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PUBLICATION RECORD

EFFECTIVE DATE	REVISION NUMBER	DESCRIPTION
05/31/2005	00	New document to establish the technical basis for the development of a radiation exposure profile for the Kansas City Plant. First approved issue. No training required. Initiated by Donald E. Bihl.
01/06/2006	00 PC-1	Approved page change revision. Incorporates changes for environmental ambient external and internal doses for unmonitored, non-radiological workers as a result of formal internal review and informal OCAS comments. Page change includes new content added to pages 7 and 22 (Sections 1.0 and 5.1.4). Removed definition of unmonitored worker on page 26 (Section 6.4.1). Minor revisions were made to pages 17, 18, 19, and 23 (Sections 4.0, 5.0, 5.1.2, 5.2 and 6.2.1). As a result of NIOSH formal review, incorporates Attachment A on pages 43 – 46, into Site Profile. This attachment describes technical basis of ambient environmental dose estimate for unmonitored workers. No sections were omitted. Retraining is not required. Initiated by Jack J. Fix. Approval: Document Owner: Signature on File 12/19/2005 Jack J. Fix, TBD Team Leader Signature on File 12/20/2005 Judson L. Kenoyer, Task 3 Manager Signature on File 12/28/2005 Kate Kimpan, Project Director Signature on File 01/06/2006 James W. Neton, Associate Director for Science

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ACRONYMS AND ABBREVIATIONS

AEC	U.S. Atomic Energy Commission
AMAD	activity median aerodynamic diameter
AP	anterior-posterior
DCF	dose conversion factor
DOE	U.S. Department of Energy
DOELAP	DOE Laboratory Accreditation Program
DU	depleted uranium
DXT	density times thickness
ESE	entrance skin exposure
GSD	geometric standard deviation
HVL	half-value layer
Hp(d)	personnel dose equivalent at depth *d* in tissue
ICRP	International Commission on Radiological Protection
ICRU	International Commission on Radiation Units and Measurements
IMBA	Integrated Modules for Bioassay Analysis
IREP	Interactive RadioEpidemiological Program
KCP	Kansas City Plant
MDL	minimum detectable level
MED	Manhattan Engineer District
NCRP	National Commission on Radiological Protection and Measurements
NIOSH	National Institute for Occupational Safety and Health
NNSA	National Nuclear Security Administration
NTA	nuclear track emulsion, type A
NVLAP	National Voluntary Laboratory Accreditation Program
OSL	optical stimulated luminescence
PA	posterior-anterior
PIC	pocket ionization chamber
RGD	radiation generating device
TLD	thermoluminescent dosimeter
U.S.C.	United States Code

1.0 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 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. § 73841(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. § 73841(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.

Section 2.0 of this Site Profile describes the Kansas City Plant (KCP) in Kansas City, Missouri. Section 3.0 discusses occupational medical doses from X-ray examinations, and Section 4.0 discusses onsite environmental doses. Sections 5.0 and 6.0 describe internal and external dosimetry, respectively, in relation to KCP.

2.0 SITE DESCRIPTION

KCP is on approximately 141 acres of the 300-acre Bannister Federal Complex 12 miles south of downtown Kansas City, Missouri, within the city limits. The Plant (Figure 1) currently shares the site with the Federal Aviation Administration, Defense Finance and Accounting Service, U.S. Marine Corps, General Services Administration, Internal Revenue Service, National Oceanic and Atmospheric Administration, and National Logistics Support Center. Built by the Navy during World War II to assemble engines for Navy fighter planes, the facility was operated by Pratt-Whitney from early 1943 until September 2, 1945. In 1947, Westinghouse began leasing the facility and the Fairfax Storage Company used part of the building as a warehouse for tires, raw rubber, sugar, and lumber. In February 1949, the U.S. Atomic Energy Commission (AEC; a DOE predecessor agency) asked the Bendix Corporation to manage KCP. Table 1 summarizes the history of the facility.

At present, KCP is a major operational facility administered by Honeywell Federal Manufacturing and Technologies/Kansas City. KCP produces non-nuclear weapons components for the nuclear weapons program of the National Nuclear Security Administration (NNSA). The principal products of KCP, since 1949, have included arming systems, fusing and firing systems, radars, power supplies, rubber, plastic and foam parts, and outer casings associated with non-nuclear components of nuclear weapons. KCP is currently the only NNSA facility for manufacturing non-nuclear components for nuclear weapons (DOE 1997).

The KCP Office reports directly to DOE Headquarters and has line management responsibility for manufacturing non-nuclear components for the NNSA. In recent years, DOE has relocated additional project roles from the Mound, Pinellas, and Rocky Flats Plant facilities to KCP (Reis 1998).

2.1 FACILITIES

The Main Manufacturing Building, constructed in 1942, is the largest facility on the KCP site with approximately 2.6 million square feet of contiguous space that houses the key manufacturing operations. The following buildings support the Main Manufacturing Building: Polymer Building, High Power Laboratory, Mold Heating and Cooling Building, Plating Building and Spray Mask Facility, Technology Transfer Center, Special Processes Building, and Manufacturing Support Building.

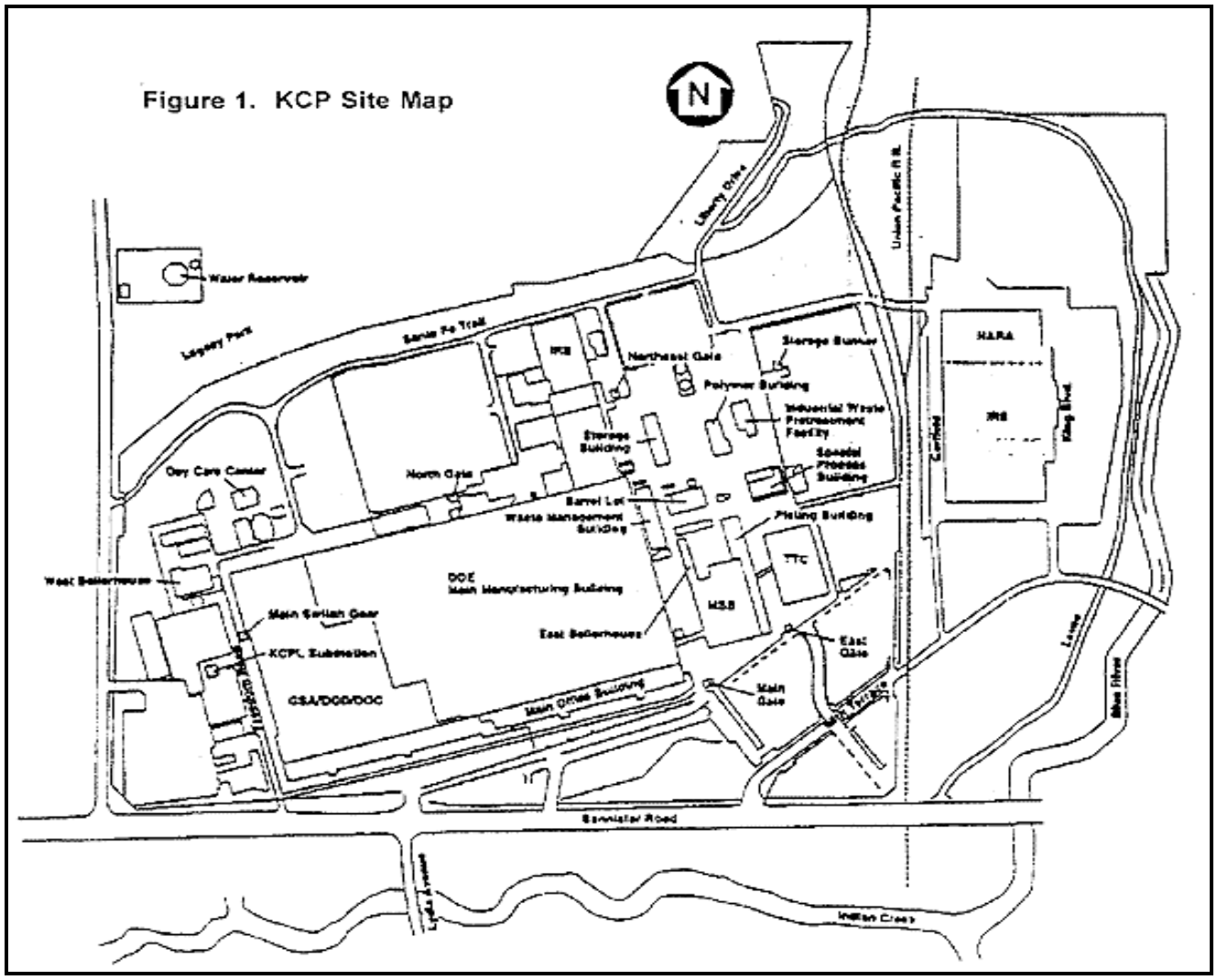

Figure 1. KCP site map.

Table 1. History of facility operations.

Year	Event
1943	Pratt-Whitney operated facility from early 1943 until September 2, 1945, to assemble engines for Navy fighter planes.
1947	Westinghouse began leasing facility, and Fairfax Storage Company used part of building as warehouse for tires, raw rubber, sugar, and lumber.
1949	KCP opens - AEC asks Bendix Corporation to manage facility and begin building non-nuclear components for nuclear weapons.
1982	Allied merger - Bendix merges with Allied Corporation.
1984	AlliedSignal merger - Allied merges with Signal companies to form AlliedSignal, Inc.
1993	Site consolidation - KCP designated as consolidated site for all non-nuclear components for nuclear weapons.
1999	Honeywell merger - AlliedSignal merges with Honeywell.

These facilities support the use of technologies for testing and evaluating engineering characteristics of materials and structures. Typical examinations include location of flaws, cracks, contaminations (chemical), and internal features; measurement of thickness, density, internal dimensions, and percentage contamination; sorting and identification of alloys; evaluation of heat treatment conditions;

and image enhancement. These capabilities are applied to a variety of objects including welds and forgings, foams, plastics, adhesives, composites, ceramics, and coatings as well as electrical component and mechanical assemblies.

Waste management operations at KCP consist mainly of hazardous nonradiological waste storage in preparation for offsite treatment or disposal. Operations can generate small quantities of low-level radioactive waste. Some waste is classified for national security reasons due to the nature of the generating process or constituents. Classified hazardous wastes are shipped off the site for sanitization and reclamation. KCP performs no onsite waste disposal. Treatment operations are limited to industrial wastewater pretreatment and selective recycling.

2.2 OCCUPATIONAL SAFETY PROGRAM

The Plant has maintained a formal procedures manual (KCD 1958); Procedure S-306 of this manual, dated November 24, 1958, defines health and safety measures for receiving, handling and storing the radioactive materials. Sections in the procedure address area access control, physical requirements, protective apparel, personal hygiene, material handling, production area, scrap and waste disposal, and injuries. This procedure was revised in 1960 (Foster 1960) and in 1962 (Foster 1962).

In 1965 an evaluation was conducted of 23 categories of workers for potential exposure to solvents, alcohols, etc., and to radiation. Table 2 summarizes this analysis for categories of workers potentially exposed to radiation (Schiltz 1965). There was a routine program of measuring uranium in urine (and lead in blood) for selected workers identified on a permanent access list to specified work areas (KCP undated). This list was used by the security, medical, and safety departments. Those on the list were authorized to work in areas in the Plant where radioactive material was handled (Nasca 2004a). This list is no longer in use.

Table 2. Categorization of KCP workers in special physical examination program working with radiation (Schiltz 1965).

Group	Description	Occupational exposures[a]
I	Radiation Workers - Departments 22, 34C, and 217-22	Could be exposed to uranium oxide and litharge (lead) powders 0-20 microns, small), and fumes from chlorinated hydrocarbon solvents.
IV	X-ray Workers, Department 213C	Could be exposed to X-ray radiation or gamma radiation from Cs-137 source (7 Ci). Some handle cured epoxy containing uranium dioxide and litharge.
VI	Radiation Workers, Departments 267, 268, and 280	Could be exposed to radiation in that they handle electronic tubes containing very small amount of radioactive material; they also handle drums that contain radioactive waste material (uranium dioxide) from Departments 22 and 34C.
X	Electron Beam Welders, Departments 45, 585, 201, 851, and Maintenance	Could be exposed to radiation from electron beam welders, which operate much like X-ray machines.
XV	Personnel Handling Neutron Sources	There is none of this work at Plant at present; however, Test Laboratory could be working with neutron sources in near future.
XX	Personnel Using Micro-Derm Thickness Gauge	Could be exposed to small amount of beta radiation.

a. Note that physicals normally include chest x-rays. Firemen and patrolmen receive physicals every 2 yr and have exit physicals. Cafeteria workers receive physicals every 3 yr but no exit physicals. All other groups receive an initial physical, annual periodic physical, and exit physical.

