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ACRONYMS AND ABBREVIATIONS

AERD	Atomic Energy Research Department
AI	Atomics International
Bq	becquerel
Ci	curie
cm	centimeter
dpm	disintegrations per minute
EEOICPA	Energy Employees Occupational Illness Compensation Program Act
ETEC	Energy Technology Engineering Center
GIF	Gamma Irradiation Facility
hr	hour
m	meter
mg	milligram
mrem	millirem
NAA	North American Aviation
NIOSH	National Institute for Occupational Safety and Health
NRC	U.S. Nuclear Regulatory Commission
RIHL	Rockwell International Hot Laboratory (later simply Hot Laboratory)
RMDF	Radioactive Material Disposal Facility
RMHF	Radiation Materials Handling Facility
s	second
SNAP	Systems for Nuclear Auxiliary Power
SRE	Sodium Reactor Experiment
SSFL	Santa Susana Field Laboratory
TBD	technical basis document
TLD	thermoluminescent dosimeter
U.S.C.	United States Code
W	watt
WBNS	Water Boiler Neutron Source
yr	year
§	section

4.1 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 (42 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.

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 environmental doses from these processes.

4.2 OPERATIONS OVERVIEW

Occupational environmental dose refers to exposures received by workers while on the site but outside the facilities at the Energy Technology Engineering Center (ETEC) from facility discharges to the atmosphere, from ambient external radiation originating in the facilities, and from inadvertent ingestion of site-generated radionuclides. This TBD describes the estimated annual intakes for inhalation exposure, estimated radiation dose from ambient external exposures, and estimated inadvertent ingestions at ETEC. Environmental measurements do not distinguish the source of emissions and therefore reflect air concentrations from nearby as well as distant sources. The estimates of emissions were useful in filling in some gaps in measurement data and were critical to estimating exposures before the start of comprehensive and routine measurement data reports.

The receptors of concern are ETEC employees who did not wear external dosimetry or who were not monitored for internal exposures. To provide the basis for estimating the environmental dose for years when monitoring did not occur or was not sufficient to apply to coworkers, this TBD provides annual intakes and ambient external dose from 1952 to 1999.

ETEC, operated by The Boeing Company and its processors, played an important role in the development of the U.S. nuclear program. ETEC operations involved research, development, and testing of nuclear reactors, nuclear support operations, and non-nuclear energy research and development (Sapere and Boeing 2005, p. 2-2).

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 Downey, California. On April 21, 1952, a small aqueous homogeneous reactor was constructed in an area that is now a loading dock; an environmental monitoring program was apparently initiated during 1952 as well. 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 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 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 used at the Downey facility were apparently for industrial radiography and were not likely a 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 research and development 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 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 in 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 (NRC) (Rutherford 2002). Subsequent occupants of the building would be unlikely to be exposed to contamination levels of importance for internal dose reconstruction.

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, were transferred to the new headquarters 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 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 both Building 101 and Building 104. These operations used 2% to 93% enriched uranium metal and composites.

Two other operations in Building 104 at the De Soto facility used 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 ceased in the late 1980s; and, following 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 Lab, analyzed radioactive samples for helium content. Operations ceased in 1995 and 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 Lab for unrestricted use. In all, 13 separate radiation surveys were conducted in Buildings 101 and 104 to demonstrate that no residual contamination existed that would be of interest for dose reconstruction (Boeing 2003).

SSFL consisted of four administrative areas as well as a buffer zone. Area IV was established by NAA at SSFL in 1953 as a nuclear research and development facility[1]. Starting in 1954 several nuclear reactors and critical assemblies were built and operated in Area IV. The Agency for Toxic Substances and Disease Registry (ATSDR) investigated radiological incidents occurring at SSFL (ATSDR 1999) and stated: "*Of all these incidents, only the Sodium Reactor Experiment (SRE) Fuel Damage incident, commonly known as "The Meltdown," resulted in a measurable release of radioactive material into the environment."* The SRE was a sodium-cooled graphite-moderated reactor in Building 143, which in 1959 had a loss of coolant that resulted in damage to 13 fuel assemblies (Sapere and Boeing 2005, pp. 2-1, 2-3, 2-5). Radioactive noble gases reached the outer containment and 28 Ci of radioactive noble gases ^{133}Xe and ^{85}Kr were released in a controlled manner and low concentrations to limit the potential dose (Rutherford 2005). A government-owned contractor-operated organization was formed to conduct non-nuclear research at the site. It was renamed ETEC in 1978. Most nuclear research programs and operations ceased in 1988. All research ended in 1998. ETEC was given the job of decontaminating and decommissioning the former nuclear facilities. Many of the facilities used in Area IV have been decontaminated and decommissioned, but some of the work is ongoing.

For simplicity, the remainder of this document uses ETEC to refer to all of the historical organizations that operated at the De Soto Avenue and Vanowen Building facilities in Canoga Park, Downey, and Area IV of the SSFL. Activities conducted at all of these ETEC sites and facilities are evaluated in this report.

4.3 INHALATION OF ONSITE AIRBORNE RADIONUCLIDES

The environmental monitoring program at Area IV was established in May 1954 before construction of the first radiological facility (Sapere and Boeing 2005, p. 3-13), with emphasis on soil, vegetation, and water sampling in the environment around Area IV. The program had been initiated in 1952 at the Downey Facility but was terminated in the move to Canoga Park (Moore, Fisher, and Rowe 1962).

[1] More recently, buildings in Area IV have had the number 4 added to the beginning of the building designation. For example, the original Building 055 would now be referred to as Building 4055.

From 1959 to present, ambient gross beta activity in air has been measured continuously in five locations. From 1963 on, gross alpha activity was measured. Ambient air samples are currently analyzed for isotopic content (Sapere and Boeing 2005, p. 3-15). No information is available for the Downey or Canoga Park facilities other than the De Soto facilities; information for the De Soto facilities is not available until ambient data are reported together with Area IV data in environmental monitoring reports. The ETEC ambient air monitoring data through 1999 are reported as gross alpha and gross beta without identification of specific radionuclides. In all cases the gross alpha and gross beta concentrations are a small fraction of the applicable limits and are indistinguishable from background. Because of this, stack emission data were used as the primary basis for inhalation intake estimates from onsite atmospheric radionuclide concentrations.

