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ACRONYMS AND ABBREVIATIONS

AEC	U.S. Atomic Energy Commission
AERD	Atomic Energy Research Department
AETR	Advanced Epithermal Thorium Reactor
AI	Atomics International
AMAD	activity median aerodynamic diameter
ATR	Advanced Test Reactor
Bq	becquerel
CF	critical or criticality facility
cm	centimeter
CMD	count median diameter
CTF	critical or criticality test facility
D&D	Decontamination and Decommissioning
DOE	U.S. Department of Energy
dpm	disintegrations per minute
DU	depleted uranium
EEOICPA	Energy Employees Occupational Illness Compensation Program Act of 2000
ETEC	Energy Technology Engineering Center
ETR	Engineering Test Reactor
EU	enriched uranium
F	fast (solubility rate)
FSCF	(SNAP) flight system critical facility
g	gram
GIF	Gamma Irradiation Facility
HEU	highly enriched uranium
HMRFSR	Heavy Metal Reflected Fast Spectrum Reactor
hr	hour
in.	inch
IREP	Interactive RadioEpidemiological Program
IVLC	*in-vivo* lung count
JAERI	Japan Atomic Energy Research Institute
keV	kiloelectron-volt
KEWB	Kinetics Experiment Water Boiler Reactor
kg	kilogram
L	liter
L-85	Alternate name for the WBNS after 1972
LLD	lower limit of detection – the lowest value where radioactivity was considered to be present with reasonable certainty

M	moderate (solubility rate)
MDA	minimum detectable amount
MFP	mixed fission product
mg	milligram
min	minute
mL	milliliter
MPBB	Maximum Permissible Body Burden
MPC	Maximum Permissible Concentration
MPLB	Maximum Permissible Lung Burden
NAA	North American Aviation, Inc.
nCi	nanocurie
NIOSH	National Institute for Occupational Safety and Health
NMDF	Nuclear Materials Development Facility
NRTS	National Reactor Testing Station
NSEC	Nuclear Science and Engineering Corporation
OCY	Old Conservation Yard
OMR	organic moderated reactor
OMRE	Organic Moderated Reactor Experiment
ORAU	Oak Ridge Associated Universities
ORNL	Oak Ridge National Laboratory
oz	ounce
pCi	picocurie
PUA	plutonium activity determined by autoradiography
PUB	plutonium activity determined using a proportional counter to count alpha particles
R&D	Research and Development
RMHF	Radiation Materials Handling Facility
RSRMS	Radiation Safety Records Management System
S	slow (solubility rate)
S10FS	SNAP 10 Flight Simulation Reactor
S2DR	SNAP 2 Development Reactor
S8DR	SNAP 8 Development Reactor
S8ER	SNAP 8 Experimental Reactor
SER	SNAP Experimental Reactor
SGR	Sodium Graphite Reactor
SNAP	Systems for Nuclear Auxiliary Power
SRE	Sodium Reactor Experiment
SSFL	Santa Susana Field Laboratory
STIR	Shield Test and Irradiation Reactor (Modified STR)
STR	Shield Test Reactor
TBC	total body count
TBD	technical basis document
TLD	thermoluminescent dosimeter
TTA	thenoyltrifluoroacetone
UAlx	uranium aluminide

UCLA	University of California, Los Angeles
UF	Uranium Fluorometric
UR	Uranium Radiometric
U.S.C.	United States Code
UST	U. S. Testing
W	watt
WBC	whole-body count
WBNS	Water Boiler Neutron Source
ZnS(Ag)	zinc sulfide scintillation crystal activated with silver
µg	microgram
µL	microliter

5.1 INTRODUCTION

Technical basis documents (TBDs) and site profile documents are general working documents that provide guidance concerning the preparation of dose reconstructions at particular sites or categories of sites. They will be revised in the event additional relevant information is obtained about the affected site(s). These documents may be used to assist the National Institute for Occupational Safety and Health (NIOSH) in the completion of the individual work required for each dose reconstruction.

In this document the word "facility" is used as a general term for an area, building, or group of buildings that served a specific purpose at a site. It does not necessarily connote an "atomic weapons employer facility" or a "Department of Energy [DOE] facility" as defined in the Energy Employees Occupational Illness Compensation Program Act [EEOICPA; 42 U.S.C. § 7384l(5) and (12)]. EEOICPA defines a DOE facility as "any building, structure, or premise, including the grounds upon which such building, structure, or premise is located … in which operations are, or have been, conducted by, or on behalf of, the Department of Energy (except for buildings, structures, premises, grounds, or operations … pertaining to the Naval Nuclear Propulsion Program)" [42 U.S.C. § 7384l(12)]. Accordingly, except for the exclusion for the Naval Nuclear Propulsion Program noted above, any facility that performs or performed DOE operations of any nature whatsoever is a DOE facility encompassed by EEOICPA.

For employees of DOE or its contractors with cancer, the DOE facility definition only determines eligibility for a dose reconstruction, which is a prerequisite to a compensation decision (except for members of the Special Exposure Cohort). The compensation decision for cancer claimants is based on a section of the statute entitled "Exposure in the Performance of Duty." That provision [42 U.S.C. § 7384n(b)] says that an individual with cancer "shall be determined to have sustained that cancer in the performance of duty for purposes of the compensation program if, and only if, the cancer … was at least as likely as not related to employment at the facility [where the employee worked], as determined in accordance with the [probability of causation] guidelines established under subsection (c) …." [42 U.S.C. § 7384n(b)]. Neither the statute nor the probability of causation guidelines (nor the dose reconstruction regulation) define "performance of duty" for DOE employees with a covered cancer or restrict the "duty" to nuclear weapons work.

As noted above, the statute includes a definition of a DOE facility that excludes "buildings, structures, premises, grounds, or operations covered by Executive Order No. 12344, dated February 1, 1982 (U.S.C. 7158 note), pertaining to the Naval Nuclear Propulsion Program" [42 U.S.C. § 7384l(12)]. While this definition contains an exclusion with respect to the Naval Nuclear Propulsion Program, the section of EEOICPA that deals with the compensation decision for covered employees with cancer [i.e., 42 U.S.C. § 7384n(b), entitled "Exposure in the Performance of Duty"] does not contain such an exclusion. Therefore, the statute requires NIOSH to include all radiation exposures in its dose reconstructions for employees at DOE facilities, including radiation exposures related to the Naval Nuclear Propulsion Program. As a result, all internal and external dosimetry results are considered valid for use in dose reconstruction. No efforts are made to determine the eligibility of any fraction of total measured exposure for inclusion in dose reconstruction.

5.1.1 Purpose

The purpose of this TBD is to describe internal dosimetry systems and practices at the Energy Technology Engineering Center (ETEC). This document discusses historical and current practices in relation to the evaluation of internal exposure data for monitored and unmonitored workers.

5.1.2 Scope

ETEC, operated by The Boeing Company and its predecessors, has played an important role in the development of the U.S. nuclear program. ETEC operations have involved research and development (R&D) in the areas of development and testing of nuclear reactors, nuclear support operations, and non-nuclear energy R&D (Sapere and Boeing 2005, p. 2-2). This TBD is part of the overall ETEC Site Profile, which describes plant facilities and processes, historical information, and environmental data in relation to dose reconstruction for ETEC workers. It contains supporting documentation to assist in the evaluation of occupational internal doses from these processes in accordance with the *Internal Dose Reconstruction Implementation Guideline* (NIOSH 2002).

The methods and concepts of measuring occupational internal doses to workers have evolved since the beginning of ETEC operations. An objective of this document is to provide supporting technical data to evaluate internal ETEC occupational doses that can reasonably be associated with worker radiation exposures covered by the EEOICPA legislation.

In addition, this document presents the technical basis of methods used to prepare ETEC worker dose information for input to the NIOSH IREP (Interactive RadioEpidemiological Program) computer code. Information on measurement uncertainties is an integral component of the approach. This document describes the evaluation of uncertainty for ETEC exposure and dose records.

5.2 INTERNAL DOSIMETRY OVERVIEW

Section 2 of this Site Profile (ORAU 2005) contains some of the following discussion of historical activities. However, this TBD discusses the information in the context of internal dosimetry.

From 1948 to 1955, the Atomic Energy Research Department (AERD) of North American Aviation, Inc. (NAA), occupied a portion of Building 001 at the Downey Plant on Lakewood Boulevard in the City of Downey, California. On April 21, 1952, AERD constructed a small aqueous homogeneous reactor in an area that is now a loading dock. The reactor and its associated “exponential pile” were apparently used as a source of neutrons for reactor physics experiments. The Water Boiler Neutron Source (WBNS) used a 93%-enriched uranyl sulfate solution and operated at power levels up to 4 W. The WBNS operated at Downey until December 1955. In 1956, it was dismantled and moved to a facility at the Santa Susana Field Laboratory (SSFL). Little information is available about the specific uses of the area after the WBNS relocation. All available records indicate that the Plant was not left in a contaminated condition, and this was confirmed by a survey in 2000 (Liddy and Rutherford 2001). Other radioisotopes at the Downey Plant were apparently used for industrial radiography and were not a likely source of internal exposure.

In 1956, NAA formed Atomics International (AI) as one of its divisions, replacing AERD. Between 1956 and 1960, AI performed nuclear R&D at a facility known as the Vanowen Building (Building 038) on Canoga Avenue in Canoga Park, California. AI designed, developed, and operated two small, aqueous, 93%-enriched uranyl sulfate research reactors, designated L-47 and L-77, at the Vanowen facility. The maximum power ratings were 5 W and 10 W, respectively. Other operations included reactor design, fuel development, and radiochemistry. L-47 operated between August 1957 and June 1958, and L-77 operated between May 1958 and February 1960. The reactors were in the southeast corner of the building at what is now a loading dock. Historical information indicates that the areas that supported nuclear operations were routinely surveyed up to and following the removal of all radioactive material. No contamination above the current limits for unrestricted use was found. This was confirmed by a 1995 survey of the facility by the U.S. Nuclear Regulatory Commission

(Rutherford 2002). Exposure of subsequent occupants to contamination levels of importance for internal dose reconstruction of the building would be unlikely.

In 1959, AI moved to a new facility on De Soto Avenue in Canoga Park. All radiological activities from the Vanowen Building, including the L-77 reactor, transferred to the new facility. Until 1983, nuclear operations were conducted at the De Soto facility in Buildings 101 and 104 (until 1984 these buildings were known as 001 and 004, respectively). Work continued at a "much-reduced level" in Building 104 until the mid-1990s. The L-77 reactor, housed in Room 416-61 of Building 104, was decommissioned and decontaminated in the late 1970s. The facility was released for unrestricted use in February 1982. Many nuclear fuel manufacturing operations were conducted in the 1960s and 1970s in Building 101 and Building 104. These operations used 2%- to 93%-enriched uranium metal and composites. Most notable of these was fuel for the Advanced Test Reactor (ATR), which utilized uranium-aluminum alloy plates. This operation was the source of internal exposures that were difficult to evaluate because of the unanticipated insolubility and small particle size (when airborne) of the alloy. Attachment A summarizes these problems.

Two other operations in Building 104 at the De Soto facility utilized radioactive material. A Gamma Irradiation Facility (GIF) used sealed ^{137}Cs and ^{60}Co sources for hardness testing and food irradiation research. Biannual leak tests of these sources confirmed that internal exposures from these devices were unlikely. Operations ended in the late 1980s and, following the removal of all equipment, the State of California released the GIF for unrestricted use in July 1999. The other Building 104 operation, the Mass Spectroscopy Laboratory or "Helium Laboratory," analyzed radioactive samples for helium content. Operations ended in 1995; the mass spectroscopy equipment was sent to Richland, Washington. In 1998, all remaining support equipment was removed and the facility was decontaminated. In October 1998, the State released the Helium Laboratory for unrestricted use. Thirteen separate radiation surveys in Buildings 101 and 104 demonstrated that no residual contamination exists that would be of interest for dose reconstruction (Boeing 2003).

SSFL consisted of four administrative areas and a buffer zone. NAA established Area IV in 1953 as a nuclear R&D facility. Starting in 1954, several nuclear reactors and critical assemblies were built and operated in Area IV. In 1959, a significant incident occurred when a sodium-cooled, graphite-moderated reactor in Building 4143 (the SRE or Sodium Reactor Experiment) had a loss of coolant, which resulted in damage to 13 fuel assemblies (Sapere and Boeing 2005, pp. 2-1, 2-3, 2-5). A government-owned, contractor-operated organization was formed to conduct non-nuclear research at the site, which was renamed ETEC in 1978. Most nuclear research programs and operations ended in 1988. All research ended in 1998. ETEC was given the job of decontaminating and decommissioning the former nuclear facilities. Many of the facilities in Area IV have been decontaminated and decommissioned, but some of this work is still ongoing.

Early 1960s AI documents describe all the elements of a comprehensive radiation safety program, including a laboratory with bioassay capability. The program was under central direction and covered the facilities operated by AI (Lang undated, 1960). There is one possible exception. The Organic Moderated Reactor Experiment (OMRE) was an AI facility at the National Reactor Testing Station (NRTS) in Arco, Idaho. NRTS apparently provided dosimetry services to AI personnel (Lang 1960). It is uncertain, based on current information, if bioassay results, if any, were sent to AI.

AI established health and safety files on each employee that contained radiation exposure records, injury records, and other "pertinent" data (Lang undated, 1960). Today, personnel radiation exposure records are in the Radiation Safety Records Management System (RSRMS), which encompasses about 170 file cabinets. In addition, the RSRMS includes records of routine surveys and environmental monitoring data (Sapere and Boeing 2005, p. 3-12).

The environmental monitoring program at Area IV was established in May 1954 prior to construction of the first radiological facility. Stack air emissions were measured in all facilities with radiological work areas or where unencapsulated or unpackaged radioactive material was handled. Gross alpha and beta activity was monitored on a weekly basis (Sapere and Boeing 2005, p. 3-13). From 1959 to the present, ambient gross beta activity in air has been measured continuously in five locations. From 1963 on, gross alpha activity was also measured. At present, ambient air samples are analyzed for isotopic content (Sapere and Boeing 2005, p. 3-15).

For simplicity, the remainder of this TBD uses the term *ETEC* to refer to all the organizations that operated at De Soto, Canoga Park, Downey, and SSFL, although technically ETEC did not exist for much of this time.

By 1959, routine urine samples were requested on Fridays, and each employee was required to submit the first voiding on Monday morning (following an absence from work of 48 hr or more). The time of the previous voiding was recorded to determine the excretion rate. If the Monday morning sample was verified as positive, a series of 24-hr samples was collected to determine the body burden. Employees were requested to fill these samples on Sunday (Kellehar 1959). Appropriate adjustments to this schedule were made for weekend work, etc. Uranium urinalyses were performed for fuel fabrication workers and gross beta or mixed fission product urinalyses were performed for many workers at SSFL and Vanowen. Commercial laboratories were available for analysis of alpha emitters other than uranium and for soft beta emitters (Alexander 1959). By 1966, most routine samples were analyzed by a vendor. Special samples were requested when an intake was suspected or if a routine sample exceeded 10% of the urinary excretion expected from one Maximum Permissible Body Burden (MPBB). Special samples were analyzed by the onsite laboratory, the vendor, or both (Kellehar 1966a).

Entry into the bioassay program at ETEC was apparently based on job assignment. By the early 1960s, the bioassay program "normally" consisted of urinalysis for personnel whose work assignments involved "potential exposure to radioactive materials." The frequency of sampling varied from one to four per year, depending on the nature of the employee's work, past exposure history, etc. (Lang undated). In 1963, a procedure called for completion of a "Request for Film Badge and Bioassay Services" form for new employees who required such services (Garcia 1963). Figure B-1 in Attachment B is an example. The Health and Safety Operation Unit completed the bottom portion, which included a place to specify bioassay for the employee.

In 1970, standards for bioassay sampling were published (Staszesky 1970). Work in areas where unencapsulated radioactive material was present required baseline and termination urine samples. A new baseline could be required for a change in job assignment. For new operations, a "pilot" bioassay program consisting of weekly urine samples could be required until a pattern was established. Regular work in these areas required a quarterly routine urine sample, but monthly samples could be required in a case of high exposure potential. Periodic fecal samples and *in vivo* counts could also be required. Employees who performed work in these areas only periodically were subject to semiannual urine samples. Personnel such as project engineers, industrial engineers, etc., who frequently entered these areas but did not perform hands-on work provided annual routine urine samples.

By the mid-1970s, the definition of who was included in the routine bioassay monitoring program had changed to "personnel whose work assignments potentially expose them to respirable-sized radioactive aerosols" (Hart 1979). By the late 1980s, the criterion was "personnel whose work assignments potentially expose them to radioactive aerosols" (Tuttle 1989). Quarterly urine sampling

was the norm through the 1980s (Hart 1979; 1980a,b,c; Eggleston 1983, 1984; Tuttle 1985, 1986a,b,c, 1988a,b, 1989).