2.3 RADIATION MONITORING PROGRAM

The KCP Safety Department had primary responsibility for industrial hygiene and health physics. Potential exposure from X-rays and isotopic beta, photon, and neutron radiation sources represented a significant concern; however, the historical emphasis appears to have involved higher potential

nonradiological exposures to solvents such as acetone, methyl ethyl ketone, trichloroethylene, alcohol, toluene, and xylene; solvent fumes; epoxy resins and amine catalysts; polyester resins and peroxide catalysts, silicones, polysulfides, polyurethane, and tolylene isocyanate catalysts; and to fumes from acid, base, and salt chemicals of a wide variety. The focus appears to have involved ongoing evaluations of industrial, chemical, and radiological workplace hazards. There are records of frequent reviews of ionizing radiation equipment to evaluate the adequacy of shielding and interlocks. The earliest survey of radiation-generating devices (RGDs) apparently occurred in 1953 through the contracted services of a certified radiological physicist to evaluate the 1-MeV and 250-keV industrial X-ray units (Hoecker 1953). There are records of contamination surveys of the workplace using forms that imply a routine evaluation (Baldwin 1966).

2.4 RADIATION SOURCES

KCP operations utilize radiation as one of the analytical tools to accurately manufacture, fabricate, and inspect non-nuclear components of nuclear weapons. Table 3 summarizes the types of radioactive sources at KCP during 1964 and 1987. Significant differences involve the extensive earlier use of radium sources and the presence of the 230-Ci ^{137}Cs instrument calibration source in 1987. The primary radiation sources at KCP involve analytical laboratory technologies for the manufacturing and testing of electronic and mechanical devices. Fabricated materials, parts, and assemblies are examined for internal flaws and defects to ensure compliance with engineering specifications and requirements.

Table 3. Types of radiation sources.

Nuclide	Use	Lists		Predominant radiation	
		1964[a]	1987[b]	Type	Major energies, keV
C-14	Calibration source, beta scope, thickness gauges	x	x	Beta	45-156
Fe-55	Testing		x	Gamma	*bremsstrahlung* to 230
Co-60	Calibration sources	x		Gamma	1,170, 1,330
Ni-63	Gas chromatograph		x	Beta	67
Sr-90	Calibration, thickness gauges		x	Beta	546
Tc-99	Calibrations		x	Beta	292
Tl-204	Beta scope, thickness gauges		x	Beta	766
Cd-109	Plating thickness gauge		x	Gamma	88
Ba-133	Neutron generator		x	Gamma	30-382
Cs-137	Calibration sources	x	x	Beta	514-1,176
				Gamma	662
Pm-147	Calibration sources, thickness gauges		x	Beta	70-256
Tl-204	Beta scope, thickness gauges		x	Beta	766
Radium	Calibration sources, thickness gauges	x	x	Alpha	4,600-4,780
				Gamma	186-610
Uranium	Calibration sources			Alpha/gamma progeny radiations from Th-230, Ra-226, etc.	
Pu-239	Calibration sources, boron analysis	x	x	Alpha	5,110-5,160
				Gamma:	39-770

a. Source: Schiltz (1964).
b. Source: KCP (1987).

In addition to the radiation sources, radiographic inspections involve the use of numerous RGDs. Table 4 lists typical RGDs. The following list summarizes radiation-generating examinations.

Table 4. Typical RGDs (Nasca 2004b).

Devices	Energy (keV)	Types	Typical use	Period of use
Industrial X-ray units	50 to 2,000 (X-ray)	Rooms & Cabinets	Radiography of parts	1950s to 2004
DXT Device[a]	12 to 200 (X-ray)	Cabinets	Density thickness	1960s to 1980s
DXT Device	Cs-137 (1 Ci)	Cabinets	Density thickness	1960s to 1980s
Electron Beam Welders	35 to 150 (X-ray)	Cabinets	Welding small parts	1960s to 2004
Electron Microscopes	30 to 200 (X-ray)	Cabinets	Analysis	1960s to 2004
Electron Beam Vacuum Deposition Systems	10 (X-ray)	Cabinets	Plating metals	1960s to 2004
Neutron generators	14.7 MeV (neutron)	Open & Cabinets	Generate microsec pulsed radiation	1960s to 2004
Neutron source	Pu-239/Be (73 mCi)	Cabinets	Boron-10 analysis	1966 to 2004
Gamma camera	Co-60 (19 Ci)	Exposure Room	Test electronic products	1950s to 1960s
Febetron Accelerator	2,300 (X-ray) pulser	Cabinets	Irradiation of electronic components	1970s to 1980s
Cesium Irradiators	Cs-137 (230 Ci)	Exposure Room	Calibration of radiation detection instruments	1950s to 1980s
Medical X-ray	125 (X-ray)	Exposure Room	Patient diagnostic tests	1960s to 1990s
Electro Curtain	175 (X-ray)	Cabinet	Radiation curing of adhesives	1980s to 1990s

a. DXT refers to density times thickness measurement devices.

- **X-Ray Radiography.** KCP uses X-ray machines of different sizes and power to examine fabricated materials for structural defects such as voids or inclusions in weld or braze joints. These capabilities are also useful for failure analysis of electrical and mechanical assemblies.

- **Neutron Radiography.** This is similar to X-ray radiography except it uses neutrons rather than X-rays. The primary advantage of neutron radiography is that high-density materials such as iron, lead, and uranium are nearly transparent to neutrons while low-density materials such as organic compounds are highly absorptive to neutrons. This technique is particularly well suited to examining foams, encapsulants, or seals through relatively thick metal cases.

- **Gauging.** Beta and X-ray techniques are used under a variety of applications to measure the thickness of materials. Utilizing the backscatter of beta radiation, thicknesses of platings and coatings can be measured.

KCP operations involve numerous analytical capabilities. Those involving ionizing radiation include nuclear magnetic resonance spectrometry, electron probe microanalysis, and scanning electron microscopy. Radiation protection surveys of equipment installations appear to have been an integral component of the safety program. For example, workplace surveys during 1960 and 1964 contain detailed evaluations of the installations, normal operating conditions, routine survey instrumentation, enclosures, safety features, and electrical safeguards (Schiltz 1960; Harrison, Meunier, and Schiltz 1964a).

2.5 RADIATION DETECTION INSTRUMENTATION

The KCP Industrial Hygiene Department appears to have had extensive radiation detection instrument capabilities. Table 5 reproduces the equipment listed in a memorandum that was apparently first written on May 19, 1964, and updated in August 1965 and March 1966 (Schiltz 1966).

2.6 RADIOLOGICAL RECORDS

KCP has developed and maintained a radiological records database that contains records for all monitored worker exposures at KCP for all years of record. The database contains exposure data for about 4,400 workers. General trends in the recorded dose are:

Table 5. Radiation detection equipment list.[a]

Description	No.
Victoreen Low Range Beta-Gamma Survey Meter, Model #592B, Range 0 - 1,000 mR/hr	2
Gas Proportional Alpha Survey Meters, Eberline Model #PAC-3G, Range 0 - 100,000 cpm	4
Beta/Gamma Survey Meter, Nuclear Chicago Corporation, Model 2586, Range 0 – 25, 0 – 250, and 0 – 2,500 mR/h	1
Low-Range Beta-Gamma Survey Meters, Victoreen Model #389C, Range 0 - 20 mR/hr	4
Fast-Slow Neutron Survey Meter, Nuclear Chicago Model #2715, Range 10 - 10^4 neutrons/cm^2	2
Air Proportional Alpha Survey Meter, Eberline Instrument Corp., Model #PAC-1A, Range 0 - 100,000 cpm	1
Tritium Monitors, Atomic Accessories, Model #TSM91, Range 0 - 100,000 mCi/cm^3 of air of tritium	2
Gamma Survey Meters, Victoreen Model 61720, Range 0 - 500 R/hr	2
High Range Gamma Survey Meters, Eberline Instrument Corp., Model Gadora 1-B, Range 0 - 5,000 R/hr	2
Victoreen Condenser R-Meter Model #570 with assortment of probes for energy and dose	2
Victoreen Model 510 Roentgen rate meter with assortment of probes	1
Gamma Radiation Monitor, Eberline Instrument Corp., Model RM-2, Range 0 - 0.2 mR/hr	1
Beta-Gamma Count Rate Meters, Victoreen Model 743, Range 0 - 60,000 cpm.	2
Eberline Alpha Gas Proportional Floor Monitors, Model FM-2G, Range 0 - 100,000 cpm	2
Pocket Ionization Dosimeters, charge readers, and approximately 170 dosimeters with ranges from 0-100 R	
Tritium Monitor, Atomic Accessories Corp., Model #TSM-91-C, BKC #29549	1
Alpha, Beta, Gamma Proportional Counter, Nuclear Measurements, Model PC3, Serial No. 275, BKC #14346	1
Proportional Counter Converter, Nuclear Measurements Corp., Model #PCC12A, Serial No. 145, BKC #17098	1
Cutie Pie Survey Meter, Model 2510, Nuclear Chicago, 0 - 2500 mR/hr	1

a. Source: Schiltz (1966) copied to Emergency Radiation Monitoring Team File.

- Positive deep, shallow, and extremity doses were first recorded in 1950.
- Positive neutron doses were first recorded in 1966.
- Before about 1959, the recorded deep and shallow doses were essentially equal.
- Recorded extremity dose was higher in 1951 and 1952 than in any other year.
- Relatively high shallow dose (in comparison with deep dose) was recorded from 1959 to 1964 and during 1973.
- Recorded neutron dose is typically equal to recorded deep dose and to recorded shallow dose. The shallow dose is typically equal to the deep dose.
- There are comparatively few records with positive neutron dose.
- The only years with recorded positive uranium bioassay results are 1959 to 1971.

It appears that some dose has been recorded for nearly all occupational categories. The higher dose records were examined. Based on information from Nasca (2004d), some doses in the KCP radiological records system have been assigned to workers even though investigations at the time of measurement have shown unexplainable causes. Given the information available, these doses should be accepted as actual.

Dose reconstructors should use recorded doses from the KCP database to supplement the hard-copy original dosimeter processing information submitted by DOE because the respective claims are often difficult to read.

3.0 OCCUPATIONAL MEDICAL DOSE

KCP has a history of required routine medical examinations of workers that include routine chest X-rays. The medical X-ray units belonged to and were operated by the KCP Medical Department. There are handwritten records of chest X-rays being taken beginning on October 7, 1949 (Todd 2004a). KCP conducted routine examinations that included medical laboratory screening, pulmonary function testing, vision screening, audiometric screening, vital signs, health history updates, and physical examinations (Todd 2004b). Before 1993, a chest X-ray was offered with every physical examination. Beginning in 1993, chest X-rays were offered every 5 yr or more frequently if worker history or physical circumstances indicated a need. In 1997, medical X-ray services were outsourced; thereafter, X-rays were not routinely offered as part of any physical examination with the exception of beryllium surveillance. The beryllium surveillance includes a chest X-ray every 5 yr (Todd 2004c). There is no evidence that photofluorographic techniques were used based on a review of X-ray films from the 1950s and 1960s. The first date of a chest X-ray being examined was April 12, 1950.

Dose Reconstruction from Occupationally Related Diagnostic X-Ray Procedures (ORAU 2003a) contains background information on X-ray doses. There is limited information about X-ray machine parameters pertinent to doses received by KCP workers. A radiation protection survey of the main medical X-ray unit on January 16, 1964 (Harris, Meunier, and Schiltz 1964b) identified the following:

- General Electric, Model 11 DA1, Serial No. 620871LDXC, 80 kVp maximum, 100 mA maximum, 0.5-mm Al inherent and 2.0-mm Al external filters with a cone.

- For chest exposures the beam is normally directed toward the patient at a distance of 72 in. For extremities, the beam is directed downward approximately 28 to 36 in. from the tabletop.

- The normal operating settings (assumed for chest X-rays) were 70 to 870 kVp and 100 mA for 0.1 s.

- Measured exposure to evaluate collimation of the beam for the Series II (chest X-ray) examination (75 kVp, 100 mA, 0.2 s) showed a well-collimated beam with a maximum exposure of 26.3 mR.