Average annual gross alpha/gross beta concentrations in facility stack emissions were the basis for estimating potential worker environmental inhalation intakes. Most of the available ETEC stack emission data include annual average gross alpha and gross beta concentrations at the stack point of release. Years with data vary by facility, but gross alpha/gross beta concentration information is available for most years between 1971 and 1999.. For years where there were no emission data, the facility-specific average for available years was used. The average of yearly data was used rather than the median because the average concentration was higher. In many cases the stack emission concentrations of gross alpha and gross beta are indistinguishable from the ambient background concentrations, indicating manmade radionuclide releases were very low.

Identification of specific radionuclides released from various facilities in stack emissions are available in data from 1988 to 1999 and were used to characterize radionuclide emissions for all years. There are no radionuclide-specific data from earlier years. The radionuclide-specific data showed that naturally occurring radionuclides were major constituents of radionuclide emissions, with ^{210}Po a main constituent of the gross alpha concentration and ^{40}K and ^{7}Be major constituents of the gross beta detected. Any uranium and thorium detected is assumed to originate from facility activities, even though it could be naturally occurring.

4.3.1 Stack Effluent Concentrations, 1988 to 1999

Radionuclide-specific, gross alpha and gross beta stack emission data are available for this period for some facilities. In Area IV these facilities include mainly Building 020[2] [RIHL, Rockwell International Hot Laboratory, later simply Hot Laboratory) and Buildings 021-022 [Radioactive Materials Disposal Facility (RMDF), Radioactive Materials Handling Facility (RMHF)], with limited information for T024 [Systems for Nuclear Auxiliary Power (SNAP) Environmental Test Facility], and T059 (SNAP 8 Development Reactor) (Moore 1989; Robinson 1998; Gaylord 1990; Rutherford 1999; Rockwell 1991, 1992, 1993, Rockwell 1996; Boeing 2000; Tuttle 1997).

For De Soto the only facility for which there is emission data is Building 004/104 (Hot Analytical Chemistry). The most complete information, and the majority of the radioactive releases, comes from Building 020 for Area IV. These effluent data show that the majority of the release during most years was from naturally occurring radionuclides: ^{210}Po for alpha-emitters and ^{40}K and ^{7}Be for beta-emitters. No data were available for 1993 and 1994.

Table 4-1 lists the radionuclides identified as being released from each facility during this period. Naturally occurring radionuclides are not included; isotopes of thorium and uranium could be naturally occurring but are not so considered in ETEC effluent reports.

[2]As noted in an earlier footnote, the original Building 020 may be referred to as Building 4020.

Table 4-1. Identified radionuclides in facility air effluents, 1988 to 1999.

Facility	Alpha emitters	Beta emitters
020, Hot Lab	Pu-238, Pu-239, U-234, U-235, U-238, Th-228, Th-230, Th-232, Am-241	Co-60, Sr-90, Cs-137
021-022, RMDF/RMHF	Pu-238, Pu-239, U-234, U-235, U-238, Th-228, Th-230, Th-232, Am -241	Co-60, Sr-90, Cs-137, H-3
T024, SNAP ETF	Th-230	Cs-137
T059, SNAP8DR	Pu-238, Pu-239, U-234, U-238, Am-241	Co-60, Cs-137
004/104, De Soto	Pu-238, Pu-239, U-234, U-235, U-238, Th-228, Th-230, Th-232, Am -241	Co-60, Sr-90, Cs-137, Pu-241

4.3.2 Stack Effluent Concentrations, 1971 to 1987

Gross alpha and gross beta effluent data are available for most years during this period for Area IV facilities Buildings 020, 021-022, and Building 055 (Nuclear Materials Development Facility), and limited data is available for Building 003 (Engineering Test Building) (Remley 1979, Eggleston 1983, 1984; Hart 1979, 1980a,b,c,d; Moore c. 1974; Heine 1973, Oldenkamp and Mills 1991; Tuttle 1985, 1986a,b,c, 1988a,b].

Data are available for De Soto Building 004/104 and 001/101 (Fuel Fabrication). Because of lack of other information, Building 001/101 was assumed to have the same radionuclide composition in effluents as Building 004/104. This is a conservative assumption because, as a fuel fabrication facility, no mixed fission products would be expected in the gross beta emissions. In years where data were not available, stack concentrations were assumed to be the average yearly gross alpha and gross beta concentrations in stack effluents from years 1971 to 1999 for which data were available.

4.3.3 Stack Effluent Concentrations, 1959 to 1970

No data were available for this period, but because operations and activities during this period took place in many of the same facilities addressed in later years, the average yearly gross alpha and gross beta concentrations in stack effluents from 1971 to 1999 were assumed for each of these years. Furthermore, the average percentage that each identified radionuclide contributed to the gross alpha or gross beta concentration determined from 1988 to 1999 data was applied to each of these years to make radionuclide-specific stack concentration estimates. Because no radionuclide-specific data were available for Building 055, the characteristic effluent radionuclide percentage from Buildings 024 and 059 were applied. Radionuclide-specific data were also unavailable for De Soto Building 001/101, so the characteristic effluent radionuclides percentages from Building 004/104 were applied.

4.3.4 Stack Effluent Concentrations, Before 1959

No data are available for the period before 1959; the facilities operating during this period were mainly small reactor facilities. The SRE, the largest reactor at ETEC, operated from 1954 to 1964 in Area IV. It is possible that reactor operations before 1959 could have released fission product activities similar to those after 1959. Therefore, fission product releases from Area IV from 1954 to 1958 are assumed to be the same as 1959 to 1964. There would have been few activities or operations with potential for release of alpha-emitting radionuclides; therefore, alpha-emitting radionuclides from 1954 to 1958 are assumed to be 10 percent of alpha-emitting radionuclide releases from 1959 to 1964. Operations at the De Soto Avenue facilities did not begin until 1959. Activities at the Downey site from 1952 to 1955 and Canoga Park Vanowen Building from 1954 to 1960 were limited and releases are conservatively assumed to be the same as the 1954 Area IV releases.