Special bioassay consisting of more frequent urine sampling was in place very early (1960), and in the mid-1970s fecal sampling was also used but "only when gross internal contamination" was suspected (Hart 1979). Using the concept of an MPBB, an excretion rate was determined by radionuclide that would indicate that one MPBB had been received. For several years prior to 1968, the policy was to restrict employees from work in potential airborne areas until their body burden was less than 25% of the MPBB (Alexander 1968a). Starting in January 1968, ETEC imposed a restriction from work in areas with potential airborne exposure (or in some cases from all radiation areas) if the bioassay results indicated the receipt of 50% or more of the MPBB. The restriction remained in place until two consecutive bioassay samples indicated that the remaining deposition was less than 25% of the MPBB (Staszesky 1970).

It appears that the first *in vivo* counting occurred in 1967 at a U.S. Atomic Energy Commission (AEC)-funded facility at the University of California, Los Angeles (UCLA). These counts were performed in conjunction with the insoluble enriched uranium (EU) intakes described in Attachment A. Although these counts were frequently referred to as whole-body counts (WBCs) or total body counts (TBCs) in site documents, they were really attempts to quantify the amount of ^{235}U activity in the lung. At that time, chest (lung) counting was in its infancy and standard calibration methods using phantoms had not been developed. ETEC ultimately determined that the results of these counts were significant overestimates. From 1968 to 1983, *in vivo* counting was done on the site using portable counters operated by Helgeson Nuclear Services, Inc. Most of these counts were also chest counts for EU. Work with unirradiated highly enriched uranium (HEU) ended in 1983, and no more chest counts occurred (Tuttle 1986b). WBCs for fission and activation products apparently occurred sporadically during this period, probably in conjunction with suspected exposures or new projects.

Even though ETEC ended research activities with radioactive materials in 1988, the internal dosimetry program continued; most of the features of the program are listed above. The ETEC bioassay program is apparently not accredited by the DOE Laboratory Accreditation Program.

The bioassay records in the individual files generally consist of:

- Individual Personnel Keysort Cards (Figure B-2, Attachment B), which were used to track the type, frequency, and week of sample collection. In addition, the form summarized individual results. The forms can be difficult to read due to the quality of the copies, and dose reconstructors should refer to the forms listed below for urine and fecal data. This form might be the only place *in vivo* data are listed. Two versions were produced – 1963–1966 and 1967–1970. The latter continued in use at least until 1983. This form was the successor to one that recorded only results prior to 1963 (example not included).

- Bioassay Data Sheets (Figures B-3 and B-4, Attachment B), which were two-page forms that contained all the information on analyses conducted by the onsite laboratory. Some, if not most, offsite analysis results were apparently transferred to this form.

- Individual Analysis Results Sheets (Figure B-5 to B-10, Attachment B), which contained all the information on analyses conducted by offsite laboratories. This form might show the calculated result (even if it was below the limit of detection), the sensitivity of the analysis, and the uncertainty.

- Wound Monitoring Report (Figure B-11, Attachment B). The entries on this form are self-explanatory.

- Individual hand-drawn or computer-generated plots of bioassay data, apparently done as follow-up to high results. Some can be difficult to read due to the quality of the copies.

- Incident workups following high bioassay results. These might contain data summaries in a more legible and understandable form than information in other locations because they were produced by health physicists familiar with the data.

5.3 *IN VITRO* MINIMUM DETECTABLE ACTIVITIES, COUNTING METHODS, AND REPORTING PROTOCOLS

5.3.1 *In Vitro* Urine Analysis

5.3.1.1 1948 – 1957

Apparently, no bioassay program existed before August 1958. Useful source-term data or facility-specific ambient airborne concentrations are probably not available. This unmonitored period includes the operation of the WBNS from 1948 to 1955 at the Downey Plant, the operation of the L-47 reactor at the Vanowen facility, and operations at SSFL from 1954 to 1957, including for example the operation of the WBNS in 1956 and 1957. Documents indicating environmental releases or incidents from these facilities have not been located. Bioassay data after 1957 in similar facilities, such as the WBNS or L-77 Reactor, might be the only data available to the dose reconstructor as an indicator of missed dose. Because this data would be from a different period, use it with caution.

5.3.1.2 1958 – 1966

According to Kellehar (1966) the bioassay program was initiated in August 1958. As mentioned above, routine bioassay was normally by urinalysis at a frequency of 3 months. At first, gross alpha or gross beta measurements were made of the samples. Specific radionuclides could be determined "where required" (Lang 1960). Some detail has been found on early urinalysis methods. In addition to the in-house laboratory capability, bioassay services were contracted to the following vendors:

- Tracerlab, Richmond, California
- Nuclear Science and Engineering Corporation (NSEC), Pittsburgh, Pennsylvania
- U. S. Nuclear, Burbank, California
- Controls for Radiation, Cambridge, Massachusetts
- Biomedical Procedures, North Hollywood, California
- BioScience Laboratories, Santa Monica, California
- Atomic Corporation of America, Panorama City, California
- U. S. Testing (UST), Richland, Washington

Information on the periods during which ETEC used these laboratories was not found. The name of the laboratory responsible for the result appears in at least some early bioassay records. Due to various problems with the other labs (Fisher 1963), it appears that BioScience Laboratories analyzed the most samples early in this period. In 1964, ETEC initiated a contract with UST, which became International Technology Analytical Services, Quanterra Environmental Services, and finally Severn Trent Laboratories. From 1965 on, UST appears to have been the main laboratory vendor. Only information on uranium analyses was found for the in-house lab.

AI Uranium Methods [A and B (fluorometric) and enriched uranium (radiometric)]
The use of "A" and "B" to designate uranium bioassay methods apparently changed over the years. In general, it should be easy to distinguish fluorometric methods if the units were reported; fluorometric results were reported in micrograms or milligrams per unit volume rather than a unit of radioactivity per unit volume. The fluorometric methods fused uranium from raw urine with sodium fluoride and measured the fluorescence when the compound was exposed to ultraviolet light. The uranium fluorescence can be quenched by certain heavy metal ions, and the primary difference between the two methods was that Method A considered these quenching effects negligible. Method B took into account possible quenching effects (Mason and Burr 1958).

Due to its higher specific activity, EU activity could be determined by counting. The procedure involved digesting the urine, removing some of the interfering cations, and electroplating the uranium onto nickel disks. The disks were counted in a proportional counter (Mason and Burr 1958).

No specific information on sensitivities for the in-house laboratory was obtained. The values for the contractor laboratories which follow should be used for dose reconstruction since they should be comparable to the in-house laboratory.

Vendor Uranium Methods [codes: 1A (fluorometric) and 1B (radiometric)]
The fluorometric method was implemented during the first half of 1948 at AEC Sites (ORAU 2004, p. 24). In May 1959, U. S. Nuclear quoted a price for fluorometric analysis with a minimum measurable concentration of 0.002 µg/mL based on a minimum volume of 1 L (Shepard 1959). Controls for Radiation, Inc., quoted 0.0001 µg/mL based on a minimum volume of 150 mL (O'Brien 1959). NSEC gave its minimum measurable concentration as 0.0002 µg/mL (Edelmann 1959).

The early radiometric methods generally used separation chemistry followed by counting on a gas-flow proportional counter or a ZnS(Ag) scintillation counter (ORAU 2004, p. 25). Shepard (1959) quoted a minimum measurable concentration of 7.5 dpm/L for radiometric determination of enriched uranium. NSEC gave its minimum measurable concentration for enriched uranium as 2 dpm/24-hr sample (NSEC 1957).

Gross Alpha
Shepard (1959) gave a minimum measurable concentration of 7.5 dpm/L for gross alpha counting. NSEC gave its minimum measurable concentration as 0.2 cpm/mL (NSEC 1957). It is assumed that this is a typographical error and 0.2 dpm/mL was intended.

Gross Beta
Shepard (1959) gave a minimum measurable concentration of 75 dpm/L for gross beta counting. NSEC gave its minimum measurable concentration as 1.0 cpm/mL (NSEC 1957). It is assumed that this is a typographical error and 1.0 dpm/mL was intended.

Mixed Fission Products
The early procedure for mixed fission products was generally to add strontium carrier to the aluminum oxide solution for the plutonium procedure, then precipitate lanthanum hydroxide. This procedure extracted rare earths and strontium with yields ranging from 90% for cerium to 23% for strontium. The dried planchet was counted for beta activity with an approximate minimum detectable amount (MDA) of 60 dpm/sample (ORAU 2004, p. 27). This procedure separated and counted radionuclides of alkaline earths and rare earths such as strontium, yttrium, barium, lanthanum, cerium, europium, and promethium. It might not have accounted for radionuclides of ruthenium, cesium, zinc, cobalt, manganese, niobium, or zirconium (ORAU 2004, p. 27). NSEC gave its minimum measurable concentration as 2.0 cpm/mL (NSEC 1957). It is assumed that this is a typographical error and

2.0 dpm/mL was intended. This is the same value NSEC quoted for gross gamma measurements, so the technique might not have used chemical separations.

Polonium-210
BioScience used electrodeposition on a nickel disk and counting in a low-background proportional counter. The sensitivity quoted was 0.001 dpm/mL (Lee 1963).

Plutonium
One of the first plutonium bioassay analysis consisted of lanthanum fluoride precipitation and thenoyltrifluoroacetone (TTA) extraction and gross alpha counting. Electrodeposition on a stainless-steel disk was followed by counting with nuclear track emulsion (autoradiography). Plutonium bioassay at ETEC apparently started around October 1966 (Kellehar 1966a). The urine maximum permissible concentration (MPC) that indicated 1 MPBB was listed as 4 dpm/24 hr. Therefore, in keeping with normal practice at the time, 0.4 dpm/24 hr would be the sensitivity required. BioScience quoted a sensitivity for plutonium bioassay of 0.00006 dpm/mL, but stated that it would have to subcontract the analyses (Lee 1963).

Strontium-90
BioScience coprecipitated ^{90}Sr and ^{90}Y with calcium as oxalates. Yttrium-90 was isolated and purified using a "milking" procedure with tributylphosphate, assayed, and reported as ^{90}Sr equivalent. The method assumed that ^{90}Sr and ^{90}Y were in secular equilibrium. If not, counting could be delayed for ingrowth. The sensitivity quoted was 0.1 dpm/mL (Lee 1963).

Tritium
Tritium was not a significant personnel exposure hazard from research reactor operations, primarily due to the low power levels. However, 15 kCi of tritium in a triple-walled container was in Room 416-51 in Building 004 at the De Soto Facility until November 1967 (Alexander 1967a). The earliest viable method for tritium analysis seems to have been liquid scintillation counting of raw urine in a scintillation cocktail. NSEC gave its minimum measurable concentration as 2,220 cpm/mL (NSEC 1957). It is assumed that this is a typographical error and 2,220 dpm/mL was intended.

Thorium
No details of early thorium analyses were recovered. Tracerlab analyzed three urine samples for AI in 1959. The detection limit was 0.2 µg/125 mL sample (Tracerlab 1959).

5.3.1.3 1967 – 1974

Partial documentation on bioassay methods from 1967 through 1974 was found. The sources of these data were mainly statements of work for bioassay services. These documents are believed to refer to services offered by UST (Kellehar 1966b, 1967, Spielman 1968, Bales 1969, Staszeski 1971). It is uncertain if these were the only bioassay services contracted during this period, but the information should be usable for dose reconstructions. Many of the methods listed in this TBD were undoubtedly in place prior to 1967.

Uranium Fluorometric (Procedure A)
The sample was acidified. A 100-µL aliquot was analyzed directly. Recovery was 93% ±18%. The detection limit was 0.5 µg/L. A fluorophotometer was used to measure the uranium.

Uranium Fluorometric (Procedure B)
Uranium was extracted from the ashed residual salts with methyl isobutyl ketone using a salting solution of acid lammonium hydroxide. A fluorophotometer was used to measure the uranium present. Recovery was 83% ±8%. The detection limit was 0.05 µg/L.

Uranium Radiometric
Uranium was isolated as in Procedure B above. It was measured by a gas-flow proportional counter or a ZnS(Ag) scintillation counter. Recovery was 83% ±8%. The detection limit was 0.5 dpm/sample. At ETEC the standard sample volume per day was 1,500 mL. The result was the total alpha activity.

Plutonium (Procedure A)
Plutonium is isolated by coprecipitation as the fluoride, extraction with TTA, and electrodeposition. Autoradiography was used to detect the plutonium. Recovery was 82% ±14%. The detection limit was 0.05 dpm/sample. At ETEC the standard sample volume per day was 1,500 mL. Plutonium results would have included activity from ^{238}Pu, ^{239}Pu, and ^{240}Pu, but not ^{241}Pu or ^{241}Am.

Plutonium (Procedure B)
Plutonium was isolated as in Procedure A, and counted in a gas-flow proportional counter. Recovery was 85% ±10%. The detection limit was 0.5 dpm/sample. At ETEC the standard sample volume per day was 1,500 mL. Plutonium results would have included activity from ^{238}Pu, ^{239}Pu, and ^{240}Pu, but not ^{241}Pu or ^{241}Am.

Tritium
Tritium was determined by liquid scintillation counting of an aliquot of the sample. Recovery was 100% ±10%. The detection limit was 5,000 dpm/mL.

Polonium
Polonium was spontaneously deposited on a silver disk from a dilute hydrochloric acid solution of the residual salts. The disk was alpha-counted. Recovery was 93% ±7%. The detection limit was 0.5 dpm/sample.

Strontium
Strontium was isolated by precipitation as the oxalate, and then as the nitrate. Yttrium was removed by a nitric acid wash. Barium was removed as the chromate. Strontium was precipitated as the carbonate and counted with a gas-flow proportional counter. This provided a gross strontium result. Recovery was 85% ±9%. The detection limit was 4 dpm/sample. After the in-growth of ^{90}Y (in 2 weeks), the ^{90}Y was isolated as the hydroxide, and then as the oxalate, which was burned to the oxide and counted in a gas-flow proportional counter. The ^{90}Sr was computed from the ^{90}Y, which was in secular equilibrium at separation time. Recovery was 78% ±11%. The detection limit was 4 dpm/ sample.

Thorium
Thorium was isolated by a double fluoride precipitation and extraction with TTA, followed by spectrophotometric determination with morin. Recovery was 78% ±12%. The detection limit was 1.0 µg for a 1,000-mL sample. Thorium-232 was determined by planchetting the thorium fraction and alpha-counting. The detection limit was 0.5 dpm/sample.

Phosphorus-32
Phosphorus-32 was separated first as the phosphomolybdate and then as the magnesium ammonium phosphate, which was planchetted and beta-counted in a gas-flow proportional counter. Recovery was 86% ±7%. The detection limit was 4 dpm/sample.

Sulfur-35
The sample was ashed to drive off tritium and carbon, and counted by liquid scintillation techniques. Recovery was 95% ± 10%. The detection limit was 10 dpm per sample for a 1-mL sample aliquot.

Carbon-14
Carbon-14 was determined by liquid scintillation techniques. Recovery was 65% ±15%. The detection limit was 10 dpm for a 1-mL sample aliquot.

Gross Alpha
Gross alpha was determined by extracting most actinides from a 9N nitric acid solution into diethyl ether. This provided recoveries from 80% to 99% of most actinides. The detection limit was 1.0 dpm/ sample.

Gross Beta
Gross beta was determined from a beta count of the ashed residual salts. A ^{40}K correction could be made. Recovery was 95% ±5%. The detection limit was 2 dpm for a 50-mL sample.

Promethium-147
Promethium-147 was chemically isolated by precipitation as the fluoride and extraction into TTA, and counted by liquid scintillation techniques. Recovery was 83% ±10%. The detection limit was 5 dpm/sample. In 1967, for at least one case of suspected promethium exposure (^{147}Pm-oxide), bioassay analyses were performed on urine and fecal samples (Alexander 1967a).

Americium and Curium
Americium and curium were isolated from contaminating actinides by HDEHP in toluene extraction out of 4N HNO_3. Americium and curium were extracted into the HDEHP toluene solution at a pH of 4.5 and back-extracted using 4N HNO_3, electrodeposited, and counted using alpha spectroscopy. The recovery was 80% ±15% with a detection limit of 0.5 dpm/sample.

Gross Fission Products
Gross fission products were precipitated as the oxalate from a basic solution, planchetted, and beta-counted in a gas-flow proportional counter. Recovery for all fission products averaged 82% ±5%. The detection limit was 5 dpm/sample based on the counting efficiency of the radiologically critical isotopes ^{90}Sr and ^{90}Y. A “gamma scan” was utilized for routine detection of certain gamma-emitting radionuclides. The gamma scan was performed by counting a sample in the well of a NaI(Tl) crystal and feeding the pulses to a multichannel pulse-height analyzer. The simultaneous measurement of more than one radionuclide using gamma energy analysis precluded making accurate statements about detection limits and precision. The laboratory offered a table of detection limits per sample based on experience; this information appears in Attachment E. Lower detection limits than those in the table were available at extra cost. Other gamma-emitting radionuclides could be determined.