- There were instructions on focusing the light field on the patient or extremities so the minimum body area would be exposed to the X-ray beam. The useful X-ray beam is further collimated by adjusting the size of the light field.

This implies an entrance skin exposure (ESE) of approximately 13 mR measured in 1964 for the usual chest posterior-anterior (PA) examination (i.e., 70 to 80 kVp, 100 mA, 0.1 s). Measurements of the ESE before 1964 were not found.

3.1 DOSE RECONSTRUCTION

The DOE dose reports for each claim should provide the schedule of actual X-ray examinations. KCP routine practices were apparently to conduct periodic physical examinations depending on the worker's job classification as follows:

- Firemen and patrolmen receive physicals every 2 yr and have exit physicals.
- Cafeteria workers receive physicals every 3 yr and have no exit physicals.
- All other groups receive an initial physical, annual periodic physicals, and an exit physical.

Dose reconstructors should use the following approach based on current information to assess dose from medical chest X-rays:

1. Before 1964, use the default values in Tables 3.3-1 and 4.0-1 of ORAU (2003b) for chest X-rays.

2. Beginning in 1964, use an ESE of 13 mR.

3. If there is no reasonable information in the claim documentation, assume an annual PA radiographic chest X-ray for each claimant's employment period from 1949 through June 1993.

4. If there is no reasonable information in the claim documentation, assume a PA radiographic chest X-ray for each 5-year period beginning in July 1993 for each claimant's employment period.

Organ Dose
Organ dose calculations for workers at KCP from 1949 to the present involve only the 14- by 17-in. PA chest film. The analysis in this Site Profile evaluated only doses from this technique. Other radiographic examinations of KCP employees that could have occurred are likely to be nonoccupational in the sense that they were associated with illness or injury and were not part of a routine examination process. There is no indication in the examined records that other diagnostic radiographic examinations routinely occurred as part of the occupational medical program.

Before 1964, the default organ dose recommendations for PA chest examinations in ORAU (2003a) were used (i.e., kerma of 2 rem) with minimal collimation due to the lack of detailed information concerning measurements of the ESE and collimation for KCP X-ray equipment. Historic documentation of measurements on January 16, 1964 does show a well-collimated system with an ESE of 13 mR. Tables 6 and 7 summarize organ doses for the pre-1964 and 1964-2004 periods, respectively, for PA chest examinations. This analysis assumed that an exposure of 1 R is equivalent to a kerma of 1 rad, 10 mGy, or 1 rem (ORAU 2003a).

3.2 UNCERTAINTY ANALYSIS

The description of error and uncertainty in ORAU (2003a) is directly applicable to evaluation of medical X-ray dose to KCP workers. Given the information available, and the apparent unavailability of measurements of the ESE for the specific X-ray equipment and diagnostic technics used before 1964 at KCP, the default options from ORAU (2003a) should be used. ORAU (2003a) recommends that dose reconstructors assume that errors are all positive (i.e., use only +30%).

3.3 INSTRUCTION GUIDE FOR DOSE RECONSTRUCTORS

KCP practice involved maintaining a log of the date and type of medical examination and a record of the associated radiography films. DOE-provided medical X-ray dose information should include a copy or transcription of information from this log. Dose reconstructors should compare the frequency of examinations in this log with the default assumptions listed in Table 8.

Dose reconstructors should use the greater number of examinations from the log or from Table 8 in preparing their dose calculations. A normal distribution should be assumed with an uncertainty of ±30% at the one-sigma confidence interval. Dose reconstructors may multiply the doses listed in

Table 6. Organ dose estimates for PA chest radiographs before 1964 assuming minimal collimation.

Organ	Dose conversion factor (mGy/Gy air kerma) (beam quality for 2.5 mm Al HVL)[a]	Organ dose (rem)
Thyroid	174	3.48E-02
Eye/brain	32	6.40E-03
Ovaries	168	2.5E-02
Urinary bladder	168	2.5E-02
Colon/rectum	168	2.5E-02
Testes	9.1	5.0E-03
Lungs	451	9.02E-02
Thymus	451	9.02E-02
Esophagus	451	9.02E-02
Stomach	451	9.02E-02
Bone surfaces	451	9.02E-02
Liver/gall bladder/spleen	451	9.02E-02
Remainder	451	9.02E-02
Female breast	49	9.80E-03
Uterus	149	2.5E-02
Bone marrow	92	1.84E-02
Skin[b]		2.72E-01

a. HVL = half-value layer; dose conversion factors (DCFs) from ORAU (2003a) and ICRP Publication 34 (ICRP 1982, Tables A.2 to A.8)
b. Skin dose was determined by multiplying ESE by a backscatter of 1.36 for an HVL of 2.5 mm Al from NCRP Report 102 (NCRP 1989, Table B-8).

Table 7. Organ dose estimates for PA chest radiographs from 1964 to 2004.

Organ	Dose conversion factor (mGy/Gy air kerma) (beam quality for 2.5 mm Al HVL)[a]	Organ dose (rem)
Thyroid	32	4.16E-04
Eye/brain	32	4.16E-04
Ovaries	1	1.30E-05
Urinary bladder	1	1.30E-05
Colon/rectum	1	1.30E-05
Testes	0.01	1.30E-07
Lungs	451	5.86E-03
Thymus	451	5.86E-03
Esophagus	451	5.86E-03
Stomach	451	5.86E-03
Bone surfaces	451	5.86E-03
Liver/gall bladder/spleen	451	5.86E-03
Remainder	451	5.86E-03
Female breast	49	6.37E-04
Uterus	1.3	1.69E-05
Bone marrow	92	1.20E-03
Skin[b]		1.77E-02

a. Dose conversion factors (DCFs) from ORAU (2003a) and ICRP Publication 34 (ICRP 1982, Tables A.2 to A.8)
b. Skin dose was determined by multiplying ESE by a backscatter of 1.36 for an HVL of 2.5 mm Al. From NCRP Report 102 (NCRP 1989, Table B-8).

Table 8. PA chest occupational X-ray frequency.

Examination	Frequency
Before 1993	Chest X-ray offered with annual physical examinations.
Beginning in 1993[a]	Chest X-rays offered every 5 yr.

a. X-rays could be more frequent if a worker's history or physical circumstances indicated.

Tables 21 and 22 in Section 6.0 by a factor of 1.3 (i.e., 30% increase) to include uncertainty (ORAU 2003a) in the analysis.

4.0 OCCUPATIONAL ENVIRONMENTAL DOSE

KCP prepared a site safety assessment report in 1995 in compliance with DOE requirements (Allied-Signal 1995); the report concluded that Plant operations had produced no undue hazard to the general public and no significant effect on the environment. It appears that there are minimal, if any, historical radiological effects. Air and water effluents have been monitored routinely to assess compliance with relevant criteria. KCP routinely handles hazardous chemicals, but there is limited handling of radioactive materials and thus essentially little likelihood of a significant occupational environmental exposure associated with releases. The Plant does generate low-level radioactive waste that includes equipment radiation sources, tritium exit signs, irradiated components, gap tubes, smoke detectors, and small amounts of cleanup materials and personal protective equipment. Mixed wastes consist of encapsulated electronic assemblies and spent solvents used in cleaning and decontaminating radioactive materials (primarily defective sources). There is no offsite environmental exposure. As such, variability calculated from background control personnel dosimeters (i.e., 2-sigma = 25 mrem) is recommended as a claimant-favorable option to assign external dose to unmonitored workers, as described in Attachment A.

The primary and apparently only substance handled in large quantities with a potential for significant environmental release involved depleted uranium (DU) from 1958 through 1970. As described in Attachment A, the recommended approach to assign an internal environmental dose to unmonitored nonradiological workers is to assume an annual intake fraction of 0.01 times the median annual measured air concentrations in Table 11 (i.e., multiply annual concentrations in Table 11 by 6.58×10^{10} to obtained pCi/day). The 0.01 factor is a "claimant favorable" estimate of the decrease in concentration based on dispersion between inside and outside air concentrations. For the period following 1970, there are no workplace air sampling data available for analysis. Since there is the potential for unknown small releases or residual contamination, such as the incident with ^{147}Pm in 1989, the same 0.01 factor is recommended using the average median annual measured air concentration (for all years) from Table 11. Beginning in 1990, there is assurance that no significant environmental releases occurred based on the multi-agency response findings to the ^{147}Pm incident.

5.0 OCCUPATIONAL INTERNAL DOSE

KCP has handled numerous types of radioactive sources (Table 3). Most of these sources are sealed or contained as components of various types of equipment. No incidents involving confirmed internal intakes have been reported (Lund 2004). The primary workplace exposure was associated with machining items containing DU oxide from 1958 to about 1971; the program that used DU oxide ended in 1972 (Nasca 2005a). No uranium was processed until 1997 when a new program was initiated. From 1997 onward, DU metal was reduced in size and shape by an electrochemical process that involves the placement of DU metal in an acid bath. The parts are rinsed with water and dried before handling. Because the uranium does not become volatile during the electrochemical process, remaining in the acid solution, there is minimal personnel internal dose hazard with this process. There is no removable contamination with this process (Nasca 2005a). Bioassay data (urine assays) and air monitoring data were found only for 1959 through 1971. Intakes of DU after 1972 are not likely.

5.1 DEPLETED URANIUM

The uranium at KCP was DU. For internal dose calculations, use the NIOSH accepted default isotopic ratios for DU that are found in the Interactive Modules for Bioassay Analysis (IMBA) computer program. Table 9 lists typical weight percents and activity fractions of uranium isotopes. When estimating intakes, dose reconstructors should assume that exposures to uranium dust were chronic.

5.1.1 Physical and Radiological Characteristics

KCP had substantial quantities of UO_2 on the site at various times. Order number ICO-020757 (Bendix 1962) shows that UO_2 was ordered in 10,000-pound lots. The relevant specifications from

Table 9. Mass and radiological characteristics of DU.

Isotope	Weight percentage[a]	Specific constituent activity in mixture[b]		
		Bq/mg	pCi/mg	dpm/mg
U-234	0.0010	2.3116	62.4757	138.696
U-235	0.1991	0.1592	4.3028	9.5523
U-236	0.0003	0.0072	0.1941	0.4308
U-238	99.7996	12.4111	335.4345	744.6647
	Totals	14.889	402.4071	893.3437

a. From IMBA computer program (Birchall et al. 2003).
b. Could vary from IMBA values due to rounding.

Specification Control No. 4542260-00 (KCP 1998) were that the minimum density should be no less than 10.8 g/cm^3, the surface area of the powder should be no greater than 1.1 m^2/g, at least 97% by weight of the material should be less than 10 µm in diameter, and 100% by weight should be less than 15 µm in diameter. These specifications are consistent with a powder having an activity median aerodynamic diameter (AMAD) of 1.175 µm and a geometric standard deviation (GSD) of 2.48.

In fitting bioassay data, dose reconstructors may wish to start with a 1-µm AMAD, a GSD of 2.5, a density of 10.97 g/cm^3, a lung solubility Type S, and f1 of 0.002 (ICRP 1995). However, it is not known what impact processes at KCP had on the particle size of uranium. Use of the default 5-µm AMAD particle size is also acceptable unless it is known that the intake was of unaltered UO_2 powder.

5.1.2 Workplace Monitoring

Table 10 summarizes alpha radiation contamination results from 1962 to 1969 in DU work areas considered to be most significant to potential worker exposure (Nasca 2004b). In addition, from 1958 to 1970 KCP workplaces were routinely monitored with air samplers for DU concentrations (Nasca 2004e). Table 11 lists the maximum measured workplace concentrations used to calculate the median and 95th-percentile statistical parameters.

Table 10. Alpha contamination levels, 1962 to 1969 (Nasca 2004b).

Facility	Work area	Measured levels (dpm/100 cm$^{2)}$	
		Average	Maximum
D/34C (D/27C)	Air lock	226	20,000
	Locker room	570	7,000
	General area	2,564	45,000
D/220-22 (D/443-20D D/216-22, D/217-20D)	Air lock	190	800
	Wash-up	180	650
	General area	425	1,000
D/22, D/20D	Air lock	206	350
	Clean area	468	2,000
	General area	892	16,000

Eating and smoking were prohibited in DU processing areas. However, at least one survey report indicated the presence of cigarette butts, candy wrappers, and coffee cups in the exclusion area (Baldwin 1966). This indicates that ingestion of DU was a possible route of occasional intake. Dose reconstructors may assume inhalation or ingestion intakes to be claimant-favorable.