4.3.5 Estimates of Potential Inhalation Intake, 1952 to 1999

The average annual stack effluent concentrations described in Section 4.3.3 were used as the basis for estimating occupational environmental inhalation intakes from facility effluent releases. The facility effluent concentrations for each year were summed. Workers were assumed to be exposed for 2,000 hr/yr with a breathing rate of 3.3×10^{-4} m^3/s (ICRP 1975). Because the stack effluent concentrations were at the point of release, a further reduction factor of 0.01 was taken to account for the lessened overall intake due to contribution from multiple, widely spaced facilities; atmospheric dispersion of stack effluent over the course of a year's exposure, and building wake effects.

Noble gases are not included in the estimates of intake because they are not included in emission monitoring data and because the potential dose from radioactive noble gases is very small. In addition and with one exception, acute release events that contribute negligibly to the annual average radionuclide concentration are not included. The exception is the inclusion of mixed fission product releases from a burned fuel slug event occurring in Building 020 in July 1975, which had a noticeable effect on the average annual gross beta concentrations (Hart 1979).

Table 4-2 lists the estimated environmental inhalation intakes in Area IV. Table 4-3 lists the estimated environmental inhalation intakes at the Canoga Park De Soto Avenue and Vanowen Building facilities and the Downey site. In the absence of measurements or studies, NIOSH guidance requires that default solubility classes and particle size values from the International Commission on Radiological Protection be used (NIOSH 2002, pp. 15–16). Facility-specific solubility and particle size data are not available. The most claimant favorable material form and solubility should be selected for each radionuclide at all sites and facilities.

Table 4-2. Annual occupational environmental radionuclide inhalation, Area IV (Bq/yr).

