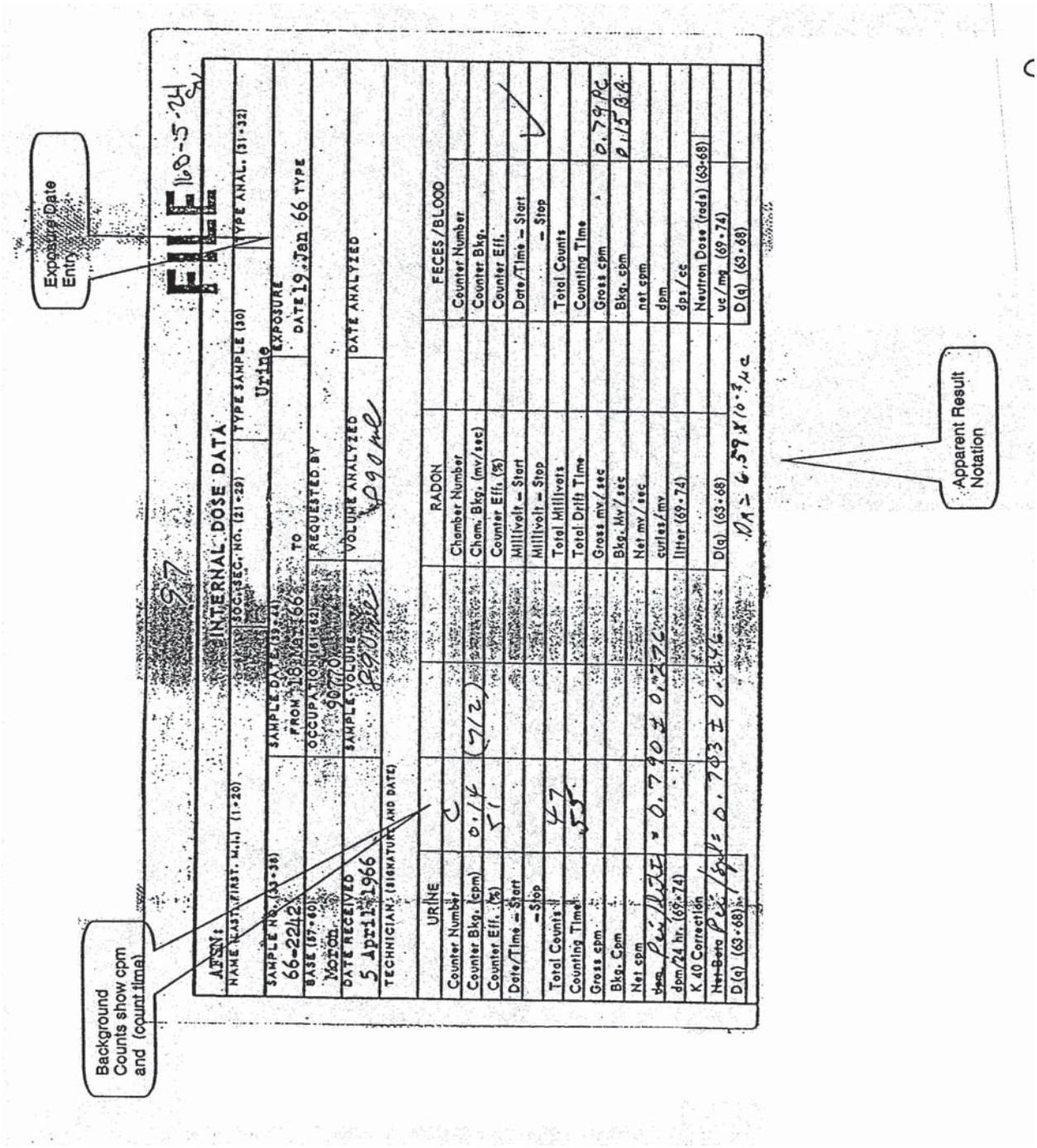

Figure B- 2. Another Example AFLC Form 1165, Internal Dose Data (May 66)

B.2.1.2 AFLC Form 1165, Radiological Sample Data (May 66)

The AFLC Form 1165, Radiological Sample Data (May 66) was apparently also used during the same time period as the previous form. However, our review indicates that this form applied primarily to samples analyzed by alpha spectrometry. Figure B-3 provides an example of this form and contains notations on several interesting features. These features include:

- **Alpha Spectrometry Counting Information:** This section of the form provides room for recording specific information about the radioactivity counting process. Entries include: identification of the radionuclide (^{236}Pu and ^{239}Pu) in separate columns; counter and efficiency (SPEC 2, 24.3); total counts and minutes for each (400, 571, 1 are the time, and the counts in the ^{236}Pu and the counts in the ^{239}Pu); background counts and time (800, 1, 1 as time, counts in the ^{236}Pu area and counts in the ^{239}Pu area). These entries are self-explanatory for the most part.
- **Elapsed Time in Days:** the time from exposure (assumed as midpoint of time at the accident site) to sample collection.
- **Exposure Time Entry:** An entry of the presumed exposure period. This example contains only the entry "66", presumably indicating the year 1966. No day or month information is entered.
- **Calculated Result:** the results of calculating the radioactivity. In this case entered as (Fci/Spl 4.5 ± 10.0) indicating 4.5 femtocuries per sample with an estimated counting error of 10.0 femtocuries per sample. Other evaluations indicate that for alpha spectrometry RHL calculated and reported the estimated error at the 68% confidence level. In this example, the error is greater than the calculated result.
- **Reported Results:** the result formally reported for this analysis. In this case the result was reported as No Detectable Activity (NDA) meaning that the sample result was less than the estimated error.

Observations about other data on this example reveal details of the processes used in analyzing samples. For instance, the Sample Volume (2000 mL) and the Volume Analyzed (1000 mL) indicate the standard practice that used one-half a submitted sample's volume thereby retaining a portion for further confirmation or reanalysis if laboratory difficulties were encountered.

B.2.1.3 AFLC Form 1165, Radiological Sample Data (Jul 67)

This data form represents an evolution of the previous two versions of the AFLC Form 1165. However, the form retains the same essential data presented on a piece of letter sized (8-1/2" × 11") card stock. This revised form retains the identifying information, but expands on and reformats the basic radioactivity counting and results information. Figure B-4 provides and example of this version of the form. Interesting features on the form are noted as before and include:

- **Gross Alpha Information:** this section contains the same information about the alpha counter data. In this case, total counts and time appear to be reversed; i.e. for TOTAL CTS AND TIME, the entries are 55 and 155. The first (55) was the RHL standard time for

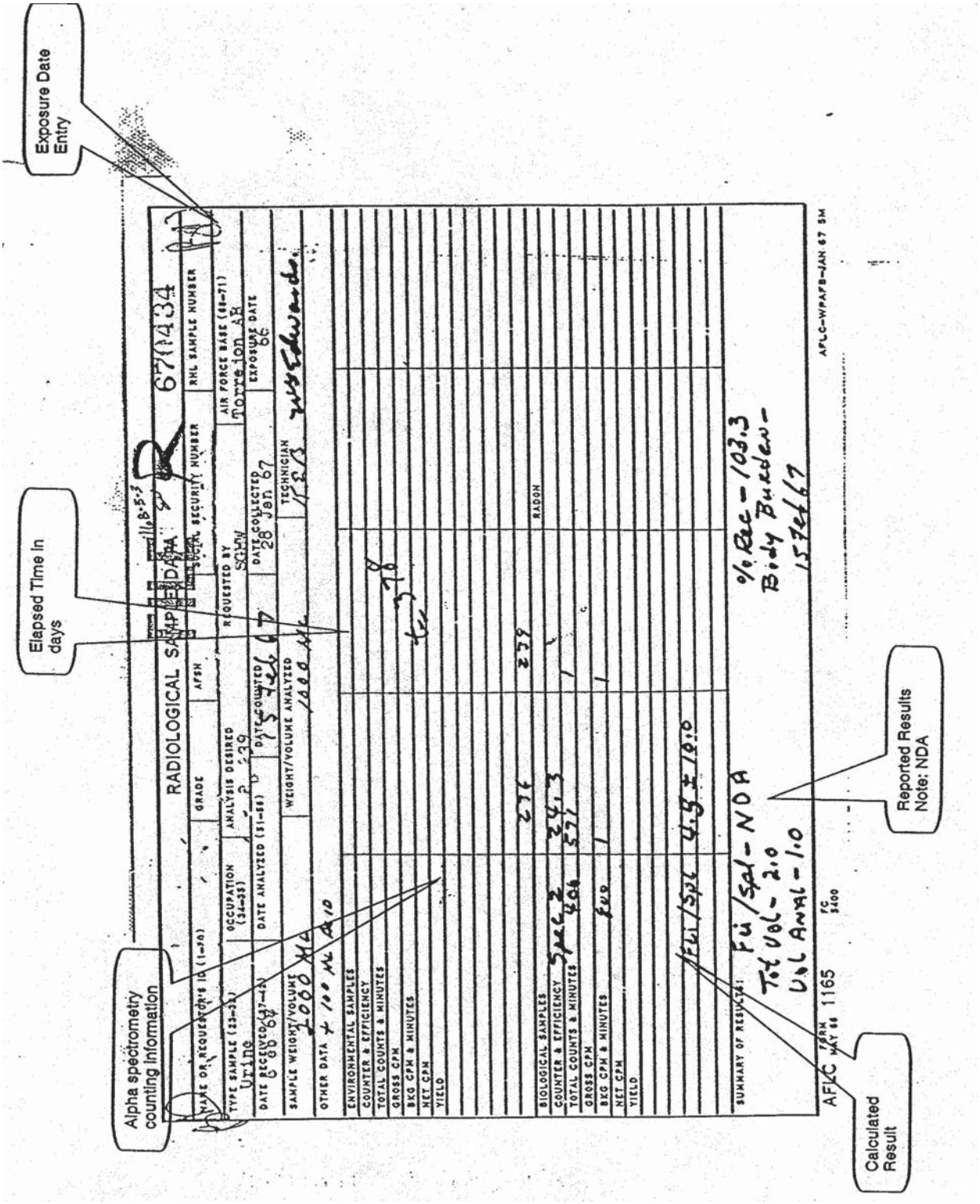

Figure B- 3. AFLC Form 1165, Radiological Sample Data (May 66)

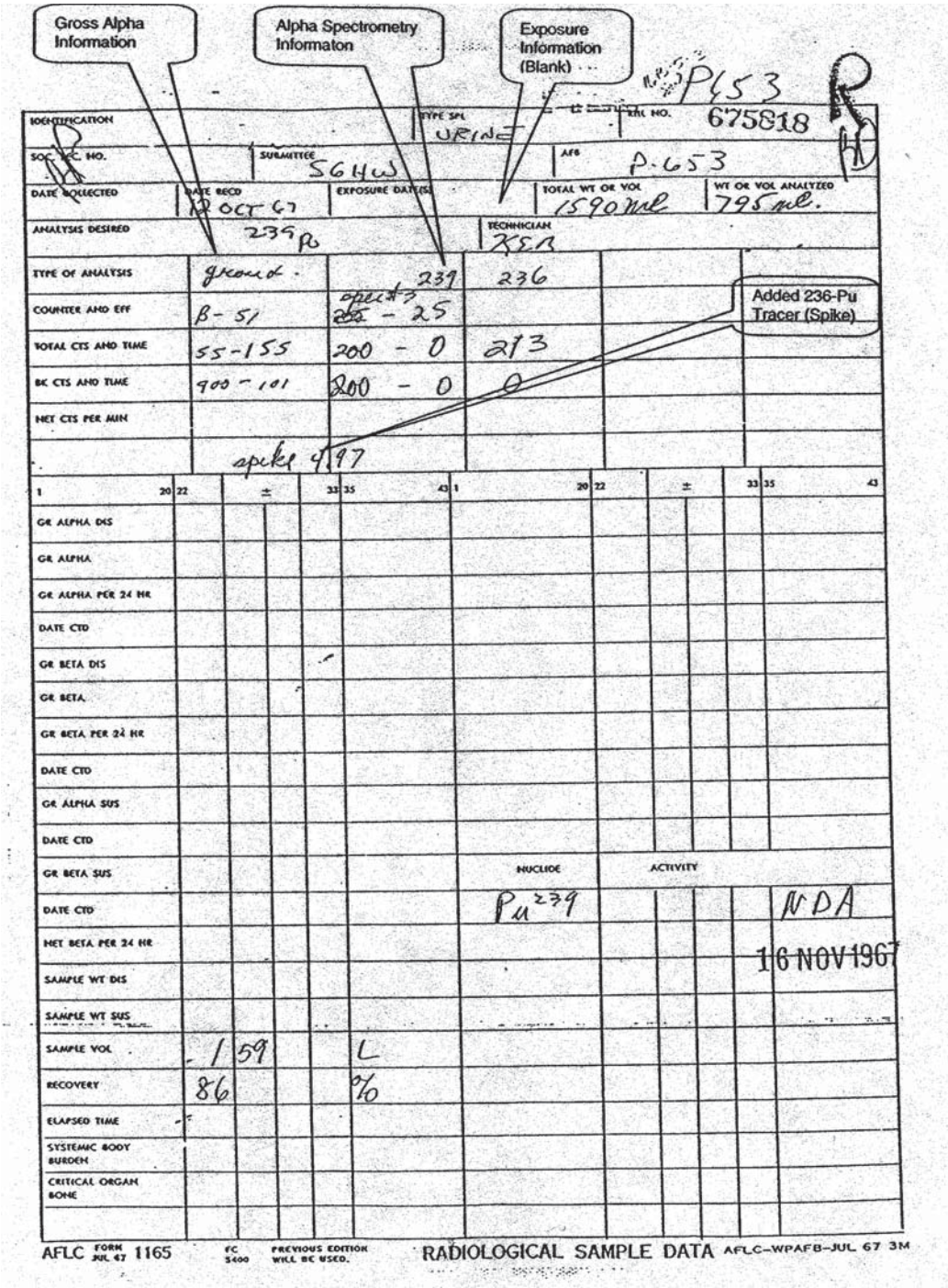

Figure B- 4. AFLC Form 1165, Radiological Sample Data (Jul 67)

counting gross alpha samples. So, the second entry (155) represents the sample counts. Similar comments apply to the background entries.

- **Alpha Spectrometry Information:** Similar information for calculating the results from the alpha spectrometry counting are included here. The counts and the counting time are interchanged as above.
- **Add ^{236}Pu Tracer (Spike):** the entry indicates the amount (in disintegrations per minute – dpm) of tracer added to the portion of the sample taken for analysis. This value is used in calculating the chemical recovery.

The preceding discussion about the forms provides the foundation for understanding the evaluation process applied to analyzing entries in the spreadsheet discussed in the next section. Clearly, consistency among the entries on the data forms and the entries in any final data set would be required. The data cards formed the only permanent record available of the actual data generated at the time of the incident. Consequently, they provided the primary means for verifying information from other sources; at least when the data on the cards were unambiguous.

B.2.1.4 Informal Data Records

An informal, handwritten record appeared in the case files of the High 26 group. That record was prepared on available paper scrap and was apparently used as source data for transfer to punched data cards. RHL used punched data cards as the primary medium for maintaining data and results for later use in organizing, sorting, reporting, and transfer to computer tape.

Figure B-5 illustrates one example of that form. The form contained an entry at the top (3826) that represents the sequential portion of the RHL assigned sample number (66-3286). The form also contains six numbered entries. The meaning of those data contained in those entries is explained in Table B-1.

Table B- 1. Data contained on informal RHL form.

No.	Meaning
1.	Urinary excretion pCi/24 hr and error
2.	Chemical Recovery (%)
3.	Total Sample Volume in Liters (L)
4.	Days elapsed from exposure to sample
5.	Day of Year Sample Completed (6256 means 256th day of 1966 or September 13, 1966
6.	Fraction of a systemic body burden

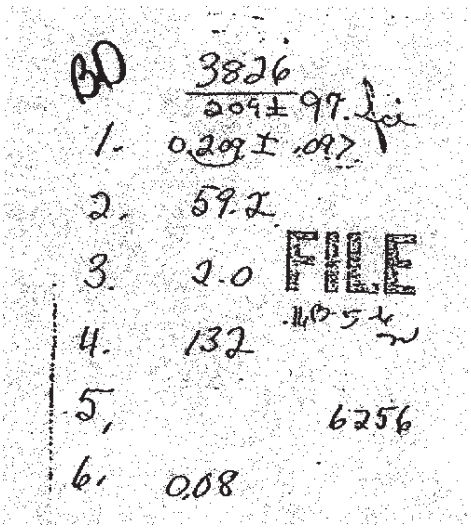

Figure B- 5. Informal RHL data form.

B.2.2. Spreadsheet

During an initial visit, IERA representatives provided a copy of a Microsoft EXCEL spreadsheet that they had prepared. The spreadsheet contained the basic data transcribed from the hardcopy data forms into the spreadsheet. Table B-2 explains the data items in the spread sheet. Figure B-6 contains an example of one page of the spreadsheet to illustrate the items of information transferred to the sheet. The individual names, Social Security Numbers, and AFSNs have been masked on this example for privacy reasons.

The spreadsheet contains information for 1,758 entries on 1,555 individuals.

Table B- 2. Data Items in IERA spreadsheet

Data Item	Meaning
Name:(Last, First, M.I)	Individual Name
SSN:	Social Security Number
AF ID # :	Air Force Service Number
Type Sample	Type of Sample – urine, nasal swipe, fecal, etc.
Type Anal.	Type of analysis performed – gross alpha, ^{239}Pu
Sample No.	Sample Number assigned by RHL
Sample Date:	Date the sample was collected.
Base:	Base of assignment of the person sampled.
Date Recived (*sic*)	Date the sample was received at RHL
Sample Volume	The total volume of the sample in Liters or milliliters
Sample Analyzed	Volume of sample used in a specific analysis procedure
Date Analyzed	The date the analysis was completed
Final Sample Result	Result in picocuries per day
Uncertainty	The counting error or uncertainty of the result (apparently 95% confidence level for gross alpha results; 68% confidence level for alpha spectrometry results.)

Although this spreadsheet does not contain any new data, it represented a substantial Air Force effort that could serve as the basis for preparing data for further evaluation and use in the dose assessment. The data added and revisions made are discussed in a later section of this appendix.

B.2.3. Reports

Additional information in the form of correspondence and written reports can provide details of the accident and the response effort, as well as insight into the approach to evaluating possible health and safety issues associated with the response effort. Several documents provided key information about those factors and formed the foundation for the pertinent analysis required of this effort. Documents that provided those kinds of key information included:

The *Palomares Summary Report* prepared by the Field Command, Defense Nuclear Agency that provides a comprehensive summary of the details of the accident, contamination levels, response efforts and limited discussions of health and safety actions (DNA 1975).

"Plutonium Deposition Registry Board, Proceedings: First Annual Meeting, 26 – 28 October 1966" prepared by the Air Force Logistics Command that described the proceedings of the first meeting of this board and reviewed key issues and discussions on the progress and future plans for the follow-up effort (Odland 1966).

Name:(Last, First, M.I)	SSN:	AF ID # :	Type Sample	Type Anal.	Sample No.	Sample Date:	Base:	Date Recived	Sample Volume	Sample Analyzed	Date Analyzed	Final Sample Result	Uncert.
Data Masked	Data Masked	Data Masked	Urine	N/A	66-2475	n/a Fr:845 10 Apr66	Torrejon	7-Apr-66	1000	1000	N/A	0.131	n/a
Data Masked	Data Masked	Data Masked	Urine	N/A	66-2867	To: 650 11 Apr 66	Torrejon	22-Apr-66	1600	1800	N/A	1.10+/-0.27	n/a
Data Masked	Data Masked	Data Masked	Urine	PU	66-1193	19-Feb-66	Torrejon	3-Mar-66	1500	1000	N/A	n/a	n/a
				Gross									
Data Masked	Data Masked	Data Masked	Urine	Alpha	66-1428	26-2-66	Moron	9-Mar-66	600	624	N/A	n/a	n/a
			Nasal										
Data Masked	Data Masked	Data Masked	Swipe	N/A	66-2525	12-Mar-66	Torrejon	6-Apr-66	n/a	n/a	N/A	n/a	n/a
												0.0639+/-	
Data Masked	Data Masked	Data Masked	Urine	N/A	66-2049	n/a	Torrejon	31-Mar-66	430	430	N/A	0.0793	n/a
Data Masked	Data Masked	Data Masked	Urine	N/A	66-2146	19-Mar-66	Torrejon	1-Apr-66	850	850	N/A	0.179+/-0.118	n/a
Data Masked	Data Masked	Data Masked	Urine	N/A	66-1811	3-Feb-66	Torrejon	28-Mar-66	720	720	N/A	0.0405	n/a
Data Masked	Data Masked	Data Masked	Urine	N/A	66-2332	19-Mar-66	Torrejon	1-Apr-66	890	890	N/A	n/a	n/a
Data Masked	Data Masked	Data Masked	Urine	N/A	66-2866	Fr:0700 6 Apr 66	Torrejon	22-Apr-66	1100	1100	N/A	1.04+/-0.26	n/a
				Gross									
Data Masked	Data Masked	Data Masked	Urine	Alpha	66-1403	26-Feb-66 Fr. 1 Apr 66	Torrejon	9-Mar-66	660	685	N/A	n/a	n/a
Data Masked	Data Masked	Data Masked	Urine	N/A	66-2885	To: 2 Apr 66	Torrejon	26-Apr-66	1500	1500	N/A	0.137+/-0.107	n/a
Data Masked	Data Masked	Data Masked	Urine	N/A	66-1097	25-Jan-66	Wiesbaden	1-Mar-66	850	850	17-Mar-66	n/a	n/a
Data Masked	Data Masked	Data Masked	Urine	N/A	66-2912	19-Mar-66	Torrejon	1-Apr-66	550	550	N/A	0.189+/-0.124	n/a
				Gross									
Data Masked	Data Masked	Data Masked	Urine	Alpha	5(66-213)		Torrejon	25-Jan-66	430	200	N/A	n/a	n/a
				Gross									
Data Masked	Data Masked	Data Masked	Urine	Alpha	23(66-231)	n/a	Torrejon	25-Jan-66	475	200	N/A	0.473+/-0.233	n/a
				Gross									
Data Masked	Data Masked	Data Masked	Urine	Alpha	25(66-233)	n/a	Torrejon	25-Jan-66	475	200	N/A	0.473+/-0.233	n/a
												0.0336+/-	
Data Masked	Data Masked	Data Masked	Urine	N/A	66-2581	23-Mar-66	Moron	11-Apr-66	1000	1000	N/A	0.0623	n/a
				Gross									
Data Masked	Data Masked	Data Masked	Urine	Alpha	66-888	5-Feb-66	Moron	18-Feb-66	970	200	N/A	3.77+/-1.36	n/a
				Gross									
Data Masked	Data Masked	Data Masked	Urine	Alpha	66-1402	26-Feb-66	Torrejon	9-Mar-66	1000	1000	N/A	1.04+/-1.88	n/a
			Nasal										
Data Masked	Data Masked	Data Masked	Swipe	N/A	66-1308	n/a	Moron	9-Mar-66	n/a	n/a	N/A	n/a	n/a
				Gross									
Data Masked	Data Masked	Data Masked	Urine	Alpha	66-496	7-Feb-66	Moron	10-Feb-66	600	200	N/A	3.89+/-1.05	n/a
Data Masked	Data Masked	Data Masked	Urine	N/A	66-2073	19-Mar-66	Torrejon	1-Apr-66	490	490	N/A	0.141+/-0.083	n/a
												0.0787+/-	
Data Masked	Data Masked	Data Masked	Urine	N/A	66-2057	9-Mar-66 Fr: 27 Mar 66	Torrejon	1-Apr-66	650	650	N/A	0.0622	n/a
Data Masked	Data Masked	Data Masked	Urine	N/A	66-2684	To:28 Mar 66	Torrejon	12-Apr-66	1100	1100	N/A	0. +/-0.	n/a
Data Masked	Data Masked	Data Masked	Urine	N/A	66-2498	n/a	Moron	7-Apr-66	950	950	N/A	0	n/a
				Gross									
Data Masked	Data Masked	Data Masked	Urine	Alpha	66-1379	28-Feb-66	Torrejon	9-Mar-66	500	520	N/A	n/a	n/a
Data Masked	Data Masked	Data Masked	Urine	N/A	66-2294	20-Mar-66	Moron	1-Apr-66	950	950	N/A	0.162+/-0.136	n/a
				Gross			Hanaw						
Data Masked	Data Masked	Data Masked	Urine	Alpha	66-732	9-Feb-66	Germany	17-Feb-66	860	200	N/A	2.62+/-1.07	n/a

Figure B- 6. Example page of IERA results spreadsheet.

An article entitled "Bioassay Experiences in Support of Field Operations Associated with Widespread Dispersion of Plutonium," in *Proceedings of Symposium on Diagnosis and Treatment of Deposited Radionuclides*, sponsored by the Hanford Environmental Research Foundation (Odland 1968a).

An article entitled "Industrial Medical Experience Associated with the Palomares Nuclear Incident" published in the Journal of Occupational Medicine that was a peer-reviewed version of the previous proceedings.

A letter by Colonel Wallace, Air Force Logistics Command Surgeon, with the subject: "Palomares Broken Arrow – Report on Medical Follow-up Program" that summarized the results of the follow-up program through January 1968 and concluded that neither additional follow-up nor meetings of the Plutonium Deposition Registry Board were required (Wallace 1968).

These documents provided a narrative overview of the approach to conducting the assessment of possible exposure to plutonium at Palomares. The discussions highlighted the issues faced, the problems encountered, and the rationale that formed the basis for the effort and decisions made throughout the period of on-site activity and subsequent follow-up. These issues are discussed in Section 2 of the main report. However, key points from that review are repeated here and serve as reference for the analyses to follow. The key points include the following.

- *Sample Contamination*. During the initial phase on site, samples were collected under less than ideal conditions that could have contaminated the sample containers and samples themselves from the blowing dust containing plutonium. In fact, RHL reported frequent episodes of gross alpha contamination on the outer surfaces of the sample containers received.
- *Sample Collection Period*. Ideally, samples should be collected for a full, 24-hour period to obtain the best representation of the daily excretion required by methods for estimating body content. In fact, most of the on-site samples were limited to 12 hours because of mission needs and difficulties keeping subjects confined for an entire 24 hours. To compensate for this, RHL corrected the result for every sample with a total volume of less than 1000 milliliters to 1200 milliliters; the volume assumed to represent the daily urine output of a normal, adult male.
- *Exposure Type and Date*. Most of the response personnel spent several weeks at the site. Their activities varied from daily presence in contaminated areas to primary work in administrative areas. As a simplifying assumption, exposures were considered as single, acute intakes that occurred on the mid-point of the period of time spent on the site.

B.3. DATA EVALUATION AND PREPARATION FOR DOSE ASSESSMENT

B.3.1. Data Evaluation

One final product from this project is a dataset, containing the estimates of the possible intake of plutonium and of the associated committed effective dose equivalent that can be loaded into the Air Force Master Radiation Exposure Registry. This process requires that the data provided undergo detailed scrutiny to determine its suitability and to identify possible consistency problems. Upon receiving the collection of data forms, spreadsheet, and reports discussed above

the data review occurred in several stages. Objectives of the review included availability of data elements required for input to chosen internal dosimetry models. The primary parameters include: the type of intake (inhalation, ingestion, skin contact), the date or dates the exposure occurred, the date of collection of nasal swab or urine samples, the duration of the urine sample collection, and the results of the sample analysis. Review of the data indicated that the hardcopy forms recorded exposure date or dates, sample date, and results for many samples. In other cases, forms did not contain all the required data. Consequently, our investigators sought alternate approaches.

First, the spreadsheet and data forms were compared to determine whether all forms were present in the spreadsheet and whether the entries were correct. The initial evaluation identified a number of problems with the spreadsheet and supporting forms as shown in Table B-3.

This initial review indicated that substantial numbers of samples lacked one or more important pieces of data such as a Sample Date or Exposure Date. The review also identified 115 data forms attached to a primary card that apparently represented a repeat analysis of the same sample or a follow-up sample for an individual. Those additional samples were not in the spreadsheet.

Following the initial review additional efforts corrected many of the missing entries through more careful analysis of the information and reasonable assumptions about the missing information.

Table B- 3. Issues with Palomares Data.

Issue	Number of Entries	Percentage
Exposure Date Not Available	402	22.7
Sample Date Not Available	445	25.1
No SSN Available	385	21.8
No Air Force ID Available	2	0.11
Sample Vol. < 600 mL	323	18.3
Sample Vol. > 1000 mL	434	24.5
Number with Additional Sampling Data (2^{nd} page)	115	6.50
Number of Cards Marked Out	2	0.11
Number of Cards Not Found	5	0.28
		Total Number of Samples = 1768

The duration of sample collection is a critical piece of data that determines the daily excretion rate of plutonium in urine. Daily excretion, as mentioned above, is the accepted parameter for estimating body content at a time following exposure. Air Force reports indicated that sample collection lasted 12 hours for many samples collected at Camp Wilson. To correct, the Air Force established a procedure that corrected the result for any urine sample of less than 1200 milliliters to 1200 milliliters. Although this may have been somewhat arbitrary, it provided a reasonable and conservative correction. The procedure was deemed conservative because it would tend to overestimate urinary excretion. For example, if an individual actually collected 900 milliliters in a 24-hour period, the correction would still be applied and the estimated daily excretion would be

increased by 25%. When other factors are equal, increasing the urinary excretion also raises the estimated body content.

Our review of the data indicated that 12-hour samples were clearly designated in 42 of the samples entered in the initial spreadsheet. Attempts to duplicate the Air Force estimate of systemic body burden revealed that the sample volume correction might have been applied inconsistently. However, this did not adversely affect any conclusions about the individuals tested. This finding does not materially affect preparation of the data for this assessment except for the samples clearly identified as 12-hour samples. This review concluded that adjustments to samples that were not designated as 12-hour samples presented were unnecessary. Therefore, recorded sample volumes were assumed to represent 24-hour output unless specifically designated as 12-hour samples.

Missing or incorrect entries for Exposure and Sample Date present additional challenges to performing a reasonable estimate of radiation dose. Careful review of the data indicated that additional analysis would be required to establish these parameters.

Other observed issues included missing SSNs, AFSNs, and other entries. Upon further analysis, it became evident that the records included information on the entire spectrum of responders – from Air Force to other Services (Army, Navy, Marines); other US agencies (State Department, Bureau of Mines), possible Spanish civilian employees of Torrejon Air Base or local citizens, and at least one media representative. Only US Air Force personnel would have AFSNs, however, entries for members of the other services had similar entries. Missing SSNs introduce some problems for integrating the results into current data systems, however the issue can be resolved.

B.3.2. Preparation of Data for Analysis

The issues identified in the previous section provided the basis for an approach to refine the data by correcting errors and inconsistencies and by developing reasonable estimates of missing data. As mentioned, this process had the primary objective of developing input data for the following parameters: exposure date, sample date, sample duration, and urinary excretion rate and its estimated error. Other inconsistencies observed in the data were also corrected to the extent possible. Each of these procedures is summarized in the following sections.