The parameters in Table 11 were based on maximum measured air concentrations at several locations in the plant. Some measurement locations were identified only by a number. When locations were indicated, most were near the walls of the work areas. Other locations were labeled as "Mixing Rm, West," "Air Lock," "Over Shower," "Rubber Mill Rm," "Mill Stack West," and "Dispersion Roll."

Table 11. Statistical parameters of measured DU in workplace air.[a,b]

	KCP measured results[a]			Lognormal fit		
	No. of	(μCi/cm^3)		Air concentration (μCi/cm^3)		
Year	measurements	Mean	Maximum	Median	95%	GSD
1958	22	7.18E-12	4.90E-11	4.01E-13	1.74E-10	4.02E+01
1959	27	8.82E-13	1.22E-11	2.89E-13	2.53E-12	3.74E+00
1960	33	1.32E-12	1.50E-11	3.41E-13	3.94E-12	4.43E+00
1961	31	1.00E-12	2.04E-11	1.97E-13	1.52E-12	3.46E+00
1962	31	7.73E-13	1.13E-11	2.50E-13	2.03E-12	3.58E+00
1963	31	1.25E-12	1.63E-11	2.47E-13	1.90E-12	3.46E+00
1964	31	2.21E-12	3.90E-11	3.91E-13	2.98E-12	3.44E+00
1965	31	1.99E-13	8.70E-13	1.05E-13	8.02E-13	3.45E+00
1966	23	7.01E-13	6.24E-12	2.00E-13	2.00E-12	4.06E+00
1967	22	1.40E-12	1.30E-11	5.70E-13	3.12E-12	2.81E+00
1968	19	1.21E-12	9.88E-12	2.31E-13	3.47E-12	5.19E+00
1969	19	1.88E-11	8.55E-11	3.88E-12	1.42E-10	8.92E+00
1970	19	7.32E-14	5.91E-13	4.02E-14	1.98E-13	2.64E+00
Average 1958-70		2.85E-12	2.15E-11	5.49E-13	2.62E-11	

a. All departments.
b. Based on maximum measured KCP workplace airborne uranium concentrations at several monitoring locations.

5.1.3 Bioassay

KCP workers were individually monitored for DU intake from 1959 to 1971 using a fluorophotometric method to measure the level of uranium in urine. KCP (1962) states that the method is sensitive to concentrations of uranium from 1×10^{-10} to 5×10^{-11} g per 0.25 g of sodium fluoride with a precision of ±10% (KCP 1962). This sensitivity equates to 0.5 to 1 μg U/L of urine. However, this sensitivity may be the theoretical best based on ultra pure water blanks, as opposed to urine blanks. The urine volume used for bioassay analysis was 0.1 ml and the lowest quantity of uranium used to determine a standard curve was 1×10^{-9} g. The urine concentration that equates to the lowest uranium quantity of the standard curve is 10 μg/L. Bioassay data for four individuals (Nasca 2005b) shows that concentrations as low as 1 μg U/L were recorded. However, most sites using fluorophotometry at this time were claiming more modest detection levels; for instance, Hanford - 4 μg/L (ORAU 2004a), Paducah – 10 μg/L (ORAU 2004b), University of Rochester (used by many AWEs) 5 – 10 μg/L (ORAU 2005). Because a definitive statement of the detection limit achieved by the Kansas City Plant was not found, an MDA of 10 μg/L is recommended.

The frequency of bioassay analysis for KCP personnel who worked with DU powders is not known. The available data are shown on an annual basis and might be the sum of one or more bioassay measurements. If the individual case information does not yield additional information, dose reconstructors should make the claimant favorable assumption that the recorded bioassay quantities represent a single bioassay measurement taken at the end of the calendar year.

By procedure (KCP undated), the workers in radiation areas 20D, 34C, and 443E-20 were to receive uranium-in-urine bioassays semiannually (May and November). Bioassay data were recorded on either the individual's film badge envelopes or the annual 3-in × 5.5-in. radiation exposure record (Nasca 2005b). The actual frequency of bioassay analysis for KCP personnel who worked with DU powders varied from person to person and from year to year (Nasca 2005b). The date and results of individual bioassay results are in each individual's dosimetry file.

The only bioassay data that are available electronically (Nasca 2004c) are the numeric sums of all bioassay measurements taken during the year for an individual and could be the sum of one or more bioassay measurements. The number of bioassay measurements that comprise the annual sum is not recorded in the electronic database. Bioassay data from four individuals were studied (Nasca 2005b) and the number of bioassay samples taken per year ranged from 0 to 6. Zero samples per year means that there are gaps in the bioassay record where one or more years of no bioassay data are bracketed by years for which there are bioassay data. In 1960 and 1961, there appear to have been more urine samples collected and the results of the samples were greater than for other years. No information was found to explain why the bioassay data for 1960 and 1961 were greater than for other years. There were no records of an incident involving DU powder in those years (Nasca 2005a).

Table 12 summarizes an analysis of the electronic bioassay records (Nasca 2004c). These data show a peak in 1960 and 1961. The peak is apparently not the result of an incident (Nasca 2005c) but could be due to a large number of bioassay samples being collected in those years for each worker. The bioassay data for 1971 are very low, indeed, they are less than the sensitivity level. The low bioassay levels may indicate that no intakes of uranium occurred during that year. As noted above, the excreta data in Table 12 represent the sum of an unstated number of bioassay measurements. An improved co-worker analysis on bioassay data is scheduled as part of the overall plan for co-worker analyses of DOE sites.

Table 12. Statistical parameters of recorded DU in urine.[a]

	Recorded annual urine concentration[b]			Lognormal fit		Chronic intakes (pCi/d)[c]		
	No. of workers	Concentration (µg/L)		Concentration (µg/L)				
Year	reported	Mean	Maximum	Median	GSD	5th	Median	95th
1959	214	4.125	52.60	2.642	2.675	1.05E+02	6.42E+02	3.92E+03
1960	281	36.58	140.	19.53	3.813	7.79E+02	4.75E+03	2.89E+04
1961	123	51.40	192.1	37.44	2.402	1.49E+03	9.10E+03	5.55E+04
1962	148	4.327	15.75	3.162	2.508	1.26E+02	7.69E+02	4.69E+03
1963	211	10.96	72.00	7.564	2.532	3.02E+02	1.84E+03	1.12E+04
1964	219	5.627	78.38	3.888	2.431	1.55E+02	9.46E+02	5.76E+03
1965	175	9.572	38.00	5.583	3.422	2.23E+02	1.36E+03	8.27E+03
1966	223	6.432	45.05	4.214	2.640	1.68E+02	1.02E+03	6.24E+03
1967	159	5.438	21.50	3.574	2.713	1.43E+02	8.69E+02	5.30E+03
1968	11	6.055	6.600	6.052	1.029	2.42E+02	1.47E+03	8.97E+03
1969	1	0.15	0.150	0.150	1.000	5.99E+00	3.65E+01	2.22E+02
1970	59	11.64	45.00	7.576	2.686	3.02E+02	1.84E+03	1.12E+04
1971	47	0.03596	0.1000	0.02993	1.903	1.19E+00	7.28E+00	4.44E+01
ALL	1,871	14.1	192.1	5.5	4.7			

a. All bioassay measurements.

b. The recorded annual sum of urine concentration is the sum of all bioassay results for the year. There is one sum for each person-year record. The listed statistics are based on the analysis of the data, which are the sums of all bioassay data for every person for that year.

c. Chronic intakes that produce the urinary excretion per day on the 365th day of intakes corresponding to the median excretion from the lognormal fit and 5th and 95th percentile intakes using a GSD of 3. Assumes 5-µm AMAD particle size; intakes for 1-µm AMAD particle size, 10.97 g/cm^3 density, and absorption type S are smaller.

Because these data are the sum of all bioassay measurements for each person in each year, they represent the maximum bioassay result possible. Also the default 5-µm AMAD particle size was used for the intake estimates; however, using the more likely 1-µm AMAD particle size and density of uranium oxide results in smaller intakes. Hence, the intakes in Table 12 are overestimates so long as a whole year's exposure is used. The different percentile intakes are used with types of exposures explained in Section 5.1.4.

Due to the nature of the work performed at KCP, and because no accident reports have been found, it is reasonable to assume that intakes of DU from 1959 through 1971 were chronic unless the individual's dosimetry records indicate otherwise.

The persons who received bioassay for uranium had Organization Codes 530001 and 531002. Table 13 lists the number of bioassay results for the two organizations and the occupations that had bioassay results. The Organization Code does not provide much information about what groups might have been exposed to airborne uranium because Code 531002 refers to the DOE contractor that operated KCP (e.g., Bendix, Allied-Signal), and Code 530001 refers to DOE workers. The data in Table 13 indicate that nearly all types of workers might have been exposed to uranium at KCP. The information in Table 13 may be of some benefit to dose reconstructors if a worker's job description and organization code are known. However, see Section 5.1.4 for more general instructions for unmonitored workers.

Table 13. Number of recorded bioassay measurements and average of measurements for 1959 through 1971.

Occupation description	Occ Code	Number of individual measurements			Bioassay measurements (µg/L)	
		Organization code				
		530001	531002	Total	Total[a]	Ave.[b]
Managers and administrators	110	10	342	352	3,210.44	9.12
Engineers	160		228	228	2,017.84	8.85
Scientists	170		15	15	35.16	2.34
Health physicists	184	3		3	12.90	4.30
Miscellaneous professionals	200		17	17	149.64	8.80
Repair technician	350		44	44	580.58	13.19
Health technician	360		3	3	16.17	5.39
Technologist, engineering	370		38	38	519.03	13.66
Miscellaneous technicians	390		47	47	418.52	8.90
Administrative support/clerical/work planners	450		60	60	717.74	11.96
Fire fighter	512		58	58	228.33	3.94
Security inspector & guard	513	3	4	7	11.25	1.61
Food service employees	521		1	1	0.00	0.00
Custodian/janitor	524		11	11	31.33	2.85
Mechanics/repairers	610		350	350	2,687.45	7.68
Electrician	643		153	153	1,371.99	8.97
Pipe fitter	645		177	177	1,566.47	8.85
Machinist	681		152	152	2,140.77	14.08
Sheet metal worker	682		2	2	0.00	0.00
Operators, plant/system/utility	690		2	2	2.45	1.23
Machine setup/operators	710	5	28	33	262.95	7.97
Welders/solderers	771		11	11	77.10	7.01
Miscellaneous precision/production workers	780	14	640	654	6,894.03	10.54
NA[c]	781		1	1	14.05	14.05
Drivers	840		28	28	217.13	7.75
Handlers/laborers/helpers	850		145	145	1,279.55	8.82
All groups				2,592	24,472.87	9.44

a. Grand total of all bioassay measurements for that occupational code for all years (1959-1971).
b. Average of recorded bioassay measurements.
c. NA – Information not available.

Table 13 also lists the sum of all bioassay results, for all years, for each occupational code and the average of the bioassay measurements for each occupational code.

The electronic record of uranium bioassay contains the sum of all measurements performed during the year for the particular individual. The electronic record does not indicate the number of samples that comprise the recorded measurement. The results of individual measurements were written on cards that are nearly illegible. For these reasons, KCP should be a candidate site for a co-workers study. The Managers and Administrators occupation category included individuals who performed job estimates and who were commonly in the work areas.

5.1.4 Unmonitored Worker

Table 12 lists the statistical parameters of measured DU concentrations in the urine of KCP workers for the years for which urine data were found and associated intakes. Table 13 provides some insights into job categories. The uncertainties in the intakes in Table 12 do not warrant assigning specific intakes to each job category; however, the data in Table 12 do lend themselves to grouping workers into 4 exposure categories: (1) workers routinely exposed to airborne or loose material, (2) workers occasionally exposed, (3) workers rarely exposed or exposed only to very low workplace airborne or contamination levels, and (4) workers with little or no potential for radiological exposure. Unmonitored workers in category 1 should be assigned the 95th percentile intakes; category 2 to the median intakes; category 3 to the 5th percentile intakes; and category 4 workers should be assigned internal exposures per the environmental section of this document. The dose reconstructor should use the information in Table 13, the worker's radiation exposure file, and the computer assisted interviews, and Technical Information Bulletin 014 (ORAU 2005c) to assign unmonitored workers to exposure categories. Generally, the occupations for which bioassay data have not been found are nurses, miscellaneous repairers/ construction workers, and equipment operators.