5.3.1.4 1975 – 1988

The following analytical methods were taken from a series of annual reports (Hart 1979, 1980a,b,c; Eggleston 1983, 1984; Tuttle 1985). The measurement “type” in parentheses appears in many personnel bioassay records. The detection limits should have improved over the years. However, a listing was not found. The reports list positive results for the year and, in some cases, follow-up results. The follow-up results were examined to see if the lowest value reported as non-zero could be determined. In general, there were not enough data to provide a satisfactory result. Therefore, dose reconstructors will have to use the detection limits in the previous section or, in some cases, on the individual results.

Uranium Radiometric and Fluorometric (UR, UF)
Uranium was extracted from an acidic solution of ashed urine using aluminum nitrate, tetrapropyl ammonium hydroxide, and methyl isobutyl ketone. The uranium was recovered by back-extracting into water by evaporating to ketone. The water solution was planchetted for alpha counting for the UR analysis. The result was the total alpha activity. Fluorometric analysis required removal of an appropriate aliquot of the water solution prior to planchetting for pelletizing with NaF-LiF. The pellet was analyzed for uranium with a fluorometer. Most uranium samples were apparently analyzed by using both techniques.

Mixed Fission Products (FP1)
Mixed fission products were precipitated from a basic oxalate media. By adjustment of pH and oxalate concentrations, elements that are amphoteric or that form oxalate complexes in the form of excess oxalate were also precipitated. Alkali metals such as ^{137}Cs did not precipitate. In addition, volatile fission products such as ^{131}I were lost. The precipitate was washed with NAOH and water and planchetted for counting.

Mixed Fission Products (FP2)
FP2 uses the same extraction procedure as FP1; however, the soluble oxalate precipitates were gamma-counted for ^{137}Cs and other gamma emitters. The results from the FP1 analysis and the FP2 analysis were summed and reported as a single value.

Mixed Fission Products (FP3)
This analysis is the same as FP2 except oxalate insoluble results were reported separately as FP3a and oxalate soluble results were reported separately as FP3b. The FP3a analysis was assumed to indicate ^{90}Sr, but other radionuclides, such as ^{60}Co, might also be detected. Further analysis was used to quantify ^{90}Sr specifically and identify interfering radionuclides if significant quantities occurred. The FP3b analysis was selective for ^{137}Cs, using gamma-ray spectroscopy (Tuttle 1988a).

Plutonium (PUA, PUB)
After reduction to plutonium (III) and (IV) with hydroxylamine hydrochloride, plutonium was precipitated with lanthanum fluoride. This isolated the plutonium from most elements, including uranium, except thorium, rare earths, and actinides. After oxidation of plutonium with sodium nitrate in acid media, extraction of plutonium was performed with 0.5 M TTA in xylene. Following extraction, the aqueous solution containing plutonium was neutralized and concentrated by heating. After oxidation of the plutonium in a basic media, it was electrodeposited on a stainless-steel disc. The plutonium activity was determined by autoradiography (PUA) for greater sensitivity or counted for alpha radiation with a proportional counter (PUB). Plutonium results would have included activity from ^{238}Pu, ^{239}Pu, and ^{240}Pu, but not ^{241}Pu. The PUA analysis was chemically selective for plutonium and excluded the ^{241}Am that is generally present (Tuttle 1986b).

Gross Beta, High Level (GBH)
The gross sample was evaporated to dryness, followed by organic digestion by hydrogen peroxide and nitric acid. Natural potassium (^{40}K) correction was determined by diluting the ashed salts to a known volume, and removing an aliquot for flame spectrophotometry. The remaining solution was evaporated to near dryness, planchetted, and counted for beta radiation with a proportional counter. The radioactivity in the urine sample due to ^{40}K was subtracted from the gross count.

Gross Alpha (GA1a)
This analysis was specific for uranium and/or plutonium, which were extracted from ashed urine salts using aluminum nitrate, tetrapropylammonium hydroxide, and methyl isobutyl ketone. Transuranics did not extract to any appreciable extent. Uranium and/or plutonium were recovered by back-

extracting into water by evaporating the ketone. The uranium or plutonium was electrodeposited on a stainless-steel disc and autoradiographed.

Gross Alpha (GA1b)
This was the same analysis as GA1a, except the extraction solution was planchetted and counted for alpha radiation with a proportional counter.

Gross Alpha (GA2)
This analysis was specific for all alpha emitters. Metabolized actinides were converted to states suitable for coprecipitation with alkaline earth phosphates by digesting the gross urine sample in 10% nitric acid. The actinides were coprecipitated with calcium phosphate by neutralizing the acid solution with ammonia. The precipitate was washed, planchetted, and counted for alpha radiation with a proportional counter.

5.3.1.5 1989 – 2005

By 1989, all R&D activities had ended. All work with radioactive materials has been in conjunction with ongoing decontamination and decommissioning activities. UST was apparently the bioassay contractor between 1989 and 1992. Then ETEC switched to Teledyne (now Teledyne Brown Engineering). No analytical procedures were recovered, but they were probably based on the earlier methods presented above. The collection methods, etc., appear to be very similar to those used in previous years. The routine sampling strategy was to use generic screening urinalyses (gross alpha and gross beta minus ^{40}K). If positive results were found, dose assessments were performed using default radionuclide mixtures or more specific analyses were performed (Barnes 1999). Periodic *in vivo* measurements were performed to detect intakes from gamma emitters likely to be present at ETEC (^{137}Cs, ^{60}Co, ^{154}Eu, etc.).

Particle size measurements were generally not performed in conjunction with routine workplace monitoring. Worker breathing zone air samples were collected (Barnes 1999), but it is unclear if this information was incorporated into an individual's dosimetry records.

Since 1992, bioassay results returned from the vendor have been entered into a computer database (Boeing Canoga Park Internal Dosimetry Tracking System). Results before 1992 are available only in hard copy (Barnes 1999). In general, the entire sample was not used for the requested analysis; therefore, multiple results for one sample are possible.

According to the health physicist responsible for internal dosimetry in 2005, no positive results have been reported since 1998. The sensitivities required for each analysis were published (Barnes 1999) and are listed in Table 5-6 in Section 5.6.

5.3.2 *In Vitro* Methods for Individual Radionuclides

The methods for individual radionuclides are covered for the specific periods under the sample types (i.e., urine and fecal).

5.3.3 Fecal Sample Analysis

Although fecal sampling was mentioned as both a routine and a special bioassay method in site documents, little detail has been found about the analytical methods used. A statement in the 1967 bioassay contract for UST indicated that all bioassay samples are wet ashed with nitric acid and

hydrogen peroxide. A salt fusion was added for fecal samples to ensure recovery of all radionuclides. An aliquot was apparently analyzed using methods similar to those discussed for urine sampling.

Detection limits for various isotopes for fecal samples were found for 1967. In the period from 1975 to 1988 when the number of total and positive tests was well-documented, only a few fecal samples were mentioned, but this might have been because this series of reports did not always make the distinction between fecal and urine, which was by far the more common bioassay medium.

In 1967, bioassay analyses were performed on fecal samples for a case of suspected promethium exposure (Alexander 1967b). Fecal samples for uranium were apparently taken in conjunction with the investigation into the uranium aluminide exposures in 1967 (see Attachment A). Tables of total samples and positive results for 1975 and 1976 show analysis "type" codes for uranium fecal samples (F-UF and F-UR), but apparently no such analyses occurred in those years (Hart 1979, 1980a). In 1982, positive PUA fecal results were reported for two individuals (one sample each) (Tuttle 1985).

Due to the lack of information found, the urine MDAs for gross alpha or gross beta listed in Table 5-3 in Section 5.6 should be converted to dpm/day and used for fecal analyses prior to 1967 if the sensitivity was not reported by the laboratory. From 1967 on, the values in Table 5-4 in Section 5.6 should provide claimant-favorable estimates because detection limits probably were lower with time.

Table 5-1 lists the frequencies for *in vitro* monitoring.

Table 5-1. Internal dose control program (*in vitro*).

Routine monitoring type	Period	Frequency
Urine, single void[a]	1958–1988	Quarterly, semiannual, annual, based on job
Urine, single void, H-3	1958–1966	Weekly, based on job

a. Sample requested on Friday, first voiding on Monday morning requested. Positive samples were verified and followed with at least one 24-hr sample collected on Sunday.

5.4 *IN VIVO* MINIMUM DETECTABLE ACTIVITIES, COUNTING METHODS, AND REPORTING PRACTICES

5.4.1 Whole-Body Counting

Although Helgeson Nuclear Services regularly visited the site to conduct chest counts for ^{235}U (see Section 5.3.2), whole-body counting for fission or activation products was apparently not part of the routine bioassay program at ETEC. Between 1975 and 1988, only 25 counts on 25 individuals were summarized in annual reports. In the same period, 385 chest counts were completed. All WBCs were reported positive for ^{137}Cs. Ten counts were performed in 1977 and 15 were performed in 1979. The counts were probably performed when intakes were suspected, but could have been project-ending counts. Although only ^{137}Cs was reported in the summaries, undoubtedly a wide spectrum was scanned that would have detected other high-energy gamma emitters, such as ^{60}Co, with a reasonably low detection level. Barnes (1999) stated that WBCs occurred "periodically."

5.4.2 Chest Counting

In 1967, the first chest (lung) counts for uranium using a medical system were performed at UCLA. The 186-keV gamma ray from the decay of ^{235}U was used to quantify the amount of EU in the lung by scintillation spectrometry (Tuttle 1986a). Calibration of this system was crude; it used a point source rather than a distributed source that would better simulate the radioactivity in a lung. The point source contained more ^{234}U than the uranium fuel, so the activity-to-mass conversion was off by about a factor of 2. Moreover, no chest-wall thickness corrections were made. The first two corrections

amounted to initial lung burdens that were about a factor of 4 too high. The correction for chest-wall thickness would have depended on each individual. This system was not used after 1967. It is uncertain to what extent corrections were made to individual employee records. Dose reconstructors could use these data for claimant-favorable overestimates. For more realistic calculations, it would be best to rely on urine and fecal data that would hopefully be available.

Starting in 1968, Helgeson Nuclear Services provided lung counting services using equipment and techniques specifically developed to measure lung deposits of uranium. The first counts in 1968 were done with 0.5-in. by 8-in. NaI(Tl) detector in the shadow shield used for whole-body counting. A crude masonite phantom was used for calibration, and included a chest-wall thickness correction. The individual was raised on an air mattress until the chest was flush with the bottom of the detector. The counts took 40 min. The results were reported in milligrams of ^{235}U ±2 sigma, with the uncertainty based on counting statistics alone (Helgeson 1968).

By 1977, two 5-in.-diameter, thin-window phoswich detectors were used, which provided a reduction in Compton scattered background by a factor of 4 in the ^{235}U region. In addition, the counting chamber was totally shielded. These changes enabled reduction of the counting time to 20 min. The calibration was similar to that described for 1968, but was cross-checked with the REMAB phantom used at Oak Ridge National Laboratory (ORNL). The minimum sensitivity was reported to be 30-60 µg of ^{235}U and 1 to 3 mg of natural or depleted uranium (Helgeson 1978, 1983). Calibration was performed with a phantom containing a known amount of ^{235}U and a known chest-wall thickness. The gross pulse-height spectrum obtained was adjusted for background by subtracting an assumed background spectrum, consisting of a straight line passing through the gross count values just below and just above the pulse-height spectrum region corresponding to 186 keV. The net count was adjusted for chest-wall thickness and converted to mass of ^{235}U, based on the calibration (Tuttle 1986a).

Until 1981, the results of lung count measurements were reported as zero if a result was below the 2-sigma uncertainty based on counting statistics. The results were reported as the actual value if equal to or greater than the 2-sigma uncertainty. Starting in 1981, Helgeson Nuclear Services was asked to report all values, regardless of the assigned uncertainty. An analysis of noncensored data along with counts of three individuals with no history of uranium exposure was reported in 1986 (Tuttle 1986a). This analysis estimated that the Helgeson results were biased high by 32.5 µg of ^{235}U when only natural levels were present. This suggested that the straight-line approach to subtracting background was not appropriate when measuring background subjects. At values approaching the maximum permissible lung burden (MPLB), the background subtraction method was less important. The MPLB for ^{235}U was 245 µg. Dose reconstructors can consider the values reported by Helgeson a claimant-favorable overestimate. For a more realistic estimate, adjust the lung count results by assuming a linear relationship of the bias between 0 and 1 MPLB, as indicated in Attachment C. Above 1 MPLB, no adjustment in the results is recommended.

Helgeson apparently also did special counts as requested by the site. These included counts for ^{241}Am and may have included counts for plutonium and thorium as well. Where these counts appear in claimant records, dose reconstructors should consider a “less than” value as the reporting level (decision level) and the MDA should be taken as twice the “less than” value.

Table 5-2 lists the frequencies for *in vivo* monitoring.

Table 5-2. Internal dose control program (*in vivo*).

Routine monitoring type	Period	Frequency
Chest count U-235 (UCLA)	1967–1968[a]	Monthly as long as count indicated > ½ of lung burden (0.02 µCi)
Chest count U-235 (ORNL)	1967[b]	One time
Chest count U-235 (Helgeson)	1968–1983 (enriched U work ended in 1983)	Three times/yr, 14-21 personnel selected for measurement each time
WBC Cs-137 (Helgeson)	1977, 1979	Uncertain[c]
WBC	1989–2005	Periodically[d]

a. Counts performed to follow up suspected enriched UAl_x exposures in the "powder room." The UCLA counter was not used after April 1968 (Tschaeche 1968a).
b. Counts of one employee as a cross-check on the UCLA calibration (Alexander 1967a).
c. It is unknown if these counts were routine or special.
d. Frequency not specified in Barnes (1999).

5.5 UNCERTAINTY

At ETEC the uncertainty for a single bioassay measurement was not reported consistently. Reviewed statements of work for bioassay services do not contain any specification for reporting uncertainty. UST results were apparently reported with a 2-sigma uncertainty if they were above the minimum detectable level. BioScience reported a 95% confidence interval (at least in some cases). The Individual Analysis Results Sheets (Figures B-5 to B-10, Attachment B) might show the uncertainty for individual measurements. Due to the calibration and other problems discussed above, the estimated error in the early UCLA lung counting results for ^{235}U was ±200% at 1 sigma. The stated uncertainty for the Helgeson lung counts was about ±25% at 1 sigma (Tschaeche 1968b).

5.6 DETECTION LIMITS

Urine results above the lower limit of detection (LLD), sensitivity, minimum detectable level, etc., were used to calculate the percentage of the MPBB received by the worker. The normalized result (1,500 mL/day) was divided by a standard excretion rate for one MPBB to produce this percentage. If possible, the worker followed with additional urine samples until the results were less than the laboratory LLD, the lowest value at which radioactivity was considered to be present with reasonable certainty (Tuttle 1989). No exact definitions of these terms have been found. In effect, the detection limit value was used as the reporting level. Most results appear to have been reported with an uncertainty of 2-sigma error or a 95% confidence interval. If the uncertainty included zero, the result was considered to be background (i.e., no radioactivity detected in the sample). Assuming the uncertainties in the background and the sample were equal, the detection limits published in site documents were closer to a decision level. That is, there is only an approximate 5% chance that results at this level are really background results (false positives). The MDAs would be approximately twice these values to ensure that there is only an approximate 5% chance that results at the MDA would not be detected (false negatives). The detection limits recovered are listed in the tables as the reporting levels. Since the detection limits in the tables were mainly collected from contract documents, the reported sensitivity ("less than" values) in the claimant records should be used to determine MDA in lieu of the values listed in these tables.

Tables 5-3 through 5-6 list MDAs and reporting levels for periods corresponding to the bioassay methods discussed in Sections 5.3 and 5.4. The reporting levels are listed in the units quoted in the references, which are generally the units of the results. However, various volumes were used to report the results. In general, the excretions assumed when reporting "per sample" or "per day" were 1,500 mL for urine samples and 135 g for fecal samples.

Table 5-3. Detection limits 1958–1966[a].