	Radionuclide												
Year	Pu-239	Pu-238	Am-241	U-234	U-235	U-238	Th-228	Th-232	Th-230	Co-60	Sr-90	Cs-137	H-3
1999	1.1E-04	2.4E-06	6.4E-06	2.0E-04	4.1E-06	4.7E-04	8.0E-06	9.1E-06	1.8E-03	1.1E-02	3.0E-02	9.0E-02	2.1E-02
1998	5.3E-04	4.2E-05	1.4E-05	5.7E-04	3.0E-05	7.7E-04	5.8E-05	1.2E-04	4.3E-03	3.5E-02	3.1E-02	1.4E-01	2.3E-01
1997	1.6E-04	1.4E-05	5.4E-06	1.6E-04	1.1E-05	3.9E-04	2.2E-05	4.0E-05	6.6E-04	4.9E-03	1.1E-03	8.1E-03	2.9E-01
1996	6.5E-05	2.0E-06	5.7E-06	1.3E-04	4.9E-06	4.9E-04	9.6E-06	9.2E-06	6.1E-04	6.3E-03	1.3E-03	1.4E-02	1.9E-01
1995	1.9E-04	1.0E-05	1.6E-05	3.3E-04	1.8E-05	1.5E-03	3.5E-05	4.0E-05	6.3E-04	8.3E-03	2.7E-03	2.0E-02	2.9E-01
1994	1.6E-04	1.4E-05	5.4E-06	1.6E-04	1.1E-05	3.9E-04	2.2E-05	4.0E-05	6.6E-04	1.4E-02	3.1E-02	9.5E-02	2.3E-01
1993	1.6E-04	1.4E-05	5.4E-06	1.6E-04	1.1E-05	3.9E-04	2.2E-05	4.0E-05	6.6E-04	1.4E-02	3.1E-02	9.5E-02	2.3E-01
1992	9.8E-05	7.9E-06	6.0E-06	1.3E-04	9.2E-06	5.2E-04	1.8E-05	2.6E-05	1.6E-04	1.9E-03	7.0E-04	4.1E-03	9.3E-02
1991	2.1E-04	1.7E-05	1.0E-05	2.6E-04	1.7E-05	8.0E-04	3.3E-05	5.2E-05	8.9E-04	5.4E-03	1.5E-03	1.3E-02	1.7E-01
1990	1.4E-04	7.6E-06	1.1E-05	2.3E-04	1.2E-05	9.7E-04	2.5E-05	2.9E-05	6.3E-04	4.4E-03	8.5E-04	1.1E-02	7.4E-02
1989	3.3E-05	1.1E-06	1.1E-06	5.2E-05	6.6E-07	5.5E-05	1.3E-06	2.8E-06	6.1E-04	5.6E-03	5.1E-04	1.1E-02	1.5E-01
1988	1.2E-04	1.0E-05	3.6E-06	1.2E-04	8.1E-06	2.5E-04	1.6E-05	3.0E-05	6.5E-04	7.2E-03	1.2E-03	1.5E-02	2.4E-01
1987	6.5E-05	3.9E-06	2.9E-06	9.1E-05	3.8E-06	2.1E-04	7.4E-06	1.2E-05	6.2E-04	1.5E-02	2.5E-03	2.7E-02	7.2E-01
1986	5.6E-05	5.2E-06	1.9E-06	5.6E-05	4.3E-06	1.4E-04	8.4E-06	1.5E-05	1.8E-04	1.5E-02	4.2E-03	3.4E-02	5.6E-01
1985	1.2E-04	8.2E-06	9.2E-06	1.8E-04	1.2E-05	8.2E-04	2.3E-05	2.9E-05	1.3E-04	1.5E-02	2.1E-02	7.0E-02	5.5E-01
1984	7.7E-05	8.0E-06	2.3E-06	6.7E-05	6.3E-06	1.7E-04	1.2E-05	2.3E-05	1.0E-04	1.0E-03	8.2E-04	3.9E-03	1.6E-02
1983	4.9E-05	4.9E-06	1.0E-06	4.2E-05	3.4E-06	6.1E-05	6.7E-06	1.4E-05	1.9E-04	1.5E-03	3.7E-04	3.5E-03	4.8E-02
1982	2.0E-04	2.3E-05	2.4E-06	1.2E-04	1.5E-05	1.0E-04	3.0E-05	6.4E-05	1.4E-04	1.8E-03	2.7E-03	9.6E-03	2.5E-02
1981	8.4E-05	8.9E-06	2.0E-06	6.9E-05	6.6E-06	1.3E-04	1.3E-05	2.5E-05	1.7E-04	4.9E-03	3.0E-03	1.5E-02	1.7E-01
1980	2.1E-04	1.4E-05	6.9E-06	2.7E-04	1.0E-05	4.3E-04	2.0E-05	3.8E-05	2.3E-03	9.6E-03	3.2E-03	2.5E-02	2.3E-01
1979	1.3E-04	1.3E-05	3.3E-06	1.0E-04	1.0E-05	2.3E-04	2.0E-05	3.8E-05	1.9E-04	5.3E-03	6.2E-03	2.1E-02	2.3E-01
1978	1.1E-04	1.1E-05	3.0E-06	9.3E-05	8.4E-06	2.2E-04	1.7E-05	3.1E-05	2.0E-04	1.3E-02	9.2E-03	4.1E-02	4.9E-01
1977	1.3E-04	1.4E-05	2.2E-06	8.9E-05	9.6E-06	1.2E-04	1.9E-05	3.8E-05	2.1E-04	5.9E-03	1.7E-03	1.5E-02	1.6E-01
1976	2.4E-04	2.8E-05	3.7E-06	1.6E-04	1.9E-05	1.9E-04	3.7E-05	7.7E-05	2.7E-04	4.0E-03	6.9E-04	1.0E-02	5.1E-02
1975	4.7E-04	5.5E-05	5.9E-06	2.9E-04	3.6E-05	2.6E-04	7.1E-05	1.5E-04	4.1E-04	1.6E-01	6.1E-01	1.7E+00	5.1E-01
1974	1.6E-04	1.4E-05	5.4E-06	1.6E-04	1.1E-05	3.9E-04	2.2E-05	4.0E-05	6.6E-04	1.4E-02	3.1E-02	9.5E-02	2.3E-01
1973	1.6E-04	1.7E-05	3.8E-06	1.3E-04	1.3E-05	2.6E-04	2.5E-05	4.9E-05	2.1E-04	3.9E-03	1.3E-03	7.5E-03	2.3E-01
1972	1.6E-04	1.6E-05	4.8E-06	1.4E-04	1.3E-05	3.6E-04	2.6E-05	4.7E-05	2.2E-04	1.0E-02	3.0E-02	8.7E-02	6.2E-02
1971	1.6E-04	1.4E-05	5.4E-06	1.6E-04	1.1E-05	3.9E-04	2.2E-05	4.0E-05	6.6E-04	1.4E-02	3.1E-02	9.5E-02	2.3E-01
1970	1.6E-04	1.4E-05	5.4E-06	1.6E-04	1.1E-05	3.9E-04	2.2E-05	4.0E-05	6.6E-04	1.4E-02	3.1E-02	9.5E-02	2.3E-01
1969	1.6E-04	1.4E-05	5.4E-06	1.6E-04	1.1E-05	3.9E-04	2.2E-05	4.0E-05	6.6E-04	1.4E-02	3.1E-02	9.5E-02	2.3E-01
1968	1.6E-04	1.4E-05	5.4E-06	1.6E-04	1.1E-05	3.9E-04	2.2E-05	4.0E-05	6.6E-04	1.4E-02	3.1E-02	9.5E-02	2.3E-01
1967	1.6E-04	1.4E-05	5.4E-06	1.6E-04	1.1E-05	3.9E-04	2.2E-05	4.0E-05	6.6E-04	1.4E-02	3.1E-02	9.5E-02	2.3E-01
1966	1.3E-04	1.4E-05	4.4E-06	1.2E-04	1.1E-05	3.4E-04	2.2E-05	4.0E-05	5.1E-05	1.1E-02	3.1E-02	8.8E-02	2.3E-01
1965	1.3E-04	1.4E-05	4.4E-06	1.2E-04	1.1E-05	3.4E-04	2.2E-05	4.0E-05	5.1E-05	1.1E-02	3.1E-02	8.8E-02	2.3E-01
1964	1.3E-04	1.4E-05	4.4E-06	1.2E-04	1.1E-05	3.4E-04	2.2E-05	4.0E-05	5.1E-05	1.1E-02	3.1E-02	8.8E-02	2.3E-01
1963	1.3E-04	1.4E-05	4.4E-06	1.2E-04	1.1E-05	3.4E-04	2.2E-05	4.0E-05	5.1E-05	1.1E-02	3.1E-02	8.8E-02	2.3E-01
1962	1.3E-04	1.4E-05	4.4E-06	1.2E-04	1.1E-05	3.4E-04	2.2E-05	4.0E-05	5.1E-05	1.1E-02	3.1E-02	8.8E-02	2.3E-01
1961	1.3E-04	1.4E-05	4.4E-06	1.2E-04	1.1E-05	3.4E-04	2.2E-05	4.0E-05	5.1E-05	1.1E-02	3.1E-02	8.8E-02	2.3E-01
1960	1.3E-04	1.4E-05	4.4E-06	1.2E-04	1.1E-05	3.4E-04	2.2E-05	4.0E-05	5.1E-05	1.1E-02	3.1E-02	8.8E-02	2.3E-01
1959	1.3E-04	1.4E-05	4.4E-06	1.2E-04	1.1E-05	3.4E-04	2.2E-05	4.0E-05	5.1E-05	1.1E-02	3.1E-02	8.8E-02	2.3E-01
1958	1.3E-05	1.4E-06	4.4E-07	1.2E-05	1.1E-06	3.4E-05	2.2E-06	4.0E-06	5.1E-06	1.1E-02	3.1E-02	8.8E-02	2.3E-01
1957	1.3E-04	1.4E-05	4.4E-06	1.2E-04	1.1E-05	3.4E-04	2.2E-05	4.0E-05	5.1E-05	1.1E-02	3.1E-02	8.8E-02	2.3E-01
1956	1.3E-04	1.4E-05	4.4E-06	1.2E-04	1.1E-05	3.4E-04	2.2E-05	4.0E-05	5.1E-05	1.1E-02	3.1E-02	8.8E-02	2.3E-01
1955	1.3E-04	1.4E-05	4.4E-06	1.2E-04	1.1E-05	3.4E-04	2.2E-05	4.0E-05	5.1E-05	1.1E-02	3.1E-02	8.8E-02	2.3E-01
1954	1.3E-04	1.4E-05	4.4E-06	1.2E-04	1.1E-05	3.4E-04	2.2E-05	4.0E-05	5.1E-05	1.1E-02	3.1E-02	8.8E-02	2.3E-01

Table 4-3. Annual occupational environmental radionuclide inhalation at De Soto Avenue, Vanowen Building, and Downey facilities. (Bq/yr).