Due to how Table 12 intakes were calculated, they must be applied to the whole year to ensure claimant-favorability, even if the energy employee worked a partial year.

Dose reconstructors will likely not find bioassay results in 1958 yet the air sample data indicate that airborne contamination existed. For 1958 for category 1 workers assume chronic intakes of 5,430 pCi/d; for category 2 workers assume chronic intakes of 891 pCi/d; for category 3 workers, 146 pCi/d. The median intake for 1958 was determined by ratio of the median air concentrations for 1958 / 1959 from the lognormal distributions in Table 11 times the median intake for 1959 from Table 12. The default GSD of 3 was applied to the median intake for 1958 to determine the 5th and 95th percentile intakes for 1958. This is consistent with the GSD for the intakes for the other years in Table 12.

Because several maximizing assumptions were used to determine the intakes in Table 12, the uncertainty distribution of doses determined from these intakes should be entered into IREP as constants.

5.2 OTHER NUCLIDES

There is apparently no indication or expectation of significant worker intakes of any other nuclide. The response to an incident on February 10, 1989 involving ^{147}Pm initially concluded, apparently in error, that an intake had occurred according to a report of what appears to be an extensive investigation (KCP 1989). According to this report, the Department of Energy, Environmental Protection Agency and the Missouri Department of Health, Radiological Health Division representatives were notified at the time of this incident. A DOE team of investigators arrived at KCP on February 14 to assume technical management of the situation. Based on what eventually were determined to be false-positive

bioassay results, the homes of 4 KCP workers were inspected and some contamination was found. There were undoubtedly many activities to identify the cause and extent of this contamination. Pm-147 is a relatively low-energy 100% beta emitting nuclide (e.g., maximum energy of 224.7 keV, average energy of 62 keV) with a half-life of 2.6 years. The primary concern with this type of nuclide would be direct skin contamination and intake. The report addresses the chronology of steps, and the results, to examine potential exposure and contamination. KCP operations do include the use of several radioactive sources that can be fragile and relatively unsealed, as occurred with the ^{147}Pm source incident. However, any intakes from KCP use of various small sources would probably be comparatively insignificant.

6.0 OCCUPATIONAL EXTERNAL DOSE

Information concerning the early history of KCP nuclear weapons assembly activities involves classified information; therefore, a clear description of events at that time is not publicly available. As described in Section 2.0, AEC operations at KCP began in 1949. The primary work activity involving external radiation exposure was fabrication and quality control testing of non-nuclear components of nuclear weapons.

6.1 BASIS OF COMPARISON

Since the initiation of the Manhattan Engineer District (MED; a DOE predecessor agency) project in the early 1940s, various concepts and quantities have been used to measure and record occupational radiation dose at the many MED/AEC/DOE facilities. A common basis of comparison has been selected to assess the consistency of the available historical recorded dose at KCP with current KCP dosimetry performance. With the known dates of changes in KCP dosimetry systems, comparisons of recorded doses before and after these changes provide an ability to assess consistency. Similar sources have been used to calibrate and conduct performance testing of dosimetry systems (AEC 1955; Unruh et al. 1967; McDonald et al. 1983). This basis, to be used in dose evaluation or reconstruction, is the personal dose equivalent, $H_p(d)$, where d identifies the depth in millimeters and represents the point of reference for dose in tissue. For weakly penetrating radiation of significance to skin dose, d is 0.07 mm and $H_p(d)$ is noted as $H_p(0.07)$. For penetrating radiation of significance to whole-body dose, d is 10 mm and $H_p(d)$ is noted as $H_p(10)$. The International Commission on Radiological Units and Measurements has recommended both $H_p(0.07)$ and $H_p(10)$ for use as the operational quantities to be recorded for radiological protection (ICRU 1993). These are the radiation quantities used in the DOE Laboratory Accreditation Program (DOELAP) since the 1980s to accredit personnel dosimetry systems in the DOE complex, including KCP (DOE 1986).

6.2 WORKPLACE EXTERNAL RADIATION FIELDS

Section 2.4 describes the primary sources of workplace radiation fields at KCP. These sources are historically associated with processes involving industrial RGDs (X-rays and electron accelerators); isotopic beta, gamma-ray, and neutron radiation-emitting sources; and DU. The isotopic sources are typically used in the manufacturing or quality control processes to monitor fabrication of non-nuclear weapons components.

6.2.1 Beta Radiation

Beta radiation fields in KCP workplaces are associated with the sources of beta-emitting nuclides listed in Table 3, uranium handled from about 1958 to 1971, and RGD-produced electrons identified in Tables 2 and 4. Potential exposures to beta radiation sources are typically limited to maintenance activities or failures. For example, a ^{147}Pm source was used to measure the thickness of a film, and

failure of the integrity of this source did result in some worker exposure and was the subject of an official investigation (KCP 1989). Worker exposure to electrons would typically be associated with an error or equipment failure.

6.2.2 Photon Radiation

Photon (X-ray and gamma) radiation associated with KCP work activities covers a broad energy range. Sources of ionizing radiation at KCP listed in Table 3 include several radioactive sources that emit photon radiation. These were typically used to check or calibrate processes to gauge thicknesses, perform instrument calibrations, and so forth. Many sources were beta and photon emitters and are of the types and source strengths typically used by mainstream industrial or process-related users. In general, doses associated with the proper, and widespread, use of these small check sources are comparatively negligible. In addition, KCP used a variety of larger photon-emitting radiation sources and X-ray RGDs, as listed in Table 4. These were typically used to radiograph parts or perform operational tasks. There were numerous X-ray machines in locations around the Plant.

The photon energy spectra in KCP workplaces have not been measured. The spectra are related to the configuration of the X-ray machines, the process, and the extent of shielding. However, regardless of the precise spectra, significant photon radiation would have been readily measured at KCP by the available dosimeter technology during all years of operation.

6.2.3 Neutron Radiation

Sources of neutron radiation at KCP involve neutron generators and nuclide (alpha, neutron) interactions, such as ^{239}PuBe sources, as listed in Table 4. The first presence of neutron-emitting nuclides apparently occurred after 1965 based on a review of workplace hazards (Schiltz 1965).

6.3 DOSIMETER TECHNOLOGY

KCP has historically used beta/photon and neutron dosimeters to measure potential radiation exposure of personnel.

6.3.1 Beta/Photon Dosimeters

Beta and photon (X-ray and gamma) dosimeters used at KCP include:

- Pocket ionization chamber (PIC)

- KCP Film Dosimeter (Bendix 1964) – a two-piece, stainless-steel, film holder with front and rear matching rectangular windows. The front and rear faces have a 1-mm-thick cadmium filter in the area of the open window. Two different types of personal dosimetry film packets are used – Kodak Type 2 for mixed beta and gamma radiation and DuPont Type 558 for X-ray and gamma radiation. The DuPont 558 film packet contains two films, Types 508 and 1290. The type 508 film has a range of 0.003 to 30 R, and Type 1290 has a range of 0.68 to 3,000 R. KCP calibration data using uranium and ^{60}Co sources are available (Bendix 1964).

- KCP (two-chip, TLD-100) thermoluminescent dosimeter (TLD) – used from 1973 to 1982.

- Eberline standard (three-chip, TLD-100) TLD – used from 1983 through 1990. This dosimeter employs one chip under a 10-mg/cm^2 filter to measure the shallow or skin dose and one or two chips under a 285-mg/cm^2 filter to measure the deep or whole-body dose (TMA 1990). The

use of this dosimeter was terminated due to its inability to pass DOELAP lower energy photon performance testing categories although it had passed National Voluntary Laboratory Accreditation Program (NVLAP) testing (Allied-Signal 1991).

- The Landauer K1 (three-chip, TLD-700) TLD – use began April 1, 1991, to measure beta, X-ray, and gamma radiation exposure to KCP workers.

- Landauer optical stimulated luminescent (OSL) aluminum-oxide dosimeters – use began in 2000.

6.3.2 Neutron Dosimeters

KCP used the Landauer Neutrack I dosimeter to measure neutron doses. This dosimeter is a polycarbonate (Lexan) neutron recoil track registration device used to monitor fast neutron interactions. The Lexan responds to neutrons by recording ionization damage caused by neutron interactions with carbon and oxygen atoms, which leaves a track. It has a uniform energy response from 3 to over 14 MeV with a threshold of 1 MeV.

The dosimeters were used to record the official dose of record and the PIC was used to provide administrative control until dosimeter results were available. Table 14 summarizes the use of personnel dosimetry techniques and exchange frequencies.

Table 14. Personnel dosimetry systems.

Period	Description	Dose measured	Routine exchange period
Beta/photon dosimeters			
1950 - 1954	KCP in-house film badge system	Beta/nonpenetrating X-ray/gamma	Weekly
1955			Biweekly
1956 - 1964			Monthly
1965 -1972			Bimonthly
1973 - 1982	KCP two-chip TLD		Bimonthly
1983 - 1990	Eberline three-chip TLDs		Quarterly
1991 - 2000	Landauer three-chip TLDs		Quarterly
2001 - 2003	Landauer OSL		Semiannual
Neutron dosimeters			
1961 - 1967	Controls for Radiation, Inc.	Neutron	Biweekly
1967 - 1973	Landauer, film		
1974 - 1982	Landauer NTA film		Quarterly
1983 - 1990	Landauer Lexan track-etch		
1991 - 2000	Landauer Neutrak I Poly carb		
2001 - 2003	Landauer Neutrak 144		Semiannual

KCP participated in DOELAP performance testing using Landauer-provided services beginning in October 1992, and most recently passed DOELAP performance testing in May 1995. As of May 1998, KCP has been exempt from DOELAP accreditation based on the low-level potential for worker dose and accreditation by NVLAP.

6.4 DOSE RECONSTRUCTION

Evaluation of KCP worker dose to ensure that the occupational dose for each worker claim is not underestimated involves assessment of:

- Potential unmonitored dose for workers who were not monitored for occupational radiation exposure throughout their employment at KCP.

- Potential missed dose for monitored workers because of missing dosimeter results, actual dose that was less than the detection capabilities of the dosimeters, or unrecorded doses in a worker's exposure history.

- Potential adjustments to the recorded dose because of considerations of the dosimetry technology, calibration methods, and workplace radiation fields that could have resulted in error in the recorded dose.

6.4.1 Potential Unmonitored Dose

Based on KCP safety policies and the recorded dose to categories of workers, it appears that monitoring occurred for most categories of workers. However, for workers without a recorded dose, it is reasonable to assume that any unmonitored dose would be less than the dose received by monitored workers. Using the measured doses for these workers is more reasonable than an analysis based on the AEC radiation protection guidelines to assign dosimeters to workers who potentially exceed 10% of the dose limit. Figure 2 shows a statistical analysis of recorded penetrating (i.e., photon) doses in a lognormal probability plot for 1950 through 2003. Table 15 summarizes for KCP recorded annual penetrating dose the arithmetic average and maximum value for all recorded penetrating doses and the lognormal probability statistical parameters for all positive recorded doses (i.e., dose > 0). Dose reconstructors should assign the ambient environmental dose to an unmonitored worker with minimal potential for radiation exposure from KCP operations, median coworker dose to an unmonitored worker with minimal likelihood of actual workplace exposure and the 95th percentile coworker dose to workers with a potential for workplace radiation exposure for each year of employment without a recorded dose. There should not, typically, be a significant neutron exposure of unmonitored workers because sources of neutron radiation were very limited.

6.4.2 Potential Missed Dose for Monitored Workers

Missed dose occurs when the dose of record is zero because the interpreted dose had a negative bias, dosimeter response was less than the minimum detectable level (MDL), or there was no dose of record for an assigned badge for a period. There is no evidence of a systematic negative bias, as could occur from background control dosimeters in locations of elevated ambient dose (ORAU 2003b). Missed dose because the response is less than the MDL is typically the most important consideration and is important for earlier years when MDLs were higher and dosimeter exchange was more frequent. Dose reconstructors should follow NIOSH (2002) guidance to calculate the missed photon dose. Methods for estimating the potential missed photon dose include:

- Method 1: Estimate the missed dose reading from other readings for the same person doing similar work for different periods (Watson et al. 1994).

- Method 2: Assign a maximum missed photon dose based on the MDL and the number of routine exchange periods (NIOSH 2002) for the respective dosimetry systems in Table 16.