B.3.2.1 Exposure Date

Exposures were assumed to be acute inhalation as discussed in the main report. The exposure date was then calculated by determining the midpoint of the time an individual spent on station. Exposure date entries on the forms included all of the following: a single date (25 Jan 66), a date range (18 Jan 66 to 30 Jan 66), an arrival date (Arr: 20 Jan 66), a month and year (Jan 66), a year (66) and a few others.

Generally, an arrival date or single date entry could be assumed to represent the beginning of exposure and that was done. The end of the exposure presented additional difficulties. For data forms that did not clearly indicate the end of the exposure period, Sample Dates for all samples for an individual were reviewed. The day before the last Sample Date was assigned as the end of exposure period. This approach seemed reasonable since the established procedure was to

collect a sample from everyone before his or her departure. In some cases, individuals may have returned to their base of assignment before providing a sample. These cases would generally represent a few days. That delay was not viewed as serious when the other difficulties and uncertainties are considered. If the last sample was collected after Camp Wilson ceased all operations on April 11, 1966, that date was used as the end of exposure.

B.3.2.2 Sample Date

Data forms did not contain Sample Date entries for 445 samples. An alternative approach was developed to provide a reasonable estimate of the Sample Date. Data on the date a sample was received at RHL and the assigned laboratory sample numbers were used to develop the estimate.

The approach compared the range of valid Sample Date entries with the Date Received at RHL and with the sequence of assigned sample numbers. Figure B- 7 illustrates the distribution of the receipt of samples at the laboratory. The results of the comparison and some additional judgement allowed the Sample Date to be estimated. Although not necessarily precise, the approach allowed reasonable estimates of the Sample Date. The derived Sample Date information was then entered into a master dataset along with the other data for each urine sample. Notations documenting the source of the Sample Date were made for each entry.

B.3.2.3 Sample Duration

Actual sample duration was documented in a very small fraction (42 samples) of the samples received. Fortunately, basic sample volume data provide the basis for making any corrections needed. As discussed above, this project elected to treat recorded sample volumes as representing 24-hour outputs unless the data forms specifically designated the samples as 12-hour samples. For those, the results were adjusted to the currently accepted nominal daily urine output (1400 mL) for Reference Man. Those adjustments were performed in the intake assessment process.

B.3.2.4 Other Parameters

Analytical results for daily urinary excretion and the estimated error were transcribed as entered on the hardcopy forms. However, in the case of samples reported as No Detectable Activity, the data forms were reviewed for the presence of other calculations of a numerical result and its estimated error. When found, these actual results were used in the analysis, even when the error value exceeded the result. This procedure applied primarily when the results of multiple samples were available, as was the case for many of the "High 26" group. In these cases, although the errors were large, they nevertheless provided order of magnitude information about the levels present and were useful comparisons to other values. Specific notes are contained in the individual case files in Volumes II and III.

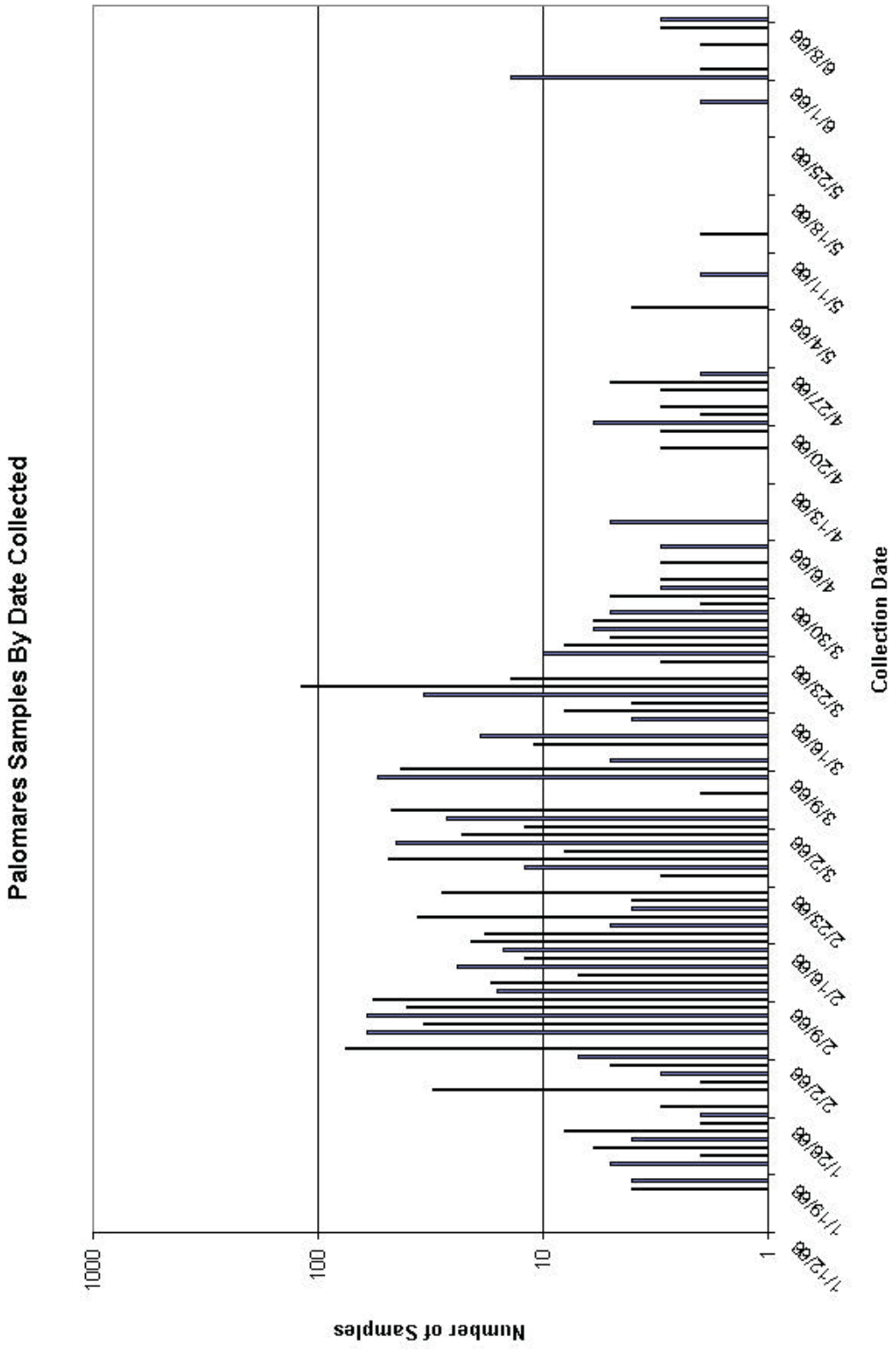

Figure B- 7. Distribution of Samples Received at RHL

B.3.2.5 Other Inconsistencies

Other inconsistencies in the dataset were also identified and corrected where possible. Although these did not affect the actual intake and dose assessments, they do affect identifying information. This review discovered inconsistencies in:

- Individual names caused by interchange of a letter or two.
- SSNs caused by typographical errors or easily identified keyboarding errors.
- Errors in designation of the analysis type, such as GrossAlpha for Gross Alpha.
- Base names caused by typing errors.

Other inconsistencies affecting only a few entries were revised as they were discovered.

B.4. SUMMARY OF THE DATA EVALUATION AND PREPARATION

After making the changes and updates discussed above the data set served as the basis for additional evaluations before processing of the intake and dose assessments. Those additional evaluations considered the amount of data available for each individual, the quality of the data, and possible issues with the data that would limit its reliability in assessing individual cases. In particular, the High 26 group had substantially more data than any other group of individuals. That group of 26 was followed-up for more than a year. Follow-up began in the summer of 1966 and continued until August and September 1967 for some of the group. Because of this, that group served as the primary group for study.

Evaluation of the data also revealed that about 115 appeared to have had their initial gross alpha analyses repeated using the alpha spectrometry technique. Or, they submitted follow-up samples upon request for analysis by alpha spectrometry. Those individuals comprised a second group that received additional evaluation of their conditions. Review of their data for reliability as indicated by adequate chemical recovery and other factors resulted in a total of 54 individuals with adequate sample data. The remaining 62 were removed because their sample results were not reported through laboratory error or other problems, or the chemical recoveries of their alpha spectrometry samples were below 40% and not considered reliable. This group was called the "Repeat Analysis" Group. Their individual cases were evaluated and the results are reported in Appendix C.2.

Of the remaining majority of samples, most represented only one sample for an individual collected while at Camp Wilson. As discussed in Appendix C.3, many of those results were quite high indicating possible contamination. Review of the data also revealed that a substantial number showed relatively low urine measurements. Their results were in the same range of urinary excretion as the individuals with the lower intakes and associated CEDEs of the High 26 and Repeat Analysis Cases. Further review of the data and assessment of a reasonable lower level of detection led to the conclusion that samples with results of less that 0.1 picocuries per day represented that reasonable lower level. Individuals with daily excretions at that level were evaluated and reported in Appendix C.3. This group, called Contamination Cutoff Cases, was not evaluated to the depth of detail as the previous cases, primarily because they had only one result for urine content. Nevertheless, the assessment provides an approximate estimate of their intake and dose.

Finally, all remaining samples were reviewed. Since their samples were collected on site and were at risk of sample contamination, the urine measurements are entered at Appendix C.4. However, no further assessment of their results was attempted.

APPENDIX D

SELECTION OF DOSE METHODOLOGY

D. SELECTION OF DOSE METHODOLOGY

D.1. REVIEW OF INTERNAL DOSIMETRY METHODS

Exposure to radiation can occur from sources of penetrating radiation outside the body, such as x-ray machines or industrial radiography sources, or from sources of radioactive materials, such as plutonium or uranium, that enter the body, locate in an internal organ or organs, and irradiate the tissues of those internal organs. The problem of calculating the dose depends on many factors such as the shape of the organ, the type of radiation, the amount of the deposit, and the distribution of the deposit. Each of these individual factors is subject to considerable variability and difficulty in determining accurately. Once a dose is calculated, effectively communicating the possible effect of the dose on health requires additional skill and effort.

The current approach to limiting radiation exposure in the United States is derived from recommendations in ICRP Publications 26 and 30. The ICRP approach uses the concept of Committed Effective Dose Equivalent (CEDE) - a cumulative dose, weighted for the contributions of individual organs, and summed over a 50-year period for workers. Quantities derived from the CEDE such as the Annual Limit on Intake (ALI) and the Derived Air Concentration (DAC) provide operational limits for workers so that the overall guidelines will not be exceeded. The ALI is the activity of a radionuclide that would irradiate a person to the limit set by the ICRP for each year of occupational exposure. The DAC is found by dividing the ALI by the volume of air inhaled (2,400 m^3) in a working year (2,000 hours) (ICRP 1979).

For internal exposures, determining the dose requires knowledge of the following questions:

- How does the material get into the body?
- Once in the body, how quickly does the material move to other organs?
- Does the material in the initial organ leave the organ or does some remain?
- Once in an organ, how does the material irradiate the organ and other organs?
- Once in an organ, how does the material move to other organs?
- Finally, how does is the material eliminated from the body if at all?

Answers to these provide the basis for developing an approach to calculate the dose to organs, the effective dose equivalent to the body, and interpreting the effects of the dose.

D.1.1. Internal Dosimetry Methods

The methods for estimating organ dose from internal radionuclides have evolved since radioactive materials were discovered and used. Until 1979, ICRP Publication 2 provided the guidelines and methodology. In 1979, ICRP Publications 26 and 30 changed the basic approach to limiting radiation, and for internal radionuclides in particular. That approach currently remains the accepted approach in the United States for purposes of regulation. However, progress in all areas of radiation effects and the behavior of radionuclides in the body have produced more recent recommendations on a number of key elements in the process as presented in ICRP Publications 54, 60 and 66. As for any dynamic area of study, continued improvements in the understanding of plutonium's behavior in the body, improved methods for estimating body

content, and more accurate mathematical models for estimating intake and dose from body content will evolve.

D.1.1.1. ICRP Publication 2 Methods

The models of ICRP-2 assumed that a single organ could be considered the critical organ; that the organ retention could be represented by a single exponential term; that the physical characteristics, such as intake parameters, transfer functions, and tissue size and weight, could be represented by "Standard Man" data; that organs could be assumed to be spherical; and that scattered radiation could be ignored. In performing the dosimetry, it was assumed that the material was distributed uniformly throughout the organ and that the energy absorbed equaled the energy emitted. Doses were limited to a specified annual dose to the critical organ.

Intakes of radionuclides were controlled by limiting "Maximum Permissible Concentration" (MPC) values in air and water for workers so that the annual dose limit to the critical organ would not be exceeded. The annual limit on dose to the critical organ applied over a 50-year intake period so that the limit would not be exceeded even if a radionuclide were taken in continuously over 50 years. An associated limit, called the "Maximum Permissible Body Burden," was that amount of a material in the body that would not exceed the annual dose limit to the critical organ. The ICRP-2 method was in effect and adopted for the Palomares accident.

D.1.1.2. ICRP-30 Models and Methods

The ICRP changed its basic recommendations and revised the system of dose limitation in ICRP Publication 26 based on risk. This approach acknowledged the availability of sufficient information about the effects of radiation to estimate risk for fatal cancer from a unit dose equivalent in exposed people and in the risk of serious disease to offspring of exposed people. The basic recommendations addressed both stochastic effects and non-stochastic effects. For stochastic effects, such as cancer and hereditary effects, risks are assumed to be directly related to dose equivalent with no threshold, meaning that the probability of the effect occurring, rather than the severity, is related to the dose equivalent. The severity of non-stochastic effects, such as cataracts and erythema, varies with dose, usually above a threshold or minimum dose.

ICRP Publication 30 provided revised dosimetry models that assume organ retention is represented by one or more exponential expressions, the critical organ concept no longer applies, the dose in an organ must consider radiation emitted by other organs in the body, and the physical characteristics are represented by "Reference Man" data in ICRP Publication 23 (ICRP 1975). The model assumes that deposition in an organ is uniform, and that the total dose is averaged over the organ.

Under the revised system, dose equivalent limits are intended to prevent non-stochastic effects and to limit stochastic effects to acceptable levels. To meet this end, an annual occupational limit of 50 rem (0.5 Sv) to any organ was established (ICRP 1979). For stochastic effects, the limit on risk is the same whether the whole body is irradiated or organs are non-uniformly irradiated. This is accomplished by assigning organ weighting factors, w_t, that represent the ratio of the risk for the effect in an organ to the risk for whole body irradiation. The limit on risk to the whole body is then determined by summing the contributions for each irradiated organ and is given by:

$$\sum_{T} w_T H_{50,\,T} \leq 5 \text{ rem } (0.05 \text{ Sv})$$

where $w_t H_{50,T}$ is called the weighted committed dose equivalent or the committed effective dose equivalent (CEDE), and $H_{50,T}$, called the committed dose equivalent (CDE), is the total dose equivalent averaged over tissue (T) in the 50 years following intake and is limited to 50 rem (0.5 Sv). Table D-1 contains the organ weighting factors from ICRP-30.

The dosimetry model calculates the absorbed dose averaged over the organ mass during 50 years following intake. It considers each radiation type and applies a radiation weighting factor, sometimes called the quality factor, which has the following value:

Q=1 for beta particles, electrons and all electromagnetic radiation.

Q=10 for fission neutrons emitted in spontaneous fission and protons.

Q=20 for alpha particles from nuclear transformations, for heavy recoil particles, and for fission fragments.

Table D- 1. ICRP-30 Tissue weighting factors, w_T (ICRP 1979).

Tissue	Weighting Factor, w_T
Gonads	0.25
Red Marrow	0.12
Lung	0.12
Breast	0.15
Thyroid	0.03
Bone Surface	0.03
Remainder 0.06 for the organs with the five highest dose.	0.30

Additional modifying factors, not discussed here, that consider irradiation from other organs and radionuclides are used to calculate the final organ dose equivalent.

For inhaled radionuclides, the Task Group on Lung Dynamics developed a respiratory tract model, which uses the approach shown in Figure D-1. That approach considers three classes (D, W, and Y) of material based on retention in the deep or pulmonary section of the lung. The classification depends on a range of retention half-times: D < 10 days; 10 days < W < 100 days; and Y > 100 days. ICRP-30 contains metabolic data for certain chemical forms of the materials.

The model defines three regions of deposition: nasal-pharyngeal (N-P), tracheo-bronchial (T-B) and pulmonary (P). Fractions initially deposited in these regions are D_{N-P}, D_{T-B}, and D_P and are based on an aerosol particle size of 1 μm. As Figure D-1 indicates, each section is divided into compartments that are associated with clearance pathways and have an established clearance half-time T and fraction F for removal of material. Compartments a, c, and e represent direct transfer to body fluids, known as the transfer compartment, for further transfer to other organs or excretion. Compartment g represents indirect transfer to body fluids through lymph nodes. For Class Y material, only some material is transferred (in compartment i) to other bodily fluids. The remainder stays indefinitely in compartment j. Compartments b, d, f and h transfer material to the

		Class					
		D		W		Y	
Region	Compartment	T Day	F	T day	F	T day	F
N-P	a	0.01	0.5	0.01	0.1	0.01	0.01
(D_{N-P} = 0.25)	b	0.01	0.5	0.4	0.9	0.4	0.00
T-B	c	0.01	0.95	0.01	0.5	0.01	0.01
(D_{T-B} = 0.08)	d	0.2	0.05	0.2	0.5	0.2	0.99
	e	0.5	0.8	50	0.15	500	0.05
P	f	n.a.	n.a	1.0	0.4	1.0	0.4
(D_P = 0.25)	g	n.a.	n.a	50	0.4	500	0.4
	h	0.5	0.2	50	0.05	500	0.15
L	i	0.5	1.0	50	1.0	1000	0.9
	j	n.a.	n.a.	n.a.	n.a.		0.1

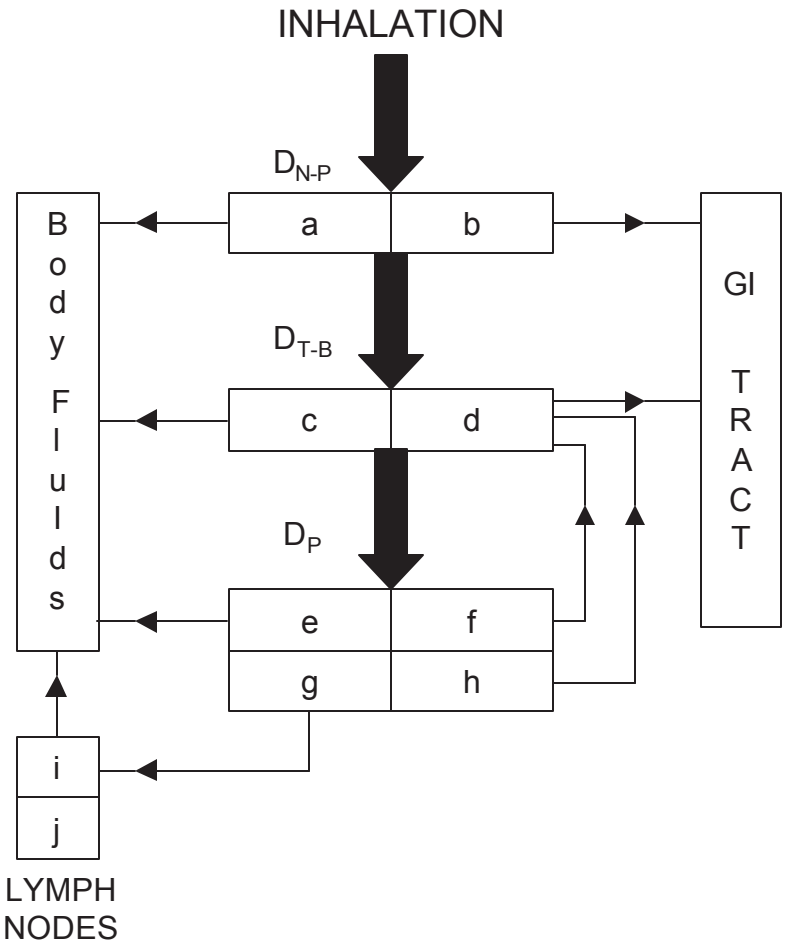

Figure D- 1. ICRP-30 Model of the respiratory tract (ICRP 1979).

gastro-intestinal tract (GI tract). Once a radionuclide reaches other organs, its behavior is then governed by the metabolic model.

The gastro-intestinal tract model is based on the fraction transferred from the GI tract to the systemic system (f_1). Since f_1 for Class Y plutonium is 0.00001, ingestion is not considered significant for evaluation of the Palomares responders and the GI tract will not be considered further.

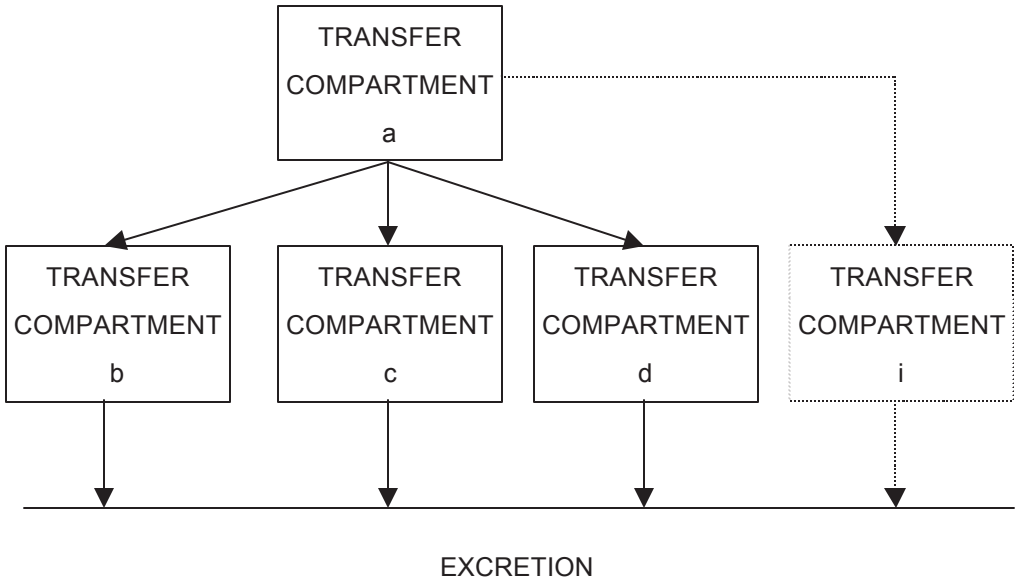

Figure D- 2. ICRP-30 Transfer Compartment Model (ICRP 1979).

Material that has been transferred to bodily fluids and other compartments of various tissues are indicated in Figure D-2, taken from ICRP-30. The time a material takes to transfer from the deposition site is represented by transfer compartment a. The clearance half-time for this compartment is 0.25 day unless stated otherwise. Each tissue that receives the radionuclide will have one or more compartments with an associated elimination rate. The model assumes that there is no feedback, or recycling, of a material to an original compartment. That means the model is a one-pass, or pass-through, model. Figure D-3 shows the ICRP-30 model for a Class Y plutonium aerosol.

Calculation of the committed dose equivalent (CDE) for a given organ is the sum of the product of two factors: U_s, the total number of transformations of the radionuclide in the source organ (S) over 50-years following intake, and SEE (T←S), the energy absorbed in the target tissue (T), modified by the quality factor, for each type of radiation emitted in S. ICRP tables of SEE values are available for estimating the committed dose equivalent for an organ.

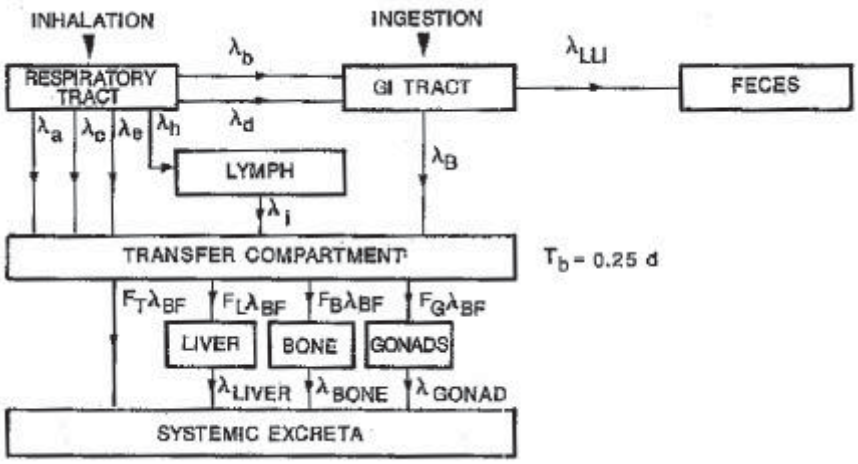

Figure D- 3. ICRP-30 Pu Metabolic Model (ICRP 1979).

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D.1.1.3. ICRP-60 and 66 Methods

Further refinement in the basic recommendations of the ICRP and in certain models have been achieved since the revisions of ICRP-26 and 30. Most notable is a revision of the Respiratory Tract Model by the Task Group on Lung Dynamics, approved by the ICRP and published in Publication 66 (ICRP 1994). That model represents an update to ICRP-30 that provides a broader scope, having been designed not only to evaluate secondary limits on intake of radionuclides by inhalation for a worker, but also to:

- Provide a realistic framework for modeling lung retention and excretion characteristics in an individual case, and the resulting lung and systemic organ doses, based on bioassay data;
- Take into account factors such as cigarette smoking and lung disease which influence lung particle retention;
- Enable knowledge of the dissolution and absorption behavior of specific materials to be used in the calculation of the lung dose, systemic absorption and excretion of the materials;
- Apply explicitly to age-dependent members of a population; and
- Calculate biologically meaningful doses in a manner that is consistent with the morphological, physiological, and radiobiological characteristics of the various tissues of the respiratory tract.

The ICRP-66 lung model consists of three parts:

- A particle deposition model,
- A particle transport model, and
- A particle absorption model.

The new lung model is fundamentally different from the lung model published in ICRP-30, which calculates only the average dose to the lungs. It accounts for the differences in

radiosensitivity of the respiratory tract tissues, and the wide range of doses they may receive, and calculates doses to the specific tissues in the respiratory tract.

The respiratory tract is represented by five regions (Figure D-4): the nasal and oral passageways termed the "extrathoracic" (ET) airways; three thoracic regions termed the Bronchial region (BB); the Bronchiolar region (bb), and the Alveolar-Interstitial region (AI, the gas exchange region); and the lymphatics associated with the extrathoracic (LN_{ET}) and thoracic airways (LN_{TH}). The model evaluates the risks of lung and other cancers by calculating the doses received by tissues in each of the regions, then summing and weighting those doses to obtain equivalent doses, and finally applying the tissue weighting factors in ICRP Publication 60 (ICRP 1991).

The new model accommodates calculating the intake of different individuals (adults and children), although that feature is not pertinent to this project. Intake depends on two factors: inhalability and breathing rate. Inhalability is the ratio of the concentration of particles or gases in air entering the respiratory tract to the concentration in ambient or surrounding air. Larger particles (20 μm and larger) have higher inertia and therefore are not inhaled as easily as smaller particles under most conditions. The breathing rate depends on age and physical activity. The model provides tables of reference values of breathing rates for men and women as well as children aged 15, 10, 5, and 1 year, and 3 months for different levels of activity. The reference values for adults were developed to simulate common activity levels in the workplace that combine periods of sitting and exercise. The "reference male worker" is assumed to spend 3% of an 8-hour work period sitting and 69% at "light exercise."

Deposition is provided for each of the five regions of the lung for the various categories of activity and breathing type – nose or mouth.

The model contains three clearance pathways: material in ET_1 clears by direct means such as nose blowing; in other regions clearance may be to GI tract and lymphs or absorption into blood. Once cleared, particle transport is represented by the model in Figure D-5 that shows 14 compartments with individual values of the particle transport rate constant. Absorption into blood is treated as a two-stage process involving dissociation into material that can be absorbed (called dissolution) and absorption into blood of soluble material and material dissociated from particles (called uptake). In addressing absorption, the model uses three material "Types": F (fast), M (moderate), and S (slow). These Types correspond to Classes D, W, and Y of ICRP-30.

The Types are characterized by the amount of deposit that enters the blood and an approximate half-life according to the following:

- Type F: 100% at 10 minutes.
- Type M: 10% at 10 minutes and 90% at 140 days.
- Type S: 0.1% at 10 minutes and 99.9% at 7000 days.

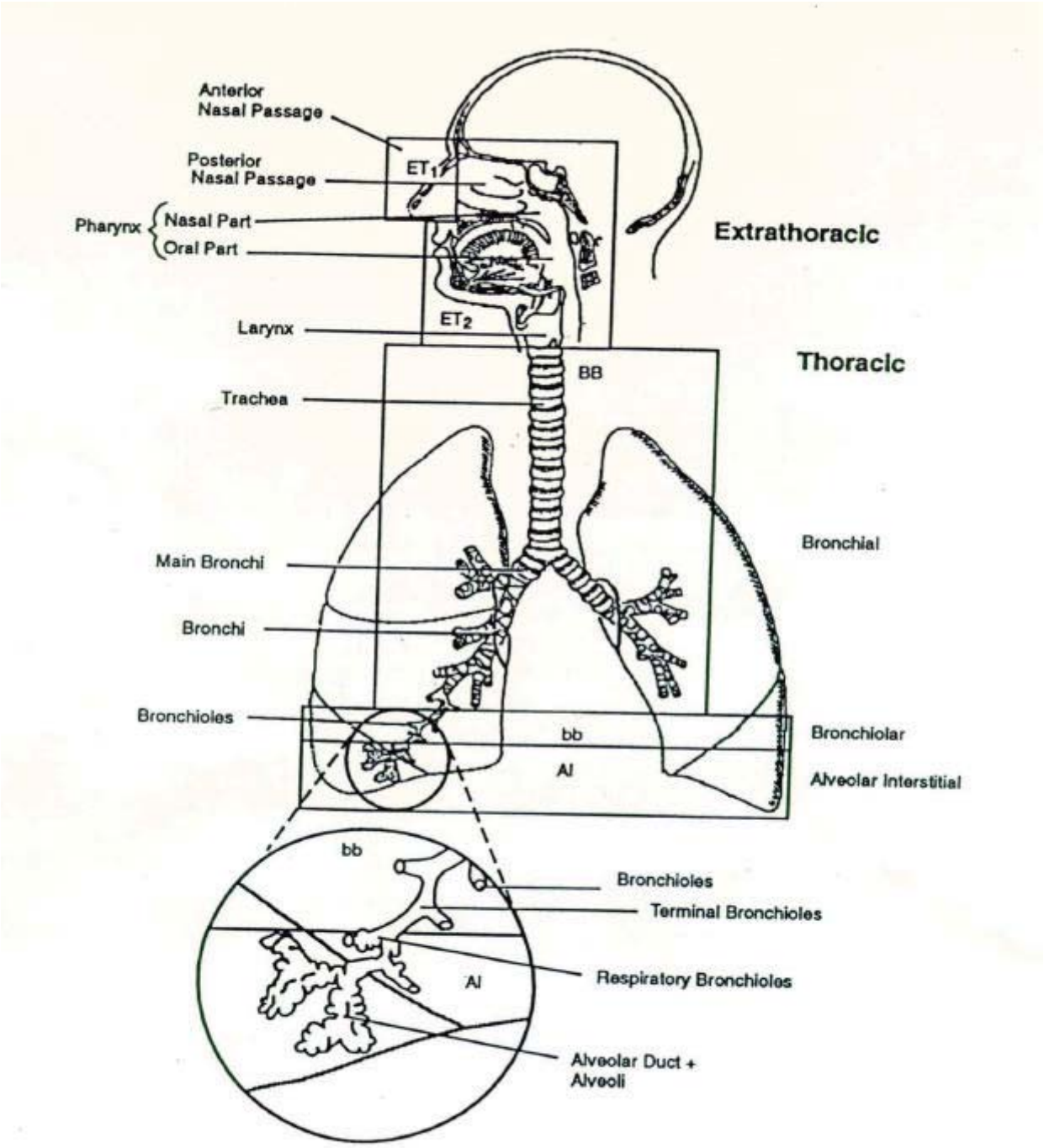

Figure D- 4. Anatomical Regions of the Respiratory Tract (ICRP 1994).