	Radionuclide												
Year	Pu-239	Pu-238	Am-241	U-234	U-235	U-238	Th-228	Th-232	Th-230	Co-60	Sr-90	Cs-137	Pu-241
1954-1999, De Soto Avenue													
1999	0	0	0	0	0	0	0	0	0	0	0	0	0
1998	9.1E-07	1.1E-05	2.6E-05	2.1E-04	1.4E-05	2.6E-05	1.1E-04	4.0E-05	4.5E-05	3.9E-05	3.6E-04	1.8E-04	0
1997	3.7E-06	4.3E-05	1.1E-04	8.6E-04	5.6E-05	1.1E-04	4.4E-04	1.6E-04	1.8E-04	3.4E-05	3.1E-04	1.5E-04	0
1996	1.8E-07	2.1E-06	5.0E-06	4.1E-05	2.7E-06	5.0E-06	2.1E-05	7.8E-06	8.7E-06	1.1E-05	1.0E-04	4.9E-05	0
1995	3.6E-07	4.2E-06	1.0E-05	8.4E-05	5.5E-06	1.0E-05	4.3E-05	1.6E-05	1.8E-05	1.9E-05	1.7E-04	8.4E-05	6.5E-03
1994	7.6E-06	8.9E-05	2.2E-04	1.8E-03	1.2E-04	2.2E-04	9.1E-04	3.3E-04	3.7E-04	5.1E-05	4.8E-04	2.3E-04	0
1993	7.6E-06	8.9E-05	2.2E-04	1.8E-03	1.2E-04	2.2E-04	9.1E-04	3.3E-04	3.7E-04	5.1E-05	4.8E-04	2.3E-04	0
1992	1.4E-06	1.6E-05	4.0E-05	3.2E-04	2.1E-05	3.9E-05	1.7E-04	6.1E-05	6.8E-05	2.6E-05	2.4E-04	1.2E-04	0
1991	1.2E-06	1.4E-05	3.4E-05	2.8E-04	1.8E-05	3.4E-05	1.4E-04	5.3E-05	5.9E-05	4.9E-05	4.6E-04	2.2E-04	0
1990	1.8E-06	2.1E-05	5.1E-05	4.2E-04	2.7E-05	5.1E-05	2.1E-04	7.9E-05	8.8E-05	9.5E-05	8.9E-04	4.3E-04	0
1989	7.6E-06	8.9E-05	2.2E-04	1.8E-03	1.2E-04	2.2E-04	9.1E-04	3.3E-04	3.7E-04	5.1E-05	4.8E-04	2.3E-04	0
1988	4.2E-06	4.9E-05	1.2E-04	9.8E-04	6.4E-05	1.2E-04	5.0E-04	1.9E-04	2.1E-04	6.8E-05	6.4E-04	3.1E-04	0
1987	5.2E-06	6.1E-05	1.5E-04	1.2E-03	7.9E-05	1.5E-04	6.2E-04	2.3E-04	2.6E-04	4.9E-05	4.6E-04	2.2E-04	0
1986	1.4E-06	1.6E-05	4.0E-05	3.2E-04	2.1E-05	4.0E-05	1.7E-04	6.2E-05	6.9E-05	5.8E-05	5.4E-04	2.6E-04	0
1985	3.3E-06	3.9E-05	9.5E-05	7.7E-04	5.1E-05	9.5E-05	4.0E-04	1.5E-04	1.6E-04	4.2E-05	3.9E-04	1.9E-04	0
1984	1.0E-05	1.2E-04	3.0E-04	2.4E-03	1.6E-04	2.9E-04	1.2E-03	4.5E-04	5.1E-04	5.5E-05	5.1E-04	2.5E-04	0
1983	4.5E-04	5.3E-03	1.3E-02	1.0E-01	6.8E-03	1.3E-02	5.4E-02	2.0E-02	2.2E-02	6.6E-04	6.2E-03	3.0E-03	0
1982	1.0E-05	1.2E-04	2.9E-04	2.3E-03	1.5E-04	2.9E-04	1.2E-03	4.5E-04	5.0E-04	4.7E-05	4.4E-04	2.2E-04	0
1981	2.2E-05	2.6E-04	6.3E-04	5.1E-03	3.3E-04	6.3E-04	2.6E-03	9.7E-04	1.1E-03	1.4E-04	1.3E-03	6.3E-04	0
1980	3.9E-06	4.6E-05	1.1E-04	9.1E-04	6.0E-05	1.1E-04	4.7E-04	1.7E-04	1.9E-04	1.9E-04	1.8E-03	8.6E-04	0
1979	1.8E-04	2.1E-03	5.0E-03	4.1E-02	2.7E-03	5.0E-03	2.1E-02	7.8E-03	8.7E-03	2.0E-04	1.8E-03	8.9E-04	0
1978	2.6E-06	3.0E-05	7.5E-05	6.0E-04	4.0E-05	7.4E-05	3.1E-04	1.1E-04	1.3E-04	1.3E-04	1.2E-03	5.7E-04	0
1977	2.2E-06	2.6E-05	6.2E-05	5.1E-04	3.3E-05	6.2E-05	2.6E-04	9.6E-05	1.1E-04	1.2E-04	1.1E-03	5.3E-04	0
1976	3.0E-04	3.5E-03	8.5E-03	6.9E-02	4.5E-03	8.5E-03	3.6E-02	1.3E-02	1.5E-02	3.6E-04	3.4E-03	1.6E-03	0
1975	2.4E-05	2.8E-04	6.9E-04	5.6E-03	3.7E-04	6.9E-04	2.9E-03	1.1E-03	1.2E-03	1.2E-04	1.1E-03	5.4E-04	0
1974	9.4E-05	1.1E-03	2.7E-03	2.2E-02	1.4E-03	2.7E-03	1.1E-02	4.1E-03	4.6E-03	1.9E-04	1.8E-03	8.9E-04	0
1973	9.7E-06	1.1E-04	2.8E-04	2.2E-03	1.5E-04	2.8E-04	1.2E-03	4.3E-04	4.8E-04	7.7E-05	7.2E-04	3.5E-04	0
1972	9.4E-05	1.1E-03	2.7E-03	2.2E-02	1.4E-03	2.7E-03	1.1E-02	4.1E-03	4.6E-03	1.9E-04	1.8E-03	8.9E-04	0
1971	9.4E-05	1.1E-03	2.7E-03	2.2E-02	1.4E-03	2.7E-03	1.1E-02	4.1E-03	4.6E-03	1.9E-04	1.8E-03	8.9E-04	0
1970	9.4E-05	1.1E-03	2.7E-03	2.2E-02	1.4E-03	2.7E-03	1.1E-02	4.1E-03	4.6E-03	1.9E-04	1.8E-03	8.9E-04	0
1969	9.4E-05	1.1E-03	2.7E-03	2.2E-02	1.4E-03	2.7E-03	1.1E-02	4.1E-03	4.6E-03	1.9E-04	1.8E-03	8.9E-04	0
1968	9.4E-05	1.1E-03	2.7E-03	2.2E-02	1.4E-03	2.7E-03	1.1E-02	4.1E-03	4.6E-03	1.9E-04	1.8E-03	8.9E-04	0
1967	9.4E-05	1.1E-03	2.7E-03	2.2E-02	1.4E-03	2.7E-03	1.1E-02	4.1E-03	4.6E-03	1.9E-04	1.8E-03	8.9E-04	0
1966	9.4E-05	1.1E-03	2.7E-03	2.2E-02	1.4E-03	2.7E-03	1.1E-02	4.1E-03	4.6E-03	1.9E-04	1.8E-03	8.9E-04	0
1965	9.4E-05	1.1E-03	2.7E-03	2.2E-02	1.4E-03	2.7E-03	1.1E-02	4.1E-03	4.6E-03	1.9E-04	1.8E-03	8.9E-04	0
1964	9.4E-05	1.1E-03	2.7E-03	2.2E-02	1.4E-03	2.7E-03	1.1E-02	4.1E-03	4.6E-03	1.9E-04	1.8E-03	8.9E-04	0
1963	9.4E-05	1.1E-03	2.7E-03	2.2E-02	1.4E-03	2.7E-03	1.1E-02	4.1E-03	4.6E-03	1.9E-04	1.8E-03	8.9E-04	0
1962	9.4E-05	1.1E-03	2.7E-03	2.2E-02	1.4E-03	2.7E-03	1.1E-02	4.1E-03	4.6E-03	1.9E-04	1.8E-03	8.9E-04	0
1961	9.4E-05	1.1E-03	2.7E-03	2.2E-02	1.4E-03	2.7E-03	1.1E-02	4.1E-03	4.6E-03	1.9E-04	1.8E-03	8.9E-04	0
1960	9.4E-05	1.1E-03	2.7E-03	2.2E-02	1.4E-03	2.7E-03	1.1E-02	4.1E-03	4.6E-03	1.9E-04	1.8E-03	8.9E-04	0
1959	9.4E-05	1.1E-03	2.7E-03	2.2E-02	1.4E-03	2.7E-03	1.1E-02	4.1E-03	4.6E-03	1.9E-04	1.8E-03	8.9E-04	0
1954-1960, Vanowen Building (all years are the same)													
	1.3E-04	1.4E-05	4.4E-06	1.2E-04	1.1E-05	3.4E-04	2.2E-05	4.0E-05	5.1E-05	1.1E-02	3.1E-02	8.8E-02	2.3E-01
1952-1955, Downey Site (all years are the same)													
	1.3E-04	1.4E-05	4.4E-06	1.2E-04	1.1E-05	3.4E-04	2.2E-05	4.0E-05	5.1E-05	1.1E-02	3.1E-02	8.8E-02	2.3E-01