Dose reconstructors should use Method 1 if there is sufficient information and only occasional missed doses, and Method 2 if significant doses are zero or missing. Use Method 2 for workers who were monitored with missing or zero recorded dose or who worked in an area or an occupation where positive dose would be expected but was not recorded.

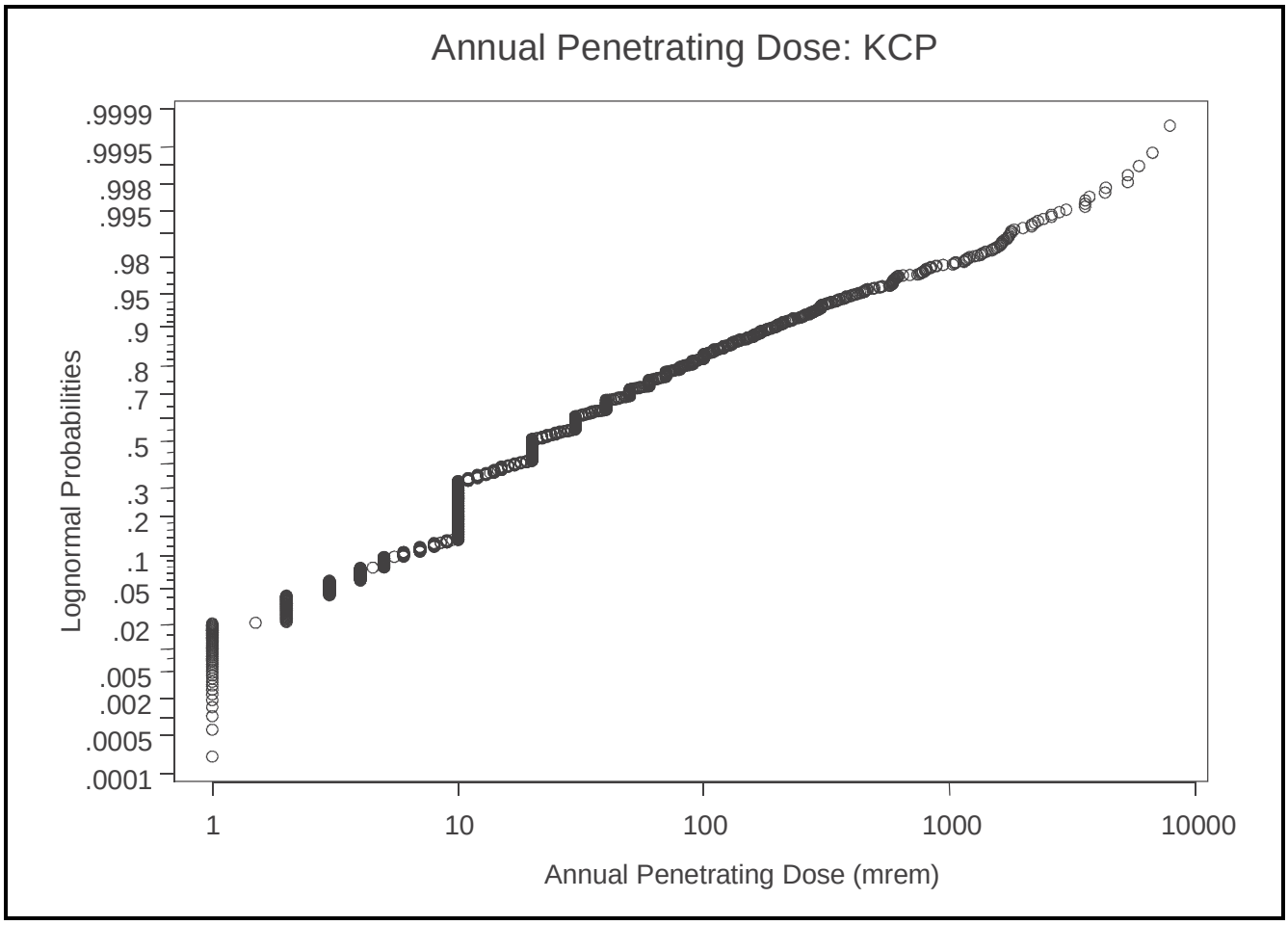

Figure 2. Lognormal probability plot of KCP recorded annual penetrating dose.

6.4.3 Neutron Dose Adjustments for Monitored Workers

KCP worker recorded positive neutron dose was recorded a total of 35 times over a period from 1966 through 1996, with most instances occurring in the 1980s with good dosimetry capabilities, from a total of 14,758 annual dose records. In all cases, except for 2, the recorded annual deep dose is equal to the recorded annual neutron implying all recorded deep dose resulted from the neutron dose. The two exceptions occurred in 1976 and 1983 with recorded deep and neutron annual doses of 64 mrem and 26 mrem, and 3 mrem and 1 mrem, respectively. As such, the recommended approach is to use a neutron to photon dose ratio of 1:1 as a conservative estimate of neutron dose to the few workers with any potential of neutron exposure. Overall, the recorded neutron dose is a relatively insignificant component of the worker dose. The recorded neutron dose must be adjusted to include conversion to the International Commission on Radiological Protection (ICRP) Publication 60 (1991) neutron weighting factor required for input of the dose into the Interactive RadioEpidemiological Program (IREP) by using the assumed neutron energy and dose fraction listed in Table 17. Because there are no workplace measurements of neutron spectra, this analysis assumed that 100% of the workplace neutron radiation is within the range of 0.1 to 2 MeV because this option provides the highest organ dose estimate. The approach can be simplified using the following expression:

Neutron dose = adjusted photon dose × *upper 95th-percentile neutron/photon dose ratio* × *ICRP 60 DCF*

Table 15. Statistical parameters of recorded penetrating annual doses.

	Arithmetic, all recorded dose			Lognormal, dose>0			
	No. of	Dose, rem		No. of	Dose, rem		
Year	workers	Mean	Maximum	workers	Median	95%	GSD
1950	46	7.09E-02	6.47E-01	29	5.01E-02	5.54E-01	4.31E+00
1951	227	2.78E-01	5.32E+00	158	4.97E-02	2.02E+00	9.52E+00
1952	233	1.91E-01	5.90E+00	220	2.45E-02	7.42E-01	7.96E+00
1953	103	1.37E-01	5.32E+00	72	2.13E-02	4.50E-01	6.38E+00
1954	65	1.61E-01	2.99E+00	21	1.19E-01	4.74E+00	9.39E+00
1955	41	9.18E-02	4.18E-01	25	1.10E-01	4.59E-01	2.39E+00
1956	26	2.89E-01	1.75E+00	26	1.35E-01	1.07E+00	3.52E+00
1957	65	5.24E-01	7.89E+00	36	1.77E-01	6.33E+00	8.80E+00
1958	301	3.90E-02	8.43E-01	89	5.44E-02	4.84E-01	3.78E+00
1959	464	8.97E-03	2.30E-01	72	3.57E-02	1.78E-01	2.66E+00
1960	1,043	1.21E-02	1.35E+00	165	4.53E-02	2.31E-01	2.70E+00
1961	948	2.39E-02	5.90E-01	400	3.17E-02	1.64E-01	2.72E+00
1962	700	1.25E-02	1.65E+00	59	4.25E-02	4.53E-01	4.22E+00
1963	597	1.18E-02	1.63E+00	100	3.65E-02	2.02E-01	2.83E+00
1964	530	6.85E-03	6.90E-01	59	3.09E-02	1.97E-01	3.08E+00
1965	436	1.35E-03	8.00E-02	26	1.66E-02	5.59E-02	2.09E+00
1966	415	2.32E-03	2.00E-01	27	2.54E-02	8.95E-02	2.15E+00
1967	370	1.71E-03	2.50E-01	20	2.06E-02	7.14E-02	2.13E+00
1968	469	1.75E-03	3.50E-01	5	1.10E-01	6.84E-01	3.04E+00
1969	577	(a)	0.00E+00	0			
1970	580	1.60E-03	3.20E-01	29	1.61E-02	7.47E-02	2.54E+00
1971	575	9.39E-04	2.00E-01	16	2.02E-02	9.09E-02	2.50E+00
1972	195	1.19E-02	4.26E-01	68	2.04E-02	8.61E-02	2.40E+00
1973	199	9.86E-02	1.81E+01	69	1.87E-02	1.09E-01	2.91E+00
1974	169	1.30E-02	8.80E-02	67	2.86E-02	6.89E-02	1.71E+00
1975	150	5.72E-03	8.60E-02	44	1.25E-02	5.96E-02	2.58E+00
1976	126	9.65E-03	2.48E-01	53	1.37E-02	7.34E-02	2.77E+00
1977	123	5.45E-03	3.00E-01	12	2.76E-02	1.72E-01	3.04E+00
1978	152	7.53E-03	1.25E-01	18	4.55E-02	2.25E-01	2.64E+00
1979	162	2.69E-03	8.30E-02	17	2.12E-02	5.64E-02	1.81E+00
1980	185	4.19E-03	1.33E-01	22	2.74E-02	8.49E-02	1.99E+00
1981	210	4.12E-03	1.20E-01	24	2.72E-02	9.27E-02	2.11E+00
1982	209	2.00E-03	5.20E-02	22	1.67E-02	3.72E-02	1.63E+00
1983	226	2.98E-03	3.14E-01	12	2.76E-02	2.06E-01	3.39E+00
1984	216	1.89E-02	3.57E+00	18	2.44E-02	2.80E-01	4.41E+00
1985	201	4.67E-02	8.66E+00	49	1.41E-02	8.34E-02	2.95E+00
1986	194	3.22E-03	5.50E-02	27	2.05E-02	4.71E-02	1.66E+00
1987	196	1.65E-03	4.00E-02	20	1.18E-02	4.82E-02	2.35E+00
1988	188	3.52E-03	1.80E-01	11	4.62E-02	1.81E-01	2.29E+00
1989	233	8.03E-04	1.60E-02	17	1.09E-02	1.37E-02	1.15E+00
1990[b]	217	1.16E-03	4.10E-02	17	1.37E-02	2.50E-02	1.44E+00

a. Recommend using values for 1968. All 1969 recorded doses = zero.
b. Recommend using values for 1990 for all subsequent years.

6.4.4 Adjustments to Recorded Deep Dose

The DOE-reported photon deep dose for KCP workers is likely to provide a reasonable estimate of the actual photon dose. The dosimetry technology used is capable of detecting and measuring the photon dose based on studies of historical performance (AEC 1955; Unruh et al. 1967). The energy of prevalent photons based on the radiation sources is readily measured, so no adjustment of recorded dose would be likely based on the response characteristics of the different dosimeters.

Table 16. Potential missed dose for monitored workers.

Dosimeter	Period	Exchange	MDL (mrem)		Maximum missed annual dose (rem)	
			Shallow	Deep	Shallow	Deep
KCP in-house film badge system	1950 - 1954	Weekly[a]	40[b]	40[b]	1.100	1.100
KCP in-house film badge system	1955	Biweekly	40	40	0.960	0.960
KCP in-house film badge system	1956 - 1964	Monthly	40	40	0.480	0.480
KCP in-house film badge system	1965 -1972	Bimonthly	40	40	0.240	0.240
KCP 2-chip TLD	1973 - 1982	Bimonthly	30	30	0.180	0.180
Eberline 3-chip TLDs	1983 - 1990	Quarterly	30	30	0.120	0.120
Landauer 3-chip TLDs	1991 - 2000	Quarterly	30	30	0.120	0.120
Landauer OSL	2001 - 2003	Semiannual	30	30	0.060	0.060

a. Earliest estimated exchange for radiation workers is weekly.
b. Estimated MDL typical of film dosimeter capabilities.

Table 17. Neutron dose fractions and associated ICRP 60 correction factors (ICRP 1991).

Description	IREP neutron energy (MeV)	Default dose fraction (%)	ICRP 60 correction factor (CF)
KCP workplace exposures	0.1 - 2 MeV	100	1.91

6.4.5 Radiation Dose Fraction

Table 18 summarizes the recommended fractions for KCP worker deep dose according to the energy categories required by IREP. For the reported deep dose, 50% is assigned to the respective categories of photon energy from 30 to 250 keV, and greater-than-250-keV. For the reported neutron dose, 100% is attributed to the 0.1- to 2-MeV category. For the reported shallow dose, the relative contribution from beta and photon radiation cannot be distinguished. As such, it is recommended that the dose reconstructor calculate the maximum dose to the organ of interest assigning 100% to the beta radiation > 15 keV category or assigning 100% to the photon radiation < 30-keV category. This provides claimant-favorable analysis of the dose to the organ of interest.