The dose to each region is determined according to ICRP's general approach of averaging the dose to target tissue in each region. Target cells in ET_1, ET_2, BB, and bb are calculated, and then modified by a risk apportionment factor that represents the relative sensitivity of the region to the whole organ. Finally, the ICRP tissue weighting factors are applied.

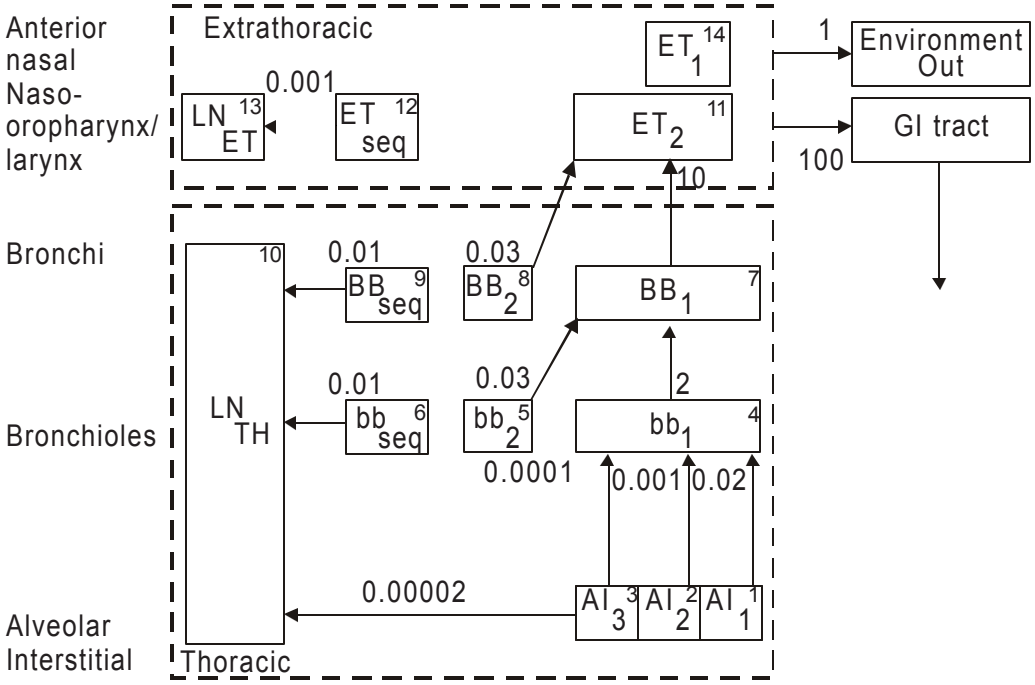

Figure D- 5. Compartment Model of ICRP-66 (ICRP 1994).

Assessment of intake presents one of the more difficult problems for estimating organ dose and the CEDE. Commonly applied methods include in-vitro bioassay of the amount of the material excreted, measurements of body content or organ content by external whole body counting, or for inhalation or ingestion, estimating the amount of material in the air or water using air or water samplers. Each method has its advantages and disadvantages. For this case, the in-vitro bioassay measurements of urine samples from 1966 and 1967 provided the best available method for assessing the intake based on a substantial amount of recorded urinary excretion results.

Organ or tissue weighting factors affect the calculation of committed effective dose equivalent from the effective dose equivalent for each organ or tissue. The ICRP's 1990 recommendations (ICRP 1991) provide weighting factors for a number of tissues that were part of the remainder in the 1979 recommendations of ICRP-26 (ICRP 1979). Table D-2 lists the tissue weighting factors of ICRP-60 as well as those of ICRP-26 for comparison. Substantial differences between the two sets of weighting factors include a reduction in the bone surface and breast factors by three times, a 67 percent increase in the thyroid factor, and assignment of factors for additional organs, including the skin of the whole body.

D.1.1.4. Effect of Respiratory Tract Model on Dose

The differences between the two ICRP models for the respiratory tract could be expected to produce differences in estimated doses. During development of the updated respiratory tract model, its performance was tested in detail to determine the affects of various parameters taken alone and in combination. Some examples of the performance of both systems provide useful information about likely differences in estimating both equivalent dose and effective dose equivalent.

Table D- 2. Tissue Weighting Factors (ICRP 1991).

	ICRP Recommendations	
Tissue or organ	1979	1990
Gonads	0.25	0.20
Red Marrow	0.12	0.12
Colon		0.12
Lung	0.12	0.12
Stomach		0.12
Bladder		0.05
Breast	0.15	0.05
Liver		0.05
Esophagus		0.05
Thyroid	0.03	0.05
Skin		0.01
Bone Surface	0.03	0.01
Remainder	30[1]	.05[2]

[1] A value of 0.06 is applicable to each of the five remaining organs or tissues receiving the highest equivalent doses.
[2] The remainder is composed of the following tissues or organs: adrenals, brain, small intestine, kidney, muscle, pancreas, spleen, thymus and uterus.

One such evaluation, reported by James (James 1994) compared the lung dose equivalent and effective dose for several categories of radionuclides, including insoluble alpha emitters, such as plutonium at Palomares. In those illustrations, James compared doses for intakes of 1 μm activity median aerodynamic diameter (AMAD) particles although ICRP recommends 5 μm AMAD for workers. For 1 μm AMAD, Type S (Class Y) ^{239}Pu, the ICRP-30 and ICRP-66 equivalent dose per unit intakes were 320 μSv/Bq and 84 μSv/Bq, respectively. The ICRP-66 equivalent dose was lower by about a factor of 3.8. For 5 μm AMAD particles, ICRP-66 estimated 50 μSv/Bq, or about 6 times lower. Calculating effective dose for the same conditions, ICRP-30 produced 60 μSv/Bq and ICRP-66 produced 16 μSv/Bq for 1 μm AMAD particles and 9.1 μSv/Bq for 5 μm AMAD particles, representing reductions of about 3.7 and 6.5, respectively. Thus, other factors being equal, the ICRP-66 respiratory tract model can produce equivalent doses that are roughly 3 to 6 times lower for the same intake than the ICRP-30 model. This difference, attributed to the modified model for lung deposition and clearance and revised tissue weighting factors – must be recognized in evaluating methods for this project.

D.1.1.5. Intake Assessment

Intake assessment presents one of the more difficult problems for estimating the dose in affected organs and the CEDE. Commonly applied methods include in-vitro bioassay of the amount of the material excreted, measurements of body content or organ content by external whole body counting, or for inhalation or ingestion, estimating the amount of material in the air or water using air or water samplers. Each method has it advantages and disadvantages. For the case at hand, in-vitro bioassay of urine samples provides the best available method for assessing the intake.

This problem is common to the models discussed above. At the present time, either or both models can assist in calculating an estimate of the intake from knowledge of in-vitro bioassay, whole body counting, or measurement of air concentrations. Assessment of intake using in-vitro bioassay is the primary method of interest in this case because urine sample results are available for those who responded.

The models discussed above provide mathematical expressions, supported by a body of reference data to determine the amount of a radionuclide that can be excreted. Special excretion functions have been derived and are recommended for specific materials (ICRP 1988). In general, the amount of a radionuclide excreted in urine per day is related to the amount of radioactivity in one or more systemic retention compartments and fractional transfer parameters from those compartments to urine or feces. For plutonium, two special models have been developed and are commonly used. These are the "Jones" model and the "Durbin" model.

The Jones model (Jones 1985; Strong and Jones 1989) describes how plutonium excretion in urine varies with time. The model is used with the standard intake models (respiratory tract, gastro-intestinal tract, and direct), and models the material leaving those models as going directly into the four Jones model compartments. The Jones model was originally developed to describe the excretion rate of plutonium following intravenous injection. However, it has been modified for use in estimating chronic and acute inhalation and ingestion exposures. The Jones model is described by the following expression:

$$E_u = \sum_{j=1}^{4} F_{jj} \exp(-k_{jj} t)$$

where E_u = urinary excretion rate of plutonium at time t, in pCi/d

F_{jj} = fraction of injected activity that excretes according to exponential term j, in pCi/d per pCi injected.

k_{jj} = rate constant for decrease of excretion for exponential term j, in d^{-1}.

t = time, d.

The Jones Model transfer parameters are provided in Table D-3.

A second model, known as the Durbin Plutonium Excretion Model (ICRP 1988) performs in a similar fashion to the Jones model. As with the Jones model, material leaving the intake models (respiratory tract, gastro-intestinal tract, and direct) is modeled as going directly to the Durbin

model excretion compartments. The Durbin model is characterized by five compartments and has the following form:

$$E_{u,\,t} = \sum_{j=1}^{5} F_{pj} \exp\,(-k_{pj}t)$$

where $E_{u,t}$ = urinary excretion rate of plutonium at time t, in pCi/d
F_{Pj} = fraction of injected activity that excretes according to exponential term j, in pCi/d per pCi injected.
k_{Pj} = rate constant for decrease of excretion for exponential term j, in d^{-1}.
t = time, d.

Table D- 3. Jones Model Transfer Parameters (Strong and Jones 1989).

Compartment	Rate Constant, d^{-1}	Fractional Excretion Rate by Compartment, d^{-1}
1	5.58×10^{-1}	4.75×10^{-3}
2	4.42×10^{-2}	2.39×10^{-4}
3	3.80×10^{-3}	8.55×10^{-5}
4	2.84×10^{-5}	1.42×10^{-5}

The Durbin Model parameters are given in Table D-4.

Table D- 4. Durbin Model Transfer Parameters (ICRP 1988).

	Urine Excretion		Fecal Excretion	
Excretion Compartment	Fractional Rate, d^{-1}	Rate Constant, d^{-1}	Fractional Rate, d^{-1}	Rate Constant, d^{-1}
1	4.1×10^{-3}	5.78×10^{-1}	6.0×10^{-3}	3.47×10^{-1}
2	1.2×10^{-3}	1.26×10^{-1}	1.6×10^{-3}	1.05×10^{-1}
3	1.3×10^{-4}	1.65×10^{-2}	1.2×10^{-4}	1.24×10^{-2}
4	3.0×10^{-5}	2.31×10^{-3}	2.0×10^{-5}	1.82×10^{-3}
5	1.2×10^{-5}	1.73×10^{-4}	1.2×10^{-5}	1.73×10^{-4}

D.1.2. Description of Computer Models

Many computer programs have been developed and are available for performing the calculations of the models discussed above. Currently more programs implement the ICRP-30 system than the ICRP-66 model. This comes as no surprise since the ICRP-30 system remains the current system for regulation of the doses from radioactive materials in the United States. However, one

objective for this project included the evaluation and recommendation or the best calculation method. Since ICRP provisions are usually adopted in the U.S., investigating at least one software program that implemented the most recent approach seemed reasonable. After some review of the available software, three programs were selected for further study – the Radiological Bioassay and Dosimetry Program (RBD) as modified for the Air Force, Code for Internal Dosimetry (CINDY), and Lung Dose Evaluation Program (LUDEP ver 2.06). This section provides a general description of each program and some salient features. Later sections discuss the approach and results of testing the methods for this report.

D.1.2.1. Radiological Bioassay and Dosimetry Program (RBD)

The RBD software package (ORNL 1993) was developed for the U.S. Army and modified for the U.S. Air Force (Version RBD/AF) by Oak Ridge National Laboratory to demonstrate compliance with Federal radiation protection guidance.

The algorithms within the RBD and RBD/AF programs are the same. The RBD/AF program contains the following changes and enhancements to RBD:

- Increased number of organs for which committed dose can be calculated.
- Replacement of the "department identifier" input with "base code."
- Addition of an identifier field for gender of individual assayed.
- The display of the allowable lifetime intake (ALI) for a radionuclide was changed to the calculation of the fraction of the ALI received by the individual.
- The format of the committed effective dose report was revised to reflect Air Force reporting requirements.

The RBD model implements the ICRP-30 lung model and a urinary excretion model adapted from Leggett and Eckerman (Eckerman 1987). The software package was designed to run interactively on an IBM-compatible personal computer. RBD consists of a data base module to manage bioassay data and a computational module that incorporates algorithms for estimating radionuclide intakes from either acute or chronic exposures. These calculated results are based on the measurement of the worker's rate of excretion of the radionuclide or the retained activity in the body using the approach contained in ICRP-30. RBD estimates an intake using a separate file for each radionuclide containing parametric representations of the retention and excretion functions. These files also contain dose-per-unit intake coefficients used to compute the committed dose equivalent. Computed results derived from bioassay data (estimates of intake and committed dose equivalent) are stored in separate databases, and the bioassay measurements used to compute a given result can be identified.

D.1.2.2. Code for Internal Dosimetry (CINDY)

The Code for Internal Dosimetry (CINDY) (v.1.4) is a menu-driven interactive computer program that was developed to address the Department of Energy Order 5480.11 and the Nuclear Regulatory Commission's Standards for Protection Against Radiation (10 CFR Part 20). The CINDY software package (PNL 1992) was developed by Pacific Northwest National Laboratory

to provide the capabilities to calculate organ dose equivalents and effective dose equivalents using the approach contained in ICRP-30.

CINDY supports calculation of organ dose equivalents, effective dose equivalents and committed effective dose equivalents; interpretation of bioassay data; and evaluation of committed and calendar-year doses from intake or bioassay measurement data.

For inhalation exposures, CINDY uses the ICRP-30 lung model and approach for calculation of organ dose equivalents and effective dose equivalents, which is described in the previous discussion of the RBD/AF model. Biokinetic models are used to estimate intakes based on bioassay data. For intake and urinary excretion of plutonium, the Jones and Durbin models are both available, as in the LUDEP program.

The metabolic and excretion models available in CINDY are:

- ICRP-30 Lung model
- ICRP-30 Gastrointestinal (GI) model
- ICRP-30 General systemic model
- Jones and Durbin Plutonium Excretion Models

CINDY uses the quality factors and tissue or organ weighting factors published in ICRP-26.

D.1.2.3. Lung Dose Evaluation Program (LUDEP ver 2.06)

The Lung Dose Evaluation Program (LUDEP) (v. 2.0) is a personal computer program for calculating internal doses using the ICRP-66 respiratory tract model. The LUDEP program runs on an IBM-compatible personal computer in a DOS or Windows environment.

LUDEP was designed initially for two applications: (1) to help the ICRP Task Group examine the ICRP-66 lung model (during its proposal stage) in detail, by testing the predictions of deposition, clearance, and retention against experimental data, and by determining the model's implications for doses to the respiratory tract; and (2) to test the practicality of implementing the model.

LUDEP calculates doses to all body organs. It includes a bioassay module that allows calculations of excreted activity and retention in the lungs, other organs, and whole body.

The model contains several built-in databases, including radionuclide decay data from Oak Ridge National Laboratory and from ICRP-38; biokinetic models from ICRP-30; and bioassay functions from ICRP-54. ICRP data are generally used as the default values within the model, although the user is given the option to input case-specific parameters.

The ICRP-66 model that is implemented in LUDEP 2.06 was designed to realistically represent the deposition of inhaled particles in the respiratory tract, the subsequent biokinetic behavior of inhaled radionuclides, and the doses delivered to the respiratory tract.

The LUDEP code allows the user to input the particle size of an airborne concentration or intake. LUDEP allows the user to input the characteristic aerosol AMAD (or activity median thermodynamic diameter - AMTD) for a given airborne concentration or intake. The code contains a biokinetic model and organ dosimetry.

The metabolic and excretion models available in LUDEP are:

- ICRP-66 Lung model
- ICRP-30 Gastrointestinal (GI) model
- ICRP-30 General systemic model
- ICRP-30 Plutonium biokinetic model
- ICRP-54 Durbin Plutonium excretion model
- Jones Plutonium Excretion Model

LUDEP allows users to choose either the quality factors or organ/tissue weighting factors published in ICRP-26, or the radiation weighting factors and organ/tissue weighting factors published in ICRP-60. The bone dosimetry is a recycling model with initial uptake onto bone surfaces, transfer from surface to bone volume, and recycling from bone and other tissues to plasma.

D.2. MODEL TESTING AND COMPARISON

Selection of a computer program to support intake and dose assessment required a set of criteria to guide the testing and evaluation process. Criteria based on the ability to perform credible assessments from the data available were a prime objective. That is, the computer tool should demonstrate an ability to produce credible results with the data from 1966 and 1967. Considering all of this, our approach recognized a need to be able to estimate plutonium intakes from urine bioassay data, to calculate committed effective dose equivalents from those intakes, and to readily accommodate the available data without major conversion efforts.

D.2.1. Performance Criteria

The major task for this project involved an attempt to calculate intake from the urine bioassay information available. Other data from the response and cleanup operation simply do not exist to support intake estimates from air sampling or other means. Studies performed by JEN for decades following that effort offer some data for developing independent intake and dose estimates using environmental data. Nevertheless, the methods for estimating intake of plutonium by inhalation from the urinary data must be evaluated for performance and ease of use. Performing the intake assessment using this approach acknowledges that sizeable uncertainties can be expected because the assessments assume the characteristics of reference man rather than the specific characteristics of the individual involved.

Calculation of the organ dose equivalents and committed effective dose equivalent for each responder based on the intake must also meet accepted performance.

Finally, the selected method must have data requirements that can be met using the available data with as few conversions as possible.

These three criteria formed the primary basis for evaluating the performance of the three computer programs.

Ease of use provided a secondary factor for evaluating each of the three programs. This factor concentrated primarily on requirements for setting up input data sets and producing output data and reports that could be manipulated easily for a number of purposes – comparing the results of

testing the three methods, evaluating trends in intakes and doses for selected groups of subjects, data plotting and report preparation.

D.2.1.1. Performance on Intake Estimates

Review of the documentation for each of the three methods indicated that all employed generally accepted excretion models, i.e., either the ICRP-54 Durbin model, or the Jones model, or both. Implementation of calculation procedures for those excretion models seemed similar in that the approaches involved solutions to differential equations to determine the excretion patterns from estimated intakes.

The common approach among the models involved:

- calculating an initial estimate of intake from urine results,
- calculation of the expected urinary output rate (pCi/d or Bq/d),
- comparison of calculated urinary excretion to measured excretion using a form of statistical goodness of fit, and
- iteration until a selected calculation error was achieved.

The three methods were initially tested with an assumed excretion of 0.1 Bq/day (27 pCi/day) excretion rate at a series of sampling times after acute inhalation intake over one year. That is, for selected days, the urinary output of Class Y (Type S) ^{239}Pu was set at 0.1 Bq/day. The results of that test are shown in Figure D-6. In those tests, LUDEP provided estimates that were typically about 2 times higher than RBD estimates and about 3.5 times higher than CINDY estimates. The committed effective dose equivalents associated with those intakes are shown in Figure D-7.

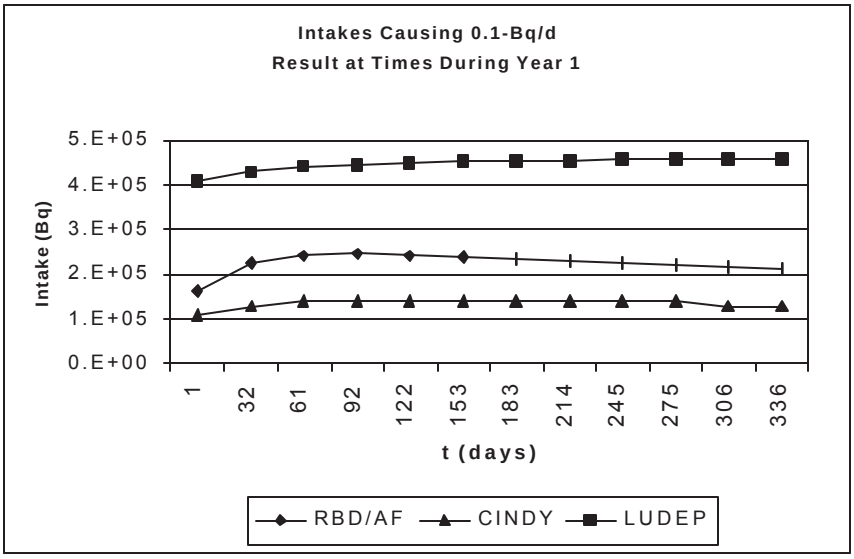

Figure D- 6. Intake estimates of the three methods.

Two of the three models (CINDY and LUDEP) offered options for weighting the measured results in performing the estimate. RBD/AF applied weighting based only on the relative contribution of multiple bioassay methods, e.g., results from urine bioassay and whole body counting.

CINDY's options include:

- Unweighted least squares: The weighting factors are assumed constant and equal, implying that the variance is independent of the magnitude of the measurement.
- Ratio of the means: The weighting factors are assumed inversely proportional to the expected value (as defined by the unit intake function). This assumption implies that the variance is proportional to the magnitude of the expected value.
- Average of the slopes: The weighting factors are assumed inversely proportional to the square of the expected value, implying that the variance is proportional to the square of the expected value.
- User-defined weights: The user supplies the estimate of the variance for each measurement value. The weighting factors are taken to be the inverse of the supplied variance.

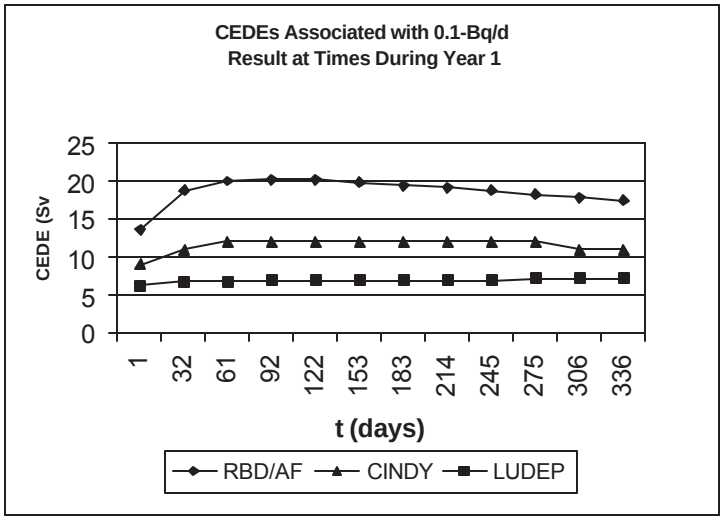

Figure D- 7. Estimated CEDE for three methods.

LUDEP offers the following options:

- Uniform absolute errors: The uncertainty values are a constant value, K.
- Uniform relative errors: Each uncertainty value is a constant proportion of the data point.
- Square root errors: Each uncertainty value is a constant multiple of the square root of the corresponding data point.
- Errors included in data set: The values of the uncertainties in the data, if known, are used.
- Logarithmic errors: Assumes the measured values fall about the true value with a log-normal distribution.

In comparing the approaches available in the two models, CINDY's "unweighted least squares", "ratio-of-the-means", "average-of-the-slopes", and "user-defined weights" methods seem to be roughly similar to LUDEP's methods using "uniform-absolute errors", "uniform-relative errors", "square-root errors", and "errors included in the data set." This conclusion results from evaluation of the discussion on weighting in the CINDY user guide (PNL 1992), summarized below.

Methods for comparing the estimated values with the measured values are based on the basic formula for weighted least-squares regression of a linear relationship with zero intercept as follows:

$$I = \frac{\sum_{i=1}^{n} w_i R_i X_i}{\sum_{i=1}^{n} w_i R_i^2} \qquad (1)$$

where I = estimated intake (pCi for acute intakes and pCi/d for chronic intakes)

w_i = least-squares regression weighting factor.

X_i = bioassay measurement for the ith data point (pCi/d for excretion and pCi for retention).

R_i = fractional retention or excretion estimate.

n = number of bioassay measurement points.

In CINDY, the four methods for intake estimation relate to four methods for defining the weighting factor, w_i. Ideally, the weighting should involve the variance of the measurement value (Bevington 1969). Each of the four methods, therefore, involves a particular assumption about the estimation of the variance.

In general, the intake estimate from the "user-defined weights" method is preferred when the input weighing factors represent good estimates of the variance of the measurement. Alternatively, the "ratio-of-the-means" intake estimate is probably the best estimate because the weighting is based on an estimate of the variance as proportional to estimated bioassay result. This method generally gives better "eyeball" fit to the bioassay data (PNL 1992).

The unweighted least-squares regression analysis is expressed by the following equation:

$$I = \frac{\sum_{i=1}^{n} X_i R_i}{\sum_{i=1}^{n} R_i^2} \qquad (2)$$

where terms are as previously defined. This method may be used when all measurement values are expected to have similar accuracy and all are significantly above the detection limits of the measurement method. This method could also be referred to as "uniform weighting" because all weights, w_i, are assumed equal in derivation of Equation 2 from Equation 1.

The "ratio-of-the-means" method is based on the assumption that the variance of the expected value is proportional to the magnitude of the expected value. The weights are expressed as follows:

$$w_i = \frac{1}{kIR_i} \tag{3}$$

where k is a constant of proportionality. Substitution of Equation 3 into Equation 1 results in the following expression for the intake estimate:

$$I = \frac{\sum_{i=1}^{n} X_i}{\sum_{i=1}^{n} R_i} \tag{4}$$

As can be seen from this expression, the intake estimate is just the ratio of the sum of the measured values to the sum of the unit intake function values. This is equivalent to the ratio of the means of the measured values and the unit intake function values (proportional to the expected values), hence, the name "ratio-of-the-means" method. Note also that from Equation 4, the sum of the measured values is equal to the sum of the expected values:

$$\sum_{i=1}^{n} X_i = \sum_{i=1}^{n} IR_i \tag{5}$$

This method is appropriate when the variance of the measurement is expected to be proportional to the measured value.

The average-of-the-slopes method is derived from Equation 1 by defining the weights as inversely proportional to the square of the unit intake function values:

$$w_i = \frac{1}{kI^2R_i^2} \tag{6}$$

The resulting expression for the intake estimate is as follows:

$$I = \frac{\sum_{i=1}^{n} \frac{X_i}{R_i}}{n} \tag{7}$$

This expression gives the average of the ratios of measurement value to unit intake function value, which is equivalent to the average of the slopes of the equation

$$X_i = IR_i \tag{8}$$

This method is appropriate when the variance of the measurement is expected to be proportional to the square of the expected value.

The user has the option of identifying the variance for each measurement data point. The fourth method (user-defined weights) uses this statistical parameter as an inverse weight in Equation 1:

$$w_i = \frac{1}{V_i} \tag{9}$$

where V_i is the user-supplied statistical parameter value for bioassay measurement i. This method allows the user to implement almost any weighting method desired based on predetermined weights. In evaluating the intake estimate using this method, only the data points having a defined value for V_i are used in the calculation.

As an example of the use of the "user-defined weights" method, consider a set of bioassay data values that includes an estimate of the standard deviation of the measurement value. The user-defined weights method can be used to provide an intake estimate based on the variance of the measurement values. To perform the analysis, the reported standard deviations are squared to provide the values for the weights to be entered into the CINDY program. This results from the assumption that the variance of the measurement is represented by the square of the standard deviation of the measurement. The code will use the inverses of the squared values as weights in Equation 1 to give an estimate of the intake with variance weighting.

As noted above, selection of the weighting method and any factors are important for reasonable results.

A number of cases were developed for testing the performance on estimating intakes. The primary data used were derived from the group of High 26 individuals from the Palomares follow-up. These were the only cases of data available with multiple bioassay measurements taken over an extended period – 12 to 18 months from the time of the accident. Unfortunately even those data raised questions about the actual dates of sampling and exposure, the reliability of results and other matters. Significant concerns arose from the use of gross alpha counting of initial samples and the possibility of contamination of samples collected on site (See Section 2 and Appendix B).

Using the bioassay data for two individuals who each had multiple samples taken, intakes and associated CEDEs were estimated by LUDEP, CINDY, and RBD/AF. The results indicated the estimated intakes were highest using LUDEP, lowest using CINDY, and intermediate using RBD/AF. The 50-year CEDEs were highest using RBD/AF, while the other two models provided lower results—in one case, LUDEP's CEDE was slightly lower than that predicted by CINDY, with the order reversed in the other case. The greatest difference in predicted CEDE was a factor of 2.2.

Using a subset of the bioassay results (all individuals with sample results greater than 10 pCi/sample from the initial spreadsheet provided by the Air Force), CEDEs were estimated by RBD/AF, CINDY, and LUDEP. As in the previously described case, RBD/AF generally predicted higher results, while those of CINDY and LUDEP were more similar.

The bioassay results for the "High 26" were modeled using CINDY and LUDEP to determine intakes and CEDEs. When CINDY doses were estimated using the "ratio-of-the-means" method, the CINDY CEDEs were higher than those predicted by LUDEP by an average factor of 13.5. When the "user-defined weights" method was used in CINDY, the CEDEs exceeded those predicted by LUDEP by an average factor of 1.5.

For CINDY and LUDEP, the estimated errors from counting, as reported on the data forms, or recalculated from the raw data on the forms, were used to calculate the statistical variance, which

was used as the input value in the "user-defined weights" option for CINDY; the counting error itself was used in the "errors included in data set" option for LUDEP. The estimated counting errors involved some inconsistency – they were reported at 95% confidence level for gross alpha results and at the 68% confidence level for alpha spectrometry; this difference was taken into account in calculating the variance used in the CINDY "user-defined weights" option. Often, the later results were reported as No Detectable Activity. In that case, a value of 0.009 pCi/day was assumed for gross alpha results, and a value of 0.003 pCi/day was assumed for alpha spectrometry results. The errors in those were set at 25% of the value; which may be somewhat low for the level of activity.