Radionuclide	Method/ description	MDA[b]	Reporting level[c]
Gross alpha	Urine	15 dpm/L (U. S. Nuclear) 4 dpm/sample (NSEC)	7.5 dpm/L 2 dpm/sample
Gross beta	Urine	150 dpm/L (U. S. Nuclear) 2 dpm/mL (NSEC)	75 dpm/L 1 dpm/mL
Gross beta (minus K-40)	Urine	0.4 dpm/mL (BioScience) 0.04 dpm/mL (UST)	0.2 dpm/mL 0.02 dpm/mL
H-3	Urine	4,440 dpm/mL (NSEC) 10,000 dpm/mL (BioScience) 10,000 dpm/mL (UST)	2,220 dpm/mL 5,000 dpm/mL 5,000 dpm/mL
MFP (gross)	Urine	4 dpm/mL (NSEC) 60 dpm/sample (Hanford) 0.2 dpm/mL (BioScience) 0.04 dpm/mL (UST)	2 dpm/mL 30 dpm/sample 0.1 dpm/mL 0.02 dpm/mL
Po-210	Urine	0.02 dpm/mL (BioScience) 0.02 dpm/mL (UST)	0.01 dpm/mL 0.01 dpm/mL
Plutonium	Urine	0.0006 dpm/mL (UST)	0.0003 dpm/mL
Sr-90	Urine	0.2 dpm/mL (BioScience) 0.04 dpm/mL (UST)	0.1 dpm/mL 0.02 dpm/mL
Thorium	Urine	0.4 µg/125 mL (Tracerlab) 0.0004 µg/mL (BioScience) 0.0004 µg/mL (UST)	0.2 µg/125 mL 0.0002 µg/mL 0.0002 µg/mL
Uranium	UF-1A Urine	0.004 µg/mL (U. S. Nuclear) 0.0002 µg/mL (Controls for Rad.) 0.0004 µg/mL (NSEC) 0.0004 µg/mL (BioScience) 0.0004 µg/mL (UST)	0.002 µg/mL 0.0001 µg/mL 0.0002 µg/mL 0.0002 µg/mL 0.0002 µg/mL
Uranium (enriched)	UR-1B Urine	15 dpm/L (U. S. Nuclear) 4 dpm/sample (NSEC) 0.012 dpm/mL (BioScience) 0.012 dpm/mL (UST)	7.5 dpm/L 2 dpm/sample 0.006 dpm/mL 0.006 dpm/mL

a. The date of the reference to the reporting levels, which are the best information available for the period 1958-1966.
b. Assumed to be twice the sensitivity (see text). The reported sensitivity ("less than" values) in the claimant records should be used to determine MDA in lieu of the values listed in this table. These values should also be used for the in-house laboratory.
c. U. S. Nuclear from Shepard (1959), Controls for Radiation, Inc., from O'Brien (1959), NSEC from Edelmann (1959) and NSEC (1957), BioScience from Lee (1963), and Tracerlab from Tracerlab (1959). UST results that were considered to be less than detectable generally have the reporting level included with the result.

5.7 UNIT CODES AND DESCRIPTION OF UNITS

The bioassay results from ETEC and its predecessor organizations are apparently not available in a computerized format. The units used to report the results are generally included in the hard-copy reports. Table 5-7 reflects this information.

5.8 EXCRETA SAMPLE KIT CODES

No codes have been found. Table 5-8 lists sample kit information summarized from various site documents.

Table 5-4. Detection limits 1967–1974[a].

Radionuclide	Method/ description	MDA[b]	Reporting level[c]
Ca-45	Urine	10 dpm/mL	5 dpm/mL
Ca-45	Urine	0.06 dpm/mL	0.03 dpm/mL
Ca-45	Feces	30 dpm/sample	15 dpm/sample
Cs-137	FP3b Urine	60 dpm/sample	30 dpm/sample
Gross alpha	Urine	0.02 dpm/mL	0.01 dpm/mL
Gross alpha	Urine	0.010 dpm/mL	0.005 dpm/mL
Gross alpha	Feces	4 dpm/sample	2 dpm/sample
Gross beta (less K-40)	Urine	0.08 dpm/mL	0.04 dpm/mL
Gross beta	Feces	4 dpm/sample	2 dpm/sample
H-3	Urine	10,000 dpm/mL	5,000 dpm/mL
I-131	Urine	0.12 dpm/mL	0.06 dpm/mL
I-131	Urine	0.4 dpm/mL	0.2 dpm/mL
MFP (gross)	Urine	0.04 dpm/mL	0.02 dpm/mL
MFP (gross)	Feces	16 dpm/sample	8 dpm/sample
Plutonium	Urine	0.8 dpm/sample (estimated)	0.4 dpm/sample (estimated)
Plutonium	Radiographic Urine	0.0006 dpm/mL	0.0003 dpm/mL
Plutonium	Procedure A Feces	2 dpm/sample	1 dpm/sample
Plutonium	Procedure B Feces	4 dpm/sample	2 dpm/sample
Polonium	Urine	0.02 dpm/mL	0.01 dpm/mL
Polonium	Feces	6 dpm/sample	3 dpm/sample
Pm-147	Urine	0.10 dpm/sample	0.05 dpm/sample
Radium	Urine	0.010 dpm/mL	0.005 dpm/mL
Radium	Feces	20 dpm/sample	10 dpm/sample
Sr-90	Urine	0.04 dpm/mL	0.02 dpm/mL
Sr-90	Feces	16 dpm/sample	8 dpm/sample
Sr-90	FP3a Urine	60 dpm/day	30 dpm/day
Thorium	Urine	0.002 µg/mL	0.001 µg/mL
Thorium	Feces	6 µg/sample	3 µg/sample
Uranium	UF Urine	0.006 µg/mL (10-mL volume)	0.003 µg/mL (10-mL volume)
Uranium	UF Urine	0.0004 µg/mL (100-mL volume)	0.0002 µg/mL (100-mL volume)
Uranium	UF Feces	4 µg/sample	2 µg/sample
Uranium (enriched)	UR Urine	0.012 dpm/mL	0.006 dpm/mL
Uranium (enriched)	UR Feces	4 dpm/sample	2 dpm/sample
U-235	IVLC (UCLA)	0.8 mg	0.4 mg
U-235	IVLC (Helgeson)	60-120 µg (depends on chest-wall thickness)	30-60 µg

a. The date of the reference to the reporting levels, which are the best information available for the period 1967-1974.
b. Assumed to be twice the sensitivity (see text). The reported sensitivity ("less than" values) in the claimant records should be used to determine MDA in lieu of the values listed in this table.
c. U-235 IVLC (UCLA) from Saxe (1967a). *In vitro* values are primarily from UST contract documents (Kellehar 1966b, 1967, Spielman 1968, Bales 1969, Staszesky 1971).

5.9 SOLUBILITY TYPE, FRACTION ACTIVITY, AND PARTICLE SIZE BY FACILITY

In the absence of any measurements or studies, NIOSH guidance requires the use of default solubility classes and particle size values from the International Commission on Radiological Protection (NIOSH 2002, pp. 15, 16). With one exception, facility-specific solubility and particle size data for ETEC has not been found. Activity fractions were not available with the exception of those for limited fuel fabrication operations. For highly enriched research reactors, values from Shleien, Slayback, and Birky (1998) have been listed as an approximation. In some cases, default assumptions from Barnes (1999, Table D-3) have been used to estimate the activation products using a ratio to the fission products. Table 5-9 lists this information.

Table 5-5. Detection limits 1975–1988[a].

Radionuclide	Method/ description	MDA[b]	Reporting level[c]
Cs-137	FP3b Urine	60 dpm/sample	30 dpm/sample
Cs-137	WBC	4 nCi	2 nCi
Plutonium	PUA Urine	0.0990 dpm/sample	0.0495 dpm/sample
Sr-90	FP3a Urine	60 dpm/day	30 dpm/day
U-235	IVLC (Helgeson)	60-120 µg (depends on chest-wall thickness)	30-60 µg
Uranium (total)	UF Urine	0.60 µg/day	0.30 µg/day
Uranium (enriched)	UR Urine	7.5 dpm/day	3.75 dpm/day

a. The date of the reference to the reporting levels, which are the best information available for the period 1975-1988.
b. Assumed to be twice the sensitivity (see text). The reported sensitivity ("less than" values) in the claimant records should be used to determine MDA in lieu of the values listed in this table.
c. Sr-90, U (total), and EU (1987, 1988) from Tuttle (1988b, 1989).

Table 5-6. Detection limits 1989–2005[a].

Radionuclide	Method/ description	MDA[b]	Reporting level[c]
Gross alpha	Urine	15 dpm/1,500 mL	7.5 dpm/1,500 mL
Gross beta	Urine	30 dpm/1,500 mL	15 dpm/1,500 mL
Gross beta, corrected for K-40	Urine	45 dpm/1,500 mL	22.5 dpm/1,500 mL
Gamma scan	Urine	60 dpm/1,500 mL	30 dpm/1,500 mL
Sr-90	Urine	6 dpm/1,500 mL	3 dpm/1,500 mL
Uranium fluorometric	Urine	15 µg/1,500 mL	7.5 µg/1,500 mL
Uranium alpha spec.	Urine	3 dpm/1,500 mL	1.5 dpm/1,500 mL
Thorium alpha spec.	Urine	3 dpm/1,500 mL	1.5 dpm/1,500 mL
Plutonium alpha spec.	Urine	3 dpm/1,500 mL	1.5 dpm/1,500 mL
Am-241	Urine	0.90 dpm/1,500 mL	0.45 dpm/1,500 mL
H-3, distillation	Urine	3000 dpm/1,500 mL	1,500 dpm/1,500 mL
H-3, electrolytic enrichment	Urine	60 dpm/1,500 mL	30 dpm/1,500 mL
WBC (Co-60)	WBC	4 nCi	2 nCi

a. The date of the reference to the reporting levels, which are the best information available for the period 1989-2005.
b. Assumed to be twice the sensitivity (see text). The reported sensitivity ("less than" values) in the claimant records should be used to determine MDA in lieu of the values listed in this table.
c. Sensitivity reported in Barnes (1999, Table D-1).

Table 5-7. Unit codes and description of units.

Computer code	Description of units
NA	NA

Several non-nuclear energy research programs were conducted in Area IV. The primary non-nuclear R&D activities were performed at the Liquid Metal Engineering Center, which became ETEC . The facilities that performed non-nuclear research are not included in Table 5-9 unless they were also used for nuclear research (Sapere and Boeing 2005, p. 2-12). Dose reconstructors should use general environmental radioactivity levels to assess personnel working in non-nuclear areas exclusively.

Table 5-8. Excreta sample kit codes.

Kit code[a]	Media	Sample description
NA	Urine	Routine samples. Single voiding collected on Monday morning before returning to work. Collected in single bottle. Also referred to as "rate samples." Positive samples were followed up with additional rate sample for verification. Time of sample collection and previous voiding were recorded. Normally collected in a 16-oz container. By 1999, 1-L polyethylene containers were used. It appears that one full container (900-1000 mL) was collected and collection time noted.
NA	Urine	24-hr samples. Used for follow-up to verified rate sample or for incidents. One or more samples could be requested. Single samples were collected at home on Sunday. Collected in 32-oz or 1-L polyethylene containers.
NA	Urine	Spot samples. For follow-up to incidents, spot sample could be collected as soon as possible. This sample was probably collected in 16-oz container.
NA	Feces	Could be requested in conjunction with urine samples as follow-up to incidents. No descriptions of historical fecal kits were located. By 1999, single voiding samples were collected in 83-oz polyethylene containers. The minimum mass that was considered adequate was 30 g.

a. NA means that no codes to identify sample kits were used at ETEC. For a time requests were color-coded but these do not appear in the individual records

In the 1960s, a test was conducted that involved dropping a 1-kg depleted uranium slug from a helicopter. Apparently the slug was never recovered. No contamination was found in the area (Sapere and Boeing 2005 [p. 2-13]). It is unlikely that this event would contribute significantly to the internal dose of occupationally exposed workers, and it has not been included.

5.10 FACILITY-SPECIFIC RADIONUCLIDE CONVERSIONS

AI fabricated fuel for the ATR and the Engineering Test Reactor (ETR) in 1966, 1967, and 1968 at the De Soto Facility in Room 1110-62, Building 101 (known at the time as Building 001). See Attachment A for details. A second fuel fabrication campaign for the ATR was completed in 1979 and 1980. These are the only two operations for which facility-specific data were made available. These data are listed in Attachment D and summarized in Table 5-10.

5.11 WORKPLACE MONITORING DATA

If bioassay data are not adequate to evaluate an individual's internal doses, dose reconstructors can use workplace monitoring data (NIOSH 2002). The following types of workplace data might be available for ETEC: breathing zone air samples, general area air samples, and surface contamination surveys. However, these data are not likely to be in individual exposure records. Data on respirator

Table 5-9. Solubility type, fraction activity, and particle size by facility.

Facility[a]	Compound[b]	RN[c]	Solubility type[d]	Particle size[d] (μ AMAD)	Activity fraction[e]
Downey Plant **Bldg. 001** WBNS reactor 1948–1955	Homogeneous water boiler-type reactors - 93%-enriched uranyl sulfate solution in H_2O. Reactors operated at low power levels (4-10 W).	Sr-90	to be determined	5	4.89E-01
		Cs-137	to be determined	5	5.10E-01
		U-234	to be determined	5	5.36E-08
		U-235	to be determined	5	1.24E-06
		U-238	to be determined	5	1.93E-08
Vanowen Fac. **Bldg. 038** L-47, L-77 reactors 1956–1960		Pu-238	to be determined	5	6.62E-04
		Pu-239	to be determined	5	1.22E-05
		Pu-240	to be determined	5	5.73E-06
		Pu-241	to be determined	5	3.33E-04
		Am-241	to be determined	5	1.16E-06
De Soto Fac. **Bldg. 104** L-77 reactor 1960–1979?					
Area IV, SSFL **Bldg. 4073** KEWB reactor 1956–1966					
Bldg. 4093 L-85 reactor 1956–1980					
De Soto Fac. **Bldg. 101** Powder room & new powder room 1966-1968 & 1979–1982	Uranium Aluminide Fuel Fabrication Alloy of uranium and aluminum	U-234	S	1	See Table 5-10
		U-235	S	1	See Table 5-10
		U-236	S	1	See Table 5-10
		U-238	S	1	See Table 5-10
Bldg. 4143 SRE 1957–1964 *Decontaminated 1974–1983* *Storage facility 1983–1999* *Demolished 1999*	High-temperature, sodium-cooled, graphite-moderated EU reactor (site of loss-of-coolant accident in 1959). Unalloyed uranium metal thermally bonded by NaK in stainless-steel tubes. 1957-July 1959, core was 2.78% EU September 1960, 2nd core began operation. Core was 7.6 % (weight) Th-232 with 93% EU (Atomics International, 1959)	Sr-90	to be determined	5	to be determined
		Cs-137	to be determined	5	to be determined
		U-234	to be determined	5	to be determined
		U-235	to be determined	5	to be determined
		U-238	to be determined	5	to be determined
		Pu-238	to be determined	5	to be determined
		Pu-239	to be determined	5	to be determined
		Pu-240	to be determined	5	to be determined
		Pu-241	to be determined	5	to be determined
		Am-241	to be determined	5	to be determined
		Co-60	to be determined	5	to be determined
		Eu-152	to be determined	5	to be determined
		Eu-154	to be determined	5	to be determined
		Th-232	to be determined	5	to be determined
		H-3			to be determined
Bldg. 4010 SER 1959–1960 S8ER 1963–1965	SNAP reactors – "fully enriched" uranium dispersed in zirconium hydride fuel rods	Sr-90	to be determined	5	4.60E-01
		Cs-137	to be determined	5	4.78E-01
		U-234	to be determined	5	5.03E-08
		U-235	to be determined	5	1.16E-06
		U-238	to be determined	5	1.81E-08
Bldg. 4024 S2DR 1961–1962 S10FS 1965–1966 SNAP TTF 1971		Pu-238	to be determined	5	6.22E-04
		Pu-239	to be determined	5	1.15E-05
		Pu-240	to be determined	5	5.38E-06
		Pu-241	to be determined	5	3.13E-04
		Am-241	to be determined	5	1.09E-06
		Co-60	to be determined	5	4.28E-03
		Eu-152	to be determined	5	4.89E-02
Bldg. 4028 STR 1961–1964 STIR 1964–1972		Eu-154	to be determined	5	6.11E-03
		H-3			1.95E-03

Table 5-9 (Continued). Solubility type, fraction activity, and particle size by facility.