Table 18. Beta, photon, and neutron radiation energies and percentages for external radiation exposures.

Description	Operations		Radiation type	Energy selection	%
	Begin	End			
Parameters to estimate dose to whole-body organs	1949	2004	Beta	> 15 keV	100[a]
			Photon	30–250 keV	50[b]
				> 250 keV	50[c]
			Neutron	0.1-2 MeV	100[d]

a. Beta particles (electrons) from DU, sealed sources, and electron accelerators are greater than 15 keV.
b. Workplace photon energies from scattered X-rays and DU are greater than 30 keV. Primary photon energies for DU are within the IREP category of 30 to 250 keV.
c. Workplace photon energies cover a broad range. Photons from sealed sources and RGDs are typically of higher energies but are also heavily shielded.
d. The neutron energy region of 0.1 – 2 MeV was selected to be claimant-favorable for whole-body dose to provide the highest calculated organ dose.

6.5 UNCERTAINTY IN PHOTON AND NEUTRON DOSE

The analysis for this Site Profile evaluated bias and uncertainty associated with the KCP external dosimetry systems.

6.5.1 Deep Photon Dose for Film Dosimeters

The uncertainty in the KCP recorded photon deep dose results from various parameters related to dosimeter response, calibration, and workplace radiation fields can only be estimated. This was calculated using guidance in NIOSH (2002) and the National Research Council analysis of dose from atmospheric nuclear tests (NRC 1989). Table 19 lists estimates of uncertainty for film dosimeters used at KCP from 1950 to 1973.

Table 19. Uncertainty for KCP film photon deep dose.

Recorded dose (mrem)	Estimated IREP Parameter 2[a]	Estimated 95% uncertainty factor	Upper 95% confidence photon deep dose (mrem)
10	2.32	5.21	52
20	1.78	3.11	62
50	1.37	1.85	93
100	1.20	1.43	143
200	1.10	1.22	244
500	1.10	1.20	600
1,000	1.10	1.20	1,200
2,000	1.10	1.20	2,400
3,000	1.10	1.20	3,600
4,000	1.10	1.20	4,800
5,000	1.10	1.20	6,000

a. Recommended input value for IREP Parameter 2 is the highest value that bounds the annual dose value (i.e., use 1.36 for doses between 50 and 100 mrem).

The values in Table 19 were calculated from guidance in NIOSH (2002) as follows:

$$K(E) = 1 + 1.96\left[\frac{\sigma(E)}{E}\right]$$

$$\sigma(E) = \frac{\sigma^*}{D_\infty \gamma} e^{\gamma E}$$

where

Upper 95% confidence photon dose = $E \times K(E)$ (roentgens)
E = dose (roentgens)
σ^* = 0.015
D_∞ = 2.8
γ = 0.25

and $K(E)$ is limited to a minimum of 1.20 (NRC 1989).

6.5.2 Deep Photon Dose for TLDs

KCP has used TLDs since 1974 to measure the deep photon dose. The uncertainty in the recorded deep photon dose TLD results was calculated using guidance in NIOSH (2002) and the National Research Council analysis of dose from atmospheric nuclear tests (NRC 1989). Table 20 lists estimates of uncertainty for the TLDs used at KCP beginning in 1974.

The respective values in Table 20 were calculated from guidance in NIOSH (2002) as follows:

$$\sigma(E) = \sqrt{\left(\frac{L_C}{1.96}\right)^2 + \left(\frac{\sigma^*}{100}(E)\right)^2}$$

where

Upper 95% Confidence Dose = $E \times K(E)$ (millirem)
E = Dose (millirem)
$K(E)$ = 1 + 1.96 * ($\sigma(E)/E$)
Lc = 30 mrem
σ^* = 10%

Table 20. Uncertainty in TLD deep photon dose.

Recorded dose (mrem)	Estimated IREP Parameter 2[a]	Estimated 95% uncertainty factor	Upper 95% confidence photon deep dose (mrem)
10	2.00	4.01	40.1
20	1.59	2.51	50.2
50	1.28	1.63	81.5
100	1.17	1.36	136
200	1.12	1.25	250
500	1.10	1.20	600
1,000	1.09	1.20	1200
2,000	1.09	1.20	2400
3,000	1.09	1.20	3600
4,000	1.09	1.20	4800
5,000	1.09	1.20	6000

a. Recommended input value for IREP Parameter 2 is the highest value that bounds the annual dose value (i.e., use 1.36 for doses between 50 and 100 mrem).

6.5.3 Alternative Approach

The approach in sections 6.5.1 and 6.5.2 based on the recorded dose may not be feasible for many workers. Dose reconstructors may incorporate consideration of uncertainty in the dose calculation for measured and missed doses as follows:

- The technology used to measure worker dose at KCP is similar to the technology used by commercial and AEC laboratory facilities. The errors in the penetrating dose are anticipated to be approximately ±30% and normally distributed. Dose reconstructors can assume that errors are all positive (i.e., use only +30%) and multiply the measured dose by a factor of 1.3 (i.e., increase of 30%) to be used for IREP Parameter 1 and to set Parameter 2 to zero.

- For missed dose, a lognormal distribution is assumed. Dose reconstructors should calculate the unmonitored dose using Section 6.4.1 and the missed dose using Section 6.4.2 to arrive at Parameter 1 input and to set Parameter 2 equal to 1.52.

6.5.4 Shallow Dose

The primary workplace radiation that contributes to KCP worker exposure is photon radiation. Dose reconstructors should apply the uncertainties in Tables 19 and 20 for film and TLD measured deep dose, respectively, as needed to the shallow dose component as well.

6.5.5 Neutron Dose Uncertainty

Few if any KCP workers have received a significant neutron dose. The recommendation to estimate the neutron dose for workers with potential for neutron exposure based on their work activities with neutron emitting equipment using a 1:1 ratio between the measured photon deep dose, and uncertainty in Tables 19 and 20, is sufficient to address uncertainty in the neutron dose considering the low probability of significant neutron dose.

6.6 ORGAN DOSE

After calculating the photon and neutron doses and their associated standard errors for each year, dose reconstructors can use the values to calculate organ doses of interest using NIOSH (2002). There are many complexities and uncertainties when applying organ DCFs to adjusted doses of record. Many of the factors that affect the dose of record have been summarized in tables in this Site Profile. ICRU (1988) indicated that film badge dosimeters, while not tissue-equivalent, can be used for personnel dosimetry though it is more difficult to ensure that the variation in response with energy and angle of incidence is correct for lower energy photon radiation. Given the many uncertainties, especially with film badge dosimetry in the 1950s through 1970s, a claimant-favorable approach is used to estimate organ dose using an anterior-posterior (AP) exposure geometry. The exposure-to-organ DCFs for an AP orientation listed in Table 21 result in a higher organ dose given the radiation effectiveness factors of the intermediate energy photons. As such, these DCFs are used to convert

Table 21. AP photon exposure-to-organ DCFs.[a]

Organ	Exposure-to-organ DCFs (AP geometry)		
	< 30 keV	30-250 keV	>250 keV
Bladder	0.175	1.244	0.883
Bone (red marrow)	0.025	0.626	0.720
Bone (surface)	0.209	1.229	0.764
Breast (female)	0.561	1.266	0.930
Colon	0.075	1.060	0.844
Esophagus	0.014	0.688	0.745
Eye	0.936	1.236	0.880
Gonads (female-ovaries)	0.047	0.955	0.819
Gonads (male-testes)	0.622	1.434	0.941
Liver	0.106	1.064	0.845
Lung	0.100	0.986	0.842
Remainder organs	0.071	0.879	0.787
Skin	0.504	0.892	0.835
Stomach	0.182	1.251	0.885
Thymus	0.288	1.408	0.892
Thyroid	0.473	1.440	0.972
Uterus	0.061	1.011	0.786

a. Source: NIOSH (2002).

recorded film badge gamma (photon) doses to organ dose. Table 22 lists the deep-dose-to-organ DCFs for an AP orientation for neutrons. In the conversion of photon and neutron doses of record to organ doses, the exposure geometry is an important consideration. The AP exposure geometry is used to estimate the organ dose to conduct an initial screening.

Some of the more common exposure geometries encountered in the workplace are defined as follows:

Table 22. AP neutron deep-dose-equivalent-to-organ DCFs.[a]

Organ	Deep-dose-equivalent-to-organ DCFs (AP geometry)		
	10-100 keV	0.1-2 MeV	2-20 MeV
Bladder	1.268	0.796	1.105
Bone (red marrow)	0.651	0.361	0.720
Bone (surface)	0.656	0.436	0.675
Breast (female)	1.111	1.145	1.121
Colon	0.947	0.490	0.912
Esophagus	0.775	0.412	0.869
Gonads (female-ovaries)	0.935	0.424	0.903
Gonads (male-testes)	1.466	1.307	1.222
Liver	0.983	0.641	0.990
Lung	0.737	0.557	0.950
Remainder organs	0.819	0.525	0.889
Skin	0.986	0.853	0.918
Stomach	1.221	0.824	1.099
Thyroid	1.066	1.086	1.123

a. Source: NIOSH (2002).

- An AP exposure is typical for an individual who works in a directional radiation field and faces the source of the radiation source, such as a nuclear weapon component, while working.

- A rotational exposure is typical of an individual who is constantly turning in a directional radiation field while working, such as when conducting inventories in the nuclear weapons storage vaults.

- An isotropic exposure is typical of a worker involved in activities involving a highly nondirectional or omnidirectional radiation field. An example of work in an omnidirectional radiation field that leads to isotropic irradiation of a worker would be maintenance activities where scattered neutrons and photon radiation are incident on the worker from all directions.

The proposed initial screening option to identify likely noncompensable cases based on claimant-favorable organ dose estimates for long-term workers is to use the organ DCFs for an AP exposure geometry as indicated in Tables 6 and 7. Claims that require a more realistic assessment to determine compensability should consider the geometries mentioned above. Appendix B of NIOSH (2002) provides DCFs to convert KCP worker photon and neutron doses to the primary organ doses for many selections of exposure geometry, target organ, and radiation quantity.

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GLOSSARY

Atomic Energy Commission
Original agency established for nuclear weapons and power production; a predecessor to the U.S. Department of Energy.

beta dose
A designation (i.e., beta) on some external dose records referring to the dose from less-energetic beta, X-ray, or gamma radiation.

beta radiation
Radiation consisting of charged particles of very small mass (i.e., the electron) emitted spontaneously from the nuclei of certain radioactive elements. Physically, the beta particle is identical to an electron moving at high velocity.

curie
A special unit of activity. One curie exactly equals 3.7×10^{10} nuclear transitions per second.

deep absorbed dose (D_d)
The absorbed dose at the depth of 1.0 centimeter in a material of specified geometry and composition.

deep dose equivalent (H_d)
The dose equivalent at the depth of 1.0 centimeter in tissue.

depleted uranium (DU)
Uranium having less than the natural mass of ^{235}U; used as components in nuclear weapons or as a surrogate for enriched uranium or plutonium in testing.

dose equivalent (*H*)
The product of the absorbed dose (*D*), the quality factor (*Q*), and any other modifying factors. The special unit is the rem. When *D* is expressed in grays, *H* is in sieverts (1 sievert = 100 rem).

dose of record
The dose files provided by DOE to NIOSH as part of the individual worker files.

dosimeter
A device used to measure the quantity of radiation received. A holder with radiation-absorbing elements (filters) and an insert with radiation-sensitive elements packaged to provide a record of absorbed dose or dose equivalent received by an individual. See *film dosimeter*, *neutron film dosimeter*, *thermoluminescent dosimeter*.

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

dosimetry system
A system used to assess dose equivalent from external radiation to the whole body, skin, and extremities. This includes the fabrication, assignment, and processing of dosimeters as well as interpretation and documentation of the results.