Using both the CINDY and LUDEP models, the sample data sets for the "High 26" were input to estimate CEDEs for each individual, using first all the samples, then excluding those that were analyzed by gross alpha, which would correspond with the early samples taken onsite. The results show that the CEDEs are generally lower when gross alpha results are excluded, averaging a 24% or 62% decrease in CINDY results (depending on weighting factor used—see next paragraph) and a 6% decrease in LUDEP results. This difference between models may be due to a noted tendency of LUDEP to weight sample results for longer times after exposure more strongly in calculations using multiple bioassay data points.

When gross alpha data were included in the CINDY model runs, the CEDE using the "ratio-of-the-means" method exceeded the CEDE using the "user-defined weights" method by an average factor of 13. The CEDE from the "user-defined weights" method exceeded the CEDE from the "ratio-of-the-means" method in only 2 of the 26 cases. When gross alpha data were excluded, the CEDE from the "ratio-of-the-means" method exceeded the CEDE from the "user-defined weights" method by an average factor of 3.4; in 3 cases the CEDE from the "user-defined weights" method exceeded the CEDE from the ratio-of-the-means" method.

In general, from other tests, the "user-defined weights" estimates tended to apply more significance to measurements taken at longer elapsed times from exposure. Coincidentally, those values were generally much lower than the early measurements and had much lower absolute values for the variance, which was estimated from the counting error.

For LUDEP, similar comparisons of the performance of the assumed errors options revealed reasonable agreement among results from the "uniform-absolute errors", the "uniform-relative errors", the "square-root errors" and the "errors included in the data set" options when applied to the actual urine results of three of the High 26 Cases Group. Those agreements were achieved for reasonable values of K (0.25 to 1), and showed agreement within about 50%, which seems acceptable considering the nature of the data. The logarithmic errors option produced estimates of intake that were 3 to 4 times higher than the other methods.

For CINDY, the "user-defined weights" method also seemed to attribute greater significance to lower values of results, yielding lower values of intake. In effect, the approach seemed to ignore other measured values. After multiple attempts to better characterize CINDY performance and consultation with its developers (Traub 2000), we concluded that the uncertainty in the estimated errors themselves contributed to this performance, and the "user-defined weights" method was no longer used. The "ratio-of-the-means" method, recommended by the CINDY user manual (PNL 1992), showed reasonable performance and was selected as the method to be used.

When other factors were held equal, intakes estimated by CINDY (using the "ratio-of-the-means" weighting method) and LUDEP (using the "errors included in the data set" option)

agreed to within a factor of two for the majority of the High 26 Cases Group. Given the variability of the data, the agreement was deemed reasonable and the performance acceptable for the type of assessment performed.

D.2.1.2. Performance on Dose Calculations

The performance evaluation tested conversion of intakes into committed dose equivalent in organs or tissues and calculation of committed effective dose equivalents with RBD/AF, CINDY and LUDEP. Testing the dose performance involved two separate efforts: basic assessments using assumed intakes, and assessments of selected cases from the High 26 Cases Group.

The basic assessment test consisted of assessments of the same set of Palomares data derived from the first 29 entries in the data listing (see Appendix B) provided by the Air Force. These data consisted of single urine measurements (generally of 10 pCi/day or more), collected at the accident site during the accident response effort. RBD, CINDY, and LUDEP calculated intakes and doses for each of the 29 cases. Committed effective dose estimates from the three programs varied by no more than a factor of about two from the highest dose to the lowest dose estimate for each case, with RBD/AF generally giving the highest estimated CEDE; LUDEP yielding the lowest; and CINDY providing intermediate dose estimates. That LUDEP produced the lowest doses seems consistent with the findings about its performance discussed above.

The second part of the testing involved actual test cases from two members of the High 26 Cases Group. Those cases had several urine measurements taken on site and during the follow-up period. RBD/AF, CINDY, and LUDEP provided estimates of the intake and dose for these two cases. These cases were calculated with several variations involving exclusion of selected urinary measurements for reasons, such as suspected contamination, possible chemical recovery issues, results below the detection limit, or simply to evaluate the behavior of the programs. The results of these tests confirmed the tendency of the methods to favor urine results with lower values, taken at long times after exposure. Generally, the CEDEs were highest for RBD/AF, lowest for LUDEP, and intermediate for CINDY. Again, results differed by no more than a factor of two. That performance seems acceptable.

Finally, CINDY and LUDEP were tested further with the entire High 26 Cases Group. In tests paralleling the intake assessments, CEDEs were also estimated with and without gross alpha results. LUDEP provided estimates that were about 30% lower than CINDY when gross alpha results were excluded, and from about 30% to 90% lower than CINDY when all urine measurements were included. Considering the nature of the data, the results are acceptable.

D.2.1.3. Ability to Satisfy Data Requirements

Parameters required for calculating estimates of intake from urine bioassay and the associated dose equivalents satisfy the model selected to perform the task. Computer software that implements the models establishes unique processes for satisfying the data input needs. The three computer methods were evaluated for the compatibility with available urine bioassay data. Primary parameters included the date of exposure, date of sample, radionuclide, type of exposure, pathway, particle characteristics, lung type or class, results and units, and sample volume, among others. The requirements of each program are discussed and compatibility with the available data assessed.

RBD

Data items and that may require assumptions to achieve compatibility include:

- **Date** – Since the Palomares data reflect exposure due to an incident, rather than a series of routine monitoring measurements, the date of the exposure incident is required. In some cases, this will have to be estimated based on the data on each dose data card, such as when a range is presented. In some cases, the date of exposure and date of sampling recorded on the dose data cards are the same. Unless adjusted based on additional information or other reasonable assumptions, this will result in an error during model execution.
- **Time** – The time of exposure does not affect the execution of the model if it is left blank.
- **Nuclide** – Data for ^{239}Pu are included in the files of the model.
- **Pathway** – As in the previous studies, inhalation exposure only can be assumed.
- **AMAD** – The default value of 1 μm can be used; range is 0.2 to 10 μm.
- **Class** – For inhalation, ^{239}Pu can be either Class W or Class Y. If it is assumed that all ^{239}Pu is in the form of PuO_2, then Class Y should be assumed, per ICRP-30.
- **Measurement date** – In some cases, this must be assumed due to incomplete data on the dose data cards.
- **Measurement time** – The time of measurement does not affect the execution of the model if it is left blank.
- **Result and Units** – The results on the dose data cards must be converted to units that are accepted by the model. For urinalysis, the options are dpm/mL, dpm/day, dpm/sample, dpm/L, μg/mL, Bq/L, Bq/day.
- **MDA** – The minimum detectable amount does not appear to be generally available on the dose data cards. A value could be estimated. The model will accept a zero value in this field.
- **Volume** – Sample volume is required if results are input in units of dpm/sample; otherwise, it can be left blank.
- **Volume/day** – The urinary volume per day is required for execution. The default is 1400 mL; the existing Palomares reports state that a value of 1200 mL was used as a default.

Overall, data are sufficiently available or can be reasonably estimated to run the RBD/AF model using the Palomares internal dose data. However, the nature of the available data could result in potentially large relative errors in time from exposure to sampling, which could have a significant impact on the validity of any resulting conclusions as to the intake and committed effective dose of a particular individual. It is however, reasonable to assume that these errors will average out over the large data set available, leading to conclusions that are more supportable for the exposure cohort as a whole.

Other model specific parameters are available as defaults appropriate for the model within the program and supporting data files. These seem reasonable or can be readily modified.

CINDY

Most data items required to perform the calculations are available and compatible. Those data items that may require assumptions to achieve compatibility include:

- **Excretion period** – Set to 24 hours if not specified otherwise.
- **Intake mode -** Acute inhalation is assumed; can be changed.
- **Date and time of intake** – Based on data reported on bioassay cards. Time set to 12:00 PM since no times were reported, however the impact is unimportant for the radionuclide involved.
- **Particle size** – 1 ìm assumed.

Overall, data are sufficiently available or can be reasonably estimated to run the CINDY model using the Palomares internal dose data. However, the nature of the available data could result in potentially large relative errors in time from exposure to sampling, which could have a significant impact on the validity of any resulting conclusions as to the intake and committed effective dose of a particular individual. It is however, reasonable to assume that these errors will average out over the large data set available, leading to conclusions that are more supportable for the exposure cohort as a whole.

LUDEP

Most data items required to perform the calculations are available and compatible. Those data items that may require assumptions to achieve compatibility include:

- **Intake -** Acute intake by the inhalation pathway can be assumed.
- **AMAD** – A value of 1 ìm can be assumed.
- **Absorption Type** – This factor introduced in ICRP-66 as F, M, or S for default absorption values corresponding to fast, medium, or slow absorption. Type S, which corresponds to the Class Y designation of PuO_2, can be assumed.
- **Time after intake (days) -** In some cases, this must be assumed due to incomplete data on the dose data cards.

Overall, data are sufficiently available or can be reasonably estimated to run the LUDEP model using the Palomares internal dose data. However, as with the programs, the nature of the available data could result in potentially large relative errors in time from exposure to sampling, which could have a significant impact on the validity of any resulting conclusions as to the intake and committed effective dose of a particular individual. It is however, reasonable to assume that these errors will average out over the large data set available, leading to conclusions that are more supportable for the exposure cohort as a whole.

The three programs provide adequate data compatibility. LUDEP uses SI units of becquerels (Bq) for radioactivity, and sieverts (Sv) for dose equivalent. However, conversion of units from picocuries per day (pCi/d) to becquerels per day (Bq/d) can be easily accommodated.

D.2.1.4. Ease of Use

With over 1,500 individual cases potentially requiring assessment, data input, result output and other manipulations can impact efficiency. Each program was assessed for features of convenience or difficulty that could impact effectiveness.

RBD/AF

Input features of RBD/AF include:

A data input screen for bioassay data with the choices for selectable entries for: gender, base code, assay, reason, nuclide, pathway, AMAD, class, in-vitro assay (measurement date, measurement time), result (unit – for urine, units can be dpm/mL, dpm/day, dpm/sample, dpm/L, ìg/mL, Bq/L, Bq/day), MDA, volume, and volume/day.

The program stores the data in files describing sets of cases, facilities or other convenient means. This allows data preparation, calculation, and reporting to be conducted as separate activities.

Output features of RBD/AF include:

Estimated intake (in Bq and ìCi), estimated intake as a percent of the ALI, ALI (in Bq), committed dose equivalent (in ìSv and mrem, by organ/tissue), and effective dose (in ìSv and mrem). An optional graph of excretion rate vs. time can also be generated.

The summary output report presents, by individual committed dose equivalent (in mrem, by organ/tissue), effective dose (in mrem).

The summary output report is presented in a space-delimited file, that is easily imported into a spreadsheet (with only minor editing required) for manipulation and sorting.

CINDY

Input features of CINDY include:

Subject identification: name, identification number, SSN, dates of birth, sex, file name prefix.

Subject/Bioassay Measurement-Specific: exclusion flag, bioassay type, bioassay radionuclide, sample end date and time, excretion period, measured value, measurement inverse weighting factor, measurement unit numerator (pCi/nCi/dpm/Bq), unit denominator type, sample size and units.

Subject/Intake Specific: exposure duration, intake mode, begin date and time of intake, end date and time of intake, particle size, facility, employer at time of intake, radionuclides of concern, intake estimate.

Run-Specific: dose report times, dose reporting limits, bioassay projection endpoint, bioassay projection report times, bioassay projection graph selections, text report selections, radiological working units options, error tolerances, radionuclide daughter handling, model selection, and model parameter values.

Output features of CINDY include:

- Several different output reports: For the current effort, useful data points are found on the subject report, which reflects data inputs and normalization, as well as the intake assessment summary and dose assessment reports.
- Intake Assessment Report: includes intake estimate, lung model details, mean residence time in each compartment of GI tract, and urinary excretion model details.
- Dose Assessment Report: includes dose equivalent, weighting factors, and organ dose equivalents, by organ; effective dose equivalent; lung model details; and systemic model details.

- Optional display of a urinary excretion curve on the monitor or printed using text characters.

CINDY output formats can be saved in formats that are easily imported into most personal computer application software.

LUDEP

- LUDEP includes data input screens for the sequence of calculations necessary to estimate an acute intake from urine bioassay data that include:
- Intake (acute or chronic, inhalation or ingestion or injection, value entered in Bq (acute) or Bq/day (chronic)), or exposure (concentration in Bq/m^3 and duration in hours);
- Deposition (AMAD (ìm));
- Absorption (F, M, or S),
- Radionuclides;
- Biokinetic model,
- Quantity to calculate (whole body retention, lung retention, urinary excretion rate, fecal excretion rate, or specified organ retention);
- Function (ICRP-54 function or enter own function);
- Number of points: days (in this case) that encompass all sampling intervals;
- Time: enter a start and stop time, in days;
- Urine Sample Activity Data: time after intake (days), measured activity (Bq), and estimated uncertainty (if known)

LUDEP does not generate a printable output report. Results are displayed on-screen. The output for the calculation of intake based on urinary bioassay sample data provides a best estimate of intake (Bq), standard error of intake (Bq), 95% confidence limit on intake, chi square test statistic, and probability.

LUDEP operates solely as an interactive, desktop program that requires substantial effort to set up and operate. Input parameters can be established for exposure scenarios, saved in files, and used for multiple cases. Organ dose results can be saved to files, as can urine excretion data. Overall, LUDEP does not provide the reporting convenience of RBD or CINDY.

D.2.2. Sensitivity of parameters

Estimated intakes and associated doses depend on the selection of the various input parameters and data. These parameters determine how the intake, biokinetic, and excretion models treat the characteristics of each case. Some of those parameters depend on the characteristics of the exposure scenario, while others depend primarily on the models themselves. In the latter case, ICRP provides recommended values for many of these parameters based on calculating estimates for reference man.

D.2.2.1. Time from Exposure to Sampling

Exposure dates and sampling dates in Palomares records have substantial uncertainty. When recorded, the data are quite specific. When not recorded, or when several samples were collected on different dates, determining a representative acute exposure date can involve an element of

subjectivity. This issue also relates to determining the type of exposure – acute or continuous – as discussed in the next section. The effect of the time between exposure and sampling on estimated intake was assessed with a simple test that varied the time only for a fixed urine excretion value. The time values were varied in increments of one month for a period of two years. Estimated intakes from CINDY varied from 15% for the first month to 7% for the second and third months with a total decrease of 18% over the two-year period. LUDEP results decreased by 5% at one month to 2% at the second month with a total decrease over the first year of 12%. At worst, the differences during the first 30 days should be less than 15 % for CINDY and about 5% for LUDEP.

D.2.2.2. Use of Multiple Bioassay Measurements.

Multiple bioassay measurements affect the estimated intake primarily through the process of obtaining the best fit of the calculated expected values of excretion to the measurements. Testing the methods showed that the selection of weighting factors in CINDY (errors in data sets in LUDEP) could have substantial effect on the intakes. The variations in those were discussed in Section D.2.1.1. The methods performed acceptably within the boundaries of the expectations for the available data.

D.2.2.3. Particle Size

Using LUDEP, the estimated intakes of inhaled ^{239}Pu particles of different AMADs were compared. In one test, the series of bioassay results for one individual were input using AMADs of 0.5, 0.6, 0.7, 0.8, and 1.0 µm. Decreasing the AMAD between 1.0 and 0.5 led to a decrease in the estimated intake; the difference over the entire range tested was less than 8% of the intake associated with an AMAD of 1.0 µm. In another evaluation, the organ dose equivalents to organs were modeled using AMADs of 1, 2.5, 5, 7.5, and 10 µm as shown in Figure D-8. In this case, the organ dose equivalents decreased more than 70% over the range from 1 to 10 µm in all organs except the ovaries and the organs of the gastrointestinal (GI) system. There was no change in the doses to the ovaries, and doses to the GI organs increased from 7 to 23%. Overall, there was a decrease of 75% in committed effective dose equivalent (Figure D-9) when AMAD was varied from 1 ìm to 10 ìm, and a decrease of 40% when the AMAD was increased from 1 µm to 5 µm.

These two comparisons indicate that using an AMAD of 1.0 :m in LUDEP leads to the highest estimated doses, and would therefore be the most conservative estimate of particle diameter. ICRP-30 recommended a default AMAD of 1.0 ìm, but ICRP-66 recommended 5 ìm as generally more representative in occupational settings in the absence of specific information. The variation of organ dose equivalents and committed effective dose equivalent with particle size is acknowledged. A value of 1 ìm AMAD was selected for modeling calculations as a conservative measure.

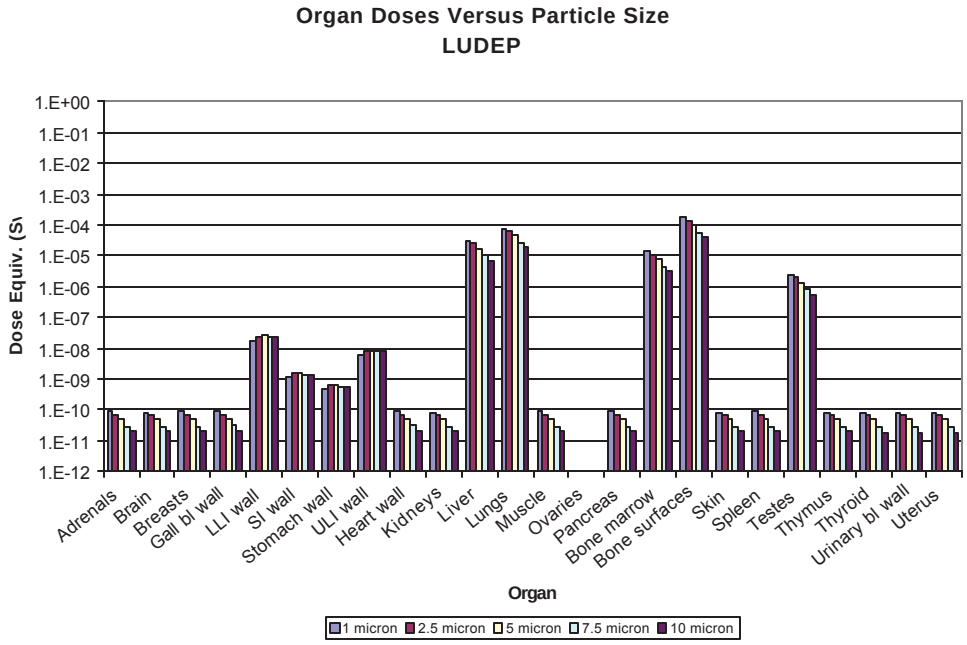

Figure D- 8. Variation of organ dose equivalent with particle size in LUDEP.

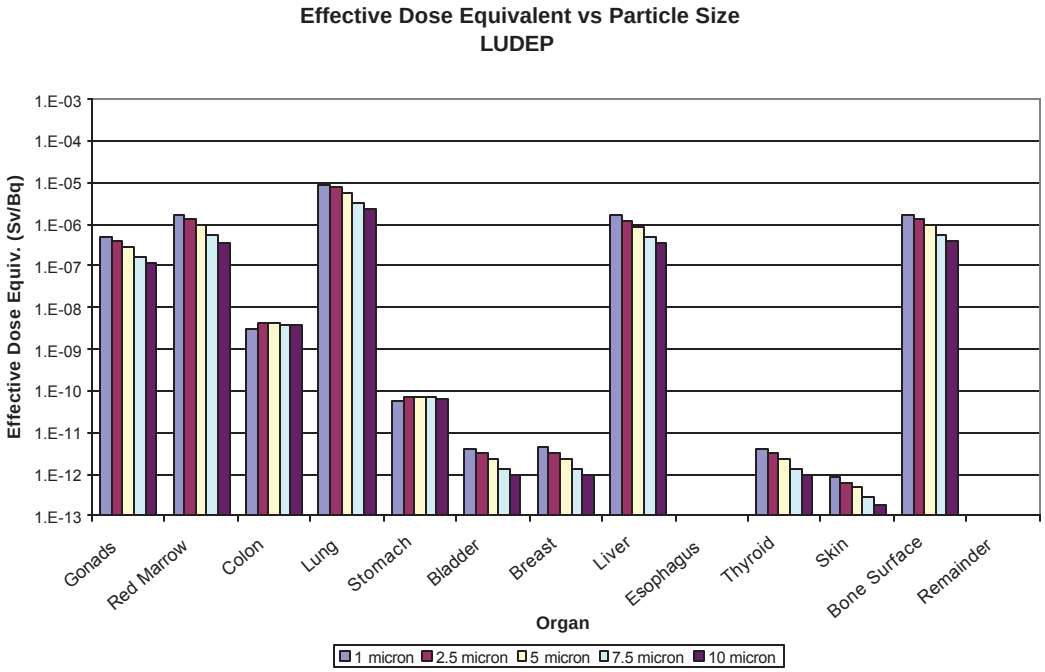

Figure D- 9. Variation of committed effective dose equivalent with particle size from LUDEP.

D.2.2.4. Type of Exposure

The data for some cases indicated possible exposures at several times during the two to three months on site. Evaluation of these cases could assume either a single acute exposure or a series of exposures similar to a continuous intake over the time. CINDY provides for either type of exposure scenario. A quantitative comparison of the two possible exposure scenarios was conducted. In all cases attempted, the estimated intake for an acute exposure was higher than the estimated intake for a continuous exposure, with an average increase of 50% and ranging up to 110%. When the range of exposure dates is reasonably well known, CINDY yields little difference in the results obtained by assuming either an acute (median exposure date) or continuous exposure. The differences in the two methods (acute vs. continuous) become greater as more assumptions are required to establish the dates of exposure. The results were very close when a range of dates was provided, varied significantly when only one date was provided, and showed the largest variation when assumptions were required for both the beginning and end of the exposure period. When only one date was entered on the bioassay data card, significant (>50%) differences resulted for the acute and continuous estimated intakes for 22 of 30 individuals. The highest difference was an 80-percent increase in estimated intake using the acute mode. When a range of dates was entered on the bioassay data card, there were no significant differences in the estimated intakes when either the acute or continuous approach was used. When no exposure date was entered on bioassay data card, significant differences occurred in intakes estimated for seven out of eight individuals, ranging from 70 to 110 percent.

The LUDEP model as currently configured requires significant additional effort to calculate continuous exposures when there is a time lapse between the end of exposure and the collection date for a bioassay sample. The number of manipulations required to perform this assessment were manageable for a few cases; however, the method was very unwieldy, and judged error-prone when applied to hundreds or thousands of cases.

In all comparisons, the estimated intake assuming acute exposure was higher than the estimated intake assuming continuous exposure, with an average increase of 50% and ranging up to 110%. These results emphasize the sensitivity of the estimated intake to the exposure date range.

D.3. MODEL ADOPTION

Taking the four factors considered above, RBD/AF, CINDY, and LUDEP all provide acceptable performance on estimating intake, calculating dose, and providing compatibility with the available data. LUDEP is somewhat less convenient for manipulating large numbers of cases and for generating outputs that can be used in other manipulations; however it implements the current ICRP respiratory tract model.

CINDY and RBD/AF implement the current regulatory system of the NRC and DOE for radiation protection, while LUDEP offers the alternative for applying the respiratory tract model and other features of recent ICRP recommendations. CINDY provides somewhat more flexibility in setup, estimating intakes, and reporting. Consequently, CINDY was chosen as the primary method for assessing the Palomares cases. LUDEP was retained as a reasonable alternate that provides complementary assessments for interesting cases and offers a much-needed point for comparison of results.

APPENDIX E

ESTIMATES FROM URINALYSIS

ESTIMATES FROM URINALYSIS

Urinalysis results existed for 1,758 samples collected from 1,555 individuals taken over a period extending from a few days to about 2 years following the accident. Earlier samples, collected on site, indicated the strong possibility of contamination. Follow-up samples, collected after personnel returned to their permanent base of assignment, showed dramatically lower concentrations. In 1968, those results demonstrated that no responder received a systemic body burden above a small fraction of the maximum permissible body burden (MPBB) – the standard for comparison at the time. This conclusion support expectations that estimates of intake and dose using currently accepted methods could support similar conclusions. This appendix discusses the urine data, provides preliminary estimates of intake and dose, and draws conclusions about the reliability of the estimates.

Estimates of intake and dose based on urinalysis for plutonium proved to be affected by numerous technical difficulties that made the results unrealistic compared to other plutonium exposure cases from industrial and environmental settings. Nevertheless, review of the extensive urinalyses confirmed the conclusions about the minimal impacts on the health of the responders made during the post-accident evaluations in 1966 through 1968. Furthermore, this effort completed a much-needed organization of the data, consistency checks and revisions, and preparation of the data for use in future evaluations.

E.1 DATA

The Air Force Institute of Environment, Safety, and Occupational Health Risk Analysis (AFIERA) and the Air Force Medical Operations Agency (AFMOA) provided records in the form of:

- Air Force Forms with laboratory analytical and exposure details of the nasal swipe and urine samples submitted and processed.
- Complete case files for the 26 individuals identified for follow-up in 1966 and commonly referred to as the "High 26".
- A Microsoft Excel spreadsheet prepared by AFIERA staff that contained the data from those Air Force Forms, and some data related specifically to the 26 individuals (referred to as the "High 26" who were considered as having the highest exposures).
- Copies of the accident response reports, USAF RHL documents on the evaluation of exposures by urinalysis, and selected publications from journals and conference proceedings.

Appendix B contains a detailed discussion of the information collected, an evaluation of the information's suitability for a dose evaluation, and adjustments made to the data for performing intake and dose calculations. The record prepared and maintained by the Air Force consisted of forms, computer spreadsheets, and written correspondence and reports of activities.

E.1.1. Forms

The USAF Radiological Health Laboratory (USAF RHL), the central laboratory for providing radiological services to Air Force units in 1966, recorded the data and results of samples processed on three series of forms: AFLC Form 1165, Internal Dosimetry Data (May 66), AFLC

Form 1165, Radiological Sample Data (May 66), and AFLC Form 1165, Radiological Sample Data (Jul 67). Although similar in design and content, these three forms evolved over the course of the laboratory effort on Palomares. The three forms recorded the data about the individual who submitted the sample, radiation measurement data for urine, radon (breath) (sic), and feces/blood samples, counting data, instrument data, and other factors; and finally the results. The Internal Dosimetry Data (May 66) form apparently served primarily for the samples processed during the initial, or on-site, phase of the response. Figure B-1, Appendix B illustrates that the May 66 version of the form contained information from samples collected in April 1966. The Radiological Sample Data (May 66) form was used to record alpha spectrometry data for most of the follow-up phase. The Radiological Sample Data (Jul 67) form was used during the end of the follow-up phase.

Consistency among the entries on the data forms and the entries in any ultimate data set would be required. The data cards formed the only permanent record available of the actual data generated at the time of the incident. Consequently, they provided the primary means for verifying information from other sources, at least when the data on the cards were unambiguous.

E.1.2. Spreadsheet

AFIERA representatives also provided a copy of a Microsoft EXCEL® spreadsheet that contained the basic data transcribed from the hardcopy data forms into the spreadsheet. Figure B-6, Appendix B contains an example of one page of the spreadsheet to illustrate its contents. The spreadsheet contains information on some 1,758 individual entries for 1,555 individuals. The spreadsheet served as a good starting point for evaluating the data contained on the hardcopy records.

E.1.3. Reports

Several other documents provided essential information about the details of the accident, the response effort, and the approach to evaluating health and safety issues during the response. These documents provided a narrative overview of the approach to assessing possible exposure to plutonium at Palomares. The discussions highlighted the issues faced, the problems encountered, and the rationale that formed the basis for the effort and decisions made throughout the period of on-site activity and subsequent follow-up. These issues are discussed in some detail in Section 2 above. The issues related to possible sample contamination, the sample collection period, and the exposure type and date formed the basis for evaluating the suitability of the data for the evaluation effort.

E.2 EVALUATION OF THE DATA

E.2.1. Condition of the Data

The data were evaluated to assess the availability of the elements required by the internal dosimetry models, including: the type of intake (inhalation, ingestion, skin contact), the date or dates the exposure occurred, the date of collection of nasal swab or urine samples, the duration of the urine sample collection, and the results of the sample analysis. Review indicated that the exposure date or dates, sample date, and results were not completely recorded for all cases. The

collection of information was reviewed by comparing the spreadsheet and data forms to determine whether all forms were present in the spreadsheet and whether the entries were correct. The initial evaluation identified a number of problems with the spreadsheet and supporting forms as shown in Table E-1.

This initial review indicated that substantial numbers of samples lacked one or more important pieces of data and identified 115 data forms that apparently represented a repeat analysis of a sample or a follow-up sample for an individual. Following the initial review, many of the missing entries were corrected through careful analysis of the information and reasonable assumptions about the missing information.

The duration of sample collection is critical to estimating the daily excretion rate of plutonium in urine. Air Force reports indicated that sample collection lasted 12 hours for many samples collected at Camp Wilson. The Air Force corrected the result for any urine sample of less than 1200 milliliters to 1200 milliliters. This conservative procedure would tend to overestimate urinary excretion. Our review indicated that 12-hour samples were clearly designated in 42 of the samples; however, attempts to duplicate the Air Force estimate of systemic body burden revealed that the sample volume correction might have been applied inconsistently. However, this did not adversely affect any conclusions about the individuals tested. Our review concluded that adjustments to samples that were not designated as 12-hour samples presented were unnecessary. Therefore, recorded sample volumes were assumed to represent 24-hour output unless specifically designated as 12-hour samples.

Table E- 1. Issues with dose records.

Issue	Number of Entries	Percentage
Exposure Date Not Available	402	22.7
Sample Date Not Available	445	25.1
No SSN Available	385	21.8
No Air Force ID Available	2	0.11
Sample Vol. < 600 mL	323	18.3
Sample Vol. > 1000 mL	434	24.5
Number with Additional Sampling Data (2nd page)	115	6.50
Number of Cards Marked Out	2	0.11
Number of Cards Not Found	5	0.28
		Total Number of Samples = 1768

Missing or incorrect entries for Exposure and Sample Date also hinder a reasonable estimate of intake and radiation dose. Additional analysis would be required to establish these parameters.

Other observed issues included missing Social Security Numbers (SSNs), Air Force Service Numbers (AFSNs), and other entries. Many of those records pertained to a broad spectrum of responders – from Air Force to other Services (Army, Navy, Marines); other US agencies (State

Department, Bureau of Mines), possible Spanish civilian employees of Torrejon Air Base or local citizens, and at least one media representative.

E.2.2. Sample Collection and Handling

Urine sampling was begun within three days of the accident. Urine sample collection on site was subject to several compromises. First, isolation of responders for 24 hours was desired and attempted but operational requirements limited the period to 12 hours or less. Opportunities for sample contamination from strong winds frequently spread dust over the base camp; decontamination procedures were not always followed; make-shift sample containers were used, and even when preferred containers were obtained, storage areas were frequently contaminated by blowing winds.

Nasal swabs were also collected and submitted to the laboratory, however, records indicated that of the 122 nasal swab records reviewed, 109 did not contain a result, 13 contained a result (8 were 0 pCi, 4 had values all below 1.5 pCi, and 1 was reported as NDA). Therefore, the nasal swab records were not used in this analysis.