Facility[a]	Compound[b]	RN[c]	Solubility type[d]	Particle size[d] (µ AMAD)	Activity fraction[e]
Bldg. 4059 S8DR 1968–1969					
Bldg. 4012 SNAP CTF 1962–1968 HMRFSR 1970–1972	Critical test facilities - SNAP development test facilities (fully enriched uranium dispersed in zirconium hydride fuel rods)	Sr-90	to be determined	5	4.89E-01
		Cs-137	to be determined	5	5.10E-01
		U-234	to be determined	5	5.36E-08
		U-235	to be determined	5	1.24E-06
		U-238	to be determined	5	1.93E-08
Bldg. 4373 SNAP CTF 1957–1963		Pu-238	to be determined	5	6.62E-04
		Pu-239	to be determined	5	1.22E-05
		Pu-240	to be determined	5	5.73E-06
Bldg. 4019 SNAP FSCF 1964–1965		Pu-241	to be determined	5	3.33E-04
		Am-241	to be determined	5	1.16E-06
Bldg. 4009 OMR CF 1958–1967	Critical test facilities – Civilian nuclear power test facilities, EU low-power reactors. HEU and DU used and stored also.	Sr-90	to be determined	5	4.60E-01
		Cs-137	to be determined	5	4.79E-01
		U-234	to be determined	5	5.04E-08
		U-235	to be determined	5	1.17E-06
SGR CF 1958–1967		U-238	to be determined	5	1.81E-08
		Pu-238	to be determined	5	6.23E-04
		Pu-239	to be determined	5	1.15E-05
High-energy rate forging 1980s		Pu-240	to be determined	5	5.39E-06
		Pu-241	to be determined	5	3.13E-04
		Am-241	to be determined	5	1.09E-06
		Co-60	to be determined	5	4.29E-03
DU storage Early 1990s		Eu-152	to be determined	5	4.90E-02
		Eu-154	to be determined	5	6.12E-03
Bldg. 4100 AETR test facility 1960–1974	Critical test facilities – thorium- and uranium-fueled reactors. 20 different reactor core configurations studied. The AETR's first 9 core configurations (through 1965) contained various amounts of U-233 and Th-232 and were driven by 93% enriched U fuel (Mountford and Morewitz, 1965). The activity fractions are the medians of the cores studied. Only very small amounts of the other radionuclides listed were likely to have been produced, but they were listed as ROCs by Boeing.	Sr-90	to be determined	5	
		Cs-137	to be determined	5	
		U-233	to be determined	5	9.81E-01
		U-234	to be determined	5	1.83E-02
		U-235	to be determined	5	5.71E-04
		U-238	to be determined	5	6.63E-06
Radiation Instrument Calibration and Radiological Sample Counting Laboratory 1985–present		Pu-238	to be determined	5	
		Pu-239	to be determined	5	
		Pu-240	to be determined	5	
		Pu-241	to be determined	5	
		Am-241	to be determined	5	
		Eu-152	to be determined	5	
		Eu-154	to be determined	5	
		Th-232	to be determined	5	1.05E-03
		H-3			
Bldgs. 4003 4163, 4041, 4654, 4689, 4653, 4606, 4773 SRE Support Complex 1954–1964	Nuclear support operations – SRE fuel assembly (uranium and thorium metal slugs), contained radioactive "hot cave," tanks, hoods, and lines until 1975. SRE fuel loaded in 1960 was 7.6 % (weight) Th-232 with 93% EU (Aomics International, 1959)	Sr-90	to be determined	5	4.60E-01
		Cs-137	to be determined	5	4.79E-01
		U-234	to be determined	5	5.04E-08
		U-235	to be determined	5	1.17E-06
		U-238	to be determined	5	1.81E-08
		Pu-238	to be determined	5	6.23E-04
		Pu-239	to be determined	5	1.15E-05
		Pu-240	to be determined	5	5.39E-06
		Pu-241	to be determined	5	3.13E-04
Analysis of SNAP fuel burn-up & irradiation experiments 1965–1973		Am-241	to be determined	5	1.09E-06
		Co-60	to be determined	5	4.29E-03
		Eu-152	to be determined	5	4.90E-02
		Eu-154	to be determined	5	6.12E-03
		Th-232	to be determined	5	to be determined
Bldg. 4005 Uranium Carbide Fuel Pilot Plant 1958-1993	Nuclear support operations – pilot plant for uranium-carbide fuel production in 1966-1977, first using DU and then EU	U-234	to be determined	5	to be determined
		U-235	to be determined	5	to be determined
		U-238	to be determined	5	to be determined

Table 5-9 (Continued). Solubility type, fraction activity, and particle size by facility.

Facility[a]	Compound[b]	RN[c]	Solubility type[d]	Particle size[d] (µ AMAD)	Activity fraction[e]
Bldg. 4011 Radiation Instrumentation Calibration Laboratory 1984–1996	Nuclear support operations – sealed sources. In 1960, a radioactive liquid spill from OMRE shipping cask occurred west of building.	Sr-90	to be determined	5	4.60E-01
		Cs-137	to be determined	5	4.79E-01
		U-234	to be determined	5	5.04E-08
		U-235	to be determined	5	1.17E-06
		U-238	to be determined	5	1.81E-08
		Pu-238	to be determined	5	6.23E-04
		Pu-239	to be determined	5	1.15E-05
		Pu-240	to be determined	5	5.39E-06
		Pu-241	to be determined	5	3.13E-04
		Am-241	to be determined	5	1.09E-06
		Co-60	to be determined	5	4.29E-03
		Eu-152	to be determined	5	4.90E-02
		Eu-154	to be determined	5	6.12E-03
Bldg. 4030 Van de Graaff Accelerator 1960–1964	Nuclear support operations – neutron source, used H-3 target	H-3			1
Bldg. 4020 Hot Laboratory 1957–1988 *D&D Completed 1999*	Nuclear support operations – Used to examine fuel from SRE, SER, S2DR, S8DR, S10FS-3, and outside reactors. Fuel disassembled or separated from cladding.	Sr-90	to be determined	5	4.60E-01
		Cs-137	to be determined	5	4.79E-01
		U-234	to be determined	5	5.04E-08
		U-235	to be determined	5	1.17E-06
		U-238	to be determined	5	1.81E-08
		Pu-238	to be determined	5	6.23E-04
		Pu-239	to be determined	5	1.15E-05
		Pu-240	to be determined	5	5.39E-06
		Pu-241	to be determined	5	3.13E-04
		Am-241	to be determined	5	1.09E-06
		Co-60	to be determined	5	4.29E-03
		Eu-152	to be determined	5	4.90E-02
		Eu-154	to be determined	5	6.12E-03
		Pm-147	to be determined	5	to be determined
Bldg. 4023 Liquid Metals Component Test Building 1962–1986	Nuclear support operations – tests with sodium loops containing radioactive contaminants	Sr-90	to be determined	5	to be determined
		Cs-137	to be determined	5	to be determined
		Co-60	to be determined	5	to be determined
		Eu-152	to be determined	5	to be determined
		Eu-154	to be determined	5	to be determined
Bldg. 4029 Radioactive Measurement Facility 1959–1974	Nuclear support operations – Leaking calibration source contaminated source well	Ra-226	to be determined	5	1
Bldg. 4055 NMDF 1967–1979	Nuclear support operations – Development work involving Pu DU work 1967	U-234	to be determined	5	to be determined
		U-235	to be determined	5	to be determined
		U-238	to be determined	5	to be determined
		Pu-238	to be determined	5	to be determined
		Pu-239	to be determined	5	to be determined
		Pu-240	to be determined	5	to be determined
		Pu-241	to be determined	5	to be determined
Bldg. 4064 Fuel Storage Facility 1958–1993 *D&D in 1997*	Nuclear support operations – Secure storage of nonirradiated fuel (EU and Pu); following removal of fissionable material in mid-1980s, radioactive waste (soil) stored until 1993.	Cs-137	to be determined	5	to be determined
		U-234	to be determined	5	to be determined
		U-235	to be determined	5	to be determined
		U-238	to be determined	5	to be determined
		Pu-238	to be determined	5	to be determined
		Pu-239	to be determined	5	to be determined
		Pu-240	to be determined	5	to be determined
		Pu-241	to be determined	5	to be determined
		Am-241	to be determined	5	to be determined
		Co-60	to be determined	5	to be determined
		Eu-152	to be determined	5	to be determined
		Eu-154	to be determined	5	to be determined

Table 5-9 (Continued). Solubility type, fraction activity, and particle size by facility.

Facility[a]	Compound[b]	RN[c]	Solubility type[d]	Particle size[d] (µ AMAD)	Activity fraction[e]
Bldgs. 4021, 4022 RMHF 1959–present	Nuclear support operations – Radioactive waste processing from onsite programs; 2005 storage area from D&D activities	Sr-90	to be determined	5	4.60E-01
		Cs-137	to be determined	5	4.78E-01
		U-234	to be determined	5	5.03E-08
		U-235	to be determined	5	1.16E-06
		U-238	to be determined	5	1.81E-08
		Pu-238	to be determined	5	6.22E-04
		Pu-239	to be determined	5	1.15E-05
		Pu-240	to be determined	5	5.38E-06
		Pu-241	to be determined	5	3.13E-04
		Am-241	to be determined	5	1.09E-06
		Co-60	to be determined	5	4.28E-03
		Eu-152	to be determined	5	4.89E-02
		Eu-154	to be determined	5	6.11E-03
		H-3			1.95E-03
Bldg. 4363 Mechanical Component Development & Counting 1956–1963	Nuclear support operations – In 1962, work done on contaminated component from SRE accident	Sr-90	to be determined	5	0.5
		Cs-137	to be determined	5	0.5
17th Street Drainage 1959?–1999	Areas of known contamination – runoff from SNAP facilities	Cs-137	to be determined	5	1.0
OCY Late 1960s–late 1970s	Areas of known contamination – spill detected in 1976 (mixed fission products suspected)	Sr-90	to be determined	5	0.5
		Cs-137	to be determined	5	0.5
Bldg. 4886 Former Sodium Disposal Facility 1956–1978	Areas of known contamination – storage drums contaminated with residual radioactivity. SRE fuel loaded in 1960 was 7.6 % (weight) Th-232 with 93% EU (Atomics International, 1959)	Sr-90	to be determined	5	to be determined
		Cs-137	to be determined	5	to be determined
		U-234	to be determined	5	to be determined
		U-235	to be determined	5	to be determined
		U-238	to be determined	5	to be determined
		Pu-238	to be determined	5	to be determined
		Pu-239	to be determined	5	to be determined
		Pu-240	to be determined	5	to be determined
		Pu-241	to be determined	5	to be determined
		Th-232	to be determined	5	to be determined
		H-3			to be determined

a. Facilities were combined for this analysis if they were similar and had a common list of radionuclides of concern.
b. If chemical compounds were not available, the best description found is listed.
c. Radionuclides are radionuclides of concern from Sapere and Boeing (2005, Tables 2-1, 2-2 and Section 2.0). Radionuclides were not included in the radionuclides of concern if they were an activation product limited to steel and rebar, had a short half-life, were a naturally occurring isotope consistent with natural background, or had not been observed in soil measurements.
d. Defaults used except for the powder room.
e. Activity fractions for fission products calculated from values published for a research reactor and activation products (Co-60 and Eu-152) calculated from Eu-154 ratios in Barnes (1999). A technical information bulletin on fission product nuclides to assign across the complex is to be published.

use are not likely to be available. Quantitative fit testing information was not located. In the case of surface contamination data, site/process-specific resuspension factors are not likely to be available.

5.12 SOURCE TERM DATA

Without bioassay or air sample data, the last resort is determination of airborne concentrations using source term evaluations (NIOSH 2002, p. 19). Data on the amount of dispersible material available does not appear to be available for ETEC.

Table 5-10. Facility-specific radionuclide conversions.

Process description	Activity per unit mass (Bq/g uranium)			
	U-234	U-235	U-236	U-238
Enriched uranium fuel fabrication, Room 1110-62 (powder room), Building 101, De Soto Facility, 1966–1967	1.35E+06	7.44E+04	7.31E+03	7.50E+02
ATR enriched uranium fuel fabrication, 1979–1982 Building 101, De Soto Facility	2.37E+06	7.44E+04	1.14E+04	6.79E+02

5.13 RADON

For dose reconstruction under EEOICPA, occupational radon exposure is exposure to radon emanating from sources other than those naturally occurring in the area. Dose reconstructors must subtract the natural background level of radon exposure from any measured values when assessing occupational exposure (NIOSH 2002, p. 32). ETEC was not a processing or storage location for large quantities of ^{226}Ra or ^{222}Rn. Radon measurements made from December 1989 to February 1990 in several facilities indicated that none of the areas measured exceeded 2 pCi/L and all but one was less than 1 pCi/L.

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GLOSSARY

Atomic Energy Commission (AEC)
An agency established by the U.S. Government for oversight of nuclear weapons and power production; a predecessor to the U.S. Department of Energy.

cladding
The outer layer of material encasing a reactor fuel element (e.g., aluminum or zirconium). Cladding promotes the transfer of heat from the fuel to the coolant and contains fission products and activation products that result from the fissioning of the fuel.

core
That part of the reactor consisting of the fuel and some of the control elements for reactor operation.

dosimetry
The science of assessing absorbed dose, dose equivalent, effective dose equivalent, etc., from external or internal sources of radiation.

fission
A nuclear transformation characterized by the splitting of a nucleus into at least two other nuclei and the release of a relatively large amount of energy.

fission product
Radionuclides resulting from fission.

ionizing radiation
Electromagnetic or particulate radiation capable of producing charged particles through interactions with matter.

in vitro
Literally, in glass; outside the living body and in an artificial environment; internal bioassay sampling, such as fecal samples or urine samples.

in vivo
Literally, in the living; in the living body of a plant or animal; bioassay sampling by whole-body counting.

isotope
Nuclides having the same number of protons in their nuclei (same atomic number), but a differing number of neutrons (different mass number).

natural uranium
Uranium that has not been through an enrichment process.

radiation
Energy transferred through air or some other media in the form of particles or waves (see ionizing radiation).

radioactivity

The spontaneous emission of radiation, generally alpha or beta particles, gamma or X-rays, or neutrons from unstable atoms.

radionuclide

A radioactive species of an atom characterized by the constitution of its nucleus specified by atomic number (the number of protons) and the mass number (equal to the number of protons plus neutrons).

type

The rate of absorption from lung to blood of inhaled radioactive materials and includes types F (fast), M (moderate), and S (slow).

zirconium

A metallic element highly resistant to corrosion and often used to make cladding for nuclear fuel. It is sometimes alloyed in small amounts in the fuel itself.

ATTACHMENT A
SUMMARY OF URANIUM FUEL FABRICATION INTERNAL DOSIMETRY ISSUES

In the fall of 1965, AI began fabricating reactor fuel elements. AI fabricated fuel for the ATR and the ETR in 1966, 1967, and 1968 at the De Soto Facility. In Room 1110-62, Building 101 (known at the time as Building 001), briquettes of an alloy of 93%-enriched uranium and aluminum, known as uranium aluminide (UAl_x), were formed in an electric arc melting furnace. These briquettes were crushed to form a powder, which was cold-pressed into compacts that became the cores of the fabricated fuel plates. The room where these activities took place was known as the "powder room."

The work was performed at six work stations: weighing station 1, a melting station, a crushing and sieving station, weighing station 2, a compact forming station, and a deburring station. The operations were performed with containment thought to be appropriate for the type of operation and the hazard of the material. From one to four air samplers operated in the room near the work stations.

In 1967, after 15 months of operation, urine bioassay data indicated that the material was probably insoluble and, therefore, that the air activity was not being compared to the appropriate MPC. Although the uranium was more than 93% ^{235}U by weight, ^{234}U accounted for more than 96% of its activity. The insoluble MPC for ^{234}U was a factor of 6 lower than the soluble MPC. This led to the conclusions that the regulatory standard (weekly average MPC) had been exceeded on a number of occasions and that equipment and procedures for controlling the airborne uranium were insufficient.

An internal investigation determined that the primary reason for the ineffective confinement of the uranium was leakage from the crushing glovebox seal, from the fume hood of weighing station 2, and from the arc furnace. Temporary measures were put in place until more permanent fixes could be implemented. These changes were put in place and the project was completed. Along with the engineering changes implemented, workers were required to wear full-face respirators and lapel air samplers (Saxe 1967b). At some point, operations were relocated to a "new powder room."

In 83 weeks of operation, the MPC for insoluble ^{234}U (1×10^{-10} µCi/cm^3) was exceeded in 33 weeks. Twenty-one personnel who had worked in the powder room were monitored by chest counting. Three of these personnel were determined to have exceeded the MPBB and received work restrictions. The results for the other workers ranged from no detectable intake to less than one MPBB.

On August 15, 1967, a report of regulatory violations was submitted to the AEC. The report included the corrective measures taken (Sax 1967b). Inquiries by the AEC led to a review of an earlier fuel fabrication project, the Japan Atomic Energy Research Institute (JAERI), which took place in Room 1110-061. This operation involved uranium metal plates enriched to 20% by weight. The activity of the material was still approximately 90% ^{234}U. This operation did not involve crushing, but deburring the edges of the plates occurred during the process. This apparently took place outside ventilated enclosures until January 1967. An investigation into the JAERI fuel operation led to the conclusion that the average weekly concentration for insoluble ^{234}U had been exceeded six times during the operation in November and December 1966. The soluble standard (6×10^{-10} µCi/cm^3) would not have been exceeded. The insoluble standard was used because of the experience with UAl_x and the lack of evidence that uranium metal should be treated differently. Using this standard, five people had been exposed to concentrations greater than the AEC MPC. None of these people were involved in the ATR or ETR project (Remley 1967). Urinary excretion was measured on all five workers. The data were consistent with short-term exposures with small lung depositions. However, one of the workers initially showed a lung burden 1.5 times the MPBB. As stated in the main text, lung counting was its infancy during this time. The estimate for this worker was later reduced to 0.5 MPBB, based

on the difference in enrichments between the JAERI material and the calibration source for the lung counter at UCLA.