DXT Units

DXT stands for *density times thickness*. When either the density or thickness is known and can be assumed constant, DXT units are used to measure the alternate parameter to a precision of about 0.5%. DXT units use either X-ray generators or ^{137}Cs radioactive sources.

exchange period (frequency)

Period (weekly, biweekly, monthly, quarterly, etc.) for routine exchange of dosimeters.

exposure

As used in the technical sense, a measure expressed in roentgens of the ionization produced by photons (i.e., gamma and X-rays) in air.

extremity

That portion of the arm extending from and including the elbow through the fingertips, and that portion of the leg extending from and including the knee and patella through the tips of the toes.

field calibration

Dosimeter calibration based on radiation types, intensities, and energies present in the work environment.

film

Generally means a film *packet* that contains one or more pieces of film in a light-tight wrapping. The film when developed has an image caused by radiation that can be measured using an optical densitometer. See *nuclear emulsion*.

film density

See *optical density*.

film dosimeter

A small packet of film in a holder that attaches to a wearer.

gamma rays

Electromagnetic radiation (photons) originating in atomic nuclei and accompanying many nuclear reactions (e.g., fission, radioactive decay, and neutron capture). Physically, gamma rays are identical to X-rays but with higher energy; the only essential difference is that X-rays do not originate in the nucleus.

gray

International System unit of absorbed dose (1 gray = 100 rad).

ionizing radiation

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

minimum detectable level (MDL)

A term used in this and other NIOSH documents to refer to a statistically determined minimum detection level, Lower Limit of Detectability (L_D), and related quantities.

neutron

A basic particle that is electrically neutral weighing nearly the same as the hydrogen atom.

neutron film dosimeter
A film dosimeter that contains an NTA film packet.

nuclear emulsion
Often referred to as NTA film and used to measure personnel dose from neutron radiation.

nuclear track emulsion, type A (NTA)
A film that is sensitive to fast neutrons. The developed image has tracks caused by neutrons that can be seen by using an appropriate imaging capability such as oil immersion and a 1,000-power microscope or a projection capability.

open window
Designation on film dosimeter reports that implies the use of little shielding. It commonly is used to label the film response corresponding to the open window area.

optical density
The quantitative measurement of photographic blackening with the density defined as $D = \mathrm{Log}_{10} (I_o/I)$.

personal dose equivalent, $H_p(d)$
Represents the dose equivalent in soft tissue below a specified point on the body at an appropriate depth *d*. The depths selected for personnel dosimetry are 0.07 millimeter and 10 millimeter for the skin and body, respectively. These are noted as $H_p(0.07)$ and *Hp(10)*, respectively.

photon
A unit or particle of electromagnetic radiation consisting of X- or gamma rays.

photon – X-ray
Electromagnetic radiation of energies between 10 and 100 kilovolts-electron whose source can be an X-ray machine or radioisotope.

quality factor, *Q*
A modifying factor used to derive dose equivalent from absorbed dose.

radiation
Alpha, beta, neutron, and photon radiation with sufficient energy to ionize atoms. See *ionizing radiation*.

radioactivity
The spontaneous emission of radiation, generally alpha or beta particles, gamma rays, and neutrons from unstable nuclei.

rem
A special unit of dose equivalent, which is equal to the product of the number of rad absorbed and the quality factor.

roentgen (R)
A unit of exposure to gamma (or X-ray) radiation. It is defined precisely as the quantity of gamma (or X-) rays that will produce a total charge of 2.58×10^{-4} coulomb in 1 kilogram of dry

air. An exposure of 1 roentgen is approximately equivalent to an absorbed dose of 1 rad in soft tissue for higher (above 100 kilovolts-electron) energy photons.

shallow absorbed dose (D_s)
The absorbed dose at a depth of 0.007 centimeter in a material of specified geometry and composition.

shallow dose equivalent (H_s)
Dose equivalent at a depth of 0.007 centimeter in tissue.

shielding
Any material or obstruction that absorbs (or attenuates) radiation and thus tends to protect personnel or materials from radiation.

skin dose
Absorbed dose at a tissue depth of 7 milligrams per square centimeter.

thermoluminescent
Property of a material that causes it to emit light as a result of being excited by heat.

thermoluminescent dosimeter (TLD)
A holder containing solid chips of material that when heated will release the stored energy as light. The measurement of this light provides a measurement of absorbed dose.

Whole-body dose
Commonly defined as the absorbed dose at a tissue depth of 1.0 centimeter (1,000 milligrams per square centimeter); however, this term is also used to refer to the recorded dose.

X-ray
Ionizing electromagnetic radiation that originates external to the nucleus of an atom.

ATTACHMENT A
EXTERNAL AND INTERNAL AMBIENT ENVIRONMENTAL DOSE ASSIGNMENT GUIDELINE

Potential external and internal radiation ambient environmental exposures for unmonitored workers were based on KCP measured workplace data.

A1.0 EXTERNAL RADIATION

No measurements of environmental external dose from KCP operations have been located. The expectation from site documents is little if any radiological ambient environmental impact. However, in spite of no evidence of an impact, it is recommended as a claimant-favorable approach that dose reconstructors assign an external dose of 25 mrem per year to an unmonitored, nonradiological worker for each year of employment at KCP. The IREP input to calculate the organ dose is shown in Table A-1. This potential external radiation dose was determined based on the 95% variability calculated from background control personnel dosimeters (i.e., 2-sigma = 25 mrem), as shown in Table A-2.

Table A-1. IREP input for ambient external dose calculation.

Exposure rate	Radiation type	Dose distribution type	Parameter 1	Parameter 2	Parameter 3
Chronic	Photons E=30-250keV	Constant	0.025	0.000	0.000

A2.0 INTERNAL RADIATION

The primary and apparently only radionuclide handled in large quantities with a potential for significant environmental release was depleted uranium (DU) from 1959 through 1971. The recommended approach to assign an ambient environmental internal dose to unmonitored nonradiological workers is based on analysis of dispersion to the outside of measured laboratory uranium concentrations. There are no windows and all potential releases were from stacks. The status of filtration for these stacks is not known. KCP representatives identified two primary laboratories used for handling depleted uranium as follows:

Department	Stack velocity (ft^3/min)	Work room area (ft^2)
Machining area (20D)	2,400	8,800
Mixing/curing (27C)	1,987	7,000

Based on this information, a "claimant favorable" estimate of dispersion between outside and inside air concentrations of 0.01 is estimated as described in Section A2.1 based on National Council on Radiation Protection and Measurements Report 123, *Screening Models for Releases of Radionuclides to Atmosphere, Surface Water, and Ground"* (NCRP 1996). An annual intake fraction of 0.01 times the median annual measured laboratory air concentrations in Table 11 (i.e., multiply annual concentrations in Table 11 by 6.58×10^{10} to obtain pCi/day intake) is used to estimate a potential daily intake. For the period following 1970, there are no workplace air sampling data available for analysis. Since there is the potential for unknown small releases or residual contamination, such as the incident with ^{147}Pm in 1989, it is recommended that the average median concentration for all years from Table 11 be used with the factor of 0.01 for each year of employment following 1970 until 1990 when there is assurance that no significant environmental releases occurred based on the multi-agency response findings to the ^{147}Pm incident. The resulting input for IMBA is shown in Table A-3.

Table A-2. KCP personnel dosimeter control dosimeter data.

Account series	Photon deep dose DDE (mrem)	Photon shallow dose SDE (mrem)	Days	mrem/day	mrem/hr
DUO	37	37	206	0.179612	0.007
DUO	34	34	210	0.163726	0.007
DUO	41	42	216	0.188441	0.008
DUO	40	39	223	0.179878	0.007
ENG	35	34	206	0.169734	0.007
ENG	35	37	210	0.16878	0.007
ENG	36	38	216	0.167176	0.007
ENG	39	38	223	0.174036	0.007
FMT	30	29	206	0.145881	0.006
FMT	29	28	210	0.135928	0.006
FMT	33	33	216	0.152259	0.006
FMT	33	32	223	0.148526	0.006
GEN	30	30	206	0.146249	0.006
GEN	32	33	210	0.151753	0.006
GEN	35	36	216	0.161556	0.007
GEN	34	34	223	0.152212	0.006
MET	53	53	206	0.259466	0.011
MET	52	53	210	0.247593	0.010
MET	59	62	216	0.271173	0.011
MET	58	57	223	0.260149	0.011
NDE	45	45	206	0.220097	0.009
NDE	48	50	210	0.226797	0.009
NDE	49	52	216	0.228593	0.010
NDE	52	52	223	0.235022	0.010
NGA	41	41	206	0.200631	0.008
NGA	37	36	210	0.174383	0.007
NGA	42	44	216	0.196286	0.008
NGA	42	41	223	0.189626	0.008
RAD	29	29	206	0.14282	0.006
RAD	32	32	210	0.153984	0.006
RAD	34	34	216	0.155472	0.006
RAD	35	35	223	0.156726	0.007
ULT	41	40	206	0.200863	0.008
ULT	41	40	210	0.194969	0.008
ULT	45	46	216	0.208939	0.009
ULT	47	46	223	0.209522	0.009
WMT	39	39	206	0.188718	0.008
WMT	35	35	210	0.168635	0.007
WMT	44	47	216	0.201806	0.008
WMT	44	44	223	0.197593	0.008
XRY	40	39	206	0.194161	0.008
XRY	43	43	210	0.203215	0.008
XRY	49	51	216	0.224537	0.009
XRY	48	47	223	0.213842	0.009
			Average =	**0.1888947**	
			Stdev =	**0.0345245**	
			year, 1-sigma	**12.60143**	**mrem**
			2-sigma	**25.20286**	**mrem**

Table A-3. IMBA input for ambient internal dose calculation.

Year	Median KCP measured air concentration		GSD
	(μCi/cm^3)	pCi/day	
1958	4.01E-13	2.64E-02	3.0
1959	2.89E-13	1.90E-02	3.0
1960	3.41E-13	2.24E-02	3.0
1961	1.97E-13	1.30E-02	3.0
1962	2.50E-13	1.65E-02	3.0
1963	2.47E-13	1.63E-02	3.0
1964	3.91E-13	2.57E-02	3.0
1965	1.05E-13	6.91E-03	3.0
1966	2.00E-13	1.32E-02	3.0
1967	5.70E-13	3.75E-02	3.0
1968	2.31E-13	1.52E-02	3.0
1969	3.88E-12	2.55E-01	3.0
1970	4.02E-14	2.65E-03	3.0
1971-89	5.49E-13	3.62E-02	3.0

A2.1 KCP DU AIR CONCENTRATIONS OUTSIDE THE PLANT FROM ACTIVITIES INSIDE THE PLANT

Calculations of the possible air concentration of DU outside the KCP facility due to activities inside the building were done based on the measured inside air concentration and engineering details of the primary laboratories. KCP work with DU was performed in two laboratories described as follows:

Department	Stack velocity (ft^3/min)	Work room area (ft^2)
Machining area (20D)	2,400	8,800
Mixing/curing (27C)	1,987	7,000

Ref: Email Nasca to Fix, 21-Oct-2005. Subject Re: Uranium handling at KCP.

For purposes of calculating dilution factors, the measured room air concentration was assumed to be 1 Bq/m^3 (a unit air concentration). The results of the calculated outside air concentration was then used to calculate the potential occupational dose for selected receptor locations.

A2.1.1 Receptor on the Roof

Using the Machining Area (20D) as an example, the quantity of DU exhausted from the room is: 1.133 Bq/s[1].

Equation 1 (Equation 2.8 from NCRP 1996) was used to calculate the air concentration downwind from the release point but on the same surface as the release point (i.e. both the source and receptor are on the roof).

$$C = B_0 \frac{Q}{\mu_h x^2} \qquad \text{(Eq. 1)}$$

Where $B_0 = 30$, Q = the release rate (1.133 Bq/s), u_h = mean wind speed (10.6 mph = 4.74 m/s), and x is the distance downwind (m). To obtain the distance where $C = 0.01$, Equation 1 is rewritten as Equation 2 and solved for x.

$$x = \sqrt{\frac{B_0 Q}{C u_h}} \qquad \text{(Eq. 2)}$$

Substitution of the numerical values given above, $C = 0.01$ when x exceeds 26.7 m (87.9 ft) and the wind always blows towards the recipient.

A2.1.2 Receptor on the Ground Downwind From the Source on the Roof

For this case the calculations use Equation 3 (Equation 2.6 from NCRP 1996)

$$C = \frac{fQP}{u} \qquad \text{(Eq. 3)}$$

Where f is the fraction of the time that the wind blows from the source to the receptor and P is the dilution factor from figure 2.2 of the NCRP report. The greatest air concentrations result when the height of the building is assumed to be zero, in which case the value of P is 3.6×10^{-2}. Substituting the numeric values given above, the value of C is 8.6×10^{-3} Bq/m^3. The dilution factor of 0.0086 is for a distance of 100 m (the closest distance plotted in the reference figure) from the source point. This analysis is based on the claimant-favorable assumption that the wind is always blowing toward the receptor.