Laboratory personnel observed alpha particle contamination on the outside of sample containers from the operational site very early in the program. This immediately raised issues about whether any alpha activity detected in urine represented material excreted by responders. Follow-up sampling was recognized as one means for resolving issues of possible contamination for persons with urine levels indicating significant exposure.

Upon receipt at the laboratory, a unique sample number was assigned, the samples were recorded into a sample logbook, and the AFLC Forms, discussed above, were completed. Attempts to locate the logbook(s) were unsuccessful. Samples were then submitted for the selected radioactivity analysis procedure.

During the follow-up sampling effort, sample containers obtained specifically for the purpose and tested for contamination were used to collect urine specimens from individuals. Whenever possible, sample collection was conducted at an Air Force medical facility under controlled conditions to reduce the likelihood of mishandling and to fulfill the need for a legitimate 24-hour collection period.

E.2.3. Sample Analysis Procedures

The USAF Radiological Health Laboratory processed the urine samples in a two-phased program – an initial phase and a resample phase. During the initial phase, samples collected on site were processed by a gross alpha counting procedure with preliminary chemical processing to extract any alpha emitting radionuclides from the bulk urine sample (Odland 1968a).

E.2.3.1. Initial Phase Procedures

During the initial phase, samples were processed for counting by: digesting a portion of the urine sample with nitric acid and hydrogen peroxide to a white residue; dissolving the residue and coprecipitation of plutonium with bismuth salts; dissolving the salts with hydrochloric acid, addition of lanthanum carrier, and coprecipitation of plutonium on lanthanum fluoride; and direct

mounting of the precipitate onto 2" stainless steel planchets for gross alpha counting (Odland 1966).

A small amount of ^{239}Pu tracer was added to pooled urine and processed in the same batch as Palomares samples. The added tracer served as an indicator of the effectiveness of plutonium recovery, which was reported to average 75.6 ± 19.6% (68% confidence) (Odland 1966).

The samples were counted in internal proportional counters optimized for detecting alpha particles. Daily checks monitored instrument response, and daily background counts were done. According to reports (Odland 1966), samples were counted for 120 minutes, and background was counted for 960 minutes. Review of the initial data indicated that samples were often counted for 55 minutes. Background was reported to range from 0.02 to 0.06 count per minute and counting chambers were decontaminated whenever the background count exceeded 0.1 count per minute.

Gross alpha results were reported in pCi/sample, where:

$$\text{pCi/sample} = \frac{(\text{gross counts/gross ctg time}) - (\text{background counts/bkgrd ctg time})}{(\text{counting efficiency})(2.22)(\text{procedural yield})}$$

Analysis of selected samples from the initial phase indicated that the results and estimated errors were calculated, recorded, and reported. The estimated errors were determined from counting data only and were reported at the 95% confidence level.

Procedural yield was determined from the results of the traced urine sample for each batch of urine processed.

E.2.3.2. Resample Phase Procedures

During the resample phase, the laboratory derived its procedures from those used for monitoring workers at other facilities handling significant quantities of plutonium. The process involved nitric acid digestion, coprecipitation of alkaline earth and plutonium phosphates, precipitation with cerium, ion exchange to remove interfering ions, and electrodeposition onto stainless steel planchets for radioactivity counting. A small quantity of ^{236}Pu was added to each sample before chemical processing to evaluate radiochemical recovery.

Radioactivity counting was conducted using alpha particle spectrometry with solid-state surface-barrier detectors in a vacuum. Count data were collected with a multichannel pulse-height analyzer. Detector efficiency and background were monitored daily. Background was counted for 800 minutes duration and samples for 100 minutes. Review of results indicated that samples were counted for 100, 200, or 400 minutes, perhaps in attempts to achieve lower detectability.

Data were accumulated in 255 storage positions. Total events in a 236-Pu band and in a 239-Pu band were determined. The activity in the counting sample was determined from the following equation:

$$\text{pCi/sample} = \frac{(\text{net cpm in } 239\text{-Pu band}) \times (\text{dpm } 236\text{-Pu added})}{(\text{net cpm in } 236\text{-Pu band}) \times 2.22}$$

$$\text{where net cpm in 239 - Pu band} = \left[\frac{\text{gross cts in 239 - Pu band}}{\text{gross ctg time}} - \frac{\text{bkg cts in 239 - Pu band}}{\text{bkg ctg time}} \right]$$

$$\text{and net cpm in 236 - Pu band} = \left[\frac{\text{gross cts in 236 - Pu band}}{\text{gross ctg time}} - \frac{\text{bkg cts in 236 - Pu band}}{\text{bkg ctg time}} \right]$$

dpm 236-Pu = activity of 236-Pu spike added to sample corrected for decay to date of count.

Corrections for sample volume to convert the result into the amount excreted in a day (24 hours) were also applied before calculating the body burden. Errors were estimated based on counting statistics and minimum detectable activity levels established and applied. Odland reported that the minimum detectable activity (MDA) as used in the program was defined as the sample activity associated with a counting error at the 95% confidence level equal to 0.95 times the sample activity (Odland 1968a). That means that any sample whose estimated error exceeded 95% of the sample activity was reported as no detectable activity (NDA).

During review of the records, assessments of the procedures indicated that the estimated errors on alpha spectrometry samples were calculated and reported at the 68% confidence level.

E.2.4. Data Preparation

E.2.4.1. Description of Changes

Adjustments to the data provided were made to fill data gaps and to overcome inconsistencies for exposure date, sample date, sample duration, and urinary excretion rate and its estimated error. Other inconsistencies observed in the data were also corrected to the extent possible.

E.2.4.1.1 Exposure Date

The exposure date was determined from the midpoint of the time an individual spent on station. Exposure date entries on the forms included all of the following: a single date, a date range, an arrival date, a month and year, a year only and a few others. Missing start dates were developed from reasonable estimates based on other recorded information, such as arrival date. Exposure end dates were derived similarly, or from recorded sample collection dates. Both of these modifications are discussed further in Appendix B.

E.2.4.1.2 Sample Date

Missing Sample Date entries for the 445 samples identified (Table E-1) were estimated with an approach that used data on sample receipt at USAF RHL and assigned laboratory sample numbers (See Appendix B). The approach recognized that receipt of samples at USAF RHL, the sample number sequence assigned, and collection date were related. Derived Sample Date information was then entered into a master data set along with the other data for each urine sample.

E.2.4.1.3 Sample Duration

Actual sample duration was documented in a very small fraction (42 samples) of the samples received. Fortunately, basic sample volume data provide the basis for making any corrections needed. As discussed above, this project elected to treat recorded sample volumes as representing 24-hour outputs unless the data forms specifically designated the samples as 12-hour samples. For those, the results were adjusted to the currently accepted nominal daily urine output (1400 mL) for Reference Man. Those adjustments were performed in the intake assessment process.

E.2.4.1.4 Other Parameters

Analytical results for daily urinary excretion and the estimated error were transcribed as entered on the hardcopy forms. However, in the case of samples reported as No Detectable Activity, the data forms were reviewed for the presence of other calculations of a numerical result and error. When found, these calculated results were used in the analysis, even when the error value exceeded the result. This procedure applied primarily when the results of multiple samples were available, as was the case for many of the High 26 Cases Group. In these cases, although the errors were large, they nevertheless provided order of magnitude information about the levels present and were useful comparisons to other values.

E.2.4.1.5 Other Inconsistencies

Other inconsistencies in the data set were also identified and corrected where possible. Although these did not affect the actual intake and dose assessments, they do affect identifying information. These reviews discovered inconsistencies in names, SSNs caused by typographical errors or keyboarding errors, errors in analysis type, inconsistent base names, and others.

E.2.5. Grouping of Cases

The majority of available records contained results from the gross alpha method on samples collected on site. Typically, one record was available for each individual and initial results indicated that intakes and doses estimated using the records would be unusually high. On the other hand, the individual records for the High 26 Cases Group generally contained several results with most from the preferred alpha spectrometry method. In between, the 115 individuals with results from alpha spectrometry follow-up analyses had more limited data. An overall approach to evaluating the cases was clearly needed.

E.2.5.1. Review of Data Available

Estimating intake from urine bioassay depends on reasonably accurate urinary excretion values that follow the expected pattern for the assumed type of exposure and Class (Type) of the contaminant. The data should be as free of artifacts as possible. The varied quality of the records cast doubt about whether reasonable estimates could be developed for all individuals. Records for the High 26 Cases Group offered the best opportunity. On the other hand, most of the records for samples collected on site raised serious questions about estimates derived from them. Some of those issues arose from initial attempts to use the High 26 records as the model for the other cases. As mentioned earlier, those studies indicated that including the results from gross alpha analyses obtained from samples collected on site produced intake estimates and doses that seemed unreasonably high. Furthermore, the pattern of results for samples collected during the resampling phase often did not follow the pattern expected for Class Y (Type S) plutonium.

Figure E-1 contains results and expected urinary excretion for one case that illustrates the difficulty. The figure shows the actual samples as data points and calculated curves for the actual CINDY fit (intake = 58,000 pCi) and reasonable "eye-ball" fits of 23,200 and 870,000 pCi. The first two samples were taken at 3 days and 59 days after the incident. This subject was one of the first responders to arrive. In addition, the last two samples, taken at 472 and 547 days after the incident were reported as NDA. They are plotted as 0.003 pCi/day for graphing purposes. The "final" fitted result was obtained by excluding the first two samples from consideration. Even for this case, the upper and lower rough estimates differ from the fitted curve by a factor of two, with associated CEDEs of approximately 10 to 270 rem (0.1 to 2.7 Sv).

The apparent difficulties with fitting urinary excretion models to the actual data required further investigation. Peer reviewers of a draft version of this report suggested that all of the data should be considered to assess whether some other form of plutonium behavior was being observed rather than the assumption of inhalation exposure to very insoluble Class Y (Type S) material. These suggestions were evaluated for this revision by considering the validity of the Class Y (Type S) assumption, by considering other routes of entry (e.g. ingestion), and by assessing the effect of the alternate approaches on all data for the High 26 group.

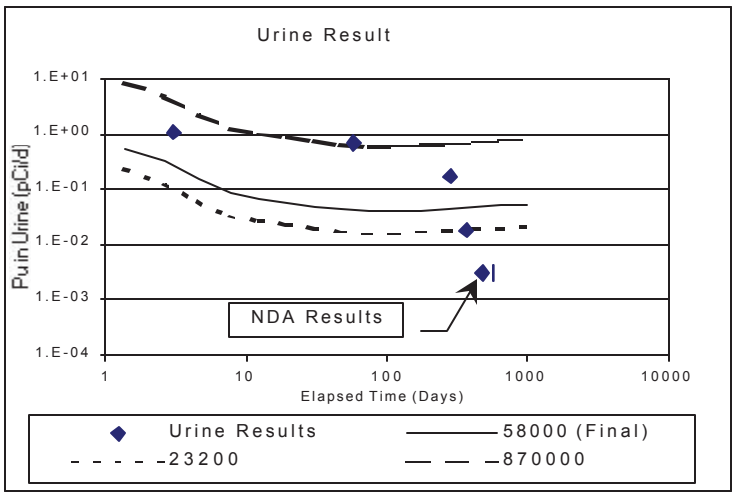

Figure E- 1. Example urinary excretion.

Regarding the conclusions about material form, numerous investigators report that plutonium produced under the conditions of the Palomares accident (i.e. explosion, and fire produce oxides of plutonium under high temperature) tend to be very insoluble (Church 2000). Furthermore, investigations at Palomares itself indicate that the material present on site consists primarily of 87% Type S (Class Y) and 13% Type M (Class W) material (Stradling 1993). Although those findings represent studies conducted at some time after the accident, it seems reasonable to expect that the solubility of plutonium would not decrease over time. Consequently, the assumption that Type S (Class Y) plutonium was the principal form present during response activities seems very reasonable.

Investigations of the behavior of the set of urine results with expected behavior involved qualitative, graphical comparisons of the dataset with the expected curve shapes for urinary excretion from inhalation of Type M (Class W) and Type S (Class Y) plutonium alone and in combination, and from ingestion of soluble and insoluble plutonium. Figure E-2 compares the urine results from the initial sampling and the re-sampling phases of the High 26 Group with the urinary excretion patterns for inhalations of Type M (Class W) plutonium, Type S (Class Y) plutonium and two combinations (one of equal amounts of Type M and Type S, and the other of 3 parts Type S and 1 part Type M). The excretion curves do not represent fits to the data. Rather they have been scaled by the amount of plutonium intake required to place them on the chart. As a matter of fact the assumed intakes are 15,000 pCi Type M, 15,000 pCi Type S, 15,000 pCi Type M plus 15,000 pCi Type S, and 5,000 pCi Type M plus 15,000 pCi Type S, respectively. The plutonium amounts are not critical for this comparison because the shapes of the curves provide the substantial observations about the behaviors.

The urine results shown in Figure E-2 seem to decrease steadily, almost monotonically, on this logarithmic presentation. However, each of the urinary excretion curves declines very rapidly at first, but then declines much more slowly. Actually, for the plutonium forms involved, there is a slight increase beginning at around 200 days that represents the continuing release of plutonium retained in the lungs combined with additional plutonium being remobilized from other organs. Most importantly, the expected excretion continues at an ever more slowly decreasing rate at times beyond 500 days after the initial rapid decrease. There are obvious differences between the data and the expected excretion.

Figure E-3, illustrates the behavior of ingested plutonium for comparison with the urine results. Again as for the inhalation case, the excretion curves differ substantially from the results. A level that seemingly predicts the excretion soon after exposure tends to over estimate excretion later. Conversely, reasonable estimates at longer times generate significant differences at the earlier times.

These discussions raise serious concerns about estimates of intake that would be derived from the data. One interpretation suggests that other, or better, models should be tried. On the other hand, the data themselves may be contribute to the difficulties; especially those from samples collected on site or soon after departing Palomares. Alternately, improvements in laboratory procedures may have contributed to the discrepancies. Conversations with USAF RHL personnel who devised and directed the urine analysis program indicated that the alpha spectrometry methods for ^{239}Pu were very much at the developmental stage for most of 1966 (Taschner 1999). Additional first-hand experience by one of this report's authors (a former director of radioanalysis at the USAF RHL from 1969 to 1976) confirms those observations as well as the difficulties in measuring such low concentrations of plutonium radioactivity (Case 2001).

Consequently methods, used in this project, excluded data from the on-site samples and attributed more significance to samples collected at later dates for the High 26 Group.

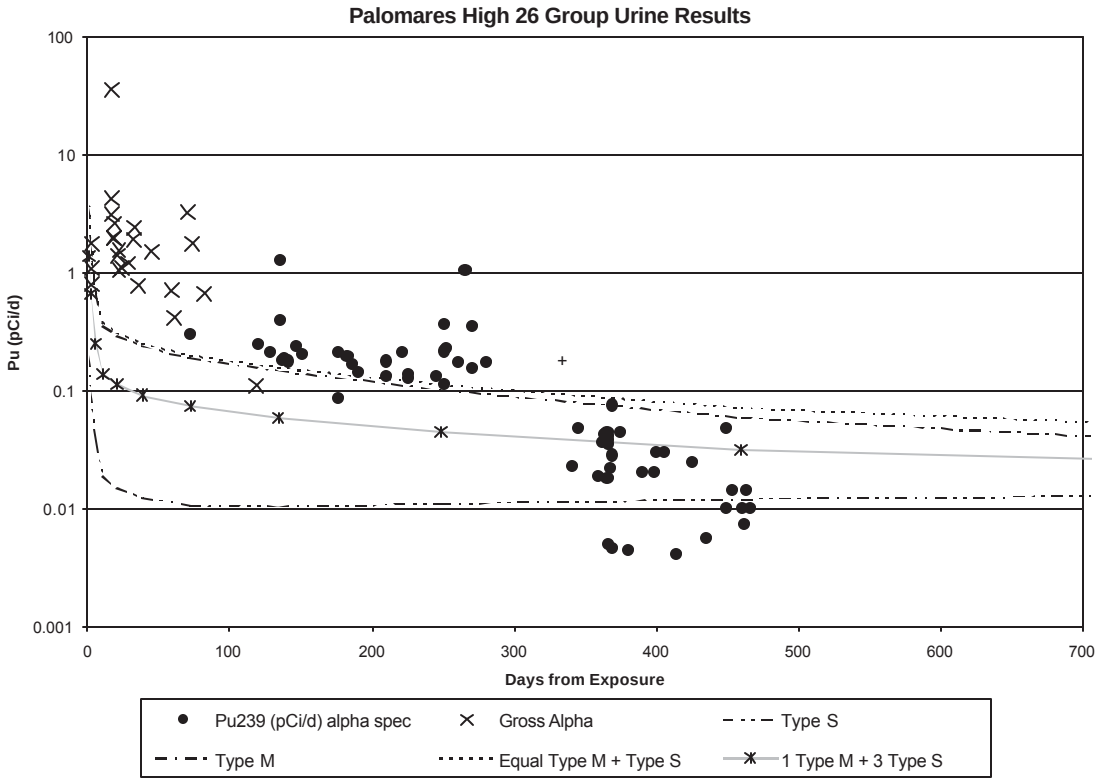

Figure E- 2. Comparison of High 26 Group urine results with excretion expected from inhalation of plutonium.

The remaining results generally fell into two categories: those with the results of some resampling; and those with one sample and often very high results. Urinary excretion results for the latter case ranged from 0.0 pCi/day to 237 pCi/day with corresponding committed effective dose equivalent of up to 6,000 rem (60 Sv) from an estimated intake of 20,000,000 pCi. If real, that intake would have produced a dose equivalent to the lung of almost 5,000 rem (50 Sv) and an effective dose equivalent of about 560 rem (5.6 Sv) in the first year alone. Both of those are 100 times higher than the applicable regulatory limits for non-stochastic (prompt) and stochastic (delayed) effects and would have produced deterministic (non-stochastic) effects. Clearly that case is extreme and alternative approaches to processing were needed.

E.2.5.2. Selection of Contamination Cutoff

Careful review of the group of data indicated that processing all of the cases would produce unrealistic estimates that would be based on potentially contaminated samples. Contamination of samples collected at the accident site continued to impact the evaluation as it did at the time of the accident. However, review of those data also indicated a substantial number of cases that had urinary results that were essentially below the detection limit or were quite low.

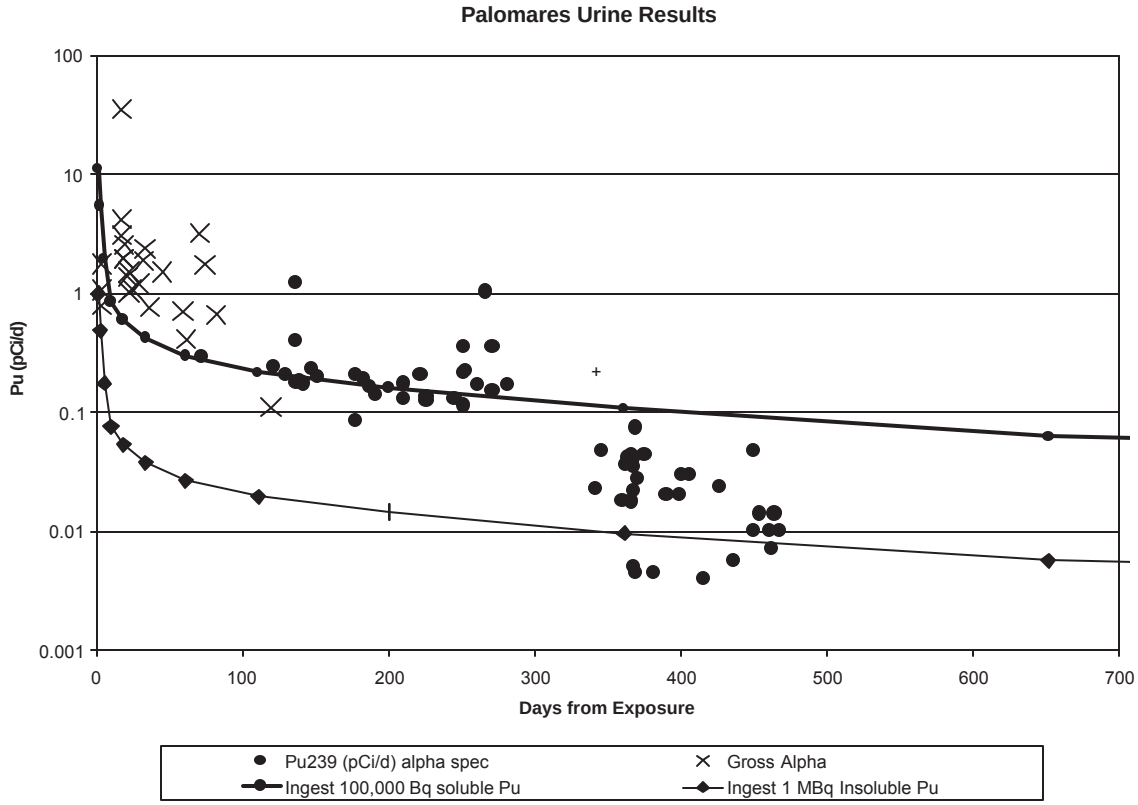

Figure E- 3. Comparison of High 26 Group Urine Results with expected excretion following ingestion.

After consultation with AFMOA, the data were reviewed again to determine whether a reasonable lower cutoff could be determined. Selection of a reasonable cutoff allows the use of professional judgement to eliminate questionable data, while at the same time, allowing reasonable estimates from apparently "uncontaminated" samples. This task was approached by evaluating selected records to calculate a Lower Limit of Detection according to current practice, and to research LLDs in use by other laboratories for similar assessments. The effort extracted sample and background counting information for 39 gross alpha samples and 3 alpha spectrometry samples. The mean and standard deviation of those were of 0.1 ± 0.1 pCi/day for gross alpha and 0.015 ± 0.003 pCi/day for alpha spectrometry. These were compared with the reported limits achieved by the combined U.S and Spanish effort to assess intakes and doses in the local population. A detection limit of 0.74 mBq/d (0.02 pCi/d) was in use from 1966 to 1985 (Iranzo 1988). That limit is essentially the same as the result obtained from Air Force data. Furthermore, the value of 0.1 pCi/day for gross alpha also seemed like a reasonable cutoff. Consequently, that value was selected as a cutoff limit. Cases with urinary excretion measurements below the level were categorized as the Contamination Cutoff Cases Group. Those with measurements above the level were categorized as the Remaining Cases Group and were not processed further in this project.

E.3 DOSE CALCULATION

E.3.1. Exposure Scenario

The type of exposure (acute or chronic; inhalation, ingestion, direct) must be known or assumed to perform a meaningful estimate of an intake of radioactive material and its associated dose equivalent. One or more of the common routes of entry (inhalation, ingestion, or direct) generally apply. Examinations of the activities that may have caused the exposure provide the clues to determining the type and route of the exposure.

As discussed above, the response to the Palomares nuclear accident involved hundreds of personnel working toward the common purpose of recovering vital materials, protecting themselves and the local populace, and restoration of the accident scene to useable and safe conditions. The accident itself released plutonium during explosions and fires that followed the impact of two of the nuclear weapons with the ground. The plutonium was released primarily as airborne dust and as residues from fire, that contaminated the ground. Since the fires essentially were out long before serious response efforts started, the main source of exposure arose from activities such as vehicle movement, handling debris during recovery, plowing fields to mix the contaminant into the soil, and vehicle movement. Persistent winds also contributed to the resuspension of contaminated soils from the ground or contaminated dusts from the surfaces of accident debris, local buildings, or agricultural crops.

Ingestion by hand to mouth transfer is a second possible route of entry. However, that route is very inefficient. Furthermore, the fraction of plutonium that enters the bloodstream from the intestines is very small (0.00001 for Type S). For reasons discussed in Sections 3.1.1.2 and 4.4.1 above, the ingestion route is not considered further.

The type of exposure was assumed to be a single acute exposure. This assumption accommodates the long time for removal of plutonium oxides from the human body. The response activity occurred from January 18, 1966 until April 3, 1966 when activities were moved from Camp Wilson to another location. Personnel on site reached a maximum in late January; tapered off during February, and then increased slightly in mid-March during the packaging of contaminated debris, soil and other wastes for disposal. Most departed the site by late March 1966. The nominal length of assignment was about two weeks. However, records indicate that some personnel stayed much longer.

E.3.2. Parameters Used in Models

Two computer methods, CINDY and LUDEP, were selected to calculate estimates of the ^{239}Pu[1] intakes and doses. CINDY applies the system described in ICRP-30 while LUDEP uses the respiratory tract model of ICRP-66 and the organ/tissue weighting factors of ICRP-60. CINDY served as the primary method and LUDEP provided alternative estimates for comparison. Both programs require selection of input parameters that control the various factors of the intake (respiratory tract), biokinetic and excretion models used in the analysis. Table E-2 contains the parameters selected for the CINDY runs. The parameters chosen represent the default values for

[1] The isotope, ^{239}Pu, is discussed as the primary isotope of interest. Commonly, ^{240}Pu that is also present in weapons material cannot be distinguished from ^{239}Pu by the counting techniques used. However, no distinction is made for this possible presence of ^{240}Pu.

an acute inhalation exposure of Class Y ^{239}Pu obtained from ICRP-30 or other recognized appropriate sources as described in the CINDY Users Guide (PNL 1992). In addition, urine sample collection times were assumed to represent a 24-hour collection unless specifically stated otherwise.

CINDY calculated the cases in a two-step process: the intake assessment mode to estimate the intake from the urine bioassay measurements, followed by the dose assessment mode to calculate the 50-year committed dose equivalent for each organ, and the 50-year committed effective dose equivalent. For some cases, CINDY was also run in the bioassay projection mode to generate a plutonium excretion curve for plotting and further analysis. Figure E-2 above represents such a plot. In addition, CINDY was run in the calendar year dose assessment mode to calculate the annual dose equivalent to specific organs for comparison with the non-stochastic limit.

For LUDEP, a similar process was used to setup the required parameters. LUDEP bases its calculations on an estimate of the intake type and intake value. Intake is estimated for a unit intake first, using a selected excretion model such as the Jones model. Then, the derived excretion model curve is fit to the measurement data to generate an estimate of the intake. Finally, the intake is used to estimate the organ dose equivalents and the committed effective dose equivalent for the exposure type (acute, inhalation), activity parameters (worker, standard worker), and model parameters. Table E-3 contains the parameters used for estimating intakes and doses for LUDEP cases.

All cases were run with standard ICRP default values for the deposition and particle transport factors except particle density, which was set at 10 g/cm^3, which is the density of PuO_2 rather than a density representative of dust. The compartment numbers for the clearance rate constant values and the deposition fractions in Table E-3 refer to Figure 5 of the main report. The compartment rate constants are the half-times (in $days^{-1}$) that material moves from the "From" compartment to the "To" compartment.

E.4 RESULTS

E.4.1. High 26 Cases Group

The High 26 Cases Group represents the collected measurement data from 26 responders who were identified for follow-up after the initial phase of sampling in 1966. The evaluation of the cases is presented with discussions of their urine bioassay measurement characteristics, the approach to performing the estimates, and a discussion of the results.

E.4.1.1. Urine Bioassay Measurement Characteristics

The High 26 Cases Group provided 127 urine samples during their on-site and resampling activities. Those 127 samples produced 25 measurements of gross alpha activity and 102 measurements of ^{239}Pu from alpha spectrometry. The 102 samples from alpha spectrometry were distributed among the 26 people as shown in Table -4. The gross alpha method reported 24 results above the minimum detectable and one result as no detectable activity.

Table E- 2. Parameters used in CINDY runs.

Parameter	Value
Subject identification	
Name	Specific to individual
ID	Set to dummy value of 1234567890
SSN	Specific to individual or 000-00-0000 if not available
Date of birth	Not available: set to dummy value of 01/01/1945
Sex:	Male (with few exceptions for obvious female names)
Intake information	
Intake exposure rate	Acute
Intake mode	Inhalation
Begin date	Specific to estimated acute exposure date for individual
Begin time	Left at default value of 00:00
Particle size (microns)	1
Facility	Palomares
Employer	U.S. Air Force
Edit/input bioassay data	
To exclude set non-blank	G or x entered if individual had a gross alpha result that was being excluded from the current model run
Bioassay type	u entered for urine
Bioassay radionuclide	Pu239
Sample end date	Sample date, specific to individual's sample
Sample end time	Left at default value of 00:00
Excretion period (hr)	24 unless dose card specifies otherwise (regardless of sample volume)
Measured value	Sample result (for units of pCi/sample) specific to individual's sample
Inverse of weighting factor	Variance of sample error (not used in methodology reported in final output)
Unit numerator	pCi
Units are per ...	[S] for sample
Sample size	Sample volume (for units of mL) specific to individual's sample
Sample size units	mL
Reference volumes	
Urine-male (mL)	1400 (not used in modeling--overridden by entries made to "excretion period" parameter)
Feces-male (g)	135 (not used in modeling—no bioassays of this type)
Intake Assessment Mode	
Radionuclides of concern	Pu239; Working units = pCi
Intake composition	Fraction inhaled = 1 ICRP-30 Class D = 0% ICRP-30 Class W = 0% ICRP-30 Class Y = 100%
Change default parameters	Radionuclide daughters: Consider? yes Select radiological units: pCi Error tolerance for integration: .0000001
Select models	Pu239: Jones excretion model

Table E- 2. Parameters used in CINDY runs.

Parameter	Value	
Dose Assessment Mode (specified period)		
Radionuclides of concern	Pu239 Working units = pCi	
Intake estimate	Quantity inhaled: in pCi, specific to individual based on results of intake assessment mode run ICRP-30 Class D = 0% ICRP-30 Class W = 0% ICRP-30 Class Y = 100%	
Change default parameters	Dose reporting times = 1 report time Report time in years = 50 Select radiological units: pCi Error tolerance for integration: .0000001	
Select models	Pu239: Jones excretion model	
Jones Excretion Model Parameters		
Compartment	**Fractional Rates (1/d)**	**Transfer rate constant (1/d)**
1	4.75×10^{-3}	0.558
2	2.39×10^{-4}	4.42×10^{-2}
3	8.55×10^{-5}	3.60×10^{-3}
4	1.42×10^{-5}	2.84×10^{-5}
Systemic Model – Pu		
Bone	Fraction from transfer compartment: 0.45 Transfer compartment clearance half-time (d) : 0.25 Organ clearance half-time (d): 18,200 Fraction reaching urine: 0.5 Fraction Reaching feces: 0.5	
Liver	Fraction from transfer compartment: 0.45 Transfer compartment clearance half-time (d): 0.25 Organ clearance half-time (d): 7,300 Fraction reaching urine: 0.5 Fraction Reaching feces: 0.5	
Testes	Fraction from transfer compartment: 0.00035 Transfer compartment clearance half-time (d): 0.25 Organ clearance half-time (d): 3,650,000 Fraction reaching urine: 0.5 Fraction Reaching feces: 0.5	
Pu f_1 values		
Inhalation	Class D: 0.001 Class W: 0.001 Class Y: 0.00001	
Ingestion	Soluble: 0.001 Insoluble: 0.00001	

Table E- 3. LUDEP Input Parameters.