4.4 AMBIENT EXTERNAL RADIATION

Ambient external radiation dose information for ETEC is available for 1975 to 1999. During the later years (1974 to 1989) of the active nuclear programs at ETEC, the ambient radiation monitoring program used complex bulb dosimeters (CaF_2:Mn). This was justified by the amount of nuclear materials handled in the operations in Area IV and De Soto and by the low levels of radiation in the environment. At the termination of all nuclear work in 1989, that level of monitoring was no longer needed and efforts were directed toward simplifying the program. The initial step included using the well-established LiF thermoluminescent dosimeters (TLDs) for personnel monitoring in radiation work.

While these dosimeters were well suited to measuring exposures in the range of interest for compliance with occupational radiation regulations (doses above background), they were somewhat insensitive for environmental measurements because the resolution in terms of dose uses increments of 10 mrem/quarter. Using these dosimeters demonstrated that environmental exposures did not reach regulatory limits, but it provided limited information on the actual exposure rates around the facilities and in the neighboring environment (Rutherford 1999).

In addition to the LiF TLDs, Rocketdyne began deploying environmental TLDs in the last quarter of 1995 that used an aluminum oxide (called *sapphire*) chip. These TLDs were capable of determining doses in increments of 0.1 mrem (in comparison to 10 mrem for the LiF badges). Proper use of the control badges supplied with these dosimeters allowed elimination of the natural and transportation exposure that occurred before, during, and after the deployment of the environmental dosimeters to measure the ambient radiation. This permitted accurate determination of the net exposure received while the environmental TLDs were exposed in the field to the ambient radiation (Rutherford 1999).

The following sections describe estimates of environmental external dose during five periods from 1954 to 1999; estimates for possible earlier exposures in Downey and Canoga Park are also provided. Workers were assumed to be exposed for 2,000 hr/yr. A baseline estimate of the environmental external dose was developed for each period, and additional dose for specific facilities during specific years was developed based on aerial survey information and specific facility fence line exposure rate information. In all cases, the minimum dose above background is 10 mrem/yr.

4.4.1 External Dose, 1996 to 1999

The sensitivity of the aluminum oxide chip in environmental TLDs is accounted for during this period. Average annual dose from offsite background dosimeters is subtracted from the average annual dose of environmental TLDs in Area IV and De Soto to provide an estimate of the baseline occupational environmental dose at the respective sites. The baseline occupational environmental dose for both De Soto and Area IV is 10 mrem/yr. In addition, environmental dosimeters near Buildings 021-022 (RMDF/RMHF) are consistently higher than the average. This is stated to be reflective of normal operations at the RMHF (Rutherford 1999). Occupational environmental doses near this facility are increased by 10 mrem/yr during this period to 20 mrem/yr.