Air samples collected prior to 1966 were reviewed, and no concentrations exceeded 1×10^{-10} µCi/cm^3 (Saxe 1967c).

A particle size study consisting of two general area air samples from the powder room indicated that the particles were less than 1 micron count median diameter (CMD) (Alexander 1967c). The actual median count diameter and geometric standard deviation were not provided in the reference; therefore, the activity median aerodynamic diameter (AMAD) could not be determined. In the only other particle size reference recovered, Baurmash (1967) measured the particle size distribution for a UO_2 grinding operation in the "processing room in Bldg. 1." The CMD reported for this operation was 0.195 micron with a geometric standard deviation of 1.66. The mass median diameter was calculated as 0.42 micron. Using a density of 10.97 g/cm^3 for UO_2, the AMAD for this aerosol is 1.39 microns. While this information is extremely limited, it points out that small particle sizes were at least possible during uranium fuel fabrication. Therefore, dose reconstructors should consider a 1-micron AMAD particle size for these operations in addition to the default assumption.

A study of the solubility of UAl_x in simulated lung fluid was completed by AI in 1968. While the original study has not been recovered, the reference to it indicates that UAl_x was about 10 times more insoluble than the most insoluble uranium compound (U_3O_8) measured by ORNL (Alexander 1968b). ICRP (1979) lists U_3O_8 as a class Y compound. The quality of the AI study is uncertain. ICRP (1994) lists the most insoluble compounds of uranium as type S. Therefore, dose reconstructors should consider UAl_x results as type S.

ATTACHMENT B
EXAMPLE INTERNAL DOSIMETRY RECORD DOCUMENTS
Page 1 of 20

PM-1
Appendix A-1
July 1, 1963
Page 7 of 13

ATOMICS INTERNATIONAL
A DIVISION OF NORTH AMERICAN AVIATION, INC.

TO: Health and Safety Laboratory Unit ADDRESS: 779-21 Headquarters #004

FROM: Health and Safety Operations Unit ADDRESS:

PHONE: DATE:

SUBJECT: Request for Film Badge and Bioassay Services

To be Completed by Employee Requiring Services

In accordance with federal, state and local regulations, a complete occupational radiation history is maintained on employees subject to personnel monitoring and/or bioassay. Therefore, please fill in the information requested below.

1. Name_____ 2. Serial #_____

3. Social Security #_____ 4. Birth Date_____

Occupational Exposure - Previous History

5. Previous Employments Involving Radiation Exposure - List Name and Address of Employers	6. Dates of Employment (From-To)	7. Periods of Exposure

NOTE: If you listed one of the Military Services as a former employer where radiation exposure was involved, please give Service Serial Number_____.

Certification: I certify that the exposure history listed in columns 5, 6, and 7 is correct and complete to the best of my knowledge and belief.

_____
Employee's Signature Date

To Be Completed by Health and Safety Operations Unit

Work Area Code #_____ Film Badge Type By _____ Bym _____

Employee's Supervisor_____

Will bioassay services be required? Yes No

Signed:_____
Health and Safety Operations

Figure B-1. Example of Request for Film Badge and Bioassay Services, 1963.

ATTACHMENT B
EXAMPLE INTERNAL DOSIMETRY RECORD DOCUMENTS
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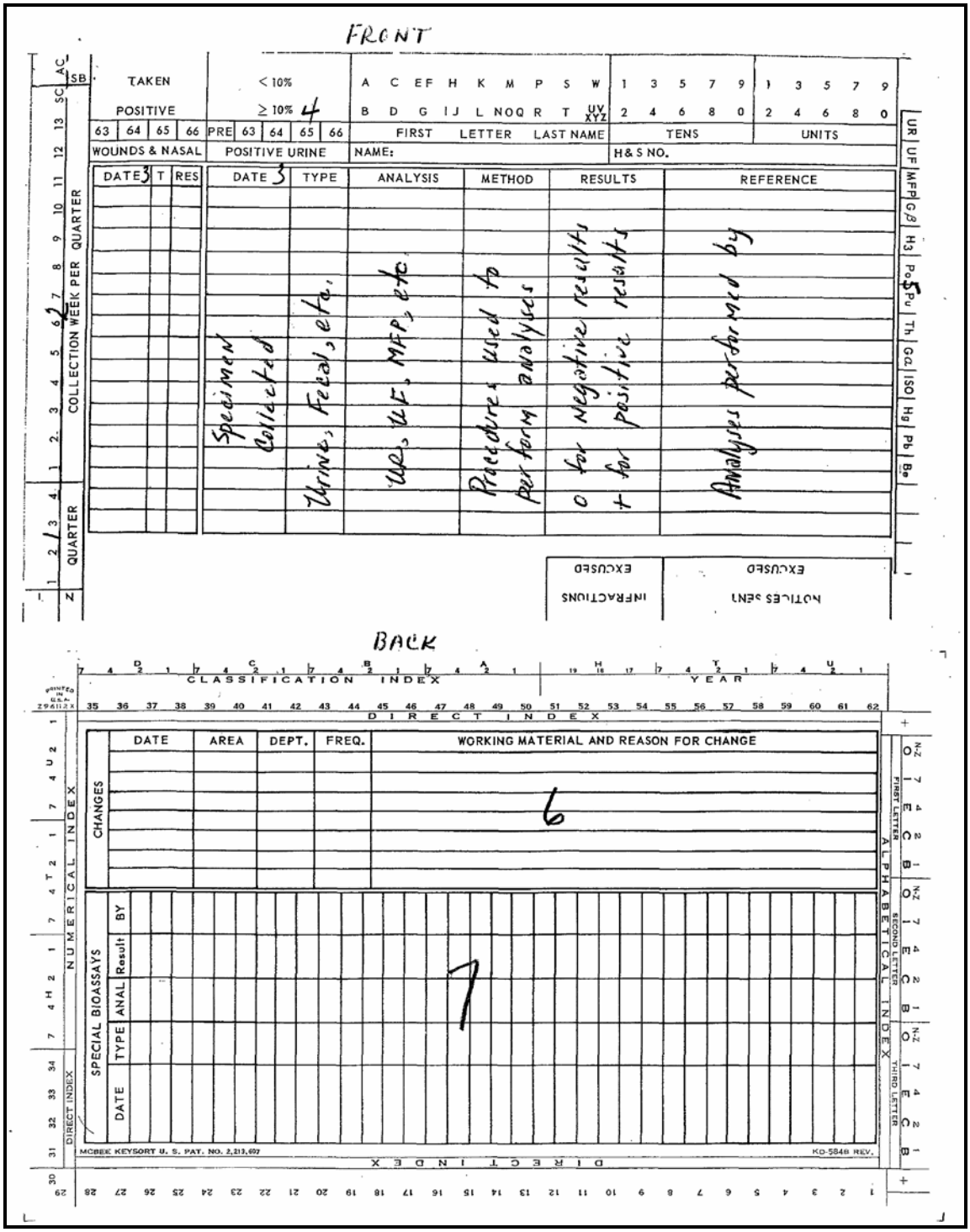

Figure B-2. Example Individual Personnel Keysort Card, 1966. (Both sides are shown. See key on following page.)

ATTACHMENT B
EXAMPLE INTERNAL DOSIMETRY RECORD DOCUMENTS
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Key to the Individual Keysort Card, Figure B-2. The following numbers correspond to the handwritten numbers on the form. The cards were set up for a 4-yr period (Kellehar 1966a).

1. Frequency
2. Week of collection
3. Date of each specimen collection, wound, and nasal smears
4. Indication of analysis results; degree of positive results in any year
5. Type of analyses required
6. Changes in work affecting analyses and/or specimen collection (reverse side)
7. Special bioassays (reverse)

The following is an edited version of the description of the form provided by ETEC:

This is the "master" record for internal dosimetry results. This lists the type of analysis and the "per day" results of those analyses.

Date is the date the sample was obtained from the individual.

Type lists the type of sample obtained. Method is a code to identify the procedure utilized. An incomplete record exists of the specific laboratory techniques associated with these methods. When results are listed as "< XXX," XXX is used as the reporting level rather than the value listed in the table.

Results are listed. In general, unless otherwise listed, they are in "dpm/day" (for radiological data) or "µg/day" (for fluorometric or other chemical analyses). Results were standardized to a 1,500-mL/day urine excretion rate or a 135-g/day fecal excretion rate. The results line up with the methods. Statistical errors were generally NOT provided, but can be obtained by manual search of the source data.

This (back) side of the Bioassay Card sometimes has additional bioassay data. No results are listed on this side of the card, but it can be useful for determining the work location of the individual and possibly the type of work being done at the time.

Special bioassays may also be listed on this side of the card. If listed on this side, it generally suggests that the bioassay was taken in response to a specific incident. However, absence of bioassay data on this side of the card should not be interpreted as an indication that the individual was never involved in an incident.

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EXAMPLE INTERNAL DOSIMETRY RECORD DOCUMENTS
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BIOASSAY DATA SHEET

☐ Routine ☐ Special - Date, Time of Incident_____
Name_____ Area_____ H.&S.#____

For Person-nel Expo-sure Record	Date Sampled	Sample Type	Analysis	Method	Quarter Dose, rem	Quarter / Yr	Reference

Sampling Data	1	Specimen H.&S.#					
	2	Type (Urine, Feces, etc.)					
	3	Date, Time of Previous Void ΔD_1					
	4	Date, Time Specimen Voided* ΔD_2					
	5	Mid-excretory to First Count ΔT_1	hrs				
	6	Vol. or Wt. of Specimen	ml-mg				
Preparation Data	7	Analysis					
	8	Method					
	9	Vol. or Wt. Analyzed (Sample)	ml-mg				
	10	Date, Time Ppt. Began ΔD_3					
	11	Start of Ppt. to First Count ΔT_2	hrs				
	12	Ppt. or Residue + Planchet Wt.	mg				
	13	Planchet Wt.	mg				
	14	Ppt. or Residue Wt.	mg				
	15	Planchet Area	cm^2				
	16	Ppt. or Residue Thickness	mg/cm^2				
Radiometric — First β Count — Additional Counts on Back	17	Date, Time Count Began $\beta\ \Delta D_4$					
	18	Total Count $\beta\ Ng_1$	c				
	19	Counting Time $\beta\ Tg_1$	min				
	20	Gross Count $\beta\ Rg_1$	c/m				
	21	K^{40} $\beta\ Rk^{**}_1$	c/m				
	22	Background Control $\beta\ Rb_1$	c/m				
	23	Net Count ± $\beta\ Rs_1$	c/m				
	24	Fssa					
	25	Efficiency Factor					
	26	Total Activity per Sample	d/m				
	27	Total Activity per Specimen	d/m				
	28	24 Hr. Volume	d/m				
	29	Analysis by					
	30	Reference					
Radiometric — First α Count — Additional Counts on Back	31	Analysis					
	32	Method					
	33	Vol. or Wt. Analyzed (Sample)	ml-mg				
	34	Date, Time, Count Began $\alpha\ \Delta D_4$					
	35	Total Count $\alpha\ Ng_1$	c				
	36	Counting Time $\alpha\ Tg_1$	min				
	37	Gross Count $\alpha\ Rg_1$	c/m				
	38	Background Control $\alpha\ Rb_1$	c/m				
	39	Net Count ± $\alpha\ Rs_1$	c/m				
	40	Fssa					
	41	Efficiency Factor					
	42	Total Activity per Sample	d/m				
	43	Total Activity per Specimen	d/m				
	44	24 hr. Volume	d/m				
	45	Analysis by					
	46	Reference					

Figure B-3. Example Bioassay Data Sheet, page 1 of 2 (used by onsite laboratory).

ATTACHMENT B
EXAMPLE INTERNAL DOSIMETRY RECORD DOCUMENTS
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Bioassay Data Sheet, page 1 (front).

The following is an edited version of the description of the form provided by ETEC:

This sheet records bioassay results when a determination was done in the AI laboratory (it also appears to have been used to log in and document vendor results that were sent in tabular form). It was designed to be a multipurpose form for use with all types of analyses.

The general concept of the sheet was that all steps of a hand calculation could be performed by entering the appropriate data and performing the indicated mathematical operations. The calculations on this side of the page were preliminary; the sample results are listed on the second page of this form.

One will occasionally find results penciled in on the top right corner of the page.

The calculations are relatively straightforward. They are typical for a radiochemical/radiological laboratory.

Multiple count blocks were provided; it appears this was used to log in sequential counts to determine half-lives of the isotopes.

Information blocks are self-explanatory.

Note that only those blocks on the form pertinent to the type of measurement were filled in. One can use this to determine the type of sample being analyzed.

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EXAMPLE INTERNAL DOSIMETRY RECORD DOCUMENTS
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Second Count	47	Specimen H.&S.#					
	48	Date, Time Count Began	ΔD_5				
	49	Hrs. From Last Count	ΔT_3	hrs			
	50	Total Count	Ng_2	***c			
	51	Counting Time	Tg_2	min			
	52	Gross Count - Rk_1 **	Rg_2	c/m			
	53	Background Control	Rb_2	c/m			
	54	Net Count	Rs_2	c/m			
Third Count	55	Date, Time Count Began	ΔD_6				
	56	Hrs. From Last Count	ΔT_4	hrs			
	57	Total Count	Ng_3	***c			
	58	Counting Time	Tg_3	min			
	59	Gross Count - Rk_1 **	Rg_3	c/m			
	60	Background Control	Rb_3	c/m			
	61	Net Count	Rs_3	c/m			
Fourth Count	62	Date, Time Count, Began	ΔD_7				
	63	Hrs. From Last Count	ΔT_5	hrs			
	64	Total Count	Ng_4	***c			
	65	Counting Time	Tg_4	min			
	66	Gross Count - Rk_1 **	Rg_4	c/m			
	67	Background Control	Rb_4	c/m			
	68	Net Count	Rs_4	c/m			
Fifth Count	69	Date, Time Count Began	ΔD_8				
	70	Hrs. From Last Count	ΔT_6	hrs			
	71	Total Count	Ng_5	***c			
	72	Counting Time	Tg_5	min			
	73	Gross Count - Rk_1 **	Rg_5	c/m			
	74	Background Control	Rb_5	c/m			
	75	Net Count	Rs_5	c/m			
Growth or Decay Results	76	Decay or Growth Factor					
	77	Daughters Present					
	78	Parent					
	79	Net Count ( ) @ΔT_1 or ΔT_2=0		c/m			
	80	Absorption Factor ( )		fssa			
	81	**Efficiency Factor** ( )					
	82	Total Activity per Sample		d/m			
	83	Total Activity per Specimen		d/m			
	84	24 Hr. Volume		d/m			
	85	Reference					

Spectrographic, Fluorometric or Colorimetric	86	Specimen H.&S.#					
	87	Analysis					
	88	Method					
	89	Vol. or Wt. Analyzed (Sample)		ml-mg			
	90	Range Setting					
	91	Scale Reading					
	92	Element Equivalent		ugm			
	93	Control		ugm			
	94	Net Sample		ugm			
	95	Net Specimen		ugm			
	96	24 Hr. Volume		ugm			
	97	Analysis by					
	98	Reference					

Remarks _____

* For a 24 hour urine specimen this is the time of the last void
** Method 2A only
*** Indicate type of radiation in each data column

Figure B-4. Example Bioassay Data Sheet, page 2 of 2 (used by onsite laboratory).

ATTACHMENT B
EXAMPLE INTERNAL DOSIMETRY RECORD DOCUMENTS
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Bioassay Data Sheet, page 2 (back).

The following is an edited version of the description of the form provided by ETEC:

This sheet records bioassay results when a determination was done in the AI laboratory (it also appears to have been used to log in and document vendor results that were sent in tabular form). It was designed to be a multipurpose form to be used with all types of analyses.

This is the "back" side of the form.

The general concept of the sheet was that all steps of a hand calculation could be performed by entering the appropriate data and performing the indicated mathematical operations. The calculations on the front side of the page were preliminary; the sample results are listed on this second page of the form.

One will occasionally find results penciled in on the top right corner of the page.

The calculations are relatively straightforward. They are typical for a radiochemical/radiological laboratory.

Multiple count blocks were provided; it appears this was used to log in sequential counts to determine half-lives of the isotopes.

Information blocks are self-explanatory.

Note that only those blocks on the form pertinent to the type of measurement were filled in. One can use this to determine the type of sample being analyzed.