Input parameters		
Intake regime	Exposure	Occupational
	Subject	Standard worker
	Intake	Acute, inhalation, 1 Bq (used to generate excretion curve)
Time	50 years	
Deposition	Exposure	Occupational
	Subject	Standard worker
	AMAD (:m)	1
	Advanced mode	All defaults except density = 10 g/cc ICRP Defaults 1. SUBJECT: Adult Male 2. ACTIVITY: Light Exercise 3. TYPE: Nose Breather 4. DISPERSION: polydisperse Physiological Parameters a) Functional Residual Capacity: 3301 cc b) Extra-thoracic Dead Space: 50 cc c) Bronchial Dead Space: 49 cc d) Bronchiolar Dead Space: 47 cc e) Height: 176 cm f) Tracheal Diameter: 1.650 cm g) First Bronchiolar Diameter: 0.165 cm Activity Related Parameters h) Ventilation Rate: 1.50 cu.m/h i) Respiratory Frequency: 20.0 /min j) Tidal Volume: 1250 cc k) Volumetric Flow Rate: 833 cc/s l) Fraction breathed through nose: 1.000 Aerosol Size Parameters m) AMAD: 1.0000 μm (changed from default of 4) n) AMTD: 0.3407 μm o) Φg: 2.43 p) Den: 10.00 g/cc (changed from default of 3) w) SF: 1.50 Deposition q. ET1 17.54 % r. ET2 22.59 % s. BB 1.38 %* t. bb 2.22 %* u. AI 13.04 % Total = 56.78% v. Fs* (BB%) = 49.76, (bb%) = 49.98%

Table E- 3. LUDEP Input Parameters.

Particle transport (See Figure 5)		
	Compartment From – To	Rate Constant (1/d)
	1 to 4	0.02
	2 to 4	0.001
	3 to 4	0.0001
	3 to 10	0.000020
	4 to 7	2.0
	5 to 7	0.03
	6 to 10	0.01
	7 to 11	10.0
	8 to 11	0.03
	9 to 10	0.01
	11 to GI	100.0
	12 to 13	0.001
	14 Out	1.0
	Compartment	Deposition Fraction
	ET_{seq}/ET_2	0.00050
	BB_{seq}/BB	0.00700
	BB_2/BB	=Fs1
	Bb_{seq}/bb	0.00700
	Bb_2/bb	=Fs2
	AI_2/AI	0.6000
	AI_3/AI	0.1000
Absorption	Selected S for default values	
Radio-nuclides	ICRP-38 database	Pu239
Biokinetic model	ICRP-30	Part 4: Pu(Y)M.mod (for Pu, class Y, male) Organs = liver, whole skeleton, testes (default for Pu239) Bone type = surface seeker (default for Pu239) Blood half life = 0.25 d (default for Pu23)
Calculations		
Excretion/ Retention (the results of this run are then entered as the bioassay function in the intake estimate mode)	Quantity to calculate	urinary excretion rate
	Select ICRP-54 function	Pu/Am (J) (this is the Jones Plutonium Excretion Model)
	Enter own function	Used defaults as follows: A(1) = 4.75E-03 t½= 1.24E+00 d A(2) = 2.39E-04 t½= 1.57E+01 d A(3) = 8.55E-05 t½= 1.82E+02 d A(4) = 1.42E-05 t½= 2.44E+04 d A(5) = 0.00E+00 t½= 0.00E+00 d A(6) to A(10) are zero Retention t½in blood: 1.000E-07
	Period of integration	1 day
	Time	1 day to 730 days
	Number of points	730
	Intervals	Linear

Table E- 3. LUDEP Input Parameters.

Intake estimation	Data filename	*.dat file for individual, showing days elapsed from exposure to sample, sample result in Bq/d, and sample error in Bq/d, as in the following example for an individual with three samples 10 0.005 0.0005 43 0.004 0.0014 78 0.001 0.001
	Assumed errors	errors included in data set
	Modify for DTPA?	no modification
	Bioassay function filename	File from excretion/retention mode run
	Estimate intake	command line, estimated intake appears on screen

Review of the procedures for calculating the radioactivity results and their errors revealed that the reported errors for gross alpha measurements represented the 95% confidence level while the reported errors for alpha spectrometry measurements represented the 68% confidence level. Since the criterion for reporting a result as no detectable activity was based on the 95% confidence limit, alpha spectrometry results may not have followed that convention. Therefore, some alpha spectrometry results may have been reported as positive when the estimated errors did not support that conclusion. Nevertheless, the approach was more likely to report a numerical result, which is preferable to the NDA report. Unfortunately, the numerical values for the laboratory's NDA were not discussed in any of the reports of the sampling and analysis effort reviewed for this project.

Table E- 4. Breakdown of alpha spectrometry samples.

Number of Samples	Number of Submitters
3	5
4	2
5	14
6	3
7	2

The measurement results from alpha spectrometry revealed that actual numerical values and the associated counting errors were calculated even when the sample was reported as NDA. Those results were used in developing these estimates when recorded on the individual data cards. The alpha spectrometry results contained 63 reported values while the remainder were reported as NDA or were not reported, apparently because of a laboratory error. Of the 63 results, 15 were less than their error at the 68% confidence level and 33 results were greater than the 68% level but less than the 95% confidence level. Only 15 results were above the 95% confidence level. This means that for 48 of the 63 reported results, zero was included in the range of possible results.

Reproducibility of the laboratory measurements was also evaluated using samples that were reprocessed. Although limited, five samples were reprocessed primarily to correct low chemical recovery. One of those was processed three times, reporting two numerical results that were less than the 68% confidence level error, and one result as NDA. Of the other four samples, three

showed differences in reported radioactivity of two to three times. The remaining sample was a valid NDA report.

In considering the impact of these apparent analytical difficulties, the levels of radioactivity of these samples (less than 0.1 pCi/d) may produce only a few detectable events during counting periods of 100 to 400 minutes recorded. For those techniques, background counting levels are also very low, usually on the order of one count in thousands of minutes. Although these levels are quite low, they can represent plutonium intakes that require evaluation.

Figure E-4 illustrates the urine results obtained from the High 26 Cases Group. Those results show the variability in measured plutonium values. The expected behavior of urinary excretion from inhalation of Class Y (Type S) ^{239}Pu and an equal mixture of Class W (Type M) and Class Y (Type S) ^{239}Pu are also shown. The results do not correspond to the expected pattern very well at all as was previously discussed in Section 4 of the main report. Consequently, attempts to fit the urinary excretion model to the measurements were expected to be difficult.

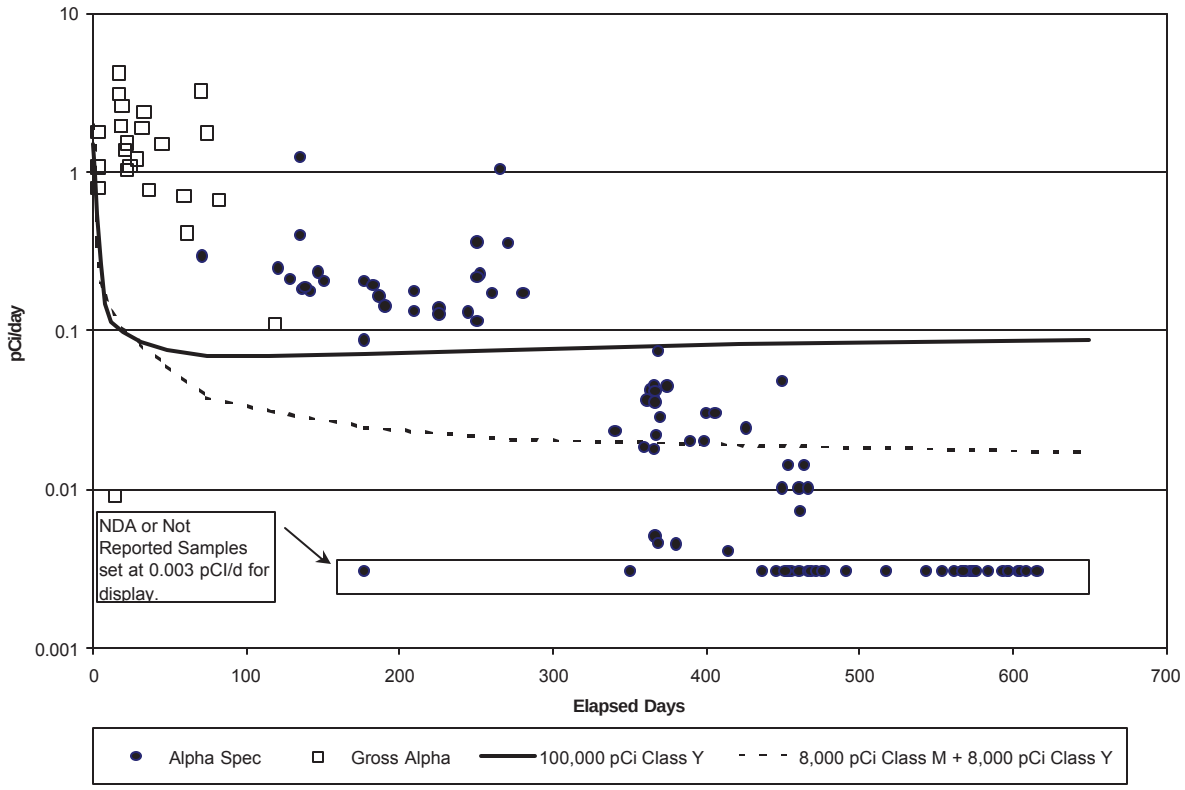

Figure E- 4. High 26 Cases Group urine results.

E.4.1.2. Approach to Estimates

The urine analysis results for the High 26 Cases Group indicated that those cases with several measurements for samples collected over the entire initial and resampling efforts could provide the best data for testing. To do this, several variations on use of the data and setup for the CINDY and LUDEP programs were used. For samples, assumptions were developed for the date

of exposure, the use of gross alpha results and the use of NDA results. For the programs, the main adjustment involved the method for weighting results during intake assessment using CINDY and LUDEP.

E.4.1.2.1 Date of Exposure

The entire High 26 Cases Group arrived during the early phases of the response effort. Some arrived the day following the accident while others arrived somewhat later. All arrived in January 1966. Some remained on site for only a few days or weeks while others remained for the entire deployment. Rather than use the midpoint of the assumed on-site period as the date of exposure for this group, their arrival date at Palomares was selected. This assumption was judged conservative since it would estimate slightly higher intakes because more days would elapse between the assumed exposure and sampling. The effect would be minimal as shown by tests of both CINDY and LUDEP (Section 3.3.1).

E.4.1.2.2 Use of Gross Alpha Measurements

Twenty-two of the 25 gross alpha results (one of the group had no gross alpha results) were from samples collected on dates that represent on-site activities. The gross alpha activity of these samples ranged from NDA to 35 pCi/d. That former result represents a very high urinary level. Tests were run that included and excluded the gross alpha results, including those collected on and off site as separate cases. The results indicated that both CINDY and LUDEP tended to produce better fits for samples with lower values and taken at longer time following the exposure.

E.4.1.2.3 Use of NDA Results

Samples reported as no detectable activity do not produce a numerical result. However, these samples indicate that their radioactivity content is near or below the level that can be measured with confidence. That is, at those levels, the analysis indicates that the radioactivity may, or may not, be present. Since many of the results obtained during the resample period were reported as NDA (see Figure E-4 above), a method was needed to make them available to CINDY and LUDEP. The available choices included careful review of the data records for entries representing a calculation of a numerical quantity for the sample that was reported as NDA. Figure B-3, Appendix B illustrates such a case. Those were used whenever possible. For the remaining samples, options included recalculation from the recorded counting data, arbitrarily setting the value to zero, or arbitrarily setting the value to the lower limit of detection (0.003 pCi/day) for alpha spectrometry samples. All of those approaches were used.

E.4.1.2.4 Weighting Factors for Urine Measurements

Section 3.2.1.1 discusses the selection of weighting factors for estimating intakes from bioassay measurements and Section 3.2.2 summarizes some performance tests. Those were confirmed for the High 26 Cases Group data. Selection of the "ratio-of-the-means" method in CINDY and the "errors included in data set" method for LUDEP provided conservative estimates of intake. That

is, the selected methods provided estimates that were balanced between being unreasonably high and artificially low.

E.4.1.3. Results

The methods applied to estimating intakes and doses described above were applied to the 26 individual cases. Some adjustments were necessary to accommodate the specific data qualities for each case. Although intake and committed dose equivalent dose to organs, and committed effective dose equivalent were estimated, they are not adopted as official estimates for any individual because of the difficulties discussed earlier in the report. This section summarizes the overall results and discusses approaches for developing estimates that are more reasonable.

The urine results for the 26 individuals in this group exhibited two common traits that could have substantially affected the intake estimates and doses. These traits were 1) an unexpectedly rapid decrease in urine concentration for follow-up program samples, and significant variation in replicate analyses of individual samples. Figure D-5 illustrates these two traits.

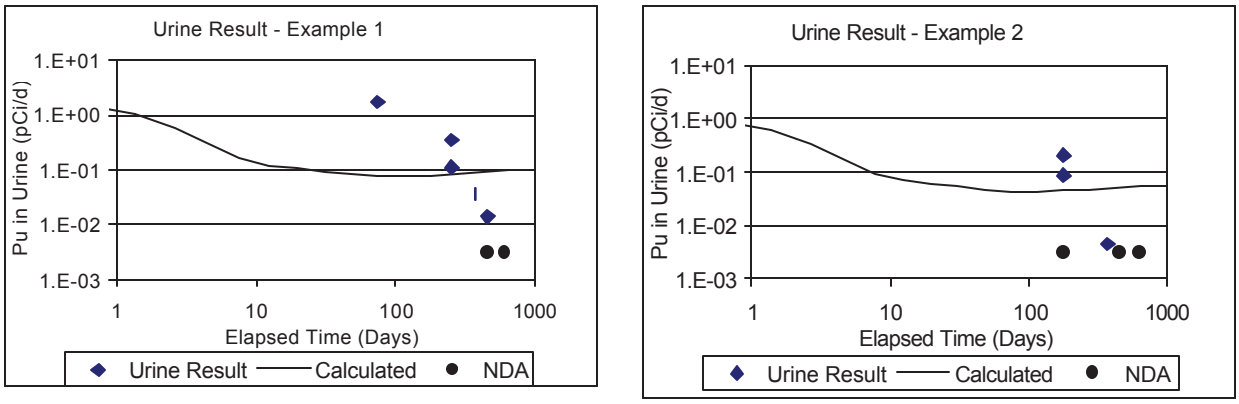

Figure E- 5. Examples of urine result characteristics.

Interestingly, most samples in this group show decreasing urinary excretion, usually reaching the non-detectable level for later samples. Of course, if those latter values are correct, then the estimated intakes and corresponding doses would be much lower than reported in this study. Alternately, the rapid decrease in value could be related to improved laboratory capability.

The variability of replicate measurements was only reported for a few samples. However, if those reported are typical of the analytical performance, then similar variability would be expected for the other samples. Unfortunately, there are no data to support this possibility.

E.4.1.3.1 Intakes and Doses from Urinalysis

For the 26 cases, the preliminary intake estimates varied from 34,000 pCi to 570,00 pCi from CINDY and 19,000 pCi to 2,600,000 pCi from LUDEP with the gross alpha results excluded in all the cases. Estimates of committed effective dose equivalent ranged from 10 rem to 170 rem (0.1 to 1.7 Sv) from CINDY and 1.3 to 180 rem (0.013 to 1.8 Sv) from LUDEP. LUDEP ranged from –83% to +150% of CINDY results. The range of differences between LUDEP results and CINDY results seems reasonable considering the variation in the data and the complexities of the

assessment. In addition to the intakes and CEDE estimates, 50-year committed dose equivalents were calculated for organs using CINDY. Those results are listed in Table E-5 to illustrate the range of estimated values. However, when compared with independent estimates from environmental data and with the results of other exposure cases, these estimates seem unreasonably high.

Table E- 5. High 26 Preliminary Intake, Committed Dose Equivalent and Committed Effective Dose Equivalent Estimates.

Subject	Intake (pCi)	CEDE	Testes	Breast	R Marrow	Lung	Thyroid	Bone Sur	Liver	Other	LL Int.	UL Int.	S Int.
Data Masked	6.8E+04	21	3.0	0.0	16.3	76.9	0.0	212.9	38.4	3.5	0.0	0.0	0.0
Data Masked	8.6E+04	26	3.7	0.0	20.6	97.2	0.0	269.2	48.6	4.5	0.0	0.0	0.0
Data Masked	6.2E+04	19	2.7	0.0	14.8	70.1	0.0	194.1	35.0	3.2	0.0	0.0	0.0
Data Masked	6.3E+04	19	2.7	0.0	15.1	71.2	0.0	197.2	35.6	3.3	0.0	0.0	0.0
Data Masked	5.60E+05	170	24.3	0.0	133.9	633.0	0.0	1753.0	316.5	29.2	0.1	0.0	0.0
Data Masked	6.5E+04	20	2.8	0.0	15.5	73.5	0.0	203.5	36.7	3.4	0.0	0.0	0.0
Data Masked	1.6E+05	49	7.0	0.0	38.3	180.9	0.0	500.9	90.4	8.3	0.0	0.0	0.0
Data Masked	1.1E+05	34	4.8	0.0	26.3	124.3	0.0	344.3	62.2	5.7	0.0	0.0	0.0
Data Masked	4.2E+04	13	1.8	0.0	10.0	47.5	0.0	131.5	23.7	2.2	0.0	0.0	0.0
Data Masked	6.4E+04	20	2.8	0.0	15.3	72.3	0.0	200.3	36.2	3.3	0.0	0.0	0.0
Data Masked	5.5E+04	17	2.4	0.0	13.2	62.2	0.0	172.2	31.1	2.9	0.0	0.0	0.0
Data Masked	4.4E+04	14	1.9	0.0	10.5	49.7	0.0	137.7	24.9	2.3	0.0	0.0	0.0
Data Masked	7.6E+04	23	3.3	0.0	18.2	85.9	0.0	237.9	43.0	4.0	0.0	0.0	0.0
Data Masked	7.2E+04	22	3.1	0.0	17.2	81.4	0.0	225.4	40.7	3.8	0.0	0.0	0.0
Data Masked	1.8E+05	55	7.8	0.0	43.0	203.5	0.0	563.5	101.7	9.4	0.0	0.0	0.0
Data Masked	2.1E+05	65	9.1	0.0	50.2	237.4	0.0	657.4	118.7	11.0	0.0	0.0	0.0
Data Masked	6.6E+04	20	2.9	0.0	15.8	74.6	0.0	206.6	37.3	3.4	0.0	0.0	0.0
Data Masked	6.8E+04	21	3.0	0.0	16.3	76.9	0.0	212.9	38.4	3.5	0.0	0.0	0.0
Data Masked	6.9E+04	21	3.0	0.0	16.5	78.0	0.0	216.0	39.0	3.6	0.0	0.0	0.0
Data Masked	3.4E+04	10	1.5	0.0	8.1	38.4	0.0	106.4	19.2	1.8	0.0	0.0	0.0
Data Masked	1.00E+05	31	4.3	0.0	23.9	113.0	0.0	313.0	56.5	5.2	0.0	0.0	0.0
Data Masked	7.1E+04	22	3.1	0.0	17.0	80.3	0.0	222.3	40.1	3.7	0.0	0.0	0.0
Data Masked	4.4E+04	14	1.9	0.0	10.5	49.7	0.0	137.7	24.9	2.3	0.0	0.0	0.0
Data Masked	5.8E+04	18	2.5	0.0	13.9	65.6	0.0	181.6	32.8	3.0	0.0	0.0	0.0
Data Masked	6.4E+04	20	2.8	0.0	15.3	72.3	0.0	200.3	36.2	3.3	0.0	0.0	0.0
Data Masked	9.9E+04	30	4.3	0.0	23.7	111.9	0.0	309.9	56.0	5.2	0.0	0.0	0.0

Annual dose equivalents to the organs and effective dose equivalent per year are shown in Figure E-6 for an intake of 34,000 pCi; the lowest intake estimated by CINDY. These curves represent the accumulation of dose to the specified organ in each year following exposure. Readers should note that the lung dose dominants for the first few years. According to this estimate, the bone dose then predominates thereafter, reaching a maximum at around 13 years following exposure and then slowly declining. These curves illustrate the need to consider both the delivery of the dose and the 50-year cumulative total when assessing the potential for health effects.

E.4.2. Repeat Analysis Cases Group

Palomares responders were placed in the Repeat Analysis Cases Group if they met one or both of the following conditions:

- They submitted an initial urine sample while on site that was analyzed for gross alpha radioactivity and then reanalyzed by alpha spectrometry for ^{239}Pu; or
- They submitted an initial sample while on site that was analyzed by gross alpha counting and then submitted one or more follow-up samples after returning to their base of assignment for analysis by alpha spectrometry.

In general, the urine measurements for this group were not as robust as those for the High 26 Cases Group and follow-up did not extend beyond an initial resampling attempt. The following sections discuss the urine measurements available for this group, the process of estimating the intakes and dose equivalents, and the results.

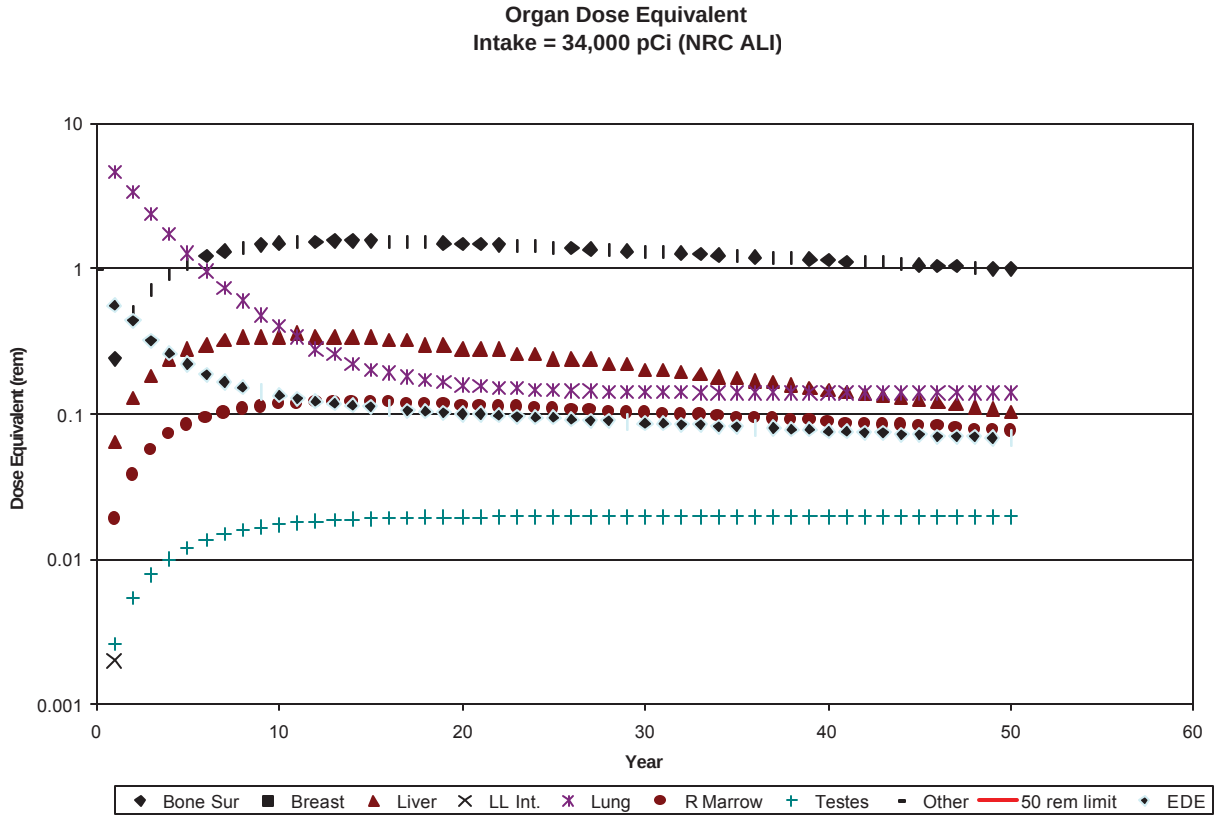

Figure E- 6. Annual organ dose equivalent for 34,000 pCi intake of ^{239}Pu.

E.4.2.1. Urine Bioassay Measurement Characteristics

The Repeat Analysis Cases Group provided 82 urine samples that produced usable results. Other samples submitted did not produce usable results for several reasons. These reasons included laboratory errors during processing and chemical recoveries that were unreported, too low to be measured or below 40%. This project established a minimum requirement for chemical recovery at 40% for alpha spectrometry samples as a reasonable lower limit for credible results. The 82 samples were collected from 54 individuals during January 17, 1966 to June 22, 1966. Figure E-7 illustrates the distribution of sample results obtained for this group. Most of the samples (88) were collected on dates (before April 3, 1966) that represent on-site activity, while 66 samples were collected after that time. The results indicate that the gross alpha and alpha spectrometry measurements are primarily greater than 0.1 pCi/d and that the two types of measurements are interspersed among one another. Gross alpha results, however, tended to have higher values than the alpha spectrometry measurements.

A more detailed review of the data indicated that the samples and analyses were distributed as shown in Table E-6. This distribution seemed to imply that most of the samples were characterized by a gross alpha measurement followed by reanalysis by alpha spectrometry in an attempt to identify the radionuclide responsible for the gross alpha result. In most cases, the alpha spectrometry result was lower than the gross alpha measurement. Twenty-three individuals were characterized by this situation. Unfortunately, resampling was not accomplished for those in this group of 23.

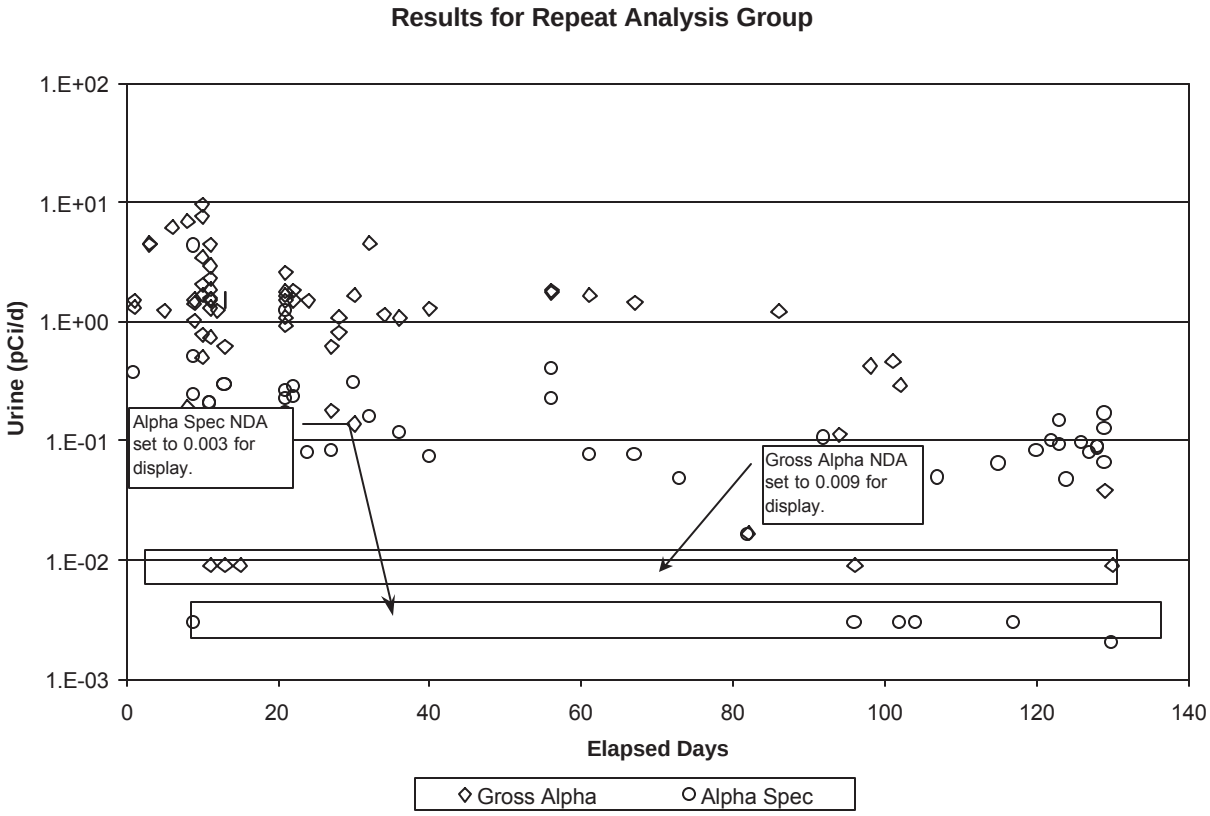

Figure E- 7. Results for Repeat Analysis Cases Group.

The remaining 31 individuals had records characterized by at least two samples with gross alpha measurements on the initial sample and gross alpha or alpha spectrometry or both on the resample. Alpha spectrometry measurements were performed on several initial samples.

Table E- 6. Distribution of Samples for the Repeat Analysis Cases Group.

Number of Samples	Number of Submitters
1	23
2	25
3	3
4	1
5	2

E.4.2.1.1 Date of Exposure

The Repeat Analysis Cases Group had exposure dates that extended over a broader range of dates than the High 26 Cases Group. However, many were among the initial responders who arrived in January 1966. Many stayed on site for one to two weeks, with some up to a month. A few may have remained until the very end of operations. Just as for the High 26 Cases Group, some sample dates were not available in their records and were assigned. Since the time on site seemed shorter and better recorded for this group, the exposure date was assumed as the midpoint of the time at Camp Wilson.

E.4.2.1.2 Use of Measurements

Many gross alpha results for resamples were not reported at all. Therefore, the approach to calculating the estimated intake assumed the following.

- Gross alpha results for samples collected on site were excluded from the analysis.
- Gross alpha results reported as NDA were included with an assumed value of 0.009 pCi/d.
- Alpha spectrometry results reported as NDA were reviewed and numerical values included if found on data cards.
- Some alpha spectrometry results that did not fit the expected urinary excretion pattern were excluded even if the sample was not collected on site.

E.4.2.1.3 Weighting Factors for Urine Measurements

Section 3.2.1.1 discusses the selection of weighting factors for estimating intakes from bioassay measurements and Section 3.2.2 summarizes some performance tests. Those were confirmed for the High 26 Cases Group data and applied to the Repeat Analysis Cases Group.