4.4.2 External Dose, 1990 to 1995

Relatively insensitive LiF chips were used in dosimeters during this period, so the baseline dose during this period was based on the minimum dosimeter sensitivity of 10 mrem/quarter or 40 mrem/yr. This minimum sensitivity baseline results in a higher annual dose than subtraction of offsite background dose from onsite environmental dosimeter results. The occupational environmental external dose near Buildings 021-022 is increased by 10 mrem/yr because of elevated dosimeter readings.

4.4.3 External Dose, 1975 to 1989

External doses during this period of active nuclear operations at ETEC were measured using the complicated CaF_2:Mn bulb dosimeters. Baseline external occupational environmental doses during this period were estimated by subtracting the average annual offsite background dose from the maximum annual onsite dose at De Soto and Area IV. Baseline doses were at the minimum of 10 mrem/yr except for 1978, when radiation from the Building 621 area, described below, was high enough that the baseline dose for the site was doubled to 20 mrem/yr.

Three documents provided quarterly facility fence line exposure rates for Building 20 (RIHL) and Buildings 021-022 (RMDF/RMHF) during 1986, 1987, and 1988 (Tuttle 1988a,b, 1989). These measurements showed the exposure rates around Building 020 to be background, but the exposure rates around the RMDF to be elevated above the baseline. These measurements were used as the basis for elevating external occupational environmental doses around the RMDF along with results of an aerial radiological survey in 1978 (Jobst 1979). The aerial survey represents a snapshot in time, but shows elevated exposure rates around the RMDF, around Building 143 the Sodium Reactor Experiment (SRE) Facility from storage of radioactive material, and around the Building 621 Radioactive Accountable Waste Storage Building from storage of radioactive liquid in a nearby tank vault. Because of the nature of the survey, these elevated exposure rates were applied only to the year 1978 for the SRE facility. However, because elevated exposure around the RMDF had occurred in other years, the aerial survey results were included in determining an average elevated dose rate for the RMDF. Because Building 621 was part of the RMDF complex, the RMDF elevated results were applied throughout this period to Building 621 as well.

4.4.4 External Dose, 1959 to 1974

No data are available for the period from 1959 to 1974, but many of the same nuclear operations continued during this period. The facilities not considered are mainly small reactor facilities, which are typically well shielded. The SRE, the largest reactor at ETEC, operated from 1954 to 1964. Building 621 was constructed in the mid-1960s (Sapere and Boeing 2005, Volume 2, Part 5). Both Building 020, the Hot Lab, and Buildings 021-022, the RMDF, began operations in 1959. Based on operating information from 1975 to 1989, estimates were made of the occupational environmental external dose for 1959 to 1974. The baseline dose above background in Area IV was assumed to be doubled to 20 mrem/yr to account for reactor operations during this period. The baseline dose in De Soto remains at 10 mrem/yr. Facility-specific dose during these years was assumed to be elevated to 20 mrem/yr for De Soto Building 001 and elevated to 80 mrem/yr for the RMDF in Area IV. During the years of SRE operations, a facility-specific dose of 40 mrem/yr was assumed for Building 143.

4.4.5 External Dose Before 1959

No data are available for the period before 1959; as noted above the facilities were mainly small reactor facilities, which are typically well shielded. The SRE, the largest reactor at ETEC, operated from 1954 to 1964 in Area IV. The baseline dose above background in Area IV was assumed to be 20 mrem/yr to account for reactor operations during this period. The facility-specific dose for Building 143 was 40 mrem/yr for 1954 to 1959. Environmental external doses from 1952 to 1955 for Downey and from 1954 to 1960 for the Canoga Park Vanowen Building are estimated to be 10 mrem/yr based on the operation of small test reactors.

4.4.6 Estimates of External Dose, 1952 to 1999

Table 4-4 lists the occupational environmental external dose estimates for Area IV, De Soto Avenue, Vanowen, and Downey sites for applicable periods from 1952 to 1999. Some facility-specific doses are provided for De Soto and in Area IV.

Table 4-4. Annual external occupational environmental dose (mrem/yr).

	De Soto		Area IV			
		Facility-specific dose		Facility-specific dose		
Year	Baseline[a]	Fuel Fab Bldg 001	Baseline[a]	RMDF Bldg 021-022	SRE Bldg 143	RAWSB Bldg 621
1999	--	--	10	20		
1998	10		10	20		
1997	10		10	20		
1996	10		10	20		
1995	40		40	50		
1994	40		40	50		
1993	40		40	50		
1992	40		40	50		
1991	40		40	50		
1990	40		40	50		
1989	10		10	20		20
1988	10		10	40		40
1987	10		10	60		60
1986	10		10	160		160
1985	10		10	70		70
1984	10		10	70		70
1983	10	20	10	70		70
1982	10	20	10	70		70
1981	10	20	10	70		70
1980	10	20	10	70		70
1979	10	20	10	70		70
1978	10	20	20	40	90	180
1977	10	20	10	70		70
1976	10	20	10	70		70
1975	10	20	10	70		70
1974	10	20	20	80		80
1973	10	20	20	80		80
1972	10	20	20	80		80
1971	10	20	20	80		80
1970	10	20	20	80		80
1969	10	20	20	80		80
1968	10	20	20	80		80
1967	10	20	20	80		80
1966	10	20	20	80		80
1965	10	20	20	80		
1964	10	20	20	80	40	
1963	10	20	20	80	40	
1962	10	20	20	80	40	
1961	10	20	20	80	40	
1960	10	20	20	80	40	
1959	10	20	20	80	40	
1958	--		20		40	
1957	--		20		40	
1956	--		20		40	
1955	--		20		40	
1954	--		20		40	
Vanowen						
1954-1960	10					
Downey						
1952-1955	10					

a. Minimum above-background baseline external dose of 10 mrem/yr.