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EXAMPLE INTERNAL DOSIMETRY RECORD DOCUMENTS
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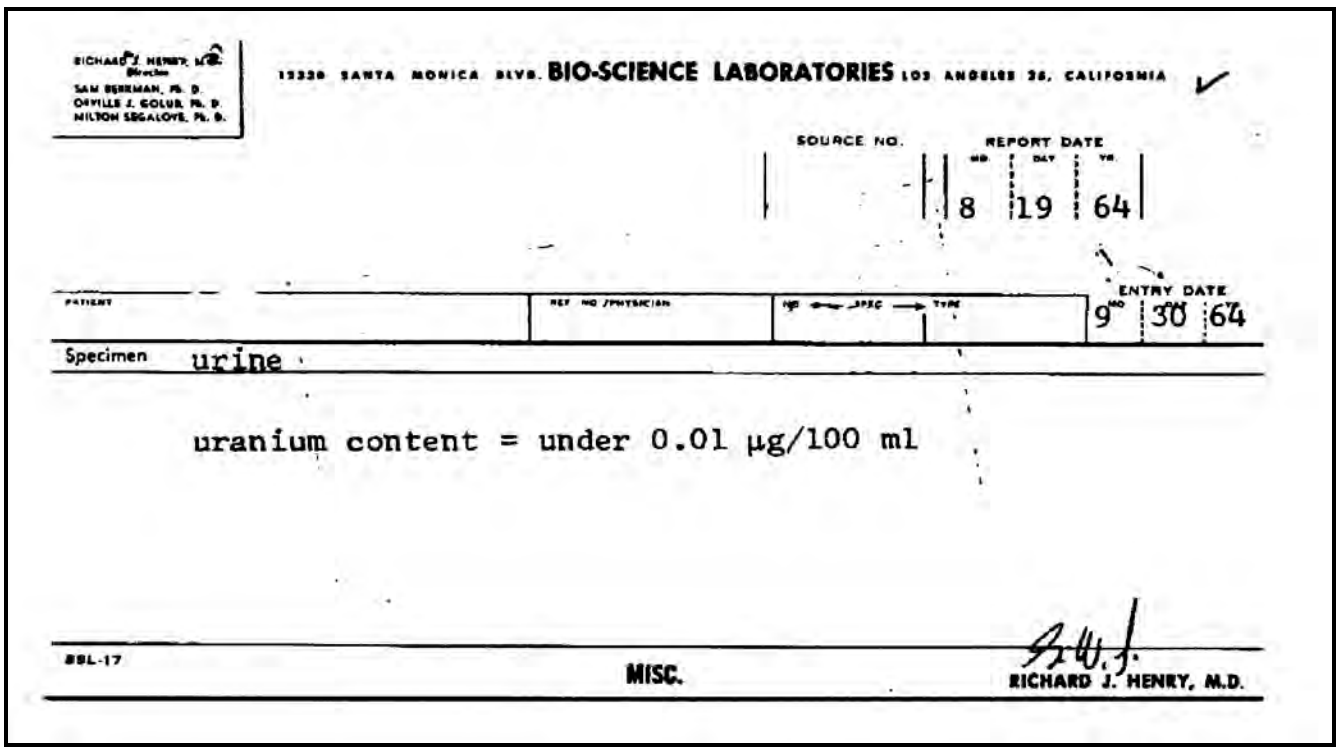

RICHARD J. HENRY, M.D.
Director
SAM BERKMAN, Ph. D.
ORVILLE J. GOLUB, Ph. D.
MILTON SEGALOVE, Ph. D.

12330 SANTA MONICA BLVD. **BIO-SCIENCE LABORATORIES** LOS ANGELES 25, CALIFORNIA

SOURCE NO.

REPORT DATE
MO. DAY YR.
8 19 64

PATIENT | REF. NO. /PHYSICIAN | AGE → SEX → TYPE

ENTRY DATE
MO. DAY YR.
9 30 64

Specimen urine

uranium content = under 0.01 µg/100 ml

BBL-17

MISC.

RICHARD J. HENRY, M.D.

Figure B-5. Example Bioassay Data Sheet, BioScience, Uranium Fluorometric.

ATTACHMENT B
EXAMPLE INTERNAL DOSIMETRY RECORD DOCUMENTS
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The following is an edited version of the description of the form provided by ETEC:

Report for Uranium fluorometric results from BioScience Laboratories. These are the "record of entry" reports from the vendor laboratory that processed urine and fecal bioassay results.

These records should be correlated to the bioassay summary record (Individual Keysort Card [Figure B-2]). The individual's name is usually written in pencil in the upper left margin.

The Source Number probably refers to an identification code for Atomics International.

The Report Date is the date the results were available.

The number in the Patient block is the Health Physics ID Number used to identify the person.

The meaning of the No. Spec. Type entry is not known at this time. It is sometimes blank.

Entry Date refers to the date that the sample was received for processing. In general it is one to two days after the date the sample was obtained by Rocketdyne from the individual. This date is obtained from the Individual Keysort Card.

Results are generally reported in micrograms U (µg) per 100 milliters of samples (ug/100 ml). On positive samples, one will usually find a value pro-rated into a "per day" rate using 1500 ml/day as a urine excretion rate, and 135 g/day for a fecal excretion rate.

By dividing the uranium radiometric values by the uranium fluorometric values, one can derive a specific activity for the sample. This specific activity can then be used to estimate the general enrichment level of the uranium the individual was exposed to. For simplicity, for low intakes, the ETEC practice was to model the intake as pure U-234 (which results in overestimation of "true" dose).

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EXAMPLE INTERNAL DOSIMETRY RECORD DOCUMENTS
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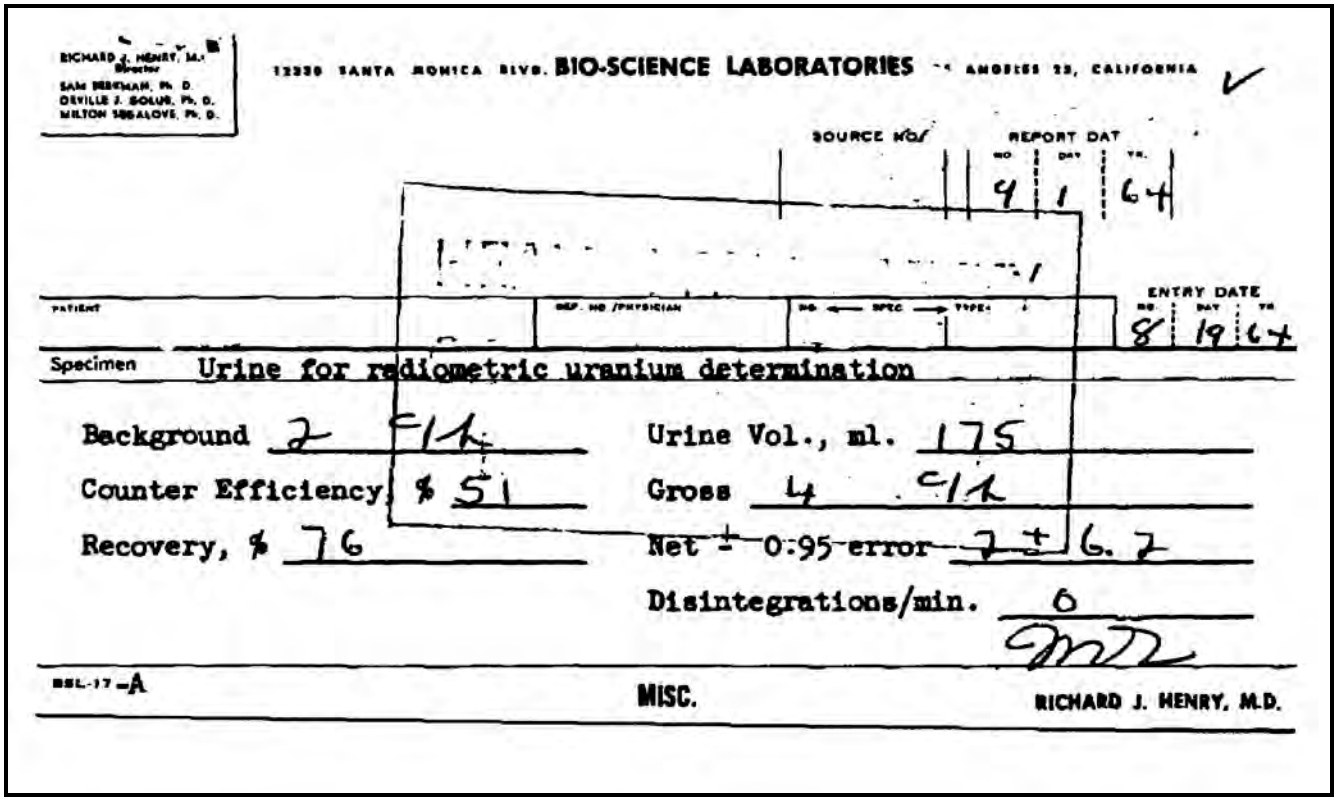

RICHARD J. HENRY, M.D.
Director
SAM BERTMAN, Ph. D.
DEVILLE J. SOLUS, Ph. D.
MILTON [illegible]LOVE, Ph. D.

12338 SANTA MONICA BLVD. **BIO-SCIENCE LABORATORIES** LOS ANGELES 25, CALIFORNIA

SOURCE NO. [illegible] REPORT DAT MO 4 DAY 1 YR 64

PATIENT [illegible] REF. NO /PHYSICIAN [illegible] NO. SPEC. TYPE: [illegible] ENTRY DATE MO 8 DAY 19 YR 64

Specimen Urine for radiometric uranium determination

Background 2 c/m
Counter Efficiency % 51
Recovery, % 76

Urine Vol., ml. 175
Gross 4 c/m
Net ± 0.95 error 2 ± 6.2
Disintegrations/min. 0

BSL-17-A

MISC.

RICHARD J. HENRY, M.D.

Figure B-6. Example Bioassay Data Sheet, BioScience, Uranium Radiometric.

ATTACHMENT B
EXAMPLE INTERNAL DOSIMETRY RECORD DOCUMENTS
Page 11 of 20

The following is an edited version of the description of the form provided by ETEC:

Report for uranium radiometric results from BioSicence Laboratories. These are the "record of entry" reports from the vendor laboratory that processed urine and fecal bioassay results.

These records should be correlated to the bioassay summary record (Individual Keysort Card [Figure B-2]). Two versions were used (see Figure B-7 for a side by side comparison).

The individual's name is usually written in pencil in the upper left margin. The Source No. generally lists a sample number. The Report Date is the date the results were available.

The number in the Patient block is the sample number to identify the person.

The meaning of the No. Spec. Type entry is not known at this time.

Entry Date refers to the date that the sample was received for processing. In general it is one to two days after the date the sample was obtained by Rocketdyne from the individual. This date is obtained from the Individual Keysort Card.

Background refers to the background on the alpha counter (generally counts/hour).

Counter efficiency is the efficiency of the counter used.

Recovery is the efficiency of the chemical recovery during the chemical preparation of the sample.

Urine Volume (ml) is the amount of sample actually analyzed.

Gross is the counter results UNCORRECTED for background.

Net - 0.95 error is the NET count rate (gross - bkgd) with a 95% statistical error indicated. If the error exceeds the net count, it is considered a non-detect. If the net exceeds the error, then it is considered a detect. The basis for this calculation of error (i.e., count times, normal or Poisson, etc.) is not available.

Disintegrations/min is the net activity determined for the sample.

In some cases, one will find a value of "XX dpm/day" written in. This is the sample result pro-rated to a daily excretion rate using a 1500 ml assumption for daily urine volume and a 135 g/day rate for feces.

Disintegrations are calculated by (Net CPM) / (Counter Efficiency/100) * (Recovery/100) [Assuming Efficiencies are written in percentage; fractional "percentages" are more likely true efficiency fractions and would not need conversion from percentage.]

By dividing the uranium radiometric values by the uranium fluorometric values, one can derive a specific activity for the sample. This specific activity can then be used to estimate the general enrichment level of the uranium the individual was exposed to. For simplicity, for low intakes, ETEC practice was to model the intake as pure U-234 (which results in overestimation of "true" dose).

ATTACHMENT B
EXAMPLE INTERNAL DOSIMETRY RECORD DOCUMENTS
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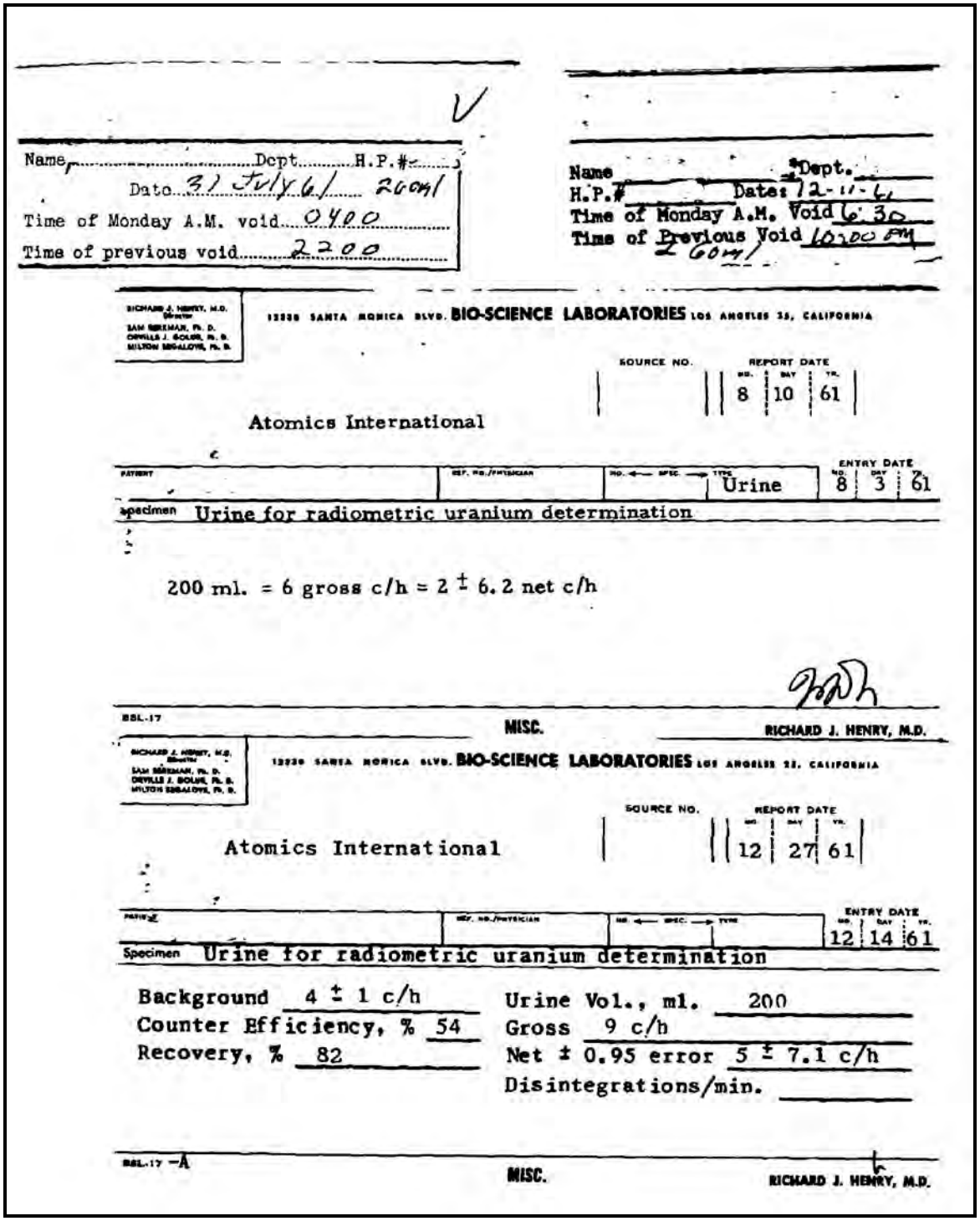

V

Name ..... Dept. ..... H.P.# .....
Date 31 July 61 200ml
Time of Monday A.M. void 0400
Time of previous void 2200

Name ..... Dept. .....
H.P.# ..... Date: 12-11-61
Time of Monday A.M. Void 6:30
Time of Previous Void 10:00 PM
200ml

RICHARD J. HENRY, M.D. Director
SAM SHERMAN, Ph. D.
ORVILLE J. SOLAR, Ph. D.
MILTON SHOLOVE, Ph. D.

12338 SANTA MONICA BLVD. **BIO-SCIENCE LABORATORIES** LOS ANGELES 25, CALIFORNIA

SOURCE NO. | REPORT DATE MO. 8 DAY 10 YR. 61

Atomics International

PATIENT | REF. NO./PHYSICIAN | NO. ← SPEC. → TYPE Urine | ENTRY DATE MO. 8 DAY 3 YR. 61

Specimen Urine for radiometric uranium determination

200 ml. = 6 gross c/h = 2 ± 6.2 net c/h

BSL-17 MISC. RICHARD J. HENRY, M.D.