E.4.2.2. Results

The methods used for estimating intakes and doses for the High 26 Cases Group were applied to the Repeat Analysis Cases Group. Some adjustments were necessary to accommodate the specific data qualities for each case. The results are anonymously listed in Table E-7. This section summarizes the overall results and discusses approaches for developing estimates that are more reasonable.

E.4.2.2.1 Intakes and Doses

For the 54 cases, the estimated intakes varied from 2,900 pCi to 1,300,000 pCi from CINDY and 11,900 pCi to 5,240,000 pCi from LUDEP with the gross alpha results excluded in all the cases. Estimates of committed effective dose equivalent ranged from 0.9 rem to 400 rem (0.009 to 4.0 Sv) from CINDY and 0.8 to 367 rem (0.008 to 3.67 Sv) from LUDEP. LUDEP results ranged from –238% to +94% of CINDY results. In addition to the intakes and ŒDE estimates, annual dose equivalents and committed dose equivalents were calculated for organs using both CINDY and LUDEP.

Table E- 7. Repeat Analysis Group Preliminary Intake, Committed Dose Equivalent, and Committed Effective Dose Equivalent Estimates.

Name	Intake (pCi)	CEDE	Testes	Breast	R Marrow	Lung	Thyroid	Bone Sur	Liver	Other	LL Int.	UL Int.	S Int.
Data Masked	1.00E+05	31	4.3	0.0	23.9	113.0	0.0	313.0	56.5	5.2	0.0	0.0	0.0
Data Masked	1.70E+05	54	7.4	0.0	40.7	192.2	0.0	532.2	96.1	8.9	0.0	0.0	0.0
Data Masked	4.40E+03	1.4	0.2	0.0	1.1	5.0	0.0	13.8	2.5	0.2	0.0	0.0	0.0
Data Masked	6.90E+04	21	3.0	0.0	16.5	78.0	0.0	216.0	39.0	3.6	0.0	0.0	0.0
Data Masked	2.30E+04	7.1	1.0	0.0	5.5	26.0	0.0	72.0	13.0	1.2	0.0	0.0	0.0
Data Masked	1.40E+05	43	6.1	0.0	33.5	158.3	0.0	438.3	79.1	7.3	0.0	0.0	0.0
Data Masked	9.40E+05	290	40.9	0.0	224.8	1062.6	0.0	2942.6	531.3	49.0	0.1	0.0	0.0
Data Masked	1.90E+05	58	8.3	0.0	45.4	214.8	0.0	594.8	107.4	9.9	0.0	0.0	0.0
Data Masked	1.10E+05	34	4.8	0.0	26.3	124.3	0.0	344.3	62.2	5.7	0.0	0.0	0.0
Data Masked	4.30E+03	1.3	0.2	0.0	1.0	4.9	0.0	13.5	2.4	0.2	0.0	0.0	0.0
Data Masked	3.10E+05	95	13.5	0.0	74.1	350.4	0.0	970.4	175.2	16.2	0.0	0.0	0.0
Data Masked	2.00E+05	61	8.7	0.0	47.8	226.1	0.0	626.1	113.0	10.4	0.0	0.0	0.0
Data Masked	1.50E+05	46	6.5	0.0	35.9	169.6	0.0	469.6	84.8	7.8	0.0	0.0	0.0
Data Masked	3.90E+05	120	17.0	0.0	93.3	440.9	0.0	1220.9	220.4	20.3	0.0	0.0	0.0
Data Masked	3.60E+05	110	15.7	0.0	86.1	407.0	0.0	1127.0	203.5	18.8	0.0	0.0	0.0
Data Masked	2.60E+04	8	1.1	0.0	6.2	29.4	0.0	81.4	14.7	1.4	0.0	0.0	0.0
Data Masked	4.40E+03	1.4	0.2	0.0	1.1	5.0	0.0	13.8	2.5	0.2	0.0	0.0	0.0
Data Masked	1.90E+05	58	8.3	0.0	45.4	214.8	0.0	594.8	107.4	9.9	0.0	0.0	0.0
Data Masked	5.50E+05	170	23.9	0.0	131.5	621.7	0.0	1721.7	310.9	28.7	0.1	0.0	0.0
Data Masked	2.90E+03	0.89	0.1	0.0	0.7	3.3	0.0	9.1	1.6	0.2	0.0	0.0	0.0
Data Masked	1.20E+05	37	5.2	0.0	28.7	135.7	0.0	375.7	67.8	6.3	0.0	0.0	0.0
Data Masked	4.40E+03	1.4	0.2	0.0	1.1	5.0	0.0	13.8	2.5	0.2	0.0	0.0	0.0
Data Masked	1.30E+06	400	56.5	0.0	310.9	1469.6	0.0	4069.6	734.8	67.8	0.1	0.0	0.0
Data Masked	9.40E+04	29	4.1	0.0	22.5	106.3	0.0	294.3	53.1	4.9	0.0	0.0	0.0
Data Masked	4.70E+03	1.4	0.2	0.0	1.1	5.3	0.0	14.7	2.7	0.2	0.0	0.0	0.0
Data Masked	1.80E+05	55	7.8	0.0	43.0	203.5	0.0	563.5	101.7	9.4	0.0	0.0	0.0
Data Masked	4.00E+05	120	17.4	0.0	95.7	452.2	0.0	1252.2	226.1	20.9	0.0	0.0	0.0
Data Masked	4.90E+04	15	2.1	0.0	11.7	55.4	0.0	153.4	27.7	2.6	0.0	0.0	0.0
Data Masked	3.20E+04	9.8	1.4	0.0	7.7	36.2	0.0	100.2	18.1	1.7	0.0	0.0	0.0
Data Masked	9.20E+04	28	4.0	0.0	22.0	104.0	0.0	288.0	52.0	4.8	0.0	0.0	0.0
Data Masked	2.50E+05	77	10.9	0.0	59.8	282.6	0.0	782.6	141.3	13.0	0.0	0.0	0.0
Data Masked	9.30E+04	29	4.0	0.0	22.2	105.1	0.0	291.1	52.6	4.9	0.0	0.0	0.0
Data Masked	1.80E+05	55	7.8	0.0	43.0	203.5	0.0	563.5	101.7	9.4	0.0	0.0	0.0
Data Masked	1.40E+05	43	6.1	0.0	33.5	158.3	0.0	438.3	79.1	7.3	0.0	0.0	0.0
Data Masked	1.30E+05	40	5.7	0.0	31.1	147.0	0.0	407.0	73.5	6.8	0.0	0.0	0.0
Data Masked	2.70E+05	83	11.7	0.0	64.6	305.2	0.0	845.2	152.6	14.1	0.0	0.0	0.0
Data Masked	6.80E+04	21	3.0	0.0	16.3	76.9	0.0	212.9	38.4	3.5	0.0	0.0	0.0
Data Masked	2.10E+05	65	9.1	0.0	50.2	237.4	0.0	657.4	118.7	11.0	0.0	0.0	0.0
Data Masked	7.70E+03	2.4	0.3	0.0	1.8	8.7	0.0	24.1	4.4	0.4	0.0	0.0	0.0
Data Masked	2.40E+05	74	10.4	0.0	57.4	271.3	0.0	751.3	135.7	12.5	0.0	0.0	0.0
Data Masked	2.70E+05	83	11.7	0.0	64.6	305.2	0.0	845.2	152.6	14.1	0.0	0.0	0.0
Data Masked	1.40E+05	43	6.1	0.0	33.5	158.3	0.0	438.3	79.1	7.3	0.0	0.0	0.0
Data Masked	1.10E+05	34	4.8	0.0	26.3	124.3	0.0	344.3	62.2	5.7	0.0	0.0	0.0
Data Masked	2.80E+04	8.6	1.2	0.0	6.7	31.7	0.0	87.7	15.8	1.5	0.0	0.0	0.0
Data Masked	9.50E+04	29	4.1	0.0	22.7	107.4	0.0	297.4	53.7	5.0	0.0	0.0	0.0
Data Masked	3.10E+05	95	13.5	0.0	74.1	350.4	0.0	970.4	175.2	16.2	0.0	0.0	0.0
Data Masked	1.10E+05	34	4.8	0.0	26.3	124.3	0.0	344.3	62.2	5.7	0.0	0.0	0.0
Data Masked	1.90E+05	58	8.3	0.0	45.4	214.8	0.0	594.8	107.4	9.9	0.0	0.0	0.0
Data Masked	1.40E+05	43	6.1	0.0	33.5	158.3	0.0	438.3	79.1	7.3	0.0	0.0	0.0
Data Masked	1.40E+05	43	6.1	0.0	33.5	158.3	0.0	438.3	79.1	7.3	0.0	0.0	0.0
Data Masked	1.20E+05	37	5.2	0.0	28.7	135.7	0.0	375.7	67.8	6.3	0.0	0.0	0.0
Data Masked	1.85E+05	55	8.0	0.0	44.2	209.1	0.0	579.1	104.6	9.7	0.0	0.0	0.0
Data Masked	4.40E+03	1.4	0.2	0.0	1.1	5.0	0.0	13.8	2.5	0.2	0.0	0.0	0.0
Data Masked	4.00E+05	120	17.4	0.0	95.7	452.2	0.0	1252.2	226.1	20.9	0.0	0.0	0.0

E.4.3. Contamination Cutoff Cases Group

The Contamination Cutoff Cases Group of analyses was created to calculate estimated intake and dose equivalent for those whose urine measurement results indicated potentially contaminated samples collected at the accident site but were below a reasonable minimum level that did not represent unusually high exposures. While the data for this group were not found especially robust, this approach allows additional cases to be evaluated. As discussed in Section 4.4.2, a level of 0.1 pCi/d was adopted as reasonable maximum level for cases included in the Contamination Cutoff Cases Group.

E.4.3.1. Urine Bioassay Measurement Characteristics

The Contamination Cutoff Cases Group contained 313 individuals who provided 344 samples. Of the 344 samples, 30 samples were collected on site, had high results and were subsequently reanalyzed. The 314 resamples produced results that were substantially below the values of the initial group of 30 samples. Of the 314 repeat samples, 13 results were produced by alpha spectrometry. Figure E-8 illustrates the distribution of the results with sample collection date. The figure also shows that the majority of samples were collected during the period of on-site activity and were susceptible to sample contamination.

E.4.3.2. Approach to Estimates

The procedures for analysis of the High 26 Cases Group were applied to the Contamination Cutoff Cases Group, except that the intakes and dose equivalents were calculated using only the CINDY program. LUDEP was not used. NDA reports were not encountered in this group.

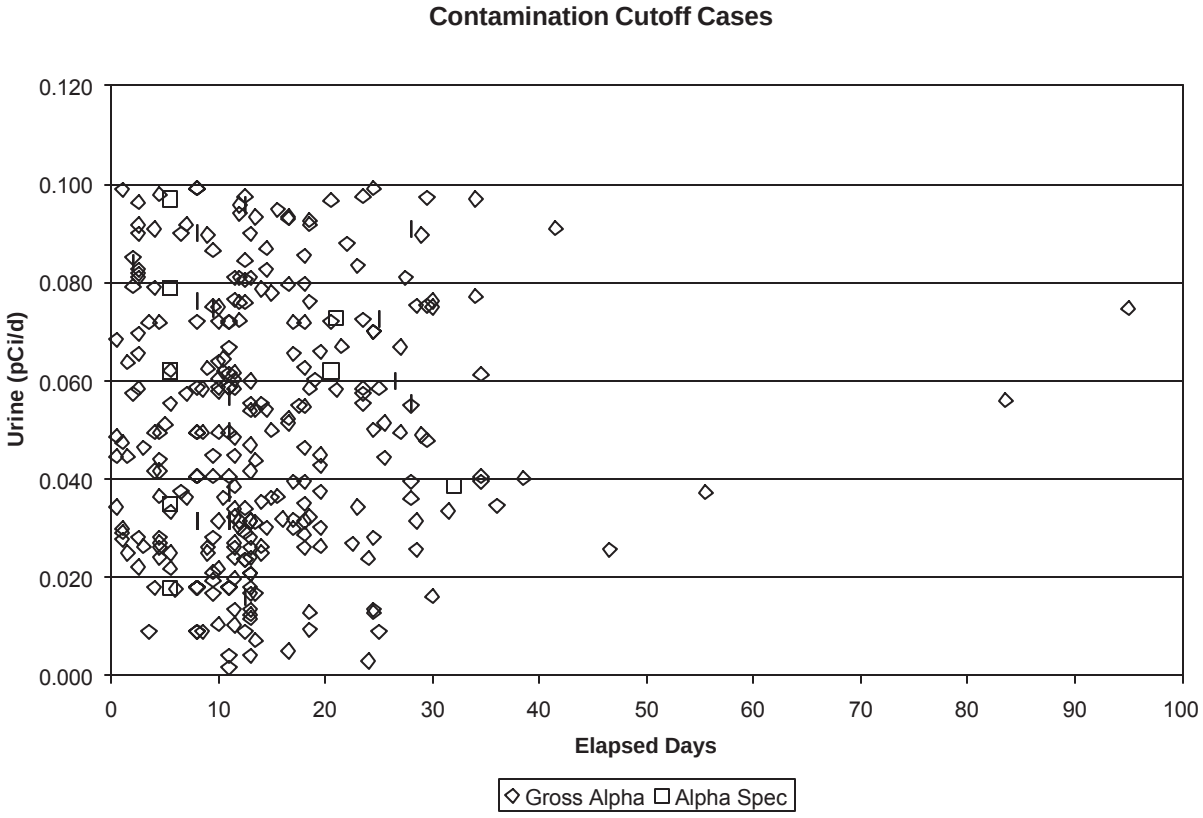

Figure E- 8. Urine results for the Contamination Cutoff Cases Group.

E.4.3.2.1 Date of Exposure

The Contamination Cutoff Cases Group had exposure dates that began over a similar range of dates to the Repeat Analysis Cases Group. Many of this group stayed on site for one to two weeks, with some up to a month. A few appeared to remain until the very end of operations. As

for the High 26 Cases Group, some sample dates were assigned. Since the time on site seem shorter and better recorded for this group, the exposure date was assumed as the midpoint of the time at Camp Wilson.

E.4.3.2.2 Use of Measurements

As mentioned in Section D-4.3.1, 30 individuals submitted more than one sample. The lowest results for any individual were used regardless of whether the analysis was performed using gross alpha counting or alpha spectrometry.

E.4.3.2.3 Weighting Factors for Urine Measurements

Each individual case contained only one measurement. Consequently, weighting factors were not a consideration for this group of assessments.

E.4.3.3. Results

The methods used for estimating intakes and doses for the High 26 Cases Group were applied to the Repeat Analysis Cases Group. Some adjustments were necessary to accommodate the specific data qualities for each case. The results for each individual are listed anonymously with the pertinent data used for calculating the estimated intake and dose equivalent in Table E-8. This section summarizes the overall results and discusses approaches for developing estimates that are more reasonable.

E.4.3.3.1 Intakes and Doses

For the 313 individuals in the Contamination Cutoff Cases Group, the estimated intakes varied from 1,500 pCi to 110,000 pCi. Estimates of committed effective dose equivalent ranged from 0.46 rem to 34 rem (0.0046 to 0.34 Sv). The higher intake and dose estimate were produced by a urine sample, taken at 25 days after the assumed exposure date, which produced a result of 0.099 pCi/d of gross alpha activity. According to the excretion function derived, the urinary content on day 25 represents approximately 9×10^{-7} of the inhalation intake. This case illustrates how urine concentrations that are even slightly above detectability can lead to sizeable estimated intakes and dose equivalents.

E.4.3.4. Remaining Cases Group

The individual cases that were not evaluated in one of the previous three groups were placed in the Remaining Cases Group. These samples included those from individuals who submitted only one sample, or from cases where some follow-up was attempted but results were inadequate because of low or no chemical recovery or laboratory error. This group contains sample measurements on 1,063 individuals for 1,219 samples. Figure E-9 illustrates the distribution of the results with positive values. The remaining results were zero, NDA, or not reported.

Table E- 8. Contamination Cutoff Group Preliminary intake, committed dose equivalent, and committed effective dose equivalent estimates.

Name	Intake (pCi)	CEDE	Testes	Breast	R Marrow	Lung	Thyroid	Bone Sur	Liver	Other	LL Int.	UL Int.	S Int.
Data Masked	1.5E+03	0.46	0.1	0.0	0.4	1.7	0.0	4.7	0.8	0.1	0.0	0.0	0.0
Data Masked	2.4E+03	0.74	0.1	0.0	0.6	2.7	0.0	7.5	1.4	0.1	0.0	0.0	0.0
Data Masked	2.5E+03	0.77	0.1	0.0	0.6	2.8	0.0	7.8	1.4	0.1	0.0	0.0	0.0
Data Masked	2.6E+03	0.8	0.1	0.0	0.6	2.9	0.0	8.1	1.5	0.1	0.0	0.0	0.0
Data Masked	2.8E+03	0.86	0.1	0.0	0.7	3.2	0.0	8.8	1.6	0.1	0.0	0.0	0.0
Data Masked	2.9E+03	0.89	0.1	0.0	0.7	3.3	0.0	9.1	1.6	0.2	0.0	0.0	0.0
Data Masked	3.2E+03	0.98	0.1	0.0	0.8	3.6	0.0	10.0	1.8	0.2	0.0	0.0	0.0
Data Masked	3.3E+03	1	0.1	0.0	0.8	3.7	0.0	10.3	1.9	0.2	0.0	0.0	0.0
Data Masked	3.4E+03	1	0.1	0.0	0.8	3.8	0.0	10.6	1.9	0.2	0.0	0.0	0.0
Data Masked	3.6E+03	11	0.2	0.0	0.9	4.1	0.0	11.3	2.0	0.2	0.0	0.0	0.0
Data Masked	3.8E+03	1.2	0.2	0.0	0.9	4.3	0.0	11.9	2.1	0.2	0.0	0.0	0.0
Data Masked	3.8E+03	1.2	0.2	0.0	0.9	4.3	0.0	11.9	2.1	0.2	0.0	0.0	0.0
Data Masked	4.0E+03	1.2	0.2	0.0	1.0	4.5	0.0	12.5	2.3	0.2	0.0	0.0	0.0
Data Masked	4.1E+03	1.3	0.2	0.0	1.0	4.6	0.0	12.8	2.3	0.2	0.0	0.0	0.0
Data Masked	4.7E+03	1.4	0.2	0.0	1.1	5.3	0.0	14.7	2.7	0.2	0.0	0.0	0.0
Data Masked	4.7E+03	1.4	0.2	0.0	1.1	5.3	0.0	14.7	2.7	0.2	0.0	0.0	0.0
Data Masked	5.0E+03	1.5	0.2	0.0	1.2	5.7	0.0	15.7	2.8	0.3	0.0	0.0	0.0
Data Masked	5.5E+03	1.7	0.2	0.0	1.3	6.2	0.0	17.2	3.1	0.3	0.0	0.0	0.0
Data Masked	5.8E+03	1.8	0.3	0.0	1.4	6.6	0.0	18.2	3.3	0.3	0.0	0.0	0.0
Data Masked	6.0E+03	1.8	0.3	0.0	1.4	6.8	0.0	18.8	3.4	0.3	0.0	0.0	0.0
Data Masked	6.1E+03	1.9	0.3	0.0	1.5	6.9	0.0	19.1	3.4	0.3	0.0	0.0	0.0
Data Masked	6.4E+03	2	0.3	0.0	1.5	7.2	0.0	20.0	3.6	0.3	0.0	0.0	0.0
Data Masked	6.4E+03	2	0.3	0.0	1.5	7.2	0.0	20.0	3.6	0.3	0.0	0.0	0.0
Data Masked	6.8E+03	2.1	0.3	0.0	1.6	7.7	0.0	21.3	3.8	0.4	0.0	0.0	0.0
Data Masked	6.8E+03	21	0.3	0.0	1.6	7.7	0.0	21.3	3.8	0.4	0.0	0.0	0.0
Data Masked	7.0E+03	2.2	0.3	0.0	1.7	7.9	0.0	21.9	4.0	0.4	0.0	0.0	0.0
Data Masked	7.9E+03	2.4	0.3	0.0	1.9	8.9	0.0	24.7	4.5	0.4	0.0	0.0	0.0
Data Masked	8.0E+03	2.5	0.3	0.0	1.9	9.0	0.0	25.0	4.5	0.4	0.0	0.0	0.0
Data Masked	8.4E+03	2.6	0.4	0.0	2.0	9.5	0.0	26.3	4.7	0.4	0.0	0.0	0.0
Data Masked	8.6E+03	2.6	0.4	0.0	2.1	9.7	0.0	26.9	4.9	0.4	0.0	0.0	0.0
Data Masked	8.7E+03	2.7	0.4	0.0	2.1	9.8	0.0	27.2	4.9	0.5	0.0	0.0	0.0
Data Masked	9.2E+03	2.8	0.4	0.0	2.2	10.4	0.0	28.8	5.2	0.5	0.0	0.0	0.0
Data Masked	9.4E+03	2.9	0.4	0.0	2.2	10.6	0.0	29.4	5.3	0.5	0.0	0.0	0.0
Data Masked	9.5E+03	2.9	0.4	0.0	2.3	10.7	0.0	29.7	5.4	0.5	0.0	0.0	0.0
Data Masked	9.8E+03	3	0.4	0.0	2.3	11.1	0.0	30.7	5.5	0.5	0.0	0.0	0.0
Data Masked	1.0E+04	3.1	0.4	0.0	2.4	11.3	0.0	31.3	5.7	0.5	0.0	0.0	0.0
Data Masked	1.0E+04	3.1	0.4	0.0	2.4	11.3	0.0	31.3	5.7	0.5	0.0	0.0	0.0
Data Masked	1.1E+04	3.4	0.5	0.0	2.6	12.4	0.0	34.4	6.2	0.6	0.0	0.0	0.0
Data Masked	1.1E+04	3.4	0.5	0.0	2.6	12.4	0.0	34.4	6.2	0.6	0.0	0.0	0.0
Data Masked	1.1E+04	3.4	0.5	0.0	2.6	12.4	0.0	34.4	6.2	0.6	0.0	0.0	0.0
Data Masked	1.1E+04	3.4	0.5	0.0	2.6	12.4	0.0	34.4	6.2	0.6	0.0	0.0	0.0
Data Masked	1.1E+04	3.4	0.5	0.0	2.6	12.4	0.0	34.4	6.2	0.6	0.0	0.0	0.0
Data Masked	1.2E+04	3.7	0.5	0.0	2.9	13.6	0.0	37.6	6.8	0.6	0.0	0.0	0.0
Data Masked	1.2E+04	3.7	0.5	0.0	2.9	13.6	0.0	37.6	6.8	0.6	0.0	0.0	0.0
Data Masked	1.2E+04	3.7	0.5	0.0	2.9	13.6	0.0	37.6	6.8	0.6	0.0	0.0	0.0
Data Masked	1.2E+04	3.7	0.5	0.0	2.9	13.6	0.0	37.6	6.8	0.6	0.0	0.0	0.0
Data Masked	1.2E+04	3.7	0.5	0.0	2.9	13.6	0.0	37.6	6.8	0.6	0.0	0.0	0.0
Data Masked	1.2E+04	3.7	0.5	0.0	2.9	13.6	0.0	37.6	6.8	0.6	0.0	0.0	0.0
Data Masked	1.2E+04	3.7	0.5	0.0	2.9	13.6	0.0	37.6	6.8	0.6	0.0	0.0	0.0
Data Masked	1.3E+04	4	0.6	0.0	3.1	14.7	0.0	40.7	7.3	0.7	0.0	0.0	0.0
Data Masked	1.3E+04	4	0.6	0.0	3.1	14.7	0.0	40.7	7.3	0.7	0.0	0.0	0.0
Data Masked	1.3E+04	4	0.6	0.0	3.1	14.7	0.0	40.7	7.3	0.7	0.0	0.0	0.0
Data Masked	1.3E+04	4	0.6	0.0	3.1	14.7	0.0	40.7	7.3	0.7	0.0	0.0	0.0
Data Masked	1.3E+04	4	0.6	0.0	3.1	14.7	0.0	40.7	7.3	0.7	0.0	0.0	0.0
Data Masked	1.3E+04	4	0.6	0.0	3.1	14.7	0.0	40.7	7.3	0.7	0.0	0.0	0.0
Data Masked	1.3E+04	40	0.6	0.0	3.1	14.7	0.0	40.7	7.3	0.7	0.0	0.0	0.0
Data Masked	1.3E+04	4	0.6	0.0	3.1	14.7	0.0	40.7	7.3	0.7	0.0	0.0	0.0
Data Masked	1.4E+04	4.3	0.6	0.0	3.3	15.8	0.0	43.8	7.9	0.7	0.0	0.0	0.0
Data Masked	1.4E+04	4.3	0.6	0.0	3.3	15.8	0.0	43.8	7.9	0.7	0.0	0.0	0.0
Data Masked	1.4E+04	4.3	0.6	0.0	3.3	15.8	0.0	43.8	7.9	0.7	0.0	0.0	0.0
Data Masked	1.5E+04	4.6	0.7	0.0	3.6	17.0	0.0	47.0	8.5	0.8	0.0	0.0	0.0
Data Masked	1.5E+04	4.6	0.7	0.0	3.6	17.0	0.0	47.0	8.5	0.8	0.0	0.0	0.0
Data Masked	1.5E+04	4.6	0.7	0.0	3.6	17.0	0.0	47.0	8.5	0.8	0.0	0.0	0.0
Data Masked	1.5E+04	4.6	0.7	0.0	3.6	17.0	0.0	47.0	8.5	0.8	0.0	0.0	0.0
Data Masked	1.5E+04	4.6	0.7	0.0	3.6	17.0	0.0	47.0	8.5	0.8	0.0	0.0	0.0
Data Masked	1.5E+04	4.6	0.7	0.0	3.6	17.0	0.0	47.0	8.5	0.8	0.0	0.0	0.0
Data Masked	1.6E+04	4.9	0.7	0.0	3.8	18.1	0.0	50.1	9.0	0.8	0.0	0.0	0.0
Data Masked	1.6E+04	4.9	0.7	0.0	3.8	18.1	0.0	50.1	9.0	0.8	0.0	0.0	0.0
Data Masked	1.6E+04	4.9	0.7	0.0	3.8	18.1	0.0	50.1	9.0	0.8	0.0	0.0	0.0
Data Masked	1.6E+04	4.9	0.7	0.0	3.8	18.1	0.0	50.1	9.0	0.8	0.0	0.0	0.0
Data Masked	1.7E+04	5.2	0.7	0.0	4.1	19.2	0.0	53.2	9.6	0.9	0.0	0.0	0.0
Data Masked	1.7E+04	5.2	0.7	0.0	4.1	19.2	0.0	53.2	9.6	0.9	0.0	0.0	0.0
Data Masked	1.7E+04	5.2	0.7	0.0	4.1	19.2	0.0	53.2	9.6	0.9	0.0	0.0	0.0
Data Masked	1.7E+04	5.2	0.7	0.0	4.1	19.2	0.0	53.2	9.6	0.9	0.0	0.0	0.0

Table E- 8. Contamination Cutoff Group Preliminary intake, committed dose equivalent, and committed effective dose equivalent estimates.