4.5 INGESTION

Two sources of occupationally derived radioactive material for ingestion are possible:

- Potable water
- Removable surface contamination from floors, objects, or soils via the hand to the mouth

Potable water is not a source of occupational radioactive material at ETEC because the ETEC facilities used either bottled water from an offsite vendor (Moore, Fisher, and Rowe 1962) or the city water supply. SSFL continues to use bottled water.

Ingestion of removable surface contamination is negligible because the intake is proportional to the amount of dirt consumed. In NRC NUREG/CR-5512, Kennedy et al. (1992, p. 6.14) assume the amount of soil ingested by an adult involved in building renovation work would be 10 mg/hr. More sedentary work would produce a smaller consumption, perhaps 50 mg/day. The other factor is the activity concentration in the ingested dirt. It is reasonable to expect that, outside of radiologically controlled areas, the removable surface contamination level or activity concentration in soil should on average approach background levels. The surface contamination level is unlikely to average an order of magnitude more than the background level for the entire workplace for the entire year.

Kennedy et al. (1992, Sections 3.2.5 and 6.3.2) define methods and assumptions for use in calculation of doses to individuals occupying a facility contaminated with residual radioactive material. An equation for calculation of ingestion dose is included. For purposes of EEIOCPA, dose is reconstructed by different algorithms than those used in this report, so only the ingestion intake is estimated here.

The method equation can be summarized in a word description as:

> *Intake equals the exposure duration multiplied by the effective transfer rate of removable contamination from surface to hands then to mouth and multiplied by the average removable surface contamination level.*

More specifically:

$$Ingestion = t \times GO \times Act$$

where

Ingestion	=	ingestion intake in Bq/yr
t	=	annual exposure duration (2,000 hr/yr)
GO	=	ingestion transfer rate of removable surface activity during building occupancy (1×10^{-4} m^2/hr)
Act	=	removable surface contamination (activity) level (Bq/m^2)

Therefore, full-time occupancy (2,000 hr/yr) in an area with a removable surface activity of a radionuclide at the level of 1 Bq/m^2 would result in an intake of 0.2 Bq of that radionuclide per year.

Inadequate data are available to estimate the median ingestion each year and at each facility, but estimates or bounding assumptions are necessary for dose reconstruction. To conservatively bound ingestion from occupational environmental sources, a worker is assumed to be a full-time occupant (2,000 hr/yr) of an area contaminated with ^{239}Pu and ^{90}Sr at 100% of their current removable contamination limits. These NRC-defined contamination limits (AEC 1974) have been essentially

unchanged for decades. The plutonium limit is 20 dpm/100 cm^2 or 33 Bq/m^2, the strontium limit is 200 dpm/100 cm^2 or 330 Bq/m^2. The resultant ingestion intake for all years and all ETEC facilities is bounded at a level of 6.7 Bq/yr of ^{239}Pu and 67 Bq/yr of ^{90}Sr, as shown in Table 4-5. The most claimant favorable material form and solubility should be selected for each radionuclide.

Table 4-5. Maximum site-wide median intakes (Bq/yr) via inadvertent soil ingestion.

Site	Years	Radionuclide[a]	
		Pu-239	**Sr-90**
Area IV	1954 - 1999	6.7	67
De Soto	1959 - 1999	6.7	67
Vanowen	1954- 1960	-	67
Downey	1952 - 1956	-	67

a. Radionuclides assumed for consistency with the predominant ones used for internal dose calculation. Pu-239 is representative of alpha-emitting radionuclides, Sr-90 representative of beta/gamma-emitting radionuclides.

4.6 UNCERTAINTY

Uncertainty is lower in more recent years and higher in earlier years when less data are available. There are few available documents with estimates of uncertainty for ETEC environmental measurements. No information is available on uncertainty of stack effluent concentrations. Many results are reported as less-than values, which indicates concentrations below the minimum detectable concentration. Qualitative estimates of inhalation intakes are 50% at 1 standard deviation from 1988 to 1999, 100% from 1971 to 1987, and 200% from 1959 to 1970.

Some of the best estimates of uncertainty are associated with the later years for external dose. Qualitative estimates of occupational environmental external dose are 20% at 1 standard deviation from 1996 to 1999; 50% from 1990 to 1995; 100% from 1975 to 1989; and 200% from 1959 to 1974.

Qualitative estimate of ingestion intakes are 100% at 1 standard deviation.

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GLOSSARY

alpha radiation

Positively charged particle emitted from the nuclei of some radioactive elements. An alpha particle consists of two neutrons and two protons (a helium nucleus) and has an electrostatic charge of +2. Alpha particles have low penetrating power and a short range (a few centimeters in air). Outside the body, the most energetic alpha particle generally fails to penetrate the dead layers of cells covering the skin or a sheet of paper. Alpha particles emitted by radionuclides inside the body are a more significant health risk.

becquerel (Bq)

International System unit of radioactivity equal to 1 disintegration per second; 1 curie equals 37 billion (3.7×10^{10}) Bq.

beta radiation

Charged particle emitted from some radioactive elements with a mass equal to 1/1,837 that of a proton. A negatively charged beta particle is identical to an electron. A positively charged beta particle is a positron. Most of the direct fission products are (negative) beta emitters. Exposure to large amounts of beta radiation from external sources can cause skin burns (erythema), and beta emitters can be harmful inside the body. Thin sheets of metal or plastic can stop beta particles.

dosimetry

Measurement and calculation of internal and external radiation doses.

exposure

(1) In general, the act of being exposed to ionizing radiation. (2) Measure of the ionization produced by X- and gamma-ray photons in air in units of roentgens.

radioactivity

Disintegration of certain elements (e.g., radium, actinium, uranium, and thorium) accompanied by the emission of alpha, beta, gamma, and/or neutron radiation from unstable nuclei.

rem

Traditional unit of radiation dose equivalent that indicates the biological damage caused by radiation equivalent to that caused by 1 rad of high-penetration X-rays multiplied by a quality factor. The average American receives 360 millirem a year from background radiation. The sievert is the International System unit; 1 rem equals 0.01 sievert. The word derives from roentgen equivalent in man; rem is also the plural.

thermoluminescent dosimeter (TLD)

Device for measuring radiation dose that consists of a holder containing solid chips of material that, when heated by radiation, release the stored energy as light. The measurement of this light provides a measurement of absorbed dose.