RICHARD J. HENRY, M.D. Director
SAM SHERMAN, Ph. D.
ORVILLE J. SOLAR, Ph. D.
MILTON SHOLOVE, Ph. D.

12338 SANTA MONICA BLVD. **BIO-SCIENCE LABORATORIES** LOS ANGELES 25, CALIFORNIA

Atomics International

SOURCE NO. | REPORT DATE MO. 12 DAY 27 YR. 61

PATIENT | REF. NO./PHYSICIAN | NO. ← SPEC. → TYPE | ENTRY DATE MO. 12 DAY 14 YR. 61

Specimen Urine for radiometric uranium determination

Background 4 ± 1 c/h
Counter Efficiency, % 54
Recovery, % 82

Urine Vol., ml. 200
Gross 9 c/h
Net ± 0.95 error 5 ± 7.1 c/h
Disintegrations/min. _____

BSL-17 -A MISC. RICHARD J. HENRY, M.D.

Figure B-7. Example Bioassay Data Sheets, BioScience, Uranium Radiometric, Showing Bottle Tags.

ATTACHMENT B
EXAMPLE INTERNAL DOSIMETRY RECORD DOCUMENTS
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The following is an edited version of the description of the form provided by ETEC:

These are two more examples of the BioScience UR records described elsewhere.

Note that these records include bottle tags listing the exact times of sample collection.

ATTACHMENT B
EXAMPLE INTERNAL DOSIMETRY RECORD DOCUMENTS
Page 14 of 20

URANIUM FLUOROMETRIC

Date Received 6-8-66

Specimen Number B

(A-B) (C) / (D) (E) = ug/sample analyzed

(4.8 - 1.5) (0.5ug) / (4695) (.80) =

A sample reading
B blank reading
C ug uranium in standard
D standard reading
E analytical yield
Signature M Lardy

UNITED STATES TESTING CO., INC.
RICHLAND, WASHINGTON

Remarks:

.66 ug / 1500 ml

4.4×10^{-4}

$\pm$ 0.8×10^{-4}

ug/ 1 ml. analyzed

* ~t 2 σ

Figure B-8. Example Bioassay Data Sheet, UST, Uranium Fluorometric.

ATTACHMENT B
EXAMPLE INTERNAL DOSIMETRY RECORD DOCUMENTS
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The following is an edited version of the description of the form provided by ETEC:

This is the vendor record for results for uranium fluorometric sampling from U. S. Testing.

The calculations are self-explanatory.

See previous discussion at BioScience UR record description for additional data.

Specimen Number contains the individual's Health Physics Identification Number.

A sample number was generally written in the center box.

Date Received is the date the sample arrived at the laboratory. This was generally one to two days after the collection date. The actual date the sample was obtained is on the Individual Keysort Card. These sheets should be correlated to that card.

ATTACHMENT B
EXAMPLE INTERNAL DOSIMETRY RECORD DOCUMENTS
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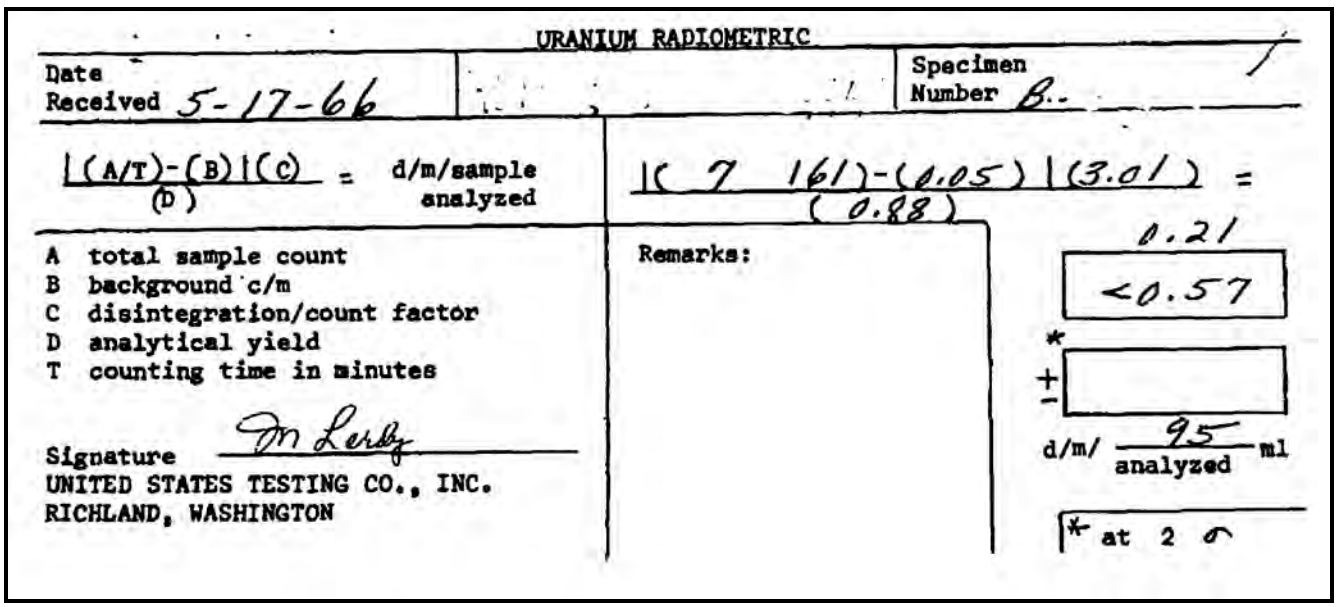

URANIUM RADIOMETRIC

Date Received 5-17-66

Specimen Number B..

$\frac{|(A/T)-(B)|(C)}{(D)}$ = d/m/sample analyzed

$\frac{|(7\ 161)-(0.05)|(3.01)}{(0.88)}$ = 0.21

A total sample count
B background c/m
C disintegration/count factor
D analytical yield
T counting time in minutes

Remarks:

<0.57

* ±

d/m/ $\frac{95}{\text{analyzed}}$ ml

Signature Jm Lerby
UNITED STATES TESTING CO., INC.
RICHLAND, WASHINGTON

* at 2 σ

Figure B-9. Example Bioassay Data Sheet, UST, Uranium Radiometric.

ATTACHMENT B
EXAMPLE INTERNAL DOSIMETRY RECORD DOCUMENTS
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The following is an edited version of the description of the form provided by ETEC:

This is the vendor record for results for uranium radiometric sampling from U. S. Testing.

The calculations are self-explanatory.

See previous discussion at BioScience UR record description for additional data.

Specimen Number contains the individual's Health Physics Identification Number.

A sample number was generally written in the center box.

Date Received is the date the sample arrived at the laboratory. This was generally one to two days after the collection date. The actual date the sample was obtained is on the Individual Keysort Card. These sheets should be correlated to that card.

ATTACHMENT B
EXAMPLE INTERNAL DOSIMETRY RECORD DOCUMENTS
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MIXED FISSION PRODUCTS

Date Received 12-8-67

Specimen Number B-

$$\frac{[(A/T) - (B)]\,(C)}{(D)} = \text{d/m/sample analyzed}$$

$$\frac{[(20/10)-(0.99)]\,(2.14)}{(0.65)} = 3.3$$

A total sample counts
T counting time in minutes
B background c/m
C disintegrations/count factor based on Sr-Y-90
D analytical yield

Remarks:

< 4.0

* + −

Signature R. H. Anderson
UNITED STATES TESTING CO., INC.
RICHLAND, WASHINGTON

d/m/ 200 analyzed ml

*at 2 σ

Figure B-10, Example Bioassay Data Sheet, UST, Mixed Fission Products

ATTACHMENT B
EXAMPLE INTERNAL DOSIMETRY RECORD DOCUMENTS
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The following is an edited version of the description of the form provided by ETEC:

This is the vendor record for results for mixed fission product sampling from U. S. Testing.

The calculations are self-explanatory.

Specimen Number contains the individual's Health Physics Identification Number.

A sample number was generally written in the center box.

Date Received is the date the sample arrived at the laboratory. This was generally one to two days after the collection date. The actual date the sample was obtained is on the Individual Keysort Card. These sheets should be correlated to that card.

These measurements were generally defaulted to Cs-137. In addition, one should be aware that Cs-137 was generally assumed to be accompanied by Sr-90 in a 50%/50% mix. Analyses of MFP results should therefore account both for Cs-137 exposures and Sr-90 exposures as being co-incident.

Note: Based upon laboratory analyses directly comparing Cs-137 and Sr-90 values, the mixture ratio for Cs-137 and Sr-90 during decontamination and decommissioning operations after 1991 were in a ratio of approximately 85% Cs-137 to 15% Sr-90. In 1999, the ratio published in the site TBD was 90% Cs-137 to 10% Sr-90 for SSFL locations other than Bldg. 4059.

ATTACHMENT B
EXAMPLE INTERNAL DOSIMETRY RECORD DOCUMENTS
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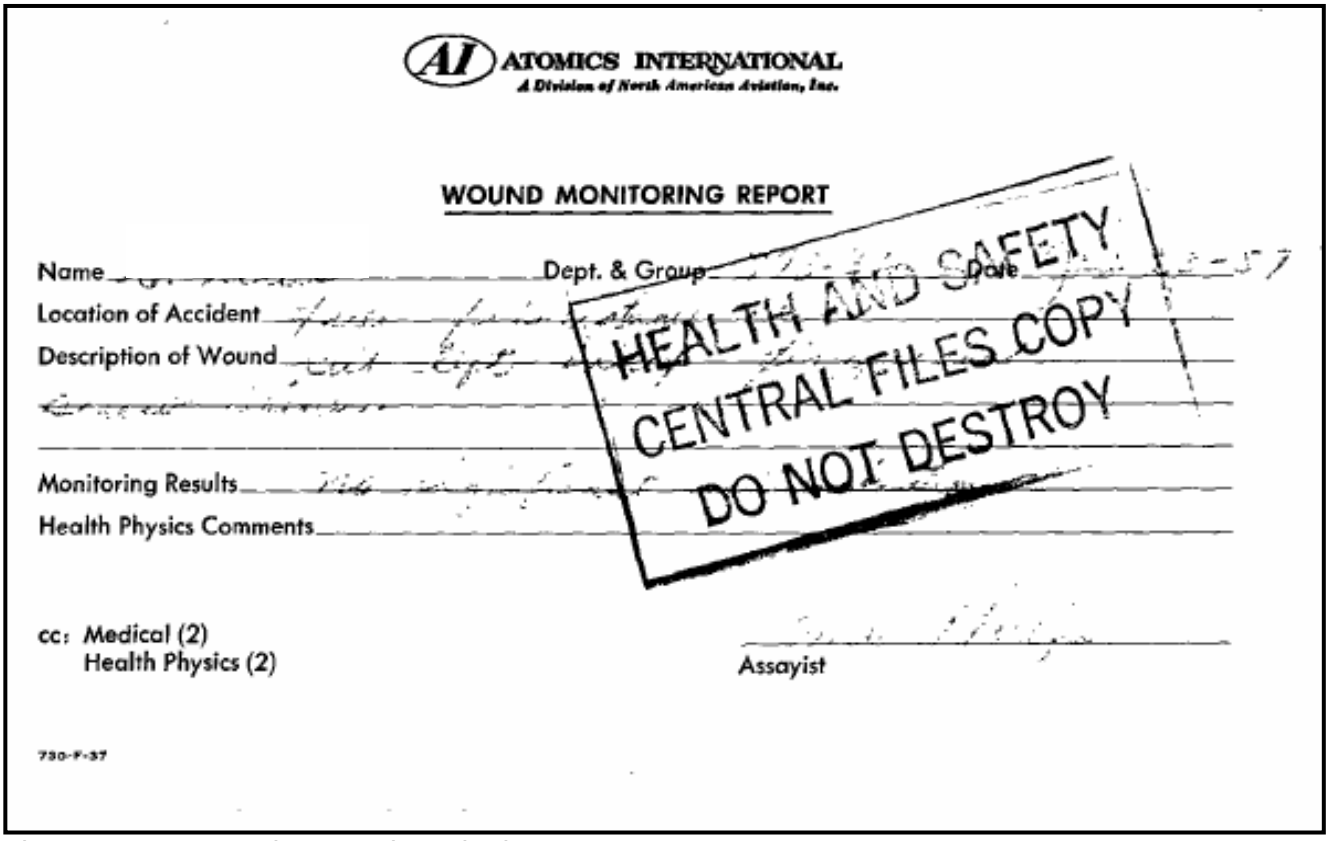

ATOMICS INTERNATIONAL
A Division of North American Aviation, Inc.

WOUND MONITORING REPORT

Name _____ Dept. & Group _____ Date _____

Location of Accident _____

Description of Wound _____

Monitoring Results _____

Health Physics Comments _____

HEALTH AND SAFETY
CENTRAL FILES COPY
DO NOT DESTROY

cc: Medical (2)
Health Physics (2)

Assayist

730-F-37

Figure B-11. Example Wound Monitoring Report.

ATTACHMENT C
ADJUSTMENT TO HELGESON NUCLEAR SERVICES U-235 LUNG COUNTS, 1981–1983

Measured (µg U-235)	Actual (µg U-235)
32.5	0
40	8.6
50	20.2
60	31.7
70	43.2
80	54.8
90	66.3
100	77.8
110	89.4
120	101
130	112
140	124
150	135
160	147
170	159
180	170
190	182
200	193
210	205
220	216
230	228
240	239
245	245

ATTACHMENT D
ISOTOPIC COMPOSITION OF ATR-ETR FUEL USED IN THE POWDER ROOM, 1966–1967

Date received	Isotopic composition (weight percent)			
	U-234	U-235	U-236	U-238
6/27/1966	0.579	93.2	0.304	5.917
6/27/1966	0.571	93.2	0.301	5.928
10/12/1966	0.516	93.15	0.310	6.010
10/13/1966	0.656	93.16	0.295	5.900
11/11/1966	0.563	93.14	0.289	6.000
12/8/1966	0.591	93.11	0.326	5.970
12/8/1966	0.627	93.13	0.286	5.960
1/20/1967	0.563	93.09	0.305	6.020
2/6/1967	0.570	93.15	0.320	5.956
2/8/1967	0.580	93.17	0.290	5.961
2/9/1967	0.550	93.14	0.310	6.001
2/10/1967	0.570	93.19	0.270	5.966
2/11/1963	0.630	93.14	0.300	5.929
3/11/1967	0.660	93.17	0.300	5.872
3/16/1967	0.610	93.13	0.350	5.912
4/16/1967	0.500	93.13	0.340	6.030
6/19/1967	0.557	93.23	0.324	5.860
6/22/1967	0.582	93.16	0.304	5.950
6/23/1967	0.600	93.14	0.290	5.972
6/23/1967	0.580	93.12	0.294	6.010
6/23/1967	0.570	93.14	0.301	6.000
Average	0.582	93.152	0.305	5.958
Standard deviation	0.039	0.033	0.019	0.048
Specific activity (µCi/g)	6,263.14	2.16	64.70	0.340
Fractional specific activity (µCi/g-U)	36.46	2.01	0.20	0.020
Fractional specific activity (Bq/g-U)	1.35E+06	7.44E+04	7.31E+03	7.50E+02

Data from Tschaeche (1968c)

ATTACHMENT D (CONTINUED)
ISOTOPIC COMPOSITION OF URANIUM USED FOR ATR FUEL FABRICATION, 1979-1982

Date	Isotopic composition (weight percent)			
	U-234	U-235	U-236	U-238
10/18/1979	1.006	93.145	0.525	5.234
1/29/1980	1.145	93.142	0.498	5.348
5/12/1980	1.007	93.141	0.472	5.379
12/3/1980	1.007	93.142	0.476	5.374
7/28/1981	1.004	93.14	0.455	5.648
11/2/1981	1.002	93.136	0.455	5.407
3/2/1982	0.9977	93.141	0.448	5.414
Average	1.024	93.141	0.476	5.401
Standard deviation	0.0534	0.0027	0.0276	0.1246
Specific activity (µCi/g)	6,263.14	2.16	64.70	0.340
Fractional specific activity (µCi/g-U)	64.141	2.012	0.308	0.018
Fractional specific activity (Bq/g-U)	2.37E+06	7.44E+04	1.14E+04	6.79E+02

Data from Moore (1979, 1980a,b,c, 1981a,b, 1982).

ATTACHMENT E
DETECTION LIMITS FOR GAMMA SCANS OF BIOASSAY SAMPLES, 1967–1975

Isotope(s)	Detection limit (dpm/sample)
Na-24	80
Co-60	75
Sc-46	60
Cs-134	175
Ba-La-140	250
Co-58	50
K-40	500
Ru-l06	500
Zn-65	100
Cu-64	300
Mn-56	70
Mn-54	60
Ga-72	140
Zr-Nb-95	20
Cs-137	50
As-76	120
Ru-103	50
Zn-69m	50
I-131	60
Cr-51	400
Np-239	80
Ce-141	60
Ce-Pr-144	500