Data Masked	1.9E+04	5.8	0.8	0.0	4.5	21.5	0.0	59.5	10.7	1.0	0.0	0.0	0.0
Data Masked	1.9E+04	5.8	0.8	0.0	4.5	21.5	0.0	59.5	10.7	1.0	0.0	0.0	0.0
Data Masked	1.9E+04	5.8	0.8	0.0	4.5	21.5	0.0	59.5	10.7	1.0	0.0	0.0	0.0
Data Masked	2.0E+04	5.8	0.9	0.0	4.8	22.6	0.0	62.6	11.3	1.0	0.0	0.0	0.0
Data Masked	2.1E+04	6.1	0.9	0.0	5.0	23.7	0.0	65.7	11.9	1.1	0.0	0.0	0.0
Data Masked	2.1E+04	6.5	0.9	0.0	5.0	23.7	0.0	65.7	11.9	1.1	0.0	0.0	0.0
Data Masked	2.1E+04	6.5	0.9	0.0	5.0	23.7	0.0	65.7	11.9	1.1	0.0	0.0	0.0
Data Masked	2.1E+04	6.5	0.9	0.0	5.0	23.7	0.0	65.7	11.9	1.1	0.0	0.0	0.0
Data Masked	2.1E+04	6.5	0.9	0.0	5.0	23.7	0.0	65.7	11.9	1.1	0.0	0.0	0.0
Data Masked	2.2E+04	6.5	1.0	0.0	5.3	24.9	0.0	68.9	12.4	1.1	0.0	0.0	0.0
Data Masked	2.2E+04	6.8	1.0	0.0	5.3	24.9	0.0	68.9	12.4	1.1	0.0	0.0	0.0
Data Masked	2.2E+04	6.8	1.0	0.0	5.3	24.9	0.0	68.9	12.4	1.1	0.0	0.0	0.0
Data Masked	2.2E+04	6.8	1.0	0.0	5.3	24.9	0.0	68.9	12.4	1.1	0.0	0.0	0.0
Data Masked	2.2E+04	6.8	1.0	0.0	5.3	24.9	0.0	68.9	12.4	1.1	0.0	0.0	0.0
Data Masked	2.3E+04	6.8	1.0	0.0	5.5	26.0	0.0	72.0	13.0	1.2	0.0	0.0	0.0
Data Masked	2.3E+04	7.1	1.0	0.0	5.5	26.0	0.0	72.0	13.0	1.2	0.0	0.0	0.0
Data Masked	2.3E+04	7.1	1.0	0.0	5.5	26.0	0.0	72.0	13.0	1.2	0.0	0.0	0.0
Data Masked	2.3E+04	7.1	1.0	0.0	5.5	26.0	0.0	72.0	13.0	1.2	0.0	0.0	0.0
Data Masked	2.4E+04	7.1	1.0	0.0	5.7	27.1	0.0	75.1	13.6	1.3	0.0	0.0	0.0
Data Masked	2.4E+04	7.4	1.0	0.0	5.7	27.1	0.0	75.1	13.6	1.3	0.0	0.0	0.0
Data Masked	2.4E+04	7.4	1.0	0.0	5.7	27.1	0.0	75.1	13.6	1.3	0.0	0.0	0.0
Data Masked	2.4E+04	7.4	1.0	0.0	5.7	27.1	0.0	75.1	13.6	1.3	0.0	0.0	0.0
Data Masked	2.4E+04	7.4	1.0	0.0	5.7	27.1	0.0	75.1	13.6	1.3	0.0	0.0	0.0
Data Masked	2.5E+04	7.7	1.1	0.0	6.0	28.3	0.0	78.3	14.1	1.3	0.0	0.0	0.0
Data Masked	2.6E+04	8	1.1	0.0	6.2	29.4	0.0	81.4	14.7	1.4	0.0	0.0	0.0
Data Masked	2.6E+04	8	1.1	0.0	6.2	29.4	0.0	81.4	14.7	1.4	0.0	0.0	0.0
Data Masked	2.6E+04	8	1.1	0.0	6.2	29.4	0.0	81.4	14.7	1.4	0.0	0.0	0.0
Data Masked	2.6E+04	8	1.1	0.0	6.2	29.4	0.0	81.4	14.7	1.4	0.0	0.0	0.0
Data Masked	2.7E+04	8.3	1.2	0.0	6.5	30.5	0.0	84.5	15.3	1.4	0.0	0.0	0.0
Data Masked	2.7E+04	8.3	1.2	0.0	6.5	30.5	0.0	84.5	15.3	1.4	0.0	0.0	0.0
Data Masked	2.7E+04	8.3	1.2	0.0	6.5	30.5	0.0	84.5	15.3	1.4	0.0	0.0	0.0
Data Masked	2.7E+04	8.3	1.2	0.0	6.5	30.5	0.0	84.5	15.3	1.4	0.0	0.0	0.0
Data Masked	2.8E+04	8.6	1.2	0.0	6.7	31.7	0.0	87.7	15.8	1.5	0.0	0.0	0.0
Data Masked	2.8E+04	8.6	1.2	0.0	6.7	31.7	0.0	87.7	15.8	1.5	0.0	0.0	0.0
Data Masked	2.8E+04	8.6	1.2	0.0	6.7	31.7	0.0	87.7	15.8	1.5	0.0	0.0	0.0
Data Masked	2.9E+04	8.9	1.3	0.0	6.9	32.8	0.0	90.8	16.4	1.5	0.0	0.0	0.0
Data Masked	2.9E+04	8.9	1.3	0.0	6.9	32.8	0.0	90.8	16.4	1.5	0.0	0.0	0.0
Data Masked	2.9E+04	8.9	1.3	0.0	6.9	32.8	0.0	90.8	16.4	1.5	0.0	0.0	0.0
Data Masked	2.9E+04	8.9	1.3	0.0	6.9	32.8	0.0	90.8	16.4	1.5	0.0	0.0	0.0
Data Masked	2.9E+04	8.9	1.3	0.0	6.9	32.8	0.0	90.8	16.4	1.5	0.0	0.0	0.0
Data Masked	2.9E+04	8.9	1.3	0.0	6.9	32.8	0.0	90.8	16.4	1.5	0.0	0.0	0.0
Data Masked	2.9E+04	8.9	1.3	0.0	6.9	32.8	0.0	90.8	16.4	1.5	0.0	0.0	0.0
Data Masked	2.9E+04	8.9	1.3	0.0	6.9	32.8	0.0	90.8	16.4	1.5	0.0	0.0	0.0
Data Masked	2.9E+04	8.9	1.3	0.0	6.9	32.8	0.0	90.8	16.4	1.5	0.0	0.0	0.0
Data Masked	2.9E+04	8.9	1.3	0.0	6.9	32.8	0.0	90.8	16.4	1.5	0.0	0.0	0.0
Data Masked	3.0E+04	9.2	1.3	0.0	7.2	33.9	0.0	93.9	17.0	1.6	0.0	0.0	0.0
Data Masked	3.0E+04	9.2	1.3	0.0	7.2	33.9	0.0	93.9	17.0	1.6	0.0	0.0	0.0
Data Masked	3.0E+04	9.2	1.3	0.0	7.2	33.9	0.0	93.9	17.0	1.6	0.0	0.0	0.0
Data Masked	3.1E+04	9.5	1.3	0.0	7.4	35.0	0.0	97.0	17.5	1.6	0.0	0.0	0.0
Data Masked	3.1E+04	9.5	1.3	0.0	7.4	35.0	0.0	97.0	17.5	1.6	0.0	0.0	0.0
Data Masked	3.1E+04	9.5	1.3	0.0	7.4	35.0	0.0	97.0	17.5	1.6	0.0	0.0	0.0
Data Masked	3.1E+04	9.5	1.3	0.0	7.4	35.0	0.0	97.0	17.5	1.6	0.0	0.0	0.0
Data Masked	3.1E+04	9.5	1.3	0.0	7.4	35.0	0.0	97.0	17.5	1.6	0.0	0.0	0.0
Data Masked	3.1E+04	9.5	1.3	0.0	7.4	35.0	0.0	97.0	17.5	1.6	0.0	0.0	0.0
Data Masked	3.1E+04	9.5	1.3	0.0	7.4	35.0	0.0	97.0	17.5	1.6	0.0	0.0	0.0
Data Masked	3.1E+04	9.5	1.3	0.0	7.4	35.0	0.0	97.0	17.5	1.6	0.0	0.0	0.0
Data Masked	3.2E+04	9.8	1.4	0.0	7.7	36.2	0.0	100.2	18.1	1.7	0.0	0.0	0.0
Data Masked	3.2E+04	9.8	1.4	0.0	7.7	36.2	0.0	100.2	18.1	1.7	0.0	0.0	0.0
Data Masked	3.3E+04	10	1.4	0.0	7.9	37.3	0.0	103.3	18.7	1.7	0.0	0.0	0.0
Data Masked	3.3E+04	10	1.4	0.0	7.9	37.3	0.0	103.3	18.7	1.7	0.0	0.0	0.0
Data Masked	3.3E+04	10	1.4	0.0	7.9	37.3	0.0	103.3	18.7	1.7	0.0	0.0	0.0
Data Masked	3.3E+04	10	1.4	0.0	7.9	37.3	0.0	103.3	18.7	1.7	0.0	0.0	0.0
Data Masked	3.3E+04	10	1.4	0.0	7.9	37.3	0.0	103.3	18.7	1.7	0.0	0.0	0.0
Data Masked	3.3E+04	10	1.4	0.0	7.9	37.3	0.0	103.3	18.7	1.7	0.0	0.0	0.0
Data Masked	3.4E+04	10	1.5	0.0	8.1	38.4	0.0	106.4	19.2	1.8	0.0	0.0	0.0
Data Masked	3.4E+04	10	1.5	0.0	8.1	38.4	0.0	106.4	19.2	1.8	0.0	0.0	0.0
Data Masked	3.4E+04	10	1.5	0.0	8.1	38.4	0.0	106.4	19.2	1.8	0.0	0.0	0.0
Data Masked	3.5E+04	11	1.5	0.0	8.4	39.6	0.0	109.6	19.8	1.8	0.0	0.0	0.0
Data Masked	3.5E+04	11	1.5	0.0	8.4	39.6	0.0	109.6	19.8	1.8	0.0	0.0	0.0
Data Masked	3.5E+04	11	1.5	0.0	8.4	39.6	0.0	109.6	19.8	1.8	0.0	0.0	0.0
Data Masked	3.5E+04	11	1.5	0.0	8.4	39.6	0.0	109.6	19.8	1.8	0.0	0.0	0.0
Data Masked	3.5E+04	11	1.5	0.0	8.4	39.6	0.0	109.6	19.8	1.8	0.0	0.0	0.0
Data Masked	3.6E+04	11	1.6	0.0	8.6	40.7	0.0	112.7	20.3	1.9	0.0	0.0	0.0
Data Masked	3.6E+04	11	1.6	0.0	8.6	40.7	0.0	112.7	20.3	1.9	0.0	0.0	0.0
Data Masked	3.6E+04	11	1.6	0.0	8.6	40.7	0.0	112.7	20.3	1.9	0.0	0.0	0.0

Table E- 8. Contamination Cutoff Group Preliminary intake, committed dose equivalent, and committed effective dose equivalent estimates.

Data Masked	4.1E+04	13	1.8	0.0	9.8	46.3	0.0	128.3	23.2	2.1	0.0	0.0	0.0
Data Masked	4.1E+04	13	1.8	0.0	9.8	46.3	0.0	128.3	23.2	2.1	0.0	0.0	0.0
Data Masked	4.1E+04	13	1.8	0.0	9.8	46.3	0.0	128.3	23.2	2.1	0.0	0.0	0.0
Data Masked	4.2E+04	13	1.8	0.0	10.0	47.5	0.0	131.5	23.7	2.2	0.0	0.0	0.0
Data Masked	4.2E+04	13	1.8	0.0	10.0	47.5	0.0	131.5	23.7	2.2	0.0	0.0	0.0
Data Masked	4.2E+04	13	1.8	0.0	10.0	47.5	0.0	131.5	23.7	2.2	0.0	0.0	0.0
Data Masked	4.3E+04	13	1.9	0.0	10.3	48.6	0.0	134.6	24.3	2.2	0.0	0.0	0.0
Data Masked	4.3E+04	13	1.9	0.0	10.3	48.6	0.0	134.6	24.3	2.2	0.0	0.0	0.0
Data Masked	4.3E+04	13	1.9	0.0	10.3	48.6	0.0	134.6	24.3	2.2	0.0	0.0	0.0
Data Masked	4.4E+04	14	1.9	0.0	10.5	49.7	0.0	137.7	24.9	2.3	0.0	0.0	0.0
Data Masked	4.5E+04	14	2.0	0.0	10.8	50.9	0.0	140.9	25.4	2.3	0.0	0.0	0.0
Data Masked	4.5E+04	14	2.0	0.0	10.8	50.9	0.0	140.9	25.4	2.3	0.0	0.0	0.0
Data Masked	4.6E+04	14	2.0	0.0	11.0	52.0	0.0	144.0	26.0	2.4	0.0	0.0	0.0
Data Masked	4.6E+04	14	2.0	0.0	11.0	52.0	0.0	144.0	26.0	2.4	0.0	0.0	0.0
Data Masked	4.7E+04	14	2.0	0.0	11.2	53.1	0.0	147.1	26.6	2.5	0.0	0.0	0.0
Data Masked	4.7E+04	14	2.0	0.0	11.2	53.1	0.0	147.1	26.6	2.5	0.0	0.0	0.0
Data Masked	4.7E+04	14	2.0	0.0	11.2	53.1	0.0	147.1	26.6	2.5	0.0	0.0	0.0
Data Masked	4.8E+04	15	2.1	0.0	11.5	54.3	0.0	150.3	27.1	2.5	0.0	0.0	0.0
Data Masked	4.9E+04	15	2.1	0.0	11.7	55.4	0.0	153.4	27.7	2.6	0.0	0.0	0.0
Data Masked	4.9E+04	15	2.1	0.0	11.7	55.4	0.0	153.4	27.7	2.6	0.0	0.0	0.0
Data Masked	4.9E+04	15	2.1	0.0	11.7	55.4	0.0	153.4	27.7	2.6	0.0	0.0	0.0
Data Masked	5.0E+04	15	2.2	0.0	12.0	56.5	0.0	156.5	28.3	2.6	0.0	0.0	0.0
Data Masked	5.0E+04	15	2.2	0.0	12.0	56.5	0.0	156.5	28.3	2.6	0.0	0.0	0.0
Data Masked	5.0E+04	15	2.2	0.0	12.0	56.5	0.0	156.5	28.3	2.6	0.0	0.0	0.0
Data Masked	5.0E+04	15	2.2	0.0	12.0	56.5	0.0	156.5	28.3	2.6	0.0	0.0	0.0
Data Masked	5.0E+04	15	2.2	0.0	12.0	56.5	0.0	156.5	28.3	2.6	0.0	0.0	0.0
Data Masked	5.1E+04	16	2.2	0.0	12.2	57.7	0.0	159.7	28.8	2.7	0.0	0.0	0.0
Data Masked	5.1E+04	16	2.2	0.0	12.2	57.7	0.0	159.7	28.8	2.7	0.0	0.0	0.0
Data Masked	5.1E+04	16	2.2	0.0	12.2	57.7	0.0	159.7	28.8	2.7	0.0	0.0	0.0
Data Masked	5.1E+04	16	2.2	0.0	12.2	57.7	0.0	159.7	28.8	2.7	0.0	0.0	0.0
Data Masked	5.1E+04	16	2.2	0.0	12.2	57.7	0.0	159.7	28.8	2.7	0.0	0.0	0.0
Data Masked	5.2E+04	16	2.3	0.0	12.4	58.8	0.0	162.8	29.4	2.7	0.0	0.0	0.0
Data Masked	5.2E+04	16	2.3	0.0	12.4	58.8	0.0	162.8	29.4	2.7	0.0	0.0	0.0
Data Masked	5.2E+04	16	2.3	0.0	12.4	58.8	0.0	162.8	29.4	2.7	0.0	0.0	0.0
Data Masked	5.2E+04	16	2.3	0.0	12.4	58.8	0.0	162.8	29.4	2.7	0.0	0.0	0.0
Data Masked	5.2E+04	16	2.3	0.0	12.4	58.8	0.0	162.8	29.4	2.7	0.0	0.0	0.0
Data Masked	5.2E+04	16	2.3	0.0	12.4	58.8	0.0	162.8	29.4	2.7	0.0	0.0	0.0
Data Masked	5.3E+04	16	2.3	0.0	12.7	59.9	0.0	165.9	30.0	2.8	0.0	0.0	0.0
Data Masked	5.3E+04	16	2.3	0.0	12.7	59.9	0.0	165.9	30.0	2.8	0.0	0.0	0.0
Data Masked	5.3E+04	16	2.3	0.0	12.7	59.9	0.0	165.9	30.0	2.8	0.0	0.0	0.0
Data Masked	5.4E+04	17	2.3	0.0	12.9	61.0	0.0	169.0	30.5	2.8	0.0	0.0	0.0
Data Masked	5.4E+04	17	2.3	0.0	12.9	61.0	0.0	169.0	30.5	2.8	0.0	0.0	0.0
Data Masked	5.4E+04	17	2.3	0.0	12.9	61.0	0.0	169.0	30.5	2.8	0.0	0.0	0.0
Data Masked	5.5E+04	17	2.4	0.0	13.2	62.2	0.0	172.2	31.1	2.9	0.0	0.0	0.0
Data Masked	5.5E+04	17	2.4	0.0	13.2	62.2	0.0	172.2	31.1	2.9	0.0	0.0	0.0
Data Masked	5.5E+04	17	2.4	0.0	13.2	62.2	0.0	172.2	31.1	2.9	0.0	0.0	0.0
Data Masked	5.5E+04	17	2.4	0.0	13.2	62.2	0.0	172.2	31.1	2.9	0.0	0.0	0.0
Data Masked	5.5E+04	17	2.4	0.0	13.2	62.2	0.0	172.2	31.1	2.9	0.0	0.0	0.0
Data Masked	5.5E+04	17	2.4	0.0	13.2	62.2	0.0	172.2	31.1	2.9	0.0	0.0	0.0
Data Masked	5.6E+04	17	2.4	0.0	13.4	63.3	0.0	175.3	31.7	2.9	0.0	0.0	0.0
Data Masked	5.6E+04	17	2.4	0.0	13.4	63.3	0.0	175.3	31.7	2.9	0.0	0.0	0.0
Data Masked	5.7E+04	18	2.5	0.0	13.6	64.4	0.0	178.4	32.2	3.0	0.0	0.0	0.0
Data Masked	5.7E+04	18	2.5	0.0	13.6	64.4	0.0	178.4	32.2	3.0	0.0	0.0	0.0
Data Masked	5.7E+04	18	2.5	0.0	13.6	64.4	0.0	178.4	32.2	3.0	0.0	0.0	0.0
Data Masked	5.7E+04	18	2.5	0.0	13.6	64.4	0.0	178.4	32.2	3.0	0.0	0.0	0.0
Data Masked	5.8E+04	18	2.5	0.0	13.9	65.6	0.0	181.6	32.8	3.0	0.0	0.0	0.0
Data Masked	5.9E+04	18	2.6	0.0	14.1	66.7	0.0	184.7	33.3	3.1	0.0	0.0	0.0
Data Masked	6.0E+04	18	2.6	0.0	14.3	67.8	0.0	187.8	33.9	3.1	0.0	0.0	0.0
Data Masked	6.1E+04	19	2.7	0.0	14.6	69.0	0.0	191.0	34.5	3.2	0.0	0.0	0.0
Data Masked	6.1E+04	19	2.7	0.0	14.6	69.0	0.0	191.0	34.5	3.2	0.0	0.0	0.0
Data Masked	6.1E+04	19	2.7	0.0	14.6	69.0	0.0	191.0	34.5	3.2	0.0	0.0	0.0
Data Masked	6.2E+04	19	2.7	0.0	14.8	70.1	0.0	194.1	35.0	3.2	0.0	0.0	0.0
Data Masked	6.2E+04	19	2.7	0.0	14.8	70.1	0.0	194.1	35.0	3.2	0.0	0.0	0.0
Data Masked	6.2E+04	19	2.7	0.0	14.8	70.1	0.0	194.1	35.0	3.2	0.0	0.0	0.0
Data Masked	6.3E+04	19	2.7	0.0	15.1	71.2	0.0	197.2	35.6	3.3	0.0	0.0	0.0
Data Masked	6.3E+04	19	2.7	0.0	15.1	71.2	0.0	197.2	35.6	3.3	0.0	0.0	0.0
Data Masked	6.3E+04	19	2.7	0.0	15.1	71.2	0.0	197.2	35.6	3.3	0.0	0.0	0.0
Data Masked	6.4E+04	20	2.8	0.0	15.3	72.3	0.0	200.3	36.2	3.3	0.0	0.0	0.0
Data Masked	6.4E+04	20	2.8	0.0	15.3	72.3	0.0	200.3	36.2	3.3	0.0	0.0	0.0
Data Masked	6.4E+04	20	2.8	0.0	15.3	72.3	0.0	200.3	36.2	3.3	0.0	0.0	0.0
Data Masked	6.4E+04	20	2.8	0.0	15.3	72.3	0.0	200.3	36.2	3.3	0.0	0.0	0.0
Data Masked	6.5E+04	20	2.8	0.0	15.5	73.5	0.0	203.5	36.7	3.4	0.0	0.0	0.0
Data Masked	6.5E+04	20	2.8	0.0	15.5	73.5	0.0	203.5	36.7	3.4	0.0	0.0	0.0
Data Masked	6.5E+04	20	2.8	0.0	15.5	73.5	0.0	203.5	36.7	3.4	0.0	0.0	0.0
Data Masked	6.6E+04	20	2.9	0.0	15.8	74.6	0.0	206.6	37.3	3.4	0.0	0.0	0.0

Table E- 8. Contamination Cutoff Group Preliminary intake, committed dose equivalent, and committed effective dose equivalent estimates.

Data Masked	7.2E+04	22	3.1	0.0	17.2	81.4	0.0	225.4	40.7	3.8	0.0	0.0	0.0
Data Masked	7.2E+04	22	3.1	0.0	17.2	81.4	0.0	225.4	40.7	3.8	0.0	0.0	0.0
Data Masked	7.3E+04	22	3.2	0.0	17.5	82.5	0.0	228.5	41.3	3.8	0.0	0.0	0.0
Data Masked	7.3E+04	22	3.2	0.0	17.5	82.5	0.0	228.5	41.3	3.8	0.0	0.0	0.0
Data Masked	7.4E+04	23	3.2	0.0	17.7	83.7	0.0	231.7	41.8	3.9	0.0	0.0	0.0
Data Masked	7.4E+04	23	3.2	0.0	17.7	83.7	0.0	231.7	41.8	3.9	0.0	0.0	0.0
Data Masked	7.5E+04	23	3.3	0.0	17.9	84.8	0.0	234.8	42.4	3.9	0.0	0.0	0.0
Data Masked	7.6E+04	23	3.3	0.0	18.2	85.9	0.0	237.9	43.0	4.0	0.0	0.0	0.0
Data Masked	7.6E+04	23	3.3	0.0	18.2	85.9	0.0	237.9	43.0	4.0	0.0	0.0	0.0
Data Masked	7.7E+04	24	3.3	0.0	18.4	87.0	0.0	241.0	43.5	4.0	0.0	0.0	0.0
Data Masked	7.7E+04	24	3.3	0.0	18.4	87.0	0.0	241.0	43.5	4.0	0.0	0.0	0.0
Data Masked	7.7E+04	24	3.3	0.0	18.4	87.0	0.0	241.0	43.5	4.0	0.0	0.0	0.0
Data Masked	7.8E+04	24	3.4	0.0	18.7	88.2	0.0	244.2	44.1	4.1	0.0	0.0	0.0
Data Masked	7.8E+04	24	3.4	0.0	18.7	88.2	0.0	244.2	44.1	4.1	0.0	0.0	0.0
Data Masked	7.9E+04	24	3.4	0.0	18.9	89.3	0.0	247.3	44.7	4.1	0.0	0.0	0.0
Data Masked	8.0E+04	25	3.5	0.0	19.1	90.4	0.0	250.4	45.2	4.2	0.0	0.0	0.0
Data Masked	8.0E+04	25	3.5	0.0	19.1	90.4	0.0	250.4	45.2	4.2	0.0	0.0	0.0
Data Masked	8.0E+04	25	3.5	0.0	19.1	90.4	0.0	250.4	45.2	4.2	0.0	0.0	0.0
Data Masked	8.1E+04	2.5	3.5	0.0	19.4	91.6	0.0	253.6	45.8	4.2	0.0	0.0	0.0
Data Masked	8.2E+04	25	3.6	0.0	19.6	92.7	0.0	256.7	46.3	4.3	0.0	0.0	0.0
Data Masked	8.3E+04	25	3.6	0.0	19.8	93.8	0.0	259.8	46.9	4.3	0.0	0.0	0.0
Data Masked	8.3E+04	25	3.6	0.0	19.8	93.8	0.0	259.8	46.9	4.3	0.0	0.0	0.0
Data Masked	8.3E+04	25	3.6	0.0	19.8	93.8	0.0	259.8	46.9	4.3	0.0	0.0	0.0
Data Masked	8.5E+04	26	3.7	0.0	20.3	96.1	0.0	266.1	48.0	4.4	0.0	0.0	0.0
Data Masked	8.5E+04	26	3.7	0.0	20.3	96.1	0.0	266.1	48.0	4.4	0.0	0.0	0.0
Data Masked	8.5E+04	26	3.7	0.0	20.3	96.1	0.0	266.1	48.0	4.4	0.0	0.0	0.0
Data Masked	8.6E+04	26	3.7	0.0	20.6	97.2	0.0	269.2	48.6	4.5	0.0	0.0	0.0
Data Masked	8.7E+04	27	3.8	0.0	20.8	98.3	0.0	272.3	49.2	4.5	0.0	0.0	0.0
Data Masked	8.7E+04	2.7	3.8	0.0	20.8	98.3	0.0	272.3	49.2	4.5	0.0	0.0	0.0
Data Masked	8.7E+04	27	3.8	0.0	20.8	98.3	0.0	272.3	49.2	4.5	0.0	0.0	0.0
Data Masked	8.8E+04	27	3.8	0.0	21.0	99.5	0.0	275.5	49.7	4.6	0.0	0.0	0.0
Data Masked	8.8E+04	27	3.8	0.0	21.0	99.5	0.0	275.5	49.7	4.6	0.0	0.0	0.0
Data Masked	8.9E+04	27	3.9	0.0	21.3	100.6	0.0	278.6	50.3	4.6	0.0	0.0	0.0
Data Masked	8.9E+04	27	3.9	0.0	21.3	100.6	0.0	278.6	50.3	4.6	0.0	0.0	0.0
Data Masked	9.0E+04	28	3.9	0.0	21.5	101.7	0.0	281.7	50.9	4.7	0.0	0.0	0.0
Data Masked	9.1E+04	28	4.0	0.0	21.8	102.9	0.0	284.9	51.4	4.7	0.0	0.0	0.0
Data Masked	9.2E+04	28	4.0	0.0	22.0	104.0	0.0	288.0	52.0	4.8	0.0	0.0	0.0
Data Masked	9.2E+04	28	4.0	0.0	22.0	104.0	0.0	288.0	52.0	4.8	0.0	0.0	0.0
Data Masked	9.3E+04	29	4.0	0.0	22.2	105.1	0.0	291.1	52.6	4.9	0.0	0.0	0.0
Data Masked	9.3E+04	29	4.0	0.0	22.2	105.1	0.0	291.1	52.6	4.9	0.0	0.0	0.0
Data Masked	9.3E+04	29	4.0	0.0	22.2	105.1	0.0	291.1	52.6	4.9	0.0	0.0	0.0
Data Masked	9.4E+04	29	4.1	0.0	22.5	106.3	0.0	294.3	53.1	4.9	0.0	0.0	0.0
Data Masked	9.4E+04	29	4.1	0.0	22.5	106.3	0.0	294.3	53.1	4.9	0.0	0.0	0.0
Data Masked	9.6E+04	29	4.2	0.0	23.0	108.5	0.0	300.5	54.3	5.0	0.0	0.0	0.0
Data Masked	9.7E+04	30	4.2	0.0	23.2	109.7	0.0	303.7	54.8	5.1	0.0	0.0	0.0
Data Masked	1.0E+05	31	4.3	0.0	23.9	113.0	0.0	313.0	56.5	5.2	0.0	0.0	0.0
Data Masked	1.0E+05	31	4.3	0.0	23.9	113.0	0.0	313.0	56.5	5.2	0.0	0.0	0.0
Data Masked	1.0E+05	31	4.3	0.0	23.9	113.0	0.0	313.0	56.5	5.2	0.0	0.0	0.0
Data Masked	1.0E+05	31	4.3	0.0	23.9	113.0	0.0	313.0	56.5	5.2	0.0	0.0	0.0
Data Masked	1.0E+05	31	4.3	0.0	23.9	113.0	0.0	313.0	56.5	5.2	0.0	0.0	0.0
Data Masked	1.1E+05	34	4.8	0.0	26.3	124.3	0.0	344.3	62.2	5.7	0.0	0.0	0.0
Data Masked	1.1E+05	34	4.8	0.0	26.3	124.3	0.0	344.3	62.2	5.7	0.0	0.0	0.0
Data Masked	1.1E+05	34	4.8	0.0	26.3	124.3	0.0	344.3	62.2	5.7	0.0	0.0	0.0
Data Masked	1.1E+05	34	4.8	0.0	26.3	124.3	0.0	344.3	62.2	5.7	0.0	0.0	0.0
Data Masked	1.2E+05	37	5.2	0.0	28.7	135.7	0.0	375.7	67.8	6.3	0.0	0.0	0.0
Data Masked	1.2E+05	37	5.2	0.0	28.7	135.7	0.0	375.7	67.8	6.3	0.0	0.0	0.0
Data Masked	1.2E+05	37	5.2	0.0	28.7	135.7	0.0	375.7	67.8	6.3	0.0	0.0	0.0
Data Masked	1.2E+05	37	5.2	0.0	28.7	135.7	0.0	375.7	67.8	6.3	0.0	0.0	0.0
Data Masked	1.5E+05	46	6.5	0.0	35.9	169.6	0.0	469.6	84.8	7.8	0.0	0.0	0.0
Data Masked	1.5E+05	46	6.5	0.0	35.9	169.6	0.0	469.6	84.8	7.8	0.0	0.0	0.0

E.4.3.5. Approach to Estimates

Intake and dose were not estimated for individuals in the Remaining Cases Group because sample contamination from on-site collection was suspected and because the sample data contained uncertainties about exposure dates and recorded sample collection dates. However, the lowest and the highest urine results of 0 and 237.9 pCi/d of gross alpha radioactivity were input to CINDY, and produced estimated intakes of 75,000 pCi to 20,000,000 pCi corresponding

to CEDEs of about 23 rem to 6,000 rem (0.23 to 60 Sv). Results of this magnitude are clearly unrealistic, not supported by the air concentrations observed at Palomares and require careful evaluation.

E.4.3.6. Results

A range of estimates for the Remaining Cases Group showed that the intakes could range from 75,000 pCi to 20,000,000 pCi with CEDEs of 23 rem to 6,000 rem (0.23 to 60 Sv). The upper end of the range represents very substantial exposures that should not be attributed to any individual without follow-up sampling to provide confirmation of the results. Additional efforts could be made to determine more details about the specific dates of assignment and duties of the individuals. These estimates indicate the possible difficulties that may be encountered when samples, contaminated from collected on site, are analyzed. Unfortunately, the possibility of contamination prevents useful evaluation of these data, especially without the benefit of follow-up samples.

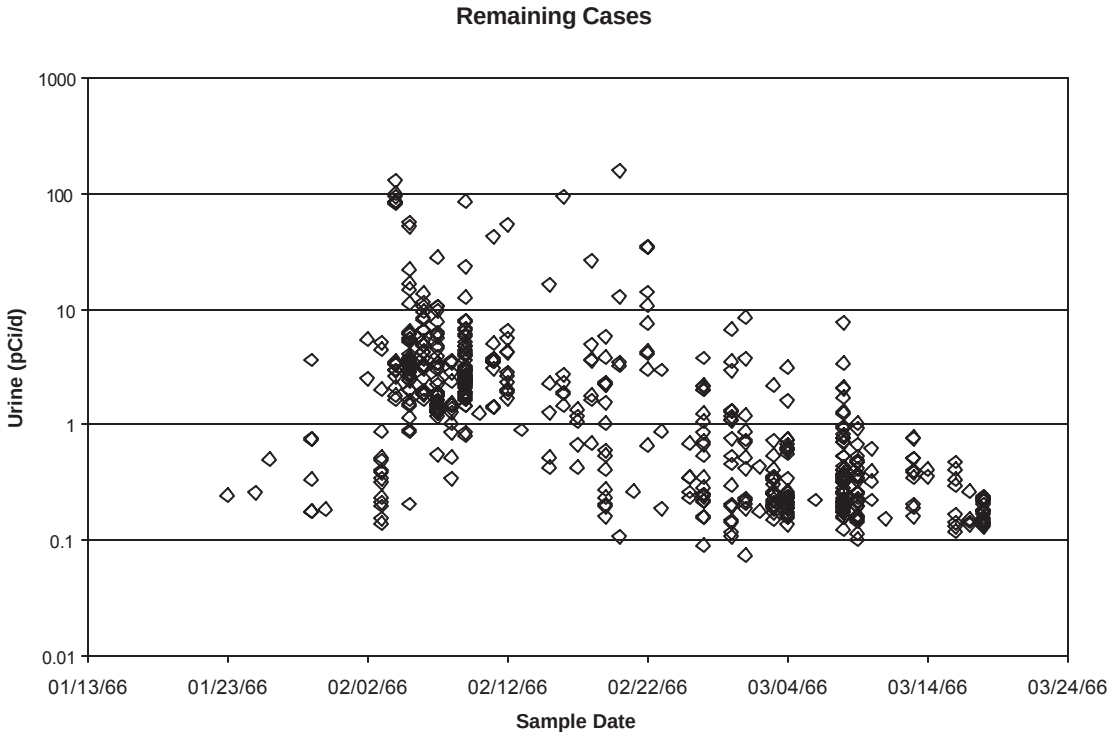

Figure E- 9. Urine results for the Remaining Cases Group.

As a final note, Figure E-9 shows a decreasing trend for the sample results. If resampling had been extended beyond the end of March 1966 as for some other groups, there is ample reason to expect that urinary excretion for this group would have followed similar patterns. Consequently, there are no more reasons to believe that this group received unusual exposures than the other groups. However, the data are simply not available to confirm the status of the individuals in this group. Therefore, follow-up sampling now for selected members of this group could provide information for re-evaluation of the possible exposures.